ORAU TEAM
Dose Reconstruction
Project for NIOSH

Oak Ridge Associated Universities I Dade Moeller & Associates I MJW Corporation

Document Title:	Document Number:	ORAUT-TKBS-0041
Site Profile for Nuclear Materials and Equipment Corporation, Apollo, Pennsylvania	Revision:	00
	Effective Date:	09/26/2008
	Type of Document:	TBD
	Supersedes:	None

Subject Expert(s):	Paul J. Demopoulos and William Joyce
Site Expert(s):	N/A

Approval:	Signature on File Dennis L. Strenge, Document Owner	Approval Date:	08/25/2008
Concurrence:	Signature on File John M. Byrne, Task 3 Manager	Concurrence Date:	08/25/2008
Concurrence:	Signature on File Edward F. Maher, Task 5 Manager	Concurrence Date:	08/25/2008
Concurrence:	Signature on File Kate Kimpan, Project Director	Concurrence Date:	08/25/2008
Approval:	Signature on File James W. Neton, Associate Director for Science	Approval Date:	09/26/2008

☒ New ☐ Total Rewrite ☐ Revision ☐ Page Change

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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.

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ACRONYMS AND ABBREVIATIONS

ADU	ammonium diuranate
AEC	U.S. Atomic Energy Commission
AWE	Atomic Weapons Employer
B&W	Babcock & Wilcox (Company)
Bq	becquerel
BZA	breathing-zone air
CFR	Code of Federal Regulations
cm	centimeter
CP	Chemical Processing
CRP	Chemical Reprocessing
d	day
DCF	dose conversion factor
DOE	U.S. Department of Energy
dpm	disintegrations per minute
DU	depleted uranium
EEOICPA	Energy Employee Occupational Illness Compensation Program Act of 2000
EU	enriched uranium
eV	electron-volt
F	fast (absorption type)
FP	fission product
ft	foot
g	gram
GA	general air
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
I	insoluble
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

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
mo	month
MOX	mixed oxide
MPC	maximum permissible concentration
mR	milliroentgen
mrad	millirad
mrem	millirem
nCi	nanocurie
NDC	Nuclear Decontamination Corporation (Building; also Laundry Building)
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
NTA	nuclear track emulsion, type A
NUMEC	Nuclear Materials and Equipment Corporation
ORAU	Oak Ridge Associated Universities
ORISE	Oak Ridge Institute for Science and Education
PA	posterior-anterior
pCi	picocurie
POC	probability of causation
QF	quality factor
R&D	research and development
RU	recycled uranium
s	second
S	slow (absorption type)
SA	specific activity
SEC	Special Exposure Cohort
SLDA	Shallow Land Disposal Area
SNM	Special Nuclear Material
SRDB Ref ID	Site Research Database Reference Identification (number)
TLD	thermoluminescent dosimeter
TRU	transuranic
U.S.C.	United States Code
wk	week

wt %	weight %
yr	year
µCi	microcurie
µg	microgram

1.0 INTRODUCTION

1.1 PURPOSE

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

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

Under EEOICPA, employment at an AWE facility is categorized as either (1) during the DOE contract period (i.e., when the AWE was processing or producing material that emitted radiation and was used in the production of an atomic weapon), or (2) during the residual contamination period (i.e., periods that NIOSH has determined there is the potential for significant residual contamination after the period in which weapons-related production occurred). For contract period employment, all occupationally derived radiation exposures at covered facilities must be included in dose reconstructions. This includes radiation exposure related to the Naval Nuclear Propulsion Program and any radiation exposure received from the production of commercial radioactive products that were concurrently manufactured by the AWE facility during the covered period. NIOSH does not consider the following exposures to be occupationally derived (NIOSH 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 plant. This site profile for NUMEC presents information useful for reconstruction of doses NUMEC employees received.

1.3 SPECIAL EXPOSURE COHORT

A Special Exposure Cohort (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 workdays aggregating at least 250 workdays or in combination with workdays within the parameters established for one or more other classes of employees in the SEC. The SEC does not include work at the Parks Township site.

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:

- 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 sources 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 SEC 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 HHS 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 non-presumptive cancers may be considered for partial dose reconstruction (DHHS 2007).

The SEC covers employees of the NUMEC Apollo facility, which may include employees who worked at both the Apollo and Parks Township facilities. This site profile covers only the Apollo NUMEC facilities and may be used to perform partial dose reconstructions for individuals who worked at the Apollo facility.

2.0 SITE DESCRIPTION

2.1 APOLLO SITE 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 water 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 site 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 Kiskimenetas 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 an analytical 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 Kiskimenetas 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).

2.2 APOLLO FACILITIES

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

2.3 APOLLO 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

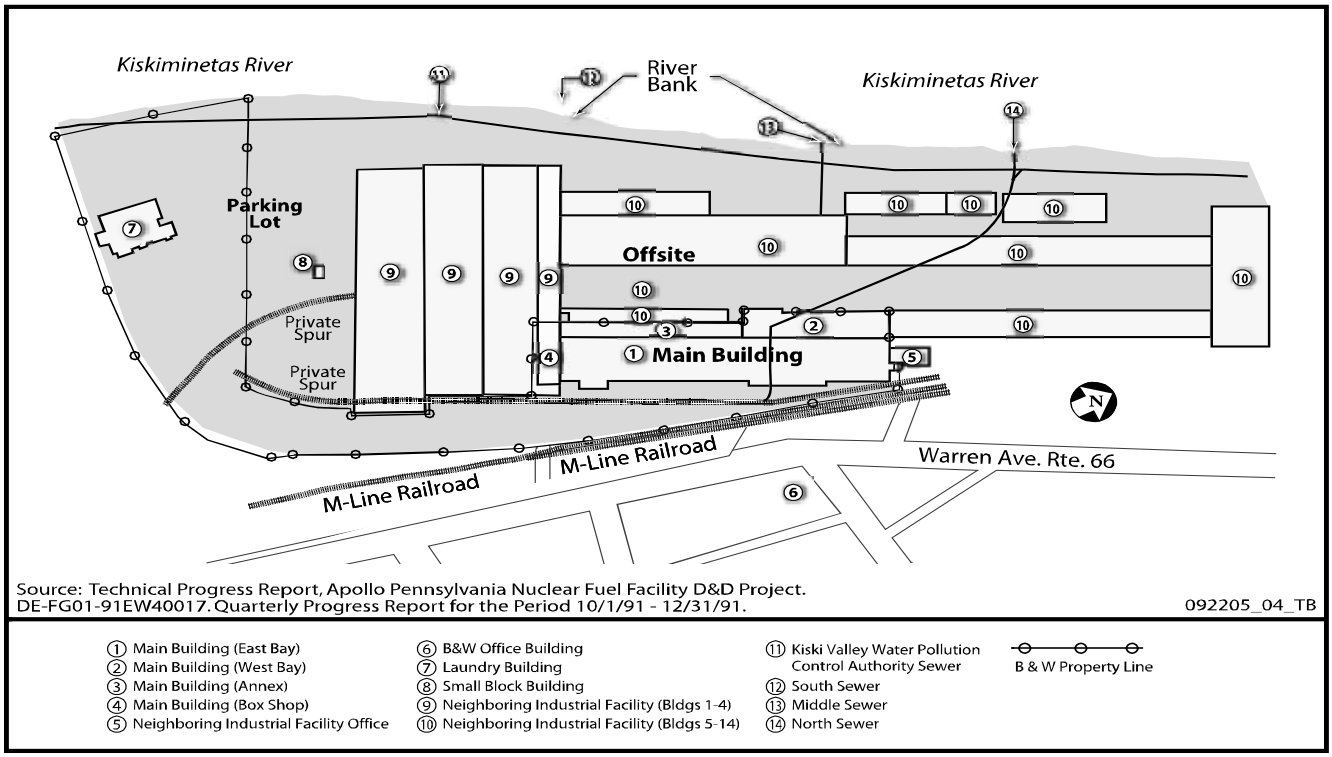

Figure 2-1. Site layout, NUMEC Apollo.

