ORAU TEAM
Dose Reconstruction
Project for NIOSH

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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.

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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)
FFTF	Fast Flux Test Facility
FP	fission product
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
IMBA	Integrated Modules for Bioassay Analysis
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
NCRP   National Council on Radiation Protection and Measurements
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

ORAU   Oak Ridge Associated Universities
ORISE   Oak Ridge Institute for Science and Education
OTIB   ORAU Team technical information bulletin

PA   posterior-anterior
pCi   picocurie
POC   probability of causation

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)

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 <u>INTRODUCTION</u>

1.1 PURPOSE

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

In this document the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer [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. § 7384I(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 2007a):

- 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 2007a). 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 Special Exposure Cohorts (SECs) for the NUMEC sites in Apollo and Parks Township, Pennsylvania.

The SECs cover employees who worked at one or both of the Apollo and Parks Township facilities. This site profile also 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 Special Exposure Cohort

An SEC 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 work 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 where the bioassay data was 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 2007c) 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 may be considered for partial dose reconstruction (Leavitt 2007).

1.3.2 Parks Township Site Special Exposure Cohort

An SEC 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 work days aggregating to at least 250 work days occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees in the SEC (Leavitt 2008).

NIOSH has determined, and the Secretary of Health and Human Services 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 where 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 2008) 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 may 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 number SNM-145 and Source Material License number C-3762 that the U.S. Atomic Energy Commission (AEC) 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).

Between 1958 and 1983, the Apollo site manufactured LEU uranium dioxide fuel for use in nuclear power plants. The process consisted of converting low-enriched UF_6 to UO_2. In 1963, a second product line was added to produce HEU (>20% ^{235}U) uranium fuel for U.S. Navy propulsion reactors. Other operations included analytical laboratories, HEU and LEU scrap recovery, forming UO_2 into pellets and other shapes, uranium storage, and research and development (R&D) (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 was 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 site is approximately 33 miles northeast of Pittsburgh in the borough of Apollo in Armstrong County, Pennsylvania. The 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. These buildings were on the east side of the site between Warren Avenue and the Kiskiminetas River. They were bordered on the north, south, and west by a metals processing facility that was not owned by B&W. 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 some 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. The parking lot area contained the Laundry Building and a Small Block Building made of concrete blocks. 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. Figure 2-1 is a general layout of the Apollo site (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
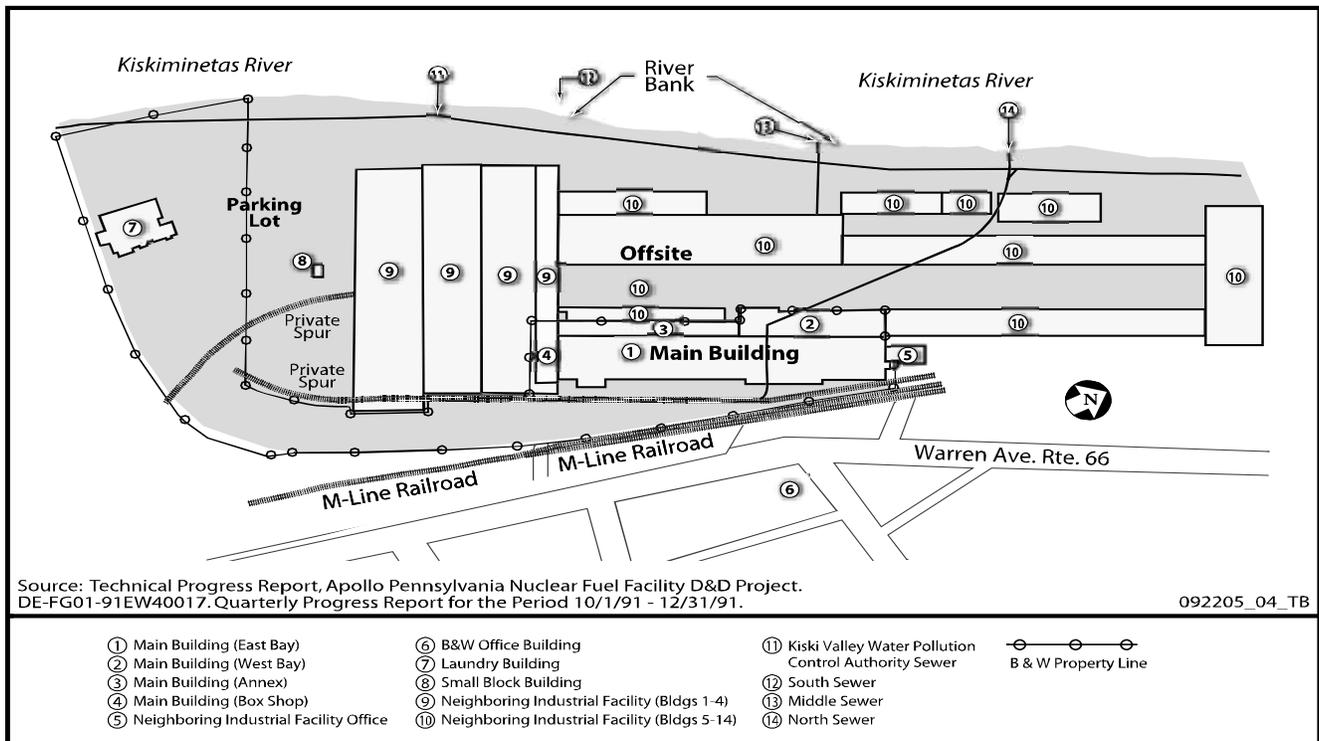
Figure 2-1. Apollo site layout.

room with the exception of the filter press section of the plant waste treatment area (NUMEC 1963). Production and process areas are described in Table 2-1. Some of these areas are illustrated in Figure 2-2.

2.1.3 Process Descriptions

Brief summaries for the principal operations are given below and additional details can be found in the series of AEC Feasibility and Health and Safety Laboratory (HASL) reports provided in the cited references (Forscher 1963; AEC 1960a,b,c, 1961a,b). Inherent in all of the operations is nuclear criticality safety that governs not only the operations and storage but also the movement of material within the facility. Nuclear criticality safety is maintained at the facility through the control of one or more of the conventional parameters of geometry, mass, concentration, and control of neutron interaction between sub-critical units. The standard administrative policy at this and other similar facilities is to control two such parameters whenever possible. For birdcage units, the two control parameters are geometry (birdcage dimensions) and a ^{235}U mass limit that can be placed within the birdcage. A review of the available literature shows that no criticality accidents have occurred during Apollo site operations.

2.1.3.1 General Process Operations

NUMEC Apollo process operations were varied. Since NUMEC Apollo acted as 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 commercial work for commercial reactors but much of the work was in the development of better fuel configurations for the burgeoning Navy nuclear program. Even though this work is not considered weapons related work the SNM and radiation level exposures resulting from this work during the AEC operational years (1957 to 1983) are to be included in radiation dose reconstruction as per NIOSH policy. Only some examples are listed below to illustrate the breadth of the SNM operations that occurred at the NUMEC Apollo site.

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 & DU) UO_2, and U_3O_8	Early 1959 to 1972
PC-1	Process chemistry	HEU, EU, DU, $(NH_4)_2U_2O_7$, UO_3, UF_6, UF_4, U nitrate, UO_2 and U_3O_8	1957–1983 HEU-1957–1978 LEU-1957–1983
PC-2	Process chemistry	HEU, EU, DU, U nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
PC-3	Process chemistry	HEU, EU, DU, U nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CP-1	Chemical processing	HEU, EU, DU, U nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CP-2	Chemical processing	HEU, EU, DU, U nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CRP-1	Chemical reprocessing	HEU, EU, DU, U 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, U 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, U 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, U 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, U 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, U nitrate, UO_2, UF_6, UF_4, U_3O_8, FPs, and TRU elements	1957–1983
Office Building	Off site	Uranium	1957–1983
Laundry facility		Co-60, Pu, Th, and U	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).
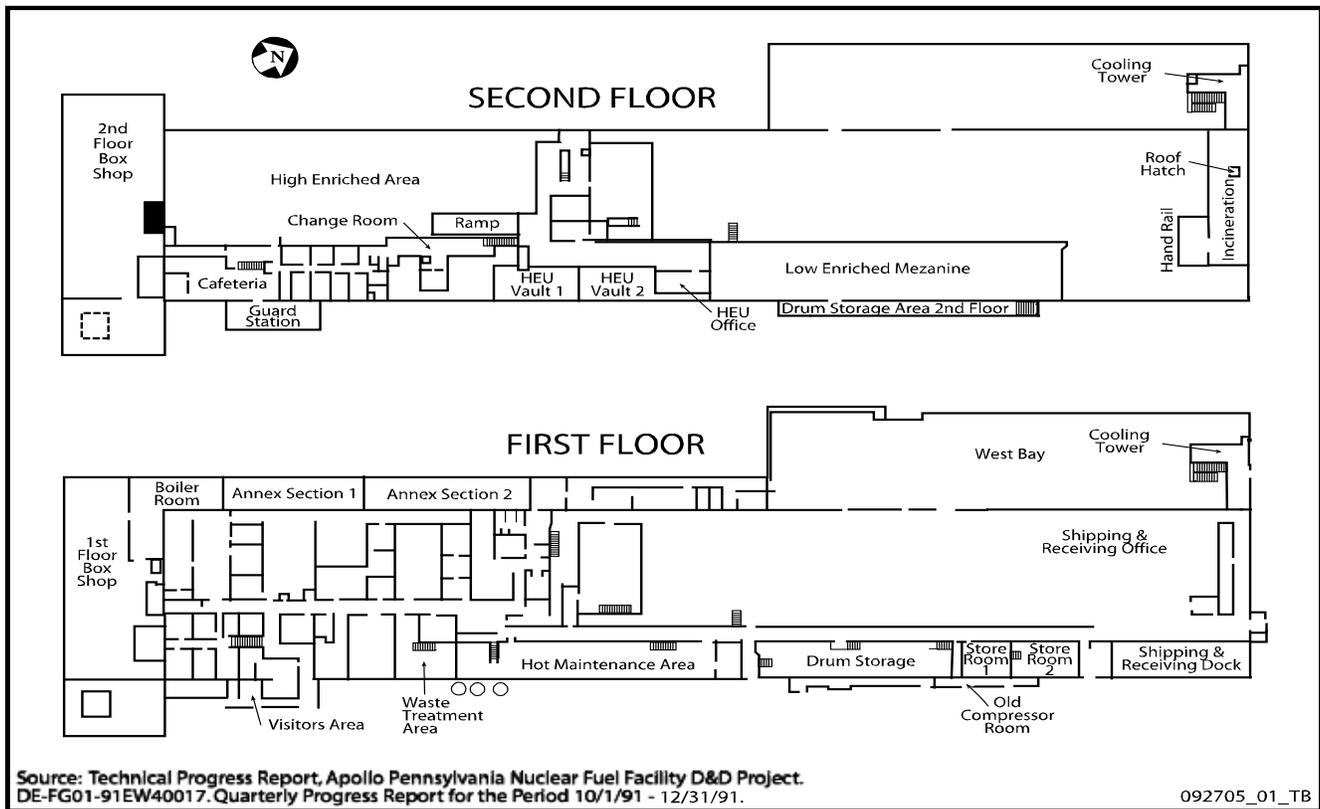
Figure 2-2. Apollo site floor plan.

