DOE Review Release 12/12/2012

Document Title:	Document Number:	ORAUT-TKBS-0041
Site Profile for Nuclear Materials and Equipment Corporation, Apollo and Parks Township, Pennsylvania	Revision:	02
	Effective Date:	11/26/2012
	Type of Document:	TBD
	Supersedes:	Revision 01 PC-1

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Approval:	Signature on File James W. Neton, Associate Director for Science	Approval Date:	11/26/2012

☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

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

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
09/26/2008	00	Approved new technical basis document for Nuclear Materials and Equipment Corporation (NUMEC) in Apollo, Pennsylvania. Incorporates formal internal and NIOSH review comments. Training required: As determined by Task Manager. Initiated by Dennis L. Strenge.
06/02/2009	01	Revised to incorporate the Parks Township facility. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by Objective Manager. Initiated by Dennis L. Strenge.
02/26/2010	01 PC-1	Page change revision to revise minimum detectable concentration for plutonium in urine value based on updated analysis of bioassay data. This resulted in changes to Section 5.2 on pages 44 and 45 and Table 5-4 on page 45. Updated references on pages 43, 44, 51, 82-83. Added reference on page 81 and updated other references on pages 82-83 in the Reference Section. No changes occurred as a result of formal internal review. Incorporates formal NIOSH review comments. Training required: As determined by Objective Manager. Initiated by Dennis L. Strenge. Approval: Signature on File 02/11/2010 Dennis L. Strenge, Document Owner Signature on File 02/11/2010 John M. Byrne, Objective 1 Manager Signature on File 02/17/2010 Edward F. Maher, Objective 3 Manager Signature on File 02/17/2010 Kate Kimpan, Project Director Signature on File 02/26/2010 James W. Neton, Associate Director for Science
11/26/2012	02	Revision initiated due to a change in guidance of dose during residual period and change in guidance on assignment of medical X-ray dose. Sections 3.0, 7.4.1, 7.4.2 were revised. Table 7-1 was deleted and Tables 7-2 (now 7-1), 7-3 (now 7-2) and 7-4 (now 7-3) were revised. A glossary was added. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by Objective Manager. Initiated by Dennis L. Strenge.

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ACRONYMS AND ABBREVIATIONS

ADU	ammonium diuranate
AEC	U.S. Atomic Energy Commission
ANL-E	Argonne National Laboratory-East
AWE	Atomic Weapons Employer
B&W	Babcock & Wilcox (Company)
BZA	breathing-zone air
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
CP	Chemical Processing
CRP	Chemical Reprocessing
d	day
DCF	dose conversion factor
DHHS	U.S. Department of Health and Human Services
DOE	U.S. Department of Energy
dpm	disintegrations per minute
DU	depleted uranium
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EU	enriched uranium
F	fast (absorption type)
°F	degrees Fahrenheit
Fab	fabrication area
FFTF	Fast Flux Test Facility
FP	fission product
fpm	feet per minute
ft	foot
g	gram
GA	general air
gal	gallon
GSD	geometric standard deviation
HASL	Health and Safety Laboratory
HEPA	high-efficiency particulate air
HEU	highly enriched uranium
HTGR	High-Temperature Gas-Cooled Reactor
hr	hour
ICRP	International Commission on Radiological Protection
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1,000 electron-volts
kg	kilogram
L	liter
lb	pound

LEU	low-enriched uranium
LLRW	low-level radioactive waste
LOD	limit of detection
M	moderate (absorption type)
m	meter
mCi	millicurie
MDA	minimum detectable activity (or amount)
MDC	minimum detectable concentration
MDL	minimum detectable level
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
min	minute
mL	milliliter
MOX	mixed oxide
MPC	maximum permissible concentration
mrad	millirad
mrem	millirem
MWd	megawatt-day
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
NTA	nuclear track emulsion, type A
NU	natural uranium
NUMEC	Nuclear Materials and Equipment Corporation
ORISE	Oak Ridge Institute for Science and Education
OW	open window
pCi	picocurie
PL	plastic
POC	probability of causation
R	roentgen
R&D	research and development
RU	recycled uranium
s	second
S	slow (absorption type)
SEC	Special Exposure Cohort
SLDA	Shallow Land Disposal Area
SNM	Special Nuclear Material
SRDB Ref ID	Site Research Database Reference Identification (number)
SRS	Savannah River Site
t	ton (metric)
TCE	trichloroethylene
TLD	thermoluminescent dosimeter
TRU	transuranic
U.S.C.	United States Code
WB	whole body

wt %	weight %
yr	year
ZPPR	Zero Power Plutonium (later Physics) Reactor
µCi	microcurie
µg	microgram
µR	microroentgen
§	section or sections

1.0 INTRODUCTION

1.1 PURPOSE

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer [AWE] facility" or a "Department of Energy [DOE] facility" as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA, as amended, provides for employees who worked at an AWE facility during the contract period and/or during the residual period.

Under EEOICPA, employment at an AWE facility is categorized as either (1) during the DOE contract period (i.e., when the AWE was processing or producing material that emitted radiation and was used in the production of an atomic weapon), or (2) during the residual contamination period (i.e., periods that NIOSH has determined there is the potential for significant residual contamination after the period in which weapons-related production occurred). For contract period employment, all occupationally derived radiation exposures at covered facilities must be included in dose reconstructions. This includes 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 covered period. NIOSH does not consider the following exposures to be occupationally derived (NIOSH 2010a):

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

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

For employment during the residual contamination period, only the radiation exposures defined in 42 U.S.C. § 7384n(c)(4) [i.e., radiation doses received from DOE-related work] must be included in dose reconstructions. Doses from medical X-rays are not reconstructed during the residual contamination period (NIOSH 2010a). It should be noted that under subparagraph A of 42 U.S.C. § 7384n(c)(4), radiation associated with the Naval Nuclear Propulsion Program is specifically excluded from the employee's radiation dose. This exclusion only applies to those AWE employees who worked during the residual contamination period. Also, under subparagraph B of 42 U.S.C. § 7384n(c)(4), radiation from a source not covered by subparagraph A that is not distinguishable through reliable documentation from radiation that is covered by subparagraph A is considered part of the employee's radiation dose. This site profile covers only exposures resulting from nuclear weapons-related work. Exposures resulting from non-weapons-related work, if applicable, will be covered elsewhere.

1.2 SCOPE

This site profile provides specific information about documentation of historical practices at the Nuclear Materials and Equipment Corporation (NUMEC) Apollo and Parks Township sites. This site profile for NUMEC presents information useful for reconstruction of doses NUMEC employees received.

1.3 SPECIAL EXPOSURE COHORTS

This section describes the classes in the Special Exposure Cohort (SEC) for the NUMEC sites in Apollo and Parks Township, Pennsylvania.

The SEC classes cover employees who worked at one or both of the Apollo and Parks Township facilities. This site profile covers both NUMEC facilities and can be used to perform partial dose reconstructions for individuals who worked at either or both facilities.

1.3.1 Apollo Site Class

An SEC class has been identified that includes all AWE employees who were monitored or should have been monitored for exposure to ionizing radiation while working at the NUMEC site in Apollo, Pennsylvania, from January 1, 1957, through December 31, 1983, for a number of days aggregating at least 250 work days or in combination with work days within the parameters established for one or more other classes of employees in the SEC (Leavitt 2007).

NIOSH has determined, and the Secretary of the U.S. Department of Health and Human Services (DHHS) has concurred, that it is not feasible to reconstruct doses for the following situations:

- Uranium internal exposure before 1960 for lack of bioassay monitoring;
- Thorium and plutonium internal exposures for lack of monitoring data, process description, and source term data;
- Potential ambient radiation dose from stack releases;
- Dose from radium-beryllium and polonium-beryllium neutron source fabrication operations; and
- Internal doses if the bioassay data were based on the NUMEC Apollo contactor, Controls for Environmental Pollution, from 1976 through 1983, because of concerns about data quality.

Although the combined petition evaluation report for petitions SEC-00047 and SEC-00080 (NIOSH 2007a) focused on the inability to estimate dose for the above situations during the period from January 1, 1957, through December 31, 1983, partial doses can be estimated for workers for whom applicable monitoring data are available. The DHHS designation for the worker class indicates that it is possible to reconstruct occupational medical dose and components of the internal dose (uranium doses starting from 1960). Therefore, individuals with nonpresumptive cancers can be considered for partial dose reconstruction (Leavitt 2007).

1.3.2 Parks Township Site Class

An SEC class has been identified that includes all AWE employees who worked at the NUMEC facility in Parks Township, Pennsylvania, from June 1, 1960, through December 31, 1980, for a number of workdays aggregating to at least 250 workdays occurring either solely under this employment or in combination with workdays within the parameters established for one or more other classes of employees in the SEC (Leavitt 2008).

NIOSH has determined, and the Secretary of DHHS has concurred, that it is not feasible to reconstruct doses for the following situations:

- Thorium internal exposures for lack of monitoring data and process descriptions;

- Internal exposures for work with irradiated fuel and fabrication of radiation sources for lack of monitoring data, process descriptions, and source term data; and

- Internal doses if the bioassay data were based on the NUMEC Apollo contactor, Controls for Environmental Pollution, from 1976 through 1980, because of concerns about data quality.

Although the SEC petition evaluation report for petition SEC-00108 (NIOSH 2008a) focused on the inability to estimate dose for the above situations during the period from June 1, 1960, through December 31, 1980, partial doses can be estimated for workers for whom applicable monitoring data are available. The DHHS designation for the worker class indicates that it is possible to reconstruct occupational medical dose and components of the internal dose when adequate monitoring data are available. Therefore, individuals with nonpresumptive cancers can be considered for partial dose reconstruction (Leavitt 2008).

2.0 SITE DESCRIPTIONS

2.1 APOLLO SITE

2.1.1 General Description and Operational History

The NUMEC Apollo nuclear fuel site operated under license SNM-145 and Source Material License C-3762 that the U.S. Atomic Energy Commission (AEC; a DOE predecessor agency) issued in 1957. From 1958 to 1983, the Apollo site was used for small-scale research and production of low-enriched uranium (LEU), highly enriched uranium (HEU), and thorium fuels. By 1963, the majority of the Apollo facility was dedicated to production of uranium fuel. The major activities at NUMEC Apollo included (1) the conversion of LEU hexafluoride (<5% ^{235}U by weight) to uranium oxide (UF_6 to UO_2) for use in light-water-moderated reactors; (2) the conversion of HEU to produce HEU (>20% ^{235}U) nuclear fuel for use in the naval reactors program; and (3) the processing of unirradiated uranium scrap (including LEU and HEU) from the AEC in the 1960s (B&WNES 1997).

In 1967, the Atlantic Richfield Company bought the Apollo facility from the original owner of NUMEC. In 1971, the facility was purchased by the Babcock & Wilcox Company (B&W), which ran the uranium fuel facility and nuclear laundry until production stopped in 1983. Decommissioning support activities began in 1978 and the Apollo site ceased all operations in 1983. Early decommissioning activities included site characterization, demolition of certain building structures, and selected soil remediation. In 1992, the U.S. Nuclear Regulatory Commission (NRC) approved the Apollo site decommissioning plan and decommissioning was complete in 1995 (B&WNES 1997).

The Apollo facility had one main bay (known as the East Bay), and three smaller attached bays known as the West Bay, the Box Shop, and the Annex. The site also included a Laundry Building and a Small Block Building in the parking lot. The Laundry Building was used for washing protective clothing from the nuclear facilities and the Small Block Building was used for storage of processing equipment. These buildings were on the east side of the site between Warren Avenue and the Kiskiminetas River. The Apollo Office Building was outside the site area across Warren Avenue. The Office Building contained a laboratory that was used to analyze radioactive and nonradioactive product. A small portion of the building basement housed operations that manufactured instruments for the production of nuclear fuels. Although the Office Building was not an original part of the Apollo Decommissioning Project, it was included as part of the project in the spring of 1993 after it was determined that some floorboards and drain lines contained uranium contamination (B&WNES 1997). The parking lot area was bounded by the Kiskiminetas River on the west, Warren Avenue on the east, and the offsite area on the north. Figure 2-1 is a general layout of the Apollo site.

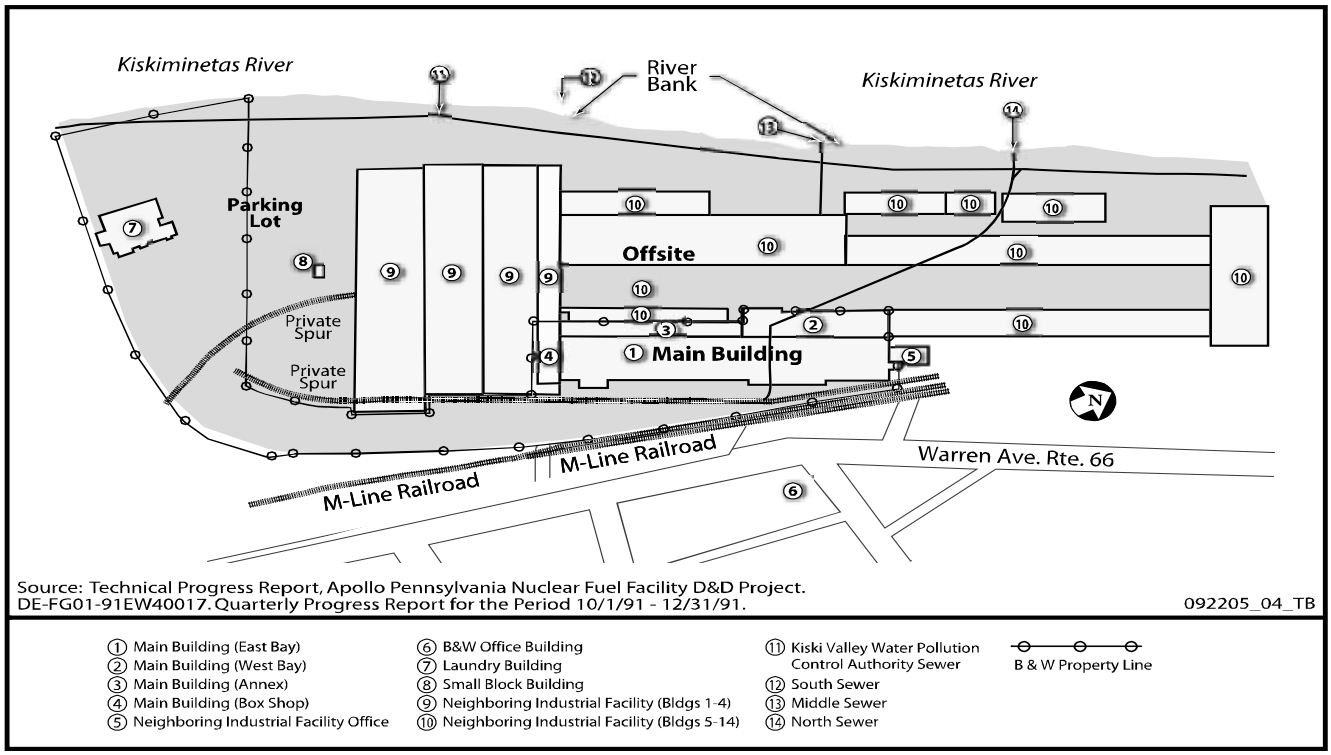

Figure 2-1. Apollo site layout (Author unknown 2004).

2.1.2 Facilities

The Apollo site was divided into production and process areas and clean areas. Personnel were required to enter through the main entrance near the parking lot. Before exiting through the main entrance, personnel were to shower if they had entered or worked in a production or process area. There were two emergency exits. Entrance into production areas was through the change room with the exception of the filter press section of the plant waste treatment area (NUMEC 1963). Production and process areas are listed in Table 2-1.

2.1.3 Process Descriptions

Brief summaries for the principal operations are provided below; additional details can be found in the series of AEC Feasibility and Health and Safety Laboratory (HASL) reports in the cited references (Forscher 1963; AEC 1960a,b,c, 1961a,b). Inherent in all the operations was nuclear criticality safety that governed not only the operations and storage but also the movement of material in the facility. A review of the available literature showed that no criticality accidents occurred during Apollo site operations.

Table 2-1. Apollo site area descriptions.

Building or area location	Description	Operations/radionuclides	Period of operation
CF-1	Ceramic fabrication	UO_2, ThO_2, (metal, powder, and oxide)	1957–1970 ThO_2: 1963–1970
CF-2	Ceramic fabrication	Uranium metal (HEU and DU) UO_2, and U_3O_8	Early 1959–1972
PC-1	Process chemistry	HEU, EU, DU, $(NH_4)_2U_2O_7$), UO_3, UF_6, UF_4, uranyl nitrate, UO_2, and U_3O_8	1957–1983 HEU: 1957–1978 LEU: 1957–1983
PC-2	Process chemistry	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983

Building or area location	Description	Operations/radionuclides	Period of operation
PC-3	Process chemistry	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CP-1	Chemical processing	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CP-2	Chemical processing	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CRP-1	Chemical reprocessing	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CRP-2	Chemical reprocessing	HEU, EU, DU, U nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CRP-3	Chemical reprocessing	Beryllium handling equipment, HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
GF-1			1957–1983
QC	Quality control		1957–1983
PS			1957–1983
A Vault	Process security material. Controlled by CP-2	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
C Vault	Receiving and shipping storage area		1957–1983
E Vault	Storage of SNM		1957–1983
F Vault	Storage for SNM of all enrichments		1957–1983
G Vault	Solution storage area, in-process storage of materials for CP-1	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
H Vault	Storage of SNM		1957–1983
Waste Treatment Area	Filter press section	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, U_3O_8, FPs, and TRU elements	1957–1983
GPH Room 2	Health and safety counting room	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, U_3O_8, FPs, and TRU elements	1957–1983
Office Building	Off site	Uranium	1957–1983
Laundry Building		Co-60, plutonium, thorium, and uranium	1960–1983 (closed 1984)
Block Building		Storage of processing equipment	1957–1983
Incinerator	Area 62 (Hoskinson H-100)	30–35 lb/hr, 300 g ^{235}U limit/batch	1960–1983

Sources: Author unknown (2004); NUMEC (1963); B&WNES (1997); Caldwell (1966, 1968a).

2.1.3.1 General Process Operations

NUMEC Apollo process operations were varied. Because the Apollo site was a research center and an all-purpose process center, a variety of radioactive materials and Special Nuclear Materials (SNM) were processed. Most of the work performed involved work for commercial reactors but much of the work was in the development of better fuel configurations for the burgeoning Navy nuclear program. Although this work is not weapons related, the SNM and radiation exposures from this work during the AEC operational years (1957 to 1983) are to be included in radiation dose reconstruction in compliance with NIOSH policy. Some examples are listed below to illustrate the breadth of the SNM operations that occurred at the Apollo site.

2.1.3.2 Uranium Process Activities

A general description of processes for various enrichments of uranium at Apollo is outlined in AEC HASL Survey Reports 82, 92, 103, 106, and 114 (*Occupational Exposure to Radioactive Dusts*

reports) which cover the period from December 1959 to January 1961 and the *Procedures for Recovery of Uranium Scrap* from 1962 (AEC 1960a,b,c, 1961a,b; NUMEC 1962). The following paragraphs provide a description of the process.

Production of UO_3 from UF_6 began with UF_6 being converted to UO_3 in the Chemical Conversion Room. The UF_6 gas was hydrolyzed to an aqueous solution of UO_2F_2 and HF. This solution was reacted with NH_4OH to form a slurry of ammonium diuranate (ADU) [$(NH_4)_2U_2O_7$]. The slurry was then pumped through a hooded pressure filter. The filter cake was transferred to drying hoods where the ADU was decomposed at a controlled temperature to UO_3. The UO_3 product was transferred in small polyethylene containers to Ceramics Fabrication for further processing (AEC 1960a, p. 3). The HASL-92 (AEC 1960b, p. 2) survey from August 1960 noted that the filter cake was dried by a rotary kiln rather than the earlier fry pan method. The HASL-114 (AEC 1961b, p. 2) survey from June 1961 noted that a calciner was added for reduction of ADU to U_3O_8. The dried cake was discharged directly from the kiln into a container, eliminating the manual transfer.

NOTE: While not specifically stated in the HASL reports, the removal of the filter cake from the pressure filter might have been a manual operation. No information could be found to indicate that the filter scraping was automated or that it was performed in a glovebox or under a filter hood.

Reduction of UO_3 to UO_2 was performed in the Ceramics Fabrication Area. UO_2 product was transferred to the Ceramic Laboratory for additional fabrication. HASL-92 (AEC 1960b) noted that the UO_3 was reduced to UO_2 in a rotary kiln rather than the reduction furnace that had been used earlier.

Ceramics fabrication was performed in the Ceramics Fabrication Area where UO_2 was hammer-milled in a ventilated enclosure and then moved to the blender glovebox where aerowax was added and the mixture was blended. The wax-UO_2 mixture was then pressed into a cake in a Drake press. The cake was placed in a glovebox where it was granulated by hand with screens to give the desired particle size. The UO_2 was loaded into shallow metal pans called “firing boats” and sintered. Sintered UO_2 was classified, weighed, and packaged.

Hammer milling, blending, granulating, pellet pressing, centerless grinding, inspecting, and packaging were performed in the Ceramics Fabrication Area.

Uranium-graphite pellet production was a temporary contract activity. U_3O_8 and graphite powder were weighed, blended in a twin shell blender, and prepressed in a Drake press in a large polyethylene tent in the ceramic laboratory. The prepressed slugs were granulated. The granulated powder was fed into the hopper of the Colton press and compact-pressed. The resultant pellets were cured, inspected, and packed for shipping. All steps after prepressing were performed in the Ceramics Fabrication Area.

R&D in techniques for coating uranium and UO_2 particles with metallic elements such as niobium, chromium, tungsten, and molybdenum by halide reduction and vapor deposition was performed. According to the 1960 and 1961 HASL survey reports (AEC 1960a,b,c,d, 1961a,b), an approximately 0.25-in. layer of UO_2 powder was placed in a furnace tube (1-in. inner diameter by 2 ft) and heated to the desired temperature. A ratio of niobium pentachloride to hydrogen was introduced. The pentachloride was reduced by the hydrogen such that niobium metal deposited on the UO_2 particles. Vibration of the system enabled uniform particle coating.

All sampling for uranium content and accountability was conducted in the Wet Chemistry, Spectrographic, and Metallographic Laboratories.

Recovery of uranium (U_3O_8) from uranium-zirconium scrap chips in oil was performed in the Chemical Reprocessing Rooms. The 1960 HASL-82 (AEC 1960a) survey report described the recovery of U_3O_8 from uranium-zirconium scrap as follows:

The oil was drained and the chips were transferred to wire mesh baskets and degreased with trichloroethane in a ventilated degreasing tank. The chips were then dissolved in hydrofluoric acid in chemical fume hoods. The batch was heated and hydrogen peroxide was added to oxidize the insoluble UF_4 to soluble UF_6. The batch was filtered and the filtrate was reduced in chemical fume hoods, with insoluble UF_4 precipitating preferentially from the solution. The solutions were filtered and the UF_4 collected in a common filter. The UF_4 was then converted to U_3O_8 by ignition.
According to *Procedures for Recovery of Uranium Scrap* (NUMEC 1962), NUMEC dissolved the scrap in two designated areas, CRP-2 and CRP-3. The product solutions from the dissolution methods were processed to generate insoluble UF_4, and were ultimately converted to the final product of U_3O_8 or UO_2.

According to the HASL-92 (AEC 1960b) survey report, the experimental development of recovering U_3O_8 (93%) by solvent extraction was under development at the Apollo site (AEC 1960b). In addition, UF_4 was being converted to U_3O_8 by ignition, with the U_3O_8 granulated manually through screens. A facility for the processing of HEU was established on the second floor near the scrap recovery area.

According to the HASL-103 (AEC 1960c) survey report, a cascade-solvent extraction uranium-zirconium recovery process was under construction. A similar type of extraction process was already in operation for Chemical Reprocessing (CRP), which consisted of leaching, feed preparation, solvent extraction, ammoniation of strip solution, precipitation, filtration of ADU slurry, kiln drying, and packaging as UO_3.

2.1.3.3 High-Temperature Gas-Cooled Reactor Critical Assembly Fuel Elements

A March 25, 1960, letter (Katine 1960) recommended approval of NUMEC Feasibility Report to fabricate 3,000 graphite fuel elements to be used in the General Atomics High-Temperature Gas-Cooled Reactor (HTGR) critical assembly. The job was to involve between 95 and 120 kg of 93%-enriched U_3O_8 supplied by another company. The total ^{235}U content of 2,850 fuel elements was to be 79.339 kg. A letter dated April 1960 to Shapiro (NUMEC) from Wesley Johnson, Deputy Manager (AEC), indicated approval of Feasibility Report for the General Atomic HTGR critical assembly fuel elements (Katine 1960).

2.1.3.4 Uranium Nitrate Solution for the University of California

A letter report dated June 9, 1961 (Katine 1961), to J. E. McLaughlin, Director, Radiation Physics Division, HASL, describes a trip to the Apollo site on June 7, 1961, to observe equipment for producing uranyl nitrate solution for the University of California. A vague description of NUMEC processes and facilities was reported. The report mentions nitric acid and aluminum nitrate solutions employed in the solvent extraction process.