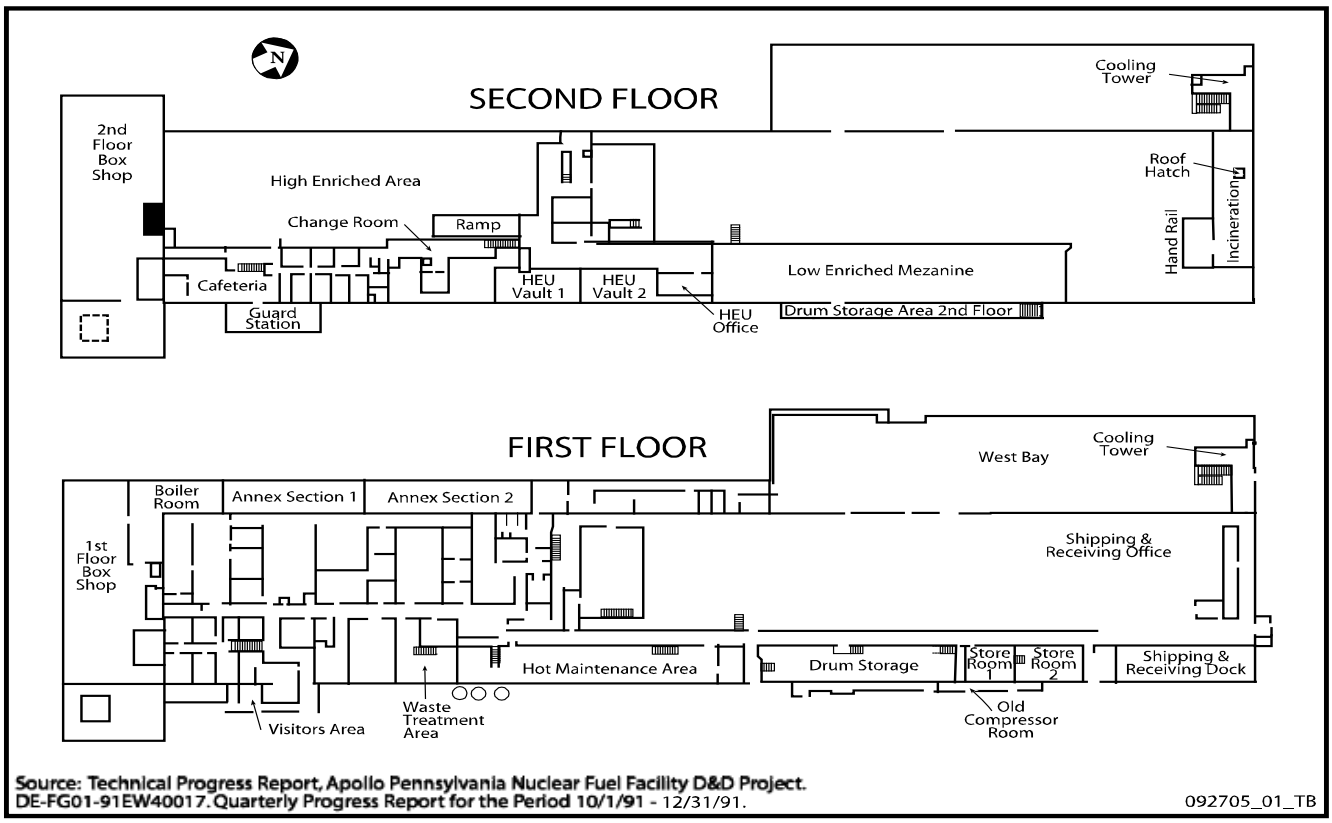

Figure 2-2. Site floor plan at NUMEC Apollo.

Table 2-1. NUMEC Apollo site area and descriptions.

Building or area location	Description	Operations/radionuclides	Period of operation
CF-1	Ceramic fabrication	UO_2, ThO_2, (metal, powder & 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 Handing Equipment, HEU, EU, DU, U nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
GF-1			1957–1983

Building or area location	Description	Operations/radionuclides	Period of operation
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 (1966a) and (1968).

birdcage. A review of the available literature shows that no criticality accidents have occurred during Apollo site operations.

2.3.1 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 (1957-1983) are to be included in radiation dose reconstruction as per NIOSH policy, since this work cannot be distinguished from AWE related work. Only some examples are listed below to illustrate the breadth of the SNM operations that occurred at the NUMEC Apollo site.

2.3.2 Various Uranium Process Activities

A general description of processes for various enrichments of uranium at Apollo is outlined in HASL Survey Reports 82, 92, 103, 106, and 114 (Occupational Exposure to Radioactive Dusts reports) which cover the period from December 1959 to January 1961 and the *Procedure for Recovery of Scrap Uranium* from 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 into 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 may have been a manual operation. No information could be found to indicate that the filter scraping was automated or that it was performed inside 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.

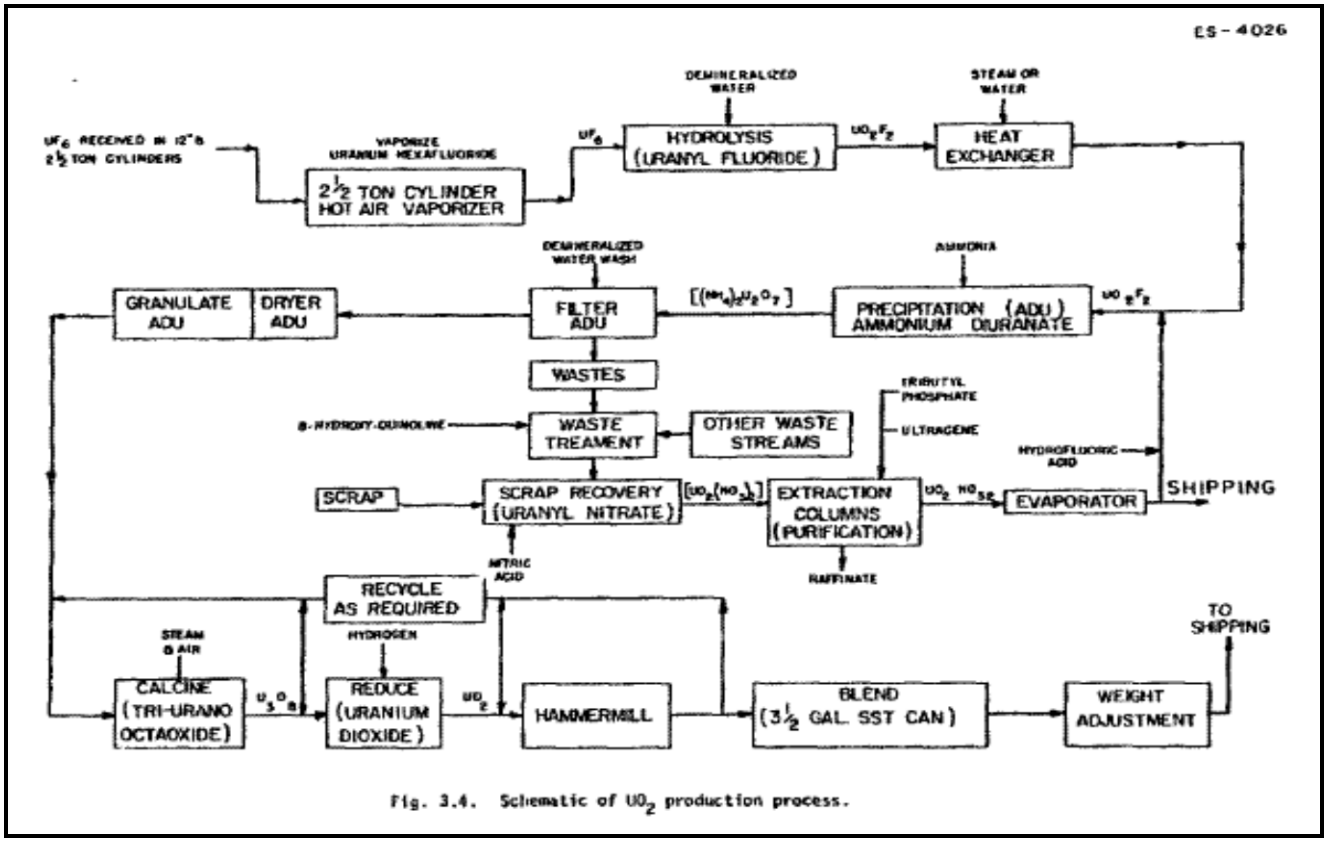

Figure 2-3. Schematic of UO_2 production process (Franke and Makhijani 1998).

