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Feed Materials Production Center – Occupational External Dose

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
04/20/2004	00	New technical basis document for Fernald Environmental Management Project (FEMP) – Occupational External Dose. Incorporates formal internal and NIOSH review comments. First approved issue. Initiated by Leo G. Faust.
03/25/2014	01	Revision initiated to resolve NIOSH comments from 2006 and to incorporate the coworker data from ORAUT-OTIB-0073 and cancel that document. Added Purpose, Scope, and Special Exposure Cohort sections. Included additional neutron information. Updated and added references including but not limited to ORAUT-OTIB-0017, ORAUT-OTIB-0012, ORAUT-OTIB-0020, ORAUT-OTIB-0055, and DCAS-TIB-0013. Incorporated changes in response to formal internal and NIOSH review comments and Sanford Cohen & Associates matrix issues. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Matthew H. Smith and Karen S. Kent.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ANSI	ANSI International (formerly American National Standards Institute)
CFR	Code of Federal Regulations
cm	centimeter
d	day
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
DOELAP	DOE Laboratory Accreditation Program
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
FEMP	Fernald Environmental Management Project
FMPC	Feed Materials Production Center
FR	Federal Register
ft	foot
g	gram
hr	hour
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiological Units
IIR	incident investigation report
in.	inch
INEL	Idaho National Engineering Laboratory
IREP	Interactive RadioEpidemiological Program
keV	kilovolt-electron, 1,000 electron volts
LOD	limit of detection
MDL	minimum detection limit
MeV	megavolt-electron, 1 million electron volts
mg	milligram
mm	millimeter
mR	milliroentgen
mrad	millirad
mrem	millirem
mrep	millirep
MTU	metric tons of uranium
NAA	neutron activation analysis
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NLO	National Lead Company of Ohio
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory

POC	probability of causation
R	roentgen
REF	Radiation effectiveness factor
rpm	revolutions per minute
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
TBD	technical basis document
TLD	thermoluminescent dosimeter
U.S.C.	United States Code
wk	week
§	section or sections

6.1 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 sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer facility" or a "Department of Energy [DOE] facility" as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility 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 Department of Energy (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program)" [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled "Exposure in the Performance of Duty." That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer "shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation[1]] guidelines established under subsection (c) …" [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation, 42 C.F.R. Pt. 82) restrict the "performance of duty" referred to in 42 U.S.C. § 7384n(b) to nuclear weapons work (NIOSH 2010a).

The statute also includes a definition of a DOE facility that excludes "buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program" [42 U.S.C. § 7384l(12)]. While this definition excludes Naval Nuclear Propulsion Facilities from being covered under the Act, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled "Exposure in the Performance of Duty"] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally-derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external occupational radiation exposures are considered valid for inclusion in a dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived (NIOSH 2010a):

- Background radiation, including radiation from naturally occurring radon present in conventional structures
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

[1] The U.S. Department of Labor (DOL) is ultimately responsible under the EEOICPA for determining the POC.

6.1.1 Purpose

The purpose of this document is to describe the external dosimetry systems and practices at the Feed Materials Production Center (FMPC) and to assist in the evaluation of occupational external exposures from processes that occurred at FMPC. FMPC was also known as the Fernald Environmental Management Project (FEMP), the Fernald Closure Project (FCP), and is now the Fernald Preserve. This document discusses the dosimeters FMPC used along with their exchange periods and other technical parameters of their use from the early 1950s to the 2000s. It provides supporting technical data to evaluate, with assumptions favorable to the claimant, occupational external doses that can reasonably be associated with radiation exposures to both monitored and unmonitored workers.

6.1.2 Scope

External radiation dosimetry refers to the measurement of radiation external to (i.e., outside) the body such as occurs with medical X-rays, cosmic rays, or radiation from naturally occurring radioactivity in the earth. Primary types of radiation typically significant to exposure to workers are beta, photon (i.e., X-ray and gamma) and neutron radiation, respectively, each with characteristic properties of origin and interaction with matter. Facilities that contain natural or manmade radionuclides have the potential for external radiation exposure of workers. External radiation dosimetry can be contrasted with internal radiation dosimetry, which is concerned with quantification of radiation exposure from radionuclides internal (i.e., inside) to the body.

This document provides a technical basis to evaluate external radiation exposure to workers that can reasonably be associated with Fernald operations under EEOICPA legislation. Consistent with NIOSH guidelines, this document identifies options to adjust historical recorded occupational external dose to account for current scientific methods and protection factors. The methods and concepts of measuring occupational external doses to workers have evolved since the beginning of Fernald operations. In particular, this document presents the methods to prepare worker dose information for input to the NIOSH Interactive RadioEpidemiological Program (IREP).

6.1.3 Special Exposure Cohort

The Secretary of the U.S. Department of Health and Human Services has designated three classes of employees at Fernald as additions to the Special Exposure Cohort (SEC):

> *All employees of DOE, its predecessor agencies, and their contractors, or subcontractors who worked at the Feed Materials Production Center in Fernald, Ohio, from January 1, 1968 through December 31, 1978, for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees included in the SEC* (77 FR 150; August 3, 2012).

It was determined that NIOSH lacked the sufficient information to allow it to estimate with sufficient accuracy the potential internal doses from exposure to thorium, which employees at this facility may have been subjected (Sebelius 2012).

> *All employees of the DOE, its predecessor agencies, and their contractors and subcontractors who worked at the Feed Materials Production Center in Fernald, Ohio, from January 1, 1954, through December 31, 1967, for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other*

> *classes of employees included in the Special Exposure Cohort* (78 FR 229; November 27, 2013).

It was determined that NIOSH lacked the sufficient information to allow it to estimate with sufficient accuracy the potential internal doses from exposure to thorium, to which employees working at this facility may have been subjected (Sebelius 2013a).

> *All employees of the Feed Materials Production Center in Fernald, Ohio, who were not employed by National Lead of Ohio, NLO, or the Department of Energy or its predecessor agencies, who worked at FMPC from January 1, 1951, through December 31, 1983, for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees included in the Special Exposure Cohort (78 FR 229; November 27, 2013).*

It was determined that NIOSH lacked the sufficient information to allow an estimate with sufficient accuracy the potential internal doses from exposure to uranium, to which employees of FMPC working at this facility may have been subjected

This dose reconstruction infeasibility for the period 1951 through 1983 applies only to subcontractors. NIOSH has access to an electronic data set that contains the results of the uranium urinalysis bioassay program for all the years of FMPC operations, and the overwhelming majority of employees of the prime contractor (National Lead of Ohio, later named NLO Inc.) have results in the bioassay data set. However, the data set does not contain bioassay results for employees of companies other than the prime contractor (i.e., non-prime contractor employees). NIOSH has obtained a limited number of bioassay samples from non-prime contractor employees through data captures, but cannot be certain that all non-prime contractor employees' bioassay data were retained by the site or captured by data capture efforts. Additionally, there are some reasons to conclude that the prime contractor did not consistently evaluate whether non-prime contractor employees should be monitored for radiation exposure because of the transitory nature of their work (Sebelius, 2013 b).

Dose reconstruction guidance in this document for the period before January 1, 1984, is presented to provide a technical basis for partial dose reconstructions for claims not compensated under the SEC (i. e., nonpresumptive cancers and SEC employment <250 days). Although it is not possible to completely reconstruct internal radiation doses for all workers for the period January 1, 1951, through December 31, 1983, NIOSH has determined, and HHS has concurred, that it is feasible to reconstruct external radiation doses for all FMPC workers for the period from January 1, 1951, through December 31, 1983 (Sebelius 2013b).

6.2 OVERVIEW OF SITE OPERATIONS

Having begun operations in 1951, FMPC was a rather late addition to the United States nuclear weapons complex, but it was a very important one. The site included chemical processing, foundries, machine shops, storage yards, a uranium recycler, and a variety of other capabilities. In addition, FMPC was assigned a mission that to date had never been performed before, at least on such a large scale. Processing such large quantities of radioactive material, in this case uranium up to 10,000 MTU annually along with small amounts of thorium (ASI ca. 1986, pp. 4–5) – with a staff of up to almost 2,900 – was a new type of endeavor.

When operational, FMPC was a large integrated facility that produced uranium metal feed materials in DOE defense program facilities throughout the United States (WMCO 1988, p. 3). FMPC used a number of processes that involved a variety of forms of uranium, such as uranium ore concentrate, uranium hexafluoride (UF_6), and recycled uranium scrap from throughout the DOE complex. The

products were “variously sized, highly purified uranium metal forms of assorted standard isotopic assays” (WMCO 1988, p. 6) ranging from depleted to slightly enriched uranium metal products. The primary facilities, referred to as plants, are described in detail in the latest revision of ORAUT-TKBS-0017-2, *Technical Basis Document for the Fernald Environmental Management Project (FEMP) – Site Description* (ORAUT 2004), and addressed in this document by reference. The radiological hazards associated with these processes and products resulted from the radioactivity of uranium, thorium, their progeny, and in some instances impurities in the recycled material.

The occupational doses FMPC workers received were a function of the physical location of the workers on the site, the process, and the type and quantities of materials. The dose also varied with the “age” (with respect to radiological decay) of the material being processed. The concerns with aging and processing involve the disequilibria of the radioactive material progeny that often are affected by the process to which the parent is subjected. For example, the temperature of the process can volatilize a progeny in the parent radioactive decay chain (such as thorium) which in turn can, until equilibrium is reestablished, affect the resulting dose rates. These and other factors are often encountered in the processing of radiological materials, all of which can affect the amount and magnitude of external dose.

Many cycles of activities took place throughout the operational lifetime of FMPC. The throughput of material varied considerably as did the sources of feed materials, one of which was ore from the Belgian Congo. This ore, pitchblende, contained large quantities of radium that required shielding (EPA 2000). Wastes from this process were stored on the site in the K-65 Silos. These silos also received waste from a site near Niagara Falls, New York. The silos became a large contributor to site background dose rates.

The introduction of recycled uranium at FMPC, mostly from the DOE Hanford Site, started in 1958 (DOE 2003, p. 75) and reached a peak in 1970. DOE (2000, p. 26) indicates that Fernald received, shipped, and processed uranium products from 1961 to 1989 that used recycled uranium feed stocks that contained constituents of concern. This material contained some “carryover” fission products and minute quantities of transuranic elements. One of the carryover products of interest for occupational external exposure is ^{99}Tc, which has 0.292 MeV beta energy at a yield of 100%. It is important because its specific activity is approximately 4 orders of magnitude greater than that of uranium, and FMPC received an estimated 135 kilograms of ^{99}Tc. Technetium-99 is a major contributor to shallow dose because it contributes most of its energies at the tissue depths that are used in defining shallow dose. The same is true of the similar beta energy in ^{234}Th, the first progeny of the ^{238}U decay series.

A more penetrating beta energy occurs with the decay of protactinium, the product of ^{234}Th decay (the first progeny radionuclide of ^{238}U decay), which undergoes a metastable state as ^{234m}Pa, which in turn decays (0.13% of time) to ^{234m}Pa with a 2.29 MeV beta at a yield of 98% and two lesser energy betas of 1 MeV each. Complete decay chain diagrams are provided in *Table of Isotopes* (Lederer and Shirley 1978) or the decay diagrams of the *Radiological Health Handbook* (PHS 1970) in figures at the end of this section.

There are several beta particles in the ^{235}U decay chain, most of which are associated with ^{231}Th, the first progeny of ^{235}U. The maximum beta energy is 0.3 MeV ±0.005, and there are several lesser energies. In addition, several gamma and X-rays are emitted during the decay process. Table 6-1 lists the radiations of major concern.