2.1.3.5 Incinerator

Combustible contaminated solid wastes were incinerated in Area 62 of the Apollo site. The system consisted of a Hoskinson H-100 incinerator with a main burner in the firebox and an afterburner in the stack just above the firebox. Both burners used natural gas for fuel (Caldwell 1968a).

Packages of contaminated waste were labeled with the ^{235}U content and were burned at a rate of 30 to 35 lb/hr. Ashes were collected in stainless-steel 1-gal containers. After cooling, the ash can was placed into a closed container and transferred to an ash handling glovebox. The ashes were sifted, sampled for uranium content, and transferred to a clean, lidded 1-gal pail which was transferred to one of the plant vaults (Caldwell 1968a).

The operator was protected during charging by a positive inflow of air through the charging door. Ash collection was enclosed in an exhausted box. All ash handling was restricted to a glovebox at

negative pressure (Caldwell 1968a). Before 1968, this might not have been the case and work in this area was perhaps the highest for intakes.

Combustible gases passed through the afterburner to a water-operated, venturi-type fume scrubber. This separated the fly ash from the gas stream. The gases were passed through a packed tower (for removal of fine particulates) and discharged through a 15-ft stack. Exhaust air from the ash handling glovebox passed through a prefilter and a high-efficiency particulate air (HEPA) filter before discharge through a roof stack (Caldwell 1968a).

2.1.3.6 Thorium Operations

According to Forscher (1963), which cites the 1963 Feasibility Report No. 47 for ThO_2, NUMEC was to complete fabrication of 626 pellets of ThO_2 with no nuclear criticality considerations necessary. NUMEC was to purchase 30 kg of ThO_2 from Davison Chemical Division of W.R. Grace Company.

NUMEC correspondence (Forscher 1963) with the AEC Oak Ridge Operations Office indicates the following plans for the fabrication of ThO_2 pellets:

1. 30 kg of ThO_2 would be transferred to the CF-1 Fabrication area.

2. Working batches of 5 kg would be processed. All powder transfers and handling would be in ventilated gloveboxes with a face velocity of 100 fpm. Material would be handled wearing latex gloves.

3. The powder would be slugged to 4-5 g/cm^3, then granulated through 14-mesh screen.

4. Each batch of powder would be blended in a “V”-type blender in a ventilated glovebox.

5. The ThO_2 pellets would be pressed using a hand press and/or automatic press in a hood with a face velocity of 100 fpm.

6. The ThO_2 pellets would be sintered in a hydrogen atmosphere with the out-gases of the furnace passing through a filtered exhaust ventilation system.

7. All pellets would be centerless ground in a ventilated hood.

8. The final product would be packaged in sausages with each sausage packaged in a polyethylene bag.

Air sampling was performed to characterize thorium exposures during this period. According to a health protection program review in 1964, thorium operations involving the blender and weighing hood were resulting in excessive airborne concentrations (Thornton and Johnson 1964).

2.1.3.7 Research Activities in the Early Years

There were research projects at the Apollo site that involved mostly the fabrication of new types of fuel in support of the Naval Reactors Branch through the Knolls Atomic Power Plant and Bettis Atomic Research Laboratory. The research involved chemical process development with various forms of uranium compounds and metal.

2.1.4 Source Term

Three main sources describe the amounts and types of radioactive material that were handled at the Apollo site: (1) Federal and State of Pennsylvania licenses for the possession and use of radioactive

materials; (2) descriptions and reviews of proposed experiments or jobs handling radioactive material in the form of HASL reports and process feasibility reports, which contain information on radionuclides, quantities, and recommended safety precautions for the described activity; and (3) inventory/material handling (accountability) records.

The use of SNM was governed by AEC regulations and licenses, under license SNM-145 and Source Material License C-3762 issued by the AEC in 1957 (Docket No 70-135). Some possession limits at different periods are listed for the Apollo site in Table 2-2.

Table 2-2. Apollo site source and SNM possession limits.

Areas	Source/chemical or physical form	Maximum possession
Processing areas, laboratories, and vaults	U-235 enrichment >5%	5,000 kg
	U-235 enrichment ≤5%	75,000 kg
	Plutonium as fully clad or encapsulated material	500 kg
Mass Spectrometry Laboratory	Uranium in any enrichment	350 g
	Plutonium in any form	0.5 g
LLRW storage areas	Within fenced areas in approved storage containers	35 g U-235
	In buildings meeting safeguards and security requirements	50 kg U-235
Nuclear Decontamination Corporation	Any byproduct material	20 mCi
	Any source material	20 g
	Any SNM	20 mCi

Source: Reitler (1972).

The Apollo site radiological source term included uranium, thorium, plutonium, and fission and activation products (Reitler 1972). At present, no definitive information is available to relate measurement of one component of the source term (e.g., plutonium) to another unmonitored component (e.g., americium) for a given area or process. Much of the work was R&D, so unique source terms could be encountered in a particular job.

Uranium. Uranium in the form of metal, oxide, and carbide was used for Apollo fabrication, reactor fuel, and research studies in the gloveboxes and laboratories (AEC 1960a,b,c, 1961a,b). The typical amounts of uranium in use in an area ranged from milligrams to hundreds of kilograms. Occasional work involving other uranium chemical forms, such as UF_6 or uranyl nitrate was conducted. Uranium forms included depleted uranium (DU), natural uranium (NU; i.e., natural enrichment), HEU (up to 93%), as well as ^{232}U, ^{233}U, and ^{236}U. Uranium from recycling operations would have included smaller activities of nonuranium isotopes, such as ^{99}Tc, ^{237}Np, and ^{239}Pu.

Thorium. Thorium dioxide use was similar to uranium use. The total mass of thorium used on site was less than that of uranium overall, but the thorium activity in an area at a given time could have been greater or less than uranium activity. Thorium dioxide was obtained from virgin thorium sources. The use of reclaimed or reconstituted thorium was not acceptable (Forscher 1963).

2.1.5 Remediation, Decontamination, and Decommissioning

The HEU processing area on the second floor of the Apollo East Bay underwent remediation from 1978 until July 1991. All remaining equipment, ventilation systems, piping, and power lines from the area were dismantled and disposed of (B&WNES 1997).

The LEU processing area in the Apollo East Bay was remediated between 1983 and 1984. During this period, the LEU processing equipment was removed and disposed of. By October 1984, all equipment was removed and sent to the Chem-Nuclear low-level radioactive waste disposal facility in Barnwell, South Carolina (B&WNES 1997).

The Laundry Building was remediated between 1984 and 1991. In 1984, the processing equipment, nonessential utilities, and miscellaneous support systems were volume-reduced, packaged, and sent to Chem-Nuclear. The Laundry Building trench, which served as a sump drain for washing machine wastewater, was removed in April 1989 (B&WNES 1997).

All equipment in the Box Shop was removed in 1976. The Small Block Building was demolished and stored in the parking lot until accepted at the on-site processing plant (B&WNES 1997).

As of August 23, 1978, NUMEC had completed decommissioning of its HEU processing at the Apollo site. All process and related equipment were removed by this date. NUMEC indicated that access to the area was limited to authorized personnel. In 1982, the NRC conducted a confirmatory survey to identify HEU that might have been present. The report indicates uranium contamination levels in grams of uranium to surface area. This report was generated to account for HEU inventory during decommissioning. The total grams remaining on and in the floors, walls, pad and ceiling were estimated to be 35,548.55 g of total uranium with about 23,743.27 g of ^{235}U (Martin 1982).

Decontamination efforts were completed from 1984 to June 1992 for the Apollo site. In June 1992, the NRC approved the Apollo decommissioning plan. Decommissioning occurred from June 1992 to 1995. NRC staff reviewed B&W groundwater monitoring data, final termination survey, and a confirmatory survey in 1996. On April 14, 1997, after notifying the Pennsylvania Department of Environmental Protection, the NRC issued a letter to B&W terminating the Apollo license (PDEP 2008).

2.1.5.1 Shallow Land Disposal Area

In October 1995, the NRC placed the adjacent B&W Shallow Land Disposal Area (SLDA) on a separate license. Until 1970, the SLDA was used as a disposal facility for the Apollo site (and possibly the Parks Township site) with about 700,000 ft^3 of waste buried in trenches (PDEP 2008). At present, options for the disposition of the SLDA are in discussion with the NRC and the former licensee. The site might become a Formerly Utilized Sites Remedial Action Program site. As of 2007, a feasibility study was being conducted by the U.S. Army Corps of Engineers (USACE 2006). The area is shown in Figure 2-2.

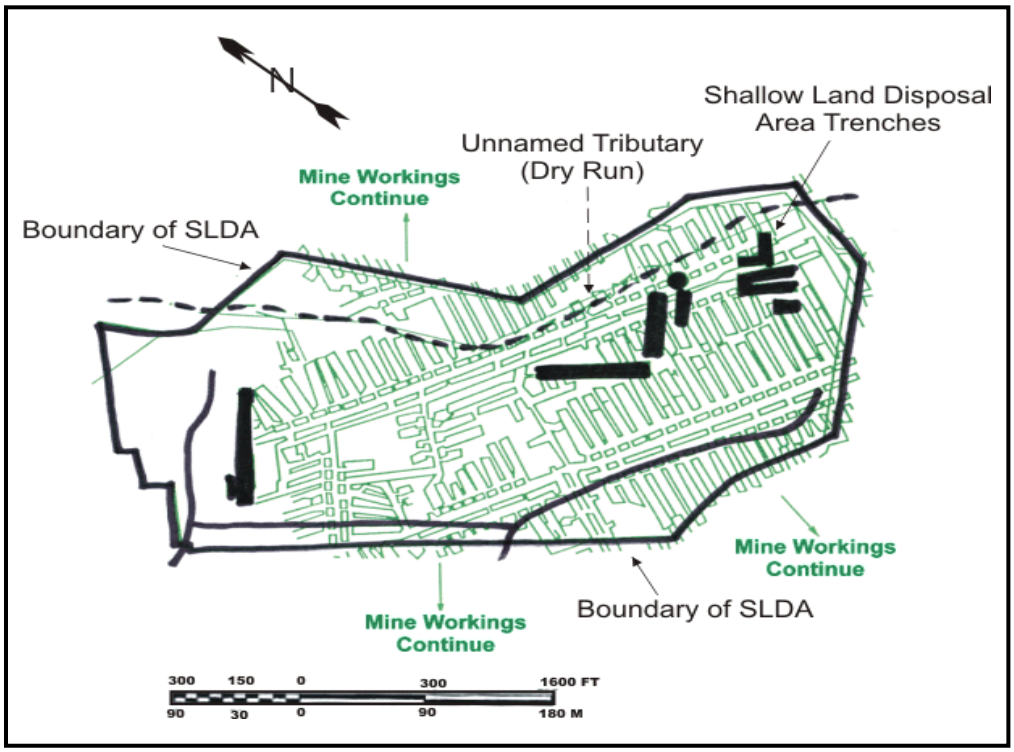

Figure 2-2. B&W Shallow Land Disposal Area (USACE 2006).

2.1.5.2 Administrative Building

The Apollo office building had been used for office space since the mid-1950s. Portions of the building had been used for an analytical laboratory and to develop and manufacture calibration sources in the 1960s and early 1970s. Both operations had been terminated by 1972.

On February 23, 1995, confirmatory surveys were conducted of the administrative building by the NRC. This included soil sampling, surface scans, and exposure rate surveys of the soil beneath the former concrete slab of the basement of the administrative building and of the surface after the area was backfilled and leveled (survey conducted May 23, 1995). All results met the guidelines for unrestricted release.

2.2 PARKS TOWNSHIP SITE

2.2.1 General Description and Operational History

The NUMEC Parks Township site (sometimes referred to as the Advanced Material Center) was on 115 acres in Parks Township, Armstrong County, Pennsylvania, along U.S. Highway 66 about 3 miles southeast of Leechburg and 30 miles northeast of Pittsburgh. The site expanded in stages throughout the 1960s. The original Building A was 20,000 ft^2. From 1961 through 1970, a major expansion of Building A (the plutonium facility) was completed in five separate expansions (61,000 ft^2). This included construction of fabrication areas Fab 2 through Fab 9 and the Drum Warehouse as well as termination and remediation of a drum storage area. The main structures were Buildings A, B, and C, the outside Storage Areas, and the Trailer Storage Area (Author unknown 2004).

Building A was constructed in 1959 and 1960 and was authorized to operate in 1961. The Hafnium Facility (part of the Building B complex) was constructed in 1960 and was operational in 1961, and the plutonium annex was completed and in operation in 1963 (for production of ^{238}Pu sources). The Metals Facility (Building B complex) was constructed in 1962 and was operational in 1963. The Machine Shop (Building B complex) was constructed in 1964. Building C, the Type II Uranium

Facility, was constructed in 1972 in the existing incinerator building (built in 1969), and preproduction of Type II fuel began in 1973 (Author unknown undated a).

The initial functions of the Parks Township facilities were fabrication of plutonium fuel, preparation of HEU fuel, and production of zirconium/hafnium bars under AEC and later NRC License SNM-414 received in March 1961, which allowed the handling of plutonium already on the site. The Parks Township site made fuel for the DOE Fast Flux Test Facility (FFTF) at the Hanford Site in the 1970s and early 1980 (Author unknown 2004). The FFTF fuel was a mixture of PuO_2 and depleted UO_2. The site also made fuel plates for the DOE Zero Power Plutonium (later Physics) Reactor (ZPPR) in the late 1960s and ZPPR-III fuel wafers (Author unknown 2004). Activities included plutonium scrap recovery, DU fabrication, HEU fuel manufacturing, source manufacturing (primarily ^{60}Co, PuBe, and AmBe), irradiated fuel sample examination, laboratory operations, and supporting nuclear power site operations. The activities were conducted in Buildings A, B, and C (Author unknown 2004). Production and process areas are summarized in Table 2-3 and discussed further in Section 2.6. Although many of the dates of operation in Table 2-3 indicate activity through 1980, some of the processes were probably terminated earlier; exact dates could not be determined from the available information. Some of the processing areas are shown in Figure 2-3, which depicts the Parks Township site layout.

Table 2-3. Parks Township site area descriptions.

Building or area location	Operations	Radionuclides	Period of operation
Building A–Plutonium processing facility	Fabrication of plutonium reactor fuel pellets, blankets, rods	PuO_2; Pu nitrate and oxalates (AmBe, PuBe, 1959–1970); alpha, beta, and gamma sources	1960–1980
A–Fab 1	Plutonium conversion, fuel fabrication for FFTF	Plutonium nitrate, plutonium oxide, depleted UO_2	1960–1980
A–East Side of Fab 1	Routine repair and maintenance of contaminated equipment	All	1960–1980
A–Fab 2	Fuel fabrication for ZPPR	Plutonium nitrate, plutonium oxide, depleted UO_2	1962–1980
A–Fab 3	Manufacturing operations, metallography, quality control of FFTF fuel	Plutonium nitrate, plutonium oxide	1963–1980
A–Fab 4	Alpha, beta, gamma, and neutron source fabrication	AmBe double encapsulated, PuBe compacted powder, Ir-192, Cs-137, Be-7, Po-210, Co-60, Am-241, Pu-238/239, PuO_2, plutonium and americium metal	1963–1980
A–Fab 5	Scrap recovery	Plutonium – various forms	1963–August 1, 1967,
	Analytical laboratory work	All, small quantities of radioactive samples	1979–1980
A–Fab 6	Scrap recovery	Plutonium – various forms	1968–1973
A–Fab 7	Fuel rod quality control tests, nonradioactive processes	All, clean and contaminated items	1968–1980
A–Fab 8	Storage	All, clean and contaminated items	1970–1980
A–Fab 9	FFTF fuel pin finishing	Encapsulated nuclear material	1970–1980
Building A–Hot Cell Room	Examination of irradiated samples, high-activity source fabrication	PuBe compacted powder, Co-60, Ir-192	1960–1969
	Storage of sources	Sealed sources, Ir-192, Cs-137, Be-7, Po-210, Co-60, Am-241, Pu-238/239, PuO_2,	1969–1980

Building or area location	Operations	Radionuclides	Period of operation
Plutonium Facility Trailer Storage Area	Storage of large quantities of plutonium and uranium	Plutonium (nonpyrophoric), DU, NU, and EU (to 5% ^{235}U)	1961–1980
Building B–Multipurpose fabrication building	DU, NU, thorium, plutonium	DU metal or alloy, U_3O_8, ^{238}Pu, ThO_2	1961–1980
Building B Hafnium Plant	Metal production	Hafnium and Zr-Be alloy (nonradioactive)	1961–1980
Building B Plutonium Annex	Conversion of ^{238}Pu nitrate to oxide	^{238}Pu nitrate, ^{238}Pu oxalate, PuO_2 powder or alloys	1963–1980
Building B–Hot Cell Room	Large source production: Postirradiation examination of test capsules and fuel pins	Co-60, Cs-137, Ir-192, and PoBe, irradiated, uranium and plutonium and other TRU elements and FPs	1961–1980
Building B–Metals Plant	First floor: small-scale metals production, fuel pellet production, materials testing	First floor: DU, UO_2, U_3O_8, UF_4, fully clad U-233, U-235, and Pu-239	1963–1974
	Second floor: Pu-238 pacemakers	Second floor: Pu-238-powered heart pacemakers	1963–1964
Building B–Machine Shop	Occasional machining of clad or unclad uranium, and clad plutonium and U-233; fabrication and repair of new and contaminated equipment from Parks Township and Apollo; machining of DU	Fully clad U-233 and Pu-239 and clad or unclad U-235 (any enrichment), primarily DU contamination; could include HEU, plutonium, thorium, and mixed FPs	1964–1980
Building C, Type II Facility or T-2 Plant	HEU processing to form sintered product	HEU (1973–1978), soluble chloride/oxide complexes, SNM oxides (UO_3, UO_2 and U_3O_8)	1969–1980
Outdoor Scrap Storage Area	Storage	UF_6 cylinders	1971–1980

a. Sources: Author unknown (2004); Author unknown (undated a); NUMEC (1963)

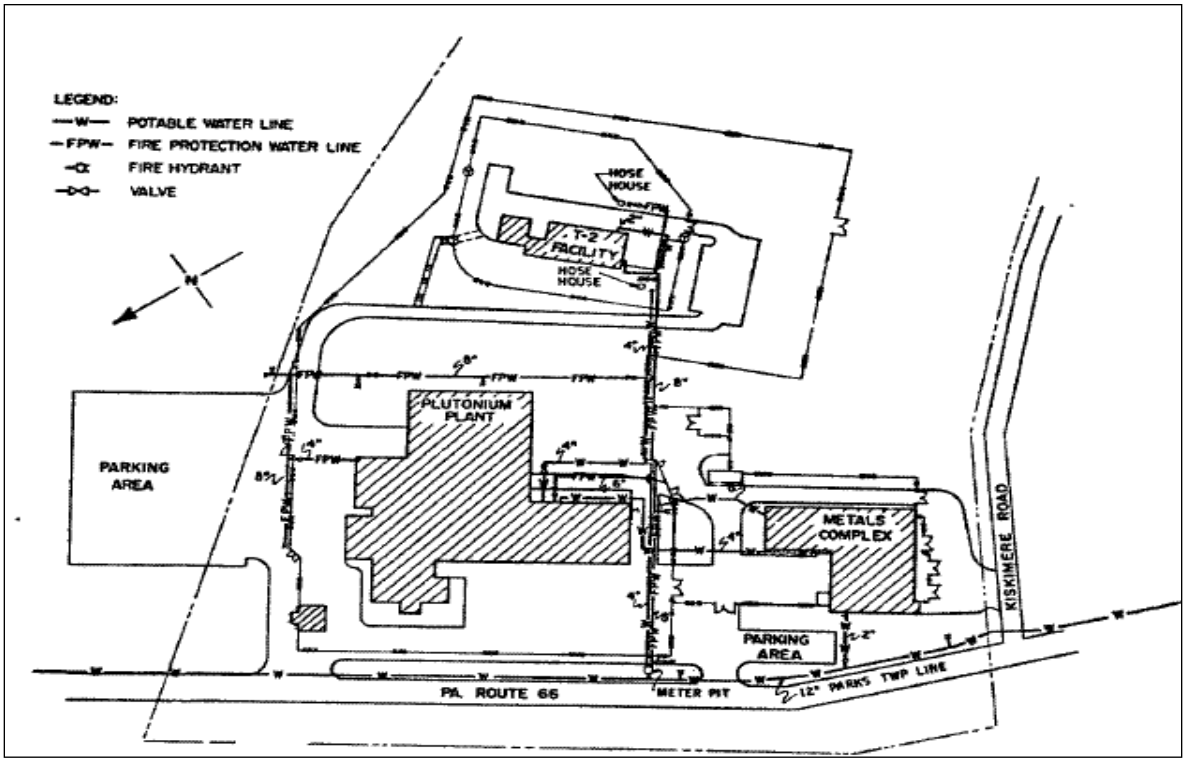

Figure 2-3. Parks Township site layout. Plutonium Plant (Building A), Metals Complex (Building B), and T-2 Facility (Building C) (Austin 1979).

2.2.2 Facilities

Building A
This building was originally a plutonium processing facility and at various times it was known as the Plutonium Laboratory, the Plutonium Building, and the NUMEC Advanced Material Center. The original portion of Building A was designed as a plutonium laboratory to perform R&D that led to plutonium-based products.

Building B
This building was a uranium processing facility with the primary radioactive material being DU, although smaller quantities of NU, thorium, and plutonium were also processed in the building. The main facilities in Building B were the Hafnium Facility, the Metals Facility, and the Machine Shop.

Building C
This facility was built east of Building A in the 1969 to 1972 timeframe. It was used to fabricate HEU fuel, called Type II fuel, from 1973 to 1978. At various times, the building was known as the Type II or T-2 Plant (Author unknown 2004).

Plutonium Plant Storage Area (Trailer)
This was a locked storage area for large quantities of plutonium and uranium.

Scrap Storage Area
This was a 150- by 80-ft outdoor area that was enclosed by cyclone fence. It was guarded 24 hr/d. Scrap was received in criticality-safe shipping containers known as birdcages and stored as received. Specific lots were moved to the Apollo site on NUMEC trucks and logged into the Apollo process storage area on the second floor of the Apollo site.

2.2.3 Process Descriptions

The information in this section is from Author unknown (undated b) unless otherwise noted.

2.2.3.1 Building A – Plutonium Plant

The original portion of Building A was designed as a plutonium laboratory to perform R&D of plutonium-based products for emerging nuclear businesses. Initial operations were authorized in 1960. Many experimental fuel forms and compositions were produced in the 1960s, including oxides, carbides, and metal alloys in the form of plates, powder, pellets, and special shapes. Work with nuclear materials in Building A was conducted in fabrication areas Fab 1 through 7 and Fab 9, and in several small laboratories adjacent to the fabrication areas. Fab 8 was used only for storage of nuclear materials. All significant work on nuclear fuel materials was done in containment systems such as radiochemical hoods and gloveboxes.

The two largest production runs of fuel were ZPPR fuel plates in the late 1960s and FFTF fuel rods in the 1970s and early 1980. More than 12,200 ZPPR fuel elements were produced under contract to Argonne National Laboratory–East (ANL-E) using plutonium-uranium-molybdenum ternary alloy plates encapsulated in stainless steel. The plutonium content was primarily ^{239}Pu, with 11,500 of the plates having 11.5% ^{240}Pu and the balance of the plates having 27% ^{240}Pu. The uranium was depleted. FFTF fuel was the largest order of fuel processed in Building A. More than 50,000 FFTF pins were made; portions of Cores 1 and 2, and the entire Cores 3 and 4. The FFTF fuel was a mixed oxide based on a 20:80 ratio by mass of PuO_2 and UO_2. The plutonium isotopic content was about 86 wt % ^{239}Pu, 12 wt % ^{240}Pu, and 2 wt % ^{241}Pu with trace amounts of ^{242}Pu and ^{238}Pu. The uranium was either natural or depleted, depending on the customer's specifications.

Plutonium scrap recovery was an integral part of Building A operations. Scrap recovery operations were conducted in Fab 5 until August 1, 1967. They were moved in 1968 to a much larger and improved operation in Fab 6, which operated throughout the 1970s and into 1980. The scrap was dissolved in concentrated nitric acid to which a small amount of hydrofluoric acid had been added. The valence of the plutonium ion in the impure nitrate solution was adjusted by the addition of small quantities of ferrous sulfamate and sodium nitrite. The plutonium nitrate solution was purified by passing through ion exchange columns. The purified solution was then concentrated by evaporation, put into bottles, and stored for shipment to the customer or for conversion to plutonium oxide. Conversion of plutonium nitrate to plutonium oxide was performed at the north end of Fab 1 in HEPA-filtered gloveboxes.

Alpha, neutron, and thermal sources were produced in Building A. The two most common neutron sources were double-encapsulated PuBe metallic sources and compacted mixtures of americium oxide and beryllium metal powders. These neutron sources were made in the Fab 4 area. A standard alpha source consisted of a plutonium oxide film that was deposited on one or both sides of a flat metal backing plate. Limited quantities of other neutron, beta, and gamma sources were made to customer specifications. The materials that were used to manufacture these specialty sources included polonium, plutonium, americium, iridium, cesium, cobalt, and beryllium. Source manufacturing always took place in HEPA-filtered gloveboxes, with the exception of high-activity sources that were fabricated in the Building A Hot Cell.