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 1960-1961 HASL survey reports, approximately 0.25-in. layer of UO_2 powder was placed into 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-gallon 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 remains in solution and part of the uranium precipitates 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 the Procedure for Recovery of Scrap Uranium, undated, NUMEC Apollo, Pennsylvania (submitted to AEC around 1962 or 1963) (NUMEC ca. 1963), incoming scrap was received in criticality safe "bird cages" and stored at Advanced Materials Center Parks Township near Leechburg, Pennsylvania, 5 miles north of 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 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-gallon 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. $BeO\text{-}UO_2$ scrap was mixed with $HF\text{-}HNO_3$. 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 UF4, and ultimately converted to the final product of U_3O_8 or UO_2.

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

A March 25, 1960, letter (Katine 1960) recommended approval of NUMEC Feasibility Report to fabricate 3000 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 to 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).

2.3.4 Uranium Nitrate Solution for the University of California

A letter report dated June 9, 1961, written by J. E. McLaughlin, Director, Radiation Physics Division, HASL, describes a trip report to the NUMEC Apollo Facility on 6-7-61 to observe equipment for producing uranyl nitrate solution for the University of California as described in Feasibility Report No. 27, dated 6-9-61. 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).

2.3.5 Apollo 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 30-35 lb/hr. Ashes were collected in stainless steel, 1-gallon 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 gallon 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 gloved box at negative pressure (Caldwell 1968a). (As of 1967 this was likely not the case and work in this area was perhaps the highest area for intakes. See section 3.1.2.)

Combustible gases passed through the after-burner to a water-operated venturi-type fume scrubber. This separated the fly ash from the gas stream. Downstream of the venturi, 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 make-up 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).

2.3.6 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 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. Slug the powder to 4-5 g/cc then granulate through 14-mesh screen.

4. Blend each batch of powder in a "V" type blender located in a ventilated glovebox.

5. Press the ThO_2 pellets using a hand press and/or automatic press within a hood with face velocity of 100 fpm.

6. Sinter the ThO_2 pellets in hydrogen atmosphere with the out gases of the furnace passing through a filtered exhaust ventilation system.

7. All pellets will be center-less ground in ventilated hood.

8. The final product was to 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 health protection program review conducted in 1964, thorium operations involving the blender and weighing hood were resulting in excessive airborne concentrations (AEC 1964).

2.3.7 Research Activities at Apollo NUMEC, 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.

2.3.8 Source Term

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 AEC in 1957 (Docket No 70-135). Some possession limits at different periods of time are listed for the Apollo site in Table 2-2.

The Apollo site radiological source term included uranium, thorium, plutonium, and fission and activation products (Reitler 1972). *Currently 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 work 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, research studies in the gloveboxes and laboratories (AEC 1960a –c and AEC 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

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
Low-level radioactive waste 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 by-product material	20 mCi
	Any source material	20 g
	Any SNM	20 mCi

Source: SNM-145 Renewal Application October 31, 1972, Reitler (1972).

was also conducted. Uranium isotopic forms included DU, normal (i.e., natural enrichment), HEU (up to 93%), as well as ^{232}U, ^{233}U and ^{236}U. Uranium from recycle operations would have included smaller activities of nonuranium isotopes, such as ^{99}Tc, ^{237}Np, ^{230}Th, and ^{239}Pu.

Thorium. Thorium dioxide use was similar to uranium. 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).

2.4 REMEDIATION, DECONTAMINATION AND DECOMMISSIONING OF NUMEC APOLLO

The HEU processing area located 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 (B&WNES 1997).

The LEU processing area located in the Apollo East Bay was remediated between 1983 and 1984. During this period, the LEU processing equipment was removed and disposed. By October of 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 that 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 "Apollo Final Technical Report," 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.

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

Table 2-3. Decommissioning criteria for NUMEC Apollo.[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).

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 may 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 (Martin 1982).

Decontamination efforts were completed from 1984 to June 1992 for the Apollo site. In June 1992, the NRC approved the Apollo decommissioning plan. Decommissioning occurred from June 1992 to 1995. NRC staff reviewed B&W'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).

2.4.1 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 Parks) with about 700,000 ft^3 of waste buried in trenches on the site (PDEP 2008). Currently options for the disposition of the SLDA are in discussion with the NRC and the former licensee. The site may become a Formerly Utilized Sites Remedial Action Program site. As of 2007 a feasibility study is being conducted by the Army Corps of Engineers (USACE 2007). A graphic of the area is provided in Figure 2-4.

2.4.2 Apollo 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 administration 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

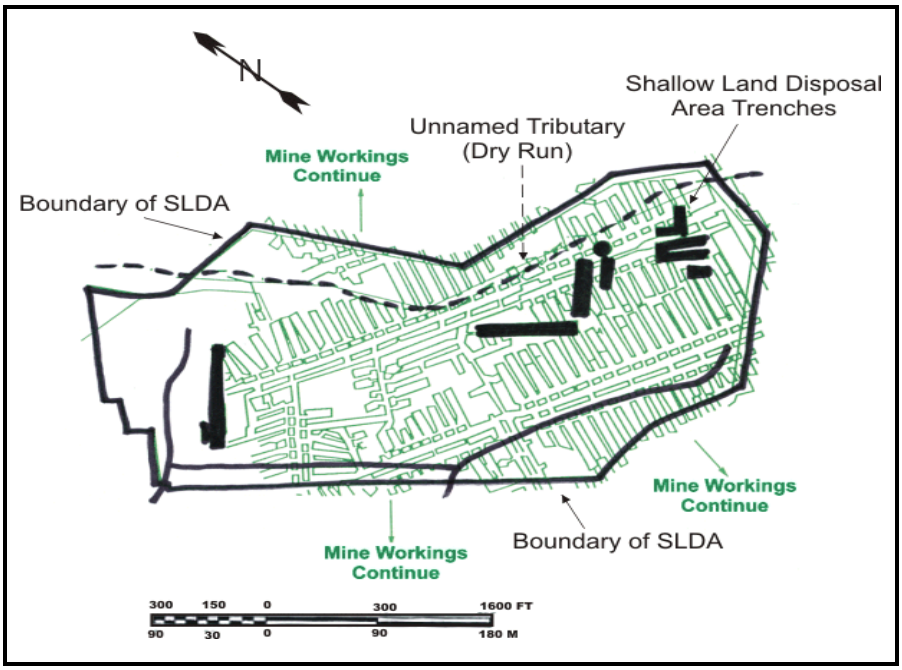

Figure 2-4. B&W Shallow Land Disposal Area (SLDA) (USAEC 2007).

southern end, the location of the instrument repair and calibration laboratory (NRC 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 that the acceptable level was approximately 15 µR/hr (NRC 1995). The reported values essentially represent background levels.

Soil samples were analyzed by the Oak Ridge Institute for Science and Education (ORISE). Their 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.

3.0 MEDICAL X-RAYS

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 at least as of 1963 that 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 (NUMEC1963).

To date no site-specific information is available for a NUMEC Apollo medical X-ray program. Therefore, medical X-ray doses should be assigned based on general guidance provided in the Technical Information Bulletin ORAUT-OTIB-0006 (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 provided in Table 6-5 of ORAUT-OTIB-0006 (ORAUT 2005a), which are reproduced here in Table 3-1. For organs not in the table, surrogate organs may be used as indicated in Table 3-2 of ORAUT-OTIB-0006 (ORAUT 2005a).

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

Organ	Annual dose (rem) 1957–1969	Annual dose (rem) 1970–1985	Annual dose (rem) 1985–present
Bladder	2.50E-02	1.00E-04	2.60E-04
Red bone marrow	1.84E-02 male 1.72E-02 female	9.20E-03	8.90E-03
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
Ovaries	2.50E-02	1.00E-04	2.60E-04
Testes	5.00E-03	1.00E-06	5.00E-07
Liver/gall bladder/spleen	9.02E-02	4.51E-02	3.37E-02
Lung	8.38E-02 male 9.02E-02 female	4.51E-02	3.37E-02
Remainder organs	9.02E-02	4.51E-02	3.37E-02
Skin	2.70E-01	1.35E-01	7.00E-02
Stomach	9.02E-02	4.51E-02	3.37E-02
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 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 likely 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 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% (ORAUT 2005a). All medical X-ray doses should be assigned as an acute exposure to photons with energy from 30 to 250 keV.

