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Site Profile for Nuclear Materials and Equipment Corporation, Apollo and Parks Township, Pennsylvania

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☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

PUBLICATION RECORD

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

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
08/25/2016	03	Revision initiated to respond to the S. Cohen & Associates review. Updated Table 2-3 to revise operating periods for Parks Township activities. Revised Section 5.1 and Table 5-3 to provide additional information on plutonium mixtures and evaluation of internal dose from plutonium mixtures and associated radionuclides. Added Section 5.2.2.4 to provide guidance on evaluation of intakes based on uranium fluorimetric bioassays. Updated Table 5-8 to include a few more MDA values for lung counts of ^{239}Pu and ^{241}Am. Guidance was added to use the maximum MDA values for years when the bioassay records do not provide an MDA value. Revised Section 5.2.4 to eliminate use of ORAUT-OTIB-0054 for evaluation of intake of associated fission and activation products. Only the doses from listed radionuclides are to be evaluated. Revised Section 5.2.5 to indicate recycled uranium contaminants should be included in the evaluation of the material solubility type that gives the highest internal dose. Added Section 6.5.2.4 and Tables 6-8 and 6-9 to provide guidance on estimation of neutron dose before 1969 using neutron-to-photon ratios. Revised Table 7-1 to include whole-body dose and skin dose with guidance to apply the isotropic exposure dose conversion factors. This is consistent with the evaluations performed for chronic lymphocytic leukemia. Included exposure to thorium in the table. Added Table 7-2 to provide guidance on assignment of onsite external dose for organs affected by shallow dose. Revised Table 7-3 to use general air sampling data rather than breathing zone data as the basis for intakes of residual uranium. Added Table 7-4 for residual thorium intakes on the same basis. Revised the air concentration for residual air concentrations at Apollo and Parks Township uranium facilities to be based on a resuspension factor of $1x10^{-5}$ (rather than 1×10^{-6}) and updated Tables 7-3, 7-4, and 7-7. Added Attachment B to provide guidance on evaluation of the various dosimetry reports available in the dosimetry records for NUMEC workers. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by Objective Manager. Initiated by Dennis L. Strenge.

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ACRONYMS AND ABBREVIATIONS

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

L	liter
lb	pound
LEU	low-enriched uranium
LLRW	low-level radioactive waste
LOD	limit of detection
m	meter
M	moderate (absorption type)
mCi	millicurie
MDA	minimum detectable activity (or amount)
MDC	minimum detectable concentration
MDL	minimum detectable level
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
min	minute
mL	milliliter
MOX	mixed oxide
MPC	maximum permissible concentration
mrad	millirad
mrem	millirem
MWd	megawatt-day
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
NS&E	Nuclear Science & Engineering
NTA	nuclear track emulsion, type A
NU	natural uranium
NUMEC	Nuclear Materials and Equipment Corporation
ORAU	Oak Ridge Associated Universities
OW	open window
pCi	picocurie
PL	plastic
PNC	Power Reactor and Nuclear Fuel Development Corporation
POC	probability of causation
R	roentgen
R&D	research and development
RU	recycled uranium
s	second
S	slow (absorption type)
SEC	Special Exposure Cohort
SLDA	Shallow Land Disposal Area
SNM	special nuclear material
SRDB Ref ID	Site Research Database Reference Identification (number)
SRS	Savannah River Site
SS	super slow (absorption type); refers to highly insoluble plutonium
t	ton (metric)
TLD	thermoluminescent dosimeter

TRU	transuranic
U.S.C.	United States Code
WB	whole body
wt %	weight %
yr	year
ZPPR	Zero Power Plutonium (later Physics) Reactor
ZPR-III	Zero Power Reactor
µCi	microcurie
µg	microgram
§	section or sections

1.0 INTRODUCTION

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

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

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

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

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

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

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

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

1.1 PURPOSE

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

1.2 SCOPE

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 8.0.

1.3 SPECIAL EXPOSURE COHORT

This section describes the classes in the SEC for the NUMEC sites in Apollo and Parks Township, Pennsylvania.

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

1.3.1 Apollo Site Class

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

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

- Uranium internal exposure before 1960 for lack of bioassay monitoring;

- Thorium and plutonium internal exposures for lack of monitoring data, process description, and source term data;

- Potential ambient radiation dose from stack releases;
- Dose from radium-beryllium and polonium-beryllium neutron source fabrication operations; and
- Internal doses if the bioassay data were based on the NUMEC contactor, Controls for Environmental Pollution, from 1976 through 1983, because of concerns about data quality.

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

1.3.2 Parks Township Site Class

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

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

- Thorium internal exposures for lack of monitoring data and process descriptions;
- Internal exposures for work with irradiated fuel and fabrication of radiation sources for lack of monitoring data, process descriptions, and source term data; and
- Internal doses if the bioassay data were based on the NUMEC contactor, Controls for Environmental Pollution, from 1976 through 1980, because of concerns about data quality.

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

2.0 SITE DESCRIPTIONS

2.1 APOLLO SITE

2.1.1 General Description and Operational History

The Apollo nuclear fuel site operated under license SNM-145 and Source Material License C-3762 that the U.S. Atomic Energy Commission (AEC; a DOE predecessor agency) issued in 1957. From 1958 to 1983, the Apollo site was used for small-scale research and production of low-enriched uranium (LEU), highly enriched uranium (HEU), and thorium fuels. By 1963, the majority of the Apollo

facility was dedicated to production of uranium fuel. The major activities at NUMEC Apollo included (1) the conversion of LEU hexafluoride (<5% ^{235}U by weight) to uranium oxide (UF_6 to UO_2) for use in light-water-moderated reactors; (2) the production of nuclear fuel from HEU (>20% ^{235}U) for the use in the naval reactors program; and (3) the processing of unirradiated uranium scrap (including LEU and HEU) from the AEC in the 1960s (B&WNES 1997).

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

The Apollo facility had one main bay (known as the East Bay) and three smaller attached bays known as the West Bay, the Box Shop, and the Annex. The site included a Laundry Building and a Small Block Building in the parking lot. The Laundry Building was used for washing protective clothing from the nuclear facilities, and the Small Block Building was used for storage of processing equipment. These buildings were on the east side of the site between Warren Avenue and the Kiskiminetas River. The parking lot area was bounded by the Kiskiminetas River on the west, Warren Avenue on the east, and the offsite area on the north. Figure 2-1 is a general layout of the Apollo site.

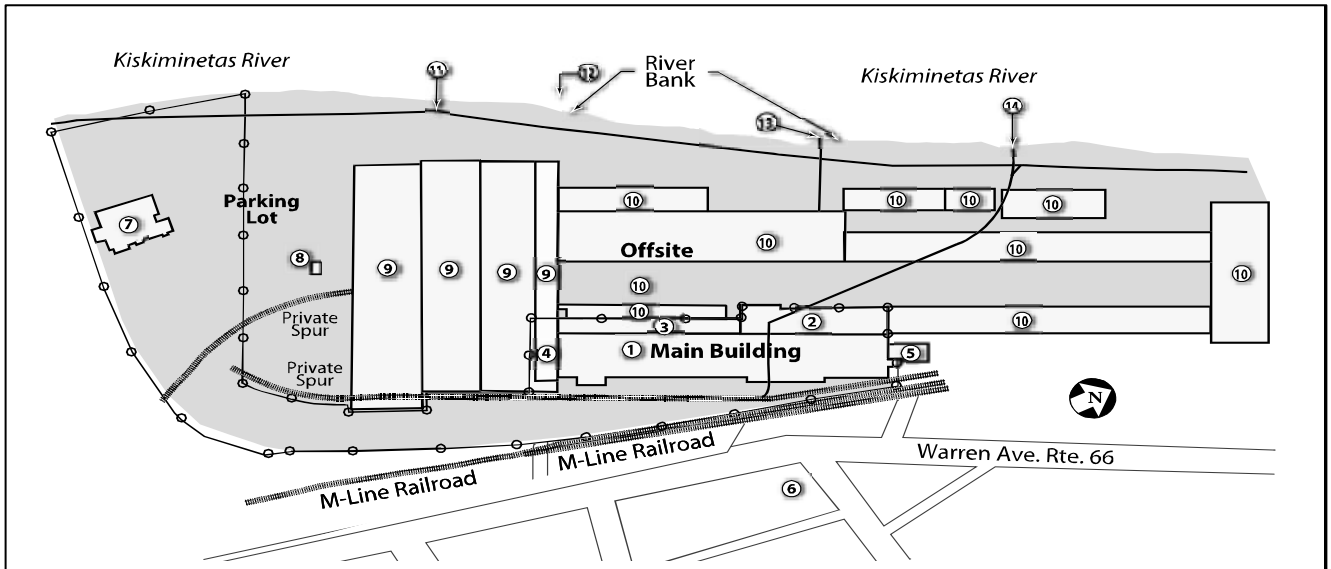

Figure 2-1. Apollo site layout (Author unknown 2004).

2.1.2 Facilities

The Apollo site was divided into production and process areas and clean areas. Personnel were required to enter through the main entrance near the parking lot. Before exiting through the main entrance, personnel were to shower if they had entered or worked in a production or process area. There were two emergency exits. Entrance into production areas was through the change room with

the exception of the filter press section of the plant waste treatment area (NUMEC 1963). Production and process areas are listed in Table 2-1.

2.1.3 Process Descriptions

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

2.1.3.1 General Process Operations

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

2.1.3.2 Uranium Process Activities

A general description of processes for various enrichments of uranium at Apollo is outlined in AEC HASL Survey Reports 82, 92, 103, 106, and 114 (Occupational Exposure to Radioactive Dusts reports AEC 1960a, 1960b, 1960c, 1961a, 1961b), which cover the period from December 1959 to January 1961, and *Procedures for Recovery of Uranium Scrap* from 1962 (NUMEC 1962). The following paragraphs provide a description of the process.

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

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

Table 2-1. Apollo site area descriptions.

Building or area location	Description	Operations/radionuclides	Period of operation
CF-1	Ceramic fabrication	UO_2, ThO_2, (metal, powder, and oxide)	1957–1970, ThO_2: 1963–1970
CF-2	Ceramic fabrication	Uranium metal (HEU and DU) UO_2, and U_3O_8	Early 1959–1972
PC-1	Process chemistry	HEU, EU, DU, $(NH_4)_2U_2O_7$), UO_3, UF_6, UF_4, uranyl nitrate, UO_2, and U_3O_8	1957–1983, HEU: 1957–1978 LEU: 1957–1983
PC-2	Process chemistry	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
PC-3	Process chemistry	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CP-1	Chemical processing	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CP-2	Chemical processing	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CRP-1	Chemical reprocessing	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CRP-2	Chemical reprocessing	HEU, EU, DU, U nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
CRP-3	Chemical reprocessing	Beryllium handling equipment, HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
GF-1	Not reported	Not reported	1957–1983
QC	Quality control	Not reported	1957–1983
PS	Not reported	Not reported	1957–1983
A Vault	Process security material. Controlled by CP-2	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
C Vault	Receiving and shipping storage area	Not reported	1957–1983
E Vault	Storage of SNM	Not reported	1957–1983
F Vault	Storage for SNM of all enrichments	Not reported	1957–1983
G Vault	Solution storage area, in-process storage of materials for CP-1	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, and U_3O_8	1957–1983
H Vault	Storage of SNM	Not reported	1957–1983
Waste Treatment Area	Filter press section	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, U_3O_8, FPs, and TRU elements	1957–1983
GPH Room 2	Health and safety counting room	HEU, EU, DU, uranyl nitrate, UO_2, UF_6, UF_4, U_3O_8, FPs, and TRU elements	1957–1983
Laundry Building	Cleaning protective clothing	Co-60, plutonium, thorium, and uranium	1960–1983 (closed 1984)
Block Building	Storage	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 (ca. 2004); NUMEC (1963); B&WNES (1997); Caldwell (1966, 1968a).

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

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

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

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

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

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

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

> *The oil was drained and the chips were transferred to wire mesh baskets and degreased with trichloroethane in a ventilated degreasing tank. The chips were then dissolved in hydrofluoric acid in chemical fume hoods. The batch was heated and hydrogen peroxide was added to oxidize the insoluble UF_4 to soluble UF_6. The batch was filtered and the filtrate was reduced in chemical fume hoods, with insoluble UF_4 precipitating preferentially from the solution. The solutions were filtered and the UF_4 collected in a common filter. The UF_4 was then converted to U_3O_8 by ignition.*

According to NUMEC (1962), scrap was dissolved in two designated areas, CRP-2 and CRP-3. The product solutions from the dissolution methods were processed to generate insoluble UF_4, and were ultimately converted to the final product of U_3O_8 or UO_2.

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

According to the HASL-103 (AEC 1960c) survey report, a cascade-solvent extraction uranium-zirconium recovery process was under construction. A similar type of extraction process was already in operation for Chemical Reprocessing (CRP), which consisted of leaching, feed preparation, solvent

extraction, ammoniation of strip solution, precipitation, filtration of ADU slurry, kiln drying, and packaging as UO_3.

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

A March 25, 1960, letter (Katine 1960) recommended approval of a NUMEC Feasibility Report to fabricate 3,000 graphite fuel elements to be used in the General Atomics high-temperature gas-cooled reactor (HTGR) critical assembly. The job was to involve between 95 and 120 kg of 93%-enriched U_3O_8 supplied by another company. The total ^{235}U content of 2,850 fuel elements was to be 79.339 kg. An April 1960 letter from AEC approved approval of Feasibility Report for the General Atomic HTGR critical assembly fuel elements (Katine 1960).

2.1.3.4 Uranium Nitrate Solution for the University of California

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

2.1.3.5 Incinerator

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

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

The operator was protected during charging by a positive inflow of air through the charging door. Ash collection was enclosed in an exhausted box. All ash handling was restricted to a glovebox at negative pressure (Caldwell 1968a). Before 1968, this might not have been the case and work in this area was perhaps the highest for intakes.

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

2.1.3.6 Thorium Operations

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

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

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

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

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

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

5. The ThO_2 pellets would be pressed using a hand press, an automatic press, or both in a hood with a face velocity of 100 ft/min.

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

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

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

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

2.1.3.7 Research Activities in the Early Years

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

2.1.4 Source Term

Three main sources describe the amounts and types of radioactive material at the Apollo site: (1) federal and State of Pennsylvania licenses for the possession and use of radioactive materials; (2) descriptions and reviews of proposed experiments or jobs handling radioactive material in the form of HASL reports and process feasibility reports, which contain information on radionuclides, quantities, and recommended safety precautions for the described activity; and (3) inventory and material handling (accountability) records.

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

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

Uranium
Uranium in the form of metal, oxide, and carbide was used for Apollo fabrication, reactor fuel, and research studies in the gloveboxes and laboratories (AEC 1960a, 1960b, 1960c, 1961a, 1961b). The typical amounts of uranium in use in an area ranged from milligrams to hundreds of kilograms.

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

Source: Reitler (1972).

Occasional work involving other uranium chemical forms, such as UF_6 or uranyl nitrate, was conducted. Uranium forms included depleted uranium (DU), natural uranium (NU; i.e., natural enrichment), HEU (up to 93%), as well as ^{232}U, ^{233}U, and ^{236}U. Uranium from recycling operations would have included smaller activities of nonuranium isotopes such as ^{99}Tc, ^{237}Np, and ^{239}Pu.

Thorium

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

2.1.5 Remediation, Decontamination, and Decommissioning

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

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

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

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

As of August 23, 1978, NUMEC had completed decommissioning of its HEU processing at the Apollo site. All process and related equipment were removed by this date. NUMEC indicated that access to the area was limited to authorized personnel. In 1982, the NRC conducted a confirmatory survey to

identify HEU that might still have been present. The report indicates uranium contamination levels in grams of uranium to surface area. This report was generated to account for HEU inventory during decommissioning. The total grams remaining on and in the floors, walls, pad and ceiling were estimated to be 35,548.55 g of total uranium with about 23,743.27 g of ^{235}U (Martin 1982).

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

2.1.5.1 Shallow Land Disposal Area

In October 1995, the NRC placed the adjacent B&W Shallow Land Disposal Area (SLDA) on a separate license. Until 1970, the SLDA was used as a disposal facility for the Apollo site (and possibly the Parks Township site) with about 700,000 ft^3 of waste buried in trenches (PDEP 2008). The site became a Formerly Utilized Sites Remedial Action Program site with management of the program transferred to the U.S. Army Corps of Engineers in 1998. As of 2007, a feasibility study was being conducted by the U.S. Army Corps of Engineers (USACE 2006). The area is shown in Figure 2-2.

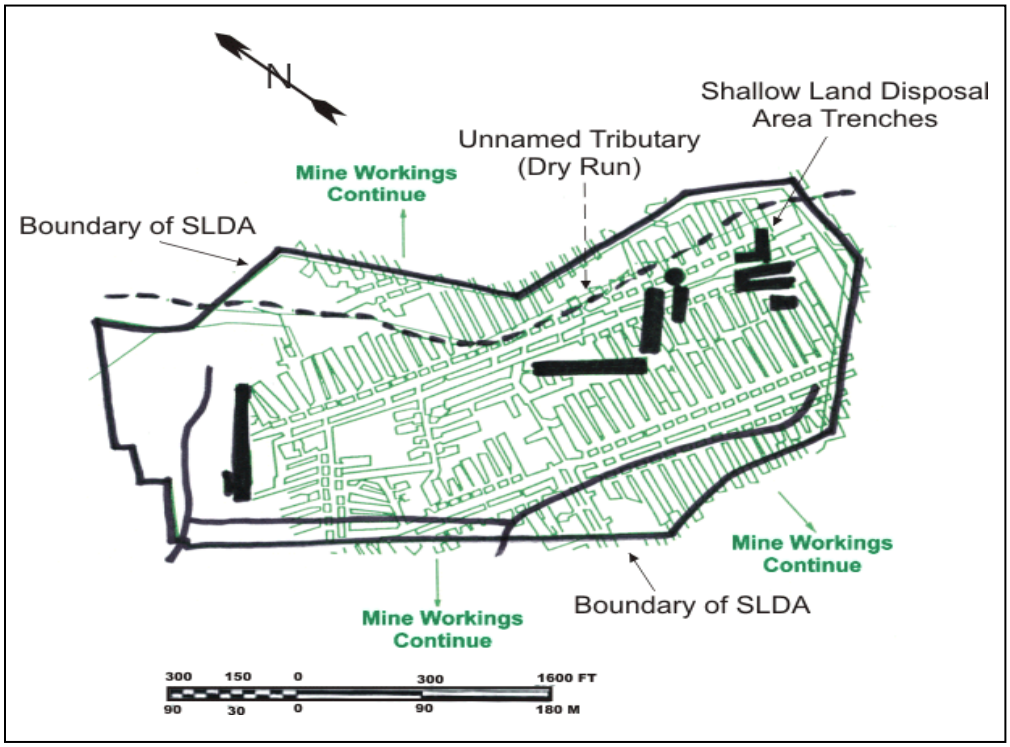

Figure 2-2. B&W shallow land disposal area (USACE 2006).

2.2 PARKS TOWNSHIP SITE

2.2.1 General Description and Operational History

The NUMEC Parks Township site (sometimes referred to as the Advanced Material Center) was on 115 acres in Parks Township in Armstrong County, Pennsylvania, along U.S. Highway 66 about 3 miles southeast of Leechburg and 30 miles northeast of Pittsburgh. The site expanded in stages throughout the 1960s. The original Building A was 20,000 ft^2. From 1961 through 1970, a major

expansion of Building A (the plutonium facility) was completed in five separate expansions (61,000 ft^2). This included construction of Fabrication Areas (Fabs) 2 through 9 and the Drum Warehouse as well as termination and remediation of a drum storage area. The main structures were Buildings A, B, and C, the outside Storage Areas, and the Trailer Storage Area (Author unknown ca. 2004).

Building A was constructed in 1959 and 1960 and was authorized to operate in 1961. The Hafnium Facility (part of the Building B complex) was constructed in 1960 and was operational in 1961, and the plutonium annex was completed and in operation in 1963 (for production of ^{238}Pu sources). The Metals Facility (Building B complex) was constructed in 1962 and was operational in 1963. The Machine Shop (Building B complex) was constructed in 1964. Building C, the Type II Uranium Facility, was constructed in 1972 in the existing incinerator building (built in 1969), and preproduction of Type II fuel began in 1973 (Author unknown undated a).

The initial functions of the Parks Township facilities were fabrication of plutonium fuel, preparation of HEU fuel, and production of zirconium/hafnium bars under AEC and later NRC License SNM-414 received in March 1961, which allowed the handling of plutonium already on the site. The Parks Township site made fuel for the DOE Fast Flux Test Facility (FFTF) at the Hanford Site in the 1970s and early 1980 (Author unknown 2004). The FFTF fuel was a mixture of PuO_2 and depleted UO_2. The site also made fuel plates for the DOE Zero Power Plutonium (later Physics) Reactor (ZPPR) in the late 1960s and Zero Power Reactor (ZPR)-III fuel wafers (Author unknown ca. 2004). The ZPR-III fuel program used 8.5% ^{240}Pu plutonium, and the ZPPR fuel program used primarily 12% ^{240}Pu (11,500 fuel plates), but also some 27% ^{240}Pu (700 fuel plates), which indicates a commercial fuel source having higher burn-up (B&WNES 1996). The highly enriched ^{240}Pu was not received on site until 1969. Activities included plutonium scrap recovery, DU fabrication, HEU fuel manufacturing, source manufacturing (primarily ^{60}Co, PuBe, and AmBe), irradiated fuel sample examination, laboratory operations, and supporting nuclear power site operations. The activities were conducted in Buildings A, B, and C (Author unknown ca. 2004). Production and process areas are summarized in Table 2-3 and discussed further in Section 2.2.2. Although many of the dates of operation in Table 2-3 indicate activity through 1980, some of the processes were probably terminated earlier; exact dates could not be determined from the available information. Scrap material was still present on site after completion of a fuel manufacturing campaign. Some of the processing areas are shown in Figure 2-3, which depicts the Parks Township site layout.

Table 2-3. Parks Township site area descriptions.