The north end of Building A was divided into two large rooms. The Hot Cell and the Cell Control Area occupied the east room, and the Hot Handling Facilities occupied the west room. The Hot Cell was a reinforced high-density concrete structure that was designed to shield personnel from gamma radiation. The Cell Control Area contained a fume hood for mixing chemicals before inserting them into the cell, and a second fume hood over the fission gas analysis equipment. A metallographic cell was abutted to the west side of the Hot Cell, just north of the sliding doors. Two small steel-walled hot

cells were in the Hot Handling Facilities room. One cell was used as a dissolving cell and the other for storage of radioactive specimens.

The interior of the Hot Cell was at a lower pressure than the exterior to prevent radioactive materials from reaching workers or the environment. Air from the Hot Cell passed through a HEPA filter before it reached the stack.

Gamma sources of ^{192}Ir and ^{60}Co, which required extensive shielding (i.e., a hot cell), and high-yield neutron sources of ^{210}Po/Be were fabricated in the Hot Cell, but the primary work in the Hot Cell was destructive postirradiation examination of test capsules and fuel pins that had been irradiated in research reactors.

Fuel processing and source manufacturing in Building A required support from other systems such as water heaters; heating, ventilation, and air conditioning; natural gas-fired boiler; air compressor; emergency generator; and a cooling tower. Building A housed a repair shop for uncontaminated equipment, a shipping and receiving area, administrative offices, and lunchroom areas. SNM was neither processed nor stored in these support areas. Routine repair and maintenance of contaminated equipment was performed in the glovebox or radiochemical fume hood where the equipment was. More extensive repairs were performed in the Warm Maintenance Area, which contained a series of ventilated HEPA-filtered gloveboxes that contained a lathe, drill press, and other required equipment.

All plutonium gloveboxes and fume hoods were removed from Building A during a 1981 to 1983 deactivation program, during which most of the effluent streams that existed during the years of plutonium fuel production were eliminated. The workload in Building A shifted to repair and refurbishment of contaminated equipment that had been used at reactor sites, building decontamination, and low-level radioactive waste (LLRW) volume reduction services for commercial customers.

Although these operations involved much smaller quantities of radioactive isotopes, they still generated radioactive contamination, so the building exhaust air continued to require HEPA filtration before exiting through roof stacks. This exhaust was monitored to ensure compliance with existing regulations. As commercial work slowed in the mid-1990s, the pace of building decontamination increased.

2.2.3.2 Building B – Multipurpose Fabrication Building

Building B was constructed in three stages beginning in 1961 when the Hafnium Facility was built to produce crystal-bar hafnium. The second stage of construction occurred in 1963 when the Metals Plant was built to the east of the Hafnium Facility. The third stage occurred in 1964 when the space between the Hafnium Facility and the Metals Plant was closed in to create the Machine Shop. Later in its life, the combined facility became known as the Metals Building and then as Building B.

DU was the primary radioactive material processed in Building B, but smaller quantities of NU, thorium, and ^{238}Pu were also processed. The DU was primarily in the form of metal or metal alloy, and the processing consisted mostly of forming (rolling, etc.) and machining operations that did not generate significant airborne emissions. A limited amount of powder products was produced at the northeast end of Building B. Plutonium-238 was processed in a room in the northwest corner. All ^{238}Pu work was performed in interconnected gloveboxes. Receiving and shipping operations were conducted in a chemical fume hood. In addition, nonradioactive metals and alloys were processed in significant quantities in Building B. Most of the work was production of crystal-bar zirconium and hafnium and zirconium-beryllium alloys.

2.2.3.2.1 The Hafnium Facility

The original product from the Hafnium Facility was crystal-bar hafnium. Crystal-bar zirconium was produced in the Hafnium Facility using a similar process. A specialty zirconium alloy product was produced consisting of zirconium-beryllium-titanium alloy powder. Metal powders of other alloys were produced in the Metals Plant with an identical process.

Under contract with AEC, ^{238}Pu nitrate was converted to an oxide product in a room in the northwest corner of the Hafnium Facility known as the Plutonium Annex. The conversion process was similar to the process for converting ^{239}Pu nitrate to fuel products in Building A, but only oxalate precipitation was used. The conversion was performed in eight HEPA-filtered gloveboxes.

2.2.3.2.2 The Machine Shop

The Machine Shop between the Hafnium Facility and the Metals Plant was used to fabricate equipment and machine metals in support of the production lines at the Apollo and Parks Township sites. The equipment in the Machine Shop included drill presses, lathes, shears, formers, grinders, polishers, welders, and sandblasting, degreasing, and other metalworking machinery. Machining of DU was performed in the Machine Shop. In addition, repair and refurbishment of equipment from the Apollo and Parks Township sites was performed. Some of this equipment contained levels of radioactivity that exceeded the criteria at that time for release for unrestricted use.

The machining operations took place on the ground floor. The second floor contained primarily offices and a training room, although a small environmental laboratory was at the south end of the second floor until 1991. After 1991, most of the second floor was used as office space until decontamination operations started in Building B in late 1996.

2.2.3.2.3 The Metals Plant

The Metals Plant was built in 1962 and was operational in 1963. The original layout of the first floor of the Metals Plant included equipment to process various metals including zirconium-beryllium, tantalum, magnesium, copper, nickel, cadmium, and uranium. The uranium operations included electroplating, melting, grinding, and powder handling involving DU.

Metals production from the Metals Plant was small scale and intermittent. Most of the processing equipment was removed for resale or disposal in 1973 and 1974.

The second floor of the Metals Plant initially contained only one office. However, over the years offices were added along with two physical and mechanical testing laboratories for quality control testing, and the Energy Conversion Laboratory (also called the R&D Laboratory) where R&D projects were performed such as the development of ^{238}Pu-fueled heart pacemakers, under an AEC sealed-source license.

2.2.3.3 Building C – Highly Enriched Uranium Processing Facility

Combined with the general expansion of Building A in 1969 and 1970, a new building was erected to the east of Building A and called the Incinerator Building. In 1972, the building was modified to include facilities for processing HEU. The building sat unused until 1973 when the company received a contract to fabricate an HEU product, and processing of SNM in the building was authorized by the AEC as an amendment to SNM-414 (Author unknown undated b).

The manufacturing operations involved dissolving HEU in a solution of hydrochloric acid (HCl) and hydrogen peroxide, then diluting the solution with demineralized water. The diluted uranium solution was fed through dialysis columns and an electrolysis cell. The solution then passed through forming

columns to create a solid sintered form (Reitler 1973a). The solid material was rinsed, dried, and placed in a furnace. The material was placed into containers and stored before being shipped to a licensed site for finishing operations. Most of the processing operations were conducted in gloveboxes, radiochemical fume hoods, or other ventilated HEPA-filtered enclosures. In addition, the room air from the building was exhausted through HEPA filters.

Materials processing produced several types of liquid wastes: process, laboratory, hexanol, utilities and blowdown, and sanitary. Uranium-rich liquid process waste was concentrated in a boildown unit and transported, along with solid waste that contained recoverable amounts of uranium, to the Apollo site for recovery.

2.2.4 Source Term

There are three main sources that describe the amounts and types of radioactive material that were handled at the Parks Township site: (1) Federal and State licenses for the possession and use of radioactive materials; (2) descriptions and reviews of proposed experiments or jobs handling radioactive material in the form of safety and process feasibility reports, which contain information about radionuclides, quantities, and recommended safety precautions for the described activity; and (3) inventory and material handling (accountability) records.

Some possession limits at different periods are listed for the Parks Township facilities in Tables 2-4 through 2-7.

The use of SNM was governed by AEC regulations and licenses under License SNM-414 issued by the AEC in 1961 (Docket No. 70-364).

The Parks Township site radiological source term included uranium, thorium, plutonium, and fission and activation products. No definitive information is available to relate measurement of one component of the source term (e.g., plutonium) to another unmonitored component (e.g., americium) for a given area or process. Much of the work was R&D, so unique source terms could be encountered in a particular job.

Uranium. Uranium in the form of metals, oxides, and carbides was used for Parks Township fuel fabrication (Building A), uranium fuel product (Building C), and reactor fuel research studies in the hot cells and laboratories. The typical amounts of uranium in use in any area ranged from milligrams to hundreds of kilograms. Work with chemical forms of uranium such as UF_6 or uranyl nitrate was occasionally conducted. Uranium forms included DU, NU, and EU (up to 93.5%), as well as ^{232}U, ^{233}U and ^{236}U. Uranium from recycling operations would have included relatively small activities of nonuranium isotopes such as ^{99}Tc, ^{237}Np, ^{230}Th, and ^{239}Pu.

Thorium. Thorium dioxide was used at the Parks Township site in preparation of special reactor fuel. The total mass of thorium that was used on site was probably less than that of uranium, but the thorium activity in use in an area at a given time could have been greater or less than uranium activity. Thorium dioxide was obtained from virgin thorium sources.

Plutonium. Chemical forms included metals, nitrates, and oxides. The heat source and heart pacemaker programs used ^{238}Pu. The heart pacemaker program used ^{238}Pu nitrate as a starting material. The reactor fuel projects used a $^{239/240}Pu$-dominated source term. There could be ^{241}Am associated with the plutonium source term; in 5 years, the ^{241}Am ingrowth would account for about 1% of the total radioactivity in a reactor source term. The ^{241}Pu activity in a heat source is initially less than 5% of the total radioactivity (ORAUT 2004).

Table 2-4. Parks Township Building A source and SNM possession limits.

Source/chemical or physical form	Period	Maximum possession	Reference
Plutonium and EU	1961–1969	Any combination of plutonium and U-235 up to 400 kg	Nussbaumer 1965
	1969–1979	Any combination of plutonium and U-235 up to 1,000 kg	AEC ca. 1965–1969
Plutonium (in nonpyrophoric form, containing at least 3 wt % Pu-240)	1979–1991	Up to 1,000 kg fissile	Rouse 1979; Austin 1981
	1991–end	Less than 200 g	Haughney 1991
EU of less than 5 wt % U-235	1969–1991	25,000 kg	AEC ca. 1965–1969; Rouse 1979; Austin 1981
EU above 5 wt % U-235	1979–1981	Possession: < 5 kg ^{235}U Use: 1 kg effective	Rouse 1979
	1981–1991	50 kg fissile	Austin 1981
EU above 5 wt % U-235 in storage	1979–1991	50 kg fissile	Rouse 1979
Uranium, any enrichment U-235	1991–end	Less than 250 g U	Haughney 1991
NU or DU and thorium	1969–1979	No limits	AEC ca. 1965–1969
NU or DU any form	1979–1991	100,000 kg U	Rouse 1979; Haughney 1991
Plutonium with greater than 5 wt % Pu-238	1969–1979	300 g	AEC ca. 1965–1969
Pu-238 as oxide or metal	1979–1991	60 g	Rouse 1979; Austin 1981
Pu-238 as sealed source	1981–1991	60 g	Austin 1981
Pu-239 as electroplated calibration or reference sources	1981–1991	10 g	Austin 1981
	1991–end	20 g	Haughney 1991
Pu-239 as evaporated calibration or reference sources	1981–end	5 g	Austin 1981
Pu-239 as encapsulated calibration or reference sources	1981–1984	230 g	Austin 1981
	1984–1991	50 g	Austin 1984
	1991–end	285 g	Haughney 1991
U-235 as evaporated calibration or reference sources	1981–end	5 g	Austin 1981; Haughney 1991; Austin 1984
U-235 as encapsulated calibration or reference sources	1984–end	5 g	Haughney 1991; Austin 1984
U-235 as electroplated calibration or references sources	1991–end	5 g	Haughney 1991
U-233	1961–1979	4 kg	Nussbaumer 1965; AEC ca. 1965–1969
U-233 as evaporated calibration or reference sources	1981–1991	1 g	Austin 1981, 1984
	1991–end	2 g	Haughney 1991
Any fissile radioactive material encapsulated to meet 49 CFR 173.398 requirements for special form material	1969–1972	300 g	AEC ca. 1965–1969
Byproduct material encapsulated	1979–end	10 Ci per source of each isotope	Rouse 1979; Austin 1984; Haughney 1991
Byproduct material any form	1979–end	20 mCi of any isotope	Rouse 1979; Austin 1984; Haughney 1991
Byproduct material any form, contaminated waste	1984–end	1,000 Ci of any isotope	Austin 1984; Haughney 1991
Byproduct material any form, contaminated waste on/in equipment	1991–end	5 Ci	Haughney 1991

Source/chemical or physical form	Period	Maximum possession	Reference
Byproduct material any form, contaminated waste on/in equipment and metallic materials from other licensees	1984–end	5,000 Ci	Austin 1984; Haughney 1991
Byproduct material any form, contamination in volume reduction services waste	1984–end	500 Ci	Austin 1984; Haughney 1991

Table 2-5. Parks Township Building B source and SNM possession limits.

Source/chemical or physical form	Period	Maximum possession	Reference
EU of 5 wt % U-235	1961–1979	5,000 kg	Puechl 1965; AEC ca. 1965–1969
	1991–end	Possession: <250 g U-235	Haughney 1991
EU above 5 wt % U-235	1961–1979	500 kg	Puechl 1965; AEC ca. 1965–1969; Rouse 1979
	1981–1991	Possession: <5 kg Use: 1 kg effective	Austin 1981
	1991–end	Possession: <700 g U-235	Haughney 1991
U-233	1961–1979	4 kg	Puechl 1965
Pu-239 with at least 3 wt % Pu-240	1961–1969	250 kg	Puechl 1965
Plutonium as fully clad, encapsulated, or otherwise contained material in operating areas or in any form in the storage vault	1969–1979	500 kg	AEC ca. 1965–1969
NU or DU and Th	1969–1979	No limits	AEC ca. 1965–1969
Plutonium nonpyrophoric form	1991–end	<200 g	Haughney 1991
Pu-238 encapsulated	1979–1981	60 g	Rouse 1979
Byproduct material any form	1979–end	20 mCi of any isotope	Rouse 1979; Haughney 1991
Byproduct material encapsulated	1979–1984	10 Ci per source of each isotope	Rouse 1979, Austin 1984
	1984–1991	5 Ci per source of each isotope	Austin 1984; Haughney 1991
	1991–end	10 Ci per source of each isotope	Haughney 1991
Byproduct material any form, contaminated waste on/in equipment	1991–end	5 Ci	Haughney 1991
Byproduct material any form, contaminated waste on/in equipment, and metallic materials from other licensees	1991–end	5,000 Ci	Haughney 1991
NU or DU, any covered or authorized activities	1991–end	100,000 kg U	Haughney 1991
Pu-239 as electroplated calibration or reference source	1981–1991	5 g	Austin 1981
	1991–end	20 g	Haughney 1991
Pu-239 as encapsulated calibration or reference sources	1981–1991	50 g	Austin 1981
	1991–end	285 g	Haughney 1991
Pu-241 as electroplated calibration or reference source	1981–1991	5 g	Austin 1981
Pu-241 as encapsulated calibration or reference source	1991–end	5 g	Haughney 1991
U-233 as evaporated calibration or reference sources	1981–1991	1 g	Austin 1981
	1991–end	2 g	Haughney 1991
U-235 as evaporated calibration or reference sources	1991–end	5 g	Haughney 1991

Source/chemical or physical form	Period	Maximum possession	Reference
U-235 as encapsulated calibration or reference source	1991–end	5 g	Haughney 1991
U-235 as electroplated calibration or reference source	1981–end	5 g	Austin 1981, Haughney 1991

Table 2-6. Parks Township Building C source and SNM possession limits.

Source/chemical or physical form	Period	Maximum possession	Reference
EU above 5 wt % U-235	1973–1978	HEU[a]	
NU or DU and thorium	1969–1979	No limits	AEC ca. 1965–1969
EU above 5 wt % U-235	1979–1991	Possession: <5 kg U-235 Use: 1 kg effective	Rouse 1979, Austin 1981
	1991–end	Possession: <700 g U-235	Haughney 1991
U-235 as encapsulated or electroplated calibration or reference sources	1981–1991	5 g	Austin 1981
	1981–1991	Possession: <5 kg Use: 1 kg effective	Austin 1981
	1991–end	Possession: <700 g U-235	Haughney 1991
Pu-239 as encapsulated calibration or reference sources	1981–1991	5 g	Austin 1981
	1991–end	285 g	Haughney 1991
Pu-239 as electroplated calibration or reference sources	1981–	5 g	Austin 1981
	1991–end	20 g	Haughney 1991
Any isotope encapsulated in one or more sealed sources	1984–1991	10 Ci	Austin 1984
Plutonium in nonpyrophoric form	1991–end	<200 g	Haughney 1991

a. Possession limits could not be found from available information, but are probably similar to the possession limits for HEU work at the Apollo facility (75,000 kg HEU).

Table 2-7. Parks Township other facility source and SNM possession limits.

Place of use	Source/chemical or physical form	Period	Maximum possession	Reference
LLRW storage areas	Radioactive fissile material in approved storage containers	1969–1984	100 g/container	AEC ca. 1965–1969
Plutonium plant outdoor storage area	EU up to 5 wt % U-235 in UF_6 cylinders	1972–1973	75,000 kg UF_6	Browne 1978
		1973–1984	200,000 kg UF_6	Browne 1978
		1984–1991	100,000 kg UF_6	Austin 1984
Plutonium plant storage trailer	Plutonium in nonpyrophoric form with at least 3 wt % Pu-241	1979–1991	Up to 1,000 kg fissile	Rouse 1979
		1991–end	<200 g	Haughney 1991
	Plutonium and U-235 in approved shipping containers with valid certificates of compliance	1984–1991	Any quantity	Haughney 1991
	EU up to 5 wt % U-235, any physical or chemical form covered by authorized activities	1979–1991	25,000 kg U	Rouse 1979
	NU or DU, any physical or chemical form covered by authorized activities	1979–end	100,000 kg U	Rouse 1979
	Uranium, any enrichment U-235	1991–end	Possession: <250 g U	Haughney 1991
	Byproduct material any form, contaminated waste	1991–end	1,000 Ci of any isotope	Haughney 1991
	Byproduct material, any form, contaminated waste on/in equipment	1991–end	5 Ci	Haughney 1991

Place of use	Source/chemical or physical form	Period	Maximum possession	Reference
	Byproduct material, any form, contaminated waste on/in equipment, and metallic materials from other licensees	1991–end	5,000 Ci	Haughney 1991
	Byproduct material, any form, contamination in volume reduction services	1984–end	500 Ci	Austin 1984, Haughney 1991
Storage areas	EU to >5 wt % U-235	1991–end	Possession: <700 g U-235	Haughney 1991
Outside storage areas	EU of any enrichment in U-235	1991–end	Possession: <350 g U	Haughney 1991
	Byproduct material, any form, contaminated waste	1991–end	1,000 Ci of any isotope	Haughney 1991
	Byproduct material, any form, contaminated waste on/in equipment, and metallic materials from other licensees	1991–end	5,000 Ci	Haughney 1991
Any state except Agreement States	Neutron irradiator source	1969–unknown	Up to 96 g plutonium as PuBe neutron source	AEC ca. 1965–1969
	Byproduct material, any form, contaminated waste on/in equipment	1991–end	5 Ci	Haughney 1991
	Byproduct material, any form, contaminated waste on/in equipment, and metallic materials from other licensees	1991–end	5,000 Ci	Haughney 1991

The two largest production runs of fuel were ZPPR fuel plates in the late 1960s and FFTF fuel rods in the 1970s and early 1980. More than 12,200 ZPPR fuel elements were produced under contract to ANL-E using plutonium-uranium-molybdenum ternary alloy plates encapsulated in stainless steel. The plutonium content was primarily ^{239}Pu, with 11,500 of the plates having 11.5% ^{240}Pu and the balance of the plates having 27% ^{240}Pu. The uranium was depleted. FFTF fuel was the largest order of fuel processed in Building A. More than 50,000 FFTF pins were made – portions of Cores 1 and 2 and all of Cores 3 and 4. FFTF fuel was a mixed oxide based on a 20:80 ratio by mass of PuO_2 and UO_2. The plutonium isotopic content was about 86 wt % ^{239}Pu, 12 wt % ^{240}Pu, and 2 wt % ^{241}Pu with trace amounts of ^{242}Pu and ^{238}Pu. The uranium was natural or depleted depending on the customer's specifications (Author unknown undated b).

Other Sources. Various sources were manufactured including AmBe, PuBe, and PoBe neutron sources; ^{192}Ir, ^{137}Cs, ^{7}Be, and ^{60}Co beta/gamma sources; ^{210}Po, ^{241}Am, and $^{238/239}Pu$ alpha sources; and ^{238}Pu heat and pacemaker sealed sources.

2.2.5 Remediation, Decontamination, and Decommissioning

The Parks Township site ceased work for DOE operations in 1980. Decontamination and decommissioning of the facilities began in 1978 at Building C, and continued through several phases for all facilities. Starting in 1994, B&W began final decontamination and decommissioning at the site to the extent permitted under the terms of its license. In January 1996, B&W submitted a site-wide decontamination and decommissioning plan and subsequent plan revisions in 1997 and 1998. In October 1998, NRC approved Revision 3.1 of the plan. Demolition and removal of all facilities was started at that time. All decommissioning activities had been completed by January 2002. All waste had been shipped to a licensed waste disposal facility, and the final status survey had been

performed. After B&W completed 2 years of groundwater monitoring that showed site groundwater was within established limits, the NRC terminated the license and released the site for unrestricted use on August 24, 2004 (PDEP 2008).

Building A
In 1980, B&W began dismantling the fuel fabrication lines to allow Building A use for other operations. Process and analytical equipment, gloveboxes, and hoods were decontaminated and removed. After the removal of this equipment, B&W used the area for commercial decontamination. In 1982, B&W used areas of the building for nuclear power site support operations. These activities continued into 1990 and involved the maintenance, testing, and refurbishment of equipment and materials that were contaminated with mixed fission and activation products. In the mid-1980s, a facility for LLRW volume reduction was under preparation, but the project was terminated in 1988 before operations were started (Author unknown undated a).

Building B
Decommissioning of the Hafnium Facility and Metals Facility started in 1976 with the removal of process equipment, which was sent for burial or offered for sale (Author unknown undated a). A radiation survey of the Metals Facility was performed during September 1980. From 1983 to 1986, the Metals Facility was used for storage of nuclear power plant spare parts (Author unknown undated a). As of 1991, the facilities were used for nondestructive assay and for calibration and testing in relation to decontamination, maintenance, and storage of nuclear industry equipment (Haughney 1991). Final decommission of the facility was included in the final site remediation, which started in 1998.

Building C
In 1978, B&W ended all HEU operations and began decommissioning efforts at the Parks Township Type II facility (Building C). Decommissioning included removal of all process and related equipment and the disposal of the contents of the discard ponds associated with Building C. Pond remediation included (1) removing the liquids and sludge and solidifying them in 55-gal drums for shipment for burial, (2) breaking up the asphalt liner for packaging in wooden boxes and steel drums for shipment for burial, and (3) packaging soil higher than background into boxes or drums for shipment for burial. Decommissioning of the process equipment started with each piece of dismantled equipment being assayed for the quantity of SNM. All possible SNM was removed. Equipment was packaged for burial. After the equipment was removed, B&W initiated a cleanup of the walls, floors, and ceilings to remove loose surface contamination. Surface areas known to contain high-level fixed contamination were chipped away and packaged for shipment for burial. The residual activity was determined to be fixed and inaccessible to diversion. Access to the building was restricted to authorized personnel, and the building's entrances were secured (Martin 1982). Initial decommissioning was completed in May 1979. During 1979, drums of U_3O_8 were temporarily stored at the facility and were shipped in 1980 (Author unknown undated a). Additional decommissioning was performed during September 1981. Final decommissioning was included in the final site remediation that started in 1998.

3.0 OCCUPATIONAL MEDICAL DOSE

The information in this section applies to the Apollo and Parks Township sites. NUMEC apparently did not have its own medical X-ray department during AEC operational years and the medical X-rays for NUMEC employees appear to have been performed at a local clinic or hospital. Therefore, in compliance with ORAUT-OTIB-0079, *Guidance on Assigning Occupational X-Ray Dose Under EEOICPA for X-Rays Administered Off Site* (ORAUT 2011a), no occupational medical dose should be assigned.

4.0 ENVIRONMENTAL OCCUPATIONAL DOSE

The Apollo site petition evaluation report for petitions SEC-00047 and SEC-00080 (NIOSH 2007a) determined that it is not feasible to reconstruct ambient environmental dose from 1957 through 1965 for the Apollo site based on limitations associated with stack monitoring data. Reliable information for the period after 1965 could not be found to bound the internal and external ambient dose, as described below.

The Parks Township site petition evaluation report for petition SEC-00108 (NIOSH 2008a) did not address ambient environmental dose, and reliable information could not be found to bound the internal and external ambient dose for the site.

4.1 ENVIRONMENTAL INTERNAL DOSE

Adequate information on environmental air concentrations near the NUMEC Apollo and Parks Township sites was not found. Therefore, no estimates of internal ambient dose can be made for workers for any period.

4.2 ENVIRONMENTAL EXTERNAL DOSE

Information on ambient external dose levels at the NUMEC Apollo and Parks Township sites were not found. Therefore, no estimates of external ambient dose can be made for workers for any period.