4.0 ENVIRONMENTAL OCCUPATIONAL DOSE

The SEC 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 based on limitations associated with stack monitoring data. Reliable information for the period after 1965 could not be found to bound the internal and external ambient dose, as described below.

4.1 ENVIRONMENTAL INTERNAL DOSE

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

4.2 ENVIRONMENTAL EXTERNAL DOSE

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

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 site 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 AT APOLLO

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 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).

Table 5-1 lists the various enrichments and chemical forms of the processed radionuclides for the Apollo facility.

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

Radionuclide or fuel	Chemical form and solubility type(s)[b]	Isotope (% in mass, where listed)	Enrichment
Uranium	UF_6, UO_2F_2, & $UO_2(NO_3)_2$ (F) UO_3 & UF_4 (M) U_3O_8 & UO_2, (S)	U-234 U-235 U-238	DU, Natural, LEU (3.5%)
Thorium[c]	ThO_2 (M, S)	Th-228, Th-232	Natural
Plutonium[d]	PuO_2 (M, S, SS)	Pu-238 0.64%, Pu-239 2.06%, Pu-240 1.07%, Pu-241 95.4%, Am-241 0.86%--Activity	Fuel grade Aged 10 yr

Radionuclide or fuel	Chemical form and solubility type(s)[b]	Isotope (% in mass, where listed)	Enrichment
Technetium or other TRU elements	Same as the Th, U or Pu matrix	Tc-99, Np-237	N/A
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	Co-60, Sr-90, Ru/Rh-106, Cs-137, Tc-99 (from recycled U)	N/A

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 likely 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 form and the enrichment varied over time at the Apollo facilities. The manufacture of uranium products occurred in most of the buildings in Apollo. See Table 2-1 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
Natural uranium	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)**			
Natural uranium	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).
c. ORAUT (2007a).

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, then either type M or S can be used to maximize the dose to the organ of concern. Also, because highly insoluble forms of plutonium may have been present, guidance in *Technical Information Bulletin: Estimating Doses for Plutonium Strongly Retained in the Lung* (ORAUT 2007b) 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 2007a). If the plutonium is type Super S, follow guidance in ORAUT (2007b) 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, an assumption of 10-yr-old fuel-grade plutonium is favorable to the claimant and reasonable (Table 5-3). This is noted in the Hanford site profile; Hanford processed much of the DOE complex plutonium (ORAUT 2007a).

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

Mixture designation	SA in (Ci/g)	Alpha activity fraction
Years of aging[b]	**10-yr**	
Pu-238	1.58E-02	1.17E-1
Pu-239	5.26E-02	3.91E-01
Pu-240	2.72E-02	2.02E01
Pu-241	1.91E+00	0.0
Pu-242	3.93E-06	2.92E-05
Am-241	3.89E-02	2.89E-01
Total alpha	1.35E-01	1.00

a. Source: ORAUT (2007a).
b. Time since separation of the Am-241 from the plutonium mix.

5.2 *IN VITRO* BIOASSAY

The bioassay program for NUMEC workers primarily involved urine and fecal sampling for isotopes of uranium, plutonium, and americium-241. Occasional analyses were performed for 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.

Several bioassay vendors were used to evaluate *in vitro* bioassay samples for the NUMEC sites. The SEC evaluation for the NUMEC Apollo site (NIOSH 2007c) 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 can not 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, except as noted below for urine bioassay data from Controls for Radiation in the early years (1961 – 1965).

5.2.1 Plutonium Urine and Feces Bioassay

Plutonium may 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 MDCs and Frequencies

Plutonium-239 was analyzed in urine from about 1962 to 1999, and ^{241}Am was analyzed starting in about 1970. The 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. Also, 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 sites. It is likely that much of the plutonium bioassay results were for work at the Parks Township site.

The accuracy of the early plutonium measurements was questioned by a manager who observed that some workers not associated with the plutonium fabrication work had higher plutonium urine results than the plutonium workers (Puechl 1963). The information indicated that the minimum detectable concentration (MDC) might have been closer to 5 dpm/L than to the reported values in measurement results. Therefore, for the first period in Table 5-4 the MDC has been assigned as 5 dpm/L. This value should be used as a minimum MDC over the listed values in the bioassay records.

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	5 dpm/L	0.02–0.12 dpm/L
9/1968–12/1975	Eberline	Pu-239	Quarterly/as needed	0.06 dpm/sample	0.03 dpm/sample
1/1970–12/75	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 5 dpm/L value is based on the uncertainty issue mentioned in the text.
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 is 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 (Barry 1964), the early analytical procedure likely 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 four 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 (Barry 1964).

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 MDCs 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 1966a). The fecal analysis continued 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/day).

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 MDA is twice the sensitivity or error.
d. Bioassay data analyzed by Controls for Environmental Pollution is not to be used in internal dose assessments.

NUMEC health physicist Roger Caldwell believed that fecal sampling was the only satisfactory method for estimating lung burdens for insoluble actinide alpha emitters classified as Y in the contemporary lung model. The most important alpha emitters included $^{239}PuO_2$, $^{241}AmO_2$, $^{234}UO_2$, and $^{232}ThO_2$ (Caldwell 1966a). 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 1966a).

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 1966a; 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 (HPS 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 since 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).

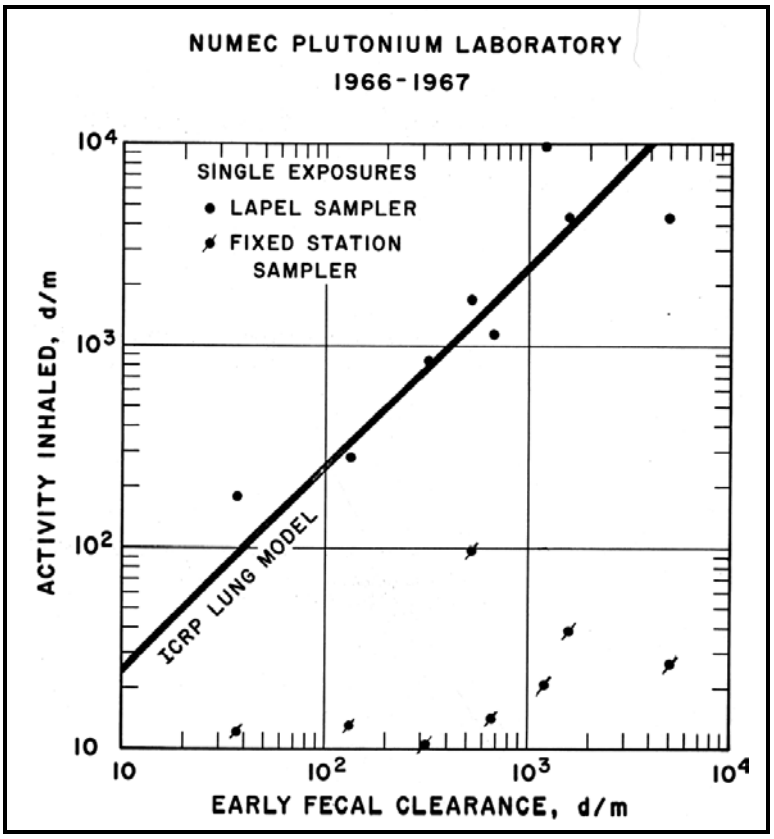

Figure 5-1. Correlation of fecal bioassay with air sampling (Caldwell, Potter, and Schnell 1967.)

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 in order 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 1966a). A number of different bioassay vendors performed plutonium urine and fecal analysis as observed from worker dosimetry records.

5.2.2 Uranium Urine and Feces Bioassay

Uranium was processed at the Apollo site. Enrichment levels varied with time and included depleted uranium (DU), natural uranium, LEU (3.5%), and HEU (93%).