Workers at FMPC who might have been exposed to the sources of radiation that are discussed in this technical basis document (TBD) were employed during the period starting in late 1951. The peak production years occurred from the late 1950s to mid-1970s, peaking in 1960 at an annual rate of approximately 10,000 metric tons of uranium (MTU) (Voillequé et al. 1995, p. 306). Individual worker monitoring methods were implemented by work locations, length of time, and facility monitoring. No

Table 6-1. Uranium beta and gamma emissions of interest (MeV).

Radionuclide	Max. beta energy (max.)	Gamma energy
U-238	None	None
Th-234	0.103 (21%)	0.063 (3.5%)
	0.193 (79%)	0.093 (4%)
Pa-234m	2.29 (98%)	0.765 (0.3%)
		1.00 (0.6%)
U-235	None	0.144 (11%)
		0.186 (54%)
		0.205 (5%)
Th-231	0.140 (45%)	0.026 (2%)
	0.220 (15%)	0.084 (10%)
	0.305 (40%)	
U-234	None	0.053 (0.2%)

early radiological policy documentation at FMPC was found during the TBD investigations. However, individual doses from personal dosimeters worn by the workers are available, and this TBD pertains to the analysis of these records. OCAS-IG-001, *External Dosimetry Implementation Guideline* (NIOSH 2007) has identified these records to represent the highest quality records for retrospective dose assessments.

6.3 BASIS OF COMPARISON

Occupational whole-body doses at the time of FMPC startup in 1951 were controlled to 0.3 R/wk and an extremity dose of 1.5 R/wk. The annual limit for maximum whole-body dose for any 1-year period was limited to 12 rem, and the annual extremity limit was 75 rem; both values had associated administrative limits that were fractions of the annual limits per calendar quarter. In 1955, the whole-body dose limits were reduced to 3 rem per 13 weeks, not to exceed 5 rem per year. The terms roentgen, rad, rem, and rep (roentgen equivalent physical) often are used interchangeably (Heatherton 1960a, p. 143), and in this document the favorable to claimant assumption was made that they are considered equal.

Various radiation dose concepts and quantities have been used to measure and record occupational dose since the start of FMPC in 1951. A basis of comparison for dose reconstruction is the concept of personal dose equivalent *Hp(d)*, where *d* identifies the depth in millimeters and represents the point of reference for dose in tissue. For penetrating radiation of significance to whole-body dose (e.g. high-energy photons), *d* = 10 mm and is noted as *Hp(10)*. For weakly penetrating radiation of significance to skin dose, *d* = 0.07 mm, and is noted as *Hp(0.07)*. These are the radiation quantities recommended by the International Commission on Radiological Units (ICRU) in Report 51 (ICRU 1993), and the radiation quantities used in the DOE Laboratory Accreditation Program (DOELAP; DOE 1986) to accredit personal dosimetry programs at DOE sites. FMPC was the first DOE site to become DOELAP accredited, in 1987. While this accreditation is of significant value in validating data from 1987 and later, there is no analogous validation for data from before 1987 and especially back to the 1950s. The accuracies of the dosimetry system(s), their recorded doses, and their comparability to current systems depend on:

- Administrative practices based on technical, statutory, and administrative requirements;
- Workplace radiation fields, materials, quantities, etc.;
- Dosimetry technologies and calibrations;
- Process technologies; and
- Training programs and practices.

6.4 HISTORICAL DOSIMETRY PRACTICES

This section provides a summary of the external dosimetry practices FMPC used throughout its history of operations and activities.

6.4.1 Administrative Practices

FMPC started operation in October 1951 (DOE 2000, p. 737) using a variety of chemical and metallurgical processes to perform its mission of supplying uranium metal products to the U.S. Atomic Energy Commission (AEC) and its successor agencies. This included fuel cores for production reactors at the Hanford and Savannah River Sites and uranium metal products for Y-12 at Oak Ridge National Laboratory (ORNL) and Rocky Flats Plant operations. Raw uranium-bearing ores, ore concentrates, and later recycled uranium compounds were received primarily from the Hanford and Savannah River Sites.

Some of the raw uranium ore (especially ores from the Belgium Congo) contained considerable amounts of radium that later became a waste problem, while recycled material had trace quantities of transuranic elements and some fission products. Fernald receipts data (DOE 2003, p. 75) presented in Table 6-2 lists the quantities of recycled materials along with other radionuclides and their sources from key shipping sites.

Figures 6-1 and 6-2 present decay schemes for ^{235}U and ^{238}U, respectively. Of prime importance is Figure 6-2 (^{238}U) because its decay scheme includes ^{234m}Pa, a major contributor to FMPC worker dose because FMPC processed only low-enriched uranium (i.e., <2% ^{235}U). For the most part, only uranium that had been subjected to processing (primarily concentration) that resulted in the disruption of its decay chain was the main feed material. Typically, only the first two progeny of ^{238}U are of importance, and in particular ^{234m}Pa with its 2.29 MeV beta energy 98% of the time. The FMPC processed only low-enriched uranium (i.e. typically <2% ^{235}U) but it varied between <0.7 and 5% (in limited quantities).

FMPC also became the storage site for thorium in the United States and processed some thorium into reactor fuel for the weapons complex. The radiological properties of thorium are different from those of uranium because it has higher energy gamma rays and a shorter time to reestablish equilibrium with its progeny after processing.

The dosimetry requirements for uranium and thorium are similar; that is, the first two progeny of ^{232}Th decay by emission of beta particles along with a few photons. Because the progeny are volatile, when thorium is processed the equilibrium is disrupted and considerable time is required for the processed material to again reach equilibrium. Figure 6-3 provides the decay scheme for ^{232}Th. Later in plant operations, ^{99}Tc became a contributor to external dose when the FMPC started processing recycled materials that included small quantities of transuranic elements and fission products (e.g., ^{237}Np, plutonium, and ^{99}Tc) (DOE 2003, p. 21). The transuranic elements shown in Table 6-2 contributed to the internal dose, but only on a limited basis. Technetium contributed primarily to external exposure (skin or extremity) dose due to the quantities present, and probably was contacted through contamination of apparel (in particular gloves).

Dosimeters for production workers were always used at FMPC. After the security credential and the dosimeter were combined, all employees wore them (WMCO 1987). However, exposures have not always been determined for all employees. During certain periods, female employees were not routinely monitored (Dugan 1974). Periods when male and female employees were monitored were:

- 1951 to 1960, male employees only;
- 1961 to 1968, male and female employees;

Table 6-2. Recycled materials receipts (DOE 2003, p. 75).

	Key shipping sites (MTU)				Contaminants			
Year	Hanford	Savannah River	West Valley	Total	Plutonium (g)	Neptunium (g)	^{236}U (kg)	Technetium (kg)
1953	0			0	0.0	0		0.0
1954	0			0	0.0	0		0.0
1955	0			0	0.0	0		0.0
1956	0			0	0.0	0		0.0
1957	0			0	0.0	0		0.0
1958	5			5	0.0	2		0.0
1959	19			19	0.1	7		0.2
1960	21			21	0.1	7		0.2
1961	50	1		51	0.2	18		0.4
1962	170			170	0.8	60		1.4
1963	1,002			1,002	4.5	351		8.0
1964	1,097	1		1,097	4.9	384		8.8
1965								
1966	2,025	202	48	2,274	9.7	752		17.1
1967	1,458	859	101	2,418	8.7	679		15.0
1968	1,692	412	168	2,273	9.1	690		15.5
1969	1,870	706	124	2,700	10.3	799		17.8
1970	2,237	22	78	2,336	10.4	798		18.2
1971	0	60	69	129	0.4	21		0.4
1972		0	31	31	0.1	4		0.1
1973			3	3	0.0	0		0.0
1974		15		15	0.0	3	17	1.2
1975								
1976		12		12	0.0	2	14	1.0
1977	0	14		15	0.0	3		0.4
1978		28		28	0.1	5	23	1.0
1979		66		66	0.1	12		0.2
1980		36		36	0.1	7	17	2.5
1981		0		0	0.0	0		0.0
1982								
1983		23		23	0.0	4		0.1
1984	706	27		733	3.2	252		5.7
1985	918			918	4.1	321		7.3
1986	1,151			1,151	5.2	403		9.2
1987	314			314	1.4	110		2.5
1988	123			123	0.6	43		1.0
1989	0			0	0.0	0		0.0
Total	**14,859**	**2,486**	**621**	**17,966**	**74.3**	**5,735**	**71**	**135**

Table 6-3. Dosimeter characteristics.

Years	Dosimeter	Filters	MDL (mrad)	Routine exchange
1951–1953	Two-element film	Open, Cd 1 mm	40	Weekly
1954–1958	ORNL dosimeter	Cu, Cd, plastic, Pb, open	30	Biweekly
1959–1985	ORNL dosimeter	Cu, Cd, plastic, Pb, open	30	Monthly
1985–1992	Commercial Panasonic TLD	Multiple	5	Monthly
1993–present	Commercial Panasonic TLD	Multiple	5	Quarterly

Actinium Series (4n + 3)*

Nuclide	Historical name	Half-life	Major radiation energies (MeV) and intensities†		
			α	β	γ
$^{235}_{92}U$	Actinouranium	7.1×10^{8}y	4.37 (18%) 4.40 (57%) 4.58c‡ (8%)	---	0.143 (11%) 0.185 (54%) 0.204 (5%)
$^{231}_{90}Th$	Uranium Y	25.5h	---	0.140 (45%) 0.220 (15%) 0.305 (40%)	0.026 (2%) 0.084c (10%)
$^{231}_{91}Pa$	Protoactinium	3.25×10^{4}y	4.95 (22%) 5.01 (24%) 5.02 (23%)	---	0.027 (6%) 0.29c (6%)
$^{227}_{89}Ac$	Actinium	21.6y	4.86c (0.18%) 4.95c (1.2%)	0.043 (~99%)	0.070 (0.08%)
98.6% / 1.4%					
$^{227}_{90}Th$	Radioactinium	18.2d	5.76 (21%) 5.98 (24%) 6.04 (23%)	---	0.050 (8%) 0.237c (15%) 0.31c (8%)
$^{223}_{87}Fr$	Actinium K	22m	5.44 (~0.005%)	1.15 (~100%)	0.050 (40%) 0.080 (13%) 0.234 (4%)
$^{223}_{88}Ra$	Actinium X	11.43d	5.61 (26%) 5.71 (54%) 5.75 (9%)	---	0.149c (10%) 0.270 (10%) 0.33c (6%)
$^{219}_{86}Rn$	Emanation Actinon (An)	4.0s	6.42 (8%) 6.55 (11%) 6.82 (81%)	---	0.272 (9%) 0.401 (5%)
$^{215}_{84}Po$	Actinium A	1.78ms	7.38 (~100%)	0.74 (~.00023%)	---
~100% / .00023%					
$^{211}_{82}Pb$	Actinium B	36.1m	---	0.29 (1.4%) 0.56 (9.4%) 1.39 (87.5%)	0.405 (3.4%) 0.427 (1.8%) 0.832 (3.4%)
$^{215}_{85}At$	Astatine	~0.1ms	8.01 (~100%)	---	---
$^{211}_{83}Bi$	Actinium C	2.15m	6.28 (16%) 6.62 (84%)	0.60 (0.28%)	0.351 (14%)
0.28% / 99.7%					
$^{211}_{84}Po$	Actinium C'	0.52s	7.45 (99%)	---	0.570 (0.5%) 0.90 (0.5%)
$^{207}_{81}Tl$	Actinium C''	4.79m	---	1.44 (99.8%)	0.897 (0.16%)
$^{207}_{82}Pb$	Actinium D	Stable	---	---	---

*This expression describes the mass number of any member in this series, where n is an integar.
Example: $^{207}_{82}Pb$ (4n + 3).....4(51) + 3 = 207
†Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series.
‡Complex energy peak which would be incompletely resolved by instruments of moderately low resolving power such as scintillators.

Data taken from: Table of Isotopes and USNRDL-TR-802.