Building or area location	Operations	Radionuclides	Period of operation
Building A–Plutonium processing facility	Fabrication of plutonium reactor fuel pellets, blankets, rods	PuO_2; Pu nitrate and oxalates (AmBe, PuBe, 1959–1970); alpha, beta, and gamma sources	1961–1980
A–Fab 1	Plutonium conversion	Plutonium nitrate, plutonium oxide, depleted UO_2	1961–1980
A–Fab 1	Fuel fabrication for FFTF	Plutonium nitrate, plutonium oxide, depleted UO_2	1972–1980
A–East Side of Fab 1	Routine repair and maintenance of contaminated equipment	All	1961–1980
A–Fab 2	Fuel fabrication for ZPR-III	Plutonium nitrate, plutonium oxide, depleted UO_2	1964–1966
A–Fab 2	Fuel fabrication for ZPPR	Plutonium nitrate, plutonium oxide, depleted UO_2	1966-1970
A–Fab 3	Manufacturing operations, metallography	Plutonium nitrate, plutonium oxide	1963–1980

Building or area location	Operations	Radionuclides	Period of operation
A–Fab 3	Quality control of FFTF fuel	Plutonium nitrate, plutonium oxide	1972–1980
A–Fab 4	Alpha, beta, gamma, and neutron source fabrication	AmBe double encapsulated, PuBe compacted powder, Ir-192, Cs-137, Be-7, Po-210, Co-60, Am-241, Pu-238/239, PuO_2, plutonium and americium metal	1963–1980
A–Fab 5	Scrap recovery	Plutonium – various forms	1963–August 1, 1967
A–Fab 5	Analytical laboratory work	All, small quantities of radioactive samples	1979–1980
A–Fab 6	Scrap recovery	Plutonium – various forms	1968–1980
A–Fab 7	Fuel rod quality control tests, nonradioactive processes	All, clean and contaminated items	1968–1980
A–Fab 8	Storage	All, clean and contaminated items	1970–1980
A–Fab 9	FFTF fuel pin finishing	Encapsulated nuclear material	1970–1980
Building A–Hot Cell Room	Examination of irradiated samples, high-activity source fabrication	PuBe compacted powder, Co-60, Ir-192	1961–1969
Building A–Hot Cell Room	Storage of sources	Sealed sources, Ir-192, Cs-137, Be-7, Po-210, Co-60, Am-241, Pu-238/239, PuO_2,	1969–1980
Plutonium Facility Trailer Storage Area	Storage of large quantities of plutonium and uranium	Plutonium (nonpyrophoric), DU, NU, and EU (to 5% ^{235}U)	1961–1980
Building B–Multipurpose fabrication building	DU, NU, thorium, plutonium	DU metal or alloy, U_3O_8, Pu-238, ThO_2	1961–1980
Building B–Hafnium Plant	Metal production	Hafnium and Zr-Be alloy (nonradioactive)	1961–1980
Building B–Plutonium Annex	Conversion of Pu-238 nitrate to oxide	Pu-238 nitrate, Pu-238 oxalate, PuO_2 powder or alloys	1963–1980
Building B–Hot Cell Room	Large source production: Postirradiation examination of test capsules and fuel pins	Co-60, Cs-137, Ir-192, and PoBe, irradiated, uranium and plutonium and other TRU elements and FPs	1961–1980
Building B–Metals Plant, first floor	Small-scale metals production, fuel pellet production, materials testing	DU, UO_2, U_3O_8, UF_4, fully clad U-233, U-235, and Pu-239	1963–1974
Building B–Metals Plant, second floor	Pu-238 pacemakers	Pu-238-powered heart pacemakers	1963–1970
Building B–Machine Shop	Occasional machining of clad or unclad uranium, and clad plutonium and U-233; fabrication and repair of new and contaminated equipment from Parks Township and Apollo; machining of DU	Fully clad U-233 and Pu-239 and clad or unclad U-235 (any enrichment), primarily DU contamination; could include HEU, plutonium, thorium, and mixed FPs	1964–1980
Building C, Type II Facility or T-2 Plant	HEU processing to form sintered product	HEU (1973–1978), soluble chloride and oxide complexes, SNM oxides (UO_3, UO_2 and U_3O_8)	1973–1978
Outdoor scrap storage area	Storage	UF_6 cylinders	1971–1980

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

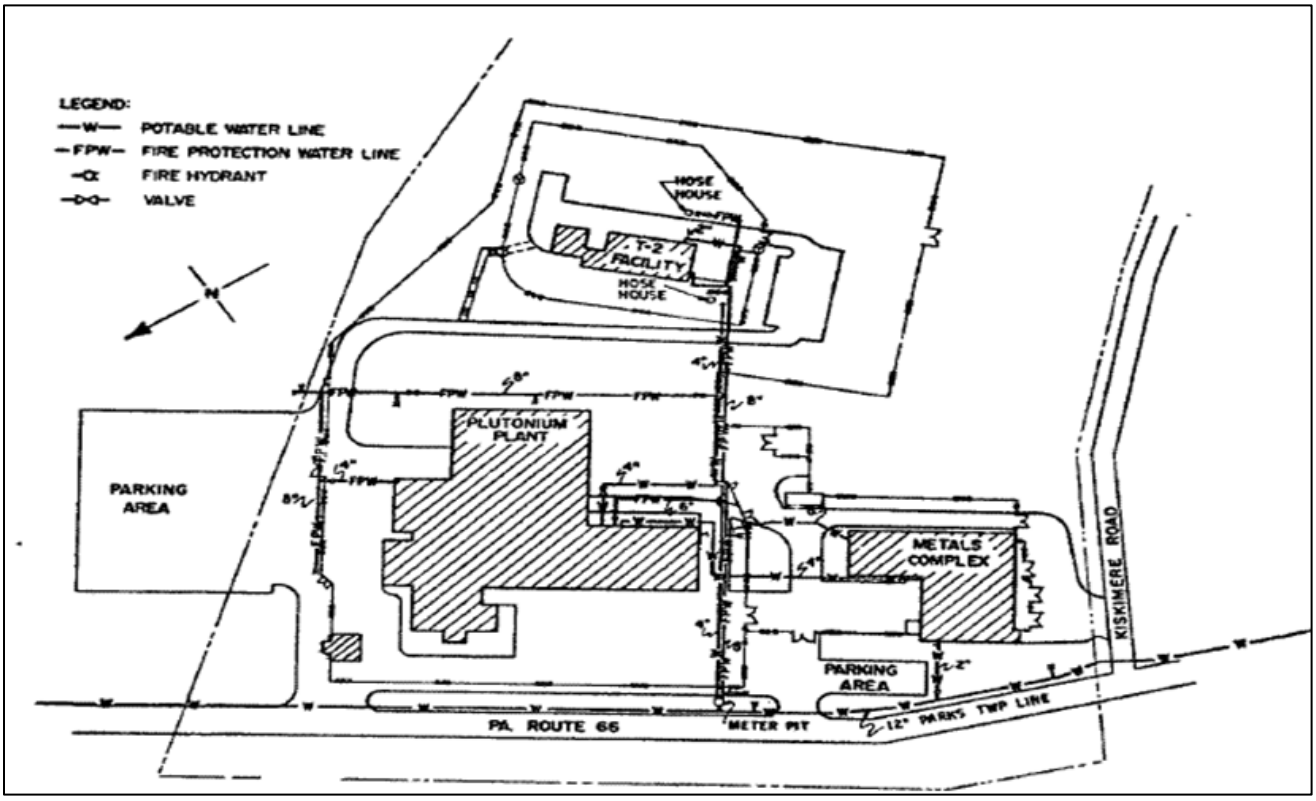

Figure 2-3. Parks Township site layout. Plutonium Plant (Building A), Metals Complex (Building B), and T-2 Facility (Building C) (Austin 1979).

2.2.2 Facilities

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

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

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

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

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

2.2.3 Process Descriptions

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

2.2.3.1 Building A – Plutonium Plant

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

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

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

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

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

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

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

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

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

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

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

2.2.3.2 Building B – Multipurpose Fabrication Building

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

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

2.2.3.2.1 The Hafnium Facility

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

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

2.2.3.2.2 The Machine Shop

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

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

2.2.3.2.3 The Metals Plant

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

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

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

2.2.3.3 Building C – Highly Enriched Uranium Processing Facility

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

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

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

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

2.2.4 Source Term

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

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

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

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

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

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

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

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

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

Source/chemical or physical form	Period	Maximum possession	Reference
Byproduct material any form	1979–end	20 mCi of any isotope	Rouse 1979; Austin 1984; Haughney 1991
Byproduct material any form, contaminated waste	1984–end	1,000 Ci of any isotope	Austin 1984; Haughney 1991
Byproduct material any form, contaminated waste on/in equipment	1991–end	5 Ci	Haughney 1991
Byproduct material any form, contaminated waste on/in equipment and metallic materials from other licensees	1984–end	5,000 Ci	Austin 1984; Haughney 1991
Byproduct material any form, contamination in volume reduction services waste	1984–end	500 Ci	Austin 1984; Haughney 1991

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

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

Source/chemical or physical form	Period	Maximum possession	Reference
Pu-239 as encapsulated calibration or reference sources	1981–1991	50 g	Austin 1981
Pu-239 as encapsulated calibration or reference sources	1991–end	285 g	Haughney 1991
Pu-241 as electroplated calibration or reference source	1981–1991	5 g	Austin 1981
Pu-241 as encapsulated calibration or reference source	1991–end	5 g	Haughney 1991
U-233 as evaporated calibration or reference sources	1981–1991	1 g	Austin 1981
U-233 as evaporated calibration or reference sources	1991–end	2 g	Haughney 1991
U-235 as evaporated calibration or reference sources	1991–end	5 g	Haughney 1991
U-235 as encapsulated calibration or reference source	1991–end	5 g	Haughney 1991
U-235 as electroplated calibration or reference source	1981–end	5 g	Austin 1981, Haughney 1991

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

Source/chemical or physical form	Period	Maximum possession	Reference
EU above 5 wt % U-235	1973–1978	HEU[a]	None
NU or DU and thorium	1969–1979	No limits	AEC ca. 1965–1969
EU above 5 wt % U-235	1979–1991	Possession: <5 kg U-235 Use: 1 kg effective	Rouse 1979, Austin 1981
EU above 5 wt % U-235	1991–end	Possession: <700 g U-235	Haughney 1991
U-235 as encapsulated or electroplated calibration or reference sources	1981–1991	5 g	Austin 1981
U-235 as encapsulated or electroplated calibration or reference sources	1981–1991	Possession: <5 kg Use: 1 kg effective	Austin 1981
U-235 as encapsulated or electroplated calibration or reference sources	1991–end	Possession: <700 g U-235	Haughney 1991
Pu-239 as encapsulated calibration or reference sources	1981–1991	5 g	Austin 1981
Pu-239 as encapsulated calibration or reference sources	1991–end	285 g	Haughney 1991
Pu-239 as electroplated calibration or reference sources	1981–end	5 g	Austin 1981
Pu-239 as electroplated calibration or reference sources	1991–end	20 g	Haughney 1991
Any isotope encapsulated in one or more sealed sources	1984–1991	10 Ci	Austin 1984
Plutonium in nonpyrophoric form	1991–end	<200 g	Haughney 1991

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

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

Place of use	Source/chemical or physical form	Period	Maximum possession	Reference
LLRW storage areas	Radioactive fissile material in approved storage containers	1969–1984	100 g/container	AEC ca. 1965–1969

Place of use	Source/chemical or physical form	Period	Maximum possession	Reference
Plutonium plant outdoor storage area	EU up to 5 wt % U-235 in UF_6 cylinders	1972–1973	75,000 kg UF_6	Browne 1978
Plutonium plant outdoor storage area	EU up to 5 wt % U-235 in UF_6 cylinders	1973–1984	200,000 kg UF_6	Browne 1978
Plutonium plant outdoor storage area	EU up to 5 wt % U-235 in UF_6 cylinders	1984–1991	100,000 kg UF_6	Austin 1984
Plutonium plant storage trailer	Plutonium in nonpyrophoric form with at least 3 wt % Pu-241	1979–1991	Up to 1,000 kg fissile	Rouse 1979
Plutonium plant storage trailer	Plutonium in nonpyrophoric form with at least 3 wt % Pu-241	1991–end	<200 g	Haughney 1991
Plutonium plant storage trailer	Plutonium and U-235 in approved shipping containers with valid certificates of compliance	1984–1991	Any quantity	Haughney 1991
Plutonium plant storage trailer	EU up to 5 wt % U-235, any physical or chemical form covered by authorized activities	1979–1991	25,000 kg U	Rouse 1979
Plutonium plant storage trailer	NU or DU, any physical or chemical form covered by authorized activities	1979–end	100,000 kg U	Rouse 1979
Plutonium plant storage trailer	Uranium, any enrichment U-235	1991–end	Possession: <250 g U	Haughney 1991
Plutonium plant storage trailer	Byproduct material any form, contaminated waste	1991–end	1,000 Ci of any isotope	Haughney 1991
Plutonium plant storage trailer	Byproduct material, any form, contaminated waste on or in equipment	1991–end	5 Ci	Haughney 1991
Plutonium plant storage trailer	Byproduct material, any form, contaminated waste on or in equipment, and metallic materials from other licensees	1991–end	5,000 Ci	Haughney 1991
Plutonium plant storage trailer	Byproduct material, any form, contamination in volume reduction services	1984–end	500 Ci	Austin 1984; Haughney 1991
Storage areas	EU to >5 wt % U-235	1991–end	Possession: <700 g U-235	Haughney 1991
Outside storage areas	EU of any enrichment in U-235	1991–end	Possession: <350 g U	Haughney 1991
Outside storage areas	Byproduct material, any form, contaminated waste	1991–end	1,000 Ci of any isotope	Haughney 1991
Outside storage areas	Byproduct material, any form, contaminated waste on or in equipment, and metallic materials from other licensees	1991–end	5,000 Ci	Haughney 1991
Any state except Agreement States	Neutron irradiator source	1969–unknown	Up to 96 g plutonium as PuBe neutron source	AEC ca. 1965–1969
Any state except Agreement States	Byproduct material, any form, contaminated waste on or in equipment	1991–end	5 Ci	Haughney 1991

Place of use	Source/chemical or physical form	Period	Maximum possession	Reference
Any state except Agreement States	Byproduct material, any form, contaminated waste on or in equipment, and metallic materials from other licensees	1991–end	5,000 Ci	Haughney 1991

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

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

2.2.5 Remediation, Decontamination, and Decommissioning

The Parks Township site ceased work for DOE operations in 1980. Decontamination and decommissioning of the facilities began in 1978 at Building C and continued through several phases for all facilities. Starting in 1994, B&W began final decontamination and decommissioning at the site to the extent permitted under the terms of its license. In January 1996, B&W submitted a sitewide decontamination and decommissioning plan and subsequent plan revisions in 1997 and 1998. In October 1998, NRC approved Revision 3.1 of the plan. Demolition and removal of all facilities was started at that time. All decommissioning activities had been completed by January 2002. All waste had been shipped to a licensed waste disposal facility, and the final status survey had been performed. After B&W completed 2 years of groundwater monitoring that showed site groundwater was within established limits, the NRC terminated the license and released the site for unrestricted use on August 24, 2004 (PDEP 2008).

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

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

to decontamination, maintenance, and storage of nuclear industry equipment (Haughney 1991). Final decommission of the facility was included in the final site remediation, which started in 1998.

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

3.0 OCCUPATIONAL MEDICAL DOSE

The information in this section applies to the Apollo and Parks Township sites. NUMEC apparently did not have its own medical X-ray department during AEC operational years, and the medical X-rays for NUMEC employees appear to have been performed at a local clinic or hospital. Therefore, in compliance with ORAUT-OTIB-0079, *Guidance on Assigning Occupational X-Ray Dose Under EEOICPA for X-Rays Administered Off Site* (ORAUT 2016a), no occupational medical dose should be assigned.

4.0 ENVIRONMENTAL OCCUPATIONAL DOSE

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

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

4.1 ENVIRONMENTAL INTERNAL DOSE

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

4.2 ENVIRONMENTAL EXTERNAL DOSE

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

5.0 OCCUPATIONAL INTERNAL DOSE

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

5.1 INTERNAL EXPOSURE SOURCES

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

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

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

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

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

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

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_3O8 and UO_2 as type S. The chemical forms and the enrichments varied over time at the NUMEC sites. The manufacture of uranium products occurred in most of the buildings at Apollo and Parks Township. See Tables 2-1 and 2-3 for more information. The dose reconstructor should use the solubility type that results in the highest dose.

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

Table 5-2. Uranium source term information.

Source term	Specific activity (pCi/µg)	U-234 activity fraction	U-234 activity	U-235 activity fraction	U-235 activity	U-236 activity fraction	U-236 activity	U-238 activity fraction	U-238 activity
NU[a]	0.683	0.489	0.334	0.023	0.016	Negligible	Negligible	0.489	0.334
93%[a]	68.1	0.968	65.9	0.030	2.04	0.002	0.136	0.0003	0.020
3.5%[a]	2.20	0.818	1.80	0.034	0.075	Negligible	Negligible	0.147	0.323
2%[b]	1.20	0.648	0.778	0.041	0.049	0.0009	0.001	0.311	0.373
Typical DU[a]	0.402	0.155	0.062	0.011	0.004	0.0005	0.0002	0.834	0.335

a. Source: Integrated Modules for Bioassay Analysis (IMBA).
b. Source: American National Standards Institute N13.22 (HPS 2013).

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 found, 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. 328-329). Older materials, even when starting out as soluble, can have a tendency to oxidize when left in contact with air. Oxides, metals, and old contamination should be treated as type S. If nothing is known about the chemical form of plutonium, either type M or S can be used to maximize the dose to the organ of concern. In addition, because highly insoluble forms of plutonium might have been present, guidance in ORAUT-OTIB-0049, *Estimating Doses for Plutonium Strongly Retained in the Lung* (ORAUT 2010a), should be followed for the evaluation of highly insoluble (type SS) 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 2015). If the plutonium is type SS, follow guidance in ORAUT (2010a) for assignment of the ^{241}Am solubility type.

There are essentially three types of plutonium-based material: (1) reactor grade, (2) weapons grade, and (3) fuel grade, which falls between reactor and weapons grades.

Fuel-grade plutonium was used for FFTF fuel fabrication and most of the ZPPR fuel fabrication (11,500 plates). However, commercial reactor-grade plutonium was also used for some ZPPR fuel fabrication (700 fuel plates) starting in April of 1969. For work in the FFTF fuel fabrication, lacking specific information on the actual composition of the processed plutonium, the composition of Hanford fuel-grade plutonium can be used because this was the source of plutonium for FFTF fuel fabrication (Author unknown 2004). The site also used Hanford 6% weapons-grade plutonium for some mixed-oxide fuel fabrictiaon. An 8.5% plutonium mix was used for a contract with Japan's Power Reactor and Nuclear Fuel Development Corporation (PNC) and for ZPR-III fuel. The activity compositions for the four grades of plutonium are listed in Table 5-3 for fuel aged up to 20 years (ORAUT 2010b, 2015). The age of plutonium to assume for a given analysis depends on the radionuclide measured in the bioassay analysis. When gross alpha, ^{238}Pu, or ^{239}Pu is measured, the dose is maximized by

assuming longer decay times (20 years). When ^{241}Am is measured and the intake is estimated using ingrowth of ^{241}Am from decay of ^{241}Pu, the dose is maximized by assuming a short (5-year) decay time. If the actual age of the fuel is known (such as from an incident investigation report), that age can be used in the intake and dose analysis. If the exposure occurred only in FFTF fuel fabrication, then the Hanford fuel-grade composition can be used. If the work location is not known (such as for plutonium scrap recovery), then use of the Hanford 6% weapons-grade plutonium will provide a dose estimate that is favorable to the claimant.

Table 5-3. Activity composition of plutonium mixtures (Ci/g).[a]

Hanford 6% weapons-grade plutonium specific activity[b]

Nuclide	Aged 0 yr	Aged 5 yr	Aged 10 yr	Aged 15 yr	Aged 20 yr
Pu-238	8.56E–03	8.23E–03	7.91E–03	7.60E–03	7.31E–03
Pu-239	5.77E–02	5.77E–02	5.77E–02	5.77E–02	5.77E–02
Pu-240	1.36E–02	1.36E–02	1.36E–02	1.36E–02	1.36E–02
Pu-241	8.24E–01	6.48E–01	5.09E–01	4.00E–01	3.15E–01
Pu-242	1.97E–06	1.97E–06	1.97E–06	1.97E–06	1.97E–06
Am-241	0.00E+00	5.83E-03	1.04E–02	1.39E–02	1.66E–02

8.5% plutonium specific activity[c]

Nuclide	Aged 0 yr	Aged 5 yr	Aged 10 yr	Aged 15 yr	Aged 20 yr
Pu-238	1.22E–02	1.17E–02	1.12E–02	1.08E–02	1.04E–02
Pu-239	5.55E–02	5.55E–02	5.55E–02	5.55E–02	5.55E–02
Pu-240	1.94E–02	1.94E–02	1.94E–02	1.94E–02	1.94E–02
Pu-241	1.78E+00	1.40E+00	1.10E+00	8.64E–01	6.80E–01
Pu-242	2.80E–06	2.80E–06	2.80E–06	2.80E–06	2.80E–06
Am-241	0.00E+00	1.26E–02	2.25E–02	3.00E–02	3.58E–02

Hanford 12% fuel-grade plutonium specific activity[b]

Nuclide	Aged 0 yr	Aged 5 yr	Aged 10 yr	Aged 15 yr	Aged 20 yr
Pu-238	1.71E–02	1.64E–02	1.58E–02	1.52E–02	1.46E–02
Pu-239	5.26E–02	5.26E–02	5.26E–02	5.26E–02	5.26E–02
Pu-240	2.72E–02	2.72E–02	2.72E–02	2.72E–02	2.72E–02
Pu-241	3.09E+00	2.43E+00	1.91E+00	1.50E+00	1.18E+00
Pu-242	3.93E–06	3.93E–06	3.93E–06	3.93E–06	3.93E–06
Am-241	0.00E+00	2.19E–02	3.89E–02	5.22E–02	6.24E–02

Commercial 27% fuel-grade plutonium specific activity[b]

Nuclide	Aged 0 yr	Aged 5 yr	Aged 10 yr	Aged 15 yr	Aged 20 yr
Pu-238	1.71E-01	1.64E-01	1.58E-01	1.52E-01	1.46E-01
Pu-239	3.41E-02	3.41E-02	3.41E-02	3.41E-02	3.41E-02
Pu-240	5.90E-02	5.89E-02	5.89E-02	5.89E-02	5.89E-02
Pu-241	1.34E+01	1.05E+01	8.28E+00	6.51E+00	5.12E+00
Pu-242	1.97E-04	1.97E-04	1.97E-04	1.97E-04	1.97E-04
Am-241	0.00E+00	9.49E-02	1.69E-01	2.26E-01	2.79E-01

a. Aging refers to the time since separation of Am-241 from the plutonium mix.
b. Source: ORAUT (2010b).
c. Source: Gerrish (1965).

5.2 IN VITRO BIOASSAY

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

Several bioassay vendors were used to evaluate in vitro bioassay samples for the NUMEC sites. The SEC evaluations for the Apollo site and the Parks Township site 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 (NIOSH 2007a, 2008a). Bioassay data from Controls for Environmental Pollution should be used only to indicate the potential for exposure to a particular radionuclide on a particular date. The data cannot be used in a dose reconstruction to evaluate intakes or assign internal dose.

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

5.2.1 Plutonium Urine and Feces Bioassay

Plutonium might have been present at the NUMEC sites in several forms that include types M, S, and possibly SS material solubility categories. The intake analysis based on bioassay monitoring results should evaluate intakes based on all three types and use the type that provides the highest dose estimate. Several different bioassay vendors performed plutonium urine and fecal analyses as observed from worker dosimetry records.

5.2.1.1 Plutonium Urine Minimum Detectable Concentrations and Frequencies

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

5.2.1.2 Plutonium Urine Analytical Procedure

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

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

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

5.2.1.3 Plutonium Fecal Minimum Detectable Concentrations and Frequencies

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

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

Table 5-4. Plutonium and americium urine bioassay MDCs and frequencies by period.[a,b]

Date	Laboratory	Analyte	Frequency[b]	MDC[c,d]	Error[e]
10/1961–12/1965	Controls for Radiation	Plutonium	Quarterly/as needed	0.28 dpm/L	0.01–0.48 dpm/L
01/1966–12/1968	Eberline	Pu-238	Quarterly/as needed	0.06 dpm/sample	0.03 dpm/sample
01/1966–12/1975	Eberline	Pu-239	Quarterly/as needed	0.06 dpm/sample	0.03 dpm/sample
01/1966–12/1975	Eberline	Am-241	Quarterly/as needed	0.06 dpm/sample	0.03 dpm/sample
01/1976–04/1980	Controls for Environmental Pollution	Pu-238	As needed	(f)	(f)
01/1976–04/1980	Controls for Environmental Pollution	Pu-239	Quarterly/as needed	(f)	(f)
01/1976–04/1980	Controls for Environmental Pollution	Am-241	Quarterly/as needed	(f)	(f)
05/1980–09/1985	Controls for Environmental Pollution	Pu-238	As needed	(f)	(f)
05/1980–09/1985	Controls for Environmental Pollution	Pu-239	Quarterly/as needed	(f)	(f)
05/1980–09/1985	Controls for Environmental Pollution	Am-241	Quarterly/as needed	(f)	(f)
1999	Quanterra	Pu-238	Unknown	0.0025–0.044 pCi/L	Not applicable
1999	Quanterra	Pu-239/240	Unknown	0.0025–0.045 pCi/L	Not applicable
11/1974–08/1975	Eberline	Gross alpha (plutonium + americium)	Quarterly/as needed	<10.0 dpm/sample	Not applicable

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

NUMEC health physicist Roger Caldwell believed that fecal sampling was the only satisfactory method for estimating lung burdens for insoluble actinide alpha emitters classified as Y in the

contemporary lung model. The most important alpha emitters included $^{239}PuO_2$, $^{241}AmO_2$, $^{234}UO_2$, and $^{232}ThO_2$ (Caldwell 1966). Caldwell calculated that easily detectable plutonium quantities were excreted in feces: 49 dpm/d PuO_2 is eliminated from a 16-nCi lung burden, 32 dpm/d by way of the feces. One-tenth of this value or 3.2 dpm/d was believed to be a suitable reference level. Workers excreting safely below this level were assumed to have nonhazardous lung burdens (Caldwell 1966).