5.2.2.1 Uranium Urine Analytical Procedure

Information in HASL report 82 (AEC 1960a) indicates that before 1960 urine samples were obtained 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 (Barry 1964), the early analytical procedure likely 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 (Barry 1964). 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 the AEC laboratory design at a temperature of 95^0 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 (Barry 1964).

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 that no absorption correction was necessary (Barry 1964).

The fluorimetric analysis would require a 5 ml sample volume and would have a sensitivity of 1 µg/L with a precision of ±10% (Barry 1964). 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.

The MDC value for the Controls for Radiation is set to 1 ug/L even though the vendor reports often indicate a value of 0.1 ug/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 Barry (1964). 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. This is consistent with the NUMEC health physics staff interpretation of the plutonium urine analysis results from Controls for Radiation, where they demonstrated that the accuracy was about 50 times less than the vendor indicated (5 dpm/day versus 0.1 dpm/day).

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

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

Date	Laboratory	Radionuclide	Frequency[b]	MDC[c]	Error
3/1961 – 2/1966	Controls for Radiation	Total uranium	Quarterly/as needed	<1 µg/L[d]	-
9/1972 – 12/1976	Eberline	Total uranium	Quarterly/as needed	<5 µg/sample	-
1/1977 – 11/1987	Controls for Environmental Pollution	Total uranium	Unknown	e	e
1999	Quanterra	Total uranium	Quarterly/as needed	<0.006 µg/L	-
4/1962 – 1/1967	Controls for Radiation	Gross alpha	Quarterly/as needed	26 dpm/L	13 dpm/L
2/1967-8/1972	Tracerlab	Gross alpha 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. Assumes the MDC is twice the error, when an MDC is not available in the records.
d. The MDC for Controls for Radiation for total uranium (1961 – 1966) is based on the reported value by Barry (1964) 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 is not to be used in internal dose assessments.

Urine samples were typically 24-hr 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).
- 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 (AEC 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 MAC (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 of 1964 this was further decreased to 150 dpm/L (AEC 1964). The natural uranium urine control limit was 50 µg/L weight basis or 75 dpm/L activity basis (AEC 1963).

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

5.2.2.3 Uranium Fecal MDCs 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/day). When results are provided as dpm/gram 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[d]
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 is 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 1966a). The fecal analysis continued until about 1985 as indicated in worker dosimetry records. Caldwell observed that some UO_2 exposures were poorly detected in urine (Caldwell 1966a). According to Caldwell, literature available at the time indicated that whole-body counting was effective for enriched uranium lung burdens greater than 7 nCi, but fecal sampling was necessary for smaller fractions of the permissible lung burden (Caldwell 1966a).

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 in order to conduct a thorium intake analysis for workers in general. If the case files include thorium measurement results, then an intake and dose assessment can be performed. Thorium was processed at the Apollo facility for a few years starting in 1963. Limited information on thorium bioassay analyses was found in the worker dosimetry records. In 1969 the error was reported as 0.1 dpm/sample for ^{232}Th (Tracerlab analysis), 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.

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 was most likely to have occurred in the Laundry facility as part of the Apollo site commercial decontamination of clothing by laundering. 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 (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. The technical information bulletin *Fission & Activation Product Assignment for Internal Dose-Related Gross Beta & Gamma Analyses* (ORAUT 2007c) 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 fission product 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 2007c).

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 site 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, ^{228}Th, ^{237}Np, ^{238}Pu, and ^{239}Pu. The intake of recycled uranium contaminant radionuclides can be estimated using the contaminant fraction values presented in Table 5-12 of the Fernald Technical Basis Document (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 2007e). 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 1966a, Caldwell 1968b). The counting performed in 1966 was provided by the University of Pittsburg Low Level Radioactivity Monitoring Facility at the Presbyterian-University Hospital, using a thin NaI crystal system (Caldwell 1966b). 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 minimum detectable amount for ^{235}U was listed as 0.08 mg for this system. The minimum detectable amount for ^{241}Am ranged from 0.13 to 0.38 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 Pittsburg 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 close proximity to the anterior chest region of the subject (BWTX 2006i, 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 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 to1992, possibly later. Uranium lung counting started regularly in December of 1971. Plutonium and americium counting started in 1966 (Caldwell 1966a) 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 1966a) 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 minimum detectable activities (MDAs) for common enrichments that may have been processed at NUMEC. Actual MDAs from workers' 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 in µg	Total uranium MDA in pCi
Natural uranium	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, ^{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.

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,i,j). Values for 1968 through 1971 are based on the Helgeson system, with remaining values for the University of Pittsburg system.
b. NR = none reported.

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

5.4 APOLLO PROCESS URANIUM AIR SAMPLING STUDIES

This discussion focuses on documented air-sampling data from five 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, 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 reports 82, 92, 103, 106, and 114. A summary of the HASL reports is listed in Table 5-10 based on information from Attachment A.

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.

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. Natural U 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, Uranium graphite pellet production, recovery of U_3O_8 HEU from U-Zr Scrap, R&D for coating Uranium particles. Natural Uranium 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 Uranium particles. Natural U 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. Uranium particle coating involved natural Uranium. No powder handling activities were evaluated.	5/1961

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

- Bioassay monitoring results
- Workplace breathing zone sampling results

- General area monitoring results
- Work location and job classification by time period
- Air concentration information from HASL reports (summarized in this section and Attachment A)
- Reports to the AEC/NRC of overexposures to airborne activity
- 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 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, use of 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 may have involved much higher air concentrations, but for a short period of time. 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 (1957-1983), 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 location 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-hr workday.

5.4.1 Apollo 40 MPC-hr Reports to AEC of Overexposures

NUMEC Health Physics reported to the AEC anytime 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 may 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, then 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, then 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, then NUMEC only reported if the bioassay data indicated an overexposure (Shapiro 1969).

5.4.2 Other Incidents

The following incidents occurred at the NUMEC facility in Apollo, Pennsylvania and may have contributed to employee exposures, but no exposure information was provided in the incident report or was reported as permissible. These incidents are in addition to those incidents/overexposures shown in Table 5-14. 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 may 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. deep of material. The material was found to be approximately 400 kilograms of dry 10% uranium by weight and 3.3-percent enriched in ^{235}U; therefore, approximately 1.32 kilograms 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 1973).

On April 20, 1974, a maximum of 6 kilograms of low-enriched UF_6 was released to the in-plant atmosphere. A pipe and valve on the suction side of a hydrolysis column re-circulating 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 APOLLO URANIUM AND PLUTONIUM AIR SAMPLING PROGRAMS

NUMEC uranium and plutonium workers wore lapel samplers starting in 1965. According to Caldwell, 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 report 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 Operations Office R&D Division. NUMEC used a Rochester Imaging Detector Laboratory gas flow proportional counter.

Before 1965, the BZA samples were likely fixed-station BZA samplers, and later NUMEC studies conducted in the 1966 -1967 time frame 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 which would indicate an average air concentration over an 8- to 24-hour period (AEC 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 (1966-1967), 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

6.1 EXTERNAL EXPOSURE SOURCES AT APOLLO

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 US

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.

6.2 RADIOLOGICAL EXPOSURE SOURCES

During NUMEC operations, occupational exposure was received primarily from handling or being in the vicinity of fissile material (either as feed or product) and from airborne radioactivity with the resultant buildup of surface contamination. Enriched UF_6 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. Little information has been obtained regarding potential exposures from handling thorium. Available information indicates PuBe neutron source production was performed at the Parks Township Site. Since there is considerable uncertainty as to whether neutron sources were manufactured at the Apollo facility and direct evidence (data) has not been found, the radiological exposures from neutron source production is not considered for the Apollo site. The primary sources of radiological external radiation exposure from operations performed at NUMEC Apollo are summarized as follows:

Source	Exposure potential
HEU production: 1957–1978	Beta radiation primarily, possibly photon dose from U progeny such as radium, etc.
LEU production: 1957–1984	As above
HEU and LEU scrap recovery	As above
UO_2 pellet production, started in 1961	As above
R&D for coating U particles: start in 1961	As above
Thorium Operations, pellet production, started in 1963	Beta radiation and more significant photon radiation
Laundry operations	Uranium and thorium residues

6.2.1 Beta Radiation

The beta dose rate on the surface of yellowcake, a natural uranium compound, just after separation is negligible, but rises steadily thereafter due to the build-up 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.