Figure 6-1. Uranium-235 decay series (PHS 1970, p. 124).

Uranium Series (4n + 2)*

Nuclide	Historical name	Half-life	Major radiation energies (MeV) and intensities†		
			α	β	γ
$^{238}_{92}U$	Uranium I	4.51×10^{9}y	4.15 (25%) 4.20 (75%)	---	---
$^{234}_{90}Th$	Uranium X_1	24.1d	---	0.103 (21%) 0.193 (79%)	0.063c‡ (3.5%) 0.093c (4%)
$^{234}_{91}Pa^{m}$	Uranium X_2	1.17m	---	2.29 (98%)	0.765 (0.30%) 1.001 (0.60%)
99.87% | 0.13%					
$^{234}_{91}Pa$	Uranium Z	6.75h	---	0.53 (66%) 1.13 (13%)	0.100 (50%) 0.70 (24%) 0.90 (70%)
$^{234}_{92}U$	Uranium II	2.47×10^{5}y	4.72 (28%) 4.77 (72%)	---	0.053 (0.2%)
$^{230}_{90}Th$	Ionium	8.0×10^{4}y	4.62 (24%) 4.68 (76%)	---	0.068 (0.6%) 0.142 (0.07%)
$^{226}_{88}Ra$	Radium	1602y	4.60 (6%) 4.78 (95%)	---	0.186 (4%)
$^{222}_{86}Rn$	Emanation Radon (Rn)	3.823d	5.49 (100%)	---	0.510 (0.07%)
$^{218}_{84}Po$	Radium A	3.05m	6.00 (~100%)	0.33 (~0.019%)	---
99.98% | 0.02%					
$^{214}_{82}Pb$	Radium B	26.8m	---	0.65 (50%) 0.71 (40%) 0.98 (6%)	0.295 (19%) 0.352 (36%)
$^{218}_{85}At$	Astatine	~2s	6.65 (6%) 6.70 (94%)	? (~0.1%)	---
$^{214}_{83}Bi$	Radium C	19.7m	5.45 (0.012%) 5.51 (0.008%)	1.0 (23%) 1.51 (40%) 3.26 (19%)	0.609 (47%) 1.120 (17%) 1.764 (17%)
99.98% | 0.02%					
$^{214}_{84}Po$	Radium C'	164μs	7.69 (100%)	---	0.799 (0.014%)
$^{210}_{81}Tl$	Radium C''	1.3m	---	1.3 (25%) 1.9 (56%) 2.3 (19%)	0.296 (80%) 0.795 (100%) 1.31 (21%)
$^{210}_{82}Pb$	Radium D	21y	3.72 (.000002%)	0.016 (85%) 0.061 (15%)	0.047 (4%)
$^{210}_{83}Bi$	Radium E	5.01d	4.65 (.00007%) 4.69 (.00005%)	1.161 (~100%)	---
~100% | .00013%					
$^{210}_{84}Po$	Radium F	138.4d	5.305 (100%)	---	0.803 (0.0011%)
$^{206}_{81}Tl$	Radium E''	4.19m	---	1.571 (100%)	---
$^{206}_{82}Pb$	Radium G	Stable	---	---	---

*This expression describes the mass number of any member in this series, where n is an integer.
Example: $^{206}_{82}Pb$ (4n + 2).....4(51) + 2 = 206
†Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series.
‡Complex energy peak which would be incompletely resolved by instruments of moderately low resolving power such as scintillators.

Data taken from: Table of Isotopes and USNRDL-TR-802.

Figure 6-2. Uranium-238 decay series (PHS 1970, p. 123).

Thorium Series (4n)*

Nuclide	Historical name	Half-life	Major radiation energies (MeV) and intensities†		
			α	β	γ
$^{232}_{90}Th$	Thorium	1.41×10^{10}y	3.95 (24%) 4.01 (76%)	---	---
$^{228}_{88}Ra$	Mesothorium I	6.7y	---	0.055 (100%)	---
$^{228}_{89}Ac$	Mesothorium II	6.13h	---	1.18 (35%) 1.75 (12%) 2.09 (12%)	0.34c‡ (15%) 0.908 (25%) 0.96c (20%)
$^{228}_{90}Th$	Radiothorium	1.910y	5.34 (28%) 5.43 (71%)	---	0.084 (1.6%) 0.214 (0.3%)
$^{224}_{88}Ra$	Thorium X	3.64d	5.45 (6%) 5.68 (94%)	---	0.241 (3.7%)
$^{220}_{86}Rn$	Emanation Thoron (Tn)	55s	6.29 (100%)	---	0.55 (0.07%)
$^{216}_{84}Po$	Thorium A	0.15s	6.78 (100%)	---	---
$^{212}_{82}Pb$	Thorium B	10.64h	---	0.346 (81%) 0.586 (14%)	0.239 (47%) 0.300 (3.2%)
$^{212}_{83}Bi$	Thorium C	60.6m	6.05 (25%) 6.09 (10%)	1.55 (5%) 2.26 (55%)	0.040 (2%) 0.727 (7%) 1.620 (1.8%)
64.0% / 36.0%					
$^{212}_{84}Po$	Thorium C'	304ns	8.78 (100%)	---	---
$^{208}_{81}Tl$	Thorium C''	3.10m	---	1.28 (25%) 1.52 (21%) 1.80 (50%)	0.511 (23%) 0.583 (86%) 0.860 (12%) 2.614 (100%)
$^{208}_{82}Pb$	Thorium D	Stable	---	---	---

*This expression describes the mass number of any member in this series, where n is an integer.
Example: $^{232}_{90}Th$ (4n).....4(58) = 232
†Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series.
‡Complex energy peak which would be incompletely resolved by instruments of moderately low resolving power such as scintillators.

Data taken from: Lederer, C. M., Hollander, J. M., and Perlman, I., Table of Isotopes (6th ed.; New York: John Wiley & Sons, Inc., 1967) and Hogan, O. H., Zigman, P. E., and Mackin, J. L., Beta Spectra (USNRDL-TR-802 [Washington, D.C.: U.S. Atomic Energy Commission, 1964]).

Figure 6-3. Thorium-232 decay series (PHS 1970, p. 121).

- 1969 to 1978, male employees only; and
- 1979 to present, male and female employees.

Female employees were not monitored during certain periods because “the potential did not exist for them to exceed 10% of the quarterly standards” (NLO 1981, p. 4; Noyes 1968a, p. 2). In the later years, when operations were oriented towards cleanup and restoration, and to meet compliance with 10 CFR Part 835, external radiation monitoring was performed for all occupational workers who entered areas at the FMPC and who had the potential to receive a 100-mrem whole-body dose in

1 year. For visitors, monitoring was performed for those who entered areas in which they had the potential to receive a 50-mrem whole-body dose in 1 year (Robinson 2002, p. 5).

Additional information is provided in Section 6.8 for reconstructing dose to unmonitored workers.

FMPC used several types of personnel dosimeters throughout its operational period. There were also several changes in occupational and administrative exposure limits during that period, including dosimeter exchange periods. Table 6-3 lists details of these changes (ORAUT 2003, pp. 8, 25).

Table 6-4 summarizes the calibration practices, calibration sources, quantity, and background correction use for FMPC dosimetry.

Table 6-4. Dosimetry calibration practices.

Years	Calibration source	Dosimetric quantity	Air or phantom	Natural background correction?
1952–?	RaU slab	Exposure & rep	Air	Yes
?–1985	RaU	Exposure & rad	Air	Yes
1985–present	Commercial	Rem	Phantom	Yes

During these periods dose or exposure limits changed with the lowering of limits as more knowledge was gained about radiation protection practices. Exposure limits to the whole body were established at 0.1 R/d in the early 1940s, and were reduced in 1948 to 0.05 R/d to the whole body and 0.3 R/wk to blood-forming organs. Skin and extremities limits were established at 0.6 and 1.5 R/wk, respectively. By 1955, whole-body exposure was reduced to 3 rem per 13-week period and 5 rem/yr from all sources of radiation. In 1959, the International Commission on Radiological Protection (ICRP) Publication 2 recommended a limit for employees over 18 years old as $5 \times (N - 18)$ rem total lifetime, where N is the age in years (ICRP 1959, p. 17). This limit included an annual restriction of 10 rem. The extremity limit remained constant at 75 rem/yr but was later reduced to 50 rem/yr. Fernald established site-specific lesser values as a guide to ensure that workers would not exceed limits. The guides were often one-fourth to one-third of the annual limits per dosimeter wearing period or per calendar quarter. In addition, exposures sometimes were controlled using other types of dosimeters such as pocket-type ion chamber “pencil dosimeters.” Time limits were based on the product of measured dose rates and time (Noyes 1968b, pp. 2–5).

While current minimum detection limits (MDLs) as listed in Table 6-3 are well defined (Cooper 1998), earlier limits were not. Because it is difficult to estimate MDLs for the early dosimetry systems, the values in this TBD are those for the analogous ORNL system. Important dose reconstruction parameters for FMPC workers are based on the following administrative practices:

- Dosimeter exchange policies,
- Dosimeter assignment policies,
- Lost dosimeter and dose assignment policies,
- Incident handling policies and information on how results were recorded, and
- Past exposure records for new employees.

6.4.2 Incidents

External radiation dose from worker involvement in incidents is included in the dose of record for Fernald workers. In addition, “Radiation Exposure Investigation” reports are documented in claimants records when site personnel investigated badge readings (NLO 1971a); these were tracked by dosimetry personnel on logsheets in the 1990s (FEMP 1990). A review of 1995 incidents indicated that incident investigation reports (IIRs) were initiated by either employees or supervisors. When

involved in or in proximity to an incident, an employee reported to the Medical Department at the end of the shift and filled out an IIR (Bogar 1986, p. 190).

6.4.3 Dosimetry Technology

The dosimetry technology at FMPC was approximately equivalent to that in use throughout the nuclear industry at the time. FMPC followed the ORNL program for dosimeter design and calibration. The exception was the lack of a requirement for neutron dosimetry at FMPC. Table 6-3 above lists data about dosimeter types, periods of exchange, and MDLs, while Table 6-4 lists calibration data over the same periods.

Extremity dosimetry at Fernald involved the use of wrist dosimeters (rather than finger dosimeters) together with application of an appropriate correction factor. Extremity exposures are believed to have been monitored using wrist film badges from approximately 1969 until 1977 (NLO 1981, p. 2). The use of thermoluminescent dosimeters (TLDs) for extremity dose measurements began on February 16, 1977 (Boback 1978, p. 1). There is some evidence that a correction factor of 3 (Abee 1951 p. 3) was used as observed in the review of early worker exposure records. Documentation of the correction factor was not established until a study by Jones determined that a factor of 2.06 times the wrist dosimeter value should be used to estimate the dose to the extremity (Jones 1988, p. 179). The wrist dosimeter in use at the time of the Jones study was a Teflon disk embedded with $CaSO_4$:Dy; however, records indicate that film previously had been used at FMPC.

The method of using wrist-to-finger ratios to estimate extremity doses is not a particularly accurate practice. This approach was used at many DOE sites, with each site determining its own correction factor. Extremity *Hp(0.07)* doses could be overestimated by this method by as much as 20% due to shielding by protective clothing on the extremities (e.g. gloves) because the wrist dosimeter is worn outside of any clothing. Therefore, the recorded extremity doses should be favorable to claimants and should provide the best estimate of *Hp(0.07)* for individual monitored employees.

This reasoning is also true for whole-body doses. After reviewing the tables in this TBD it is possible to determine that the preponderance of the radiation consists of beta particles, and that while this form of radiation can deliver substantial doses to bare skin in proximity it does not penetrate deeply into the body. The dose rate from the photon component from the radioactive decay of uranium is “minor compared to the beta dose rate” (Alvarez et al. 1984, p. 9). In addition, protective measures such as distance, shielding, clothing, gloves, etc., reduce beta dose rates appreciably without excessive bulk by approximately 20%.