Table 5-5. Plutonium fecal bioassay MDCs and frequencies by period.[a,b]

Date	Laboratory	Analyte	Frequency[b]	MDC[c]	Error
01/1966–01/1977	Eberline	Pu-239, Pu-238 or Am-241	Quarterly/as needed	0.1 dpm/sample	0.05 dpm/sample
05/1975–09/1975	Eberline	Gross alpha (plutonium + americium)	As needed	0.1 dpm/sample	0.05 dpm/sample
02/1977–10/1985	Controls for Environmental Pollution	Pu-239, Pu-238 or Am-241	Quarterly/as needed	(d)	(d)

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

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

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

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

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

5.2.2 Uranium Urine and Feces Bioassay

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

5.2.2.1 Uranium Urine Analytical Procedure

Information in HASL-82 (AEC 1960a) indicates that before 1960 urine samples were obtained at the Apollo site on a monthly and bimonthly basis, with the commercial laboratory Nuclear Engineering and

Sciences Corporation performing the urine analysis. The results frequently included high values that ranged from 50 to 150 µg/L of uranium in urine. However, the available dosimetry records do not contain urine bioassay results from before late 1959.

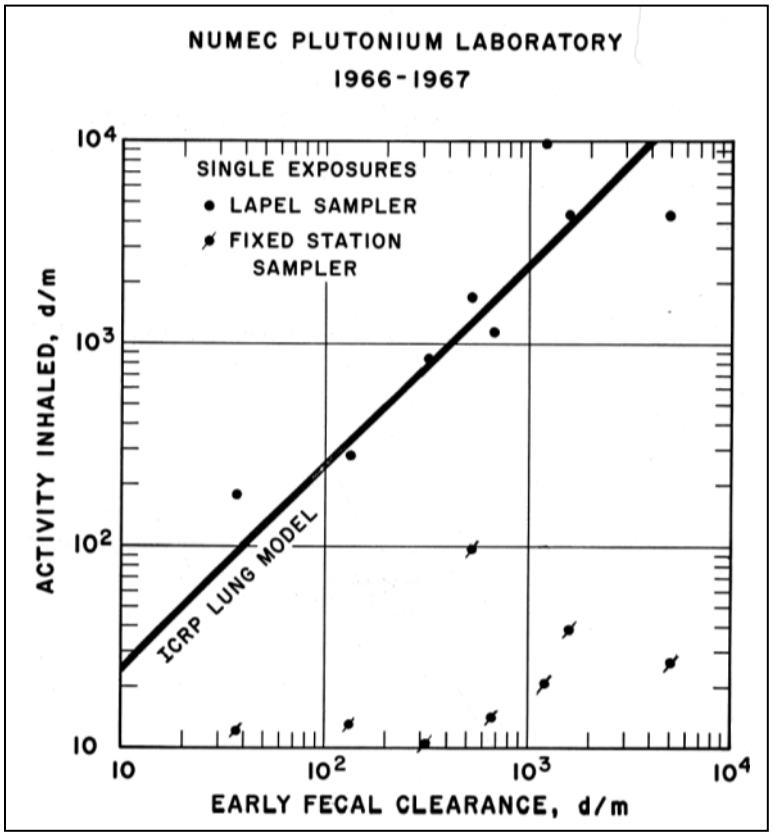

Figure 5-1. Correlation of fecal bioassay with air sampling (Caldwell, Potter, and Schnell 1967).

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

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

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

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

5.2.2.2 Urine MDCs and Frequencies

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

Table 5-6. Uranium urine bioassay MDCs and frequencies by method and period.[a,b]

Date	Laboratory	Analyte	Frequency[b]	MDC[c]	Error
03/1961–02/1966	Controls for Radiation	Total uranium	Quarterly/as needed	<1 µg/L[d]	Unknown
09/1972–12/1976	Eberline	Total uranium	Quarterly/as needed	<5 µg/sample	Unknown
01/1977–11/1987	Controls for Environmental Pollution	Total uranium	Unknown	(e)	(e)
1999	Quanterra	Total uranium	Quarterly/as needed	<0.006 µg/L	Unknown
04/1962–01/1967	Controls for Radiation	Gross alpha	Quarterly/as needed	26 dpm/L	13 dpm/L
02/1967–08/1972	Tracerlab	Gross alpha uranium	Quarterly/as needed	0.2 dpm/sample	0.1 dpm/sample
09/1972–01/1974	Eberline	Gross alpha uranium	Quarterly/as needed	<50.0 dpm/sample	0.05 dpm/mL
02/1974–04/1974	Eberline	Gross alpha uranium	Quarterly/as needed	<10 dpm/sample	Unknown
02/1974–12/1976	Eberline	Gross alpha uranium	Quarterly/as needed	2 dpm/sample	1 dpm/sample
03/1964–06/1967	Controls for Radiation	EU	Quarterly/as needed	4 dpm/L	2 dpm/L
07/1967–08/1972	Tracerlab	EU	Quarterly/as needed	0.2 dpm/sample	0.1 dpm/sample
01/1977–02/1987	Controls for Environmental Pollution	EU	Quarterly/as needed	(e)	(e)

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

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

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

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

- Wet analytical chemistry personnel every 3 months,
- Nonradiation workers annually,
- Production workers every 4 to 6 weeks maximum (NUMEC 1963),
- Maintenance personnel every quarter (NUMEC 1963),
- All other (radiation) personnel every 6 months (NUMEC 1963), and
- At the discretion of health and safety in the event of an incident such as a uranium hexafluoride release (NUMEC 1963).

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

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

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

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

5.2.2.3 Uranium Fecal Minimum Detectable Concentrations and Frequencies

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

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

Table 5-7. Uranium fecal bioassay MDCs and frequencies by period.[a]

Date	Laboratory	Radionuclide	Frequency[b]	MDC[c]
06/1967–06/1972	Tracerlab	Radiometric uranium	Quarterly/as needed	2 dpm/sample
07/1972–01/1976	Eberline	Total uranium	Quarterly/as needed	<5 µg/sample
02/1976–10/1985	Controls for Environmental Pollution	Total uranium	Quarterly/as needed	(d)

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

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

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

5.2.2.4 Uranium Fluorimetric Bioassay Evaluations

The evaluation of intakes based on uranium fluorimetric bioassays requires conversion from units of mass to activity. This requires information on the ^{235}U enrichment level because the specific activity of the uranium material varies with enrichment. If the enrichment is known for the exposed worker, then the specific activity for that enrichment should be used as defined in Table 5-2. If the enrichment is not known, then an estimate of activity that is favorable to the claimant can be made by assuming the material to be HEU (93%). This is a reasonable approach because HEU was used frequently at the Apollo facilities in the early years when urine bioassays were performed by fluorimetric analysis.

If the bioassay results contain both fluorimetric and radiometric results, then the radiometric results should be used because the radiometric analysis method provides a more sensitive estimate of uranium activity and no conversion of units is required.

5.2.3 Thorium Exposures

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

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

5.2.4 Mixed Fission Products

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

Exposure to FPs at the Apollo site was most likely to have occurred in the Laundry Building as part of the commercial decontamination of clothing by laundering. Exposure to FPs at the Parks Township site would most likely be related to source fabrication (^{60}Co and ^{137}Cs). The radionuclides representing mixed FPs could have included fission and activation products representative of reactor operations. Possible radionuclides include ^{60}Co, ^{90}Sr, ^{99}Tc [a recycled uranium (RU) contaminant], ^{137}Cs, ^{106}Ru/Rh, and possibly others. No information is available on the methods used to analyze mixed FPs in urine. Urine bioassay data for mixed FPs should be used, if included in the case files, to estimate intakes of FPs. The mixed FP urine bioassay results do not indicate if the measurements are based on beta or gamma analysis, so the intake should be based on ^{90}Sr as a representative beta emitter and ^{137}Cs as a representative gamma emitter and the analysis providing the highest dose should be used.

5.2.5 Unmonitored Radionuclides from Recycled Uranium

The uranium processed at the Apollo and Parks Township sites might have included RU. This material would contain contamination radionuclides formed during fission and activation processes when the material was irradiated in production or test reactors. The spent fuel elements were reprocessed to recover the uranium, which was returned to the DOE inventories along with trace contaminants that included ^{99}Tc, ^{237}Np, and ^{239}Pu. The intake of RU contaminant radionuclides can be estimated using the guidance in Section 5.3.2 and activity fractions in Tables 5-33 and 5-34 of ORAUT-TKBS-0017-5, *Technical Basis Document for the* Feed Materials Production Center – *Occupational Internal Dose* (ORAUT 2016b). The assignment of material solubility type is based on the guidance in Table 3-1 of ORAUT-OTIB-0060 (ORAUT 2014a). The determination of the form of uranium (types F, M, or S) most favorable to the claimant must include the recycled components for each uranium solubility type. Because plutonium is a minor contaminant in the RU matrix, the consideration of type SS is not necessary for evaluation of internal dose from this source of plutonium.

5.3 IN VIVO COUNTING

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

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

The procedure for lung counting used by the University of Pittsburgh Low Level Radioactivity Monitoring Facility included a standard stretcher technique that was used with two 5- by 3-in. NaI(Tl) dual-crystal low-energy detectors positioned above the stretcher near the anterior chest region of the subject (Boyd 2006g, pp. 92–97). The calibration was for 0.5 keV per channel, and the count time was 40 minutes for gross counts and background. Background correction was made using spectra 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 the Los Alamos National Laboratory using a detector configuration identical to the one used by the Laboratory. The calibration factor was adjusted for attenuation due to variation in the subject's chest wall thickness as measured with an encephaloscope. The evaluation of ^{239}Pu activity was based on the assumption that only ^{239}Pu was present and that all 17-keV X-rays were from ^{239}Pu. The difficulty in measuring the low-energy X-rays results in MDA values that represent significant lung burdens.

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

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

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

a. Based on U-235 MDA of 63 µg.

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

Table 5-9. In vivo MDAs for ^{239}Pu and ^{241}Am (nCi).[a]

Year	Pu-239 minimum	Pu-239 maximum	Pu-239 counts	Am-241 minimum	Am-241 maximum	Am-241 counts
1968	Not reported	Not reported	Not reported	0.13	0.38	17
1969	Not reported	Not reported	Not reported	0.16	0.16	1
1970	10	10	1	Not reported	Not reported	Not reported
1971	Not reported	Not reported	Not reported	Not reported	Not reported	Not reported
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	0.06	0.22	96
1975	4.8	19.9	133	0.11	0.21	104
1976	5.0	20.3	109	0.11	0.19	91
1977	4.4	19.6	113	0.09	0.19	88
1978	4.7	19.0	132	0.10	0.19	100
1979	5.16	24.3	168	0.08	0.26	132
1980	5.03	28.2	132	0.09	0.21	94
1981	7.21	27.8	55	0.12	0.20	31
1982	7.12	34.3	77	0.12	0.21	44
1983	9.41	15.6	6	0.12	0.16	4
1984	8.67	22.32	9	0.12	0.15	5
1985	8.84	31.07	31	0.11	0.22	29

a. From a review of worker dosimetry records (Boyd 2006a, 2006c, 2006d, 2006g, 2006h, 2006i). Values for 1968 through 1971 are based on the Helgeson system with remaining values from the University of Pittsburgh system.

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. If values are not provided in the records, then the maximum value from Table 5-9 should be used (nominally 35 nCi for ^{239}Pu and 0.40 nCi for ^{241}Am).

5.4 REPORTS OF OVEREXPOSURES AND INCIDENTS

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

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

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

5.4.1 Apollo Site Incidents

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

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

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

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

5.4.2 Parks Township Site Incidents

Incidents occurred at the Parks Township site and might have contributed to employee exposures. Information about individual involvement in incidents is likely to be in worker dosimetry records and should be consulted for evaluation of intakes of radionuclides during such events.

5.5 URANIUM AND PLUTONIUM AIR SAMPLING PROGRAMS

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

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

According to the 1963 NUMEC Health and Safety Manual, average or weighted airborne exposure studies were performed on every new operation and repeat studies were made on old operations on a

frequent basis (NUMEC 1963). According to a health protection program review in 1963, 75 short-term BZA samples were obtained in Apollo process buildings every week but no routine GA air samples were taken that would indicate an average air concentration over an 8- to 24-hour period (Hervin and Pryor 1963).

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

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

6.0 OCCUPATIONAL EXTERNAL DOSE

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

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

6.1 EXTERNAL EXPOSURE SOURCES

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

Parks Township site operations began in about 1959; DOE operations ended in 1980. The initial function of the Parks Township facilities was fabrication of plutonium fuel, preparation of HEU fuel, and production of zirconium-hafnium bars under AEC/NRC License SNM-414, received March 1961, which allowed the handling of plutonium already on the site. The Parks Township site made fuel for the FFTF that consisted of a mixture of PuO_2 and depleted UO_2. It also made fuel plates for the ZPPR in the late 1960s and ZPPR-III fuel wafers. Activities included plutonium scrap recovery, DU

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

6.2 WORKPLACE RADIATION FIELDS

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

Table 6-1. Workplace potential exposures.

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

6.2.1 Beta Radiation

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

6.2.2 Photon Radiation

Photon radiation, typically of lower energy, is characteristic of plutonium operations. Thorium emits significant higher energy photon radiation. Uranium has comparatively less significant photon radiation with dose rates of about 1.2 mrad/hr in contact with fresh yellowcake. However, during the buildup of the ^{234}Th and ^{234}Pa progeny in fresh yellowcake, the radiation levels increased somewhat for several months after yellowcake production. Photon exposure rates were estimated to be about 4 mrad/hr at 30 cm from a drum of aged yellowcake (NIOSH 2014, p. 27).

6.2.3 Neutron Exposures

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

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

6.3 DOSIMETER TECHNOLOGY

NUMEC historically used beta/photon and neutron dosimeters to measure potential WB beta/photon, WB neutron, and extremity beta/photon exposures to personnel. A summary of the NUMEC dosimetry systems and periods of use is presented in Table 6-2. A description of the dosimetry report types that might be included in the NUMEC dosimetry records is given in Attachment B.

6.3.1 Beta/Gamma Dosimeters

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

6.3.2 Neutron Dosimeters

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

6.3.3 Limits of Detection

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

assigned in relation to the dosimetry service providers, periods of use, dosimeter exchange frequencies, and estimated MDLs.

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

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

Beta/Photon Dosimeters – WB

Period	Monitoring technique	Dosimeter description
1957–05/1968	Photographic film badge provided by Nuclear Science & Engineering, Controls for Radiation, Eberline, or Landauer	Film badges contained single film packet. Three filters (front and back) were incorporated into film badge for energy dependence: cadmium, aluminum, and lead. Beta and photon radiation capabilities were similar to other dosimetry systems at that time as presented in the 1954 AEC dosimeter performance study (AEC 1955).
06/1968–1975	NUMEC or Eberline TLD	Comprised of two TLD-700 chips, two TLD-600 chips, and one CaF_2 for monitoring beta, X-ray, and gamma exposure.
1976–present	Landauer or equivalent TLD (Z1 dosimeter -1990)	Comprised of three TLD-700 chips for monitoring beta, X-ray, and gamma exposure. Insensitive to neutron radiation.

Beta/Photon Dosimeters – Wrist and Ring

Period	Monitoring technique	Dosimeter description
07/1963–ca. 05/1968	Landauer (type M – wrist beta-gamma) film badges or equivalent	Film dosimeter known as type M responsive to beta and gamma radiation.
Ca. 06/1968–1983	TLD wrist badge	Comprised of three TLD-100 chips.
07/1991–12/1991	Teledyne Isotopes TLD Badge	TLD badge for monitoring beta and gamma exposure (BWXT 1991). Details of the dosimeter are not available, other than detection limits.

Neutron Dosimeters – WB

Period	Monitoring technique	Dosimeter description
1957–05/1968	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 1,000-power microscope or projection capability.
06/1968–1995	Landauer Neutrak Extended Range dosimeter (types I8, I1, or RI)	Combined TLD albedo neutron monitor with track recoil device [CR-39 (allyl diglycol carbonate)] that responds to neutron radiation through proton recoil events. The dosimeter is responsive to a neutron energy range of about 0.0001 to 10 MeV. Dosimeter response to thermal neutron radiation was subtracted to yield fast neutron dose. The Neutrak ER has an albedo element with above-described elements. Qualitative relationship was derived to determine ratios of neutrons of various energies. The RI badge was capable of monitoring beta, X-ray, gamma, and neutrons.
07/1991–12/1991	Teledyne Isotopes TLD badge	Combined gamma, beta, and neutron TLD (BWXT 1991). Details of the dosimeter are not available, other than detection limits.

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

Vendor or processor/area monitored	Period of use	LOD[a] (rem)	Annual missed dose[b] (rem) (frequency in parenthesis)
Nuclear Science & Engineering or Controls for Radiation film and NTA film	1957–1963	0.03 photons[c] 0.03 beta[c9] 0.05 neutrons	0.18 beta & photons (monthly) 0.30 neutrons [fast] (monthly)
Landauer film and NTA film	1964–1965	0.03 photons[c] 0.03 beta[c] 0.05 neutrons	0.18 beta & photons (monthly) 0.30 neutrons [fast] (monthly)
Eberline film and NTA film	1966–06/1968	0.03 photons[c] 0.03 beta[c] 0.05 neutrons	0.18 beta & photons (monthly) 0.30 neutrons [fast] (monthly)
NUMEC or Eberline (Film-Apollo) until about 1970 and TLD for neutron	07/1968–1975	0.02 photons[d] 0.02 beta[d] 0.05 neutrons	0.12 beta & photons (monthly) 0.30 neutrons (monthly)
Landauer TLD and Teledyne Isotopes	1976–1995	0.01 photons[e] 0.01 beta[e] 0.02 neutrons	0.06 beta & photons (monthly) 0.12 neutrons (monthly)

a. Estimated LODs for each dosimetry technology. Dose levels were recorded at values less than the MDLs.
b. Annual missed dose calculated based on the MDL/2 method from NIOSH (2007b).
c. LODs for photons and beta during these periods are based on Hanford dosimeter values (ORAUT 2010b).
d. LOD during this period is probably twice the recording level of 0.010 rem.
e. Landauer LOD values from Boyd (2006j) and Teledyne Isotopes LOD values from BWXT (1991).

6.3.4 Radiological Records

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

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

Table 6-4. Annual occupational radiation exposures at the Apollo site (numbers of individuals with WB doses in each range (Breuer 1977, 1978).

Year	Total number monitored	Number with measured dose	<0.1 rem	0.1–0.25 rem	0.25–0.5 rem	0.5–0.75 rem	0.74–1.0 rem	1.0–12 rem	>12.0 rem
1976	42	42	27	14	1	0	0	0	0
1977	39	39	15	16	6	0	2	0	0

6.4 LIMITATIONS IN MEASURED DOSE

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

6.4.1 Low-Energy Photons

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

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

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

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

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

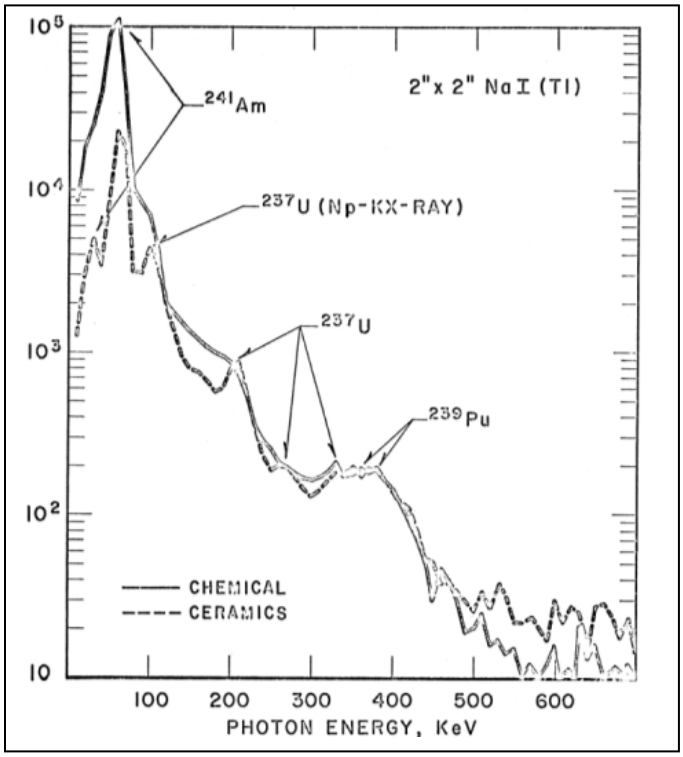

Figure 6-1. Gamma spectrum external to plutonium chemical processing line and plutonium ceramics line (Caldwell and Judd 1966, p. 14).

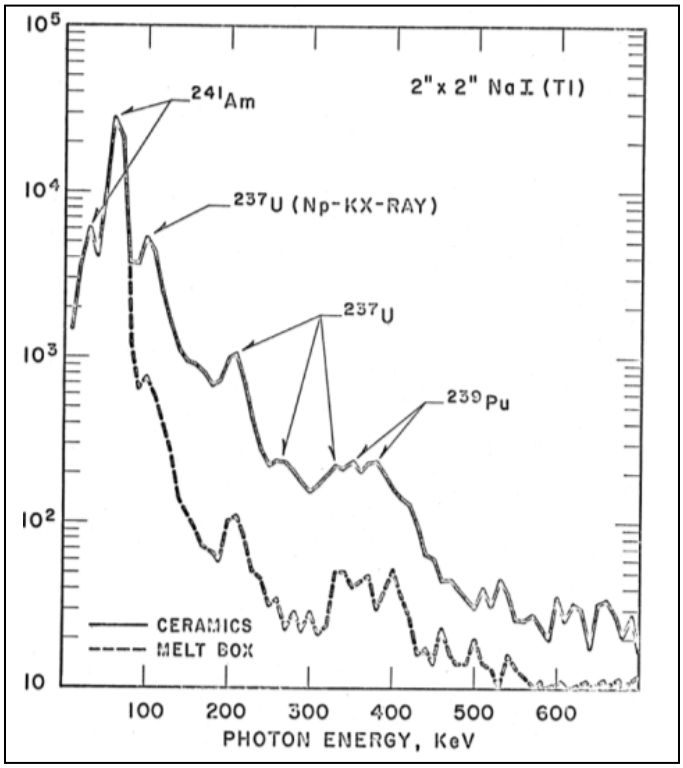

Figure 6-2. Gamma spectrum external to plutonium ceramics line and plutonium melt glovebox (Caldwell and Judd 1966, p. 15).