6.2.2 Photon Radiation

Photon exposure rates are about 1.2 mrad/hr in contact with fresh yellowcake, but during the build-up of the uranium daughters ^{234}Th and ^{234}Pa in fresh yellowcake, the radiation levels increase somewhat for several months after yellowcake production. Photon exposure rates are estimated to be approximately 1.0 mrad/hr at 30 cm from a drum of aged yellowcake (NIOSH 2007d, Table 4). Little site-specific information has been obtained about thorium, which does emit significant photon radiation.

6.2.3 Neutron Exposures

Uranium compounds can be a source of neutrons from both spontaneous fission occurring in the isotopes of uranium and from alpha-neutron reactions with low atomic number materials, such as oxides and impurities. Neutron exposures from yellowcake, a natural uranium compound, are

considerably lower than the photon exposures and are, therefore, not generally considered significant based on analyses presented in ORAU-OTIB-0024 titled "*Technical Information Bulletin: Estimation of Neutron Dose Rates from Alpha-Neutron Reactions in Uranium and Thorium Compounds"* (ORAUT 2005b). This 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 feet is about 0.003 mrem/hr for natural uranium, 0.016 mrem/hr for 5% enriched uranium, and 0.45 mrem/hr for 97+% enriched uranium. Little site-specific information has been obtained regarding thorium, which can also be a source of neutron radiation from alpha-neutron interactions. Similarly, little site-specific information has been obtained about PuBe neutron source production.

6.3 PERSONNEL MONITORING

From a cursory review of claimant records not all personnel were required to wear dosimeters but the work areas were monitored based on the available claimant information. For example, Nuclear Science and Engineering Corporation performed the personnel film dosimeter service for several years. In September 1963 Controls for Radiation Company provided a monthly service for 11 area film dosimeters that were positioned in process areas.

6.3.1 Beta/Gamma Radiation

Records of beta/gamma radiation doses to individual workers from personnel dosimeters are available for NUMEC operations beginning in 1957 for Apollo as observed from a review of claimant records. Doses received by these dosimeters were recorded at the time of measurement and the results routinely reviewed by NUMEC operations and radiation safety staff to assess compliance with radiation control limits. The *External Dose Reconstruction Implementation Guide* (NIOSH 2007b) indicates that these records represent the highest quality records for retrospective dose assessments.

Film dosimetry was used from 1959 until about 1969, and then was replaced with thermoluminescent dosimeters (TLDs). There is evidence of NUMEC concern about film dosimetry over response to the low energy photons from plutonium (Caldwell 1966b). This issue seemingly resulted in a change in vendor to Landauer 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 subsequently provided by Landauer beginning in 1976. Table 6-1 summarizes the respective venders and period of service; although the B&W records indicate the data work sheets and data for the TLD program, the actual calibration and dose calculation algorithm is not readily apparent.

6.3.2 Neutron Radiation

The source of neutron radiation at the NUMEC Apollo facility would be limited to work with HEU, and recycled uranium. Workers were monitored for neutron exposures likely with nuclear track emulsion, Type A (NTA) film from the respective commercial venders until about 1968, and then using TLDs thereafter.

Criticality dosimetry monitoring with an array of area critical assemblies that fed into a central system existed from at least 1963. As of at least 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) used by NUMEC are expected to be similar to contemporary commercial vendor capabilities. Landauer film badges were reported in

January 1961 with a lower reporting limit of 10 mrem for gamma, 40 mrem from beta, 15 mrem from neutrons and 20 mrem from fast neutrons, based on information in claimant files. However, other documentation indicates that film dosimeter MDLs in the workplace are higher. This document recommends using MDLs of 40 mrem for gamma and beta radiation, and 50 mrem for neutron radiation (ORAUT 2006a, 2007d). Table 6-1 summarizes the venders, frequency, MDLs, and the annual missed dose.

Table 6-1. MDLs and potential missed photon, beta, or neutron dose.

Vendor or processor/area monitored	Period of use	MDL[b] (rem)	Annual missed dose[c] (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[a]	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[d] 0.02 beta[d]	0.12 beta photons (monthly)
		0.02 neutrons[d]	0.12 neutrons (monthly)
Landauer TLD *whole body*	1976–1995	0.01 photons 0.01 beta	0.06 beta photons (monthly)
		0.01 neutrons	0.06 neutrons (monthly)

a. Landauer (1964).
b. Estimated MDLs for each dosimetry technology. Dose levels were recorded at values of less than the MDLs.
c. Annual missed dose calculated based on the MDL/2 method from NIOSH (2007a).
d. MDL during this period is likely twice the recording levels.

Capabilities of dosimetry technology used by commercial or DOE Laboratory vendors were comparable in performance since similar films and calibration procedures were used. Characteristics of Landauer dosimetry capabilities are presented in Table 6-2 for beta, gamma, and neutron radiation monitoring (ORAUT 2006b).

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. The film badges likely had a gamma and X-ray: 30 keV to 20 MeV; beta: over 1.5 MeV, typical of vendor offered dosimetry.
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. The TLD badges likely had a gamma and X-ray: 30 keV to 20 MeV; beta: over 1.5 MeV, typical of vendor offered dosimetry.
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. Replaced Landauer G1 - Film. The TLD badges likely had a gamma and X-ray: 30 keV to 20

Period	Monitoring technique	Dosimeter description
		MeV; beta: over 1.5 MeV, typical of vendor offered dosimetry.
Beta/photon dosimeters – Wrist and Ring		
July 1963–about 5/1968 Film	Landauer Type M (wrist beta-gamma) badges or equivalent.	Type M dosimetry was a film badge. Gamma and X-ray: 30 keV to 20 MeV; beta: over 1.5 MeV. The film badges likely had a gamma and X-ray: 30 keV to 20 MeV; beta: over 1.5 MeV, typical of vendor offered dosimetry.
About 6/1968–1983	K5 wrist badge TLD	Comprised of 3 TLD-100 chips. The TLD badges likely had a gamma and X-ray: 30 keV to 20 MeV; beta: over 1.5 MeV, typical of vendor offered dosimetry.
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 1000X power microscope or projection capability.
6/1968–1995	Landauer I8, I1 or RI Neutrak Extended Range dosimeter.	Combined TLD albedo neutron monitor with track recoil device (CR-39 [allyl diglycol carbonate]) that responds to proton recoil events. Neutron energy range was approximately 1 x 10^{-6} to 10 MeV. Albedo 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. This badge was combined with G1 to make dosimeter, known as R1 that monitored beta, X-ray, gamma, and neutrons.

Source: ORAUT (2006b).

6.3.4 Radiological Records

In compliance with 10 CFR 20.407, NUMEC was required to submit reports of personnel exposure information to the AEC/NRC. Only two records from 1976 and 1977 were found in publicly available databases at the NRC that provide personnel exposure information (Bruer 1977, 1978). The annual whole-body doses were segregated into dose bins with the number of workers in each bin. The results of the research of NRC records are shown in Table 6-3.

Table 6-3. Annual occupational radiation exposures at the Apollo facility (Bruer 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

In addition, an AEC inspection report showed the following highest beta-gamma and neutron exposures by quarter in 1973. Table 6-4 shows these findings.

Table 6-4. Highest beta-gamma and neutron exposures measured for 1973 (Nelson 1974).

Quarter	Highest beta-gamma (mrem)	Highest neutron (mrem)
1	141	169
2	153	215
3	94	773
4	111	352

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

Facilities like the Apollo site handling primarily uranium would not have significant exposure to low energy photons (ORAU 2006e). If there is an indication that the worker was exposed to low-energy photons from plutonium work, then the guidance in this section should be followed.

Experience at NUMEC since the mid 1960s indicated the limitations of film external dosimetry for the monitoring of plutonium low-energy photons. Caldwell (1966b) 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
- Effective energy of 400 keV.