Individual exposure records indicate an “open window” design for personnel monitoring devices that allowed both beta and photon radiations to reach the measuring element (film or TLD). Some DOE sites, including FMPC, incorporated a security credential in the dosimeter holder that in some instances covered the open window of the dosimeter. However, FMPC did not cover the open window with their security credential (NLO 1981, p. 2), which provided more accurate results than if the window had been covered.

An additional radiological concern at several locations at FMPC occurred when workers were subjected to high levels of radioactive material-bearing dust. This widespread source of contamination was a concern for personal dosimeters, so at times the dosimeters were enclosed in plastic bags for protection against dust contamination. The manner in which these contaminated dosimeters were handled was not identified, but this should not be an issue in dose reconstruction because the dosimeters were calibrated in plastic bags and no adjustments were made to the dosimeter results for either *Hp(0.07)* or *Hp(10)*.

An evaluation of original recorded doses for FMPC workers based on these parameters should yield a good (best available) estimate of *Hp(10)*. Where necessary, *Hp(0.07)* for those individual workers who came in direct contact with radiological source materials can also be obtained because open window results were recorded with no adjustments to those readings.

6.4.3.1 Multielement Film Dosimeters

At startup in late 1951, FMPC used the two-element beta/photon dosimeter that was developed at the Metallurgical Laboratory at the University of Chicago (Pardue, Goldstein, and Wollan 1944). This dosimeter was basically the same as the Clinton Laboratory dosimeter in use at Oak Ridge (Wilson et al. 1990). It used DuPont 552 film, an open window, and a cadmium filter. The open window was recorded as skin dose, and beta plus gamma with the filtered response was recorded as whole-body gamma dose. The Pardue dosimeter was processed off the site from startup in 1951 to June 1952, when onsite processing began. In mid-1953, FMPC converted to an advanced ORNL dosimeter, which consisted of an open window along with cadmium, lead, copper, and plastic filters. However, there are discrepancies in the documentation about the use of this dosimeter at FMPC. A health protection review of National Lead of Ohio Company (NLO, the Fernald operating contractor at the time) on May 14 to 16, 1963, indicated that the ORNL-type badge and DuPont 545 and 558 film packets were used. The 545 packet contained the 555 film dosimeter film and the 558 contained the 508 and the high-range 1290 dosimeter films. All NLO employees, AEC area office personnel, and visitors were monitored in this manner (Johnson and Heacker 1963, p. 5). A letter dated September 11, 1981, in response to a Dosimetry Assessment Fact Sheet states that in "January 1961, the ORNL badge meter, Model II was put into service at this site" (NLO 1981, p. 2). From 1951 through 1960, the film packet was sealed in plastic and placed in a metal case that was attached to the worker's security badge. A portion of the film was not covered by the metal case and served as a means of monitoring skin dose. Regardless, all designs were calibrated for each batch of film, with the optical density of the film was determined after irradiation. The unknown exposure was compared optically to the known calibration and recorded accordingly. Therefore, any multielement dosimeter would be adequate because only the open window and the element that provided approximately 1,000 mg/cm^2 shielding was used.

6.4.3.2 Thermoluminescent Dosimeters

TLDs were introduced in or around 1978 or 1979, but only on an experimental basis. An exception was the extremity dosimeter program, which introduced TLDs in 1977; however, no data were found in the literature that described these TLDs other than "they were the Teledyne Teflon impregnated with calcium sulfate type" (NLO 1981, p. 2). Therefore, this TBD has assumed that the TLD was used in a wrist dosimeter configuration, and that dose calculations involved use of a modifying factor to provide some estimate of actual extremity exposure. The reviewed sample of records showed that while extremity doses were often near limits, the imposed whole-body restrictions limited worker extremity exposure to less than the extremity limit. It was concluded that the dose of record is the best to use for reconstruction of the extremity dose. At FMPC, extremity doses were calculated by correcting the wrist dosimeter results using a modifying factor of 3 for film and 2.06 in accordance with the Jones (1988) study. Jones (pp. 177, 202) also concluded that these measured workplace values should be reduced by 14%. However, FMPC did not retrofit the calculated extremity dose based on the new modifying reduction factor; therefore, the extremity ratios that were applied at Fernald are favorable to the claimant.

Whole-body TLDs using the commercial Panasonic system were placed in service in 1983 for testing (Bogar 1986, pp. 177, 179; Adams 1984, p. 153). The system was subjected to an extensive study starting in the fall of 1981 (Plato and Miklos 1982, p. 74). The study included screening of 1,800 dosimeters, calibration of a manual reader, development of correction factors for each of the four TLD chips in the dosimeter (E1 to E4), and the development of an algorithm and the calibration of FMPC

^{137}Cs source. The TLD was the Panasonic UD-802 model, some of which were modified by NLO by removal of the plastic filter over element E2 on the front of the dosimeter. This modification was to improve the difference in the E1:E2 ratio to provide a better differential of the beta energies.

The algorithm that was developed by this study proved to be less than adequate, although the system did satisfy ANSI International Standard N13.11-1983 (ANSI 1983). The system lacked sufficient precision in estimating beta energies. Alvarez et al. (1984) found that, on average, skin dose was in error by about 1%. At a given location, this error could vary between –36% and +45%, and the review of the algorithm for calculating skin dose indicated that individual TLD readings of approximately 4% caused variation in skin doses of 15% to 25%. This led to the development of a new algorithm and the previously mentioned change in the plastic filter in the dosimeter. These changes had an impact on skin dose and in some cases eye dose, but they had no effect on deep dose because neither the filter nor the evaluation of deep dose was changed.

The appropriate TLD correction factors (based on the Gesell algorithm) were retroactively applied to dosimetry records from the beginning of TLD implementation (i.e., the period from 1983 to 1985) as described in Bogar (1986) and Adams (1984). Both references describe the deficiencies in the original TLD algorithm that was developed by Plato, and the studies, comparisons, and fieldwork that were undertaken to develop a more precise algorithm (the Gesell algorithm). The deficiencies of the Plato algorithm affected the precision of electron dose measurements and resulted in overestimates of electron dose.

6.4.4 Calibration

It is always prudent and technically defensible to calibrate to the same types and energies of the radiations to be measured. For FMPC, this would involve uranium and its progeny, in some cases ^{99}Tc, and (perhaps early in FMPC operations) ^{226}Ra. Potential errors in recorded doses depend not only on the response of the specific dosimeter to the radiation to which it is exposed and calibrated, but also on the dosimeter geometry, how it is worn, and the simple variables in shielding that are afforded by clothing and other materials. The shielding effect is especially significant when the radiations are primarily beta particles or low-energy photons, both of which are predominant with uranium.

6.4.4.1 Beta/Photon Dosimeters

FMPC dosimeters were originally calibrated using a slab of natural uranium for beta energy and low-energy photons and ^{226}Ra for gamma energy. The uranium calibration determined the nonpenetrating or skin dose by measuring the film density behind the open window of the dosimeter. The penetrating dose was determined by measuring the film density behind the metal filter. All calibrations were made “in air” (i.e., no phantom) and for each batch of film. From a review of the available documentation, exposures were made for ^{226}Ra gamma radiation at 50, 150, 450, 1,356, and 4,077 mR. Uranium slab exposures were made at 40, 80, 160, 320, 640, and 3,840 mrep (NLO 1971b, 1974, 1975, 1977). By January 1977, calibration values had changed to a range of 25 to 4,077 mR in 10 increments for ^{226}Ra and to a range of 20 to 3,840 mrep for beta emissions (for a uranium slab). All dosimeters were calibrated with and without “bags.” It was not determined when bags were first incorporated in the calibration process, but calibration data from September 1974 indicated that five gamma exposures ranging from 50 to 4,077 mR ^{226}Ra and 10 uranium slab exposures ranging from 40 to 8,000 mrep were conducted “with bags.”

This TBD assumes that when FMPC began the practice of using bags for contamination control, attendant calibration procedures were revised to accommodate the new practice. Small changes in film density can mean large changes in recorded exposure. Therefore, it is desirable to calibrate

dosimeters using the same radiological sources and energies to which worker dosimeters will be exposed; this practice was implemented at FMPC.

6.4.4.2 FMPC Beta/Photon Dosimeter

As previously stated, FMPC beta/photon dosimeters were calibrated using ^{226}Ra for gamma energy and uranium for beta and low-energy photons. All calibrations were in air (i.e., no phantom was used). This practice continued until the change to TLDs in 1983. When participation in DOELAP performance testing was required in the mid- to late 1980s (DOE 1986, p. 12), FMPC began using ^{137}Cs as the gamma calibration source while continuing to use the uranium slab. In addition, the use of a phantom was required at this time. Whether a change was determined to be necessary in the recorded penetrating radiation dose as a consequence of this change has not been determined. It is probable that a decrease in *Hp(10)* would result from these actions because most other sites experienced similar results. No change in the recorded dose is proposed to account for what could be a 10% decrease in dose.

The use of the millirep unit is somewhat unique to FMPC because it declined in use elsewhere after the 1950s. There are few references, including the *Radiological Health Handbook* (PHS 1970), so this TBD assumes that a rep is approximately 93 ergs/g of tissue. Because FMPC used this term interchangeably with rad (100 ergs/g of any receptor), there is a further inherent conservativeness of approximately 7%.

6.4.5 Workplace Beta/Photon Radiation Fields

Alvarez et al. (1984, pp. 9, 13) characterized typical FMPC radiation fields in preparation for introduction of the TLD system. The radiation fields consisted of a complex mixture of beta, X-rays, and gamma energies. These were supplemented by higher energy gamma radiation associated with ^{226}Ra transitions that account for the dose rates associated with the K-65 Silos.

6.4.5.1 FMPC Beta/Photon Dosimeter Response Testing

No data or evidence has been identified of early response testing of FMPC dosimeters, but the site used both the Pardue and ORNL dosimeters since the beginning of operations. Tests of these or similar dosimeters indicated an overestimate for *Hp(10)* with energies greater than 100 keV for the two-element dosimeter and an underestimate of dose for energies less than 100 keV (unless an adjustment is made using a fraction of the response of the shielded portion of the dosimeter). It is reasonable to assume that the early versions of the FMPC dosimeters reacted similarly to the ORNL dosimeter system, given that FMPC used the ORNL system until changing to a commercial system in 1985.

In December 1983, there was an intersite comparison of the FMPC system between FMPC, Pacific Northwest Laboratories, and the Idaho National Engineering Laboratory (INEL) (Hayes 1983, p. 1; Author unknown ca. 1983). The dosimeters were exposed in air, mounted on a 0.75-in. plywood disk and rotated at 2 rpm during exposure to several different uranium or uranium-bearing materials. Conclusions reached as a result of these tests were:

- NLO skin dose results were usually high but satisfactory.
- Penetrating dose compared to within ± a few percent (<10%), and NLO results were usually high.
- All three systems were satisfactory for adequate determination of both types of exposures.

After these tests, NLO conducted several projects to improve its ability to determine incident beta energies. NLO contracted with INEL to characterize FMPC radiation fields (Alvarez et al. 1984, p. 9). These actions, along with the conclusion and attendant suggestions, resulted in changing the algorithm that was developed by Plato and Miklos (Plato 1983, p. 3; Cooper 1988, p. 7). Documentation indicates that official use of TLDs began on January 1, 1985, but the tests took place in late 1983 and early 1984 (NLO, 1984, p. 1-2). The documentation review for preparation of this TBD determined that FMPC used the total response of the unshielded or open window for skin dose and the response behind the shield for penetrating dose before the introduction of the TLD system. This approach is consistent with accepted practices of the time throughout the weapons complex, especially in non-plutonium facilities. This results in a conservative dose estimate that is favorable to claimants.