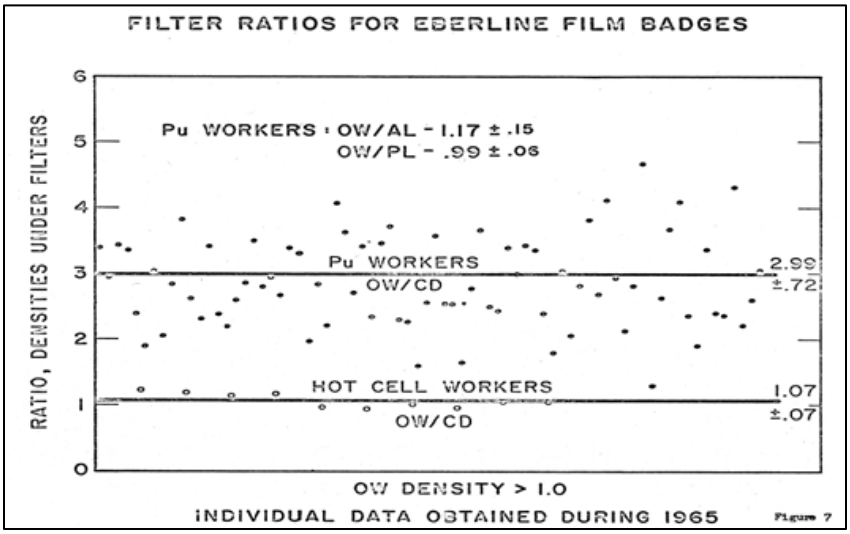

Figure 6-3. Dosimeter filter ratios (Caldwell and Judd 1966, p. 18).

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

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

2. Plutonium workers were exposed to a wide range of gamma energies. The upper end of the spectrum would produce OW:CD ratios close to 1.0. The lower gamma energies would produce an OW:CD ratio of 3.0 or greater. One plutonium worker might be exposed to an entirely different effective energy than another due to shielding, working distance, and other geometry factors. If the OW:CD ratio was less than 2.0, the reported dose was accepted. If the OW:CD ratio was greater than 2.0, NUMEC would use a plutonium spectrum calibration curve that represented a typical plutonium gamma spectrum.

3. The OW:PL ratio was about 1.0 and was within a 6% standard deviation. This meant that the large OW:CD ratio was not due to beta radiation.

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

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

6.4.2 Neutron Radiation

NTA film has a characteristic decreasing response to neutron radiation at energies below about 500 to 800 keV, depending on the extent of photon fogging and the overall process to develop and read the tracks (ORAUT 2014c). However, at this time, the neutron dosimeter readings should be used without correction for this effect (see also section 6.5.2.4).

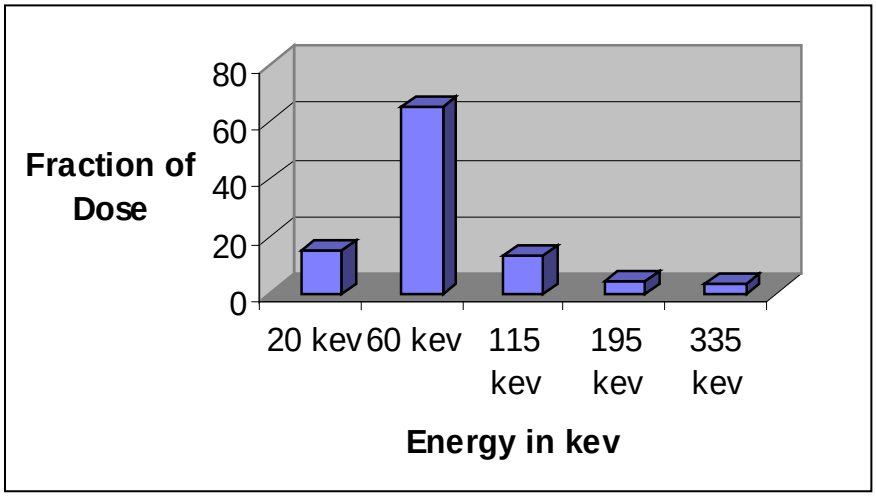

Figure 6-4. Relative contribution by energy group to personnel exposure during plutonium fuel fabrication from 1,000 MWd/t plutonium (Caldwell and Judd 1966, p. 19).

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

Energy	NUMEC plutonium (^{241}Am and ^{233}U)	Hanford plutonium
Photon <30 keV	15	25
Photon 30–250 keV	82	75
Photon >250 keV	3	0
Beta >15 keV	100	100

6.5 DOSE RECONSTRUCTION RECOMMENDATIONS

6.5.1 Recorded Dose Practices

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

Table 6-6. Recorded dose practices.

Dosimeter type	Period	Dosimeter measured quantities	Compliance dose quantities
Photon/electron film dosimeter + NTA neutron dosimeter	1957–1968	Gamma (G) Neutron (N) Beta (B)	WB or total = gamma (photon) + neutron Beta separate Extremity = gamma (+ neutron)
Photon/electron film dosimeter + TLD neutron dosimeter	1968–1983	Deep = gamma (DBG) and neutron (N) Shallow beta gamma (SBG)	WB = gamma + neutron Skin = beta Extremity = gamma + neutron
Photon/electron/neutron Panasonic TLD + CR-39 neutron dosimeter	1983–present	Deep Shallow	Skin = beta + soft gamma and neutron WB = photon + neutron Extremity = gamma + neutron

6.5.2 Adjustments to Recorded Dose

6.5.2.1 Beta Dose Adjustments

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

Table 6-7. Interpretation of reported data.

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

The guidance from ORAUT-OTIB-0017, *Interpretation of Dosimetry Data for Assignment of Shallow Dose* (ORAUT2005c) is as follows:

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

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

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

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

6.5.2.2 Photon Dose Adjustments

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

6.5.2.3 Neutron Weighting Factor Adjustments

Recorded NUMEC neutron doses are assumed to have been based on quality factors in National Council on Radiation Protection and Measurements Report 38 (NCRP 1971). The quality factors in Report 38 were compared with the neutron weighting factors in ICRP Publication 60 (ICRP 1991) to arrive at factors to convert the recorded dose to equivalent Publication 60 neutron doses as required by OCAS-IG-001 (NIOSH 2007b). A dose multiplier of 1.91 should be used for the 0.1- to 2-MeV energy range (ORAUT 2014c). This range includes 100% for HEU, EU, NU, and plutonium work

locations. The recorded neutron doses during fuel work and neutron doses estimated using the neutron-to-photon ratios should, therefore, be multiplied by a factor of 1.91 (for ICRP Publication 60 correction).

6.5.2.4 Neutron Dose During NTA Film Period

NUMEC used NTA film to monitor for neutron exposure until approximately June 1968 when TLDs were introduced at the site. The NTA film dosimeters had a reduced response to neutrons at energies below about 500 to 800 keV (ORAUT 2014a). The missed neutron dose due to the energy threshold effect is accounted for by use of neutron-to-photon ratios.

A study was performed in 1975 (BWTX 1975) using TLDs to determine the photon and neutron doses while working in the FFTF fuel fabrication area of the Parks Township Plutonium Facility. In addition, dosimeters were placed at various fixed locations. The results of this study indicated the neutron-to-photon ratio for plutonium workers had a geometric mean of 0.34 and a GSD of 1.71. This ratio is supported in the Parks Township Plutonium Facility 1979 to 1981 Health and Safety reports (BWTX 1981; 1982a), which indicate that neutron-to-photon ratios varied from 0.23 to 0.42 with an average of 0.33. The reported glovebox ratios had a geometric mean of 1.00 and a GSD of 1.49.

The difference between the ratios is likely a result of shielding on the gloveboxes that would have reduced the photon component reaching the dosimeter.

Further, information was obtained from a September 1969 event in which a worker involved with manufacturing neutron sources had a neutron-to-photon ratio of 2.33 (determined using estimated neutron dose values) (Caldwell 1968d).

Based on this information, NUMEC workers with neutron dosimetry based on NTA film dosimeters (i.e., through June 1968) should have neutron dose assigned using the ratio most appropriate for their work and job location (i.e., use a ratio of 0.34 for typical workers, ratio of 1.00 for glovebox workers, and ratio of 2.33 for workers involved with manufacturing neutron sources). This dose is assigned using a lognormal distribution with the GSD provided, except for the manufacturing neutron source ratio, which should be applied as a constant. If the worker's recorded neutron dose is higher than the neutron dose calculated from applying the ratio to the photon dose, the recorded neutron dose should be assigned.

6.5.3 Missed and Unmonitored Dose

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

6.5.3.1 Shallow Dose and Deep Dose

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

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

6.5.3.2 Neutron Dose

The potential missed neutron dose can be estimated from LOD values in Table 6-3 for monitored workers using the same approach. If the neutron-to-photon method is used to evaluate the neutron dose per guidance in Section 6.5.2.4, then use the neutron-to-photon ratio to assign missed neutron dose. When measured NTA film dose is used per guidance in Section 6.5.2.4, then use the LOD method to assign missed neutron. If monitoring records do not include neutron dosimetry information, then neutron dose should not be applied. For workers who were likely to be exposed to neutrons, and for whom no neutron dosimetry is available, a partial dose reconstruction would result. Exposure to uranium hexafluoride cylinders is a possible source of neutron exposure at the Apollo site.

6.5.4 Uncertainty

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

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

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

6.5.5 Radiation Dose Fraction

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

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

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

Description	Begin	End	Radiation type	Energy selection	Percent
Uranium facilities	01/01/1957	12/31/1983	Beta	>15 keV	100
Uranium facilities	01/01/1957	12/31/1983	Photon	30–250 keV	100
Plutonium facilities	01/01/1959	12/31/1980	Photon	30–250 keV	100
Plutonium facilities	01/01/1959	12/31/1980	Neutron	0.1–2 MeV	100
Thorium handling	01/01/1957	12/31/1983	Beta	>15 keV	100
Thorium handling	01/01/1957	12/31/1983	Photon	30–250 keV	25
Thorium handling	01/01/1957	12/31/1983	Photon	>250 keV	75
Photon and neutron sources	01/01/1957	12/31/1983	Beta	>15 keV	100
Photon and neutron sources	01/01/1957	12/31/1983	Photon	30–250 keV	50
Photon and neutron sources	01/01/1957	12/31/1983	Photon	>250 keV	50
Photon and neutron sources	01/01/1957	12/31/1983	Neutron	0.1–2 MeV	100

7.0 ESTIMATION OF EXPOSURE TO RESIDUAL ACTIVITY

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

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

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

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

7.1 EXTERNAL DOSE FROM RESIDUAL ACTIVITY IN THE WORKPLACE

7.1.1 Apollo Site

Because all operations at the Apollo site had stopped by 1983, the only exposures would have been from residual activity. The potential external dose during the residual period has been estimated based on the mean surface concentration at the end of the operational period as described in Section 7.4, derived in support of internal dose estimates during the residual period. The analysis resulted in a mean surface concentration for uranium of 6.39×10^5 dpm/m^2 (GSD = 6.95) and a mean surface concentration of thorium and progeny of 1.54×10^5 dpm/m^2 (GSD = 5.0). These values 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 DCF for exposure to uniform activity on a ground plane:

Dose (rem/yr) = residual level (dpm/m^2) × DCF [(rem/hr)/(dpm/m^2)] × exposure time (hr/yr) (7-1)

The external DCF for exposure to isotopes of uranium, thorium, and short-lived progeny is provided in Federal Guidance Report 12 (Eckerman and Ryman 1993). The median annual external dose from exposure to residual surface contamination is listed in Table 7-1 for WB exposure and skin exposure. Values are presented for uranium activity, thorium activity, and the total from both sources. These values are based on an exposure time of 2,000 hr/yr. The dose to a specific organ can be obtained by multiplying the WB exposure value by the isotropic exposure DCF for the organ of interest. The listed WB dose values are based on the effective WB dose from Federal Guidance Report 12 using ICRP Publication 60 weighting factors (ICRP 1991). The values have been adjusted to ensure that application of the exposure isotropic dose factors results in an organ dose that is not less than the Federal Guidance Report 12 value for the organ.

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

Organ	Uranium annual dose[a]	Thorium annual dose	Total annual dose
Whole body	0.0005	0.0032	0.004
Skin	0.0354	0.0088	0.044

a. Values are input into IREP as a lognormal distribution; value listed is the geometric mean; the GSD is 5.0.

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

The skin dose value represents the dose at 1 m above the ground. This provides an overestimate of dose to the skin for cancers above the waist and an underestimate of dose for cancers below the waist. For dose to shallow organs (i.e., breast, testes, and penis), the skin dose values should be assigned with a correction factor of 0.30 (ORAUT 2005c). The shallow dose to the lens of the eye should be based on the skin dose values assigned with a correction factor of 0.33 (ORAUT 2010d).

The method for assignment of external dose from residual materials at the Apollo site is as indicated in Table 7-2. The following is based on guidance in the technical information bulletin for shallow dose (ORAUT 2005c) and the procedure for occupational onsite external dose (ORAUT 2006). This method includes application of the exposure isotropic DCFs and results in a best estimate of dose.

Table 7-2. Assignment of external onsite residual dose.

Organ(s)	Dose assignment
All except skin, breast, testes, and penis	Listed WB dose times organ isotropic exposure DCF
Skin	Listed skin dose
Breast, testes, and penis	Listed WB dose times organ isotropic exposure DCF
Breast, testes, and penis	30% of listed skin dose
Lens of eye	33% of listed skin dose

The dose should be entered into the IREP input as a lognormal distribution with a GSD of 5.0 (IREP Parameter 2) as photons of energy from 30 to 250 keV. This provides a favorable estimate of POC for all organs, even though some of the photon energy is likely to be of higher energy. Although much of the dose to the skin is from electrons, assignment of the dose as photons provides a result that is favorable to claimants.

7.1.2 Parks Township Site

Because the residual period for the Parks Township site includes the period from 1981 through 1983, when the Apollo site was still in operation, it is possible that Parks Township workers were exposed to DOE work if they visited the Apollo site. If dosimeter readings are available for these years of the residual period, the dose should be based on the recorded and missed dose (Section 6.0) unless it is

known that the worker did not visit the Apollo site. There might be external dosimetry data for later years of the residual period, but operations then were not part of the nuclear weapons-related program and are not covered for dose reconstruction under EEOICPA (Leiton 2011).

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

An estimate of the external dose from residual plutonium activity at the Parks Township site has been made using the same approach as for the Apollo site uranium external dose. The mean surface concentration at the end of the operational period as described in Section 7.4, derived in support of internal dose estimates during the residual period, was 2.26×10^4 dpm/m^2 (GSD = 4.97). This value can be used to estimate the annual external dose to workers exposed to the residual activity. The annual dose is evaluated using Equation 7-1 and DCFs for exposure to uniform activity on a ground plane. The annual dose was evaluated using the Hanford fuel grade plutonium isotopic compositions in Table 7-6 as a function of material age. The highest dose was obtained for the 20-year-aged material for all organs. However, all doses were less than 0.001 rem. Therefore, no external dose need be assigned for the residual period for Parks Township workers based on potential exposure to plutonium for Building A.

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

7.2 EXTERNAL AMBIENT DOSE FROM RESIDUAL ACTIVITY

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

7.3 OCCUPATIONAL MEDICAL DOSE

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

7.4 INTERNAL DOSE FROM RESIDUAL ACTIVITY IN THE WORKPLACE

7.4.1 Apollo Site

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

ORAUT-OTIB-0070, *Guidance in Dose Reconstruction During Residual Radioactivity Periods at Atomic Weapons Employer Facilities* (ORAUT 2012b), describes methods to estimate intake of

radionuclides during residual periods. The method relates residual surface contamination to workplace air concentration during periods after completion of DOE work.

The majority of the DOE-related work at the Apollo site involved scrap recovery and fuel fabrication operations in the Apollo uranium facility (East Bay of the Main Building). The HASL reports documented the average workplace air concentration to which workers were exposed during 1960 and 1961 (AEC 1960a, 1960b, 1960c, 1961a, 1961b). The results of these studies are described in Section 5.0.

The mean average daily air concentration can be used to estimate the residual surface concentration using guidance from ORAUT (2012b). The mean average air concentration is based on the long-term average using GA monitoring results for the site. The most appropriate data is that taken using continuous air monitors that include all variations in air concentration over the year. Such data was found for uranium for the year 1967 for several of the operations at the Apollo uranium facility. Limited data was found for 1966 for the Hammer Mill and the ceramics area, and for a few periods in 1962 and 1963 for the CRP-3 beryllium glovebox area and process area. The results from these measurements and the HASL reports indicate an air concentration value that is favorable to claimants to use for the residual period is represented by the maximum median value of 329 dpm/m^3. The maximum estimated GSD for all datasets is 6.95 for the HASL GA data. Therefore, the residual activity is based on a lognormal distribution with a median of 329 dpm/m^3 and a GSD of 6.95. The deposition amount is estimated using a deposition velocity of 0.00075 m/s with deposition assumed to occur for 30 days (NIOSH 2011). Using this approach, a surface concentration of uranium is estimated as follows:

$$329\ \text{dpm/m}^3 \times 2{,}592{,}000\ \text{s/30 d} \times 0.00075\ \text{m/s} = 6.39 \times 10^5\ \text{dpm/m}^2 \qquad (7\text{-}2)$$

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

The air concentration for each year is estimated using a resuspension factor (ORAUT 2012b) of 1×10^{-5}/m. Application of this resuspension factor and the above-described depletion factors to the residual contamination level of 6.39×10^5 dpm/m^2 results in the air concentration and annual intakes in Table 7-3. The intake evaluation is based on exposure for 2,000 hr/yr and an inhalation rate of 1.2 m^3/hr.

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

Year	Air concentration (dpm/m^3)	Inhalation Intake (dpm/yr)	Ingestion Intake (dpm/yr)[a]
1984	6.39	1.53E+04	16,400
1985	5.00	1.20E+04	12,900
1986	3.92	9.40E+03	10,100
1987	3.07	7.36E+03	7,900
1988	2.40	5.77E+03	6,200
1989	1.88	4.51E+03	4,800
1990	1.48	3.54E+03	3,800
1991	1.16	2.78E+03	3,000
1992	0.90	2.16E+03	2,300
1993	0.71	1.70E+03	1,800
1994	0.55	1.33E+03	1,430
1995	0.43	1.04E+03	1,120

a. Doses are assigned as a lognormal distribution with a GSD of 6.95.

The intakes in Table 7-3 can be used to estimate the internal dose to the target organ for the years of employment for the worker. The estimated internal doses are assigned as a lognormal distribution with a GSD of 6.95, corresponding to the distribution of the 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 (F, M, and S), and the dose from the highest type should be used in the IREP input.

Although uranium-aluminum alloy was present at the Apollo scrap recovery facility, the form of the material was not likely to be an inhalation hazard and modeling an intake of uranium aluminide is not necessary for the Apollo site. The internal dose analysis should include the potential inadvertent ingestion of uranium activity based on guidance in OCAS-TIB-009, *Estimation of Ingestion Intakes* (NIOSH 2004).

The daily intake rate (dpm/d) is estimated to be 0.2 times the average daily air concentration (dpm/m^3). Using the air concentration at the end of the operating period of 329 dpm/m^3, an intake rate of 16,400 dpm/yr is obtained for 250 workdays per year for the first year of the residual period (1984). This intake rate is reduced for subsequent years by the depletion factors listed in Table 4-2 of ORAUT (2012b). This provides an assessment of ingestion intake that is favorable to claimants.

Similar data was found for thorium operations in 1964 and 1965, which were also conducted in the Apollo uranium facility. The GA samples were not continuous, but were taken for periods from a few minutes to about an hour in areas where work was being conducted with thorium. The 71 data points indicated a median thorium concentration of 112 dpm/m^3, a mean of 160 dpm/m^3, and a GSD of 2.4. The samples were counted immediately at the end of the sample period and also after several hours to allow for decay of natural atmospheric radon and progeny. The results used in this analysis were those taken after decay because they represent the alpha activity from long-lived radionuclides. Based on this data, residual activity for thorium can be evaluated using a lognormal distribution of 112 dpm/m^3 and a GSD of 5.0 based on recommendations of Davis and Strom (2008). The higher GSD is used as it includes uncertainties associated with measurement errors and other errors not represented by the variability in the numerical values alone. The thorium at Apollo was received as thorium oxide that had undergone only one separation. The measured activity was assumed to be ^{232}Th with the minimum ^{228}Th following separation, an assumption favorable to claimants. This results in an initial activity of 79 dpm/m^3 ^{232}Th. Because exposures in the residual period are about 20 to 30 years after the initial processing of thorium at NUMEC, the thorium is assumed to be in equilibrium with all progeny (ORAUT 2014b). The activity of thorium and each progeny isotope (after decay to the start of the residual period) is assumed to be 79 dpm/m^3. The activity of ^{232}Th is assigned to the

progeny ^{228}Th, ^{228}Ra, ^{228}Ac, ^{224}Ra, and ^{212}Pb (radon is assumed to remain in the thorium matrix so ^{212}Pb is present). The deposition amount is estimated using a deposition velocity of 0.00075 m/s with deposition assumed to occur for 30 days (NIOSH 2011). Using this approach, a surface concentration of ^{232}Th is estimated as follows:

$$79 \text{ dpm/m}^3 \times 2{,}592{,}000 \text{ s/30 d} \times 0.00075 \text{ m/s} = 1.54 \times 10^5 \text{ dpm/m}^2 \quad (7\text{-}3)$$

This mean surface concentration is described as a level at the end of the operating period that is favorable to claimants.

The thorium air concentration for each year is estimated using a resuspension factor (ORAUT 2012b) of 1×10^{-5}/m as discussed above for uranium. Application of this resuspension factor and the above-described depletion factors to the residual thorium contamination level of 1.32×10^6 dpm/m^2 results in the air concentration and annual intakes in Table 7-4. The intake evaluation is based on exposure for 2,000 hr/yr and an inhalation rate of 1.2 m^3/hr. The potential ingestion intake as also included in the table. The intake activities in the table are assigned to ^{232}Th, ^{228}Th, ^{228}Ra, ^{228}Ac, ^{224}Ra, and ^{212}Pb.

Table 7-4. Thorium air concentration and annual intake in the Apollo site residual period.

Year	Air concentration (dpm/m^3)	Inhalation Intake (dpm/yr)	Ingestion Intake (dpm/yr)[a]
1984	1.54	3.69E+03	4,000
1985	1.20	2.89E+03	3,100
1986	0.94	2.26E+03	2,400
1987	0.74	1.77E+03	1,900
1988	0.58	1.39E+03	1,500
1989	0.45	1.08E+03	1,200
1990	0.35	8.52E+02	900
1991	0.28	6.67E+02	700
1992	0.22	5.20E+02	600
1993	0.17	4.09E+02	400
1994	0.13	3.20E+02	340
1995	0.10	2.50E+02	270

a. Doses are assigned as a lognormal distribution with a GSD of 5.0.

7.4.2 Parks Township Site

The following information provides a method for estimating exposures during the residual radiation period due to plutonium and uranium contamination. Valid bioassay data for plutonium are unlikely to be available during the residual period. However, monitoring data can be used instead of the default intake assumptions given below to limit dose, as appropriate (NIOSH 2008b). Bioassay monitoring data for uranium might be available for the first 3 years of the Parks Township residual period (1981 through 1983). If the worker might have worked at the Apollo site during this period, the data could relate to DOE work and should be valid for use in assignment of internal dose. For workers who were known to have only worked at the Parks Township site during this period, the bioassay data should only be used to limit internal dose.

Guidance in ORAUT-OTIB-0070 describes methods to estimate intake of radionuclides during residual periods (ORAUT 2012b). The method relates residual surface contamination to workplace air concentration during periods after completion of DOE work.

The majority of the DOE-related work at the Parks Township site involved plutonium fuel fabrication. While no formal air monitoring studies are available for these activities, an assessment of bounding

general area air activity in the facilities can be made through a review of available air sampling data. Starting in October 1967, NUMEC reported personnel exposures above regulatory thresholds to the AEC (Caldwell 1967b). A bounding representation of air activity at the Parks Township site was determined based on a review of these reports. The dataset includes 105 reported values between 1966 and 1982. The reported values were given as MPC-hours (for plutonium exposure), which were converted to dpm/m^3 by multiplying by the MPC and dividing by the number of hours given for the reported value. Values reported at the MPC were set to the MPC air concentration. The resultant median air concentration was estimated to be 11.6 dpm/m^3 with a GSD of 4.97 (assuming a lognormal distribution). The air concentration values and the lognormal fit of the data are shown in Figure 7-1.