There is no information on use of plutonium at the Apollo site. However, a study performed at the Parks Township site evaluated photon fields from plutonium work. A photon spectroscopy survey was conducted at the plutonium laboratory to try to ascertain the energy spectrum of the NUMEC Parks Township plutonium work environment. Even though it was suspected that the 17- and 60-keV gamma radiation groups would dominate, uncertainties in geometry, self absorption and incidental shielding material would make theoretical or model calculations difficult (Caldwell 1966b).

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 absorbed in great part by the glovebox walls (Caldwell 1966b, 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 because the plutonium in the chemical processing line had aged an additional 2 years whereas the plutonium in the melt box had just been received (Caldwell 1966b).

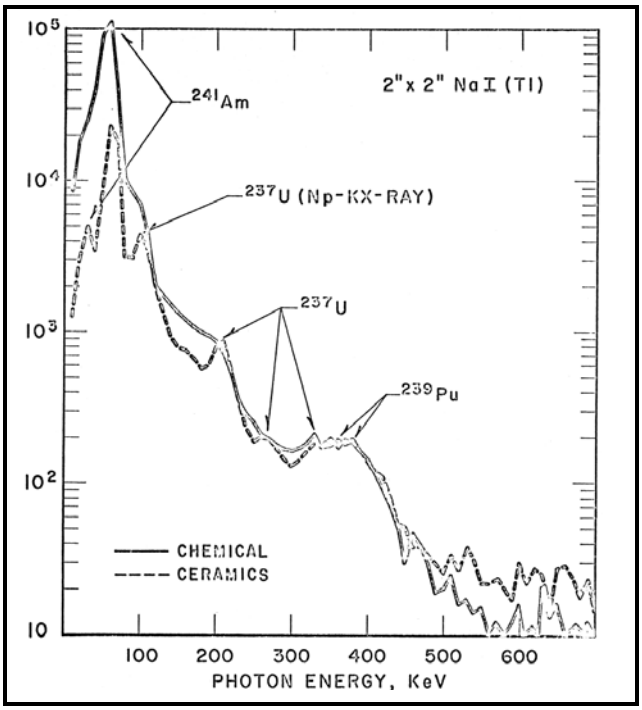

Figure 6-1. Gamma spectrum external to plutonium chemical processing line and plutonium ceramics line (Caldwell 1966b, figure 3).

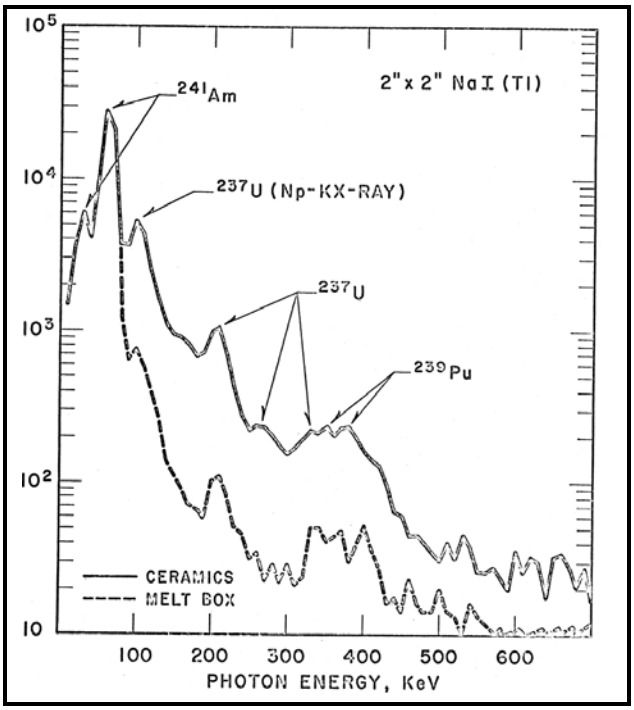

Figure 6-2. Gamma spectrum external to plutonium ceramics line and plutonium melt glovebox (Caldwell 1966b, figure 4).

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

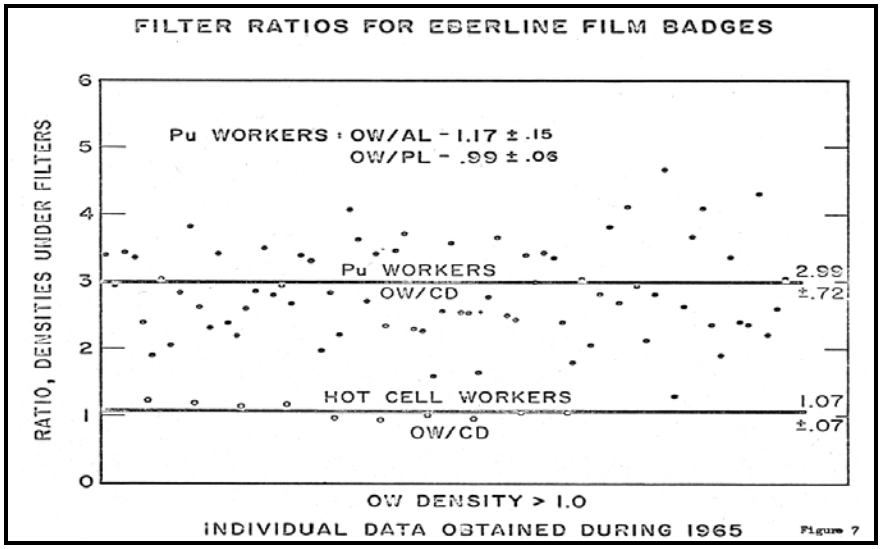

Figure 6-3. Dosimeter filter ratios (Caldwell 1966b).

The following interpretation was made by Caldwell (1966b):

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

2. Plutonium workers were exposed to a wide range of gamma energies. The upper end of the spectrum would produce OW/CD ratios close to 1.0. The lower gamma energies would produce an OW/CD ratio of 3.0 or greater. One plutonium worker may 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 into their analysis.

Caldwell (1966b) presented an assessment of the energy dose fraction for personnel exposure due to typical plutonium fuel fabrication from 1,000 MWD/T 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 1966b). 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 1966b). 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.

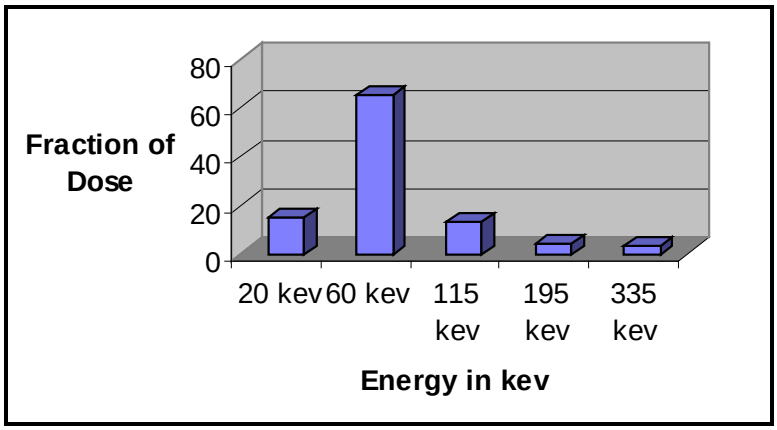

Figure 6-4. Relative contribution by energy group to personnel exposure during plutonium fuel fabrication from 1000 MWD/T plutonium (Caldwell 1966b, p. 18, Figure 8).

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 is a poor detector of neutron energies below 500 to 800 keV (ORAUT 2006c). Table 6-6 contains neutron dose correction factors for NTA film developed for dose reconstruction at Argonne National Laboratory-East (ANL-E) to account for missed low-energy neutrons. The facilities and activities at the Laboratory appear to encompass the activities and facilities at NUMEC. An NTA film energy threshold correction factor of 1.5 is recommended for NUMEC due to the presence of uranium fuel and critical assemblies. If the information on a worker indicates that work with plutonium occurred, then the energy threshold correction factor of 4 would be appropriate.

Table 6-6. NTA energy threshold correction factors.

Facility type	NTA film neutron dose correction factors[a]
U Fuel and critical assemblies (and U-Th MOX)	1.5
Pu handling facilities (and U-Pu MOX)	4

a. Adapted from ORAUT (2006c).