2.1.3.2   Various Uranium Process Activities

A general description of processes for various enrichments of uranium at Apollo is outlined in AEC Health and Safety Laboratory (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 about 1963. (AEC 1960a,b,c, 1961a,b; NUMEC ca. 1963). 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. A 5-in. diameter UF_6 cylinder was removed from a shipping/storage container known as a "birdcage" and placed in a heating mantle connected to the system, and brought to operating temperature and pressure. The UF_6 gas was fed at a controlled rate into a water-circulating column where the UF_6 was hydrolyzed to an aqueous solution of UO_2F_2 and HF. This solution was then piped to a second column where it 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 scraped off, placed in shallow metal containers called fry pans, and transferred to drying hoods where the ADU was decomposed at a controlled temperature to UO_3. The UO_3 was transferred from the pans to small polyethylene containers in an unvented glovebox and subsequently transferred 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 means of a rotary kiln rather than the previous fry pan method. 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 then 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 or reduction room where UO_3 from the Chemical Conversion plant was loaded into a reduction tube and reduced to UO_2 by passing wet hydrogen through the heated tube. UO_2 was removed from the reduction tube, placed into a polyethylene container in a glovebox, weighed and transferred to the Ceramic Laboratory for additional fabrication. HASL-92 (AEC 1960b) noted that the UO_3 was reduced to UO_2 using a rotary kiln instead of the reduction furnace.

A schematic illustrating the UO_2 production process is presented in Figure 2-3.
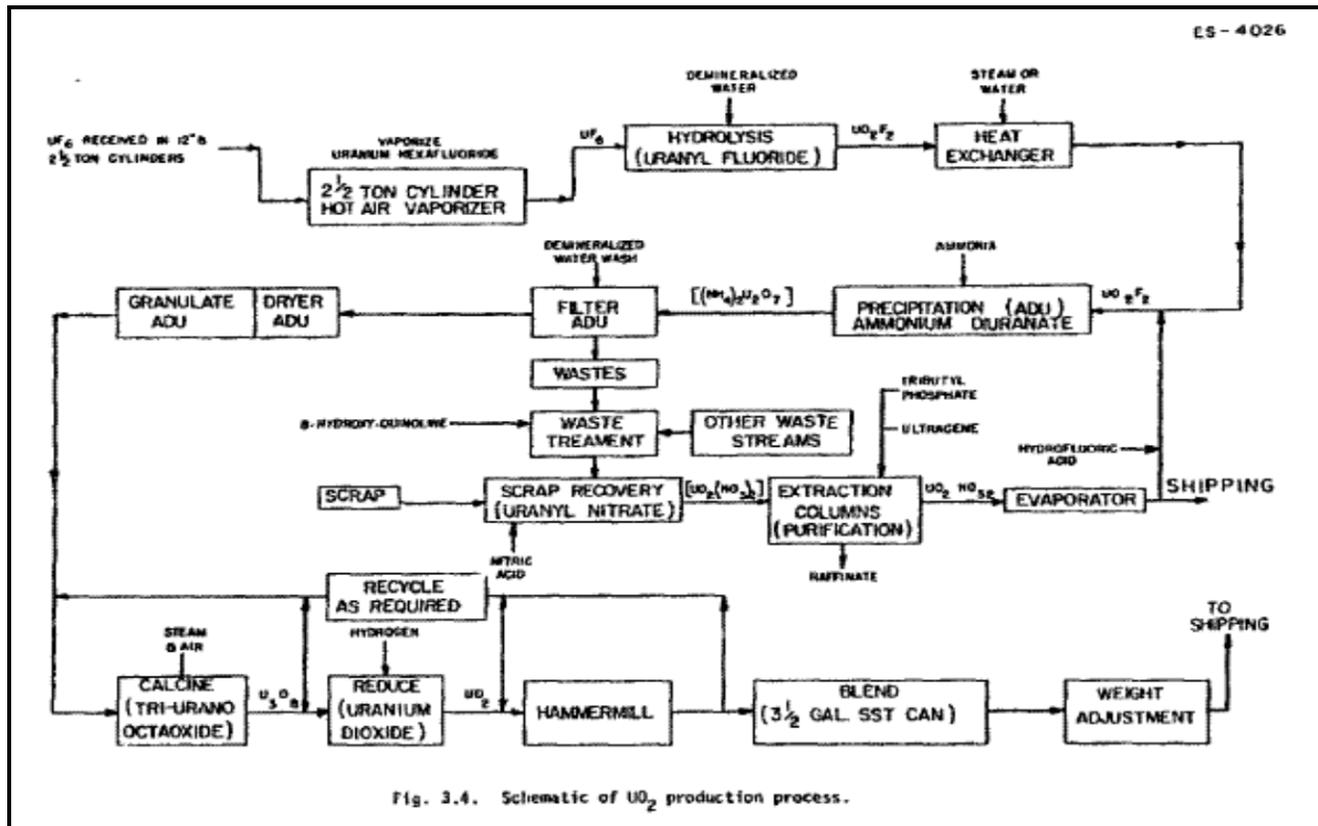
Fig. 3.4. Schematic of UO_2 production process.

Figure 2-3. UO_2 production process at Apollo site (Franke and Makhijani 1998).

Ceramics fabrication was performed in the Ceramics Fabrication Area where UO_2 was hammer-milled in a ventilated enclosure then moved to the blender glovebox where aerowax was added and the mixture 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 then loaded into shallow metal pans called "firing boats" and sintered. Sintered UO_2 was then 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, then blended in a twin shell blender and prepressed by means of a Drake press in a large polyethylene tent located in the ceramic laboratory. The prepressed slugs were subsequently fed into a granulator and 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 to 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 then introduced. The pentachloride was reduced by the hydrogen so that niobium metal deposited on the UO_2 particles. Vibration of the system allowed for 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 was performed in the Chemical Reprocessing Rooms. The 1960 HASL-82 (AEC 1960a) survey report described the recovery of uranium (U_3O_8) from uranium-zirconium scrap as follows:

One-gal cans containing uranium-zirconium chips and oil were opened. The oil was drained and the chips transferred to wire mesh baskets. Chips were then degreased with trichloroethane in a ventilated degreasing tank. The chips were then transferred to 6-L beakers and dissolved in hydrofluoric acid in chemical fume hoods. The zirconium remained in solution and part of the uranium precipitated as UF_4. After complete dissolution of chips, the batch was heated and hydrogen peroxide added to oxidize the insoluble UF_4 to soluble UF_6. The batch was then filtered to remove any remaining insoluble material. The filtrate was then 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 the HASL-92 (AEC 1960b) survey report, the experimental development of recovering U_3O_8 (93%) by solvent extraction was under development at Apollo (AEC 1960b). In addition, UF_4 was being converted to U_3O_8 by ignition with the U_3O_8 subsequently 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.

According to *Procedures for Recovery of Uranium Scrap* (NUMEC ca. 1963), incoming scrap was received in criticality-safe birdcages and stored at Advanced Materials Center Parks Township near Leechburg, Pennsylvania, 5 miles north of the Apollo site on Route 66. Specific lots were then moved to the Apollo site on NUMEC trucks and logged into the Apollo process storage area located on the second floor of the Apollo site. One drum at a time was then carried through the checking process and wheeled to the head end of the processing area. An accountability check of the information supplied by the shipper was then performed, including a gross piece count. Net weight checks were performed on chips, fuel elements, and fuel assemblies and then recorded on Form CRP-2-A. Pickle liquors and other solutions containing over 5 g/L were received in Eversafe 5-in. containers (dissolved scrap procedure followed for liquids). Pickle liquors and other solutions containing less than 5 g/L were received in 55-gal drums (dissolved scrap procedure followed for liquids). Degreasing of material was performed as necessary. Fines were removed from oil as necessary (NUMEC ca. 1963).

NUMEC dissolved the scrap in two designated areas: CRP-2 and CRP-3. Dissolution was performed in dissolving hoods in 5-in.-diameter, stainless-steel beakers. The total quantity in each hood was

limited to 350 g of ^{235}U. Uranium-zirconium chips and pieces were mixed with hydrofluoric acid. U-Al alloy pieces were mixed with nitric acid. UO_2 and U_3O_8 were mixed with nitric acid. BeO-UO_2 scrap was mixed with HF-HNO_3. The resulting solutions were placed into a mixing column and the container was then sampled, assayed, and weighed (NUMEC ca. 1963). The product solution from the various dissolution methods was processed to generate insoluble UF_4, and ultimately converted to the final product of U_3O_8 or UO_2.

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. Fuel element composition and specifications were provided in the letter. The total ^{235}U content of 2,850 fuel elements was to be 79.339 kg. There is some process description in the letter but not much detail. 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 NUMEC Apollo Facility 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 (Katine 1961).

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 equipped with a main burner in the firebox and an after-burner in the stack just above the firebox. Both burners used natural gas for fuel (Caldwell 1968a).

Packages of contaminated waste awaiting incineration were stored in designated areas. Each package was labeled with the ^{235}U content. Packages were burned at a rate of 30 to 35 lb/hr. Ashes were collected in stainless steel, 1-gal containers. The ash receiver can was changed when a total of 300 g of ^{235}U had been charged or when it was full. 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. The pail was labeled and placed on a temporary storage rack pending laboratory analysis. On receipt of analytical data, the pail label was completed with the uranium (^{235}U) content and 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). Prior to 1968, this may not have been the case and work in this area was perhaps the highest area 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. Downstream of the scrubber, the gases passed through a packed tower where fine particulates were scrubbed from the gas stream by a counter-current flow of water. These gases were then discharged through a 15-ft stack. A makeup air duct was used to maintain negative head on the scrubber and for controlling the flow of flue gases through the system. Exhaust air from the ash handling glovebox passed through a prefilter and final 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 NUMEC 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 having 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 by the site to characterize thorium exposures during this period. According to a health protection program review conducted 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 various research projects conducted 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 reactor research laboratory. The research involved chemical process development with various forms of uranium compounds and metal.

2.1.4   <u>Source Term</u>

There are three main sources that describe the amounts and types of radioactive material that were handled at the Apollo 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 HASL reports, and process feasibility reports, which contain information regarding 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 number SNM-145 and Source Material License number 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: SNM-145 Renewal Application October 31, 1972, 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 any 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 NUMEC 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 any one area ranged from milligrams to hundreds of kilograms. Occasional work involving other uranium chemical forms, such as UF_6 or uranyl nitrate was also conducted. Uranium forms included 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 onsite was less than that of uranium overall, but the thorium activity in use in an area at any 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 of the equipment was removed and sent to Chem-Nuclear (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's trench, which served as a sump drain for washing machine wastewater, was removed in April 1989 (B&WNES 1997).

All of the 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 processing plant (B&WNES 1997).

Soil acceptable decommissioning criteria are described in the *Final Technical Report, Apollo Decommissioning Project, Apollo, Pennsylvania*, but residual period building contamination levels

were not covered (B&WNES 1997). The decommissioning criteria that were followed for the Apollo site are presented in Table 2-3.

Table 2-3. Decommissioning criteria for Apollo site.[a]

Radionuclide	Release concentration (pCi/g)
Total TRU	25
Sr-90	5
Tc-99	220
Co-60	8
Am-241	30
Th-232	10

a.   B&WNES (1997).

The background exposure rates were 9.5 µR/hr and the average concentration in the soil was 4 pCi U/g.