5.0 OCCUPATIONAL INTERNAL DOSE

Occupational internal dose is the dose received by an individual from an intake of radioactive material while performing tasks in buildings and structures at the Apollo and Parks Township sites or from activities outside the buildings, such as handling materials in storage yards. This section describes NUMEC internal dosimetry systems and practices and provides supporting data to evaluate internal doses that can reasonably be associated with worker radiation exposures covered by EEOICPA. The health and safety coverage for both sites were administered by one department. This section covers exposure at both facilities because it is difficult to distinguish bioassay results between the sites.

5.1 INTERNAL EXPOSURE SOURCES

The primary sources of internal radiation exposure at the Apollo site were uranium, with some potential for exposure to plutonium or thorium dust from the manipulation and chemical processing of those materials during uranium scrap recovery and fuel fabrication processes. Uranium enrichment levels included depleted, natural, low (3.5%), and high (93%). Exposure to mixed fission and activation products was possible at some locations (Laundry Building).

The sources of internal radiation exposure at the Park Township site were uranium, plutonium, and thorium and chemical processing of those materials during plutonium scrap recovery and fuel fabrication processes. Uranium enrichment levels at the site included depleted, natural, low (3.5%), and high (93%). Exposure to other radionuclides was possible for workers who were involved in source fabrication (^{7}Be, ^{60}Co, ^{137}Cs, ^{192}Ir, ^{210}Po, ^{241}Am, and ^{238}Pu.)

Table 5-1 lists the enrichments and chemical forms of processed radionuclides for the Apollo and Parks Township sites.

ICRP (1994) lists UF_6, UO_2F_2, and $UO_2(NO_3)_2$ (uranyl nitrate) as type F; UF_4 and UO_3 as type M; and U_3O_8 and UO_2 as type S. The chemical forms and the enrichments varied over time at the NUMEC sites. The manufacture of uranium products occurred in most of the buildings at Apollo and Parks

Table 5-1. Fuel types, chemical form, isotope, and enrichment of NUMEC process material.[a]

Radionuclide or fuel	Chemical form and solubility type(s)[b]	Isotope (% in mass, where listed)	Enrichment
Uranium	UF_6, UO_2F_2, & $UO_2(NO_3)_2$ (F) UO_3 & UF_4 (M) U_3O_8 & UO_2 (S)	U-234 U-235 U-238	DU, NU, LEU (3.5%), HEU
Thorium[c]	ThO_2 (M, S)	Th-228, Th-232	Natural thorium
Plutonium[d]	PuO_2 (M, S, SS)	Pu-238 0.64%, Pu-239 2.06%, Pu-240 1.07%, Pu-241 95.4%, Am-241 0.86%--Activity	Fuel grade Aged 10 years
Technetium or other TRU elements	Same as the thorium, uranium, or plutonium matrix	Tc-99, Np-237	Not applicable
MOX[e]	PuO_2 (M, S, SS) UO_2 (M, S)	20% PuO_2 and 80% UO_2 [7% plutonium – fuel grade/5% plutonium – weapons grade]	About 4.5% ^{235}U
Fission and activation products	Unknown	Be-7, Co-60, Sr-90, Ru/Rh-106, Cs-137, Tc-99 (from ruthenium), Ir-192	Not applicable

a. Sources: Author unknown (2004), NUMEC (1963).
b. SS refers to highly insoluble plutonium (type Super S).
c. All thorium work was with unirradiated thorium material.
d. Only small amounts of plutonium were licensed for the Apollo site.
e. Mixed oxide (MOX) work was probably limited to the Parks Township site.

Township. See Tables 2-1 and 2-4 for more information. The dose reconstructor should use the solubility type that results in the highest dose.

Table 5-2 lists NUMEC-specific uranium source term information for various enrichments. For a given uranium process, the mass of (long-lived) uranium released to air does not change because of enrichment.

Table 5-2. Uranium source term information.

Uranium source term	Reference	Specific activity (pCi/µg)	Activity fractions			
			U-234	U-235	U-236	U-238
NU	IMBA[a]	0.683	0.489	0.023	-	0.489
93.%	IMBA[a]	68.1	0.968	0.030	0.002	0.0003
3.5%	IMBA[a]	2.20	0.818	0.034	-	0.147
2%	HPS[b]	1.20	0.648	0.041	0.0009	0.311
Typical DU	IMBA[a]	0.402	0.155	0.011	0.0005	0.834
Uranium source term	**Reference**	**Specific activity (pCi/µg)**	**Specific constituent activity in mixture (µCi/g, nCi/mg, or pCi/µg)**			
NU	IMBA[a]	0.683	0.334	0.016	-	0.334
93.%	IMBA[a]	68.1	65.9	2.04	0.136	0.020
3.5%	IMBA[a]	2.20	1.80	0.075	-	0.323
2%	HPS[b]	1.20	0.778	0.049	0.001	0.373
Typical DU	IMBA[a]	0.402	0.062	0.004	0.0002	0.335

a. IMBA = Integrated Modules for Bioassay Analysis software.
b. American National Standards Institute N13.22 (HPS 1995).

Many forms of plutonium were possible over the years including metal and oxides. Because the feasibility reports for the recovery or manufacture of plutonium have not been located, the exact amount processed of each chemical form is not known.

In general, plutonium oxides, carbides, and hydroxides are absorption type S; nitrates and other compounds are type M (ICRP 1995, p. 299). Older materials, even when starting out as soluble, can have a tendency to oxidize when left in contact with air. Oxides, metals, and old contamination should

be treated as type S. If nothing is known about the chemical form of plutonium, either type M or S can be used to maximize the dose to the organ of concern. In addition, because highly insoluble forms of plutonium might have been present, guidance in ORAUT-OTIB-0049, *Estimating Doses for Plutonium Strongly Retained in the Lung* (ORAUT 2010a), should be followed for the evaluation of highly insoluble (Super S) plutonium. Americium-241 is a component of plutonium contamination and should be modeled in the lung the same as the plutonium matrix in which it has grown. In other words, the americium should be treated as absorption type S if the plutonium is type S (ORAUT 2010b). If the plutonium is type Super S, follow guidance in ORAUT (2010a) for assignment of the ^{241}Am solubility type.

There are essentially three types of plutonium-based material: reactor grade, weapons grade, and fuel grade, which falls between reactor and weapons grades. For this section, lacking specific information on the actual composition of the processed plutonium, the composition of Hanford plutonium can be used because this was the source of plutonium for FFTF fuel fabrication (Author unknown 2004). The activity composition for Hanford reference fuel-grade plutonium (12%) is listed in Table 5-3 for fuel aged up to 20 years (ORAUT 2010b). The age of plutonium to assume for a given analysis depends on the radionuclide measured in the bioassay analysis. When ^{238}Pu or ^{239}Pu is measured, the dose is maximized by assuming longer decay times (20 years). When ^{241}Am is measured and the intake is estimated using ingrowth of ^{241}Am from decay of ^{241}Pu, the dose is maximized by assuming a short (5-year) decay time. A best estimate of intake can be made by assuming a 10-year decay time because this is midway between the possible low and high ages of plutonium from the Hanford Site. If the actual age of the fuel is known (such as from an incident investigation report), that age can be used in the intake and dose analysis.

Table 5-3. Activity composition of Hanford reference fuel-grade plutonium mixture (12%).[a]

Mixture designation	**Specific activity (Ci/g)**				
Years of aging[b]	**0**	**5**	**10**	**15**	**20**
Pu-238	1.71E-02	1.64E-02	1.58E-02	1.52E-02	1.46E-02
Pu-239	5.26E-02	5.26E-02	5.26E-02	5.26E-02	5.26E-02
Pu-240	2.72E-02	2.72E-02	2.72E-02	2.72E-02	2.72E-02
Pu-241	3.09E+00	2.43E+00	1.91E+00	1.50E+00	1.18E+00
Pu-242	3.93E-06	3.93E-06	3.93E-06	3.93E-06	3.93E-06
Am-241	0.00E+00	2.19E-02	3.89E-02	5.22E-02	6.24E-02

a. Source: ORAUT (2010b).
b. Time since separation of Am-241 from the plutonium mix.

5.2 *IN VITRO* BIOASSAY

The bioassay program for NUMEC workers primarily involved urine and fecal sampling for isotopes of uranium, plutonium, and ^{241}Am. Occasional analyses were performed for fission products (FPs) and ^{232}Th. The reported bioassay data generally include a measurement error that indicates the detection level. The bioassay analyses are described in the following sections.

Several bioassay vendors were used to evaluate *in vitro* bioassay samples for the NUMEC sites. The SEC evaluations for the Apollo site (NIOSH 2007a) and the Parks Township site (NIOSH 2008a) indicated that Controls for Environmental Pollution has been implicated in the falsification of data and that its bioassay analyses provided to NUMEC cannot be considered reliable. Bioassay data from Controls for Environmental Pollution should be used only to indicate the potential for exposure to a particular radionuclide on a particular date. The data cannot be used in a dose reconstruction to evaluate intakes or assign internal dose.

The *in vitro* bioassay records for individuals nearly always include an indication of the detection level for the measurement. Dose reconstructors should use the listed detection level information in evaluation of intakes for specific radionuclides when available.

5.2.1 Plutonium Urine and Feces Bioassay

Plutonium might have been present at the NUMEC sites in several forms that include type M, type S, and possibly type Super S material solubility categories. The intake analysis based on bioassay monitoring results should evaluate intakes based on all three types and use the type that provides the highest dose estimate.

5.2.1.1 Plutonium Urine Minimum Detectable Concentrations and Frequencies

Plutonium-239 was analyzed in urine from about 1962 to 1999, and ^{241}Am was analyzed starting in about 1966. The minimum detectable concentrations (MDCs) are listed in Table 5-4 for NUMEC facilities. If an MDC value is needed before the dates in the table, the values for the earliest date should be used. No bioassay monitoring results were found between 1985 and 1999. In addition, because one health physics department was responsible for the bioassay program at both sites, it is difficult to determine from the reported results if the employee worked at the Apollo or Parks Township site. It is likely that much of the plutonium bioassay results were for work at the Parks Township site.

5.2.1.2 Plutonium Urine Analytical Procedure

Information on the specific procedure used to analyze for plutonium in urine is not known. Based on bid specifications (Author unknown undated c), the early analytical procedure probably consisted of drying 500 mL of urine to dryness with nitric acid (HNO_3). The residue was reevaporated successively with HNO_3 and then 30% hydrogen peroxide (H_2O_2) and washed again with HNO_3. The ash was dissolved in 2N HNO_3 and transferred to a lusteroid centrifuge cone. Hydroxylamine hydrochloride, lanthanum carrier, and hafnium were added, and the plutonium was coprecipitated with LaF_3. After centrifuging, the precipitate was dissolved in aluminum nitrate solution and the plutonium oxidized to plutonium (IV) with sodium nitrite ($NaNO_2$). Plutonium was extracted into 2-thenoyltrifluoroacetone and back-extracted into 8N HNO_3. The aqueous phase was evaporated on a planchet and flamed to remove organic residue. The planchet was counted in a Nuclear Measurement Corporation gas flow proportional counter for 4 hours. The minimum sample volume was 500 mL. Because nearly weightless samples were obtained in the procedure, no absorption corrections were made. The sensitivity for this procedure was expected to be about 0.44 ±0.20 dpm/L in 1964 (Author unknown undated c).

Procedures used to analyze for plutonium in urine in later years are not known.

From a review of the worker dosimetry records, once per quarter seemed to be the average frequency. Special bioassays were ordered for workers who exceeded 40 maximum permissible concentration-hours (MPC-hr) of exposure or nose wipes exceeding 25 dpm.

5.2.1.3 Plutonium Fecal Minimum Detectable Concentrations and Frequencies

The analytical procedure for plutonium fecal analysis has not been located. The estimated MDCs are listed in Table 5-5 for NUMEC facilities.

Fecal sampling was initiated in January 1966 at the NUMEC facilities. Three goals of the program were (1) the early detection of acute inhalation exposures, (2) estimation of detected lung burdens, and (3) screening for potential chronic exposures (Caldwell 1966). The fecal analyses continued until about 1985 as indicated in worker dosimetry records. The results, reported as dpm/sample, should be considered equivalent to the daily excretion rate (dpm/d).

Table 5-4. Plutonium and americium urine bioassay MDC, frequency, and period.[a,b]

Date	Laboratory	Radionuclide	Frequency[b]	MDC[c,d]	Error[e]
10/1961–12/1965	Controls for Radiation	Plutonium	Quarterly/as needed	0.28 dpm/L	0.01–0.48 dpm/L
1/1966–12/1968	Eberline	Pu-238	Quarterly/as needed	0.06 dpm/sample	0.03 dpm/sample
1/1966–12/1975	Eberline	Pu-239	Quarterly/as needed	0.06 dpm/sample	0.03 dpm/sample
1/1966–12/75	Eberline	Am-241	Quarterly/as needed	0.06 dpm/sample	0.03 dpm/sample
1/1976–4/1980	Controls for Environmental Pollution	Pu-238	As needed	(f)	(f)
1/1976–4/1980	Controls for Environmental Pollution	Pu-239	Quarterly/as needed	(f)	(f)
1/1976–4/1980	Controls for Environmental Pollution	Am-241	Quarterly/as needed	(f)	(f)
5/1980–9/1985	Controls for Environmental Pollution	Pu-238	As needed	(f)	(f)
5/1980–9/1985	Controls for Environmental Pollution	Pu-239	Quarterly/as needed	(f)	(f)
5/1980–9/1985	Controls for Environmental Pollution	Am-241	Quarterly/as needed	(f)	(f)
1999	Quanterra	Pu-238	Unknown	0.0025–0.044 pCi/L	-
1999	Quanterra	Pu-239/240	Unknown	0.0025–0.045 pCi/L	-
11/1974–8/1975	Eberline	Gross alpha (plutonium + americium)	Quarterly/as needed	<10.0 dpm/sample	-

a. Based on review of worker dosimetry reports in Boyd (2006a,b,c,d,e,f).
b. Records indicate quarterly monitoring for plutonium workers, unless an intake was suspected, initiating more frequent special sample analyses.
c. Assumes the MDC is twice the reported error
d. The MDC for Controls for Radiation plutonium measurements is twice the 95th percentile of the reported error values for zero result measurements (LaBone 2010).
e. Error values are the error reported (as plus-or-minus values) for zero measurement values.
f. Bioassay data analyzed by Controls for Environmental Pollution are not to be used in internal dose assessments.

NUMEC health physicist Roger Caldwell believed that fecal sampling was the only satisfactory method for estimating lung burdens for insoluble actinide alpha emitters classified as Y in the contemporary lung model. The most important alpha emitters included $^{239}PuO2$, $^{241}AmO2$, $^{234}UO2$, and $^{232}ThO_2$ (Caldwell 1966). Caldwell calculated that easily detectable plutonium quantities were excreted in feces: 49 dpm/d PuO_2 is eliminated from a 16-nCi lung burden, 32 dpm/d by way of the feces. One-tenth of this value or 3.2 dpm/d was believed to be a suitable reference level. Workers excreting safely below this level were assumed to have nonhazardous lung burdens (Caldwell 1966).

Table 5-5. Plutonium fecal bioassay MDC, frequency, and period.[a,b]

Date	Laboratory	Radionuclide	Frequency[b]	MDC[c]	Error
1/1966–1/1977	Eberline	Pu-239, Pu-238 or Am-241	Quarterly/as needed	0.1 dpm/sample	0.05 dpm/sample
5/1975–9/1975	Eberline	Gross alpha (plutonium + americium)	As needed	0.1 dpm/sample	0.05 dpm/sample
2/1977–10/1985	Controls for Environmental Pollution	Pu-239, Pu-238 or Am-241	Quarterly/as needed	(d)	(d)

a. Based on review of worker dosimetry reports in Boyd (2006a,b,c,d,e,f).
b. Records indicate quarterly monitoring for plutonium workers, unless an intake was suspected, initiating more frequent special sample analyses.
c. Assumes the MDC is twice the sensitivity or error.
d. Bioassay data analyzed by Controls for Environmental Pollution are not to be used in internal dose assessments.

Caldwell noted that fecal sampling should be performed after a person had been away from exposure (e.g., plutonium nitrate) for at least 2 days and that individuals would have to be removed from any possible UO_2 exposure for at least 7 days before fecal data could be used to estimate long-term lung burdens (Caldwell 1966; Caldwell, Potter, and Schnell 1967).

Caldwell analyzed the correlation between lapel breathing-zone air (BZA) sampling and early fecal clearance of plutonium and uranium. There was good agreement between the proposed International Commission on Radiological Protection (ICRP) lung model (Bates et al. 1966) and lapel sampler data (Caldwell, Potter, and Schnell 1967).

NUMEC health physicists used BZA and general air (GA) sample results to screen for possible exposures. If an exposure occurred (based on BZA sample or incident), bioassays of usually both fecal and urine samples were collected and correlated with BZA samples. The suspected exposed worker was removed from radiation work and fecal and urine samples were collected. This was the method employed by the mid-1960s because it was noticed by the NUMEC health physics group that fecal sampling was well correlated to the contemporary lung model and lapel or BZA results (see Figure 5-1) (Caldwell, Potter, and Schnell 1967).

The basic fecal sample procedure was that employees were given a quart plastic refrigerator carton, a small roll of tape, paper bag, and a written set of instructions. Employees took the bioassay kit home to prepare the sample. After depositing the sample in the carton, employees replaced the lid and sealed it with tape. The carton was placed in the paper bag and brought back to the laboratory to ship to the bioassay vendor. NUMEC added formaldehyde as requested by the vendor (Caldwell 1966). A number of different bioassay vendors performed plutonium urine and fecal analyses as observed from worker dosimetry records.

5.2.2 Uranium Urine and Feces Bioassay

Uranium was processed at both the Apollo and Parks Township sites. Enrichment levels varied with time and included DU, NU, LEU (3.5%), and HEU (93%).

5.2.2.1 Uranium Urine Analytical Procedure

Information in HASL-82 (AEC 1960a) indicates that before 1960 urine samples were obtained at the Apollo site on a monthly and bimonthly basis, with the commercial laboratory Nuclear Engineering and Sciences Corporation performing the urine analysis. The results frequently included high values that ranged from 50 to 150 µg/L of uranium in urine. However, the available dosimetry records do not contain urine bioassay results before late 1959.

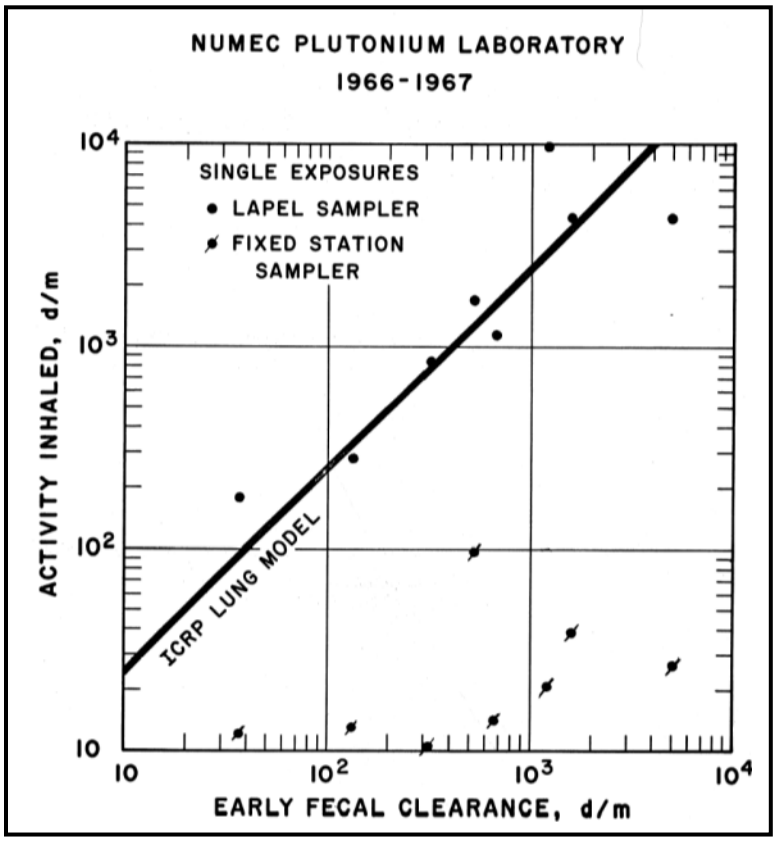

Figure 5-1. Correlation of fecal bioassay with air sampling (Caldwell, Potter, and Schnell 1967).

Available uranium bioassay data indicate uranium was analyzed in urine from about late 1959 through 1988 and in 1999 based on available bioassay reports. Information on the specific procedure used to analyze for uranium in urine is not known. Based on a bid specification (Author unknown undated c), the early analytical procedure probably consisted of taking 0.5 g of sodium carbonate ($NaHCO_3$) and adding 125 mL of urine and adjusting the pH with ammonia hydroxide (NH_4OH). After 2 hours the sample was centrifuged and the precipitated proteins, with the calcium and magnesium salts, were discarded. The supernatant was evaporated to dryness with HCl and HNO_3, then with hydrogen peroxide (H_2O_2), and finally with HNO_3, to ensure destruction of all organic matter (Author unknown undated c). The residue was taken up in 0.1N HNO_3 and added to a plating cell. A buffer solution containing ammonium oxalate, sodium phosphate, and ferrous ammonium sulfate was added and the pH adjusted to 5. The uranium was plated on a nickel disk anode in an electrodeposition unit of AEC laboratory design at a temperature of 95°F and 2 amps of current for 1 hour. The nickel disk was then dried and counted in a Nuclear Measurements Corporation gas flow proportional counter of the PC series (Author unknown undated c).

With a sample volume of 125 mL in a minimum counting time of 1 hour, the sensitivity was expected to be 12 ±3.2 dpm/L at a 90% confidence level. The recovery was expected to average 88% and an accuracy of 100 ±15%. The sample counted was to all intents and purposes weightless, so no absorption correction was necessary (Author unknown undated c).

The fluorimetric analysis would require a 5-mL sample volume and would have a sensitivity of 1 µg/L with a precision of ±10% (Author unknown undated c). However, the reported detection limit was 0.1 µg/L for results reported by Controls for Radiation, and 5 µg/L for results reported by Eberline in available bioassay reports.

No information on sample analysis methods for other periods is available.

5.2.2.2 Urine MDCs and Frequencies

The MDCs and approximate frequencies for the uranium urine bioassays are listed in Table 5-6 for NUMEC facilities. These values are based on review of bioassay monitoring results (Boyd 2006a,b,c,d,e,f). If an MDC value is needed before the dates listed in the table, the values for the earliest date should be used. The measurements based on activity (gross alpha, gross alpha uranium, and EU) should be evaluated as total uranium activity.

Table 5-6. Uranium urine bioassay MDC, frequency, and period.[a,b]

Date	Laboratory	Radionuclide	Frequency[b]	MDC[c]	Error
3/1961–2/1966	Controls for Radiation	Total uranium	Quarterly/as needed	<1 µg/L[d]	-
9/1972–12/1976	Eberline	Total uranium	Quarterly/as needed	<5 µg/sample	-
1/1977–11/1987	Controls for Environmental Pollution	Total uranium	Unknown	(e)	(e)
1999	Quanterra	Total uranium	Quarterly/as needed	<0.006 µg/L	-
4/1962–1/1967	Controls for Radiation	Gross alpha	Quarterly/as needed	26 dpm/L	13 dpm/L
2/1967–8/1972	Tracerlab	Gross alpha uranium	Quarterly/as needed	0.2 dpm/sample	0.1 dpm/sample
9/1972–1/1974	Eberline	Gross alpha uranium	Quarterly/as needed	<50.0 dpm/sample	0.05 dpm/mL
2/1974–4/1974	Eberline	Gross alpha uranium	Quarterly/as needed	<10 dpm/sample	-
2/1974–12/1976	Eberline	Gross alpha uranium	Quarterly/as needed	2 dpm/sample	1 dpm/sample
3/1964–6/1967	Controls for Radiation	EU	Quarterly/as needed	4 dpm/L	2 dpm/L
7/1967–8/1972	Tracerlab	EU	Quarterly/as needed	0.2 dpm/sample	0.1 dpm/sample
1/1977–2/1987	Controls for Environmental Pollution	EU	Quarterly/as needed	(e)	(e)

a. Based on review of worker dosimetry reports in Boyd (2006a,b,c,d,e,f).
b. Records indicate quarterly monitoring for uranium workers, unless an intake was suspected, initiating more frequent special sample analyses.
c. When an MDC is not available in the records, assumes the MDC is twice the error.
d. The MDC for Controls for Radiation for total uranium (1961 to 1966) is based on the reported value (Author unknown undated c) and should be used as a minimum value in place of the reported values in the individual bioassay records.
e. Bioassay data analyzed by Controls for Environmental Pollution are not to be used in internal dose assessments.

The MDC value for the Controls for Radiation is set to 1 µg/L even though the vendor reports often indicate a value of 0.1 µg/L. Therefore, the value in Table 5-6 has been set to 1 µg/L based on the reported value for the analytical method by Author unknown (undated c). This value should be used as a minimum value in place of the values in the bioassay records; larger reported values can be used in the intake assessment.

Care should be taken in evaluating the Controls for Radiation reported urine bioassay results because the records could have errors in the reported units. The reported values are in units of micrograms per milliliter although sometimes are listed as micrograms per liter.