6.4.5.2 FMPC Workplace Beta/Gamma Dosimeter Response

Results of tests of FMPC dosimeters that were used during the 1960s included the conclusion that the half-value thickness of absorption of UX-2 (^{234}Pa) beta energy was approximately 110 mg/cm^2 (Heatherton 1960b, p. 19). It was determined that "the combined dose rate from the surface of uranium metal in equilibrium with its two daughters, UX-1 (^{234}Th) and UX-2 (^{234}Pa), is about 240 mrad/hr." It was also determined that approximately 95% of the surface dose rate, or approximately 228 mrad/hr, originated from the UX-2 in the metal. The processing of the metal resulted in separation of uranium progeny, which produced much higher dose rates in portions of the product, process equipment, and byproducts. The reason for this increase in dose rates is the loss of self-shielding that is afforded by the mass of the in-process uranium. Further studies involved the absorption of such materials as film wrapping paper, polyethylene, cardboard, and Lucite. These materials were used because they were assumed to be nearly tissue equivalent. It was determined that the half-value thickness for tissue was approximately 110 mg/cm^2 and, therefore, that the dose to the eyes or gonads was approximately 15% of the skin dose. It was also determined that coveralls that were worn by workers (about 30 mg/cm^2) reduced uranium beta exposure to the skin by approximately 20% (Hinnefeld 1983, p. 43). Figure 6-4 summarizes these data.

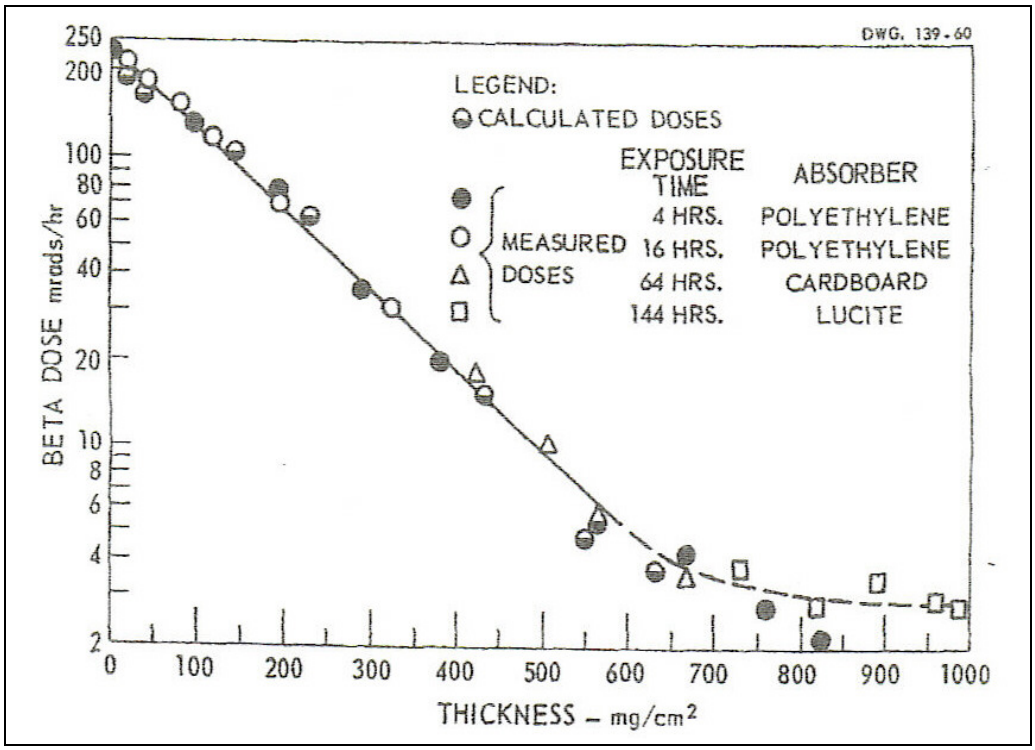

Figure 6-4. Beta dose vs. skin depth (Heatherton 1960b, p. 47).

While it was not explicitly stated in the documentation (Heatherton 1960b), it is assumed that the dosimeter in use at FMPC was the ORNL version and that the film was a DuPont type (Johnson and Heacker 1963, p. 5). There are some general data of results between film and TLD dosimeters from November and December 1982. Review of these data indicate that the two types of dosimeters did not agree and that the ratio of film to TLD varied with the location of the exposure. In some instances the ratio was greater than 1 and in others it was less than 1. Most often, film results provided higher dose estimates than TLD results, which supports the conclusion that early film dosimeter results are favorable to claimants. However, TLD results have been documented to be more accurate than film dosimeters and provide a more representative measure of the true exposures (this is particularly true when measuring beta and low-energy photons). ICRU Report 43 concludes that TLDs, when placed under an appropriate absorber "…will constitute a dosimeter having a spectral and angular response close to the ideal" while, about film, it states, "Dosimeters based on sensing elements that are not tissue equivalent (e.g. photographic film) can also be used, though in general it is more difficult to ensure that the variation of response with energy and angle of incidence is correct. Such problems are often enhanced for beta and low-energy x rays" (ICRU 1988, p. 9).

The forms of radiation at FMPC varied from plant to plant with Plants 5 and 9 exhibiting the highest potential workplace dose rates. These plants were involved with metal reduction, casting, and rolling, and these processes generated the separation and migration of progeny ^{234}Th and ^{234}Pa (UX-1 and UX-2). As stated above, ^{234}Pa contributes approximately 95% of the total beta dose rate. Therefore, any location in the process where this material accumulated resulted in the potential for higher exposure rates. Other areas of potential high radiation exposure included areas where progeny contaminated other materials (i.e., crucibles, saws, and rolling mills) or where large quantities of the parent material were present.

Studies in May 1984 (Boback 1984, p. 160) of various activities in Plant 5 indicated that the whole-body dose rate *Hp(10)* ranged from 0.1 to 1 mrem/hr. Studies in November and December 1982 using the same criteria (i.e., 80% of the time workers were performing their jobs) indicated that dose rates ranged from 0.08 to 16.5 mrem/hr. These dose rates were established for Plants 5, 6, and 9, with Plant 9 (Green 1981, p. 4) exhibiting the highest rate and Plant 5 the lowest. The dose rate values were derived using dosimeter data and applying the 80% work rate rule. These values were not intended to be precise measurements of actual dose rates and resultant doses to workers, but rather as representations of the dose rate ranges that were present. These and other studies resulted in the use of shielding and establishment of limits on materials that could be present at any time in certain locations. There is discussion in the documentation of the use of improved housekeeping to assist in reducing worker exposure (Boback 1984, pp. 145–146). There is also discussion of process changes, (i.e., Lucite face shields, rubber matting, ingots surface rinsing) that were used during "ingot pickling" to remove "beta emitting daughters." The *FEMP External Dosimetry Technical Basis Manual* contains information on dosimeter response data after 1984 (Robinson 2002).

Table 6-5 lists the lower limits of detection for DOELAP categories for FMPC, and Table 6-6 lists the maximum viable dose ranges for DOELAP categories.

Table 6-5. Lower limits of detection for DOELAP categories.

Category	***Hp(0.07)* (mrem)**	***Hp(10)* (mrem)**
Cesium	<5	<5
M30 (X-ray L)	<16	<16
S60 (X-ray M)	<16	<16
M150 (X-ray H_1)	<16	<16
M150 (X-ray H_2)	<16	NA
Sr/Y-90 (Beta)	<16	NA

Table 6-6. Maximum viable category dose ranges.

Category	*Hp(0.07)* (rem)	*Hp(10)* (rem)
Acc gamma (Cs)	5,150	5,150
Acc X-ray H_1	7,050	7,350
X-ray L	5,400	2,250
X-ray M	5,750	5,350
X-ray H_1	7,050	7,350
X-ray H_2	7,050	7,050
Gamma (Cs)	10.3	10.3
B\B+G	1,257	10.3
X-ray L+B	10.2	5.9
X-ray M+B	5,188	10.2
X-ray H_1+B	5,625	11.2
X-ray H_2+B	5,513	11
X-ray L+G	10.4	6
X-ray M+G	10.6	10.4
X-ray H_1+G	11.3	11.4
X-ray H_2+G	11.3	11.3

The angular dependence of the FMPC dosimeter satisfied the DOELAP Angular Dependence Requirements (Hinnefeld 1989, p. 163). The system was DOELAP-accredited in 1987 by meeting all requirements at that time. It has maintained accreditation as indicated in the *FEMP External Dosimetry Technical Basis Manual* (Robinson 2002, p. 6). FMPC has further improved its dosimetry by the addition of an improved computer for the system that enables it to generate element correction factors for each TLD chip in each dosimeter. New correction factors are generated on a biannual basis.

Table 6-7 lists average annular responses for the DOELAP energies of importance at FMPC.

The *Hp(10)* response is accurate for up to ±60 degrees, while *Hp(0.07)* varies drastically (as would be expected). Clothing and other barriers can offer some reduction in *Hp(0.07)*, but no adjustments were made to dosimeter results for any exposure [e.g., *Hp(10)* or *Hp(0.07)*]. The logical conclusion is that the recorded doses are favorable to claimants. The selection of beta and photon energies for the major FMPC facilities is summarized in Table 6-8. In the absence of a claimant specific work location, or if the claimant had multiple work locations in a given year, dose reconstructors should use the favorable to claimant default assumptions of 40% 30- to 250-keV and 60% >250-keV photon energy ranges along with 100% >15-keV energy for electrons and 100% 0.1- to 2-MeV energy range for neutrons, as applicable.

Alvarez et al. (1984) also estimated dose rates typical for the spectra that were measured at several locations on the plant. Those values are 0.35 mR/hr for low-energy photons <20 keV (typical for shallow skin dose) and 0.86 mR/hr for photons >30 keV and <300 keV. For energies >300 keV a deep dose rate of 1.4 mR/hr was calculated. These values are provided here for example only and if ratioed yield 13%, 33%, and 54%, respectively, for the energy categories used in dose reconstruction. The average photon radiation dose would be much lower and is “almost negligible compared with the beta skin dose,” as previously stated.

6.4.6 Neutron Dosimetry

The documentation for FMPC did not include any reference to neutron dosimetry with the exception of high-range gamma-sensitive film. This film was packaged with the sensitive 508 film and exchanged on an annual basis. There was some concern expressed in AEC audit letters (Johnson and Heacker

Table 6-7. Angular response for DOELAP energies of importance.

Energy[a]	Angle	Plane[b]	*Hp(10)* Rd/Do avg.[c]	*Hp(0.07)* Rd/Do avg.[c]
M30	0	H	1.02	1.012
M30	±20	H	1.012	0.997
M30	±40	H	0.967	0.955
M30	±60	H	0.769	0.764
M30	±60	V	0.928	0.918
S60	0	H	1.019	1.021
S60	±20	H	1.028	1.031
S60	±40	H	1.021	1.017
S60	±60	H	0.964	0.967
S60	±60	V	0.893	0.896
M150	0	H	1.026	1.026
M150	±20	H	1.061	1.068
M150	±40	H	1.102	1.102
M150	±60	H	1.02	1.02
M150	±60	V	1.058	1.058
Cs137	0	H	0.964	0.964
Cs137	±20	H	0.973	0.972
Cs137	±40	H	0.966	0.966
Cs137	±60	H	0.934	0.934
Cs137	±60	V	0.890	0.891
Sr90/Y90	0	H	0.995	N/A
Sr90/Y90	±20	H	0.982	N/A
Sr90/Y90	±40	H	0.720	N/A
Sr90/Y90	±60	H	0.367	N/A
Sr90/Y90	±60	V	0.340	N/A
M30	±60	V	0.928	0.918
S60	0	H	1.019	1.021
S60	±20	H	1.028	1.031
S60	±40	H	1.021	1.017
S60	±60	H	0.964	0.967
S60	±60	V	0.893	0.896
M150	0	H	1.026	1.026
M150	±20	H	1.061	1.068
M150	±40	H	1.102	1.102

a. The entries for M30, S60, and M150 represent calibration energy categories utilized by DOELAP.
b. H = horizontal; V = vertical.
c. Rd = "dose read" meaning the value read from the dosimeter; Do = dose observed meaning the value given by DOELAP. N/A = not applicable.