The median plutonium air concentration from the above-described study and the guidance from ORAUT (2012b) were used to estimate the residual surface concentration. The deposition amount was estimated using a deposition velocity of 0.00075 m/s, with deposition assumed to occur for 30 days (NIOSH 2011). Using this approach, a surface concentration of plutonium was estimated as follows:

$$11.6 \text{ dpm/m}^3 \times 2{,}592{,}000 \text{ s/30 d} \times 0.00075 \text{ m/s} = 2.26 \times 10^4 \text{ dpm/m}^2 \quad (7\text{-}4)$$

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

The air concentration for each year was estimated using a resuspension factor of 1×10^{-6}/m (ORAUT 2012b). Application of this resuspension factor and the above-described depletion factors to the residual contamination level of 2.26×10^4 dpm/m^2 resulted in the air concentration and annual intakes in Table 7-5. The intake evaluation was based on exposure for 2,000 hr/yr and an inhalation rate of 1.2 m^3/hr.

Because the plutonium is based on gross alpha GA monitoring results, the activity represents the total alpha activity. The dose from the residual activity should be evaluated using the fractional isotopic alpha activity of each radionuclide given in Table 7-6. Activity fractions for this conversion are given in Table 7-6 for the plutonium used for ZPR-III and PNC, the Hanford fuel-grade plutonium used for FFTF and ZPPR fuel fabrication, and for commercial reactor-grade plutonium used for some of the ZPPR fuel fabrication. The total alpha activity values in Table 7-5 are to be multiplied by the activity fractions in Table 7-6 to determine the intake of each radionuclide. The age of the material must also be considered because the maximum dose (per unit alpha activity) is obtained for the less aged material. The beginning of plutonium processing at the NUMEC plutonium facility was in 1962. Therefore, the age of the residual material in 1981 could have ranged from zero to 20 years or more if aged material was used in 1962. The minimum age would be zero years if it was deposited at the end of the operating period in 1980. Guidance on selection of material age is given in ORAUT (2014a). If the exposure occurred only in FFTF fuel fabrication, then the Hanford fuel-grade composition may be used. If the work location is not known (such as for plutonium scrap recovery), or if the work involved ZPPR fuel fabrication, then both fuel types should be considered and the fuel type giving the highest dose should be used.

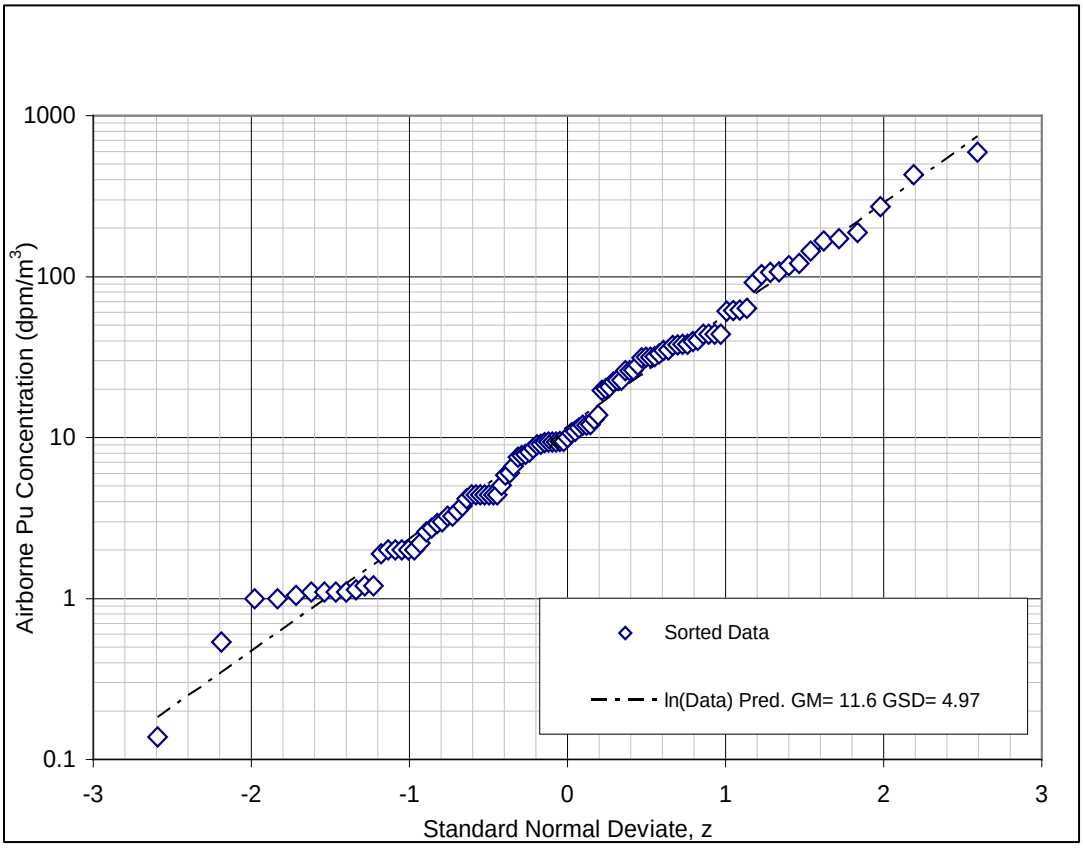

Figure 7-1. Reported plutonium air concentrations.

The intakes in Table 7-5 can be used to estimate the internal dose to the target organ for the years of employment for the worker. The estimated internal doses are assigned as a lognormal distribution with a GSD of 4.97, which corresponds to the distribution of the air concentrations that were used to estimate the annual plutonium intake. The dose should be evaluated for types M and S plutonium, and the dose from the highest type should be used in the IREP input. Because the residual plutonium is in the form of aged material, type SS plutonium should be considered and adjustments made based on guidance in ORAUT-OTIB-0049 (ORAUT 2010a).

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

Table 7-5. Plutonium air concentration and annual alpha activity intake in the Parks Township site residual period.

Year	Air concentration (dpm/m^3)	Inhalation intake (dpm/yr)[a]	Ingestion intake (dpm/yr)[a]
1981	0.023	54	580
1982	0.018	42	450
1983	0.014	33	360
1984	0.011	26	280
1985	0.008	20	220
1986	0.0066	16	170
1987	0.0052	13	130
1988	0.0041	10	100
1989	0.0032	7.6	82
1990	0.0025	6.0	64
1991	0.0020	4.7	50
1992	0.0015	3.7	39
1993	0.0012	2.9	31
1994	0.0009	2.3	24
1995	0.00074	1.8	19
1996	0.00058	1.4	15
1997	0.00045	1.1	12
1998	0.00035	0.84	9.0
1999	0.00028	0.67	7.1
2000	0.00022	0.52	5.6
2001	0.00017	0.41	4.4
2002	0.00013	0.32	3.4
2003	0.00010	0.25	2.7
2004	0.00008	0.20	2.1

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

Table 7-6. Alpha activity fraction for plutonium material at the Parks Township plutonium facility.

Hanford fuel-grade plutonium

Isotope	Aged 0 yr	Aged 5 yr	Aged 10 yr	Aged 15 yr	Aged 20 yr
Pu-238	0.18	0.14	0.12	0.10	0.093
Pu-239/240	0.82	0.68	0.59	0.54	0.51
Pu-241 (beta)	31.9	20.6	14.2	10.2	7.5
Am-241	0.00	0.19	0.29	0.35	0.40

ZPR-III and PNC plutonium

Isotope	Aged 0 yr	Aged 5 yr	Aged 10 yr	Aged 15 yr	Aged 20 yr
Pu-238	0.14	0.12	0.10	0.09	0.09
Pu-239/240	0.86	0.76	0.69	0.65	0.62
Pu-241 (beta)	20.4	14.1	10.1	7.5	5.6
Am-241	0.00	0.13	0.21	0.26	0.30

Commercial reactor-grade plutonium

Isotope	Aged 0 yr	Aged 5 yr	Aged 10 yr	Aged 15 yr	Aged 20 yr
Pu-238	0.65	0.46	0.38	0.32	0.29
Pu-239/240	0.35	0.26	0.22	0.20	0.18
Pu-241 (beta)	50.6	29.7	19.7	13.8	10.0
Am-241	0.00	0.27	0.40	0.48	0.55

The intakes in Table 7-5 relate to DOE work in Building A of the Parks Township site. Work with uranium in Building C could also have contributed to internal dose for workers during the residual period. However, no information was found to allow determination of the residual uranium activity from work in Building C. For workers who are known to have been exposed from Building C residual activity, the internal dose can be evaluated based on intakes given in Table 7-3, with intakes shifted back 3 years to account for the difference in residual period between the Apollo and Parks Township sites. These intakes are given in Table 7-6 for all years of the Parks Township residual period. If the work location is not known for a Parks Township worker, then the higher of the doses based on plutonium (Table 7-5) and uranium (Table 7-7) can be assigned as favorable to the claimant.

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

Year	Air concentration (dpm/m^3)	Inhalation Intake (dpm/yr)[a]	Ingestion Intake (dpm/yr)[a]
1981	6.39	1.53E+04	16,400
1982	5.00	1.20E+04	12,900
1983	3.92	9.40E+03	10,100
1984	3.07	7.36E+03	7,900
1985	2.40	5.77E+03	6,200
1986	1.88	4.51E+03	4,800
1987	1.48	3.54E+03	3,800
1988	1.16	2.78E+03	3,000
1989	0.90	2.16E+03	2,300
1990	0.71	1.70E+03	1,800
1991	0.55	1.33E+03	1,430
1992	0.43	1.04E+03	1,120
1993	0.34	8.16E+02	870
1994	0.27	6.38E+02	680
1995	0.21	5.00E+02	540
1996	0.16	3.91E+02	420
1997	0.13	3.07E+02	330
1998	0.10	2.39E+02	260
1999	0.079	1.89E+02	200
2000	0.061	1.47E+02	160
2001	0.048	1.15E+02	123
2002	0.038	9.02E+01	97
2003	0.029	7.07E+01	76
2004	0.023	5.54E+01	59

a. Doses are assigned as a lognormal distribution with a GSD of 6.95.

8.0 ATTRIBUTIONS AND ANNOTATIONS

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

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GLOSSARY

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

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

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

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

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

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

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

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

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

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

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

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

excluding uranium mining and milling, and (B) is designated by the Secretary of Energy as an atomic weapons employer for purposes of the [Energy Employees Occupational Illness] compensation program.

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

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

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

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

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

byproduct
Material left over from a nuclear, physical, or chemical process designed to produce a particular substance. Examples include the tailings or wastes from the extraction or concentration of uranium or thorium from ore.

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

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

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

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

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

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

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

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

decay products
See *progeny*.

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

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

deep dose
See *personal dose equivalent*.

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

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

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

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

dosimetry
Measurement and calculation of internal and external radiation doses.

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

electron

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

enriched uranium (EU)

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

enrichment

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

environmental occupational dose

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

error

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

exchange period (frequency)

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

exposure

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

external dose

Dose received from radiation emitted by sources outside the body.

film dosimeter

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

fission product (FP)

(1) Radionuclides produced by fission or by the subsequent radioactive decay of radionuclides. (2) Fragments other than neutrons that result from the splitting of an atomic nucleus.

gamma radiation

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

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

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

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

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

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

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

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

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

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

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

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

internal dose
Dose received from radioactive material in the body.

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

in vitro bioassay

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

in vivo bioassay

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

ionizing radiation

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

limit of detection (LOD)

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

low-level radioactive waste (LLRW)

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

minimum detectable activity or amount (MDA)

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

minimum detectable concentration (MDC)

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

minimum detectable level (MDL)

See *minimum detectable activity*.

missed dose

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

mixed oxide (MOX)

Nuclear fuel that contains both plutonium oxide and uranium oxide.

monitoring

Periodic or continuous determination of the presence or amount of ionizing radiation or radioactive contamination in air, surface water, groundwater, soil, sediment, equipment surfaces, or personnel (for example, bioassay or alpha scans). In relation to personnel,

monitoring includes internal and external dosimetry including interpretation of the measurements.

natural uranium (NU)
Uranium as found in nature, approximately 99.27% ^{238}U, 0.72% ^{235}U, and 0.0054% ^{234}U by mass. The specific activity of this mixture is 2.6 × 107 becquerel per kilogram (0.7 microcurie per gram).

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

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

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

nuclear track emulsion, type A (NTA)
Film made by the Eastman Kodak Company that is sensitive to fast neutrons. The developed image has tracks caused by neutrons that become visible under oil immersion with about 1,000-power magnification. The number of tracks in a given area is a measure of the dose from that radiation.

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

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

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

occupational medical dose
Dose from X-ray procedures performed for medical screening of workers as part of an occupational health program. Doses from X-rays used to diagnose diseases or injuries, even if incurred on the job, are not considered occupational and are therefore not eligible to be included in dose reconstruction under the Energy Employees Occupational Illness Compensation Program Act of 2000.

open window (OW)
Area of a film dosimeter that has little to no radiation shielding (e.g., only a holder and visible light protection). The open window measures nonpenetrating as well as penetrating dose,

which minimizes the potential for beta radiation to contribute to the interpreted penetrating dose. See *film dosimeter*.

penetrating dose

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

personal dose equivalent [*Hp*(d)]

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

photon radiation

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

probability of causation (POC)

For purposes of dose reconstruction for the Energy Employees Occupational Illness Compensation Program Act of 2000, the percent likelihood, at the 99th percentile, that a worker incurred a particular cancer from occupational exposure to radiation.

progeny

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

radiation

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

radiation source

(1) Any object or substance that emits radiation. (2) Package of radioactive material constructed to have specific radiation properties used, for example, for medical purposes or to calibrate dosimeters.

radiation worker

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

radioactive

Of, caused by, or exhibiting radioactivity.

radioactive waste

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

radionuclide

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

recycled uranium (RU)

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

resuspension

Process by which the wind lifts small particulates (generally from soil) into the air, which for radionuclides can result in an exposure hazard. Radionuclides released to the air undergo a cycle of suspension in the air, deposition on the ground, resuspension, and redeposition. However, the initial suspension process is generally included in resuspension if the initial contaminating event did not result from deposition of airborne material.

shallow dose

See *personal dose equivalent*.

shielding

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

solubility type

See *absorption type*.

soluble

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

source term

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

Special Exposure Cohort (SEC) [42 U.S.C. § 7384l(14)]

… "member of the Special Exposure Cohort" means a Department of Energy employee, Department of Energy contractor employee, or atomic weapons employee who meets any of the following requirements:

(A) The employee was so employed for a number of work days aggregating at least 250 work days before February 1, 1992, at a gaseous diffusion plant located in Paducah, Kentucky, Portsmouth, Ohio, or Oak Ridge, Tennessee, and, during such employment—

(i) was monitored through the use of dosimetry badges for exposure at the plant of the external parts of employee's body to radiation; or

(ii) worked in a job that had exposures comparable to a job that is or was monitored through the use of dosimetry badges.

(B) The employee was so employed before January 1, 1974, by the Department of Energy or a Department of Energy contractor or subcontractor on Amchitka Island, Alaska, and was exposed to ionizing radiation in the performance of duty related to the Long Shot, Milrow, or Cannikin underground nuclear tests.

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

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

special nuclear material (SNM)

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

standard deviation

Square root of the variance, or the measure of spread in a group of numbers. The sample standard deviation is the square root of the sample variance. This means that it has the same linear units as the original data values or a measure of central tendency, instead of the squared units of the sample variance.

thermoluminescence

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

thermoluminescent dosimeter (TLD)

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

thermoluminescent dosimeter chip

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

transuranic (TRU) elements

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

uncertainty

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

unmonitored dose

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

whole-body counter

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

whole-body (WB) dose

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

X-ray

See *X-ray radiation*.

X-ray radiation

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

ATTACHMENT A
DOSIMETRY REPORTS

TABLE OF CONTENTS

LIST OF FIGURES

ATTACHMENT A
DOSIMETRY REPORTS (continued)

The external dosimetry records for NUMEC come in a variety of report types that change over the years. External dose monitoring has been provided by the NUMEC health physics department and the commercial suppliers Nuclear Science & Engineering (NS&E), R. S. Landauer Jr. and Company, and Eberline Instrument Corporation (Section 6.0). The following discussion describes the various reports found in NUMEC external dosimetry records. Although NS&E was reported to provide external dosimetry services sometime from 1959 to 1968, no NS&E reports have been found.

A.1 REPORTS PROVIDED BY LANDAUER

The NUMEC records might contain summary reports for one monitoring period with dose values for several workers as well as annual reports for one worker. Annual reports for one individual were provided by Landauer in the early 1960s. Figure A-1 is an example of this report for 1964 by quarter of the gamma dose and the lifetime dose. Section 8 of the report indicates the dose is gamma dose. Section 9 of the report indicates the dose was measured by film badge. Each report provides only one type of dose: gamma, beta, or neutron. The lifetime total dose values can be useful in reconciling gaps in dosimetry information. The entries can be assumed to represent the badge exchange period, in this case quarterly.

ATTACHMENT A
DOSIMETRY REPORTS (continued)

RECORD OF

CURRENT OCCUPATIONAL EXTERNAL RADIATIONAL EXPOSURE

ACCOUNT NO. 2314

PREPARED BY
R. S. LANDAUER JR. & COMPANY

PREPARATION DATE 60165 (MO. DAY YR.)

PARTICIPANT NO. 0010

1. NAME (LAST, FIRST AND MIDDLE)

2. SOCIAL SECURITY NUMBER — CARD 1 — CARD 2

3. SEX: MALE - 1, FEMALE - 2

4. DATE OF BIRTH: MONTH / DAY / YEAR

5. NAME OF LICENSEE OR REGISTRANT: NUCLEAR MAT&EQUIP

6. REGISTRATION NUMBER

7. LICENSE NUMBER

OCCUPATIONAL EXPOSURE

8. DOSE RECORD FOR:
X OR GAMMA: 1
BETA:
NEUTRONS:
CODE: WHOLE BODY - 1; SKIN OF WHOLE BODY - 2; HANDS AND FOREARMS - 3; FEET AND ANKLES - 4

9. METHODS OF MONITORING
X OR GAMMA: 1
BETA:
NEUTRONS:
CODE: FILM BADGE - 1; POCKET CHAMBER - 2; CALCULATIONS - 3; SOLID STATE - 4; OTHER - 5

10. PERIOD OF EXPOSURE FROM MO.	DAY	YR.	TO MO.	DAY	YR.	11. X OR GAMMA DOSE FOR THE PERIOD (rem) DECIMAL	12. BETA DOSE FOR THE PERIOD (rem) DECIMAL	13. NEUTRON DOSE FOR THE PERIOD (rem) DECIMAL	TOTAL DOSE FOR THE PERIOD (rem) DECIMAL
1	01	4	3	31	4	150			150
4	01	4	6	30	4	140			140
7	01	4	9	30	4	180			180
10	01	4	12	31	4	390			390

LIFETIME ACCUMULATED DOSE

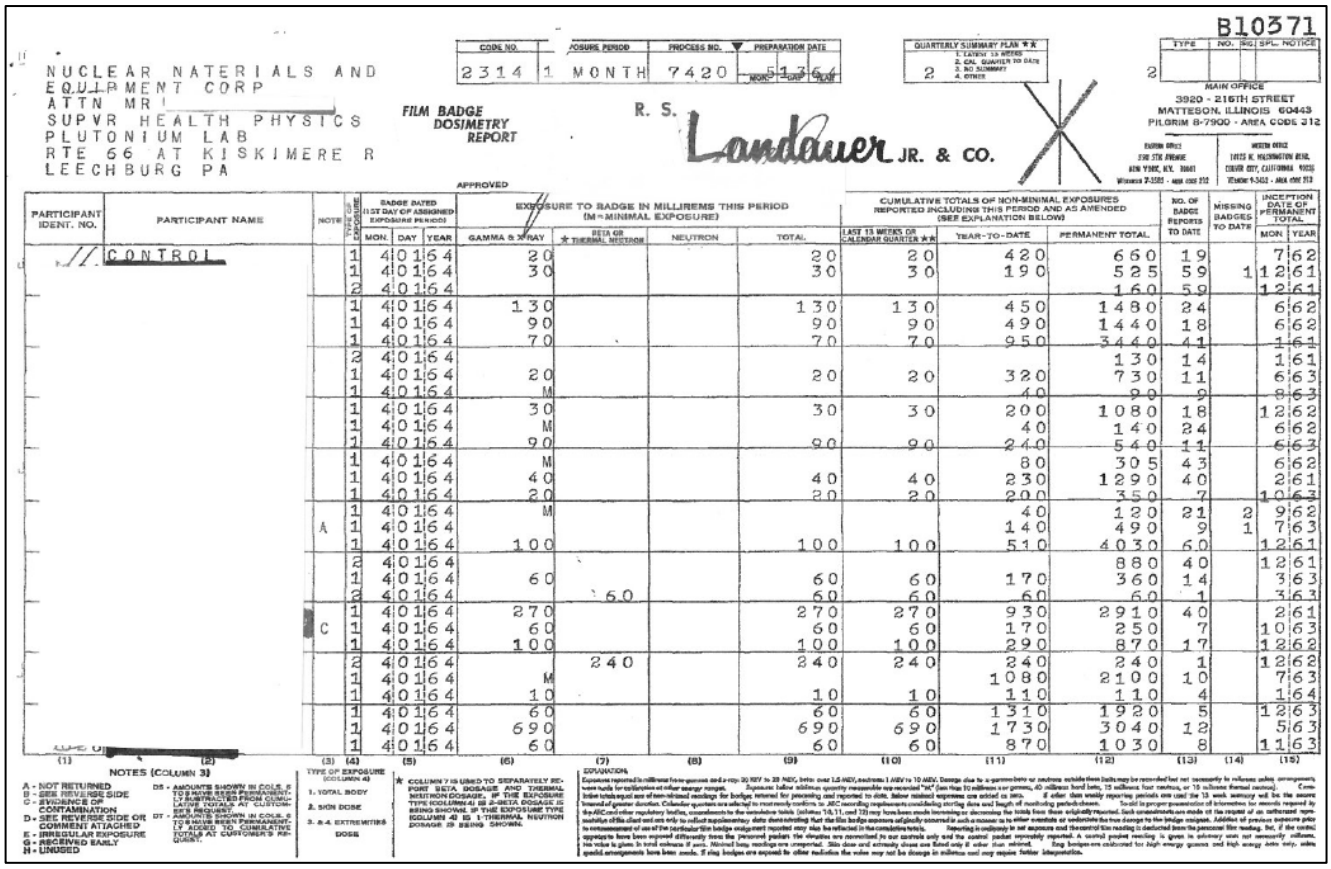
14. PREVIOUS TOTAL (rem) DECIMAL	15. TOTAL QUARTERLY DOSE DATE MO.	DAY	YR.	(rem) DECIMAL	16. TOTAL ACCUMULATED DOSE (rem) DECIMAL	17. PERM. ACC. DOSE S(N-18) (rem)	18. UNUSED PART OF PERM. ACC. DOSE (rem) DECIMAL
300	3	31	4	150	450		
450	6	30	4	140	590		
590	9	30	4	180	770		
770	12	31	4	390	1160		

R. S. LANDAUER JR. & CO. 3920 216TH STREET MATTESON, ILLINOIS 60443

Figure A-1. Landauer annual report for one worker (showing gamma, 1964).(NUMEC 1964a, p. 3)

ATTACHMENT A
DOSIMETRY REPORTS (continued)

Landauer provided results of dosimeters for groups of workers for each badge exchange period. The form code 2314 indicates workers on a monthly badge exchange frequency and form code "2315" indicates workers on a half-month badge exchange frequency. Examples of these forms are given in Figures A-2 and A-3.