Urine samples were typically 24-hour samples. The frequency for uranium urine samples was in general:

- Wet analytical chemistry personnel every 3 months,

- Nonradiation workers annually,
- Production workers every 4 to 6 weeks maximum (NUMEC 1963),
- Maintenance personnel every quarter (NUMEC 1963),
- All other (radiation) personnel every 6 months (NUMEC 1963), and
- At the discretion of health and safety in the event of an incident such as a uranium hexafluoride release (NUMEC 1963).

Although the above information indicates nonradiation workers were monitored annually, many worker files contain no record of bioassay monitoring. From a cursory review of the worker records, once per month seemed to be the highest frequency, although an average frequency was closer to once per quarter for uranium workers. Special bioassays were ordered for those workers who exceeded 40 MPC-hr of exposure or nose wipes exceeding 25 dpm.

There were as many as about 100 urine bioassay analyses each month. In the early years (to about 1964), urine samples were normally analyzed on a weight basis and then a radiometric analysis was performed if the level approached 50 µg/L. As stated above, the urinary control levels were 50 µg/L and/or 500 dpm/L for HEU (93%). According to the 1963 program review, the records for the few years before 1963 indicated that there had been no restrictions as a result of the personnel monitoring program (Hervin and Pryor 1963). However, during a hazard evaluation conducted by the AEC in 1959, a number of personnel had a urine concentration result between 50 and 150 µg/L (AEC 1960a). In later years urine was analyzed using one or both methods (weight basis and radiometric basis.)

The maximum allowable concentration in urine was 500 dpm/L for 93%-enriched ^{235}U (NUMEC 1963). At some time in 1963 this was decreased to 300 dpm/L, and by October 1964 this was decreased further to 150 dpm/L (Thornton and Johnson 1964). The NU urine control limit was 50 µg/L weight basis or 75 dpm/L activity basis (Hervin and Pryor 1963).

By the mid-1960s, both fecal and urine bioassay samples were being collected by NUMEC to determine the appropriate clearance model. The permissible NU urine level of 75 dpm/d was used (Caldwell, Potter, and Schnell 1967).

5.2.2.3 Uranium Fecal Minimum Detectable Concentrations and Frequencies

The analytical procedure for uranium fecal analysis has not been located. The MDCs and approximate frequencies for the uranium urine bioassays are listed in Table 5-7 for the Apollo and Parks Township sites. The fecal analysis results reported as dpm/sample should be considered equivalent to the daily excretion rate (dpm/d). When results are provided as dpm/g along with the sample weight, the daily excretion value is based on the total sample activity evaluated as the product of the sample weight and the reported activity concentration.

Fecal sampling (in addition to urine sampling) began on a large scale at the Apollo uranium plant in June 1966 (Caldwell 1966). The fecal analyses continued until about 1985 as indicated in worker dosimetry records. Caldwell observed that some UO_2 exposures were poorly detected in urine (Caldwell 1966). According to Caldwell, literature available at the time indicated that whole-body (WB) counting was effective for EU lung burdens greater than 7 nCi, but fecal sampling was necessary for smaller fractions of the permissible lung burden.

Table 5-7. Uranium fecal bioassay MDC, frequency, and period.[a]

Date	Laboratory	Radionuclide	Frequency[b]	MDC[c]
6/1967–6/1972	Tracerlab	Radiometric uranium	Quarterly/as needed	2 dpm/sample
7/1972–1/1976	Eberline	Total uranium	Quarterly/as needed	<5 µg/sample
2/1976–10/1985	Controls for Environmental Pollution	Total uranium	Quarterly/as needed	(d)

a. Based on review of worker dosimetry reports in Boyd (2006a,b,c,d,e,f).
b. Records indicate quarterly monitoring for uranium workers, unless an intake was suspected, initiating more frequent special sample analyses.
c. The MDC for radiometric uranium (Tracerlab) is based on a reported error value of about 1 dpm/sample, multiplied by 2.
d. Bioassay data analyzed by Controls for Environmental Pollution are not to be used in internal dose assessments.

Caldwell used a permissible fecal excretion rate of 50 dpm/d for uranium assuming the ICRP recommended 380-day half-time for chronic UO_2 exposures (Caldwell, Potter, and Schnell 1967).

By 1972 or later, Caldwell believed that fecal sampling for all radionuclides was a valuable tool for early assessment of inhalation exposures but that information on the urine-to-fecal-excretion ratio was necessary for the complete interpretation of urine data. Caldwell found that the most important use of fecal-sampling data was for estimating the magnitude of single inhalations of uranium from accidental exposures. For uranium plant operations, Caldwell believed that lung burdens should be based on urine sampling or *in vivo* counting (Caldwell ca. 1972).

5.2.3 Thorium Exposures

There is not sufficient air-sampling or urinalysis information available for the NUMEC sites to conduct a thorium intake analysis for workers in general. If the case files include thorium measurement results, an intake and dose assessment can be performed. Thorium was processed at the Apollo site for a few years starting in 1963 and at the Parks Township site in the early 1960s. Limited information on thorium bioassay analyses was found in worker dosimetry records. In 1971, the error was reported as 0.1 dpm/sample for ^{232}Th (Tracerlab analysis) for a 100-mL urine sample, which provides a minimum detectable activity (MDA) value of 0.2 dpm/sample. The fecal analysis error for the same workers was reported as 0.1 dpm/sample, which provides an MDA value of 0.2 dpm/sample.

The thorium oxides, carbides, and hydroxides are absorption type S; nitrates and other compounds are type M. The dose reconstructor can assume either type M or S (ICRP 1994) to maximize the internal dose. The internal dose is evaluated for intake as ^{232}Th. Because these bioassay analyses were specific for ^{232}Th, consideration should be given to the ^{228}Th that would be present from decay of ^{232}Th. Based on ORAUT (2012a), after preparation of the thorium dioxide from ore, the amount of ^{228}Th initially decreases and later builds in from continued decay of the ^{232}Th. The recommended assumption of an 80% ratio of ^{228}Th to ^{232}Th is appropriate for cases in which the time from initial purification is unknown. The internal dose is evaluated based on the estimated intake of ^{232}Th plus an equilibrium activity of 80% as ^{228}Th.

5.2.4 Mixed Fission Products

The records indicate urine bioassay analyses were performed occasionally for mixed FPs from 1962 through 1968. The MDC for these analyses was about 5 dpm/sample throughout the period, with sample analysis provided by Controls for Radiation. If bioassay records are found in case files with results provided by Controls for Environmental Pollution, the results should not be used to estimate intake of mixed fission products.

Exposure to FPs at the Apollo site was most likely to have occurred in the Laundry Building as part of the commercial decontamination of clothing by laundering. Exposure to FPs at the Parks Township site would most likely be related to source fabrication (^{60}Co and ^{137}Cs). The radionuclides representing mixed FPs could have included fission and activation products representative of reactor

operations. Possible radionuclides include ^{60}Co, ^{90}Sr, ^{99}Tc [a recycled uranium (RU) contaminant], ^{137}Cs, ^{106}Ru/Rh, and possibly others. No information is available on the methods used to analyze for mixed FPs in urine. Urine bioassay data for mixed FPs should be used, if included in the case files, to estimate intakes of FPs. ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gamma Analyses* (ORAUT 2007a) can be used to determine the radionuclide appropriate for the dose calculation. This technical information bulletin requires specification of the decay time for the fission product mixture. Because little is known about the FP material likely to be present at the Apollo site, a decay time of 1 year should be assumed to provide an assessment of dose that is favorable to claimants (ORAUT 2007a).

The urine bioassay results do not indicate if the measurements are based on beta or gamma analysis, so the intake should be based on both methods according to guidance in ORAUT-OTIB-0054 (ORAUT 2007a).

5.2.5 Unmonitored Radionuclides from Recycled Uranium

The uranium processed at the Apollo and Parks Township sites might have included RU. This material would contain contamination radionuclides formed during fission and activation processes when the material was irradiated in production or test reactors. The spent fuel elements were reprocessed to recover the uranium, which was returned to the DOE inventories along with trace contaminants that included ^{99}Tc, ^{237}Np, and ^{239}Pu. The intake of RU contaminant radionuclides can be estimated using the contaminant fraction values in Table 5-12 of the occupational internal dose technical basis document for the Fernald Environmental Management Project (ORAUT 2004). The estimated intake of each radionuclide is obtained by multiplying the estimated uranium intake by the appropriate contaminant fraction. The assignment of material solubility type is based on the guidance in Table 5-2 of ORAUT-OTIB-0060 (ORAUT 2007b). Because plutonium is a minor contaminant in the recycled uranium matrix, the consideration of type Super S plutonium is not necessary for evaluation of internal dose from this source of plutonium.

5.3 *IN VIVO* COUNTING

In vivo or lung counting for ^{239}Pu, ^{241}Am, uranium, and some FPs was started in about 1966 for incident evaluation (Caldwell 1966, 1968b). The counting in 1966 was provided by the University of Pittsburgh Low Level Radioactivity Monitoring Facility at the Presbyterian-University Hospital, using a thin NaI crystal system (Caldwell and Judd 1966). The bioassay records indicate this is the facility where the majority of routine WB counts were performed for NUMEC workers starting in 1969.

In 1968 and 1971, Helgeson performed WB counts on individuals for fission products, ^{235}U, ^{241}Am, with ^{239}Pu estimated from the ^{241}Am results based on expected activity ratios for ^{239}Pu/^{241}Am (Caldwell 1968b). The MDA for ^{235}U was listed as 0.08 mg for this system. The MDA for ^{241}Am ranged from 0.13 to 0.38 nCi for individual measurements at the 2-sigma level. The ^{239}Pu activity was estimated using an activity ratio ranging from 9 (ZPPR fuel) to 19.

The procedure for lung counting used by the University of Pittsburgh Low Level Radioactivity Monitoring Facility included a standard stretcher technique that was used with two 5- by 3-in. NaI(Tl) dual crystal low energy detectors positioned above the stretcher near the anterior chest region of the subject (Boyd 2006g, pp. 92–97). The calibration was for 0.5 keV per channel and the count time was 40 minutes for gross counts and background. Background correction was made using spectra obtained from unexposed individuals. Minor differences in the potassium and cesium body burdens were corrected by normalizing the spectra at an energy region from 90 to 125 keV. Activity calibrations were obtained from data published by Los Alamos National Laboratory using a detector configuration identical to the one used by the Laboratory. The calibration factor was adjusted for attenuation due to variation in the subject's chest wall thickness as measured with an encephaloscope. The evaluation of ^{239}Pu activity was based on the assumption that only ^{239}Pu was

present and all 17-keV X-rays were from ^{239}Pu. The difficulty in measuring the low-energy X-rays results in MDA values that represent significant lung burdens (ORAUT 2007a).

Lung counts were performed from about 1966 to 1992, and possibly later. Uranium lung counting started regularly in December 1971. Plutonium and americium counting started in 1966 (Caldwell 1966) and on a regular basis in 1968. FPs were counted intermittently. Lung counts are in general not as reliable as urinalysis (or fecal analysis – Caldwell 1966) for routine monitoring. However, this monitoring was routine and was used to assess routine exposures to transuranic (TRU) elements and FPs and to further analyze results from accidental acute and routine chronic intakes. Table 5-8 lists uranium lung-counting MDAs for common enrichments that might have been processed at NUMEC. Actual MDAs from worker records should be used if available because the MDA for a measurement is dependent on the chest wall thickness, which varies by individual. The MDA for ^{235}U was about 63 µg, as indicated from the cursory review of worker dosimetry records in 1971 and later years, which is a reasonable default MDA value.

Table 5-8. Lung-counting MDAs of uranium based on enrichment in picocuries.[a]

Uranium source term	Total uranium MDA (µg)	Total uranium MDA (pCi)
NU	8.75E+03	5.98E+03
93.00%	6.77E+01	4.61E+03
3.50%	1.80E+03	3.96E+03
2%	3.15E+03	5.09E+03
Typical DU	3.17E+04	1.27E+04
RU (1% ^{235}U)	6.30E+03	5.73E+03

a. Based on U-235 MDA of 63 µg.

Table 5-9 is a summary of *in vivo* MDAs for ^{239}Pu and ^{241}Am based on a review of claimant files. The results are generally reported as WB counts in the dosimetry records. Data after 1985 are sparse in the bioassay records.

Table 5-9. *In vivo* MDAs for ^{239}Pu, ^{241}Am.[a]

	Pu-239 MDA (nCi)			Am-241 MDA (nCi)		
Year	Minimum	Maximum	Counts	Minimum	Maximum	Counts
1968	NR[b]	NR[b]	NR[b]	0.13	0.38	17
1969	NR[b]	NR[b]	NR[b]	NR[b]	NR[b]	NR[b]
1970	NR[b]	NR[b]	NR[b]	NR[b]	NR[b]	NR[b]
1971	NR[b]	NR[b]	NR[b]	NR[b]	NR[b]	NR[b]
1972	9.0	11.5	3	0.13	0.13	1
1973	5.6	15.6	46	0.11	0.21	28
1974	5.44	21.3	122	0.09	0.22	96
1975	4.8	19.9	133	0.11	0.21	104
1976	5.0	20.3	109	0.11	0.19	91
1977	4.4	19.6	113	0.09	0.19	88
1978	4.7	19.0	132	0.10	0.19	100
1979	5.16	24.3	168	0.08	0.26	132
1980	5.03	28.2	132	0.09	0.21	94
1981	7.21	27.8	55	0.12	0.20	31
1982	7.12	34.3	77	0.12	0.21	44
1983	9.41	15.6	6	0.12	0.16	4
1984	8.67	22.32	9	0.12	0.15	5
1985	8.84	31.07	31	0.11	0.22	29

a. From a review of worker dosimetry records (Boyd 2006a,c,d,g,h,i). Values for 1968 through 1971 are based on the Helgeson system, with remaining values for the University of Pittsburgh system.
b. NR = none reported.

The *in vivo* bioassay records for individuals nearly always include an indication of the detection level for the measurements where the radionuclide was not detected. The detection levels are reported as "less than" values. Dose reconstructors should use the listed *in vivo* detection level information in evaluation of intakes for specific radionuclides.

5.4 APOLLO PROCESS URANIUM AIR SAMPLING STUDIES

This discussion focuses on documented air sampling data from five HASL reports for the Apollo site. The information in the reports can be useful in providing an estimate of the likely intake for a worker when job description and location are known. A general description of processes for various enrichments of uranium at Apollo is outlined in HASL Survey Reports 82, 92, 103, 106, and 114 (*Occupational Exposure to Radioactive Dusts* reports), which cover the period from December 1959 to January 1961 and the Procedure for Recovery of Scrap Uranium from 1962 (AEC 1960a,b,c, 1961a,b; NUMEC 1962). HASL survey reports contain results for loose and total alpha samples, GA samples, fixed-station and weighted BZA samples, and some personnel protective equipment, ventilation description, and general observations of activities. Attachment A lists the results of all BZA surveys for HASL Survey Reports 82, 92, 103, 106, and 114. A summary of the HASL reports is listed in Table 5-10 based on information from Attachment A.

Table 5-10. HASL report summary.

Report	Description of report	Date
HASL-82 (AEC 1960a)	Production of UO_2 from UF_6, UO_2 pellet formation, uranium-graphite pellet production, recovery of U_3O_8 HEU from uranium-zirconium scrap, R&D for coating uranium particles. NU was used in coating studies. Fully enriched (93%) HEU was used in the other processes.	12/1959
HASL-92 (AEC 1960b)	Production of UO_2 from UF_6, UO_2 pellet formation, uranium-graphite pellet production, recovery of U_3O_8 HEU from uranium-zirconium scrap, R&D for coating uranium particles. NU was used in coating studies. Fully enriched (93%) HEU was used in the other processes.	6/1960
HASL-103 (AEC 1960c)	Production of UO_2 from UF_6, UO_2 pellet formation, uranium-graphite pellet production, recovery of U_3O_8 HEU from uranium-zirconium scrap, R&D for coating uranium particles. NU was used in coating studies. Uranium (1.8% to 93%) was used in the other processes.	10/1960
HASL-106 (AEC 1961a)	Processing (93%) UO_3 from UF_6 and 93% U_3O_8 from uranium-zirconium, powder handling of HEU (93%) in the Ceramics Laboratory. Pressing EU (3.5%), centerless grinding and sintering EU and HEU (3.5% and 93%) were performed in the Ceramics Fabrication Area. Chemical reprocessing of EU (1.8%) and coating of uranium particles HEU (93%) were in operation.	12/1960
HASL-114 (AEC 1961b)	CP-2 chemical processing (3.4%) UO_3 from UF_6, CRP-2 chemical reprocessing (5.7%) U_3O_8 from uranium-zirconium, Ceramics Fabrication (5.7%) EU. Uranium particle coating involved NU. No powder handling activities were evaluated.	5/1961

The air samples consisted of radioactive particulates collected on filters from breathing zones and general areas during processing. The alpha activity measured on the filter was used to determine the airborne alpha activity concentrations. When multiple samples at a location were collected, the AEC used the mean air concentration in subsequent calculations. The AEC matched air concentration determinations with information about worker categories, locations, tasks, and time at each location or task.

When estimating the intake for a specific worker, the dose reconstructor should look for all available information related to intakes of uranium. The information can include:

- Bioassay monitoring results,

- Workplace breathing zone sampling results,
- General area monitoring results,
- Work location and job classification by period,
- Air concentration information from HASL reports (summarized in this section and Attachment A),
- Reports to the AEC/NRC of overexposures to airborne activity, and
- Reports to the worker of overexposures and work restrictions.

The bioassay monitoring results and workplace breathing zone results provide the best information because the data relate to the exposed individual. The reports to the AEC/NRC and work restriction letters also relate directly to the individual. The information related to work location and job classification is useful to establish potential for intakes, and for correlation to the HASL air sampling results. The HASL results have been analyzed to determine statistical information on air concentrations, as presented in Attachment A. The analysis of the reported doses for all individuals from all HASL reports was based on the assumption of the data being represented by a lognormal distribution. The highest value (6,300 dpm/m^3) is assumed to represent the 95% value and the lowest value (7 dpm/m^3) is assumed to represent the 5% value. The resulting distribution has a median value of 210 dpm/m^3 and a geometric standard deviation (GSD) of 7.91. This representation provides an estimate of air concentration that is more favorable to claimants than a strict numerical evaluation of the data. Several reported very high values are not well captured in a standard statistical analysis.

Case files might contain information on breathing zone air monitoring for individuals. This information can be used to establish potential for exposure and estimates of intakes for specific work tasks. In general, bioassay monitoring data should be used to estimate intakes if available.

The results in the HASL reports represent estimates of daily exposures to airborne uranium. These include specific work tasks that might have involved much higher air concentrations, but for a short period. Although it is not possible to identify in specific detail an individual worker's occupancy versus uranium airborne exposures to estimate uranium intakes during the entire operational period for the Apollo site (1957 to 1983), it is believed that the results of the HASL BZA results can give a reasonable upper bound intake value. The HASL studies resulted in improvements to working conditions, reducing the overall exposure of workers to airborne uranium.

The median value from the HASL reported average daily exposures (applied as a lognormal distribution with a GSD of 7.91) would provide a reasonable estimate of the distribution of uranium air concentrations during the 1960-through-1983 period for individuals whose specific work locations are not known. This intake should be limited to periods when the individual was likely to be involved with uranium work. It should not be applied to periods when the worker was on work restriction because of previous high exposures. This intake would represent an intake that is very favorable to claimants for individuals who did not work routinely in the Apollo uranium facility processes, such as health physics personnel, inspectors, and maintenance workers.

The inhalation intake of uranium can be estimated from the air concentration by multiplying by the breathing rate of 1.2 m^3/hour and the annual period of exposure (hours).

In the case where inhalation intakes are calculated from air concentrations, ingestion intakes are also to be considered. NIOSH (2004) states that the daily ingestion rate in picocuries can be estimated by multiplying the daily concentration in picocuries per cubic meter by a factor of 0.2 for an 8-hour workday.

5.5 REPORTS OF OVEREXPOSURES AND INCIDENTS

NUMEC Health Physics reported to the AEC any time a radiation worker exceeded 40 MPC-hr in a workday or a workweek. Individual dosimetry records should indicate if a worker exceeded the 40-MPC-hr limit. The records should also indicate if the overexposed individual was placed on work restrictions to limit internal and external radiation dose. This information would be useful in evaluation of bioassay data to indicate periods during which intakes might have occurred, and when intakes were unlikely. Individual dosimetry records should be used to reconstruct intakes on an individual basis whenever possible.

Overexposures were required to be reported to the AEC/NRC. Overexposures were measured in terms of MPC-hours. If calculated MPC-hours exceeded 40 for a week, it was considered an overexposure. MPC-hours were related to inhalation of uranium or plutonium suspended in the air.

To protect the workers, half-face and full-face respirators were available and used during certain operations. NUMEC used routine nasal smears and bioassay samples as proof of protection. A nasal smear exceeding 100 dpm acted as a flag to indicate possible inadequate protection or potential misuse of a respirator; it was assumed that no protection was afforded by the respirator and a bioassay was conducted. If a high nasal smear coincided with an impermissible air sample, it was assumed that an overexposure had occurred and NUMEC reported it in compliance with 10 CFR Part 20 requirements. If a high nasal smear could not be corroborated by a high air sample, NUMEC reported only if the bioassay data indicated an overexposure (Shapiro 1969).

5.5.1 Apollo Site Incidents

The following incidents occurred at the NUMEC site in Apollo, Pennsylvania, and might have contributed to employee exposures, but no exposure information was provided in the incident report or was reported as permissible. Information about individual involvement in incidents is likely to be in the workers' dosimetry records, and should be consulted for evaluation of intakes of radionuclides during such events.

In February 1963, a fire occurred when a polyethylene bottle containing recoverable powdered scrap uranium carbide stored under aqueous aluminum nitrate solution exploded from overpressure and the contents spontaneously ignited. Five bottles were damaged, containing about 8.8 kg of HEU. NUMEC estimated about 0.5 kg of HEU might have been lost. No information is provided about worker exposures due to this incident (George 1963).

During an investigation of a ventilation problem in the CRP-1 process area, it was discovered that the CP-1/CRP-1 ammonia fume scrubber exhaust duct had become plugged with 18 in. of material. The material was found to be about 400 kg of dry 10% uranium by weight and 3.3% enriched in ^{235}U; therefore, about 1.32 kg of ^{235}U were present in the duct. The material was removed from the duct. Routine inspections of the ductwork were put in place and a HEPA filter installed (Reitler 1973b).

On April 20, 1974, a maximum of 6 kg of low-enriched UF_6 was released to the in-plant atmosphere. A pipe and valve on the suction side of a hydrolysis column recirculating pump failed and blew out from the penton pipe, releasing the water from the bottom of the hydrolysis column, thereby releasing the UF_6. Nasal smears were taken from all personnel involved, and all were within permissible limits (Fink 1974).

5.5.2 Parks Township Site Incidents

The following incidents occurred at the NUMEC site in Parks Township and might have contributed to employee exposures. Information about individual involvement in incidents is likely to be in worker

dosimetry records and should be consulted for evaluation of intakes of radionuclides during such events. The following discussion is not intended to be a complete list of all incidents at the site.

In January 1966, a glovebox exploded when a worker struck a sparker to light a propane torch. The explosion blew out the glovebox and knocked the worker to the floor. He then ran from the room, with the box gloves still on his arms, spreading contamination. A nasal smear was taken, and he was decontaminated until the remaining activity was unremovable. Emergency personnel in protective clothing entered the room within minutes and extinguished the fire that had started in the damaged glovebox. All 150 personnel in the building at the time of the explosion were given nasal smears. Only the worker involved in the incident was found to have been seriously exposed (Caldwell, Potter, and Schnell ca. 1969).

In November 1966, an explosion in a glovebox resulted from a planned decomposition of hydrogen peroxide during a procedure test. The explosion shattered a glovebox window, sprayed the operator with nitric acid and plutonium, and caused physical injuries to a hand and an eye. After showering, the operator was transported to a local hospital for medical attention. The total uptake of plutonium was estimated to be less than 10 times the maximum permissible body burden (Crocker and Cleveland 1966; Caldwell, Potter, and Schnell ca. 1969).

In January 1967, a technician cut open an iridium source in a glovebox that did not have the proper negative pressure gradient. The cutting operation resulted in airborne release of more than 75 Ci of ^{192}Ir as fine particles; the two technicians in the area received significant inhalation exposure. The incident was discovered about an hour after the release when the technicians went through a radiation monitor on their way to lunch. After showering, they were checked with a beta and gamma counter, and high levels were discovered in the chest area that could not be removed by scrubbing. Lung counts were taken, but the workers had to stand outside the lung counter to avoid swamping the detectors. Estimated lung doses were 14 and 45 rem for the two technicians (Caldwell, Potter, and Schnell ca. 1969).

In December 1967, a technician amputated his right hand while operating a milling machine in a plutonium glovebox. The accident occurred when the technician's box glove was caught in the 4-in. cutter tool of a clausing milling machine. Contamination was limited to the severed wrist, and there was no significant release to the Plutonium Plant at large (Caldwell, Potter, and Schnell ca. 1969).

5.6 URANIUM AND PLUTONIUM AIR SAMPLING PROGRAMS

NUMEC uranium and plutonium workers wore lapel samplers starting in 1965. The primary purpose of air sampling was determination of personal exposure (Caldwell, Potter, and Schnell 1967). Sample duration using lapel samplers was one 8-hour shift at about 2 to 4 L/min. BZA sampling was performed during the HASL surveys from December 1959 to January 1961, and BZA sampling was observed in the 1963 health protection program review conducted by the AEC Oak Ridge Office R&D Division (Hervin and Prior 1963). NUMEC used a Rochester Imaging Detector Laboratory gas flow proportional counter.