1963, p. 5) that mention "the badge also contains components to evaluate personnel exposure from criticality accidents" (an event that never occurred at Fernald). There was no established need for neutron dosimetry at FMPC even though there were large quantities of UF_4 and UF_6. Enrichments were low enough (typically <2% ^{235}U) that alpha-neutron reactions were limited. The limitations of nuclear track emulsion, type A, were well documented including an MDL of approximately 40 mrem for fast neutrons. Based on studies and calculations it was concluded that fast neutron exposures at FMPC would be less than the MDL (Cooper 1998, p. 14). The purpose of the following section is to discuss and develop a neutron-to-photon ratio for estimating neutron doses at Fernald. A possible source of low-level neutron exposure at Fernald is the alpha-neutron reaction from the uranium alpha particle interactions with fluorine atoms. This reaction primarily occurs with the production and storage of UF_4 (green salt). The areas at Fernald that produced and/or stored green salt include the

Table 6-8. Beta and photon energies and percentages.

Building	Description	Radiation	Energy (keV)	Percentage
Plants 2/3	Production of UO_3	Beta	>15	100
		Photon	>30 and <250	40
			>250	60
Plant 7[a]	Reduction of UF_6 to UF_4	Beta	>15	100
		Photon	>30 and <250	40
			>250	60
Plant 1	Sampling plant	Beta	>15	100
		Photon	>30 and <250	40
			>250	60
	Refinery	Beta	>15	100
		Photon	>30 and<250	40
			>250	60
Pilot Plant[b]	Scrap recovery	Beta	>15	100
		Photon	>30 and <250	40
			>250	60
	UF_6 to UF_4 reduction	Beta	>15	100
		Photon	>30 and <250	40
			>250	60
Plant 8[b]	Scrap recovery	Beta	>15	100
		Photon	>30 and <250	40
			>250	60
Plant 5[b]	Metal production	Beta	>15	100
		Photon	<30	13
			>30 and <250	33
			>250	54
Plant 6[b]	Metal fabrication	Beta	>15	100
		Photon	<30	13
			>30 and <250	33
			>250	54
Plant 9[b,c]	Special products	Beta	>15	100
		Photon	<30	13
			>30 and <250	33
			>250	54
Plant 4[c]	UF_4	Beta	>15	100
		Photon	>30 and<250	40[c]
			>250	60[c]

a. Operated only 2 years from June 1954 to May 1956.
b. Alvarez et al. (1984, p. 17).
c. For Plants 4, 8, 9, and the Pilot Plant, the years processing thorium were: Plant 4, 1954; Plant 8, 1967–1971; Plant 9, 1954–1955; and Pilot Plant, 1964–80. For workers employed there, the energy range is 25% >30 and <250 keV and 75% >250 keV.

Pilot Plant, Plant 4, Warehouse 4B, and any other warehouse at Fernald for which the stored material is unknown. The neutron-to-photon ratio described below should only be applied to workers who were exposed to uranium fluoride materials (i.e. UF_4, UF_6) at Fernald.

6.4.6.1 Development of the Neutron-to-Photon Ratio

Neutron exposures at Fernald were evaluated in 1998 in Warehouse 4B where the storage of enriched UF_4 (green salt) in arrays of drums had the potential for generating measurable neutron fluxes. To determine if neutron doses in Warehouse 4B were of significant magnitude to warrant special neutron monitoring for workers who were involved in the inspection and packaging project, long-term neutron dose rates were measured at various locations using Landauer Neutrak-ER area dosimeters. The Landauer dosimeter is a dual-element device that uses a lithium albedo chip and a CR-39 foil so that an extended range of neutron energies can be monitored. A set of six Neutrak-ER

dosimeters were placed in fixed locations in the warehouse. To provide confidence that the long-term dosimeter measurements were representative of the dose rates in the building, and that there were no elevated hot spots of neutron dose rates, a special area survey was performed. This survey was performed using a Nuclear Research Corporation Model NP-2 "snoopy" portable neutron meter (Robinson 2001).

Table 6-9 provides the neutron dose rate survey results. Seven measurements were taken using the 10-minute integration period at locations near and among the various drums of material. As indicated in the table, none of the dose rate measurements exceeded the minimum detectable dose rate of 0.02 mrem/hr. The gamma dose rates were subsequently measured at the above locations. With the exception of the control point, the dose rates were found to vary from 3 to 6 mrem/hr on contact and from 2 to 4 mrem/hr at 1 ft.

Table 6-9. Measured neutron dose rates (Robinson 2001).

Location	Measured neutron dose (mrem/hr)	Location	Measured neutron dose[a] (mrem/hr)
Control point	0.001	Enriched Area	0.009
Red drum array	0.002	Enriched Area	0.018
Red drum array	0.003	Drum Conveyor	0.005
Red drum array	0.014		

a. All results are below the calculated minimum detectable neutron dose rate of 0.02 mrem/hr. These values are included here for completeness.

It was concluded that the external gamma dose rates within the established radiation area were consistently greater than 10 times that of the highest detected neutron dose rate.

To develop a neutron-to-photon ratio for UF_4 (green salt), photon survey data was also required. Photon surveys were conducted in 2001 on 56 drums of UF_4 (Ward and Hopson 2001, pp. 2–6). The drum midline dose rate varied from below detection limits to 1.5 mrem/hr, with a mean and standard deviation of 0.79 ±0.26 mrem/hr on contact. The photon dose rate at the bottom of the drum ranged from below detection limits to 3.5 mrem/hr, with a mean and standard deviation of 1.71 ±0.82 mrem/hr. These two datasets were combined to estimate an overall uncertainty. The photon dose rate variability is primarily the result of differences in the fill level of the individual drums. The top of the drums were discarded from this analysis because the measured general photon dose rate was below detection limits. After combining the datasets, a distribution was fit to the data. The combined datasets resulted in a lognormal distribution with a geometric mean of 1.1 mrem/hr and a geometric standard deviation of 1.7.

The neutron-to-photon ratio ($R_{n/\gamma}$) is calculated by dividing the neutron dose rate in Table 6-9 by the photon dose rate:

$$R_{n/\gamma} = \frac{Neutron\ Dose\ Rate}{Photon\ Dose\ Rate} \quad (6\text{-}1)$$

To propagate the uncertainty, a Monte Carlo simulation was conducted using distributions in place of each constant dose rate value. Figure 6-5 depicts the frequency distribution of the simulated neutron-to-photon ratios for depleted and low-enriched uranium.

Statistical parameters of the distributions in Figure 6-5 are provided in Table 6-10. The geometric mean of the low-enriched uranium neutron-to-photon distribution was 0.10 with a geometric standard deviation of 1.71 and an upper 95-th percentile confidence of 0.23. This distribution is combined with measured and missed dose distributions using Monte Carlo methods described in ORAUT-OTIB-0012, *Technical Information Bulletin: Monte Carlo Methods for Dose Uncertainty Calculations* (ORAUT 2005a).

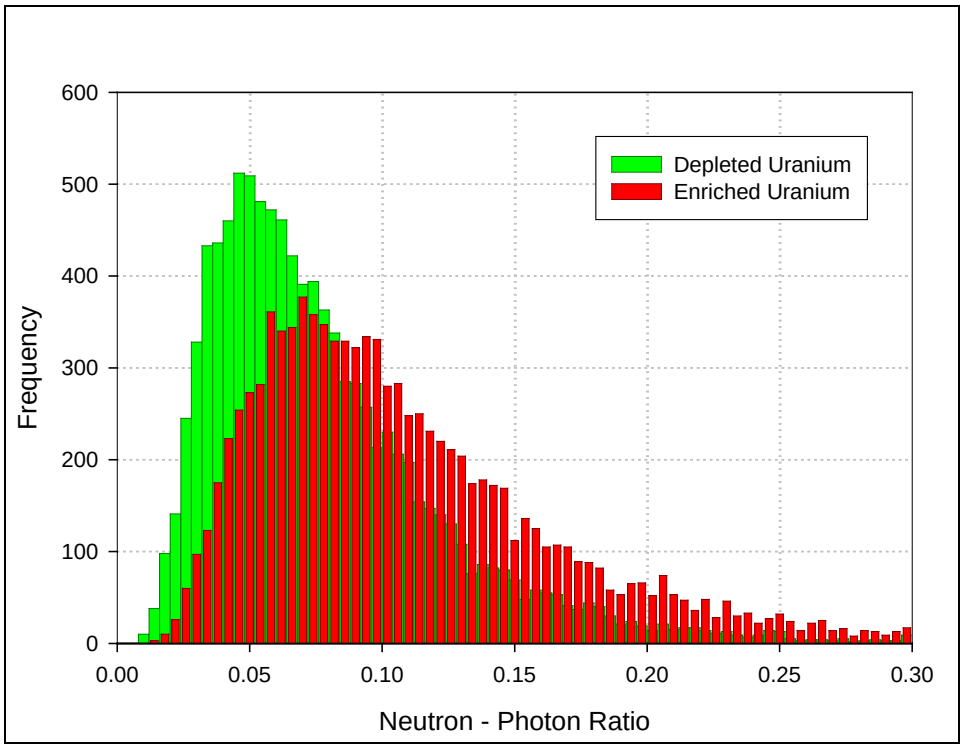

Figure 6-5. Comparison of neutron-to-photon ratio distributions for depleted and low-enriched UF_4.

Table 6-10. Statistical parameters of neutron-to-photon ratio distributions.

Enrichment	Geometric mean	Geometric standard deviation	Upper 95th percentile
Depleted uranium	0.07	1.74	0.17
Low-enriched uranium	0.10	1.71	0.23

Multiple approaches can be used to estimate the neutron dose using the neutron-to-photon ratios as presented. The factors that affect an individual's neutron dose include: (1) the quantity of uranium processed, (2) the enrichment, and (3) the time an employee worked in a process or storage area. The recorded photon dose is also a function of the quantity of processed uranium and the exposure duration. The uranium enrichment has some small effect on the recorded photon dose in that enriched uranium generally has a higher photon dose rate than depleted uranium. This effect is ignored in this analysis for simplicity and is considered a favorable to claimant simplification because a higher photon dose rate would decrease the neutron-to-photon ratio rather than increase the ratio (see Equation 6-1). Using the measured and missed photon dose should correctly account for changes in exposure that resulted from decreases in uranium production and changes in an individual's exposure duration.

Through the years, Fernald workers conducted operations with enriched uranium, natural uranium, and depleted uranium. Figure 6-6 depicts the percentage of total uranium that was received and processed by Fernald in each of these three categories (DOE 2003).

The receipt for the maximum year (1967) was 24,400 MTU.

It should be noted that although natural uranium is not presented in Table 6-10 or calculated in this analysis, the neutron-to-photon ratio for natural uranium would fall in between the depleted and low-enriched ratios. This is because the alpha-neutron reaction is directly proportional to the alpha

activity of the sample. Depleted uranium has the least amount of ^{235}U, which has a significantly higher specific activity than ^{238}U and therefore an overall lower specific activity when comparing natural and enriched uranium. Natural uranium has a slightly higher specific activity in comparison with depleted uranium, and enriched uranium has the highest specific activity of the three forms.

As can be observed from the Figure 6-6, most of the uranium work before about 1965 was with natural uranium with a decrease beginning around 1960, at which time the percentage of enriched uranium increased. By 1970, relatively little work with natural uranium was conducted at Fernald. Beginning in the mid-1960s, work with depleted uranium began to increase. By the mid 1970s, work with depleted uranium constituted the majority of the uranium work at Fernald. During the transition period (1965 to 1975), work with enriched uranium initially increased so that by 1968 enriched uranium work constituted about 50% to 60% of the uranium work at Fernald. After this peak, work with enriched uranium decreased to about 10% to 20% of the total uranium work at Fernald.