NUCLEAR NATERIALS AND EQUIPMENT CORP
ATTN MR [redacted]
SUPVR HEALTH PHYSICS
PLUTONIUM LAB
RTE 66 AT KISKIMERE R
LEECHBURG PA

CODE NO. 2314 | EXPOSURE PERIOD 1 MONTH | PROCESS NO. 7420 | PREPARATION DATE MON. 5 DAY 17 YEAR 64

QUARTERLY SUMMARY PLAN ★★ 2

B10371

TYPE 2 | NO. | REC | SPL. NOTICE

MAIN OFFICE
3920 - 216TH STREET
MATTESON, ILLINOIS 60443
PILGRIM 8-7900 - AREA CODE 312

FILM BADGE DOSIMETRY REPORT

R. S. Landauer JR. & CO.

APPROVED

PARTICIPANT IDENT. NO.	PARTICIPANT NAME	NOTE	TYPE OF EXPOSURE	BADGE DATED (1ST DAY OF ASSIGNED EXPOSURE PERIOD) MON.	DAY	YEAR	EXPOSURE TO BADGE IN MILLIREMS THIS PERIOD (M=MINIMAL EXPOSURE) GAMMA & X-RAY	BETA OR ★THERMAL NEUTRON	NEUTRON	TOTAL	CUMULATIVE TOTALS OF NON-MINIMAL EXPOSURES REPORTED INCLUDING THIS PERIOD AND AS AMENDED (SEE EXPLANATION BELOW) LAST 13 WEEKS OR CALENDAR QUARTER ★★	YEAR-TO-DATE	PERMANENT TOTAL	NO. OF BADGE REPORTS TO DATE	MISSING BADGES TO DATE	INCEPTION DATE OF PERMANENT TOTAL MON.	YEAR
[redacted]	CONTROL		1	4	01	64	20			20	20	420	660	19		7	62
			1	4	01	64	30			30	30	190	525	59	1	12	61
			2	4	01	64							160	59		12	61
			1	4	01	64	130			130	130	450	1480	24		6	62
			1	4	01	64	90			90	90	490	1440	18		6	62
			1	4	01	64	70			70	70	950	3440	41		1	61
			2	4	01	64							130	14		1	61
			1	4	01	64	20			20	20	320	730	11		6	63
			1	4	01	64	M			40		90	90	9		8	63
			1	4	01	64	30			30	30	200	1080	18		12	62
			1	4	01	64	M			40		140	140	24		6	62
			1	4	01	64	90			90	90	240	540	11		6	63
			1	4	01	64	M			80		305	305	43		6	62
			1	4	01	64	40			40	40	230	1290	40		2	61
			1	4	01	64	20			20	20	200	350	7		10	63
			1	4	01	64	M			40		120	120	21	2	9	62
		A	1	4	01	64				140		490	490	9	1	7	63
			1	4	01	64	100			100	100	510	4030	60		12	61
			2	4	01	64							880	40		12	61
			1	4	01	64	60			60	60	170	360	14		3	63
			3	4	01	64		60		60	60		60	1		3	63
			1	4	01	64	270			270	270	930	2910	40		2	61
		C	1	4	01	64	60			60	60	170	250	7		10	63
			1	4	01	64	100			100	100	290	870	17		12	62
			2	4	01	64		240		240	240	240	240	1		12	62
			1	4	01	64	M			80		1080	2100	10		7	63
			1	4	01	64	10			10	10	110	110	4		1	64
			1	4	01	64	60			60	60	1310	1920	5		12	63
			1	4	01	64	690			690	690	1730	3040	12		5	63
			1	4	01	64	60			60	60	870	1030	8		11	63

(1) (2) (3) (4) (5) (6) (7) (8) (9) (10) (11) (12) (13) (14) (15)

NOTES (COLUMN 3)
A - NOT RETURNED
B - SEE REVERSE SIDE
C - EVIDENCE OF CONTAMINATION
D - SEE REVERSE SIDE OR COMMENT ATTACHED
E - IRREGULAR EXPOSURE
G - RECEIVED EARLY
H - UNUSED
DS - AMOUNTS SHOWN IN COLS. 6 TO 9 HAVE BEEN PERMANENTLY RESTRICTED FROM CUMULATIVE TOTALS AT CUSTOMER'S REQUEST.
DT - AMOUNTS SHOWN IN COLS. 6 TO 9 HAVE BEEN PERMANENTLY ADDED TO CUMULATIVE TOTALS AT CUSTOMER'S REQUEST.

TYPE OF EXPOSURE (COLUMN 4)
1. TOTAL BODY
2. SKIN DOSE
3. & 4. EXTREMITIES DOSE

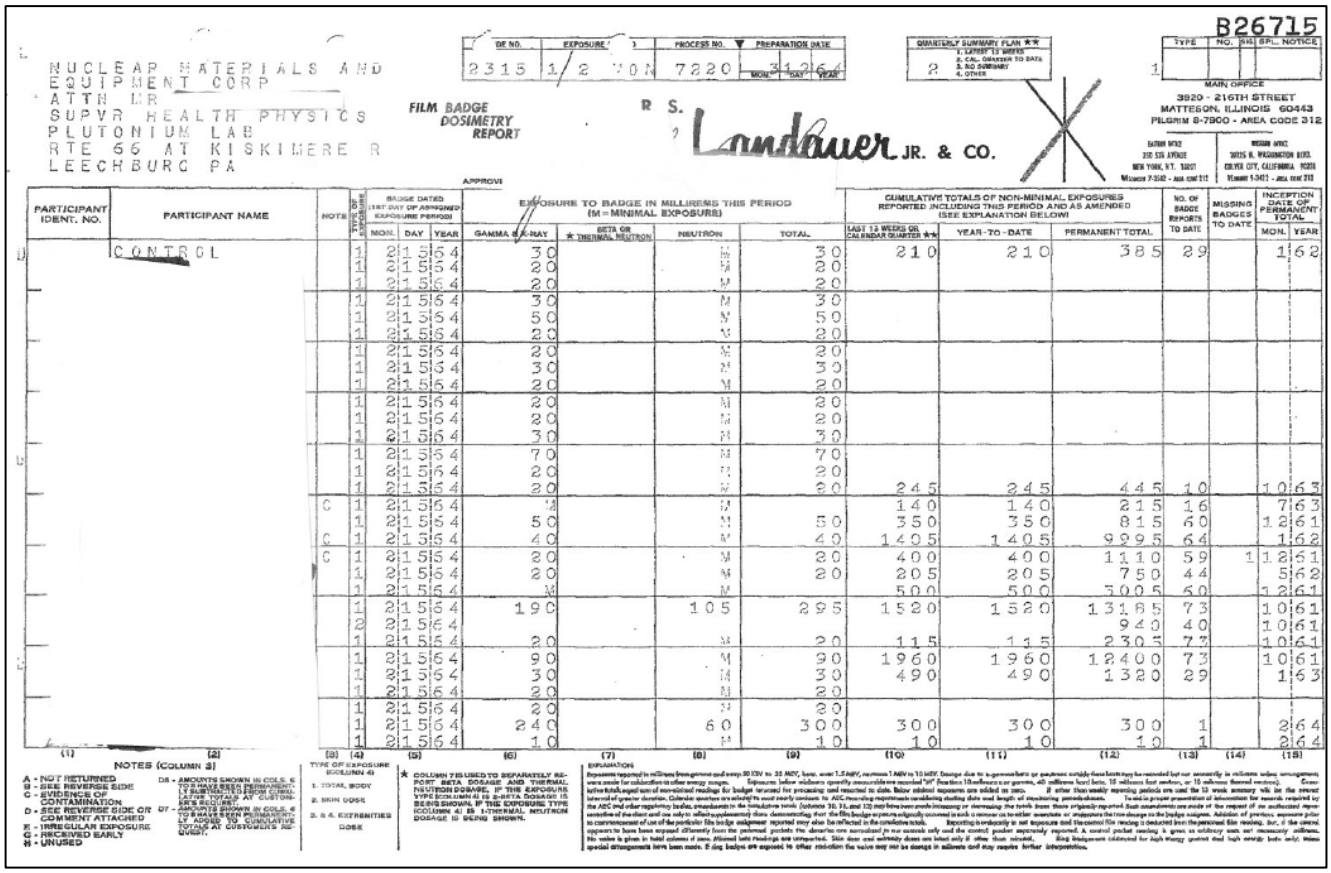
★ COLUMN 7 IS USED TO SEPARATELY REPORT BETA DOSAGE AND THERMAL NEUTRON DOSAGE, IF THE EXPOSURE TYPE (COLUMN 4) IS 2-BETA DOSAGE IS BEING SHOWN. IF THE EXPOSURE TYPE (COLUMN 4) IS 1-THERMAL NEUTRON DOSAGE IS BEING SHOWN.

EXPLANATION

Figure A-2. Landauer monthly dose results (code 2314) (NUMEC 1964b, p. 72).

Dose values are reported as Gamma and X-Ray, Beta or Thermal Neutron, Neutron, and Total for the period. Column 4 of the report indicates the type of exposure. A 1 indicates total body, 2 indicates skin exposure, and 3 and 4 indicate extremity. The Beta or Thermal Neutron column is beta dose if the exposure type is 2 (skin); it is thermal neutron if the exposure type is 1 (total body). If a worker was monitored for both total body and skin exposures, there would be two lines for the individual in the report. The report also indicates the cumulative total dose for the current quarter (or previous 13 weeks), the current year, and the lifetime totals. Note that the lifetime totals represent the totals recorded and reported by Landauer and might not include all dose recorded by the site. However, it is possible the site provided additional dose values to Landauer that are included in the totals as indicated by a note in column 3. An additional feature of the report is the information on the far right side. This includes the month and year that monitoring was started (in the Landauer records) along with the number of badge periods from the start of monitoring until the present report. Missing badge reports are indicated. This information can be useful in reconciling dosimetry records and the number of badge cycles to include in the dose reconstruction.

ATTACHMENT A
DOSIMETRY REPORTS (continued)



NUCLEAR MATERIALS AND EQUIPMENT CORP
ATTN MR
SUPVR HEALTH PHYSICS
PLUTONIUM LAB
RTE 66 AT KISKIMERE R
LEECHBURG PA

FILM BADGE DOSIMETRY REPORT

R. S. Landauer JR. & CO.

B26715

DE NO.	EXPOSURE	PROCESS NO.	PREPARATION DATE
2315	1/2 MON	7220	31 26 (MON. DAY YEAR)

QUARTERLY SUMMARY PLAN ★★: 2

MAIN OFFICE
3920 - 216TH STREET
MATTESON, ILLINOIS 60443
PILGRIM 8-7900 - AREA CODE 312

PARTICIPANT IDENT. NO.	PARTICIPANT NAME	NOTE	EXPOSURE CODE	BADGE DATED (LAST DAY OF ASSIGNED EXPOSURE PERIOD) MON.	DAY	YEAR	EXPOSURE TO BADGE IN MILLIREMS THIS PERIOD (M = MINIMAL EXPOSURE) GAMMA & X-RAY	BETA OR ★ THERMAL NEUTRON	NEUTRON	TOTAL	CUMULATIVE TOTALS OF NON-MINIMAL EXPOSURES REPORTED INCLUDING THIS PERIOD AND AS AMENDED (SEE EXPLANATION BELOW) LAST 13 WEEKS OR CALENDAR QUARTER ★★	YEAR-TO-DATE	PERMANENT TOTAL	NO. OF BADGE REPORTS TO DATE	MISSING BADGES TO DATE	INCEPTION DATE OF PERMANENT TOTAL MON.	YEAR
	CONTROL		1	2	15	64	30		M	30	210	210	385	29		1	62
			1	2	15	64	20		M	20							
			1	2	15	64	20		M	20							
			1	2	15	64	30		M	30							
			1	2	15	64	50		M	50							
			1	2	15	64	20		M	20							
			1	2	15	64	20		M	20							
			1	2	15	64	30		M	30							
			1	2	15	64	20		M	20							
			1	2	15	64	20		M	20							
			1	2	15	64	20		M	20							
			1	2	15	64	30		M	30							
			1	2	15	64	70		M	70							
			1	2	15	64	20		M	20							
			1	2	15	64	30		M	30	245	245	445	10		10	63
		C	1	2	15	64	M		M		140	140	215	16		7	63
			1	2	15	64	50		M	50	350	350	815	60		12	61
		C	1	2	15	64	40		M	40	1405	1405	9995	64		1	62
		C	1	2	15	64	20		M	20	400	400	1110	59		11	61
			1	2	15	64	20		M	20	205	205	750	44		5	62
			1	2	15	64	M		M		500	500	3005	60		12	61
			1	2	15	64	190		105	295	1520	1520	13165	73		10	61
			2	2	15	64							940	40		10	61
			1	2	15	64	20		M	20	115	115	2305	73		10	61
			1	2	15	64	90		M	90	1960	1960	12400	73		10	61
			1	2	15	64	30		M	30	490	490	1320	29		1	63
			1	2	15	64	20		M	20							
			1	2	15	64	20		M	20							
			1	2	15	64	240		60	300	300	300	300	1		2	64
			1	2	15	64	10		M	10	10	10	10	1		2	64

(1) (2) (3) (4) (5) (6) (7) (8) (9) (10) (11) (12) (13) (14) (15)

NOTES (COLUMN 3)
A - NOT RETURNED
B - SEE REVERSE SIDE
C - EVIDENCE OF CONTAMINATION
D - SEE REVERSE SIDE OR COMMENT ATTACHED
E - IRREGULAR EXPOSURE
G - RECEIVED EARLY
H - UNUSED

DB - AMOUNTS SHOWN IN COLS. 6 TO 9 HAVE BEEN PERMANENTLY SUBTRACTED FROM CUMULATIVE TOTALS AT CUSTOMER'S REQUEST.
DT - AMOUNTS SHOWN IN COLS. 6 TO 9 HAVE BEEN PERMANENTLY ADDED TO CUMULATIVE TOTALS AT CUSTOMER'S REQUEST.

TYPE OF EXPOSURE (COLUMN 4)
1. TOTAL BODY
2. SKIN DOSE
3. 4. 5. EXTREMITIES DOSE

★ COLUMN 7 IS USED TO SEPARATELY REPORT BETA DOSAGE AND THERMAL NEUTRON DOSAGE. IF THIS EXPOSURE TYPE (COLUMN 4) IS 2-BETA DOSAGE IS BEING SHOWN. IF THE EXPOSURE TYPE (COLUMN 4) IS 1-THERMAL NEUTRON DOSAGE IS BEING SHOWN.

EXPLANATION

Figure A-3. Landauer semi-monthly dose results (code 2315) (NUMEC 1964b, p. 86).

A.2 REPORTS PROVIDED BY EBERLINE

One type of Eberline report was found in the dosimetry records. This report provides the dosimeter results for one reporting period for several workers. Figure A-4 shows results of Eberline film badges for workers at the Parks Township plutonium laboratory. The doses are reported for gamma, beta, and neutron exposures. For the gamma and beta columns, a blank indicates zero dose. For neutron exposures, a blank in the Neutron Tracks columns indicates no monitoring for neutrons. A value of zero, a positive value, or an entry of M (less than detection limit) indicates the worker was monitored with a neutron film badge and a value should be in the neutron dose column (or blank if the dose was zero). Two lines are included for each worker. The values depend on the section of the report. In the Dose for Period section, the top values give the gamma, beta, and neutron doses. The lower value in the first column is the sum of the gamma and neutron dose. When both the upper and lower values of the first column are the same, there is no neutron dose. In the next section the upper values are for Accumulated Dose for Cal Qtr and the lower values are for Accumulated Dose for Cal. Yr. This section gives WB dose (gamma plus neutron), skin dose, and extremity dose. A blank in the extremity column does not indicate a zero value because there might have been no monitoring for extremity doses.

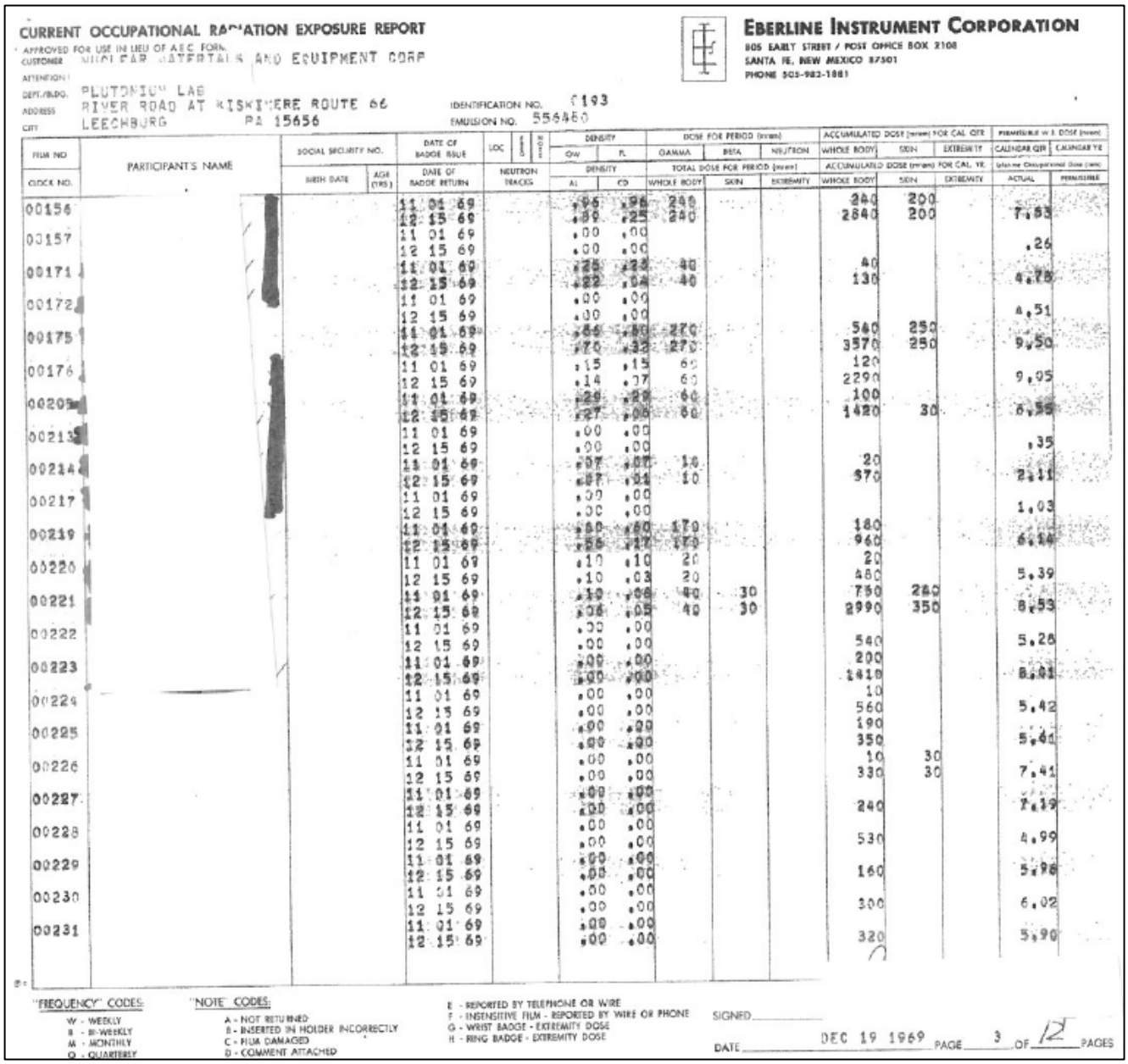
The lower line in the last section of the report gives the Lifetime Occupational Dose (rem). Other entries in this section are usually blank and provide no useful information.

ATTACHMENT A
DOSIMETRY REPORTS (continued)

The report lists entries for date of badge issue and badge return. The issue date is generally the first of the month. The return date is generally about 2 weeks after the end of the month in which the badge was issued. The wear period is likely for just the month in which the badge was issued because the return date likely represents the time the badge was returned to Eberline for processing.

The quarterly, annual, and lifetime totals can be useful in reconciling gaps in dosimetry information.

Although the report has a column for badge exchange frequency, this column seldom has an entry.



CURRENT OCCUPATIONAL RADIATION EXPOSURE REPORT

APPROVED FOR USE IN LIEU OF AEC FORM

CUSTOMER NUCLEAR MATERIALS AND EQUIPMENT CORP

ATTENTION

DEPT./BLDG. PLUTONIUM LAB

ADDRESS RIVER ROAD AT KISKIMERE ROUTE 66

CITY LEECHBURG PA 15656

IDENTIFICATION NO. 0193

EMULSION NO. 554460

EBERLINE INSTRUMENT CORPORATION
805 EARLY STREET / POST OFFICE BOX 2108
SANTA FE, NEW MEXICO 87501
PHONE 505-982-1881

FILM NO. / CLOCK NO.	PARTICIPANT'S NAME	SOCIAL SECURITY NO. / BIRTH DATE	AGE (YRS.)	DATE OF BADGE ISSUE / DATE OF BADGE RETURN	LOC	NEUTRON TRACKS	DENSITY OW	DENSITY FL	DENSITY A1	DENSITY C0	DOSE FOR PERIOD (mrem) GAMMA	BETA	NEUTRON	TOTAL DOSE FOR PERIOD (mrem) WHOLE BODY	SKIN	EXTREMITY	ACCUMULATED DOSE (mrem) FOR CAL. QTR. WHOLE BODY	SKIN	EXTREMITY	ACCUMULATED DOSE (mrem) FOR CAL. YR. WHOLE BODY	SKIN	EXTREMITY	PERMISSIBLE W.B. DOSE (mrem) CALENDAR QTR.	CALENDAR YR. ACTUAL	PERMISSIBLE
00156				11 01 69			.95	.95			240						240	200		2840	200			7.83	
				12 15 69			.89	.25			240						2840	200							
00157				11 01 69			.00	.00																.26	
				12 15 69			.00	.00																	
00171				11 01 69			.25	.22			40						40			130				4.78	
				12 15 69			.22	.04			40						130								
00172				11 01 69			.00	.00																4.51	
				12 15 69			.00	.00																	
00175				11 01 69			.86	.80			270						540	250		3570	250			9.50	
				12 15 69			.70	.32			270						3570	250							
00176				11 01 69			.15	.15			60						120			2290				9.05	
				12 15 69			.14	.07			60						2290								
00205				11 01 69			.29	.29			60						100			1420	30			6.55	
				12 15 69			.27	.05			60						1420	30							
00213				11 01 69			.00	.00																.35	
				12 15 69			.00	.00																	
00214				11 01 69			.07	.07			10						20			370				2.11	
				12 15 69			.07	.01			10						370								
00217				11 01 69			.00	.00																1.03	
				12 15 69			.00	.00																	
00219				11 01 69			.60	.60			170						180			960				6.14	
				12 15 69			.56	.17			170						960								
00220				11 01 69			.10	.10			20						20			480				5.39	
				12 15 69			.10	.03			20						480								
00221				11 01 69			.10	.06			40		30				750	280		2990	350			8.53	
				12 15 69			.06	.05			40		30				2990	350							
00222				11 01 69			.00	.00																5.28	
				12 15 69			.00	.00																	
00223				11 01 69			.00	.00																8.01	
				12 15 69			.00	.00																	
00224				11 01 69			.00	.00																5.42	
				12 15 69			.00	.00																	
00225				11 01 69			.00	.00																5.41	
				12 15 69			.00	.00																	
00226				11 01 69			.00	.00																7.41	
				12 15 69			.00	.00																	
00227				11 01 69			.00	.00																7.19	
				12 15 69			.00	.00																	
00228				11 01 69			.00	.00																4.99	
				12 15 69			.00	.00																	
00229				11 01 69			.00	.00																5.98	
				12 15 69			.00	.00																	
00230				11 01 69			.00	.00																6.02	
				12 15 69			.00	.00																	
00231				11 01 69			.00	.00																5.90	
				12 15 69			.00	.00																	

"FREQUENCY" CODES:
W - WEEKLY
B - BI-WEEKLY
M - MONTHLY
Q - QUARTERLY

"NOTE" CODES:
A - NOT RETURNED
B - INSERTED IN HOLDER INCORRECTLY
C - FILM DAMAGED
D - COMMENT ATTACHED
E - REPORTED BY TELEPHONE OR WIRE
F - INSENSITIVE FILM - REPORTED BY WIRE OR PHONE
G - WRIST BADGE - EXTREMITY DOSE
H - RING BADGE - EXTREMITY DOSE

SIGNED

DATE DEC 19 1969 PAGE 3 OF 12 PAGES

Figure A-4. Eberline report for one exchange period (NUMEC 1969, p. 4).