Before 1965, the BZA samples were probably fixed-station BZA samplers; later NUMEC studies conducted in the 1966 to 1967 timeframe indicated that there was little difference between fixed-station BZA and GA samplers. The correspondence between lapel sampler data and early fecal clearance for plutonium showed very good agreement, but fixed-station BZA samplers and general area air sampling usually underestimated airborne concentrations. Fifty percent of the lapel air sample results at the Apollo site showed concentrations seven times greater than stationary air samples. The median of the ratio of lapel BZA to GA concentrations results was found to be ~7 at the Apollo and Parks Township sites (Caldwell, Potter, and Schnell 1967).

According to the 1963 NUMEC Health and Safety Manual, average or weighted airborne exposure studies were performed on every new operation and repeat studies were made on old operations on a frequent basis (NUMEC 1963). According to a health protection program review conducted in 1963, 75 short-term breathing-zone air samples were obtained in Apollo process buildings every week but no routine general area air samples were taken that would indicate an average air concentration over an 8- to 24-hour period (Hervin and Pryor 1963).

Not all employees were assigned lapel samplers. Lapel samplers were used as a “diagnostic tool” and provided to personnel whose work activities were likely to result in a local "micro-climate" of radio-aerosol. Localized airborne exposure conditions existed during such activities as moving a contaminated beaker from one hood to another or working in a glovebox that had a pinhole leak in a glove (Caldwell, Potter, and Schnell 1967). During the NUMEC respirator effectiveness study (1966 to 1967), whenever a BZA sample indicated an exposure, the worker was removed from radiation work and fecal and urine samples were collected (Caldwell and Schnell 1968).

The MPCs in the NUMEC Health and Safety Manual (NUMEC 1963) were 1×10^{-10} µCi/mL or 220 dpm/m^3 for in-plant airborne uranium.

6.0 OCCUPATIONAL EXTERNAL DOSE

This section describes NUMEC external dosimetry monitoring practices and provides supporting technical data to evaluate external occupational doses based on available dosimetry information. Although, DHHS has determined that there is insufficient information to either: (1) estimate the maximum radiation dose, for every type of cancer for which radiation dose are reconstructed, that could have been incurred under plausible circumstances by any member of the class; or (2) estimate the radiation doses to members of the class more precisely that a maximum dose estimate, at the Apollo and Parks Township sites during the operational period (Leavitt 2007, 2008), partial dose reconstructions can be completed using any external monitoring data in an individual’s file (and that can be interpreted using NIOSH dose reconstruction processes or procedures) for Parks Township workers.

The NUMEC Health and Safety organization provided general radiological safety, criticality safety, instrumentation, and personnel dosimetry support to the Apollo nuclear fuel and the Parks Township plutonium facilities and associated operations, as presented in correspondence about resolution of safety issues at the sites (Caldwell 1967a, 1968c).

6.1 EXTERNAL EXPOSURE SOURCES

The Apollo Nuclear Fuel Facility started operations in 1957 with the small-scale production of HEU and LEU fuel. Between 1958 and 1983, the Apollo site produced LEU dioxide fuel for use in commercial nuclear reactors. The process consisted of conversion of UF_6 to UO_2. In 1963, an additional production line was added to produce HEU fuel for U.S. naval propulsion reactors. From 1958 through the 1960s, NUMEC processed unirradiated EU scrap under license from the AEC (NIOSH 2007a). Smaller operations consisted of analytical laboratories, UO_2 pellet production, and R&D into coating techniques for uranium particles (B&WNES 1997). HEU operations at the Apollo site were discontinued in 1978, and LEU and all other processing operations that involved radioactive materials had ended by the end of 1983. In the mid-1960s, NUMEC was involved in production of thorium oxide (ThO_2) pellets for use in nuclear fuel.

Parks Township site operations began in about 1959; DOE operations ended in 1980. The initial function of the Parks Township facilities was fabrication of plutonium fuel, preparation of HEU fuel, and production of zirconium-hafnium bars under AEC/NRC License SNM-414, received March 1961, which allowed the handling of plutonium already on the site. The Parks Township site made fuel for the FFTF that consisted of a mixture of PuO_2 and depleted UO_2. It also made fuel plates for the

ZPPR in the late 1960s and ZPPR-III fuel wafers. Activities included plutonium scrap recovery, DU fabrication, HEU fuel manufacturing, source manufacturing (primarily ^{60}Co, ^{192}Ir, PuBe, and AmBe), irradiated fuel sample examination, laboratory operations, and supporting nuclear power site operations. In 1980, B&W began dismantling the fuel fabrication lines to enable the area to be used for commercial decontamination and possibly LLRW volume reduction operations until the early 1990s. In 1982, B&W used areas of the building for nuclear power site support operations.

6.2 WORKPLACE RADIATION FIELDS

Occupational exposures were primarily associated with NUMEC activities with plutonium, thorium, and HEU to produce reactor fuel. Fissile material arrived in approved shipping and storage cylinders and was present in various forms (liquid, powder, or metal) to be converted for use in nuclear fuel. Available information indicates PuBe neutron source production was performed at the Parks Township site. There was some fission and activation product exposure (Caldwell and Judd 1966). The primary sources of external radiation exposure from operations at NUMEC are summarized in Table 6-1.

Table 6-1. Workplace potential exposures.

Source	Exposure potential
Plutonium fuel fabrication: 1959–1980	Gamma, X-ray, and neutron radiation primarily
HEU production: 1957–1978	Beta radiation primarily, possibly photon dose from uranium progeny such as radium, etc.
Source manufacturing	Gamma and neutron radiation depending on source
LEU production: 1957–1984	As above
Mixed plutonium and EU fuel fabrication	As above
HEU and LEU scrap recovery	As above
UO_2 pellet production started in 1961	As above
R&D for coating uranium particles started in 1961	As above
Thorium operations and pellet production started in 1963	Beta radiation and more significant photon radiation
Laundry operations	Uranium and thorium residues

6.2.1 Beta Radiation

Beta radiation associated with plutonium and thorium fuel operations is expected to be comparatively minimal. The beta dose rate for uranium operations such as on the surface of yellowcake (an NU compound) just after separation is negligible, but rises steadily thereafter due to the buildup of the ^{238}U decay products ^{234}Pa and ^{234}Th. A few months after chemical separation, when equilibrium is reached, the beta dose rate from yellowcake is about 150 mrad/hr. There would typically be mixed beta and photon radiation associated with fission and activation products.

6.2.2 Photon Radiation

Photon radiation, typically of lower energy, is characteristic of plutonium operations. Thorium emits significant higher energy photon radiation. Uranium has comparatively less significant photon radiation with dose rates of about 1.2 mrad/hr in contact with fresh yellowcake. However, during the buildup of the ^{234}Th and ^{234}Pa progeny in fresh yellowcake, the radiation levels increased somewhat for several months after yellowcake production. Photon exposure rates were estimated to be about 1 mrad/hr at 30 cm from a drum of aged yellowcake (NIOSH 2010b, Table 7).

6.2.3 Neutron Exposures

Neutron exposures might have occurred from both spontaneous fission in isotopes of uranium or plutonium and from alpha-neutron reactions with low atomic number materials such as oxides and impurities. Neutron exposures from plutonium occur and levels are generally described in the *Guide*

of Good Practices for Occupational Radiological Protection in Plutonium Facilities (DOE 1998). Neutron exposures from thorium and uranium such as yellowcake are considerably lower than the photon exposures and are, therefore, not generally considered significant based on analyses in ORAUT-OTIB-0024, *Estimation of Neutron Dose Rates from Alpha-Neutron Reactions in Uranium and Thorium Compounds* (ORAUT 2005a). That document describes the expected neutron dose rates from the various forms of uranium compounds. For a large cylinder of uranium hexafluoride, the dose rate at 3 ft is about 0.003 mrem/hr for NU, 0.016 mrem/hr for 5% EU, and 0.45 mrem/hr for +97% EU.

6.3 DOSIMETER TECHNOLOGY

NUMEC historically used beta/photon and neutron dosimeters to measure potential WB beta/photon, WB neutron, and extremity beta/photon exposures to personnel. A summary of the NUMEC dosimetry systems and periods of use is presented in Table 6-2.

6.3.1 Beta/Gamma Dosimeters

Nuclear Science & Engineering, Controls for Radiation, Eberline, or Landauer provided film dosimeter services to NUMEC from 1959 until about 1968, when thermoluminescent dosimeters (TLDs) were implemented. There is evidence of NUMEC concern about film dosimetry over-response to the low-energy photons from plutonium (Caldwell and Judd 1966). Landauer began providing dosimeter service to NUMEC in 1964. Eberline provided dosimetry service beginning in 1966, and NUMEC apparently ran an in-house TLD program beginning in about 1968. The dosimetry service was again provided by Landauer beginning in 1976. External results were found for July to December 1991 with dosimetry provided by Teledyne Isotopes (BWXT 2006).

6.3.2 Neutron Dosimeters

Workers were monitored for neutron exposures with nuclear track emulsion, Type A (NTA) film from commercial venders until about 1968, and with TLDs thereafter. In addition, criticality dosimetry monitoring was done with an array of area critical assemblies that fed into a central system. This system existed from at least 1963; in September 1963, each visitor and employee was issued an indium foil criticality dosimeter as part of each security badge (NUMEC 1963).

6.3.3 Limits of Detection

External dosimetry technology minimum detectable levels (MDLs) are expected to be similar to contemporary commercial vendor capabilities. Examination of dose reports for individual dosimeter exchange periods and workers shows recorded doses as low as 2 mrem for photons (Boyd 2006a, p. 6), which is certainly less than a statistically based MDL. However, other documentation indicates that film dosimeter MDLs in the workplace are higher. The film badge dosimetry at NUMEC was likely similar to dosimeters used at Hanford during the period 1957 to 1968 (ORAUT 2010c). Therefore, the recommended MDLs for estimation of missed dose are 30 mrem for gamma and beta radiation through 1968, and 50 mrem for neutron radiation for periods to 1975 (ORAUT 2010c,d). For neutron radiation after 1975, the MDL is reduced to 20 mrem based on studies at the Hanford Site (Fix et al. 1981) and Savannah River Site (SRS) (Taylor et al. 1995) that indicated the MDL was closer to 10 mrem for neutron exposures to fast neutrons from ^{252}Cf. The 20-mrem value is consistent with the SRS site profile (ORAUT 2005b) value for this period because work with plutonium was similar to that at the NUMEC Plutonium Facility. For estimating the potential annual missed dose in accordance with OCAS-IG-001 (NIOSH 2007b) for monitored workers, Table 6-3 summarizes the annual potential missed dose to be assigned in relation to the dosimetry service providers, periods of use, dosimeter exchange frequencies, and estimated MDLs.

Table 6-2. Dosimetry for external whole-body, wrist, and extremity exposures.

Period	Monitoring technique	Dosimeter description
Beta/photon dosimeters – whole body		
1957–5/1968	Photographic film badge	Film badges contained single film packet. Three filters (front and back) were incorporated into film badge for energy dependence: cadmium, aluminum, and lead. Beta and photon radiation capabilities are similar to other dosimetry systems at that time as presented in the 1954 AEC dosimeter performance study (AEC 1955).
6/1968–1975	NUMEC or Eberline TLD	Comprised of 2 TLD-700 chips, 2 TLD-600 chips, and 1 CaF_2 for monitoring beta, X-ray, and gamma exposure.
1976–present	Landauer or equivalent TLD (Z1 dosimeter -1990)	Comprised of 3 TLD-700 chips for monitoring beta, X-ray, and gamma exposure. Insensitive to neutron radiation.
Beta/photon dosimeters – wrist and ring		
July 1963–about 5/1968	Landauer (Type M – wrist beta-gamma) film badges or equivalent.	Film dosimeter known as Type M responsive to beta and gamma radiation.
About 6/1968– 1983	TLD wrist badge	Comprised of 3 TLD-100 chips.
7/1991 – 12/1991	Teledyne Isotopes TLD Badge	TLD badge for monitoring beta and gamma exposure (BWXT 2006). Details of the dosimeter are not available, other than detection limits.
Neutron dosimeters – whole body		
1957–5/1968	NTA film badge	Film badges using NTA films: Fast neutrons undergoing elastic collision with content of emulsion or cellulose acetate base material produce recoil protons, which are recorded as photographic tracks in emulsion. Track density is a linear function of dose. Developed image exhibits tracks caused by neutrons, which can be viewed using appropriate imaging method (i.e., oil immersion) and 1000-power microscope or projection capability.
6/1968–1995	Landauer Neutrak Extended Range dosimeter (types I8, I1, or RI)	Combined TLD albedo neutron monitor with track recoil device [CR-39 (allyl diglycol carbonate)] that responds to neutron radiation through proton recoil events. The dosimeter is responsive to a neutron energy range of about 0.0001 to 10 MeV. Dosimeter response to thermal neutron radiation was subtracted to yield fast neutron dose. The Neutrak ER has an albedo element with above-described elements. Qualitative relationship was derived to determine ratios of neutrons of various energies. The RI badge was capable of monitoring beta, X-ray, gamma, and neutrons.
7/1991 – 12/1991	Teledyne Isotopes TLD badge	Combined gamma, beta, and neutron TLD (BWXT 2006). Details of the dosimeter are not available, other than detection limits.

Source: ORAUT (2011b).

Performance of dosimetry technology at many commercial and AEC laboratory service providers was tested in 1954 by the AEC (AEC 1955). Characteristics of dosimetry systems at the NUMEC sites are listed in Table 6-2 for beta, gamma, and neutron radiation monitoring.

Table 6-3. MDLs and potential missed photon, beta, or neutron dose.

Vendor or processor/area monitored	Period of use	MDL[a] (rem)	Annual missed dose[b] (rem) (frequency in parenthesis)
Nuclear Science & Engineering or Controls for Radiation film and NTA film *whole body*	1957–1963	0.03[c] photons 0.03[c] beta	0.18 beta photons (monthly)
		0.05 neutrons	0.30 neutrons [fast] (monthly)
Landauer film and NTA film *whole body*	1964–1965	0.03[c] photons 0.03[c] beta	0.18 beta photons (monthly)
		0.05 neutrons	0.30 neutrons [fast] (monthly)
Eberline film and NTA film *whole body*	1966–6/1968	0.03[c] photons 0.03[c] beta	0.18 beta photons (monthly)
		0.05 neutrons	0.30 neutrons [fast] (monthly)
NUMEC or Eberline (Film-Apollo) until about 1970 and TLD for neutron *whole body*	7/1968–1975	0.02 photons[d] 0.02 beta[d]	0.12 beta photons (monthly)
		0.05 neutrons	0.30 neutrons (monthly)
Landauer TLD and Teledyne Isotopes *whole body*	1976–1995	0.01 photons[e] 0.01 beta[e]	0.06 beta photons (monthly)
		0.02 neutrons	0.12 neutrons (monthly)

a. Estimated MDLs for each dosimetry technology. Dose levels were recorded at values less than the MDLs.
b. Annual missed dose calculated based on the MDL/2 method from NIOSH (2007b).
c. MDLs for photons and beta during these periods are based on Hanford dosimeter values (ORAUT 2010c).
d. MDL during this period is probably twice the recording level of 0.010 rem.
e. Landauer MDL values from Boyd (2006j) and Teledyne Isotopes MDL values from BWXT (2006).

6.3.4 Radiological Records

A single dosimetry program was conducted at NUMEC. Records of radiation doses to individual workers from personnel dosimeters worn by the worker and coworkers are available for NUMEC operations beginning in 1957 for Apollo and 1959 for Parks Township as observed from a review of claimant records. Doses that were received by these dosimeters were recorded at the time of measurement and routinely reviewed by the NUMEC operations and radiation safety staff for compliance with radiation control limits. OCAS-IG-001 (NIOSH 2007b) indicates that these represent the highest quality records for retrospective dose assessments. Not all workers were assigned radiation dosimeters. Workers who received less than 25% of the quarterly dose limits in 10 CFR Part 20 were not required to be monitored (Boyd 2006a, p. 6). However, even though claimant records show that not all personnel were assigned dosimeters at all times, the records show that work areas were monitored.

Substantial worker-specific dose data have been received from NUMEC. Shallow, deep, neutron, and extremity doses are typically available. A computerized records system was implemented in October 1975 (Boyd 2006a, p. 7), and records for previous years are in hard-copy form. In addition, NUMEC was required to submit routine dose reports of personnel exposure information to the AEC and NRC for terminating employees (Boyd 2006f) as well as annual statistical data, such as those listed in Table 6-4 for 1976 and 1977 (Breuer 1977, 1978).

Table 6-4. Annual occupational radiation exposures at the Apollo site (Breuer 1977, 1978).

Year	Total number monitored	No. with measured dose	Number of individuals with WB doses in the ranges (rem)						
			<0.1	0.1–0.25	0.25–0.5	0.5–0.75	0.74–1.0	1.0–12	>12.0
1976	42	42	27	14	1	0	0	0	0
1977	39	39	15	16	6	0	2	0	0

6.4 LIMITATIONS IN MEASURED DOSE

Potential limitations in measured dose with NUMEC dosimetry capabilities include low-energy photons and neutron radiation.

6.4.1 Low-Energy Photons

Experience at NUMEC since the mid-1960s indicates there are potential limitations of the film dosimeter to measure accurately low-energy photon radiation such as that in NUMEC plutonium facilities. Caldwell and Judd (1966) indicated that photon radiation from plutonium could be considered to be in three effective energy groups:

- 17-keV X-rays that had a low penetrating ability,
- Effective energy of 60 keV from plutonium and its progeny including ^{241}Am, and
- Effective energy of 400 keV.

A spectroscopy survey at the Parks Township site evaluated photon fields from plutonium work. Surveys were conducted of the plutonium chemical processing line and ceramics line. The 60-keV peak from ^{241}Am was found to predominate. The 17-keV X-rays did not produce a peak and must have been substantially absorbed by the glovebox walls (Caldwell and Judd 1966, p. 4). Higher energy peaks at 208, 267, and 333 keV were produced by ^{237}U. The photon energy spectrum is shown in Figure 6-1.

A survey of the ceramics line and plutonium-uranium molybdenum alloy melt box line indicates a predominance of 60-keV ^{241}Am gamma radiation. The gamma energy spectrum is shown in Figure 6-2. However, the relative amount of 60-keV radiation is a factor of 6 higher for the ceramics glovebox in comparison with the melt box. This is attributed to the plutonium in the chemical processing line having aged an additional 2 years, whereas the plutonium in the melt box had just been received (Caldwell and Judd 1966).

An analysis of the Eberline film dosimeter response for open window (OW) versus aluminum (AL), plastic (PL), and cadmium (CD) filters was made as shown in Figure 6-3.

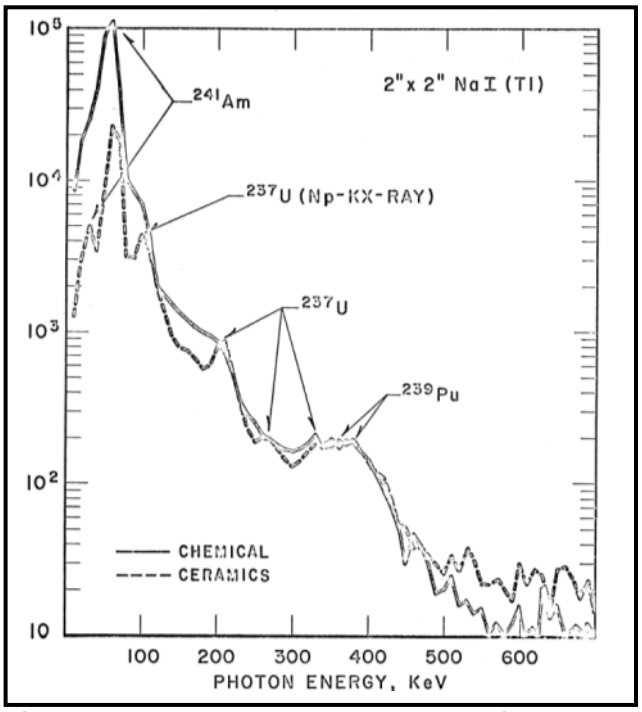

Figure 6-1. Gamma spectrum external to plutonium chemical processing line and plutonium ceramics line (Caldwell and Judd 1966, Figure 3).

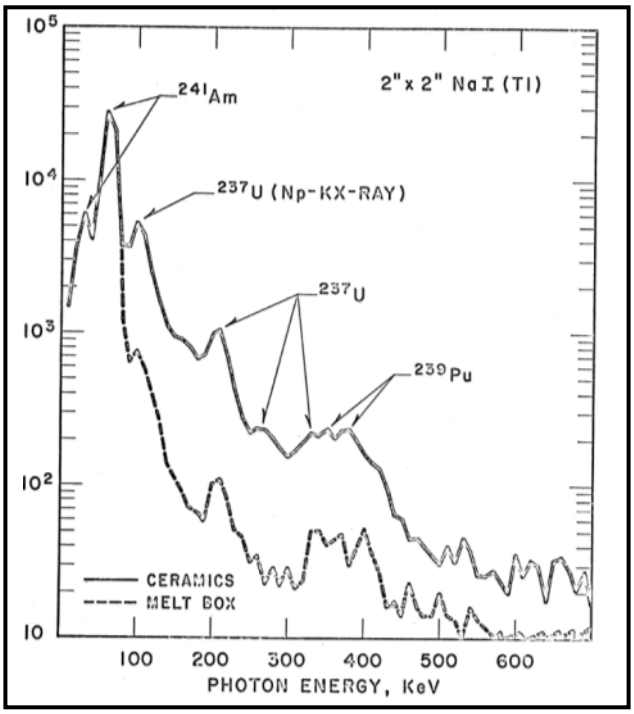

Figure 6-2. Gamma spectrum external to plutonium ceramics line and plutonium melt glovebox (Caldwell and Judd 1966, Figure 4).

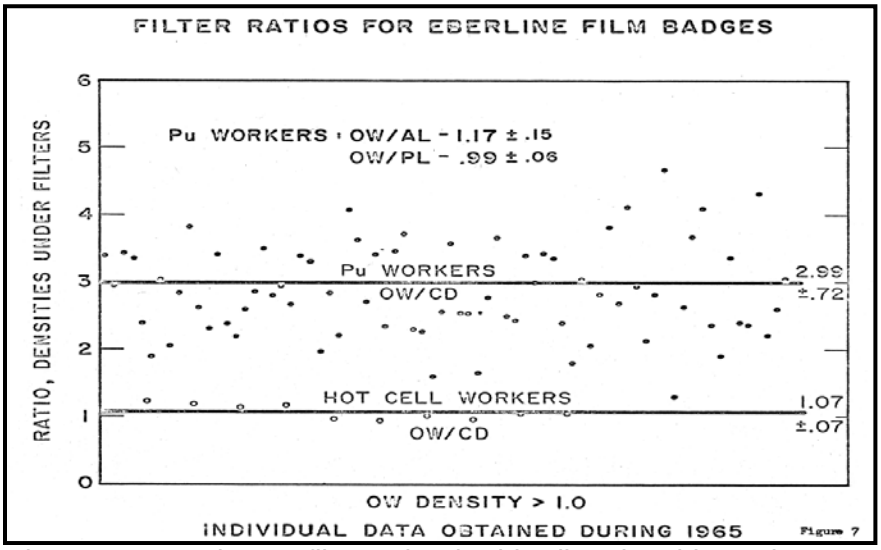

Figure 6-3. Dosimeter filter ratios (Caldwell and Judd 1966).

The following interpretation was made by Caldwell and Judd (1966):

1. Hot cell workers are exposed to ^{60}Co and FPs. The energy response of the film badge is constant above 200 keV. The dose was taken directly from a ^{60}Co calibration curve. If the OW:CD ratio was close to 1.0, the reported dose was accepted.

2. Plutonium workers were exposed to a wide range of gamma energies. The upper end of the spectrum would produce OW:CD ratios close to 1.0. The lower gamma energies would produce an OW:CD ratio of 3.0 or greater. One plutonium worker might be exposed to an entirely different effective energy than another due to shielding, working distance, and other

geometry factors. If the OW:CD ratio was less than 2.0, the reported dose was accepted. If the OW:CD ratio was greater than 2, NUMEC would use a plutonium spectrum calibration curve that represented a typical plutonium gamma spectrum.

3. The OW:PL (plastic) ratio was about 1 and was within a 6% standard deviation. This meant that the large OW:CD ratio is not due to beta radiation.

4. The OW:AL ratio was sensitive to X-ray exposures, but NUMEC did not incorporate this in its analysis.

Caldwell and Judd (1966) presented an assessment of the energy dose fraction for personnel exposure due to typical plutonium fuel fabrication from 1,000 MWd/t of plutonium. Sixty-five percent of the dose was from the ^{241}Am 60-keV gamma. Less than 7% was from the highest energy groups (Caldwell and Judd 1966). The summary of the energy dose fraction is shown in Figure 6-4. The gonadal dose was 50% of the WB or trunk dose due to the effect of the steel bottom of the plutonium gloveboxes (Caldwell and Judd 1966). Table 6-5 summarizes the gamma energy distribution for NUMEC plutonium in comparison with Hanford plutonium. Beta energies are included as well as ^{233}U and ^{241}Am, which have similar overall photon and beta properties.