Another decommissioning study conducted by Oak Ridge Institute for Science and Education (ORISE) for the NRC in 1993 indicates that the main building ground area exposure rates were between 9 to 12 µR/hr and between 10 to 13 µR/hr in the south parking lot. Total uranium concentrations ranged from 6.5 to 2,200 pCi/g for single point samples and 5.3 to 280 pCi/g for the grid block soil samples (Adams 1993).

As of August 23, 1978, NUMEC had completed decommissioning of its HEU processing at Apollo. All process and related equipment were removed by this point in time. NUMEC indicated that access to the area was controlled to authorized personnel. In 1982, the NRC conducted a confirmatory survey to identify the remaining 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 for Apollo remaining on and in the floors, walls, pad and ceiling was estimated as 35,548.55 g of total uranium with about 23,743.27 g of ^{235}U (NRC 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's groundwater monitoring data, final termination survey, and a confirmatory survey in 1996. On April 14, 1997, after notifying the Pennsylvania Department of Environmental Protection, 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). A graphic of the area is provided in Figure 2-4.
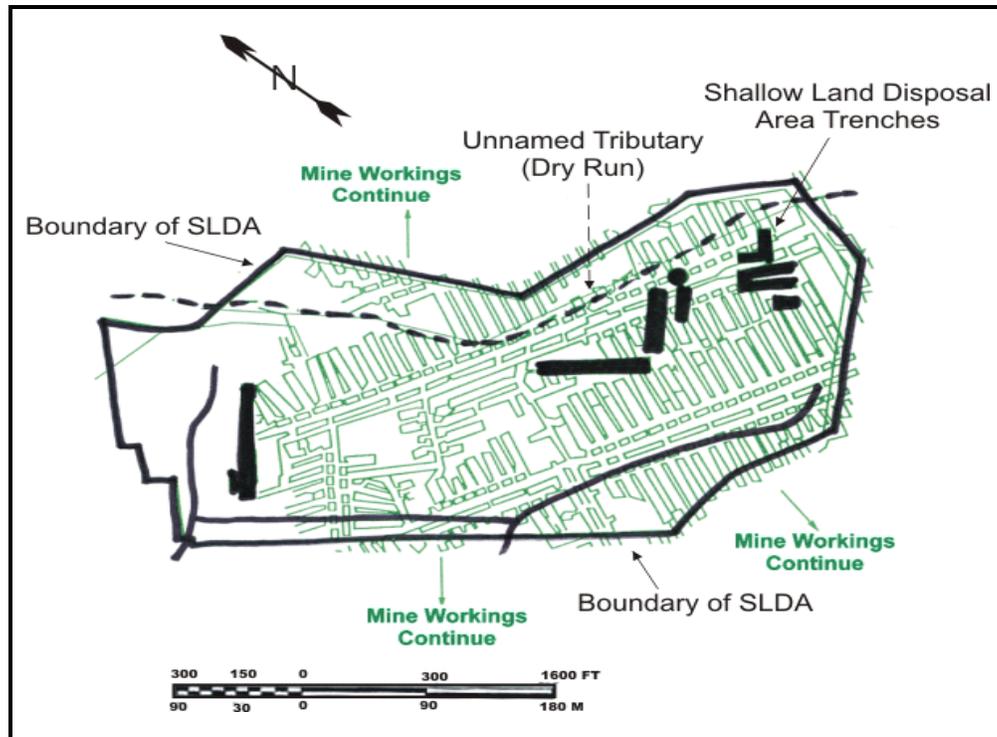
Figure 2-4. B&W Shallow Land Disposal Area (USACE 2006).

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 laboratory 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 for the soil beneath the former concrete slab of the basement of the administrative building and of the surface once the area was backfilled and leveled (survey conducted May 23, 1995). All results were found to meet the guidelines previously established for unrestricted release. Special emphasis was made to survey closely the northeast area of the administrative building where the wet laboratory existed and the southern end, the location of the instrument repair and calibration laboratory (Bores 1995). Exposure rate measurements, including background, ranged from 10 to 12.5 µR/hr, with an average of approximately 11 µR/hr. The guideline was 5 µR/hr above background, meaning the acceptable level was approximately 15 µR/hr (Bores 1995). The reported values essentially represent background levels.

Soil samples were analyzed by ORISE. The results indicated that no ^{235}U was found above the detection limit (not stated). Total uranium was not found statistically above the counting uncertainties and measured less than 3.5 pCi U/g of soil. Total thorium concentrations were found to range from 1.51 to 2.02 pCi/g with an average of 1.86 pCi/g. The guidelines for unrestricted release of soil with residual concentrations of EU and thorium are 30 pCi U/g and 10 pCi Th/g of soil, respectively. All soil concentrations were found to be well within these guidelines. The background concentrations were NOT subtracted.

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 approximately 3 miles southeast of Leechburg and 30 miles northeast of Pittsburgh. The Parks Township site consisted of Buildings A, B, and C and 85 acres of undeveloped land. NUMEC leased the land from Kiski Valley Enterprises in 1959 and eventually purchased the land in the early 1970s. The site was 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 expansion included construction of 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 with the plutonium annex 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 NUMEC Parks Township facilities were fabrication of plutonium fuel, the preparation of HEU fuel, and the production of zirconium/hafnium bars under AEC and later NRC License No. 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 Parks Township facility 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 2-4 and discussed further in Section 2.6. Although many of the dates of operation in Table 2-4 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 illustrated in Figure 2-5, which shows the Parks Township site layout.

2.2.2 Facilities

Building A
This facility 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. The layout of Building A is show in Figure 2-6.

Table 2-4. Parks Township site area descriptions.

Building or area location	Operations	Radionuclides	Period of operation
Building A–Plutonium processing facility	Fabrication of Pu reactor fuel pellets, blankets, rods	PuO_2, Pu nitrate and oxalates (AmBe, PuBe, 1959–1970), various alpha, beta, and gamma sources	1960–1980
A–Fab 1	Pu conversion, fuel fabrication for FFTF	Pu nitrate, Pu 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	Pu nitrate, Pu oxide, depleted UO_2	1962–1980
A–Fab 3	Manufacturing operations, metallography, quality control of FFTF fuel	Pu nitrate, Pu 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, Pu and Am metal	1963–1980
A–Fab 5	Scrap recovery	Pu various forms	1963–August 1, 1967,
	Analytical laboratory work	All, small quantities of radioactive samples	1979–1980
A–Fab 6	Scrap recovery	Pu 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
Plutonium Facility Trailer Storage Area	Storage of large quantities of plutonium and uranium	Pu (nonpyrophoric), DU, NU, and EU (to 5% ^{235}U)	1961–1980
Building B–Multipurpose fabrication building	DU, NU, Th, Pu	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, U and Pu and other TRU elements and FPs	1961–1980

Building or area location	Operations	Radionuclides	Period of operation
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 and could include HEU, Pu, Th, 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), and NUMEC (1963)
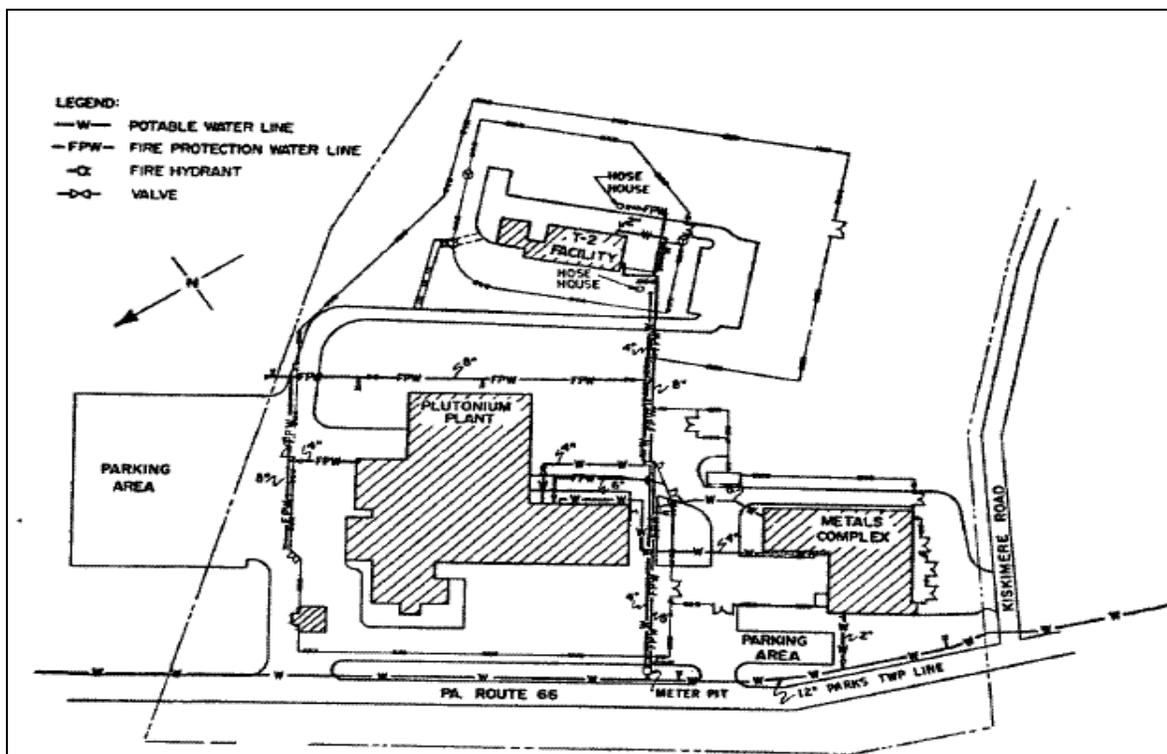
Figure 2-5. Parks Township site layout. Plutonium Plant (Building A), Metals Complex (Building B), and T-2 Facility (Building C) (Austin 1979).