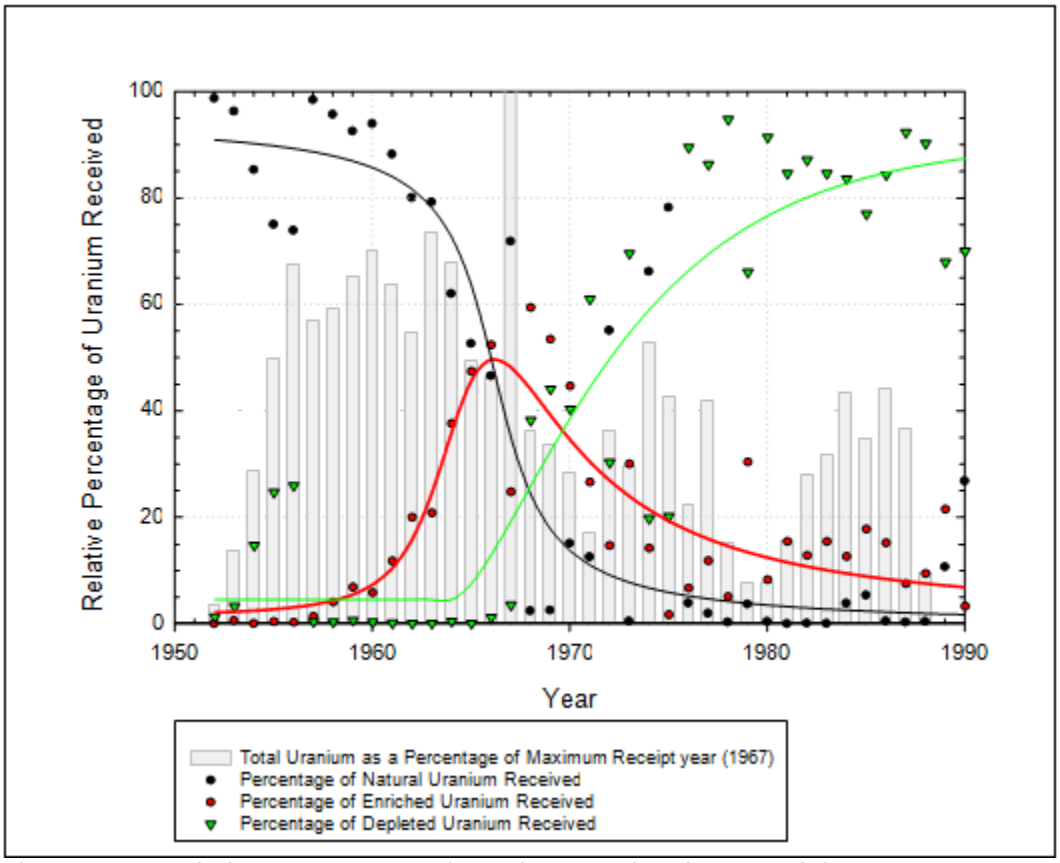

Figure 6-6. Relative percentage of uranium receipts by material category (natural, enriched, and depleted) (Voillequé et al. 1995, p. 689 and DOE 2000.

While the data in Figure 6-6 could be used to estimate annual neutron-to-photon ratios based on the percentage of enriched, natural, and depleted uranium work, to do so would unreasonably complicate the dose reconstruction and introduce additional uncertainty. The additional uncertainty stems from whether an employee might have worked predominately with one type of material or another. Because this uncertainty cannot be properly tracked without detailed job history and material tracking information, and to simplify the dose reconstruction, the low-enriched uranium neutron-to-photon ratio should be used. This assumption will tend to slightly overestimate the actual neutron-to-photon ratio and is considered a reasonable but necessary favorable to claimant assumption given the limited data available.

While there is a low probability that all of an employee's photon dose would result from exposure to green salt (especially because Fernald is known to have processed large quantities of uranium metal, yellow cake, black oxide, etc.), the probability is not zero and cannot be excluded on an individual basis without significant job history information. As a result, the neutron-to-photon ratio should be applied to all measured and missed photon dose for employees who worked in the Pilot Plant, Plant 4, any warehouse or other area known to store UF_4 or any of the onsite warehouses for which the stored material is not known. This favorable to claimant assumption is necessary to account for the uncertainty in the origin of the photon dose.

6.4.6.2 Neutron Activation Analysis

From 1955 until 1978, a variety of neutron sources were used to perform analytical neutron activation analysis (NAA) procedures. These sources included:

- RaBe source (1955 to 1967) (Ross 1967),
- Cockcroft-Walton accelerator (1964 to 1978) (Hervin and Johnson 1964, p. 8; NLO 1980, p. 1), and
- ^{252}Cf source (1972 to1975) (Deily 1983; Boback 1975, p. 4).

Exposure to these sources would have been limited to workers in the Technical Division. Evidence of work with these sources would be found in claimant interview files or other workplace documentation from DOE. In relation to assigning neutron dose for NAA activities – if applicable – the neutron-to-photon ratios in Table 6-11 can be used. The recommended ratios for work with the RaBe source and the Cockcroft-Walton accelerator are based on gamma and neutron survey data that were taken by FMPC staff (Starkey 1963, Burgett and Thomas 1965, p. 13–24; Klein and Ross 1965). The ratio for the ^{252}Cf source is based on gamma and neutron measurements that were made at ORNL for similar work with ^{252}Cf (ORAUT 2007).

Table 6-11. NAA source neutron-to-photon ratios

Source	Ratio	Reference
RaBe	3.0	Starkey 1963
Cockcroft-Walton accelerator	4.1 ± 2.3[a]	Burgett and Thomas 1965, Klein and Ross 1965
Cf-252	2.1	ORAUT 2007

a. Represents an average of the neutron-to-photon ratio for measured gamma and neutron dose rate values at the operator's position with moderating water tank filled and unfilled and a use factor of 25% based 10 hours of use per week.

6.4.6.3 Workplace Neutron Radiation Fields

To date, specific neutron energy spectra of UF_4 have not been found or modeled. According to the DOE *Health Physics Manual of Good Practices for Uranium Facilities* (Rich et al. 1988, p. 39), neutrons of approximately 2 MeV are generated from uranium alpha particle interactions with fluorine. If the neutrons are generated at this energy, through scattering interactions with surrounding materials, some fraction of the neutrons will have energy less than 2 MeV.

The POC calculations in IREP use a radiation effectiveness factor (REF) to estimate the POC. The most favorable to claimant neutron energy group (i.e., the neutron energy group with the largest REF) is the 0.1- to 2.0-MeV energy group. Because the neutrons at Fernald are assumed to be generated (born) in this energy group and although some fraction will scatter to lower (less favorable to claimant) energy groups, all of the neutron dose should be assumed to result from the 0.1- to 2-MeV energy group. In the absence of data, this is a reasonable and favorable to claimant assumption.

In situations that could have involved neutron activation analysis work, a variety of sources might have been used. Information about sources would be found in claimant interview files or other workplace documentation from DOE. The default neutron dose fractions are given in Table 6-12.

Table 6-12. Default neutron dose fractions.

Facilities/activities with neutron radiation	Source	Neutron energy (MeV)	Default dose %
Pilot Plant, Plant 4, and warehouses	UF_4, UF_6, and various enrichments	0.1 to 2.0	100%
NAA	RaBe[a]	0.1 to 2.0 2.0 to 20	10% 90%
NAA	Cockcroft-Walton accelerator[b]	0.1 to 2.0 2.0 to 20	5% 95%
NAA	Cf-252[c]	<0.01 0.1 to 2.0 2.0 to 20	25% 57% 18%

a. NBS (1960).
b. Burgett and Thomas (1965).
c. ORAUT (2007).

6.5 RECORDED DOSES

FMPC recorded both skin and penetrating doses by determining film densities behind the open window and a single filter of approximately 1,000 mg/cm^2. The FMPC historical dose record practices are given in Table 6-13.

Table 6-13. Historical recorded dose practices.[a]

Year	Dosimeter measured quantities	Compliance dose quantities
1951–1953 Film	OW: mrep S (Cd): mR	Skin = OW+S WB = S
1954–1985 Film	OW: mR S: mrem	Skin = OW+S WB = S
1985–present TLD	Nonpenetrating (Npen) Penetrating (Pen)	Skin = Npen+WB WB = Pen

a. OW = open window; S = shielded; WB = whole body.

6.6 ADJUSTMENTS TO RECORDED DOSE

Corrections to the FMPC reported dose are required due to uncertainties in the recorded data and lack of significant data, especially before 1980. To satisfy the favorable to claimant criteria for dose reconstruction and considering the variability in radiation due to the FMPC processes, it is suggested that the corrections in the following discussion be considered.

A review of available data and documentation identified no mention that suggested the need for corrections to recorded whole-body dose. However, due to uncertainties in the dosimeter responses and the extended use of the two-element (or effectively two-element) dosimeter, a correction to pre-1985 recorded dose data is recommended. While there was a change to a multielement dosimeter, the penetrating dose was evaluated by the response behind the heavy metal filter. The heavy metal filter was cadmium in both dosimeters, which attenuated the lower energy photons (see Table 6-8, Figure 6-7 (Alvarez et al. 1984), and Figure 6-8 (Thorton et al. 1961)) and resulted in an underestimated response behind that filter for measured dose and *Hp(10)*. The response of the ORNL dosimeter, as shown in Figure 6-8, indicates a general under-response below 100 keV, followed by a sharp over-response at 70 keV, followed by essentially no transmission of photons below 60 keV. Based on the spectroscopy data in Figure 6-7, approximately 10% of the gamma energy range likely falls into the under-response area of Figure 6-8. Because most, but not all,

penetrating radiations are above 100 keV, an adjustment is necessary for pre-1985 recorded penetrating whole-body doses due to the contribution to *Hp(10)* from low-energy photons including the L X-rays from both uranium and thorium. It is estimated that a correction equal to 10% of the <250 keV values in Table 6-8 be added to the *Hp(10)* dose due to the contribution of these low-energy photons to penetrating dose that are absorbed in the thick filter.