6.4.2 Neutron Radiation

NTA film has a characteristic decreasing response to neutron radiation at energies below about 500 to 800 keV, depending on the extent of photon fogging and the overall process to develop and read the tracks (ORAUT 2006). However, at this time, the neutron dosimeter readings should be used without correction for this effect.

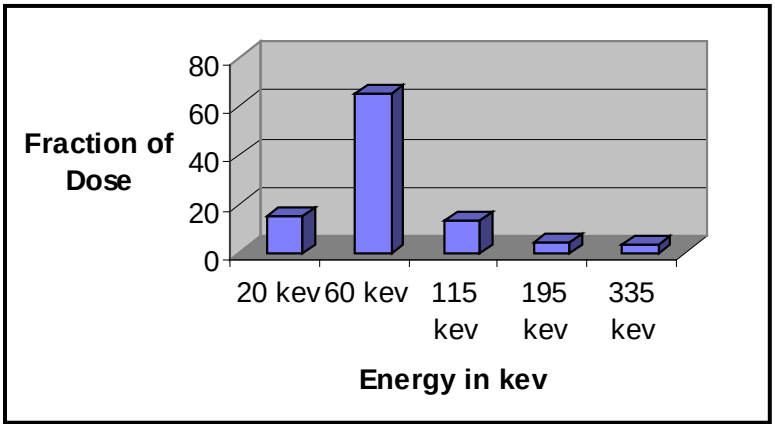

Figure 6-4. Relative contribution by energy group to personnel exposure during plutonium fuel fabrication from 1,000 MWd/t plutonium (Caldwell and Judd 1966, p. 18, Figure 8).

Table 6-5. Plutonium photon (and beta) energy factors.

Energy–photon	NUMEC plutonium, (^{241}Am & ^{233}U)	Hanford plutonium
<30 keV	15	25
30 – 250 keV	82	75
>250 keV	3	0
Energy–beta	**NUMEC plutonium, (^{241}Am & ^{233}U)**	**Hanford plutonium**
>15 keV	100	100

6.5 DOSE RECONSTRUCTION RECOMMENDATIONS

6.5.1 Recorded Dose Practices

Recorded and reported dose practices are summarized in Tables 6-6 and 6-7.

Table 6-6. Recorded dose practices.

Period	Dosimeter measured quantities	Compliance dose quantities
Photon/electron film dosimeter + NTA neutron dosimeter		
1957–1971	Gamma (G) Neutron (N) Beta (B)	WB or total = gamma (photon) + neutron Beta separate Extremity = gamma (+ neutron)
Photon/electron film dosimeter + TLD neutron dosimeter		
1972–1983	Deep = gamma and neutron (DBG) Shallow beta gamma (SBG)	WB = gamma + neutron Skin = beta Extremity = gamma + neutron
Photon/electron/neutron–Panasonic TLD + CR-39 neutron dosimeter		
1983–present	Deep Shallow	Skin = beta + soft gamma and neutron WB = photon + neutron Extremity = gamma + neutron

6.5.2 Adjustments to Recorded Dose

6.5.2.1 Beta Dose Adjustments

Beta and nonpenetrating dose was usually reported before 1975. In general, nonpenetrating radiation doses should be assigned as <30-keV photons if the employee worked with or around plutonium; otherwise, >15-keV electrons (beta) should be assigned (ORAUT 2005c).

Table 6-7. Interpretation of reported data.

Period	Reported quantity	Description	Interpretation of zeroes	Interpretation of blanks (no data)	Rollup of individual and annual data	Monitored/ unmonitored
1957–1971	R or rem	Reported WB doses include gamma and neutron doses	Zeroes were generally not reported. Reported zeroes should be interpreted as meaning less than MDL.	The absence of data should be interpreted as individual was monitored with zero result.	Photon WB dose, neutron WB dose, shallow skin dose, total deep WB dose	All employees with significant exposure potential were monitored
1972–1982	rem	Reported WB doses qualified as either photon or neutron	Zeroes were generally not reported. Reported zeroes should be interpreted as meaning less than MDL.	The absence of data should be interpreted as individual was monitored with zero result.	Photon WB dose, neutron WB dose, shallow skin dose, total deep WB dose	All employees with significant exposure potential were monitored
1983–present	rem	Photon deep, neutron deep, and skin dose reported.	Zeroes were typically reported. Reported zeroes should be interpreted as meaning less than MDL.	No data or blanks should be interpreted as individual was monitored with zero result.	Photon WB dose, neutron WB dose, shallow skin dose, total deep WB dose	All employees with significant exposure potential were monitored

The guidance from ORAUT (2005c) is as follows:

If the nature of the nonpenetrating dose is unknown, consider the following guidance:

1. For a likely noncompensable case, it is acceptable to assume the nonpenetrating dose is associated with <30-keV photons, because this maximizes the probability of causation (POC).

2. For a likely compensable case, it is acceptable to assume the nonpenetrating dose is associated with >15-keV electrons, because this minimizes the POC.

3. If the compensability decision might hinge on this issue, and if the partitioning of the nonpenetrating dose cannot be decided based on the available information, additional research might be required.

6.5.2.2 Photon Dose Adjustments

No adjustment is recommended for NUMEC recorded shallow and deep doses and photon radiation. The existing recorded doses provide a realistic estimate of the actual doses. Exposure dose conversion factors are applied to reported and missed photon doses to estimate dose to specific internal organs.

6.5.2.3 Neutron Weighting Factor Adjustments

Recorded NUMEC neutron doses are assumed to have been based on quality factors in National Council on Radiation Protection and Measurements Report 38 (NCRP 1971). The quality factors in Report 38 were compared with the neutron weighting factors in ICRP Publication 60 (ICRP 1991) to arrive at factors to convert the recorded dose to equivalent ICRP Publication 60 neutron doses as required by OCAS-IG-001 (NIOSH 2007b). A dose multiplier of 1.91 should be used for the 0.1- to 2-MeV energy range (ORAUT 2006). This range includes 100% for HEU, EU, NU, and plutonium work locations. The neutron doses from fuel work recorded for personnel at NUMEC using NTA should, therefore, be multiplied by a factor of 1.91 (for ICRP Publication 60 correction).

6.5.3 Missed and Unmonitored Dose

The potential for missed dose exists when workers are exposed to radiation at levels below the detection limit of their personnel dosimeters.

6.5.3.1 Shallow Dose and Deep Dose

The assignment of missed dose based on dosimetry records is performed using guidance in OCAS, IG-001, *External Dose Reconstruction Implementation Guideline* (NIOSH 2007b). Using this guidance, a dose equal to the limit of detection (LOD) divided by 2 is assigned for each dosimetry measurement that is recorded as less than the LOD/2 including zero values. The LOD values for NUMEC dosimeters are listed in Table 6-3.

For cases involving the skin as the target organ, guidance in ORAUT-OTIB-0017, *Interpretation of Dosimetry Data for Assignment of Shallow Dose* (ORAUT 2005c) should be followed for assignment of missed shallow and deep doses.

6.5.3.2 Neutron Dose

The potential missed neutron dose can be estimated from LOD values in Table 6-2 for monitored workers using the same approach.

If monitoring records do not include neutron dosimetry information, then neutron dose should not be applied. For workers who were likely to be exposed to neutrons, and for whom no neutron dosimetry

is available, a partial dose reconstruction would result. Exposure to uranium hexafluoride cylinders is a possible source of neutron exposure at the Apollo site.

6.5.4 Uncertainty

Dose reconstructors can incorporate consideration of uncertainty in the dose calculation for measured and missed doses as follows:

- The technology used to measure worker dose at NUMEC is similar to the technology that was used by commercial and AEC/NRC laboratory facilities. The errors in the penetrating dose are anticipated to be about ±30% and normally distributed. For noncompensable cases, the dose reconstructor can assume that errors are all positive (i.e., use only +30%) and multiply the measured dose by a factor of 1.3 (i.e., increase of 30%) to be used for Interactive RadioEpidemiological Program (IREP) Parameter 1 and to set Parameter 2 to zero (NIOSH 2007b). A constant distribution is applied.

- For missed dose, a lognormal distribution is assumed. Dose reconstructors should calculate the unmonitored dose or missed dose to arrive at Parameter 1 input and to set Parameter 2 equal to 1.52 (NIOSH 2007b). A lognormal distribution is applied.

6.5.5 Radiation Dose Fraction

Uranium represents the primary exposure hazard to NUMEC workers. Naturally occurring uranium is primarily a beta radiation hazard with an accepted surface dose rate of about 233 mrad/hr. The IREP input category for beta radiation is >15 keV. There is a small photon dose component of <10 mrem/hr (DOE 2001). As naturally occurring uranium is enriched, the photon dose is lowered but the spectra become correspondingly more energetic. The average energy of the spectra can increase from solid or liquid uranium sources because these can provide substantial shielding resulting in proportionally greater attenuation of lower energy photons. Exposure to thin layers of uranium on a surface will have a higher proportion of lower energy photons. The recommendation is to assign the photon dose as 100% to the 30-to-250-keV category to result in a higher calculated organ dose under most situations.

Apollo site workers had limited potential for exposure to radioactive sources in addition to uranium. These include thorium, plutonium, and photon sources such as radium, ^{192}Ir, ^{137}Cs, and ^{60}Co. Mixed FP exposure could have occurred at the Apollo Laundry Building that provided commercial laundering of contaminated clothing. Generally recommended categories for IREP input for the measured and assigned components of radiation dose are listed in Table 6-8 unless there is claim-specific information on the source of radiation exposure.

Table 6-8. Beta, photon, and neutron radiation energies and percentages for IREP input.

Description	Dates		Radiation type	Energy selection	Percent
	Begin	End			
Uranium facilities	1/1/1957	12/31/1983	Beta	>15 keV	100
	1/1/1957	12/31/1983	Photon	30–250 keV	100
Plutonium facilities	1/1/1959	12/31/1980	Photon	30–250 keV	100
	1/1/1959	12/31/1980	Neutron	0.1–2 MeV	100
Thorium handling	1/1/1957	12/31/1983	Beta	>15 keV	100
	1/1/1957	12/31/1983	Photon	30–250 keV	25
	1/1/1957	12/31/1983		>250 keV	75
Photon and neutron sources	1/1/1957	12/31/1983	Beta	>15 keV	100
	1/1/1957	12/31/1983	Photon	30–250 keV	50
	1/1/1957	12/31/1983		>250 keV	50
	1/1/1957	12/31/1983	Neutron	0.1–2 MeV	100

7.0 ESTIMATION OF EXPOSURE TO RESIDUAL ACTIVITY

The Apollo site stopped manufacturing nuclear fuel in 1983. Final decommissioning of the facilities was completed in 1995. For the period of residual contamination, employees of subsequent owners and operators of this facility are covered under EEOICPA. The residual period for the Apollo site covers the period from 1984 through 1995, and the residual period for the Parks Township site covers the period from 1981 through 2004.

The uranium work at Parks Township was with HEU in the Type II facility (Building C). The equipment in this building was removed in 1978, and by May 1979 the remaining surface contamination was fixed and inaccessible to diversion. The effluent reports after decontamination indicate contaminated liquid effluents from residual material in drains but no airborne emissions. This would suggest there is a potential for external exposure from residual DOE material, but inhalation exposure would be minimal.

The plutonium facility at Parks Township (Building A) was decontaminated and the equipment removed in 1980. However, the building continued to be used for non-DOE activities. Residual activity (from DOE operations) could have remained and caused exposure to workers. There was probably not much use of the buildings after the late 1980s, but NUMEC was licensed to have nuclear material on the site until final decommissioning was approved in about 1998. The license probably was kept in place to cover residual activity.

The following sections provide guidance for assignment of dose for the residual period.

7.1 EXTERNAL DOSE FROM RESIDUAL ACTIVITY IN THE WORKPLACE

7.1.1 Apollo Site

Because all operations at the Apollo site had stopped by 1983, the only exposures would be from residual activity. The potential external dose during the residual period has been estimated based on the mean surface concentration at the end of the operational period as described in Section 7.4, derived in support of internal dose estimates during the residual period. The analysis resulted in a mean surface concentration of 4.97×10^6 dpm/m^2 (GSD = 7.91). This value can be used to estimate the annual external dose to workers exposed to the residual activity. The annual dose is evaluated as follows, using a dose conversion factor (DCF) for exposure to uniform activity on a ground plane.

$$\textit{Dose}\ (\text{rem/yr}) = \textit{residual level}\ (\text{dpm/m}^2) \times \textit{DCF}\ (\text{rem/dpm/m}^2\text{/hr}) \times \textit{exposure time}\ (\text{hr/yr}) \quad (7\text{-}1)$$

The external dose conversion factor for exposure to isotopes of uranium and short-lived progeny is provided in Federal Guidance Report 12 (Eckerman and Ryman 1993). The median annual external dose from exposure to residual surface contamination is listed in Table 7-1 for organs considered in the Federal Guidance Report. These values are based on an exposure time of 2,000 hours per year.

Table 7-1. External annual dose from residual surface contamination at the Apollo site.

Organ	Annual dose[a] rem
Adrenals	0.001
Bladder wall	0.001
Bone surface	0.002
Brain	0.001
Breast	0.001
Esophagus	0.001
Stomach wall	0.001
Small intestine wall	0.001
Upper large intestine wall	0.001
Lower large intestine wall	0.001
Kidneys	0.001
Liver	0.001
Lungs	0.001
Muscle	0.001
Ovaries	0.001
Pancreas	0.001
Red marrow	0.001
Skin	0.276
Spleen	0.001
Testes	0.001
Thymus	0.001
Thyroid	0.001
Uterus	0.001

a. Values are input into IREP as a lognormal distribution; value listed is the geometric mean; the GSD is 7.91.

The dose values are for exposure to NU because this provides a higher external dose than other enrichments (except DU). This provides a dose estimate that is favorable to claimants because most uranium at the Apollo uranium facility was NU or EU. NU provides a higher external dose, per unit activity, because significant contributions come from the short-lived progeny of ^{238}U (^{234}Th and ^{234m}Pa).

The skin dose value represents the dose at 1 meter above the ground. This provides an overestimate of dose to the skin for cancers above the waist and an underestimate of dose for cancers below the waist.

The dose should be entered into the IREP input as a lognormal distribution with a GSD of 7.91 (IREP Parameter 2) as photons of energy from 30 to 250 keV as indicated in Table 6-10. This provides a favorable estimate of POC for all organs, even though some of the photon energy is likely to be of higher energy.

7.1.2 Parks Township Site

Because the residual period for the Parks Township site includes the period of 1981 – 1983, when the Apollo site was still in operation, it is possible that Parks Township workers were exposed to DOE work if they visited the Apollo site. If dosimeter readings are available for these years of the residual period, the dose should be based on the recorded and missed dose (Section 6.0) unless it is known that the worker did not visit the Apollo site. External dosimetry data might exist during later years of the residual period; however, operations were not part of the nuclear weapons-related program, and are not covered for dose reconstruction under EEOICPA (Leiton 2011).

Work at the Parks Township site included work with plutonium in Building A and work with HEU in Building C. Estimates of external dose from these two materials are described below. If it is not known where the energy employee worked during the residual period, the dose based on uranium exposures can be assigned as a favorable to claimant assumption.

An estimate of the external dose from residual plutonium activity at the Parks Township site has been made using the same approach used for the Apollo site uranium external dose. The mean surface concentration at the end of the operational period as described in Section 7.4, derived in support of internal dose estimates during the residual period, was 2.74×10^5 dpm/m^2 (GSD = 4.97). This value can be used to estimate the annual external dose to workers exposed to the residual activity. The annual dose is evaluated using Equation 7-1, using DCFs for exposure to uniform activity on a ground plane. The annual dose was evaluated using the plutonium isotopic compositions given in Table 5-3 as a function of material age. The highest dose was obtained for the 20-year aged material for all organs. However, all doses were less than 0.001 rem. Therefore, no external dose need be assigned during the residual period for Parks Township workers based on potential exposure to plutonium for Building A.

The work at the Parks Township site included work with HEU in Building C. No information has been found related to air concentrations in Building C during the operating period. Because the Apollo uranium operations covered a longer period than the Building C operations, and because high levels of airborne uranium were present during the Apollo operations (AEC 1960a,b,c, 1961 a,b), the residual activity levels for the Apollo site provide a favorable to claimant estimate of external dose from residual activity at the Parks Township HEU operations in Building C. The annual external dose values for the Parks Township residual period are the same as those given in Table 7-1.

7.2 EXTERNAL AMBIENT DOSE FROM RESIDUAL ACTIVITY

All unmonitored workers are assigned external dose as described in Section 7.1. The assigned external dose would cover any additional ambient external dose, and the assignment of ambient dose is not necessary.

7.3 OCCUPATIONAL MEDICAL DOSE

During the residual period, medical X-ray doses are not to be included in the dose reconstruction because the work is not directly related to DOE employment.

7.4 INTERNAL DOSE FROM RESIDUAL ACTIVITY IN THE WORKPLACE

7.4.1 Apollo Site

The following information provides a method for estimating exposures during the residual radiation period due to uranium contamination. Valid bioassay data are unlikely to be available during the residual period. However, monitoring data can be used instead of the default intake assumptions given below to limit dose, as appropriate (NIOSH 2008b).

Guidance in *Dose Reconstruction During Residual Radioactivity Periods at Atomic Weapons Employer Facilities* (ORAUT 2012b) describes methods to estimate intake of radionuclides during residual periods. The method relates residual surface contamination to workplace air concentration during periods after completion of DOE work.

The majority of the DOE-related work at the Apollo site was involved with scrap recovery and fuel fabrication operations in the Apollo uranium facility (East Bay of the Main Building). The HASL reports documented the average workplace air concentration to which workers were exposed during 1960 and 1961 (AEC 1960a,b,c, 1961a,b). The results of these studies are described in Section 5. The

resulting mean average daily air concentration was estimated to be 210 dpm/m^3 with a GSD of 7.91 (assuming a lognormal distribution).

The mean average daily air concentration can be used to estimate the residual surface concentration, using guidance from ORAUT (2012b). The annual deposition amount is estimated using a deposition velocity of 0.00075 m/s, with deposition assumed to occur for 1 year. Using this approach, a surface concentration of uranium is estimated as follows.

$210\ dpm/m^3 \times 31{,}536{,}000\ s/yr \times 0.00075\ m/s = 4.97 \times 10^6\ dpm/m^2$

This mean surface concentration is described as a level at the end of the operating period that is favorable to claimants. The deposited material is assumed to be resuspended and inhaled during the residual period. The amount of resuspension is assumed to be reduced with time due to fixing of the material on surfaces and to depletion (ORAUT 2012b). The depletion factors applied to each year are listed in Table 4-2 of ORAUT (2012b). The depletion factors indicated for the residual concentration at the end of the operational period should be used for the first year; the remaining years should be reduced by factors listed in Table 4-2 of ORAUT (2012b).

The air concentration for each year is estimated using a resuspension factor (ORAUT 2012b) of 1×10^{-6}/m. Application of this resuspension factor and the above-described depletion factors to the residual contamination level of 4.97×10^6 dpm/m^2, results in the air concentration and annual intakes in Table 7-2. The intake evaluation is based on exposure for 2,000 hr/yr and an inhalation rate of 1.2 m^3/hr.

Table 7-2. Uranium air concentration and annual intake in the Apollo site residual period.

Year	Air concentration (dpm/m^3)	Inhalation Intake (dpm/yr)	Ingestion Intake (dpm/yr)[a]
1984	4.97	1.19E+04	10,500
1985	3.89	9.33E+03	8,200
1986	3.04	7.31E+03	6,400
1987	2.38	5.72E+03	5,000
1988	1.87	4.48E+03	3,900
1989	1.46	3.50E+03	3,100
1990	1.15	2.75E+03	2,400
1991	0.90	2.16E+03	1,900
1992	0.70	1.68E+03	1,500
1993	0.55	1.32E+03	1,200
1994	0.43	1.03E+03	910
1995	0.34	8.09E+02	710

a. Doses are assigned as a lognormal distribution with a GSD of 7.91.

The intakes in Table 7-2 can be used to estimate the internal dose to the target organ for the years of employment for the worker. The estimated internal doses are assigned as a lognormal distribution with a GSD of 7.91, corresponding to the distribution of the average daily air concentrations used to estimate the annual uranium intake. The uranium intake is represented as ^{234}U in the dose estimate. The dose should be evaluated for the three uranium material solubility types of F, M, and S, with the dose from the highest type used in the IREP input.

Although uranium-aluminum alloy was present at the Apollo scrap recovery facility, the form of the material was not likely to be an inhalation hazard and modeling an intake of uranium aluminide is not necessary for the Apollo site. The internal dose analysis should include the potential inadvertent

ingestion of uranium activity, based on guidance in OCAS-TIB-009, *Estimation of Ingestion Intakes* (NIOSH 2004).

The daily intake rate (dpm/d) is estimated to be 0.2 times the average daily air concentration expressed in units of dpm/m^3. Using the air concentration at the end of the operating period of 210 dpm/m^3, an intake rate of 10,500 dpm/yr is obtained for 250 workdays per year for the first year of the residual period (1984). This intake rate is reduced for subsequent years by the depletion factors listed in Table 4-2 of ORAUT (2012b). This provides an assessment of ingestion intake that is favorable to claimants.

7.4.2 Parks Township Site

The following information provides a method for estimating exposures during the residual radiation period due to plutonium and uranium contamination. Valid bioassay data for plutonium are unlikely to be available during the residual period. However, monitoring data can be used instead of the default intake assumptions given below to limit dose, as appropriate (NIOSH 2008b). For the first three years of the Parks Township residual period (1981 – 1983), bioassay monitoring data for uranium may be available. If the worker may have worked at the Apollo site during this period, the data may be related to DOE work and would be valid for use in assignment of internal dose. For workers who were known to have only worked at the Parks Township site during this period, the bioassay data should only be used to limit internal dose.

Guidance in ORAUT-OTIB-0070, *Dose Reconstruction During Residual Radioactivity Periods at Atomic Weapons Employer Facilities* (ORAUT 2012b) describes methods to estimate intake of radionuclides during residual periods. The method relates residual surface contamination to workplace air concentration during periods after completion of DOE work.

The majority of the DOE-related work at the Parks Township site involved plutonium fuel fabrication. While no formal air monitoring studies are available for these activities, an assessment of bounding general area air activity in the facilities can be made through a review of available air sampling data. Starting in October 1967, NUMEC reported personnel exposures above regulatory thresholds to the AEC (Caldwell 1967b). A bounding representation of air activity at the Parks Township site was determined based on a review of these reports. The data set includes 105 reported values between 1966 and 1982. The reported values were given as MPC-hours (for plutonium exposure), which were converted to dpm/m^3 by multiplying by the MPC and dividing by the number of hours given for the reported value. Values reported at the MPC were set to the MPC air concentration. The resultant median air concentration was estimated to be 11.6 dpm/m^3 with a GSD of 4.97 (assuming a lognormal distribution). The air concentration values and the lognormal fit of the data are shown in Figure 7-1.

The median plutonium air concentration from the above-described study was used to estimate the residual surface concentration along with the guidance from ORAUT (2012b). The annual deposition amount was estimated using a deposition velocity of 0.00075 m/s, with deposition assumed to occur for 1 year. Using this approach, a surface concentration of plutonium was estimated as follows.

$$11.6 \text{ dpm/m}^3 \times 31{,}536{,}000 \text{ s/yr} \times 0.00075 \text{ m/s} = 2.74 \times 10^5 \text{ dpm/m}^2$$

This surface concentration is favorable to claimants at the end of the operating period. The deposited material was assumed to be resuspended and inhaled during the residual period. The amount of resuspension was assumed to reduce with time due to fixing of the material on surfaces and to depletion (ORAUT 2012b). The depletion factors that were applied to each year are described in Table 4-2 of ORAUT (2012b). The depletion factors indicated for the residual concentration at the end of the operational period should be used for the first year; the remaining years should be reduced by factors listed in Table 4-2 of ORAUT (2012b).

The air concentration for each year was estimated using a resuspension factor of 1×10^{-6}/m (ORAUT 2012b). Application of this resuspension factor and the above-described depletion factors to the residual contamination level of 2.74×10^{5} dpm/m^2 resulted in the air concentration and annual intakes in Table 7-3. The intake evaluation was based on exposure for 2,000 hr/yr and an inhalation rate of 1.2 m^3/hr.

Because the plutonium is based on gross alpha general air monitoring results, the activity represents the total alpha activity. The dose from the residual activity should be evaluated using the fractional isotopic activity of each radionuclide given in Table 5-3. The values in this table must be converted to fractional alpha activity for evaluation of dose. The age of the material must also be considered because the maximum dose (per unit alpha activity) is obtained for the shorter aged material. The beginning of plutonium processing at the NUMEC plutonium facility was in 1962. Therefore, the age of the residual material could have ranged from zero to 20 years or more if aged material was used in 1962. The minimum age would be zero years if it was deposited at the end of the operating period in 1980. A favorable to claimant approach would be to use the zero aged material activity fractions for the period from 1981 – 1984, the 5-year activity fractions from 1985 – 1989, the 10-year activity fractions from 1990 – 1994, the 15-year activity fractions from 1995 to 1999, and the 20-year activity fractions for 2000 – 2004. A minimizing approach would be to use the 20-year activity fractions for all years.