Building B

This complex 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.
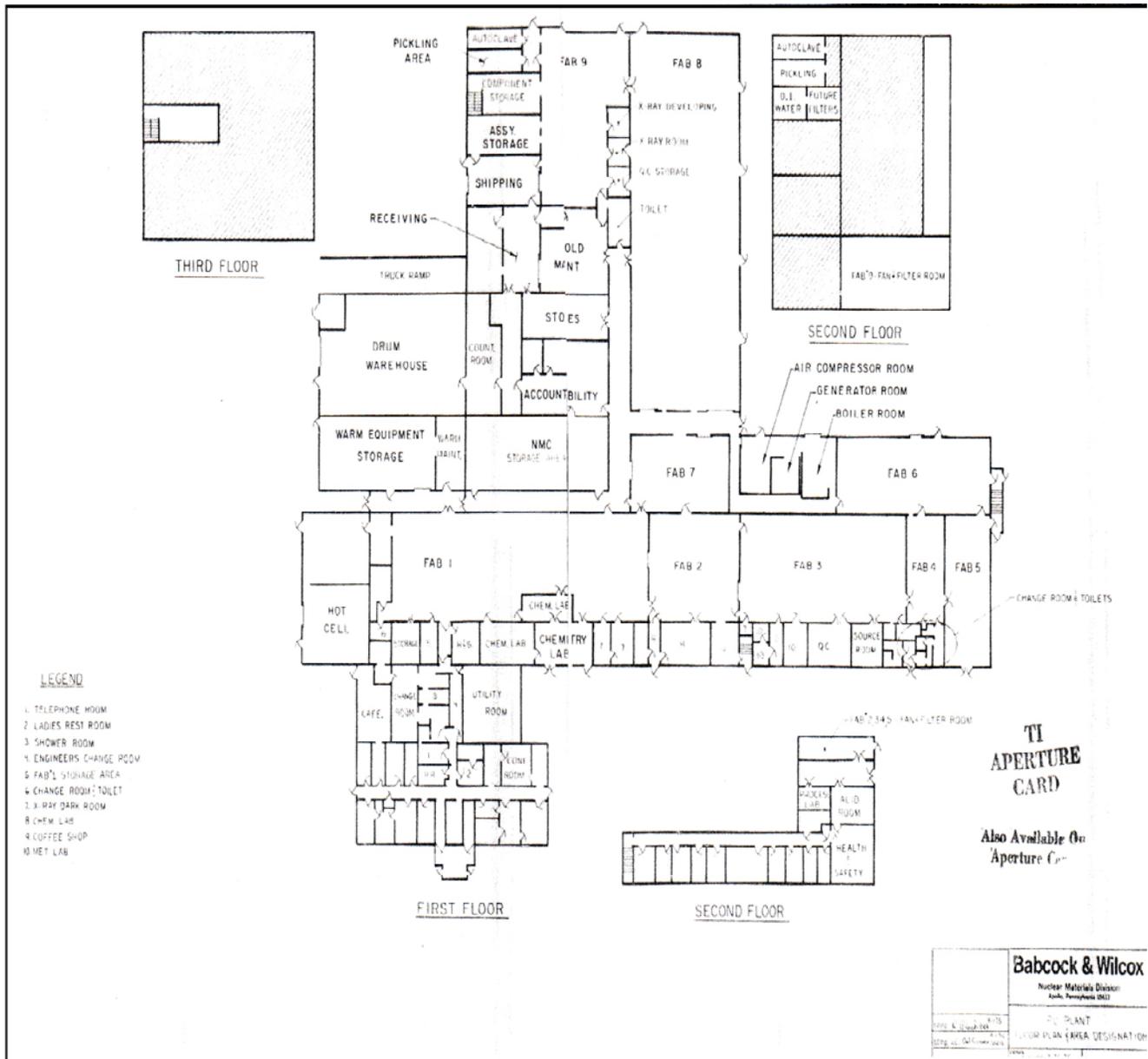
Figure 2-6. Parks Township Plutonium Plant (Building A) layout (Austin 1979).

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 that was 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 approximately 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 dependent 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 into plutonium oxide.

Conversion of plutonium nitrate to plutonium oxide was performed at the north end of Fab 1 in HEPA-filtered gloveboxes. The conversion process started with small batches of plutonium nitrate to which either oxalic acid or hydrogen peroxide was added. The same equipment could be used for either precipitation process. The filter cake from the precipitation and filtration steps was transferred to a muffle furnace for calcining to PuO_2. The PuO_2 was placed in metal cans and stored for shipment back to the customer or for use in Building A fuel fabrication.

The chemical, physical, mechanical, and radiochemical properties of in-process fuel, finished fuel forms, and radiation sources were determined on small samples in several small laboratory rooms adjacent to the west side of Fabs 1, 2, 3, and 4. Separate laboratories existed for wet chemistry, metallography and mechanical properties, and radiochemistry. An analytical chemistry laboratory was installed in Fab 5 after removal of the scrap recovery equipment, but the Fab 5 laboratory only operated for about 6 months during 1979 to 1980. Only laboratory-size quantities of reagent-grade chemicals were used in these laboratories.

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 was made that consisted of a plutonium oxide film that was deposited on one or both sides of a flat metal backing plate. Limited quantities of various other neutron, beta, and gamma sources were made to specific 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, except for 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 also located 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 the workers or the environment. Air from the Hot Cell passed through a HEPA filter before reaching the stack. Two liner boxes were used in the cell for performing work on materials that contained alpha emitters. Each box had inlet and output filters and the air exiting the boxes passed into the main cell exhaust system, which was HEPA filtered. Two other ventilation systems served hot cell operations. One system exhausted the metallography and dissolver cell in the Hot Handling Facilities room, and the fission gas fume hood in the Cell Control Area. The other system exhausted the chemical fume hood in the Cell Control Area.

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.

Significant quantities of chemicals were not used in Hot Cell operations, although various reagents were used for metallography and cleaning of sealed sources. Solid waste from inside the Hot Cell was packaged and disposed of at approved disposal sites. High-activity liquid waste was solidified for disposal as solid radioactive waste. Low-activity liquid waste, such as from washing the cell walls before a manned entry, was sent to an outside underground tank. This tank was fed by two floor drains in the Hot Cell and one in the Hot Handling Facilities room. The tank liquid was periodically pumped back into the Building A low-level liquid waste discharge system. Eighteen dry storage units (5-in.-diameter aluminum pipes set in concrete) were located outside the north end of the building just west of the former Hot Cell underground tank. The dry storage units were used to temporarily store containers of radioactive material going into or out of the hot cells.

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. A small supply room for chemicals was originally located near the metallographic room west of Fab 1, but was moved later to Fab 8. Large containers of chemicals (bulk chemicals) were stored in several locations including outside the building. 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. The Warm Maintenance Area was near

the east side of Fab 1. Most of the equipment used in the gloveboxes and radiochemical fume hoods was modified before use to facilitate both maintenance and repair within the containments.

All the 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 then 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 and final stage of construction 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 first as the Metals Building and then as Building B.

DU was the primary radioactive material that was 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 were produced at the northeast end of Building B. Plutonium-238 was processed in a room in the northwest corner of Building B. All ^{238}Pu work was performed in interconnected gloveboxes. Receiving and shipping operations were conducted in a chemical fume hood. Nonradioactive metals and alloys were also processed in significant quantities in Building B. The majority of the work was production of crystal-bar zirconium and hafnium and zirconium-beryllium alloys.

The chemicals that were used in Building B were those required to support the radioactive materials and specialty metals processing and manufacturing. This included the use of nitric and oxalic acids in ^{238}Pu processing, materials such as trichloroethylene (TCE), and iodine in crystal-bar production.

2.2.3.2.1   The Hafnium Facility

The original product from the Hafnium Facility was crystal-bar hafnium. Crystal-bar hafnium was produced by reacting hafnium sponge with nonradioactive iodine to form hafnium iodide gas. The gas was introduced into a vessel that contained a high-purity hafnium wire. A reaction between the gas and the heated wire dissociated the gas and deposited the hafnium on the wire to form crystal-bar hafnium. After reaction, the iodine gas was condensed and cycled back through the operation. The hafnium bar was packaged and shipped to customers. Crystal-bar zirconium was also produced in the Hafnium Facility using a similar process.

A specialty zirconium alloy product was produced by hydriding ingots of beryllium and zirconium with hydrogen gas in a furnace. The resultant hydride was ground into powder, heated under vacuum to dissociate the hydride, milled, and sieved. The dehydrided powder was blended with titanium powder to produce homogeneous lots of zirconium-beryllium-titanium alloy powder. The blended powder was

pressed into solid rings, packaged, and shipped. Metal powders of other alloys were also produced in the Metals Plant using an identical process.

Under contract with AEC, ^{238}Pu nitrate was converted into an oxide product in a room in the northwest corner of the Hafnium Facility known as the Plutonium Annex. The conversion process for the ^{238}Pu nitrate was very similar to the process for converting ^{239}Pu nitrate into fuel products in Building A, but only oxalate precipitation was used. The conversion was performed in eight HEPA-filtered gloveboxes. Both products and wastes were shipped to government sites. Wastewater from the sink, shower, and janitor sink were discharged into one of two interconnected 1,000-gal underground concrete tanks. The tanks were sampled to verify the water met AEC and Pennsylvania discharge criteria before release to the Kiskiminetas River. A laboratory in the west side of the Hafnium Facility opened in 1991 and operated until the building was ready for final decontamination.

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, sandblasting, decreasing, and other associated metalworking machinery. Machining of DU was performed in the Machine Shop. In addition, the repair and refurbishment of equipment from the Apollo and Parks Township facilities 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 all took place on the ground floor. The second floor contained primarily offices and a training room, although a small environmental laboratory was in use 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 a variety of equipment to:

1. Melt zirconium-beryllium rods;

2. Heat treat tantalum and zirconium billets or plates for drawing or rolling;

3. Extrude copper-clad zirconium and uranium billets;

4. Forge uranium and zirconium products;

5. Hot roll tantalum- and boron-containing stainless steels;

6. Cold roll magnesium, tantalum, and zirconium;

7. Vacuum anneal metals;

8. Draw and rotary sway zirconium and uranium products;

9. Blend zirconium-beryllium powders and press into rods for later electric arc melting;

10. Air induction melt of stainless-steel billets that contained boron;

11. Vacuum induction melt of DU-molybdenum ingots that were sent to Building A for remelting into plutonium-uranium-molybdenum alloy fuel plates;

12. Centerless grinding of uranium and zirconium products;

13. Powder processing of depleted U_3O_8 compacts;

14. Electroplate copper, nickel, or cadmium cladding onto DU products;

15. Manufacture and encapsulate sodium carbonate wafers and hafnium- or boron-containing stainless steel control rods;

16. Pickle metal products after vapor (TCE) degreasing and cleaning with nitric acid (occasionally with HNO_3-HF) and/or a caustic solution; and

17. Rinse with deionized water.

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. During the mid-1970s, a portion of the high bay area was used to machine nonradioactive fan components.

The second floor of the Metals Plant initially contained only one office. However, over the years other 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 to 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 a 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 and hydrogen peroxide, then diluting the solution with demineralized water. The diluted uranium solution was fed through dialysis columns and an electrolysis cell. The uranium solution was 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 another licensed site for finishing operations. The majority 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 blow-down, 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 facility for recovery.

Hexanol from the forming columns was recycled through both continuous and batch recovery systems. The continuous recovery system combined spent hexanol and a small amount of fresh hexanol from the storage tank, and then added small quantities of ammonia gas. The solution was fed to a wash column where contact with water removed the impurities in the hexanol. The hexanol

was distilled to remove absorbed water before being returned to the forming columns. The water was sent to a 15,000-gal holding tank for monitoring before pumping to the evaporator-concentrator. The bottoms from the evaporator-concentrator were pumped to one of two asphalt-lined holding ponds on the east side of Building C, and the vapors passed through a scrubber where volatile chlorides were removed using a sodium hydroxide solution. The spent scrubber solution was discharged to the holding ponds.

Shower and sink waste was piped to a holding tank for monitoring. The waste was piped to the septic tank if it was less than 1% of the uranium limits set in the then-applicable 10 CFR Part 20 for liquid effluents to unrestricted areas. Waste that exceeded the administrative limit was pumped to an evaporator-concentrator for boildown before recovery at the Apollo facility. Sanitary waste from commodes and urinals was piped to a septic tank that fed a 1,600-ft^2 leach field.

Gaseous effluent from the evaporator-concentrator was sent to a scrubber for treatment before release to the atmosphere. The gaseous effluent from the evaporator-concentrator consisted of air, carbon dioxide, nitrogen, and water vapor that were scrubbed with sodium hydroxide before discharge to the environment. Flue gases were discharged directly to the atmosphere.

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 2-5 through 2-8.