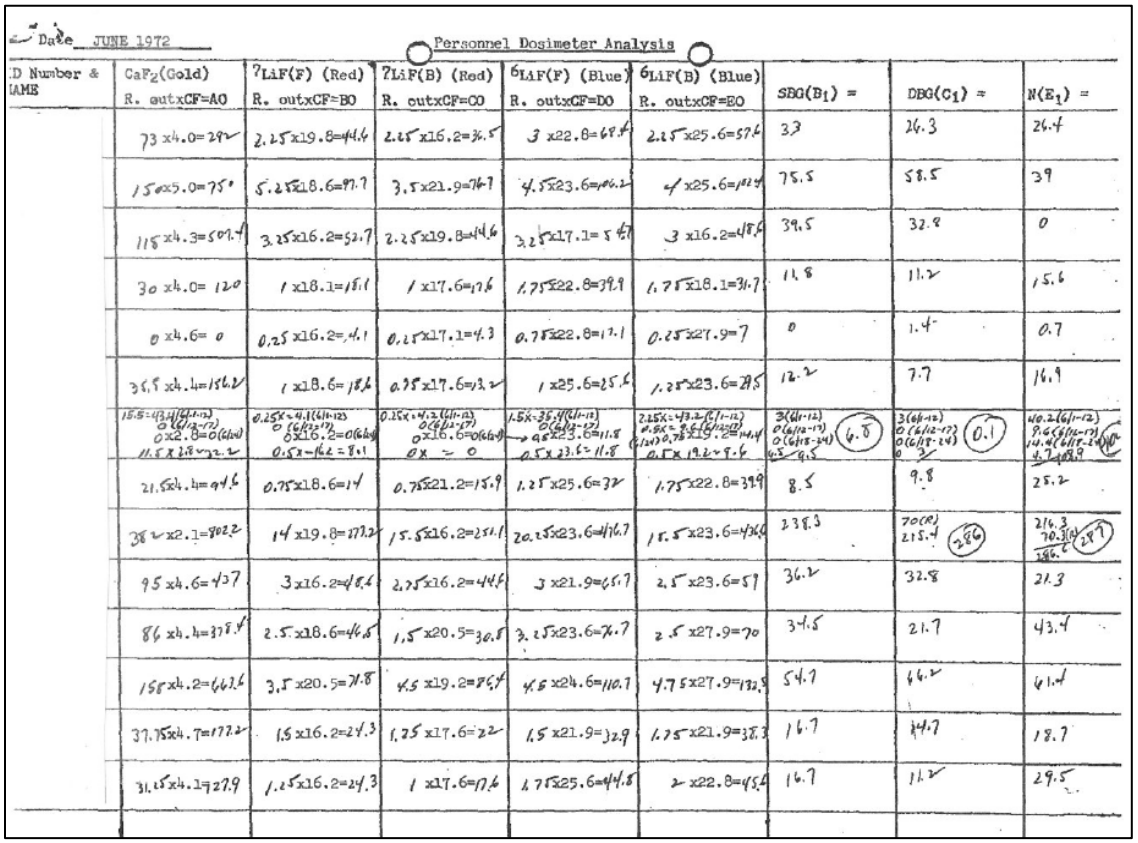
ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3 NUMEC DOSIMETRY REPORTS

NUMEC provided external dose monitoring for much of the covered periods for both the Apollo and Parks Township facilities. However, no reports are available from before 1964 when the records include the Landauer reports as described above. For some of the periods when Landauer and Eberline provided dosimetry, there could also be records of dosimetry from NUMEC. In these cases, the dose reconstructor must reconcile the two sets of dosimetry records. The records of dosimetry from NUMEC are discussed in this section.

A.3.1 Monthly Dose Calculation Sheets

This report indicates the calculations performed to convert dosimeter readout data to external dose from shallow beta/gamma, deep gamma, and neutron exposures. An example of a monthly calculation sheet is given in Figure A-5 for June of 1972. The headings indicate the results are for TLD badges. The dose values are reported to the tenth of an mrem. These values might be included in other summary reports but be rounded to whole numbers of mrem. There are reports for the months of 1970 to 1975. Some of the dose entries indicate “see special report,” and no values are given for these individuals (see Figure A-6). There is generally one line for each worker with dose values on the right side of the page. Note that one worker has entries for three monitoring periods, with along with total dose for the month. When this is noted in the records, the worker was issued three dosimeters during the month and the total dose from the three dosimeters was reported. For such cases, assignment of zero doses should include consideration of all reported monitoring periods during the month.



Date JUNE 1972 — Personnel Dosimeter Analysis

ID Number & NAME	CaF_2(Gold) R. outxCF=A0	^{7}LiF(F) (Red) R. outxCF=B0	^{7}LiF(B) (Red) R. outxCF=C0	^{6}LiF(F) (Blue) R. outxCF=D0	^{6}LiF(B) (Blue) R. outxCF=E0	SBG(B_1) =	DBG(C_1) =	N(E_1) =
	73 x4.0=292	2.25x19.8=44.6	2.25x16.2=36.5	3 x22.8=68.4	2.25x25.6=57.6	3.3	26.3	26.4
	15 x5.0=75	5.25x18.6=97.7	3.5x21.9=76.7	4.5x23.6=106.2	4 x25.6=102.4	75.5	58.5	39
	118 x4.3=507.4	3.25x16.2=52.7	2.25x19.8=44.6	3.25x17.1=55.6	3 x16.2=48.6	39.5	32.8	0
	30 x4.0=120	1 x18.1=18.1	1 x17.6=17.6	1.75x22.8=39.9	1.75x18.1=31.7	11.8	11.2	15.6
	0 x4.6=0	0.25x16.2=4.1	0.25x17.1=4.3	0.75x22.8=17.1	0.25x27.9=7	0	1.4	0.7
	35.5 x4.4=156.2	1 x18.6=18.6	0.75x17.6=13.2	1 x25.6=25.6	1.25x23.6=29.5	12.2	7.7	16.9
	15.5=43.4 (4/11-12) 0 (6/12-17) 0x2.8=0 (6/18-24) 11.5x2.8=32.2	0.25x=4.1 (6/11-12) 0 (6/12-17) 0x16.2=0 (6/18-24) 0.5x=16.2=8.1	0.25x=4.3 (6/11-12) 0 (6/12-17) 0x16.6=0 (6/18-24) 0x ~ 0	1.5x=38.4 (6/11-12) 0 (6/12-17) → 0.5x23.6=11.8 0.5x23.6=11.8	2.25x=57.2 (6/11-12) 0.5x=8.6 (6/12-17) (6/24) 0.75x29.2=21.4 0.5x 19.2=9.6	3 (6/11-12) 0 (6/12-17) 0 (6/18-24) 6.5 6.5 (6.8)	3 (6/11-12) 0 (6/12-17) 0 (6/18-24) 0 3 (0.1)	40.2 (6/11-12) 9.6 (6/12-17) 14.4 (6/18-24) 4.7 28.9
	21.5x4.4=94.6	0.75x18.6=14	0.75x21.2=15.9	1.25x25.6=32	1.75x22.8=39.9	8.5	9.8	25.2
	38.2x2.1=80.2	14 x19.8=277.2	15.5x16.2=251.1	20.25x23.6=476.7	15.5x23.6=436.4	238.3	70 (R) 215.4 (286)	214.3 70.3 (R) 286.6 (29.7)
	95 x4.6=437	3 x16.2=48.6	2.75x16.2=44.6	3 x21.9=65.7	2.5 x23.6=59	36.2	32.8	21.3
	86 x4.4=378.4	2.5 x18.6=46.5	1.5 x20.5=30.8	3.25x23.6=76.7	2.5 x27.9=70	34.5	21.7	43.4
	158 x4.2=663.6	3.5 x20.5=71.8	4.5 x19.2=86.4	4.5 x24.6=110.7	4.75x27.9=132.5	54.7	66.2	61.4
	37.75x4.7=177.2	1.5 x16.2=24.3	1.25 x17.6=22	1.5 x21.9=32.9	1.75x21.9=38.3	16.7	14.7	18.7
	31.25x4.1=127.9	1.25x16.2=24.3	1 x17.6=17.6	1.75x25.6=44.8	2 x22.8=45.6	16.7	11.2	29.5

Figure A-5. Monthly calculation for June 1972 (Boyd 2006b, p. 235).

ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3.2 Monthly Special Report Calculation Sheet

This report shows calculation of external dose for workers with special dosimetry badges (Figure A-6.) The report is for one individual for the fourth quarter of 1974. The results for each badge wear period are given for shallow dose (SBG), deep gamma dose (DBG), and neutron dose (n). A summary of doses is listed at the bottom of the page.

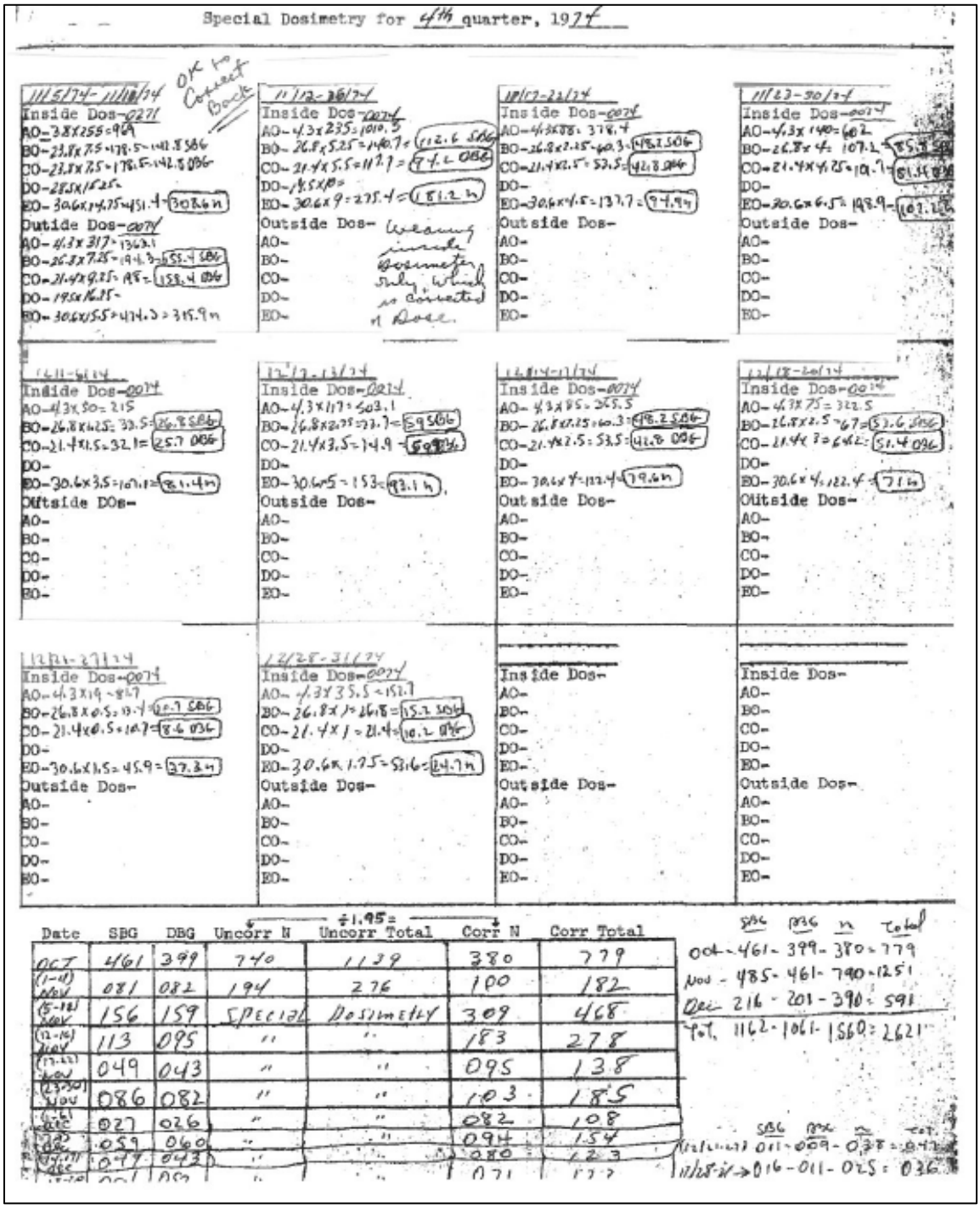


Figure A-6. Special dosimetry calculation sheet (Boyd 2006b, p. 72).

ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3.3 Monthly Summary Gamma and Neutron

This report gives the total monthly dose to individuals as handwritten results of WB dose from gamma and neutron (if any) exposures (Figure A-7). The Instrument location indicates TLDs were used to monitor external dose. Although the report has columns for each day of the month, only the monthly totals are given. The reported values include error bounds, which appear to be just 20% of the reported value.

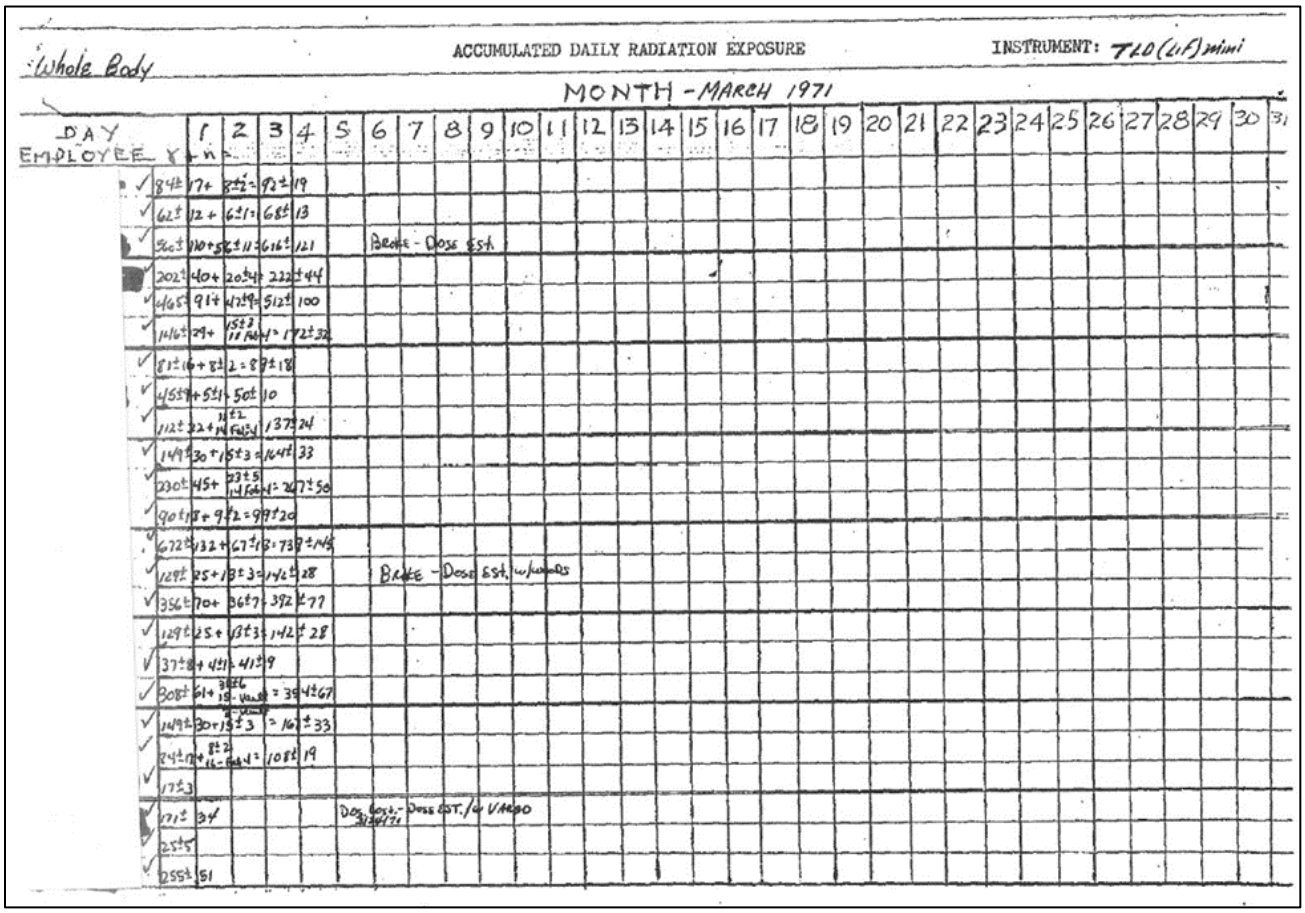


Figure A-7. Monthly summary of gamma and neutron (Boyd 2006b, p. 426).

ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3.4 Monthly Monitoring Results by Wear Period

Monthly reports with daily dose entries might be available for the late 1960s to the early 1970s giving doses reported as they were identified (Figure A-8). The report is for WB exposures from February 1968. The entries include two numbers; the top number is the dose for the current period and the bottom number is the cumulative dose to the current day of the month. The final monthly total dose is listed on the left side of the form. This particular form includes a monthly total based on the TLD results and another total based on film badges. The Instrument: entry in the upper right corner of the form indicates the reported daily values are based on TLDs. The totals on the left for the TLD agree with the daily cumulative totals on the form. The upper left of the form indicates the dose values are for WB. Other reports might be available for finger ring doses. Care must be taken when reviewing the forms to be sure the doses are for the WB. Finger ring doses are only useful for evaluation of dose to cancers on the hands.

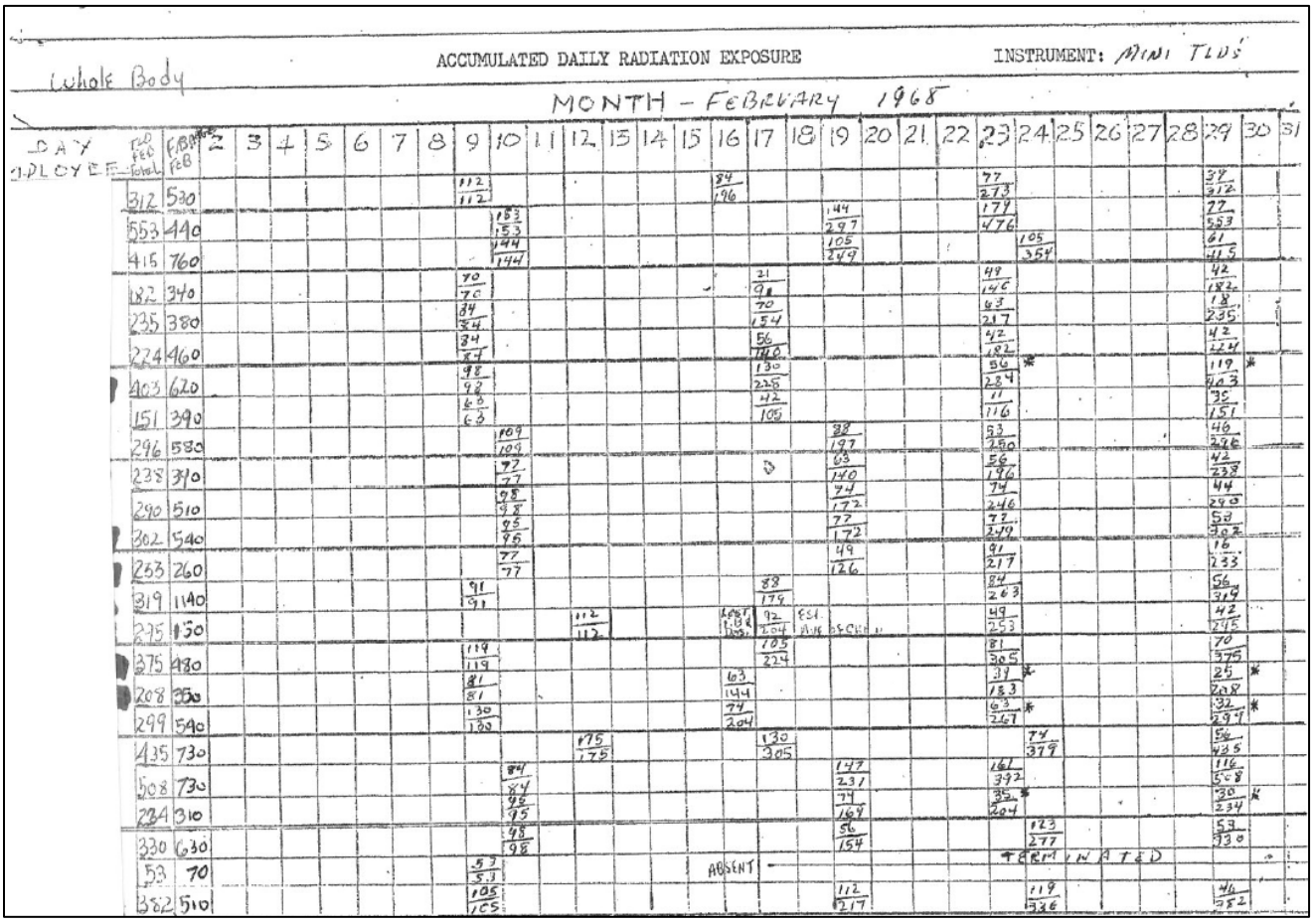


Figure A-8. Monthly monitoring results by wear period (NUMEC 1968a, p. 6).

ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3.5 Annual Report with Monthly and Quarterly Summaries

This report gives an annual summary for workers by month, quarter, and year of WB external exposure (Figure A-9). The report gives the monthly dosimeter results for workers at the Parks Township ZPPR fuel production operation through the end of 1968. Other similar reports might be available that give results for only a part of the year. The workers on this form are grouped by job category. The first set is for Acc't (material accounting) workers and the second set is for Ceramics workers. This information could be useful for identifying work locations. The dose values could include both gamma and neutron dose, but the breakdown is not indicated. The column Total Dose Thru 1968 gives the lifetime dose for each individual (one line per individual worker). The total quarterly, annual, and lifetime dose values can be useful in reconciling gaps in dosimetry information.