6.5 DOSE RECONSTRUCTION RECOMMENDATIONS

6.5.1 Recorded Dose Practices

Recorded and reported dose practices are summarized in Tables 6-7 and 6-8.

Table 6-7. Recorded dose practices.

Year	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 6-8. 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 zero 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

Period	Reported quantity	Description	Interpretation of zeroes	Interpretation of blanks (no data)	Rollup of individual and annual data	Monitored/ unmonitored
1983–present	rem	Photon deep, neutron deep, and skin dose reported.	Zeroes were typically reported. Reported zero 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

Landauer documentation also indicates that the Landauer or other badges used in the 1960s at NUMEC had a beta energy threshold on the order of 1.5 MeV resulting in an under reporting of shallow dose equivalent. Consequently, the recorded nonpenetrating film results should be multiplied by 2.0 to determine the corrected beta dose (ORAUT 2005c). Nevertheless, if a NUMEC employee worked with a source term that included primarily beta emitting radionuclides with a maximum beta energy less than 1.5 MeV, the personal beta dose would be essentially unmonitored.

Beta and nonpenetrating dose was usually reported before 1975. In general, nonpenetrating radiation doses should be assigned as <30-keV photons if the employee worked with or around Plutonium, otherwise, >15-keV electrons (beta) should be assigned (ORAUT 2005c). The guidance from ORAU 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 may hinge on this issue, and if the partitioning of the nonpenetrating dose cannot be decided based on the available information, additional research may 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 estimates of the actual doses.

6.5.2.3 Neutron Dose Adjustments

Because very little neutron spectral information is available, this analysis assumed that the neutron doses reported were calculated using the quality factors (QFs) from National Council on Radiation Protection and Measurements Report 38 (NCRP 1971). The Report 38 QFs were averaged over the ICRP Publication 60 (ICRP 1991) energy groups. A dose multiplier of 1.91 was used for conversion of the Report 38 dose equivalent to the Publication 60 equivalent dose for the 0.1 to 2 MeV energy range (ORAUT 2006c), which includes 100% for HEU, EU, natural uranium, and others. Consequently, the neutron doses recorded on film for personnel at NUMEC should be multiplied by a factor of 3 to account for changes in QFs and NTA film energy threshold as shown in Table 6-9.

Table 6-9. Neutron dose correction factors.[a]

Facility type	NTA film neutron dose correction factors[a]	ICRP 38 to ICRP 60 adjustment factor	Total neutron dose correction factor
U fuel and critical assemblies (and U-Th MOX)	1.5	1.91	3

a. Adapted from ORAUT (2006c).

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 the External Dose Reconstruction Implementation Guide (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 a zero or that is below the LOD/2. The LOD values for NUMEC dosimeters are provided in Table 6-1.

For cases involving the skin as the target organ, guidance in the Technical Information Bulletin *Interpretation of Dosimetry Data For Assignment of Shallow Dose* (ORAUT 2005c) 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 6-1 for monitored workers. The missed dose should be adjusted based on the LOD/2 method described above for shallow and deep dose.

If monitoring records do not included neutron dosimetry information, but the worker is suspected of being exposed to neutrons, the neutron dose can be estimated using a neutron-to-photon dose ratio. Measurements or a study for a neutron to gamma ratio for uranium work has not been located. A neutron to gamma ratio of 0.2 has been determined for uranium processing in gaseous diffusion plants from measurements made by Cardarelli (1997). Exposure to uranium hexafluoride cylinders is a possible source of neutron exposure at the Apollo site.

6.5.4 Uncertainty

Dose reconstructors may 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 non-compensable 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 (ORAUT 2006d). 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 (ORAUT 2006d). 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 becomes correspondingly more energetic. The average energy of the spectra can increase from solid or liquid uranium sources since these may 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. This includes thorium, plutonium, and photon sources such as radium, ^{192}Ir, ^{137}Cs and ^{60}Co. Mixed fission product exposure could occur at the 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 6-10 unless there is specific claim specific information regarding the source of radiation exposure.

Table 6-10. Beta, photon, and neutron radiation energies and percentages for IREP input.

Description	Dates		Radiation type	Energy selection	Percent
	Begin	End			
Uranium facilities	1/1/1957	12/31/1983	Beta	> 15 keV	100
	1/1/1957	12/31/1983	Photon	30-250 keV	100
Thorium handling	1/1/1957	12/31/1983	Beta	> 15 keV	100
	1/1/1957	12/31/1983	Photon	30-250 keV	75
	1/1/1957	12/31/1983		>250 keV	25
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 Energy Employees Occupational Illness Compensation Program Act. The residual period for the NUMEC Apollo plant covers the period from 1984 through 1995.

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

7.1 EXTERNAL DOSE FROM RESIDUAL ACTIVITY IN THE WORKPLACE

If dosimeter readings are available for the residual period, then 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 may 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, then 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 (rem/yr) = residual level (dpm/m^2) x DCF (rem/dpm/m^2/hr) x exposure time (hr/yr)

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 2000 hours per year. The dose values are for exposure to natural uranium as this provides a higher external dose than other enrichments (except depleted uranium). This provides a favorable to claimant dose estimate because most uranium at the Apollo uranium facility was natural or enriched uranium. Natural uranium 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).

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	0.378
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 4.1.

The dose should be entered into the IREP input as a lognormal distribution with a geometric standard deviation of 7.91 (parameter 2) as photons in energy range 30 – 250 keV as indicated in Table 6-10. This provides a favorable estimate of probability of causation for all organs, even though some of the photon energy is likely to be of higher energy.

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. Therefore, medical X-ray doses should not be assigned after 1983.

7.4 INTERNAL DOSE FROM RESIDUAL ACTIVITY IN THE WORKPLACE

If bioassay data is available during the residual period, then that 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. However, bioassay data is unlikely to be available after 1985.

When bioassay data is not available, and the worker may have worked in areas containing residual activity, the following method may be used to assign internal dose.

Guidance in the Technical Information Bulletin Dose Reconstruction during Residual Radioactivity Periods at Atomic Weapons Employer Facilities (ORAUT 2008) 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. 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 geometric standard deviation 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 2008). The annual deposition amount is estimated using a deposition velocity of 0.00075 m/s, with deposition assumed to occur for one 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 2008). The depletion factors applied to each year are described in Table 3-1 of ORAUT (2008a). 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 2008) of $1x10^{-6}$ m^{-1}. Application of this resuspension factor, and the depletion factors described above to the residual contamination level of $1.605x10^{6}$ dpm/m^2, results in the air concentration and annual intakes given in Table 7-2. The intake evaluation is based on exposure for 2000 hours per year and an inhalation rate of 1.2 m^3/hr.

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

Year	Air concentration, dpm/m3	Intake, dpm/year
1984	4.97	1.19E+04
1985	0.149	3.58E+02
1986–1995	0.00348	8.34E+00

The intakes in Table 7-2 can be used to estimate the internal dose to the target organ for the years of employment for the worker. The estimated internal doses are assigned as a lognormal distribution with a geometric standard deviation 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 at the NUMEC Apollo facility.

The internal dose analysis should include the potential inadvertent ingestion of uranium activity, based on guidance in OCAS (2004). The daily intake rate (dpm/day) 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/day is obtained. This value should be applied for all years of the residual period, without depletion, as the transfer of material from the contaminated surfaces may not be reduced with time to the same extent that resuspension is reduced. This provides a favorable to claimant assessment of ingestion intake.

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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ATTACHMENT A
BREATHING-ZONE AIR SAMPLE SURVEY DATA

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Table A-1. 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
Center less Grinder	2	23
Coated Sphere Operator	6	31
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CRP-3 Filter and Feed Prep Operator	4	39
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a. Data from AEC (1960c).

ATTACHMENT A
BREATHING-ZONE AIR SAMPLE SURVEY DATA

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
Center less 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).

ATTACHMENT A
BREATHING-ZONE AIR SAMPLE SURVEY DATA

Table A-3. 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).

ATTACHMENT A
BREATHING-ZONE AIR SAMPLE SURVEY DATA

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-Center less 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).

ATTACHMENT A
BREATHING-ZONE AIR SAMPLE SURVEY DATA

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-Center less 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	6	22	Normal

a. Data from AEC (1961b).