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

The Parks Township site radiological source term included uranium, thorium, plutonium, and fission and activation products. <u>No definitive information is currently available to relate measurement of one component of the source term (e.g., plutonium) to another unmonitored component (e.g., americium) for any given area or process.</u> 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 NUMEC 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 one area ranged from milligrams to hundreds of kilograms. Work with chemical forms of uranium such as UF_6 or uranyl nitrate was also 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 any given time could have been greater or less than uranium activity. Thorium dioxide was obtained from virgin thorium sources.

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

Source/chemical or physical form	Period	Maximum possession	Reference
PU and EU	1961–1969	Any combination of Pu and U-235 up to 400 kg	Nussbaumer 1965
	1969–1979	Any combination of Pu and U-235 up to 1,000 kg	AEC 1969
PU (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 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
U, any enrichment U-235	1991–end	Less than 250 g U	Haughney 1991
NU or DU and Th	1969–1979	No limits	AEC 1969
NU or DU any form	1979–1991	100,000 kg U	Rouse 1979, Haughney 1991
Pu with greater than 5 wt % Pu-238	1969–1979	300 g	AEC 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 1969,
U-233 as evaporated calibration or reference sources	1981–1991	1 g	Austin 1981, Austin 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 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
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 2-6. 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	5000 kg	Puechl 1965, AEC 1969
	1991–end	Possession: <250 g U-235	Haughney 1991
EU above 5 wt % U-235	1961–1979	500 kg	Puechl 1965 , AEC 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
Pu as fully clad, encapsulated, or otherwise contained material in operating areas or in any form in the storage vault	1969–1979	500 kg	AEC 1969
NU or DU and Th	1969–1979	No limits	AEC 1969
Pu 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
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 2-7. 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 Th	1969–1979	No limits	AEC 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
Pu 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 2-8. 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 1969
Pu 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
Pu plant storage trailer	Pu 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
	Pu 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
	U, 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
	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

Place of use	Source/chemical or physical form	Period	Maximum possession	Reference
Any state except Agreement States	Neutron irradiator source	1969–unknown	Up to 96 g Pu as PuBe neutron source	AEC 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

Plutonium. Chemical forms included metals, nitrates, and oxides. The heat source and heart pacemaker programs used ^{238}Pu. The ^{238}Pu 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).

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 that was 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 approximately 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 dependent 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 Parks Township 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 that site groundwater was within established limits, 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 to be used 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 to 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 still 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 ceased all HEU operations and commenced 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 that were associated with Building C. Pond remediation included (1) removing the liquids and sludge, solidifying them in 55-gal drums for shipment to burial, (2) breaking up the asphalt liner for packaging in wooden boxes and steel drums for shipment to burial, and (3) packaging soil higher than background into boxes or drums for shipment to 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 then packaged for burial. After the equipment was removed, B&W then initiated a cleanup of the walls, floors, and ceilings to remove any loose surface contamination. Those surface areas known to contain high-level fixed contamination were chipped away and packaged for shipment to 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 (NRC 1982). Initial decommissioning was completed in May 1979. During 1979, drums of U_3O_8 were temporarily stored at the facility and were shipped out 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 MEDICAL X-RAYS

The information in this section applies to both the Apollo and Parks Township sites. From a cursory review of claimant medical records, the X-ray section of the physical forms seems not to be filled out for most of the claimant files. NUMEC apparently did not have its own medical X-ray department during AEC operational years. However, some claimant case files indicate some posterior-anterior (PA) chest X-rays were given. The X-ray was read off site by a radiologist, and the radiologist's report could be dated up to 5 months after the date of the physical examination. This could have been the reason why the X-ray documentation was not retained in the medical records. There is no evidence that lateral chest X-rays or lumbar spine X-rays were given to NUMEC employees routinely or for screening purposes as evidenced from the review of the claimant records. It was indicated that, at least as of 1963, annual physical examinations were given as well as pre-employment and termination exams (NUMEC 1963). Even though there is no current evidence that routine medical X-rays were performed at NUMEC, it is reasonable to assume that one X-ray should be assigned for preemployment, one at termination, and annually in the interim. A PA chest X-ray was given to employees who worked with beryllium twice a year as indicated in the NUMEC (Apollo) health and safety manual (NUMEC 1963).

To date no site-specific information is available for a NUMEC medical X-ray program. Therefore, medical X-ray doses should be assigned based on general guidance provided in the Oak Ridge Associated Universities (ORAU) Team technical information bulletin (OTIB) ORAUT-OTIB-0006, *Dose Reconstruction for Occupational Required Diagnostic X-Ray Procedures* (ORAUT 2005a). All X-rays should be assumed to be PA chest X-rays. Photofluorography was not likely because this method was suitable for screening large groups of people at one time. Because the medical X-rays for NUMEC employees appear to have been performed at a local clinic or hospital, the use of photofluorography was unlikely and should not be assumed to have occurred unless specifically indicated in the medical records. The organ dose from PA chest X-rays should be based on the values in Table 6-5 of ORAUT-OTIB-0006, which are reproduced here in Table 3-1. For organs not in the table, surrogate organs can be used as indicated in Table 3-2 of ORAUT-OTIB-0006.

Table 3-1. Annual organ doses due to the assumed annual medical PA chest X-ray[a].

Organ	Annual dose (rem) 1957–1969	Annual dose (rem) 1970–1985	Annual dose (rem) 1985–present
Adrenal gland	9.02E-02	4.51E-02	3.37E-02
Bladder	2.50E-02	1.00E-04	2.60E-04
Red bone marrow	1.84E-02 male 1.72E-02 female	9.20E-03 male 8.60E-03 female	8.90E-03 male 8.60E-03 female
Bone surface	9.02E-02	4.51E-02	3.37E-02
Breast	9.80E-03	4.90E-03	5.80E-03
Colon/rectum	2.50E-02	1.00E-04	2.60E-04
Esophagus	9.02E-02	4.51E-02	3.37E-02
Eye/brain	6.40E-03	3.20E-03	3.90E-03
Kidney	9.02E-02	4.51E-02	3.37E-02
Liver/gall bladder/spleen	9.02E-02	4.51E-02	3.37E-02
Lung	8.38E-02 male 9.02E-02 female	4.19E-02 male 4.51E-02 female	3.14E-02 male 3.37E-02 female
Ovaries	2.50E-02	1.00E-04	2.60E-04
Pancreas	9.02E-02	4.51E-02	3.37E-02
Remainder organs	9.02E-02	4.51E-02	3.37E-02
Stomach	9.02E-02	4.51E-02	3.37E-02
Testes	5.00E-03	1.00E-06	5.00E-07
Thymus	9.02E-02	4.51E-02	3.37E-02
Thyroid	3.48E-02	3.20E-03	3.90E-03
Uterus	2.50E-02	1.30E-04	2.60E-04

a.   Based on ORAUT (2005a).

The dose to the skin is a function of the location of the cancer site on the body as indicated in Table 3-2.

Table 3-2. Annual organ doses to the skin from the assumed annual medical PA chest X-ray.

Skin area	Annual dose (rem) 1957–1969	Annual dose (rem) 1970–1985	Annual dose (rem) 1985–present
Back and sides of chest to iliac crest, back of shoulders, back and sides of neck, upper arms, elbows, forearms, and palms and backs of hands	2.70E-01	N/A	N/A
Back and sides of chest from base of neck to lowest rib, back of shoulders	N/A	1.35E-01	7.00E-02
Face and front of neck	6.40E-03	3.20E-03	3.90E-03
Front of chest (to iliac crest), front of shoulders	5.90E-03	N/A	N/A
Front of chest from base of neck to lowest rib, front of shoulders	N/A	2.94E-03	2.39E-03
Back and sides of head, ears, buttocks	2.70E-02	N/A	N/A
Back and sides of head and neck, ears, buttocks, upper arms, elbows, forearms, and hands	N/A	1.35E-02	7.00E-03
Front lower abdomen from iliac crest to pubis	5.90E-04	N/A	N/A
Front lower abdomen from lowest rib to pubis	N/A	2.94E-04	2.39E-04
Thighs to knees	7.70E-05	3.88E-05	2.75E-05
Knees and below	2.80E-05	1.42E-05	1.00 E-05

The frequency of medical X-rays should be based on the frequency of routine physical examinations as provided in the employee's records. One PA chest X-ray should be assumed to have been provided along with each routine physical, even though, as stated above, the routine physical records will probably not indicate that the employee received an X-ray. If the frequency of routine physical exams is not provided in the employee records, the one PA chest X-ray should be assumed to have been given during a prehire physical, an annual physical, and at termination. If there is indication that the employee worked regularly with beryllium, then two PA chest X-rays should be assigned each year.

The PA chest X-ray values in Table 6-5 of ORAUT-OTIB-0006 (ORAUT 2005a) can be assigned as an overestimating approach by multiplying the values by an uncertainty factor of 1.3, represented as a constant distribution. For assignment of a best estimate, the values can be entered as the mean of a normal distribution with a standard deviation of 30%. All medical X-ray doses should be assigned as an acute exposure to photons with energies from 30 to 250 keV.

4.0   ENVIRONMENTAL OCCUPATIONAL DOSE

The Apollo site petition evaluation report for petitions SEC-00047 and SEC-00080 (NIOSH 2007c) determined that it is not feasible to reconstruct ambient environmental dose from 1957 through 1965 for the NUMEC Apollo facility based on limitations that are 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 2008) also did not address ambient environmental dose, and reliable information could not be found to bound the internal and external ambient dose for the Parks Township facility.

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 at the NUMEC Apollo and Parks Township sites.

5.0 OCCUPATIONAL INTERNAL DOSE

Occupational internal dose is the dose received by an individual from an intake of radioactive material while performing tasks within buildings and structures at the NUMEC 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 technical data to evaluate internal occupational doses that can reasonably be associated with worker radiation exposures covered by the EEOICPA legislation. The health and safety coverage for NUMEC Apollo and Parks Township were administered under one department. The information in 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 NUMEC Apollo site were uranium, with some potential for exposure to plutonium, or thorium dust produced 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 facility).

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 Park Township included depleted, natural, low (3.5%), and high (93%). Exposure to other radionuclides was also 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 various enrichments and chemical forms of the processed radionuclides for the Apollo and Parks Township facilities.

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
U	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
Th[c]	ThO_2 (M, S)	Th-228, Th-232	Natural thorium
Pu[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 yr
Tc or other TRU elements	Same as the Th, U, or Pu 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% Pu – fuel grade/5% Pu – 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 RU), Ir-192	Not applicable

a. Sources: Author unknown (2004) and 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 facility.
e. Mixed oxide (MOX) work was probably limited to the Parks Township facility.

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

facilities. The manufacture of uranium products occurred in most of the buildings at Apollo and Parks Township. See Tables 2-1 and 2-4 for further 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. Also, 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 2008a) 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 2010a). If the plutonium is type Super S, follow guidance in ORAUT (2008a) 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 grade. For this section, lacking any specific information on the actual composition of the processed plutonium, the composition of Hanford plutonium can be used as this was the source of plutonium for FFTF fuel fabrication (Author Unknown 2004). The activity composition for Hanford reference fuel-grade plutonium (12%) is given in Table 5-3 for fuel aged up to 20 years (ORAUT 2010a). The age of plutonium to assume for a given analysis depends on the radionuclide measured in the bioassay analysis. When plutonium, Pu-238, or Pu-239 is measured, the dose is maximized by assuming longer decay times (20 years). When Am-241 is measured and the intake is estimated using ingrowth of Am-241 from decay of Pu-241, 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 as this is midway between the possible low and high ages of

plutonium from Hanford. If the actual age of the fuel is known (such as from an incident investigation report), then that age can be used in the intake and dose analysis.