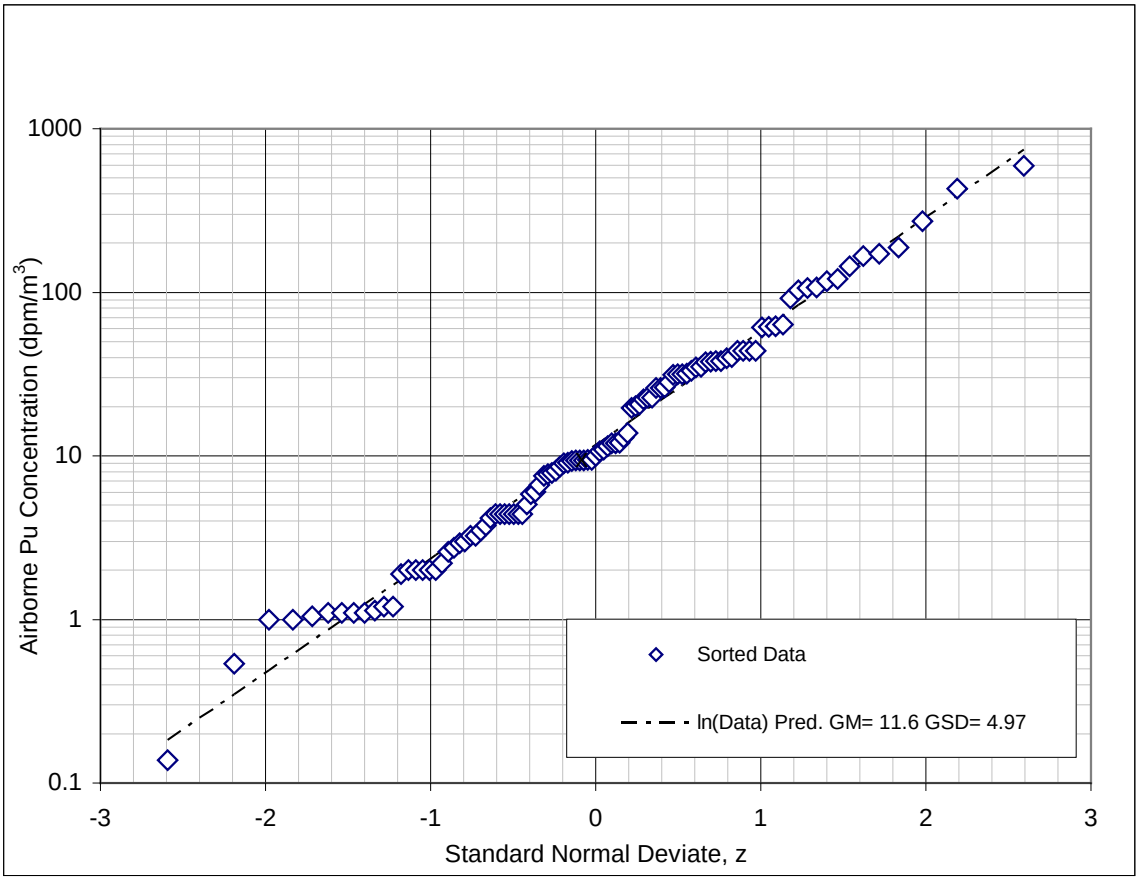

Figure 7-1. Reported plutonium air concentrations.

The intakes in Table 7-3 can be used to estimate the internal dose to the target organ for the years of employment for the worker. The estimated internal doses are assigned as a lognormal distribution with a GSD of 4.97, which corresponds to the distribution of the average daily air concentrations that were used to estimate the annual plutonium intake. The dose should be evaluated for the two plutonium material solubility types of M and S, and the dose from the highest type should be used in

the IREP input. Because the residual plutonium is in the form of aged material, type Super S plutonium should be considered and adjustments made based on guidance in ORAUT (2010a).

The internal dose analysis should include the potential inadvertent ingestion of plutonium activity, based on guidance in NIOSH (2004). The daily intake rate (dpm/d) is estimated to be 0.2 times the average daily air concentration in units of dpm/m^3. Using the air concentration at the end of operations of 11.6 dpm/m^3, an intake rate of 580 dpm/yr is obtained for ^{239}Pu for 250 workdays per year for the first year of the residual period (1981). This intake rate is reduced for subsequent years by the depletion factors listed in Table 4-2 of ORAUT (2012b). This provides an assessment of ingestion intake that is favorable to claimants.

Table 7-3. Plutonium air concentration and annual intake in the Parks Township site residual period.

Year	Air concentration (dpm/m^3)	Inhalation Intake (dpm/yr)[a]	Ingestion Intake (dpm/yr)[a]
		^{239}Pu	^{239}Pu
1981	0.27	658	580
1982	0.21	516	450
1983	0.17	404	360
1984	0.13	316	280
1985	0.10	248	220
1986	0.081	194	170
1987	0.063	152	130
1988	0.050	119	100
1989	0.039	93	82
1990	0.030	73	64
1991	0.024	57	50
1992	0.019	45	39
1993	0.015	35	31
1994	0.011	27	24
1995	0.0089	21	19
1996	0.0070	16.8	15
1997	0.0055	13.2	12
1998	0.0043	10.3	9.0
1999	0.0034	8.1	7.1
2000	0.0026	6.3	5.6
2001	0.0021	4.9	4.4
2002	0.0016	3.9	3.4
2003	0.0013	3.0	2.7
2004	0.0010	2.4	2.1

a. Doses are assigned as a lognormal distribution with a GSD of 4.97.

Table 7-4. Activity fraction for plutonium material at the Parks Township plutonium facility.

Mixture designation	Fraction of alpha activity				
Years of aging	0	5	10	15	20
Pu-238	0.18	0.14	0.12	0.10	0.093
Pu-239/240	0.82	0.68	0.59	0.54	0.51
Pu-241	31.9	20.6	14.2	10.2	7.5
Am-241	0.000	0.19	0.29	0.35	0.40

The intakes in Table 7-3 relate to DOE work conducted in Building A of the Parks Township site. Work with uranium in Building C could also contribute to internal dose for workers during the residual period. However, no information was found to allow determination of the residual uranium activity from work performed in Building C. For workers who are known to be exposed from Building C residual activity, the internal dose can be evaluated based on intakes given in Table 7-2, with intakes shifted back three years to account for the difference in residual period between the Apollo and Parks Township sites. These intakes are given in Table 7-5 for all years of the Parks Township residual period. If the work location is not known for a Parks Township worker, then the higher of the doses based on plutonium (Table 7-3) and uranium (Table 7-5) can be assigned as a favorable to claimant approach.

Table 7-5. Uranium air concentration and annual intake in the Parks Township residual period.

Year	Air concentration (dpm/m^3)	Inhalation Intake (dpm/yr)[a]	Ingestion Intake (dpm/yr)[a]
1981	4.97	1.19E+04	10500
1982	3.89	9.33E+03	8200
1983	3.04	7.31E+03	6400
1984	2.38	5.72E+03	5000
1985	1.87	4.48E+03	3900
1986	1.46	3.50E+03	3100
1987	1.15	2.75E+03	2400
1988	0.90	2.16E+03	1900
1989	0.70	1.68E+03	1500
1990	0.55	1.32E+03	1200
1991	0.43	1.03E+03	910
1992	0.34	8.09E+02	710
1993	0.26	6.34E+02	560
1994	0.21	4.96E+02	440
1995	0.16	3.89E+02	340
1996	0.13	3.04E+02	270
1997	0.10	2.38E+02	210
1998	0.077	1.86E+02	160
1999	0.061	1.47E+02	130
2000	0.048	1.14E+02	100
2001	0.037	8.95E+01	79
2002	0.029	7.01E+01	62
2003	0.023	5.50E+01	48
2004	0.018	4.30E+01	38

a. Doses are assigned as a lognormal distribution with a GSD of 7.91.

8.0 ATTRIBUTIONS AND ANNOTATIONS

All information requiring identification was addressed via references integrated into the reference section of this document.

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GLOSSARY

absorption
In external dosimetry, process in which radiation energy is imparted to material. In internal dosimetry, movement of material to blood regardless of mechanism.

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

accuracy
The characteristics of an analysis or determination that ensures that both the bias and precision of the resultant quantity will remain within the specified limits.

activity
Amount of radioactivity. The International System unit of activity is the becquerel (1 disintegration per second); the traditional unit is the curie [37 billion (3.7×10^{10}) becquerels].

activity fraction
Proportion of the total activity due to a particular radionuclide.

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

air sampling
Collection of samples of the ambient atmosphere to detect or measure the presence of radioactive material in the air.

albedo dosimeter
Thermoluminescent dosimeter that measures the thermal, intermediate, and fast neutrons scattered and moderated by the body or a phantom from an incident fast neutron flux.

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

ambient atmosphere
Depending on context, the air external to buildings, in the outside environment, or the air that surrounds an individual.

americium–beryllium (AmBe)
Common neutron source created by homogeneously mixing ^{241}Am and beryllium powders. Neutrons are produced when ^{241}Am alpha particles interact with beryllium nuclei.

atomic weapons employer (AWE) [42 U.S.C. § 7384l(5)]
Entity other than the United States, that—(A) processed or produced, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining and milling, and (B) is designated by the Secretary of Energy as an

atomic weapons employer for purposes of the [Energy Employees Occupational Illness] compensation program.

attenuation
Process by which absorption and scattering reduces the number of particles or photons passing through a body of matter.

beta dose
Designation (i.e., beta) on some records for external dose from beta and less-energetic X-ray and gamma radiation, often for shallow dose or dose to the lens of the eye.

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

bioassay
Measurement of amount or concentration of radionuclide material in the body (*in vivo* measurement) or in biological material excreted or removed from the body (*in vitro* measurement) and analyzed for purposes of estimating the quantity of radioactive material in the body.

body burden
Amount of radioactive material in an individual's body at a particular point in time.

byproduct
Material left over from a nuclear, physical, or chemical process designed to produce a particular substance.

calcine
(1) Dry solid (grainy or granular) product of a chemical process that removes liquids from a solution. (2) Process for creating the chemical reaction that removes liquids from a solution.

calibration
Adjustment or determination of the response or reading of an instrument relative to a standard or a series of conventionally true values.

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

cohort
Group of individuals selected for inclusion in a study. See *Special Exposure Cohort*.

confidence level
The interval about an estimate of a stated quantity within which the value of the quantity is expected to be with a specified probability. See *uncertainty*.

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

control rod
Neutron-absorbing device in a reactor used to slow or speed the reaction.

decay

(1) Disintegration of atomic nuclei from spontaneous radioactivity including alpha, beta, and neutron radiation, often accompanied by gamma radiation. (2) Decrease in the amount of radioactive material over time due. See *half-life*.

decay products

See *progeny*.

decommissioning

Removal of a facility from service, usually involving decontamination of radioactivity to specified levels and often involving demolition of the facility.

decontamination

Reduction or removal of radioactive material from a structure, area, object, or person. Decontamination can occur through (1) treating the surface to remove or decrease the contamination or (2) allowing natural radioactive decay to occur over a period of time.

deep dose

See *personal dose equivalent*.

depleted uranium (DU)

Uranium with a percentage of ^{235}U lower than the 0.7% found in natural uranium.

dose

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

dose equivalent (*H*, DE)

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

dosimeter

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

dosimetry

Measurement and calculation of internal and external radiation doses.

dosimetry system

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

electron

Basic atomic particle with negative charge and a mass 1/1,837 that of a proton. Electrons surround the positively charged nucleus of the atom. See *element*.

enriched uranium (EU)

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

enrichment

Isotopic separation process that increases the percentage of a radionuclide in a given amount of material above natural levels. For uranium, enrichment increases the amount of ^{235}U in relation to ^{238}U. Along with the enriched uranium, this process results in uranium depleted in ^{235}U. See *depleted uranium* and *enriched uranium*.

environmental occupational dose

Dose received from radiation site-related activities (i.e., above normal background levels) while on a site, which is often recorded by monitoring stations in specific areas or along the boundaries of facilities (e.g., plant stack emissions).

error

Difference between the correct, true, or conventionally accepted value and the measured or estimated value. Sometimes used to mean estimated uncertainty. See *accuracy* and *uncertainty*.

exchange period (frequency)

Period (weekly, biweekly, monthly, quarterly, etc.) for routine exchange of dosimeters.

exposure

(1) In general, the act of being exposed to ionizing radiation. See *acute exposure* and *chronic exposure*. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens.

external dose

Dose received from radiation emitted by sources outside the body.

film dosimeter

Package of film for measurement of ionizing radiation exposure for personnel monitoring purposes. A film dosimeter can contain two or three films of different sensitivities, and it can contain one or more filters that shield parts of the film from certain types of radiation. When developed, the film has an image caused by radiation measurable with an optical densitometer. Also called film badge.

fission product (FP)

Radionuclides produced by fission or by the subsequent radioactive decay of radionuclides.

gamma radiation

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

geometric standard deviation (GSD)

In probability theory and statistics, the geometric standard deviation describes the spread of a set of numbers whose preferred average is the geometric mean.

glovebox
Enclosure with special rubber gloves through which an operator can handle radioactive or toxic material without risk of injury or contamination normally operated at a slightly reduced pressure so that air leakage, if any, is inward.

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

high-temperature gas-cooled reactor (HTGR)
Nuclear reactor cooled with helium.

highly enriched uranium (HEU)
Uranium enriched to at least 20% ^{235}U for use as fissile material in nuclear weapons components and some reactor fuels.

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

ingestion
Process of taking a substance into the body through the mouth.

insoluble
Having very low solubility. No material is absolutely insoluble. See *absorption type* and *soluble*.

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

Integrated Modules for Bioassay Analysis (IMBA)
Computer program that uses bioassay results and other information to calculate intakes of radionuclides and subsequent doses.

Interactive RadioEpidemiological Program (IREP)
Computer program that uses a person's calculated annual organ doses and other information (e.g., gender, age at diagnosis, and age at exposure) to calculate the probability of causation of a specific cancer for a given pattern and level of radiation exposure.

internal dose
Dose received from radioactive material in the body.

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

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

***in vivo* bioassay**

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

ionizing radiation

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

limit of detection (LOD)

Minimum level at which a particular device can detect and quantify exposure or radiation. Also called lower limit of detection and detection limit or level. See *minimum detectable level*.

low-level radioactive waste (LLRW)

Unwanted radioactive materials that do not require shielding during normal handling or transport because of their low activity. Mildly radioactive material is usually disposed of by incineration and burial.

minimum detectable activity or amount (MDA)

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

minimum detectable concentration (MDC)

Minimum detectable activity (or amount) in units of concentration. See *minimum detectable activity*.

minimum detectable level (MDL)

See *minimum detectable activity.*

missed dose

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

mixed oxide (MOX)

Nuclear fuel that contains both plutonium oxide and uranium oxide.

monitoring

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

natural uranium (NU)

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

neutron film dosimeter

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

neutron radiation

Radiation that consists of free neutrons unattached to other subatomic particles emitted from a decaying radionuclide. Neutron radiation can cause further fission in fissionable material such as the chain reactions in nuclear reactors, and nonradioactive nuclides can become radioactive by absorbing free neutrons.

nonpenetrating dose

Dose from beta and lower energy photon (X-ray and gamma) radiation that does not penetrate the skin. It is often determined from the open window dose minus the shielded window dose. See *dose*.

nuclear track emulsion, type A (NTA)

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

nuclide

Stable or unstable isotope of any element. Nuclide relates to the atomic mass, which is the sum of the number of protons and neutrons in the nucleus of an atom. A radionuclide is an unstable nuclide.

occupational dose

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

occupational environmental dose

Dose received while on the grounds of a site but not inside a building or other facility.

occupational medical dose

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

open window (OW)

Area of a film dosimeter that has little to no radiation shielding (e.g., only a holder and visible light protection). See *film dosimeter*.

penetrating dose

Dose from moderate to higher energy photons and neutrons that penetrates the outer layers of the skin. See *dose*.

personal dose equivalent [$H_p(d)$]

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

photon radiation

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

probability of causation (POC)

For purposes of dose reconstruction for the Energy Employees Occupational Illness Compensation Act, the percent likelihood, at the 99th percentile, that a worker incurred a particular cancer from occupational exposure to radiation.

progeny

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

radiation

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

radiation source

Any object or substance that emits radiation.

radiation worker

Employee who works on, with, or in the proximity of radiation-producing machines or radioactive materials.

radioactive

Of, caused by, or exhibiting radioactivity.

radioactive waste

Radioactive solid, liquid, and gaseous materials for which there is no further use. Wastes are generally classified as high-level (with radioactivity as high as hundreds of thousands of curies per gallon or cubic foot), low-level (in the range of 1 microcurie per gallon or cubic foot), intermediate level (between these extremes), mixed (also contains hazardous waste), and transuranic.

radionuclide

Radioactive nuclide. See *radioactive* and *nuclide*.

recycled uranium (RU)

Uranium first irradiated in a reactor then recovered through chemical separation and purification. RU contains minor amounts of transuranic material (e.g., plutonium and neptunium) and fission products (e.g., technetium) or uranium products (e.g., ^{236}U) after purification.

resuspension

Process by which the wind lifts small particulates (generally from soil) into the air, which for radionuclides can result in an exposure hazard. Radionuclides released to the air undergo a cycle of suspension in the air, deposition on the ground, resuspension, and redeposition.

However, the initial suspension process is generally included in resuspension if the initial contaminating event did not result from deposition of airborne material.

shallow dose
See *personal dose equivalent*.

shielding
Material or obstruction that absorbs ionizing radiation and tends to protect personnel or materials from its effects.

solubility type
See *absorption type*.

soluble
In relation to health physics, refers to the speed with which radionuclides naturally dissolve in lung fluids. See *absorption type* and *insoluble*.

source term
Description of the types and quantities of radioactive materials. The source term is usually specified as a rate of exposure or an amount of radioactivity (i.e., becquerels or curies) sometimes by specific radionuclide. Often includes distinctions in chemical and physical forms and history of the material.

Special Exposure Cohort (SEC) [42 U.S.C. § 7384l(14)]
… "member of the Special Exposure Cohort" means a Department of Energy employee, Department of Energy contractor employee, or atomic weapons employee who meets any of the following requirements:

(C) (i) Subject to clause (ii), the employee is an individual designated as a member of the Special Exposure Cohort by the President for purposes of the compensation program under section 7384q of this title.

(ii) A designation under clause (i) shall, unless Congress otherwise provides, take effect on the date that is 180 days after the date on which the President submits to Congress a report identifying the individuals covered by the designation and describing the criteria used in designating those individuals.

Special Nuclear Material (SNM)
Plutonium or uranium enriched to a higher-than-natural assay including ^{239}Pu, ^{233}U, uranium containing more than the natural abundance of ^{235}U, or any material artificially enriched in one of these isotopes.

standard deviation
Square root of the variance, or the measure of spread in a group of numbers.

thermoluminescence
Property that causes a material to emit light as a result of heat.

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

thermoluminescent dosimeter chip

Small block or crystal of lithium fluoride in a thermoluminescent dosimeter. A TLD-600 dosimeter contains a chip made from more than 95% ^{6}Li for neutron radiation detection, and a TLD-700 dosimeter contains a chip made from more than 99.9% ^{7}Li for photon and beta radiation detection.

transuranic (TRU) elements

Elements with atomic numbers above 92 (uranium). Examples include plutonium and americium.

uncertainty

Standard deviation of the mean of a set of measurements. The standard error reduces to the standard deviation of the measurement when there is only one determination. See *accuracy*, *confidence level*, and *error*.

unmonitored dose

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

whole-body counter

Equipment used to perform *in vivo* bioassay. Radiation emitted from radioactive material deposited throughout the body is measured.

whole-body (WB) dose

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

X-ray

See *X-ray radiation*.

X-ray radiation

Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.

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Table A-1. HASL-82 weighted BZA survey, December 1959.[a]

Operator	Number of persons	Average daily weighted exposures in dpm/m^3
UF_6 to UO_3 Operator (Kiln and Filter)	4	790
Ceramics Reduction Furnace Operator	1	560
Ceramics Laboratory Operator	3	6,300
Sinter Furnace Operator	3	130
Ceramics laboratory–Group Leader	3	940
Co-precipitator Operator	2	950
U-Zr Recovery Operator–Chip	1	43
U-Zr Recovery Operator–Dissolving	1	49
U-Zr Recovery Operator–Reduction	1	39
Wet Chemistry Laboratory–Group Leader	1	37
Wet Chemistry Laboratory–Chemist	2	37
Wet Chemistry Laboratory–Technician	3	37
Gas Analysis–Chemist	1	7.5
Gas Analysis–Technician	1	7.5
Spectrographic Laboratory–Group Leader	1	17
Spectrographic Laboratory–Technician	3	17
Metallographic Group Leader	1	7
Grinding and Polishing Technician	1	7
Coated Sphere Operator	3	12

a. Data from AEC (1960a).

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Table A-2. HASL-92 weighted BZA survey, June 1960.[a]

Operator	Number of persons	Average daily weighted exposures in dpm/m^3
UF_6 to UO_3 Operator (Kiln and Filter)	3	370
UF_6 to UF_3 Operator (Hydrolysis and Waste Filter)	6	73
Ceramic Reduction–Kiln Operator	3	96
Powder Preparation–Ceramics Fabrication	1	820
Ceramics Laboratory Operator	1	5,500
Sinter Furnace Operator	3	94
Centerless Grinder	2	100
Ceramic Fabrication–Prepress Operator (Graphite)	2	6,300
Ceramic Fabrication–Press Operator (Graphite)	2	490
Ceramic Fabrication–Pellet Press Operator (Uranium)	2	73
Ceramics Laboratory–Group Leader	3	600
Inspection – Graphite	6	57
Inspection – Uranium	2	57
U-Zr Recovery Operator–Extraction	2	160
U-Zr Recovery Operator–Powder	1	820
Wet Chemistry Laboratory (Group Leader, Chemists, Technicians)	10	9
Gas Analysis–(Chemist, Technician)	2	39
Spectrographic Laboratory–(Group Leader, Technician)	6	19
Metallographic (Group Leader and Technician)	3	10
Coated Sphere Operator	6	11
Maintenance	21	110
Sweepers	2	150
Sampler	3	110

a. Data from AEC (1960b).

Table A-3. HASL-103 weighted BZA survey, October 1960.[a]

Operator	Number of persons	Average daily weighted exposures in dpm/m^3
Ceramics Laboratory	3	36
Ceramic Laboratory–Group Leader	3	24
Ceramics Fabrication–Reduction Kiln Operator	3	65
Ceramics Fabrication–Pellet Press	1	19
Sinter Furnace Operator	3	16
Ceramics Reduction Tube Operator	1	590
Centerless Grinder	2	23
Coated Sphere Operator	6	31
CRP-3 Leach Operator	4	26
CRP-3 Filter and Feed Prep Operator	4	39
CRP-3 Extraction Operator	3	28
CRP-3 ADU Filter and Reduction Kiln Operator	3	27

a. Data from AEC (1960c).

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Table A-4. HASL-106 weighted BZA survey, December 1960.[a]

Operator	Number of persons	Average daily weighted exposures in dpm/m^3	% U
Ceramics Laboratory Technician	2	680	93
Ceramics Laboratory Powder Production	6	910	93
Ceramics Laboratory Clerk	1	310	93
Ceramics Laboratory Group Leader	1	310	1.8-93
Ceramics Fabrication–Centerless Grinder	2	190	3.5
Ceramics Fabrication–Pellet Press	2	120	3.5
Ceramics Fabrication–Sinter Furnace	3	120	3.5-93
Ceramics Fabrication–Quality Control	4	61	3.5
CP-2 (UF_6 to UO_2) Filter and Drying	3	190	93
CP-2 (UF_6 to UO_2) Hexdrolysis	3	140	93
U-Zr powder	3	220	93
U-Zr Extraction Cascade	3	180	93
CRP-3 Extractor	3	110	1.8
CRP-3 Precipitate and Filter	1	82	1.8
CRP-3 Helper	1	88	1.8
Coated Sphere Operator	6	46	93

a. Data from AEC (1961a).

Table A-5. HASL-114 weighted BZA survey, May 1961.[a]

Operator	Number of persons	Average daily weighted exposures in dpm/m^3	% U
Ceramics Laboratory Reduction Tubes	1	130	5.7
Ceramics Fabrication–Pellet Press	2	35	5.7
Ceramics Fabrication–Centerless Grinder	2	200	5.7
Ceramics Fabrication–Sinter Furnace	3	27	3.4-5.7
Ceramics Fabrication–Quality Control	2	33	5.7
Ceramics Fabrication–Group Leader	2	31	5.7
CP-2 (UF_6 to UO_3) Hydrolysis	3	24	3.4
CP-2 Ammoniation and Filtering ADU	3	27	3.4
CP-2 Filtrate	3	20	3.4
CP-2 Calciner	3	57	3.4
CP-2 Reduction kiln Operator	3	31	3.4
CRP-2 (U-Zr) Dissolving and Filtering	3	39	5.7
CRP-2 Ammoniation, ADU Filter Conversion	3	27	5.7
CRP-2 Helper	3	33	5.7
CRP-2 Extraction Cascade	3	27	5.7
Coated Sphere Operator	6	22	Normal

a. Data from AEC (1961b).