ZPPR-EXTERNAL RADIATION EXPOSURES

Name	Date of Birth	Per Acc. Dose 5 (N-18) Thru Rem '68	Total Dose Thru 1968	Unused Part of Per Acc Thru Dose '68	1st Qrt				2nd Qrt				3rd Qrt				4th Qrt				1968 Total
					Jan	Feb	Mar	Tot	Apr	May	June	Tot	July	Aug	Sept	Tot	Oct	Nov	Dec	Tot	
(Acc't)																					
		122.500	13.538	108.962	.430	.530	.380	1.340	.205	.430	.260	.895	.240	.070	.450	.760	.171	.272			
		70.000	18.898	51.102	.935	.913	.374	2.222	.606	.802	.606	2.014	.890	1.322	.637	2.849	.461	.766	.652	1.879	8.464
		13.750	TERM	-	.352	.484	.407	1.243	.370	.489	.378	1.237	.652	.866	.562	2.080	.664	.210	TERM.	-	-
		27.500	6.233	21.267	.330	.429	.363	1.122	.345	.225	.340	.910	.596	1.015	.813	2.424	.722	.508	.426	1.656	6.112
		7.500	6.197	1.303	.160	1.050	.230	1.440	.380	.340	.360	1.080	.740	.510	.340	1.590	.405	.467	.725	1.597	5.707
		27.500	2.581	24.919	.370	.200	.130	.700	.116	.130	.130	.376	.220	.160	.200	.580	.278	.293	.354	.975	2.581
		100.000	-	-	-	-	-	-	-	-	.010	.010	.250	.010	.030	.290	.010	.020			
		18.750	0.852	17.898	-	-	-	-	-	-	-	-	-	-	-	-	-	.353	.499	.852	0.852
(CERAMICS)																-	ISSUED TLDs - OCT 1, 1968				
		56.250	16.538	41.862	.330	.340	.590	1.260	.510	.500	.650	1.660	.750	.850	.590	2.190	.302	1.209	.967	2.478	7.588
		41.250	16.597	24.653	.460	.230	.330	1.020	.570	.360	.400	1.330	.400	.220	.270	.890	.289	.587	.581	1.457	4.697
		10.000	6.963	3.037	.100	.150	.370	.620	.180	.290	.260	.730	.280	.150	.360	.790	.419	.272	.272	.963	3.103
		37.500	TERM	-	.280	.380	.650	1.310	.360	1.860	.700	2.920	1.240	.430	.430	2.100	.544	.418	.055	TERM.	
		52.500	10.799	41.701	.300	.350	.280	.930	.710	.270	.560	1.540	.340	.800	.330	1.470	.596	.725	1.028	2.349	6.289
		30.000	26.573	3.427	.000	.260	.050	.310	.320	.860	.572	1.752	.440	.228	.433	1.101	.340	.532	.478	1.350	4.513
		TERM.	-	-	.230	.230	.330	.790	.120	.380	.610	1.110	.410	.360	.300	1.070	.280	TERM, 10-25-68			-

Figure A-9. Annual report with monthly and quarterly summaries (NUMEC 1968b, p. 6).

ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3.6 Annual Report with Quarterly Summaries

This report gives a summary of the WB doses by quarter for 1968 (Figure A-10). The report is for the Advanced Materials Center, which is the Parks Township site. The report has columns for WB dose, extremity dose, and skin dose. Blanks indicate the worker was not monitored for a particular dose. The total quarterly and annual dose values can be useful in reconciling gaps in dosimetry information. The report of quarterly values does not necessarily indicate the worker was on a quarterly badge exchange frequency. Each line of the report is for one worker. The report indicates that all workers were monitored for WB dose, but not all were monitored for extremity dose, and only the first was monitored for skin dose.

Nuclear Materials and Equipment Corporation
Apollo, Pennsylvania
Advanced Materials Center
10 CFR 20.407 (b)(2) Annual Personnel Exposure Summary Report
Year: 1968

Date: 7/25/69

Name	Soc. Sec. No.	Whole Body Dose (Rems) Calendar Quarter 1	2	3	4	Annual Total	Hand and Forearm Dose (Rems) Calendar Quarter 1	2	3	4	Annual Total	Skin Dose (Rems) Calendar Quarter 1	2	3	4	Annual Total
		2.482	1.801	1.640	2.433	8.356	10.80	10.80	11.050	8.484	41.134	2.482	1.801	1.640	2.433	8.356
		2.047	1.318	2.121	1.588	7.074	6.205	6.00	6.00	6.00	24.205					
		1.366	1.396	1.925	2.582	7.269	9.040	6.350	4.980	5.002	25.372					
		1.926	1.095	1.886	1.795	6.702										
		2.361	2.063	1.151	1.296	6.871										
		1.898	.944	1.033	1.206	5.081	10.295	7.550	7.100	13.966	38.911					
		3.028	1.153	.815	1.343	6.339	17.450	17.650	10.200	8.905	54.205					
		1.602	.861	.698	1.212	4.373										

Signed: [redacted]
Advanced Materials Center
Health Physicist

Figure A-10. Annual report with quarterly summaries (Caldwell 1969, p. 3).

ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3.7 Annual Report by Month with Gamma and Neutron Doses

This report gives the annual summary of penetrating dose by month and quarter (Figure A-11). There are three entries for each individual and each period. The top value is the gamma WB dose, the middle value is the neutron dose, and the bottom value is the sum of the gamma and neutron doses. The report also gives the lifetime WB dose (gamma plus neutron) through the end of the reporting period.

GAMMA
NEUTRON
TOTAL

DATE December 1971 — EXTERNAL RADIATION SUMMARY

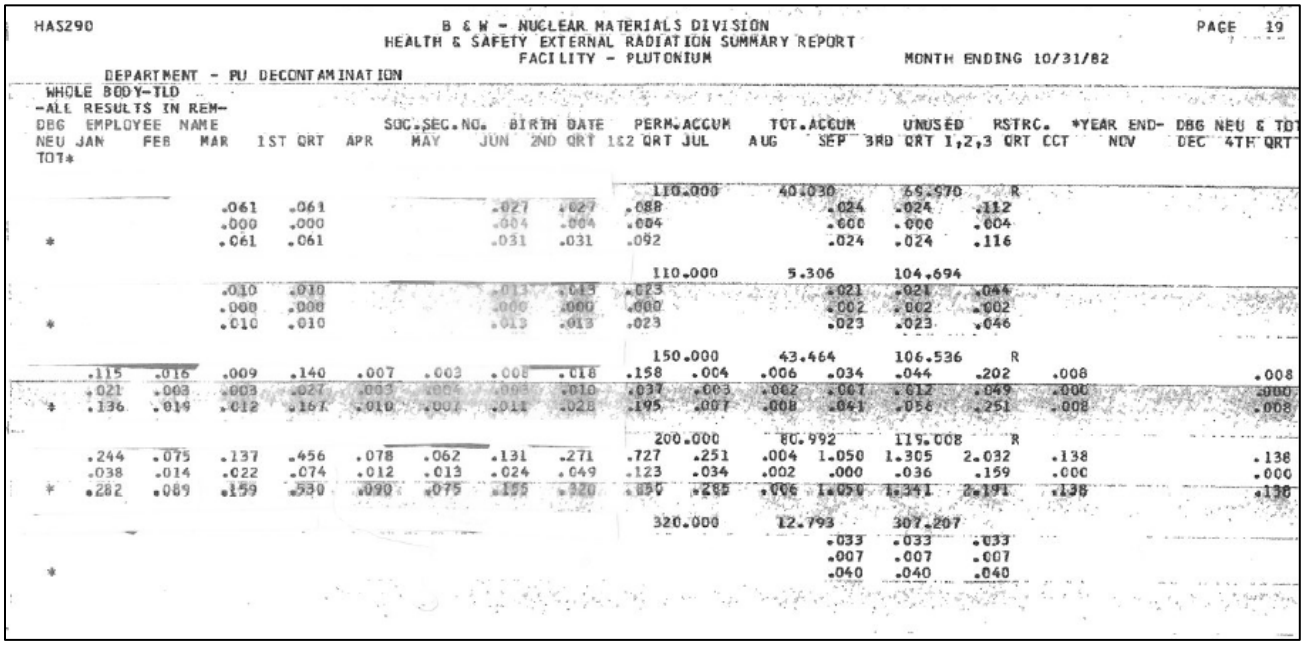
NAME	DATE OF BIRTH	PER ACC DOSE 5(N-18) REM	QTR	TOTAL DOSE THRU Dec 71	UNUSED PART OF DOSE THRU Dec	1st QUARTER (mrem) Jan	Feb	Mar	Tot	2nd QUARTER (mrem) Apr	May	Jun	Tot	3rd QUARTER (mrem) Jul	Aug	Sep	Tot	4th QUARTER (mrem) Oct	Nov	Dec	Tot	To for yea 197
						081	118	202	401	025	091	137	253	330	339	107	776	253	246	166	665	209
						008	012	020	040	075	198	282	555	427	299	143	869	182	125	142	449	191
		32.500		19.066	13.434	089	130	222	441	100	289	419	808	757	638	250	1645	435	371	308	1114	400
						470	728	465	1663	239	269	202	710	055	070	105	230	108	182	222	512	311
		81.250		46.666	34.584	047	086	047	180	362	400	324	1086	395	102	123	620	110	119	153	382	226
						517	814	512	1843	601	669	526	1796	450	172	228	850	218	301	375	894	538
						106	101	112	319	033	149	104	286	193	095	106	394	048	039	080	167	116
		63.750		26.346	37.404	011	010	025	046	084	218	232	534	256	525	288	1069	319	170	263	752	240
						117	111	137	365	117	367	336	820	449	620	394	1463	367	209	343	919	356
						062	084	149	295	023	020	018	061	023	020	017	060	023	034	012	069	485
		47.500		11.318	36.182	006	008	015	029	126	157	100	383	111	111	098	320	123	075	077	275	785
						068	092	164	324	149	177	118	444	134	131	115	380	146	109	089	344	127
						330	308	672	1310	246	143	123	512	167	441	110	718	225	208	158	591	813
		63.750		45.459	18.291	033	031	067	131	365	216	288	869	265	410	239	914	155	163	154	472	238
						363	339	739	1441	611	359	411	1381	432	851	349	1632	380	371	312	1063	551
						095	162	129	386	073	071	262	406	066	057	051	174	048	045	070	163	112
		56.250		27.843	28.407	010	016	013	039	143	244	272	659	167	148	113	428	104	080	105	289	141
						105	178	142	425	216	315	534	1065	233	205	164	602	152	125	175	452	254
						162	190	356	708	312	239	156	707	230	273	168	671	234	349	237	820	290
		145.000		31.705	113.295	016	019	036	071	431	369	292	1092	271	352	283	906	239	134	169	542	261
						178	209	392	779	743	608	448	1799	501	625	451	1577	473	483	406	1362	551
						078	078	129	285	039	052	073	164	079	068	047	194	044	039	051	134	77
		177.500		28.533	148.967	008	008	013	029	177	192	366	735	255	208	096	559	109	098	104	311	163
						086	086	142	314	216	244	439	899	334	276	143	753	153	137	155	445	241
						154	121	037	312	016	132	150	298	121	145	148	414	144	124	113	381	140
		27.500		14.824	12.676	015	012	004	031	029	192	230	451	130	130	090	350	185	085	090	360	119
						169	133	041	343	045	324	380	749	251	275	238	764	329	209	203	741	259
						101	225	250	576	107	066	085	258	152	112	134	398	067	042	112	221	155
		136.250		65.615	70.635	000	484	192	676	335	351	251	937	444	468	250	1162	428	231	166	825	360
						101	809	442	1352	442	417	336	1195	596	580	384	1560	495	273	278	1046	515
						050	073	084	207	025	046	092	163	023	032	041	096	018	021	021	060	58
		62.500		12.585	49.915	005	007	008	020	057	097	148	302	038	051	039	128	017	015	033	065	51
						055	080	092	227	082	143	240	465	061	083	080	224	035	036	054	125	104
						035	463	209	767	024	077	029	130	093	058	070	221	132	038	070	240	135
		147.500		96.560	50.940	000	1036	567	1603	099	337	100	536	138	288	127	553	510	061	151	722	341
						095	1499	776	2370	123	414	129	666	231	346	197	774	642	099	221	962	477
						050	123	062	235	081	236	241	578	033	Laid	Off	033	045	281	242	568	141
		43.750		14.564	29.186	005	012	006	023	139	300	361	800	136			136	064	228	206	498	145
						055	135	068	258	220	556	602	1378	169			169	109	509	448	1066	287

Figure A-11. Annual report by month with gamma and neutron doses (Boyd 2006b, p. 350).

ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3.8 Annual Report with Monthly Results Later Years

This report gives the dosimetry results for several workers through part of a year (Figure A-12). The dose values include three entries per period and worker. The top value is the deep gamma dose, the second value is the neutron dose, and the bottom value is the sum of the gamma and neutron doses. The entries for the first two individuals indicate the workers were on a quarterly badge exchange frequency with entries given for March, June, and September. The next two individuals were on a monthly badge exchange frequency with entries for each month. The last individual was monitored starting in September. This possibly indicates a new employee, or the start of monitoring for a previously employed worker. This report is for Parks Township plutonium facility workers involved with "Pu Decontamination." Because the period is in the residual period for the Parks Township facility, the dose values for this report would not be included in a dose reconstruction. However, if it is known the worker spent time at the Apollo facility, the values could be included in the assigned doses. The records could contain similar reports for the Apollo and Parks Township facilities for earlier years for which the dose values would be included.



HAS290 — B & W - NUCLEAR MATERIALS DIVISION — PAGE 19
HEALTH & SAFETY EXTERNAL RADIATION SUMMARY REPORT
FACILITY - PLUTONIUM — MONTH ENDING 10/31/82
DEPARTMENT - PU DECONTAMINATION
WHOLE BODY-TLD
-ALL RESULTS IN REM-

DBG NEU	EMPLOYEE NAME / SOC.SEC.NO. / BIRTH DATE	JAN	FEB	MAR	1ST QRT	APR	MAY	JUN	2ND QRT	1&2 QRT	PERM.ACCUM JUL	AUG	TOT.ACCUM SEP	3RD QRT	UNUSED 1,2,3 QRT	RSTRC. CCT	*YEAR END- NOV	DEC	DBG NEU & TOT 4TH QRT	TOT*
											110.000		40.030		69.970	R				
				.061	.061			.027	.027	.088			.024	.024	.112					
				.000	.000			.004	.004	.004			.000	.000	.004					
	*			.061	.061			.031	.031	.092			.024	.024	.116					
											110.000		5.306		104.694					
				.010	.010			.013	.013	.023			.021	.021	.044					
				.000	.000			.000	.000	.000			.002	.002	.002					
	*			.010	.010			.013	.013	.023			.023	.023	.046					
											150.000		43.464		106.536	R				
		.115	.016	.009	.140	.007	.003	.008	.018	.158	.004	.006	.034	.044	.202	.008			.008	
		.021	.003	.003	.027	.003	.004	.003	.010	.037	.003	.002	.007	.012	.049	.000			.000	
	*	.136	.019	.012	.167	.010	.007	.011	.028	.195	.007	.008	.041	.056	.251	.008			.008	
											200.000		80.992		119.008	R				
		.244	.075	.137	.456	.078	.062	.131	.271	.727	.251	.004	1.050	1.305	2.032	.138			.138	
		.038	.014	.022	.074	.012	.013	.024	.049	.123	.034	.002	.000	.036	.159	.000			.000	
	*	.282	.089	.159	.530	.090	.075	.155	.320	.850	.285	.006	1.050	1.341	2.191	.138			.138	
											320.000		12.793		307.207					
													.033	.033	.033					
													.007	.007	.007					
	*												.040	.040	.040					

Figure A-12. Annual report with monthly results later years (BWXT 1982b, p. 33).

ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3.9 Worker Lifetime Summary

The records could include a summary of the exposures received by an individual during all periods of employment (Figure A-13). This report includes summaries of external doses. External doses from gamma, beta, and neutrons are usually included. The report includes the employment period and the monitoring periods. The sample indicates an employment period that started and ended outside the monitoring period (Period of Exposure). This information can be useful in reconciling gaps in dosimetry records. The report also indicates the SNM license number under which the work was conducted. License SNM-145 indicates employment at the Apollo facility, and license SNM-414 indicates employment at the Parks Township facility. Some employees are indicated to have worked at both facilities.

BABCOCK & WILCOX
NUCLEAR MATERIALS AND MANUFACTURING DIVISION

OCCUPATIONAL RADIATION EXPOSURE HISTORY

Name _____ Social Security No. _____

Date of Birth _____ Period of Employment 2-15-60 to 11-30-80

License No. SNM-145 XXX

SNM-414 ☐

EXTERNAL RADIATION EXPOSURE INFORMATION

Period of Exposure (From-To)	Part of Body	Gamma Dose Mrem	Beta Dose Mrem	Neutron Dose Mrem	Total Dose Mrem
11-18-68 to 6-30-75	Whole Body	342	-	281	623
11-18-68 to 6-30-75	Skin	-	186	-	186
11-18-68 to 6-30-75	Extremities	-	-	-	-

Figure A-13. Worker lifetime summary (Boyd 2006d, p. 9).

A.3.10 Cumulative Dose through November of 1971

This report was prepared to show the lifetime doses to workers when the company was purchased by B&W (Figure A-14). The dose values are lifetime doses through November of 1971. Values are given for shallow dose (SBG), deep gamma dose (DBG), and neutron dose. The total WB dose is the sum of the deep gamma and neutron doses. Extremity exposures are also reported. The lifetime totals can be useful in reconciling gaps in dosimetry information.

ATTACHMENT A
DOSIMETRY REPORTS (continued)

External Exposure of Employees Prior to B&W Acquisition 11/[illegible]1, not in personnel exposure file.

Soc. Sec. No.	Per. of Exposure	Whole Body SBG	DBG + N = Total	Extremities Gamma	Neutron	Total
	5/4/72	000	090 000 090		None	
	7/1/71	000	002 017 019		None	
			370 370			
	2/15/61	312	11822 673 12495	4780	000	4780
	9/11/68	000	630 015 645		None	
	7/30/68	000	2173 186 2359	4950	000	4950
	7/22/68	217	6005 919 6924		None	
	3/1/64	084	5651 119 5770		None	
	4/1/67	640	11447 2160 13607	12524	1093	13617
	8/5/68	000	500 000 500		None	
	2/1/64	000	11855 775 12630	360	000	360
			250 250			
			102 102			
	7/1/71	003	013 013 026		None	
[redacted]						
	10/6/68	000	4234 352 4586	3128	203	3331
	6/7/67	079	11952 1138 13090	11497	372	11869
	10/11/66	038	5890 223 6113		None	
	7/1/65	018	741 013 754		None	
	11/1/67	1158	15294 3093 18387	14698	2244	16942
	12/15/61	037	1887 032 1919		None	
	6/17/63	211	11125 899 12024		None	
	2/15/61	172	7697 310 8007	1647	000	1647
	10/1/65	000	3860 000 3860		None	
	2/8/67	000	10545 004 10549		None	
	7/1/71	085	054 040 094		None	
	6/22/67	000	1380 000 1380		None	
	2/15/61	2103	16006 1098 17104	2740	000	2740
	2/15/61	003	4270 720 4990		None	
			10190 10190			
	7/1/71	092	054 000 054		None	
	1/16/68	010	148 013 161		None	
	1/1/68	388	10750 1488 12238	14525	284	14809
	11/18/68	000	145 022 167		None	
	1/29/61	682	36641 29423 66064	54387	41917	96304
	7/1/71	002	008 000 008		None	
	9/15/62	1252	40345 3440 43785	54372	3033	57405
	7/1/71	000	004 017 021		None	
	10/7/68	000	5281 453 5734	3490	254	3744
	1/1/68	000	7319 629 7948	15346	373	15719
	11/9/66	000	2400 000 2400		None	
	1/1/64	033	1018 024 1042			
	4/19/67	084	9305 136 9441			
	5/1/71	020	021 006 027			
	7/1/71	000	004 000 004			
	5/20/63	000	6040 000 6040			
	6/17/63	000	16843 766 17609			
	9/16/66	013	664 011 675			
	2/9/67	040	1414 252 1666			
	10/15/61	017	1198 034 1232			
	10/15/61	014	1085 075 1160		none	
	11/1/66	023	682 019 701		None	
	10/12/67	000	13548 1335 14883	18089	709	18798
	10/30/73					
			590 590			
	9/16/68	000	4515 499 5014	7960	299	8259

These Exposure totals were transferred to B&W Computer personnel sheet listing under Column Past exposure accumulation.

Figure A-14. Cumulative dose through November 1971 (Boyd 2006a, p. 3).

ATTACHMENT A
DOSIMETRY REPORTS (continued)

A.3.11 Plutonium Guard Summary 1971 – 1977

A summary of doses for workers who served as plutonium guards at the Parks Township plutonium laboratory during 1971 through 1977 is included in the NUMEC records (Figure A-15). If dosimetry records are not included for a worker that was involved on the security force at the Parks Township plutonium laboratory during this period, then an analysis of the information in this report could be of use in estimating external dose.

Dosimeter Results of Pu Guards in mrem.

Quarter	1971 2	3	4	1972 1	2	3	4	1973 1	2	3	4	1974 1	2	3	4	1975 1	2	3	4	1976 1	2
	017	017	019	021	020	023	023	046	051	035	047	059	068	083	046	021	033	019	041	008	004
	-	-	-	-	-	-	-	-	-	-	-	-	-	091	093	027	042	026	040	015	017
	016	018	015	023	015	025	026	049	036	036	046	030	023	219	127	033	031	021	037	027	016
	-	-	-	-	-	-	-	-	-	-	-	-	-	042	045	025	026	031	038	010	002
	024	018	026	016	019	024	030	023	032	029	034	037	051	028	032	021	032	031	031	013	004
	023	012	018	023	021	024	024	062	058	023	021	036	068	169	048	027	033	036	036	010	004
	012	020	023	026	023	026	027	035	050	030	030	036	058	021	038	003	021	032	033	009	008
	062	023	025	032	016	024	041	054	051	063	030	036	023	021	038	018	029	028	041	010	004
	021	030	036	028	016	029	025	046	059	044	050	059	071	067	062	040	039	028	042	016	005
	009	015	018	024	021	034	016	054	033	032	038	033	079	039	056	028	039	031	046	014	006
	015	025	017	024	022	029	042	051	050	032	056	043	055	046	063	037	042	029	036	015	007
	025	036	029	017	-	-	-	-	017	029	041	051	055	037	101	119	033	030	044	069	006
	037	029	023	021	019	035	031	043	058	026	042	051	160	028	038	032	032	028	035	013	005
	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	032	039	012	006
	-	-	-	-	-	-	-	-	-	-	-	-	-	043	041	028	060	042	042	013	008
	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	024	030	036	017	006
	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	030	035	033	038	016	006
	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	020
	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	035	012	007
	-	-	-	-	-	-	-	-	-	-	-	-	-	037	068	020	039	037	041	009	006
	-	-	-	-	-	-	-	-	-	-	-	-	-	028	046	020	021	028	028	008	004

Removed individual dosimeters at the end of the second quarter of 1976 and issued one dosimeter to be worn by guards during their routine patrol of the Pu plant.

Guard Patrol Dosimeter:	3/76	4/76	1/77	2/77	3/77	4/77
	068	Lost	042	070	039	046

No dose recorded to the individuals. Dosimeter issued primarily in case of criticality. Reason for removing dosimeters from individuals is due to lack of sufficient dosimeters. 10 CFR 20 states that anyone receiving less than 25% of the quarterly dose is not required to wear a dosimeter.

Figure A-15. Plutonium guard summary, 1971 to 1977 (Boyd 2006a, p. 6).

A.4 GUIDANCE FOR DOSE ASSIGNMENT

The assignment of dose depends on the information available. If the claim files have reported dosimeter readings for all dosimeter cycles, then the assignment of dose is not a problem because the reported dosimeter values can be used as provided. The dose reconstructor still needs to review the dosimeter reports to ensure that all zero dosimeter readings are included because some of the reports have a blank rather than a reported value of zero. The dosimeter reports need to be carefully reviewed along with the employee's work locations to determine if there was potential for neutron exposure and if neutron dose needs to be assigned. The NUMEC monthly calculation sheets (Figures A-5 and A-6), NUMEC monthly report sheets (Figure A-7), Landauer reports (Figures A-1, A-2, and A-3), Eberline reports (Figure A-4), annual summary reports (Figures A-11 and A-12), and

ATTACHMENT A
DOSIMETRY REPORTS (continued)

lifetime reports (Figures A-13 and A-14) could give an indication if the employee was monitored for neutron exposure.

The claims usually do not have data for all dosimeter assignments. For some claims, there might not be complete agreement among the dose values reported in NUMEC, Landauer, and Eberline reports. When the dosimeter information is incomplete, the dose reconstructor should attempt to reconcile the available information to ensure that the reported total dose (if available) has been assigned. For employees with both deep photon and neutron exposures, the relative amount of each dose type needs to be reviewed when the missing dose is assigned to periods with missing dosimeter readings. Most reported WB dose values for work at the Parks Township plutonium laboratory and source fabrication facility potentially include neutron dose.