5.2   *IN VITRO* BIOASSAY

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

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-yr	5-yr	10-yr	15-yr	20-yr
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 (2010a).
b. Time since separation of the Am-241 from the plutonium mix.

Several bioassay vendors were used to evaluate *in vitro* bioassay samples for the NUMEC sites. The SEC evaluations for the NUMEC Apollo site (NIOSH 2007c) and the NUMEC Parks Township site (NIOSH 2008) 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 only be used 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 facilities 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 providing 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 prior to the dates listed in the table, the values for the earliest date should be used. Note that 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 the Apollo and Parks Township sites, it is difficult to determine from the reported bioassay 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.

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	Pu	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 (Pu + Am)	Quarterly/as needed	<10.0 dpm/sample	-

a. Based on review of worker dosimetry reports in BWXT (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 2010a).
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.

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 HNO_3. The residue was re-evaporated successively with nitric acid (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 any 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 are 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 those workers exceeding 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) the screening for potential chronic exposures (Caldwell 1966). The fecal analysis continued

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 (Pu + Am)	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 BWXT (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.

until about 1985 as indicated in worker dosimetry records. The fecal analysis results reported as dpm/sample should be considered equivalent to the daily excretion rate (dpm/d).

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 $^{239}PuO_2$, $^{241}AmO_2$, $^{234}UO_2$, 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).

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 then correlated with BZA samples. The suspected exposed worker was removed from radiation work and both 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 analysis as observed from worker dosimetry records.
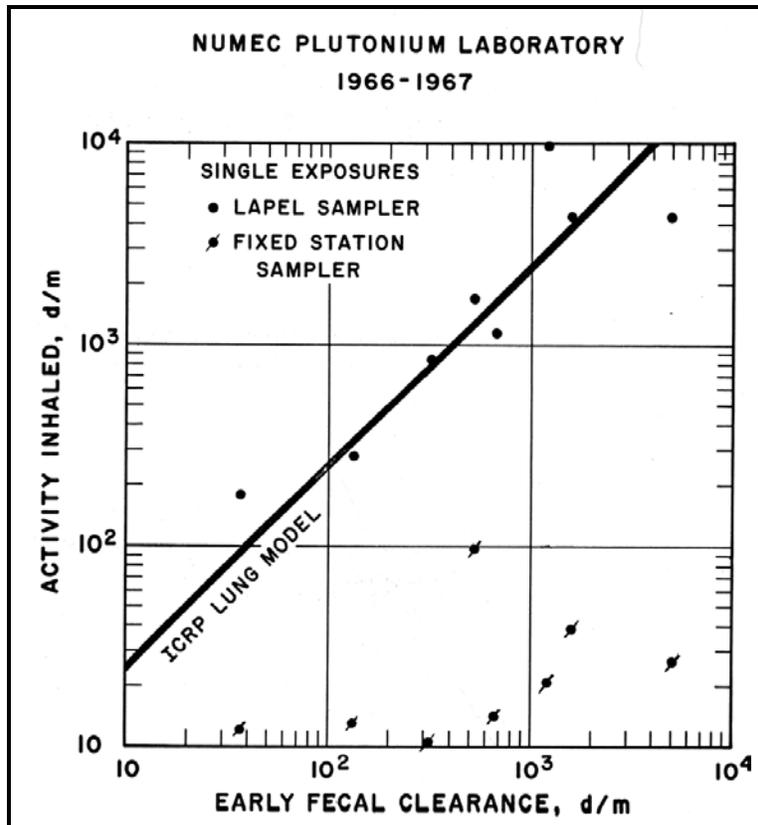
Figure 5-1. Correlation of fecal bioassay with air sampling
(Caldwell, Potter, and Schnell 1967).

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 (AEC 1960a). However, the available dosimetry
records do not contain urine bioassay results before late 1959.

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 hydrochloric acid (HCl) and nitric acid
(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 electro-deposition 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 (BWXT 2006a,b,c,d,e,f). If an MDC value is needed prior to the dates listed in the table, the values for the earliest date should be used. The measurements based on activity (gross alpha, gross alpha U, 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 U	Quarterly/as needed	<1 µg/L[d]	-
9/1972–12/1976	Eberline	Total U	Quarterly/as needed	<5 µg/sample	-
1/1977–11/1987	Controls for Environmental Pollution	Total U	Unknown	(e)	(e)
1999	Quanterra	Total U	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 U	Quarterly/as needed	0.2 dpm/sample	0.1 dpm/sample
9/1972–1/1974	Eberline	Gross alpha U	Quarterly/as needed	<50.0 dpm/sample	0.05 dpm/ml
2/1974–4/1974	Eberline	Gross alpha U	Quarterly/as needed	<10 dpm/sample	-
2/1974–12/1976	Eberline	Gross alpha U	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 BWXT (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 reported in the bioassay records; larger reported values may be used in the intake assessment.

Care should be taken in evaluating the Controls for Radiation reported urine bioassay results as the records could have errors in the reported units. The reported values are in units of µg/ml although sometimes are listed as µg/L.

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 exceeding 40 MPC-hr of exposure or nose wipes exceeding 25 dpm.

There were approximately up to 100 urine bioassay analyses conducted 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 past 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 of 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 during 1963 this was decreased to 300 dpm/L and by October 1964 this was further decreased 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 being 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 NUMEC facilities. 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 also based on the total sample activity evaluated as the product of the sample weight and the reported activity concentration.

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 U	Quarterly/as needed	<5 µg/sample
2/1976–10/1985	Controls for Environmental Pollution	Total U	Quarterly/as needed	(d)

a. Based on review of worker dosimetry reports in BWXT (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.

Fecal sampling (in addition to urine sampling) began on a large scale at the NUMEC 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 counting was effective for EU lung burdens greater than 7 nCi, but fecal sampling was necessary for smaller fractions of the permissible lung burden (Caldwell 1966).

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 facilities 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 facility 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 the 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 the bioassay analyses are

reported specific for ^{232}Th, consideration should be given to the ^{228}Th that would be present from decay of ^{232}Th. Based on information in ORAUT (2006d), 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 where 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 bioassay records indicate urine bioassay analysis was 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 being 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 facility 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 both 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 analytical methods used to analyze for mixed FPs in urine. Urine bioassay data for mixed FPs should be used, when 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 a favorable to claimant assessment of dose (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 the technical information bulletin.

5.2.5   Unmonitored Radionuclides from Recycled Uranium

The uranium processed at the Apollo and Parks Township sites may have included recycled uranium. 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 recycled uranium 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 made based on the guidance provided in Table 5-2 of ORAUT-OTIB-0060 (ORAUT 2007b). Because the plutonium contamination 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 performed 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 whole-body counts were performed for NUMEC workers starting in 1969.

In 1968 and 1971, Helgeson performed whole-body 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 in proximity to the anterior chest region of the subject (BWXT 2006g, pp. 92–97). The calibration was for 0.5 keV per channel and the count time was 40 min for both 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 representing 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 also 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 when 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 whole-body counts in the dosimetry records. Data after 1985 are sparse in the bioassay records.

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 separate HASL reports by the AEC for the Apollo site. The information in the HASL 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,

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

Year	Pu-239 MDA (nCi)			Am-241 MDA (nCi)		
	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 (BWXT 2006a,c,d,g,h,j). Values for 1968 through 1971 are based on the Helgeson system, with remaining values for the University of Pittsburgh system.
b.   NR = none reported.

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 about 1963 (AEC 1960a,b,c, 1961a,b; NUMEC ca. 1963). 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-82,- 92, -103, -106, and -114. A summary of the HASL reports is listed in Table 5-10 based on information from Attachment A.

The air samples consisted of collection of radioactive particulates 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),

Table 5-10. HASL report summary.

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

- 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 for the individual 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 tables in Attachment A. The analysis of the reported doses for all individuals from all HASL reports has been 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 a more favorable to claimant estimate of air concentration than a strict numerical evaluation of the data. The reported values have several very high values that 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 when available.

The results presented 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 NUMEC Apollo (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 a very favorable to claimant intake for individuals who did not work routinely in the Apollo uranium facility processes, such as health physics staff, 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 in a workweek. The individual dosimetry records should indicate if an individual exceeded the 40 MPC-hr limit. The dosimetry 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 when intakes might have occurred, and when intakes were unlikely. The 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-hrs exceeded 40 for a week, it was considered an overexposure. MPC-hrs 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, and it was assumed that no protection was afforded by the respirator and a bioassay was conducted. When a high nasal smear coincided with an impermissible air sample, it was assumed that an overexposure had occurred and NUMEC reported the overexposure according to 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 facility 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 regarding individual involvement in incidents is likely to be

contained 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. A total of five bottles were damaged, containing approximately 8.8 kilograms of HEU. NUMEC approximated about 0.5 kg of HEU might have been lost. No information is provided regarding 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 approximately 400 kg of dry 10% uranium by weight and 3.3% enriched in ^{235}U; therefore, approximately 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, thus 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 facility in Parks Township and might have contributed to employee exposures. Information about individual involvement in incidents is likely to be contained 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 out of 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 during the period from December 1959 to January 1961, and BZA sampling was observed in the 1963 health protection program review conducted by 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, and 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 Apollo showed concentrations seven times greater than the stationary air samples. The median of the ratio of lapel BZA to GA concentrations results was found to be ~7 at the NUMEC uranium and plutonium facilities (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 that an exposure had occurred, the worker was removed from radiation work and both 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 of 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 existing NIOSH dose reconstruction processes or procedures) for NUMEC-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 respective sites (Caldwell 1967a, 1968c).

6.1   EXTERNAL EXPOSURE SOURCES

The NUMEC Apollo Nuclear Fuel Facility started operations in 1957 with the small-scale production of HEU and LEU fuel. Between 1958 and 1983, the Apollo facility produced LEU dioxide fuel for use in commercial nuclear reactors. The process consisted of conversion of uranium hexafluoride (UF_6) to uranium dioxide (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 2007c). 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 NUMEC Apollo were discontinued in 1978, and LEU and all other processing operations involving radioactive materials had ceased 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.

The NUMEC Parks Township site operations began in about 1959; DOE operations ended in 1980. The initial function of the NUMEC Parks Township facilities was fabrication of plutonium fuel, the preparation of HEU fuel, and the production of zirconium-hafnium bars under AEC/NRC License No. 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. The Parks Township facility 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 allow 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 at NUMEC 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 also 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. NUMEC workplace potential exposures.

Source	Exposure potential
Pu fuel fabrication: 1959–1980	Gamma, X-ray, and neutron radiation primarily.
HEU production: 1957–1978	Beta radiation primarily, possibly photon dose from U progeny such as Ra, etc.
Source manufacturing	Gamma and neutron radiation depending on source
LEU production: 1957–1984	As above
Mixed Pu 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 U particles started in 1961	As above
Th operations and pellet production started in 1963	Beta radiation and more significant photon radiation
Laundry operations	U and Th 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 approximately 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 approximately 1 mrad/hr at 30 cm from a drum of aged yellowcake (NIOSH 2007d, 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 2005b). 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 whole-body beta/photon, whole-body 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.

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 such 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 2006k). 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 approximately 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 2006k). Details of the dosimeter are not available, other than detection limits.

Source: ORAUT (2006a).

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 dosimetry results were also found for the period July – December 1991 with dosimetry provided by Teledyne Isotopes (BWXT 2006k).

6.3.2   Neutron Dosimeters

Workers were monitored for neutron exposures with nuclear track emulsion, Type A (NTA) film from the respective commercial venders until about 1968, and then monitored using 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 which (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 cycles and workers shows recorded doses as low as 2 mrem for photon dose (BWXT 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. A review of historical film badge dosimetry by the National Research Council (National Research Council 1989) recommended use of 40 mrem for film dosimeters during the nuclear weapons testing period. Therefore, the recommended MDLs for estimation of missed dose are 40 mrem for gamma and beta radiation through 1968, and 50 mrem for neutron radiation for periods to 1975 (ORAUT 2006b, 2010b). 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 (WSRC 1995) that indicated the MDL was closer to 10 mrem for neutron exposures to fast neutrons from californium-252. The 20 mrem value is consistent with the Savannah River Site profile (ORAUT 2005c) value for this period where work with plutonium was similar to that at the NUMEC Plutonium Facility.   For purposes of 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 6-3. 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	0.04 photons 0.04 beta	0.24 beta -photons (monthly)
		0.05 neutrons	0.30 neutrons [fast](monthly)
Landauer film and NTA film *whole body*	1964–1965	0.04 photons 0.04 beta	0.24 beta photons (monthly)
		0.05 neutrons	0.30 neutrons [fast](monthly)
Eberline film and NTA film *whole body*	1966–6/1968	0.04 photons 0.04 beta	0.24 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[c] 0.02 beta[c]	0.12 beta photons (monthly)
		0.05 neutrons	0.30 neutrons (monthly)
Landauer TLD and Teledyne Isotopes *whole body*	1976–1995	0.01 photons[d] 0.01 beta[d]	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 of less than the MDLs.
b. Annual missed dose calculated based on the MDL/2 method from NIOSH (2007b).
c. MDL during this period is likely twice the recording level of 0.010 rem.
d. Landauer MDL values from BWXT (2006i) and Teledyne Isotopes MDL values from BWXT (2006k)

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 described in Table 6-2 for beta, gamma, and neutron radiation monitoring.

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 records 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 (BWXT 2006a, p. 6). However, even though claimant records show that not all personnel were assigned dosimeters at all times, the records do 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 (BWXT 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 (BWXT 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 facility (Breuer 1977, 1978).

Year	Total number monitored	No. with measured dose	Number of individuals with whole-body 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 accurately measure low-energy photon radiation such as that present 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 photon spectroscopy survey at the NUMEC 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 presented in Figure 6-1.

A survey of the ceramics line and plutonium-uranium Mo alloy melt box line indicates a predominance of 60-keV ^{241}Am gamma radiation. The gamma energy spectrum is presented 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 OW (open window) versus AL (aluminum), PL (plastic), and CD (cadmium) filters was made as noted in Figure 6-3.

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.
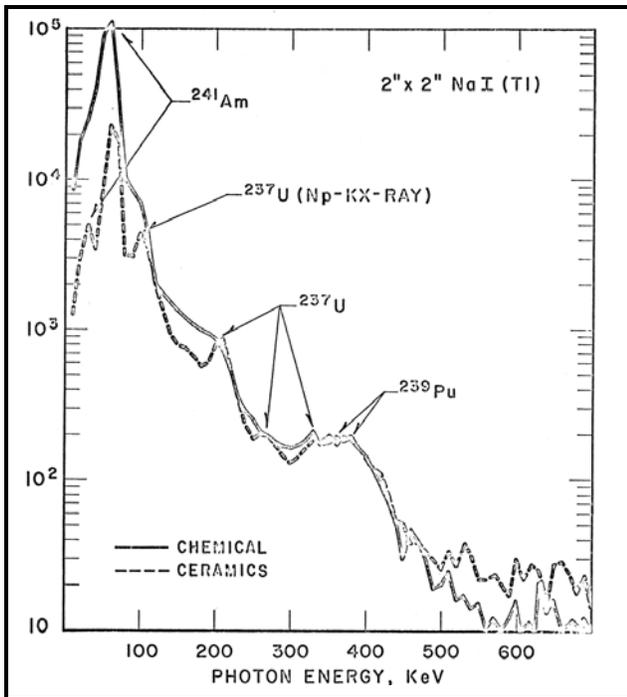
Figure 6-1. Gamma spectrum external to plutonium chemical processing line and plutonium ceramics line (Caldwell and Judd 1966, Figure 3).
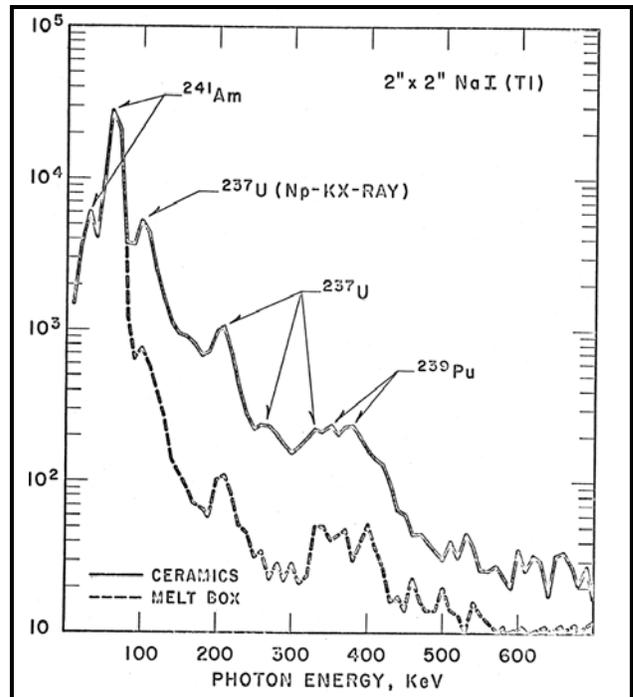
Figure 6-2. Gamma spectrum external to plutonium ceramics line and plutonium melt glovebox (Caldwell and Judd 1966, Figure 4).
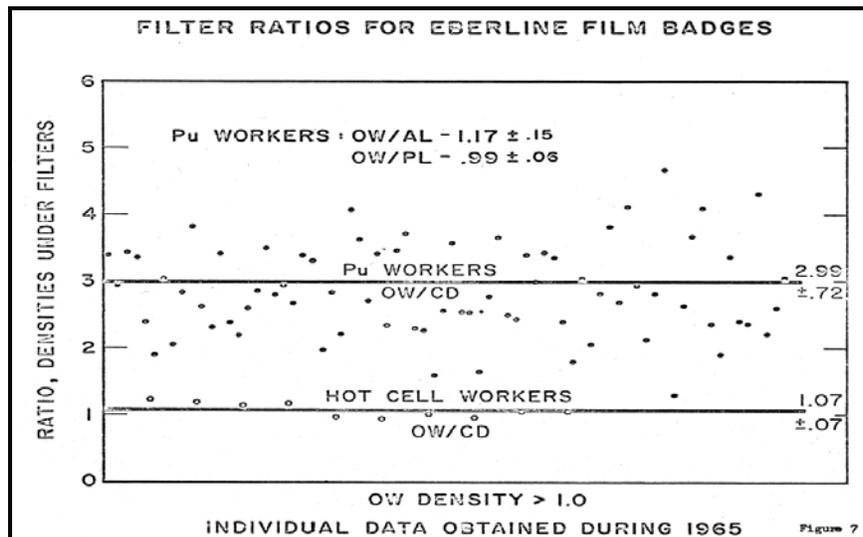
Figure 6-3. Dosimeter filter ratios (Caldwell and Judd 1966).

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 presented in Figure 6-4. It was noted that the gonadal dose was 50% of the whole-body 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.
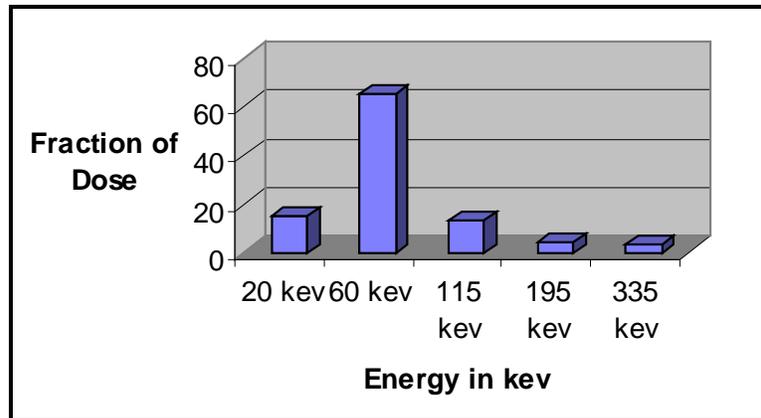
Figure 6-4. Relative contribution by energy group to personnel exposure during plutonium fuel fabrication from 1,000 MWd/t plutonium (Caldwell and Judd 1966, p. 18, Figure 8)

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

6.4.2   Neutron Radiation

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

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 & neutron WB = photon + neutron Extremity = gamma + neutron

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

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 2005d).

The guidance from ORAUT (2005d) 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, as 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, as 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 are considered to provide a realistic estimate of the actual doses.

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 (NCRP) 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 2006c). 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 provided 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 2005d) should also 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 exposed to neutrons, and for which 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 used by commercial and AEC laboratory facilities. The errors in the penetrating dose are anticipated to be approximately ±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 approximately 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 2000). 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 30-250 keV category to result in a higher calculated organ dose under most situations.

NUMEC Apollo workers had limited potential for some exposure to a variety of 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 occur at the Apollo laundry facility that provided commercial laundering of contaminated clothing. Generally recommended categories for IREP input for the measured and assigned components of radiation dose are presented in Table 6-8 unless there is claim-specific information regarding 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			
U facilities	1/1/1957	12/31/1983	Beta	>15 keV	100
	1/1/1957	12/31/1983	Photon	30–250 keV	100
Pu facilities	1/1/1959	12/31/1980	Photon	30–250 keV	100
	1/1/1959	12/31/1980	Neutron	0.1–2 MeV	100
Th 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 B&W Apollo facility ceased manufacturing nuclear fuel in 1983. Final decommissioning of the facilities was completed in 1995. During the period of residual contamination, employees of subsequent owners and operators of this facility are also covered under the EEOICPA. The residual period for the NUMEC Apollo plant covers the period from 1984 through 1995, and the residual period for the NUMEC Parks Township plant 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 considered to be 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

If dosimeter readings are available for the residual period, the dose should be based on the recorded and missed dose, as described in Section 6. The external dose from residual activity would be included in measured dose based on dosimeter readings for the workers. However, the recorded dose might also include dose from current activities that are not related to DOE work, and, therefore, represent an overestimate of external dose.

If external dosimetry information is not available for the residual period, the following method is provided to estimate a favorable to claimant external dose.

An estimate of the mean surface concentration at the end of the operational period is 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.

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

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 provided 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. The dose values are for exposure to NU as this provides a higher external dose than other enrichments (except DU). This provides a favorable to claimant dose estimate because most uranium

Table 7-1. External annual dose from residual
surface contamination.

Organ	Annual dose[a] rem
Adrenals	1.09E-03
Bladder wall	1.19E-03
Bone surface	2.99E-03
Brain	1.14E-03
Breast	1.56E-03
Esophagus	1.01E-03
Stomach wall	1.19E-03
Small intestine wall	1.12E-03
Upper large intestine wall	1.15E-03
Lower large intestine wall	1.16E-03
Kidneys	1.20E-03
Liver	1.19E-03
Lungs	1.26E-03
Muscle	1.48E-03
Ovaries	1.11E-03
Pancreas	1.06E-03
Red marrow	1.24E-03
Skin	3.78E-01
Spleen	1.20E-03
Testes	1.55E-03
Thymus	1.17E-03
Thyroid	1.31E-03
Uterus	1.11E-03

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

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

If dosimeter readings are available for the residual period, then the dose should be based on the recorded and missed dose (Section 6.0). The external dose from residual activity would be included in measured dose based on dosimeter readings for the workers. However, the recorded dose might also include dose from current activities that are not related to DOE work and, therefore, might represent an overestimate of external dose.

If external dosimetry information is not available for the residual period, an external dose can be estimated based on reported external dose distributions for monitored Parks Township employees obtained from the NRC REIRS database for 1981 through 1993 (Table 7-2) (Guido 2008). For each year, the annual dose at the 50th and 95th percentile is shown with correction for missed dose. Dose values for the 50th and 95th percentile were calculated by finding the numerical position of each percentile value within the histogram to find the correct bin, and then extrapolate to the point within the

bin to calculate the dose. The method chosen to determine the 95th percentile and 50th percentile is inherently favorable to claimants. It is likely the doses represented in each bin of the histogram would have been skewed towards the lower end of the bin (i.e. lognormally distributed). In the analysis it was assumed that the dose points were equally spaced between the lower and upper limits of the bin which should have resulted in a higher calculated dose. The contribution from missed dose was included by assuming monthly badge exchanges with 12 zero readings for the 50th percentile entries of zero dose and 11 zero readings for positive annual results. For each zero reading, a dose of half the limit of detection was added to the positive dose for the year (NIOSH 2007b). The badge exchange frequency and limit of detection are based on information provided in Table 6-3.

Assignment of external dose for unmonitored workers should follow guidance in the Technical Information Bulletin: Use of Coworker Dosimetry Data for External Dose Assignment (ORAUT 2008c). The 50th-percentile doses may be used as a best estimate of a worker's penetrating dose when professional judgment indicates the worker was likely exposed to intermittent low levels of external radiation. For routinely exposed workers who were expected to have been monitored, the 95th-percentile dose should be applied. The dose values should be treated as constant values. Adjustments to the dose values (e.g. dose conversion factors) should be made as described in Section 6.0. The 1993 values can be applied to unmonitored employees in that year and subsequent years.

The dose values in Table 7-2 include only penetrating dose as non-penetrating dose information was not available from the NRC REIRS database. Therefore, the dose to the skin from non-penetrating radiation cannot be evaluated using this data. However, it is likely any non-penetrating dose from beta radiation coming from residual activity on floors or walls would be small compared to the dose from penetrating radiation because of the distance between the residual activity and the worker.

Table 7-2. External dose distribution for monitored Parks Township employees.

Year	Number of workers with dose less than (mrem)										Annual Dose at listed percentile[a]	
	0.00	0.10	0.25	0.50	0.75	1.00	2.00	3.00	4.00	5.00	50th	95th
1981	0	12	7	5	4	0	8	1	1	0	0.305	2.060
1982	0	9	7	4	1	3	7	1	0	0	0.305	1.917
1983	1	3	2	0	0	0	0	0	0	0	0.122	0.310
1984	84	15	8	2	0	0	0	0	0	0	0.060	0.254
1985	91	107	20	16	1	8	5	0	0	0	0.086	0.841
1986	210	74	17	6	1	0	0	0	0	0	0.060	0.239
1988	117	99	33	13	7	4	8	4	1	0	0.081	0.998
1989	181	79	28	15	2	3	2	0	0	0	0.060	0.410
1990	183	89	22	9	1	0	1	0	0	0	0.060	0.283
1991	258	108	31	16	6	1	9	0	0	0	0.060	0.482
1992	237	143	25	17	1	4	1	0	0	0	0.060	0.339
1993	318	109	35	20	9	3	2	0	0	0	0.060	0.423

a. Listed value includes missed dose

7.2   EXTERNAL AMBIENT DOSE FROM RESIDUAL ACTIVITY

All unmonitored workers are assigned external dose as described in Section 7.1. If the worker is monitored (the assigned external dose based on monitoring records), or unmonitored and 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   EXTERNAL MEDICAL X-RAY 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

If bioassay data are available during the residual period, those data should be used to estimate and assign internal dose for the worker. Internal doses using bioassay data for this period should be evaluated using the guidance in Section 5.0. However, bioassay data are unlikely to be available after 1985.

If bioassay data are not available and the worker might have worked in areas containing residual activity, the following method can be used to assign internal dose.

Guidance in *Dose Reconstruction During Residual Radioactivity Periods at Atomic Weapons Employer Facilities* (ORAUT 2008b) 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 that workers were exposed to 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 the OTIB (ORAUT 2008b). 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.

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

This mean surface concentration is described as a favorable to claimant level at the end of the operating period. 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 also due to depletion (ORAUT 2008b). The depletion factors applied to each year are described in Table 3-1 of ORAUT (2008b). The depletion factors indicate the residual concentration at the end of the operational period is to be used for the first year, the second year is reduced by a factor of 0.03, and the third and remaining years are reduced by a factor of 0.0007 (representing a constant concentration after the third year).

The air concentration for each year is estimated using a resuspension factor (ORAUT 2008b) of 1×10^{-6}/m. Application of this resuspension factor and the above-described depletion factors to the residual contamination level of 1.605×10^6 dpm/m^2, results in the air concentration and annual intakes in Table 7-3. The intake evaluation is based on exposure for 2,000 hr/yr and an inhalation rate of 1.2 m^3/hr.

Table 7-3. Uranium air concentration and annual intake in the residual period.

Year	Air concentration (dpm/m^3)	Intake (dpm/yr)
1984	4.97	1.19E+04
1985	0.149	3.58E+02
1986–1995	0.00348	8.34E+00

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 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 NUMEC Apollo facility.

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 as 0.2 times the average daily air concentration expressed in units of dpm/m^3. Using the air concentration for the first year after the end of operations (1984) from Table 7-2, an intake rate of 0.99 dpm/d is obtained. This value should be applied for all years of the residual period, without depletion, because the transfer of material from the contaminated surfaces might not be reduced with time to the same extent that resuspension is reduced. This provides an assessment of ingestion intake that is favorable to claimants.

7.4.2   Parks Township Site

If bioassay data are available for the residual period, those data should be used to estimate and assign internal dose for the worker. Internal doses that are estimated from bioassay data for this period should be evaluated using the guidance in Section 5.0.

If bioassay data are not available and the worker might have worked in areas of residual activity, the following method can be used to assign internal dose.

Guidance in ORAUT-OTIB-0070, *Dose Reconstruction During Residual Radioactivity Periods at Atomic Weapons Employer Facilities* (ORAUT 2008b) 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 facility 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, 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.
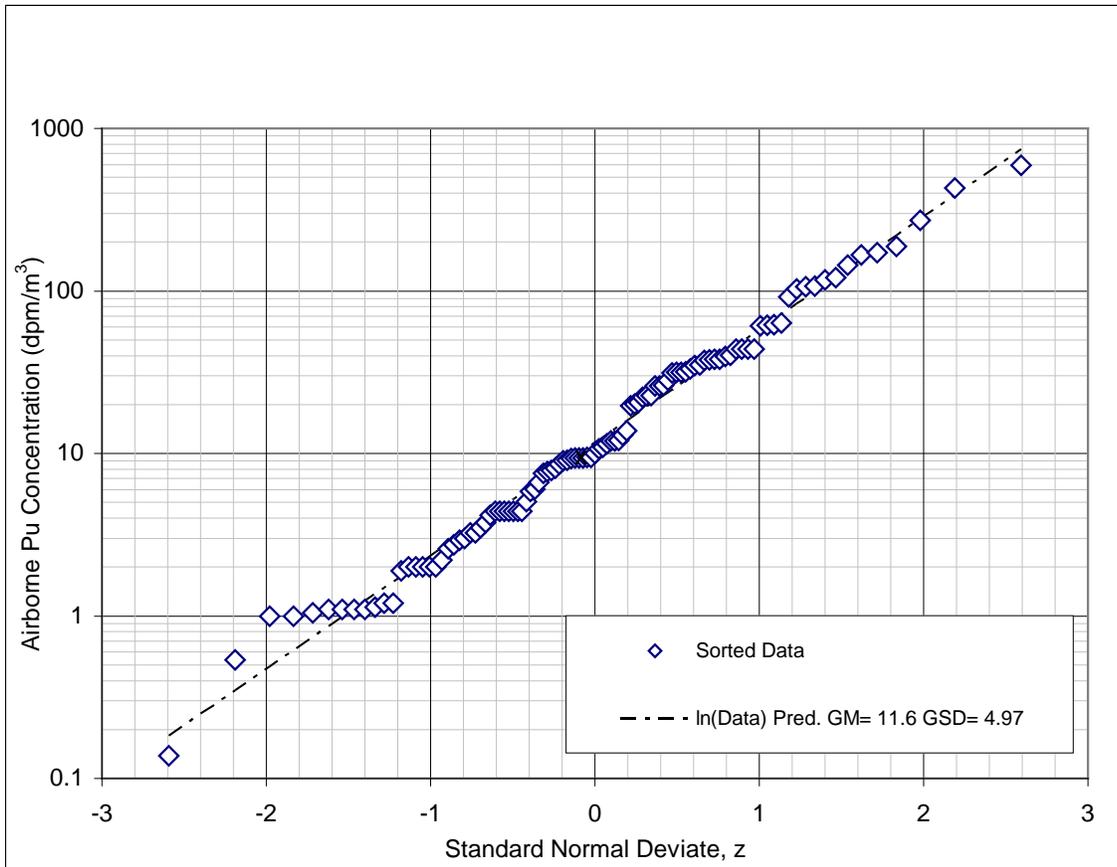
Figure 7-1. Reported plutonium air concentrations

The median plutonium air concentration from the above-described study was used to estimate the residual surface concentration along with the guidance from ORAUT (2008b). 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 also to depletion (ORAUT 2008b). The depletion factors that were applied to each year are described in Table 3-1 of ORAUT (2008b). The depletion factors indicated the residual concentration at the end of the operational period should be used for the first year; the second year should be reduced by a factor of 0.03, and the third and remaining years should be reduced by a factor of 0.0007 (which represents a constant concentration after the third year).

The air concentration for each year was estimated using a resuspension factor of 1×10^{-6}/m (ORAUT 2008b). 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 7-4. The intake evaluation was based on exposure for 2,000 hr/yr and an inhalation rate of 1.2 m^3/hr.

Table 7-4. Plutonium air concentration and
annual intake in the residual period.

Year	Air concentration (dpm/m^3)	Intake (dpm/yr)
1981	0.27	658
1982	0.008	20
1983–1995	0.0002	0.5

The intakes in Table 7-4 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 intake is represented as ^{239}Pu in the dose estimate. 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 (2008a).

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 as 0.2 times the average daily air concentration in units of dpm/m^3.

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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LIST OF TABLES

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).

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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).

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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).