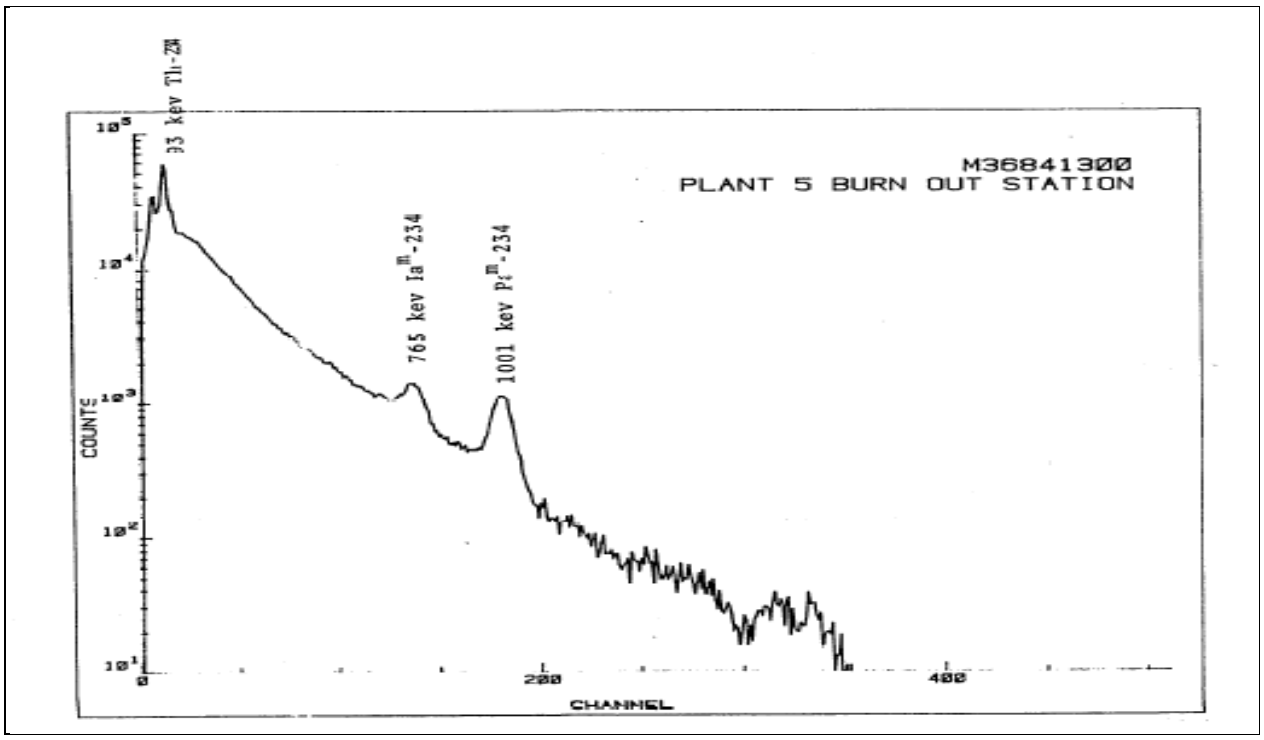

Figure 6-7. Plant 5 gamma spectrum (Alvarez et al. 1984).

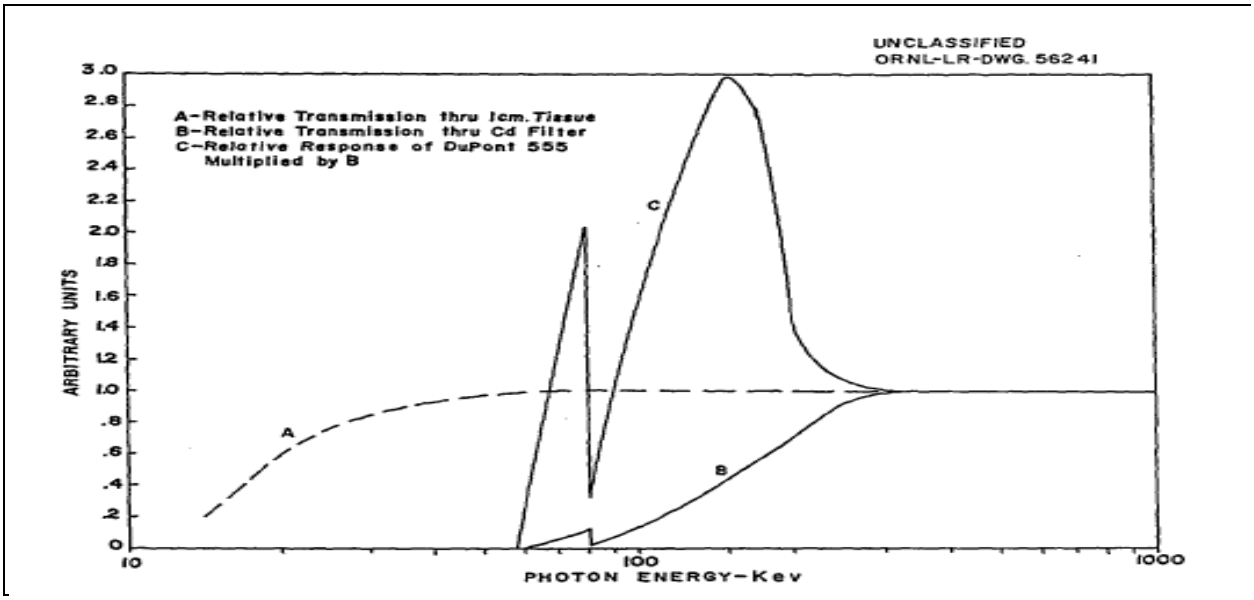

Figure 6-8. ORNL film dosimeter response (Thorton et al. 1961).

6.6.1 Adjustment for Neutron Quality Factor

The typical practice in 1995 was to use neutron flux to dose rate conversion factors in National Council on Radiation Protection and Measurements (NCRP) Report 38 (NCRP 1971). Some DOE

facilities, however, used a standard quality factor of 10 to convert from absorbed dose (rad) to dose equivalent (rem). For dose reconstruction purposes, the Report 38 doses must be converted to ICRP Publication 60 values (ICRP 1991). The conversion factor from NCRP Report 38 to ICRP Publication 60 is 1.91 [see ORAUT-OTIB-0055, *Technical Basis for Conversion from NCRP Report 38 Neutron Quality Factors to ICRP Publication 60 Radiation Weighting Factors for Respective IREP Input Neutron Energy Ranges* (ORAUT 2006a)]. The favorable to claimant assumption of using a factor of 2 should be applied to the neutron-to-photon ratio to convert the estimated dose to an ICRP Publication 60 dose equivalent value.

6.6.2 Dose Conversion Factors

Exposures were measured for the years that film dosimeters were used. Therefore, to obtain organ doses (as required for dose reconstruction), the exposure-to-organ dose conversion factors in OCAS-IG-001, *External Dose Reconstruction Implementation Guide* (NIOSH 2007), should be used for organs that are affected only by penetrating radiation (e.g., organs other than the skin, breast, and testes). Doses to the skin, breast, and testes (and any other cancer location potentially that is affected by nonpenetrating radiation) are determined based on both gamma and nonpenetrating columns; gamma doses are assigned as photons with an energy range consistent with information in this document, and nonpenetrating doses are assigned as electrons >15 keV with corrections to account for clothing attenuation or other considerations. Further guidance is provided in ORAUT-OTIB-0017, *Technical Information Bulletin: Interpretation of Dosimetry Data for Assignment of Shallow Dose* (ORAUT 2005b).

6.7 MISSED DOSE

6.7.1 Missed Beta/Photon Dose

It can be assumed with some certainty that there have been missed doses in the recorded doses for FMPC workers. This could have resulted when a dosimeter was lost, if a worker was not monitored for short intervals of employment, or if a zero was entered because the dosimeter result was less than the MDL. Various methods were used to estimate lost dosimeter results such as using coworker results, products of time and dose rate, or previously recorded results for similar work. All of these required considerable review and examination.

Missed dose from MDLs is especially important when there were short exchange periods, generally through the 1950s and 1960s. That period also had higher MDLs. NIOSH (2007) describes options to calculate the missed dose. The recommended option is to estimate a potential missed dose where one-half of the limit of detection (LOD/2) is multiplied by the number of zero or less than LOD/2 dose results. This procedure is applicable to both *Hp(0.07)* and *Hp(10)* because the same dosimeter was used for both until the introduction of a TLD finger dosimeter. The MDL per period and exchange frequencies along with the product (LOD × exchanges) are listed in Table 6-14. The other values can be obtained from these data and the individual exposure record.

Table 6-14. Missed beta/photon dose.

Years	MDL (LOD) (mrem)	LOD/2 (mrem)	Exchange frequency	Product (mrem/yr)
1951–1953	40	20	Weekly	2,080
1954–1958	30	15	Biweekly	780
1959–1984	30	15	Monthly	360
1985–present	5	2.5	Quarterly	20

6.7.2 Unmonitored Neutron Dose

Determination of neutron dose, both measured and missed, should be done using the guidance regarding neutron to photon ratios given in Section 6.4.6.1.

As an example, suppose an employee worked in Plant 4 at Fernald in 1965. The employee's measured photon dose was 400 ±80 mrem, and there were three zero monthly dosimeter readings. The energy employee's total photon dose would be 445 ±82 mrem. [Since the LOD/2 = 15 mrem, 3 zero readings X 15 mrem = 45 mrem (missed photon dose) + 400 mrem (measured photon dose) is 445 mrem]). Applying the low-enriched neutron-to-photon ratio and the conversion to ICRP Publication 60 radiation weighting factors (ICRP 1991) would result in an estimated neutron dose of 89 mrem, (445 mrem total photon dose x 0.1 x 2). The neutron dose distribution is a lognormal distribution with a geometric standard deviation of 1.77.

Summary of reasonable but favorable to claimant assumptions:

- Neutron-to-photon ratio developed using the average midline and bottom photon dose rates.
- Neutron-to-photon ratio developed using the low-enriched uranium neutron dose rate.
- Assumption that all recorded and missed photon dose was the result of exposure to enriched UF_4 material.
- Assumption that all of the neutron energies are between 0.1 and 2 MeV.
- Assumption of a quality factor of 10 to convert rad to rem.

6.8 UNMONITORED DOSE

As noted previously, in a response to a dosimetry assessment fact sheet NLO indicated that female employees at FMPC at times were not routinely monitored (NLO 1981). This situation existed although the female workers wore a combined security and dosimeter badge. It is reasonable to postulate that there could have been other circumstances where workers (both women and men) might not have been monitored.

6.8.1 Coworker Assigned Photon Dose

An estimated photon dose to unmonitored Fernald workers can be determined from monitored coworkers. ORAUT-OTIB-0020 (ORAUT 2011a) provides general instructions to evaluate the measured and missed doses for monitored Fernald workers to arrive at a favorable to claimant dose to be assigned to unmonitored workers from 1952 to 2005. If it is determined necessary to assign unmonitored dose after 2005, then dose reconstructors should apply 2005 coworker doses for 2006 through the present. Attachment A contains the details of the evaluation of Fernald coworker dose to be assigned to unmonitored workers. These measured doses include an analysis of the missed dose, which is particularly significant for the earlier years with higher LODs and frequent dosimeter exchanges (weekly or biweekly).

6.9 ORGAN DOSE

Once the *Hp(10)* doses have been calculated for each year, these values are used according to OCAS-IG-001 (NIOSH 2007) to calculate the organ dose distribution. OCAS-IG-001 describes the method for calculating organ dose using identified exposure geometries. For the period at Fernald from startup through 1984, the "Exposure (R) to Organ Dose (HT)" dose conversion factors should be

applied to the recorded and missed dose. For 1985 and later years, the "Deep Dose Equivalent (*Hp(10)*) to Organ Dose (HT)" dose conversion factors should be applied.

Worker orientation is a primary consideration for this process, but no definitive method is available to evaluate this factor. Therefore, Dose Reconstructors should use the guidance in Section 4.4, *Exposure Energy and Geometry* of OCAS-IG-001 (NIOSH 2007) for determination of the appropriate selection for DCF geometry.

6.10 UNCERTAINTY

No information was identified to address the uncertainty in the positive recorded photon dose for FMPC workers during the years that film dosimeters were used. Since the FMPC film dosimeters were of the Pardue and Oak Ridge designs, an uncertainty value of ± 30% is recommended (ORAUT 2007). The uncertainty information for the FMPC multielement TLD in Table 6-15 is based on DOELAP accreditation data from a facsimile (Author unknown 2003).

Table 6-15. Uncertainty.

FMPC dosimetry system	Uncertainty
Film (1951-1984)	1.3
Multielement thermoluminescent (1985–present)	1.06

6.11 GEOMETRIC CORRECTION FACTOR

Consideration should be given to geometry when performing dose reconstruction for uranium facility workers who worked with uranium metals, powders, or residues, and for workers who worked on equipment that was contaminated with uranium. An underestimation of the measured and missed photon doses could occur if the energy employee wore the dosimeter on the upper chest or lapel and not in the central area of the chest or on the waist. The organs in the lower torso region are most affected. These include, but are not limited to, the stomach, liver, kidney, ureter, gall bladder, pancreas, small intestine, large intestine, rectum, ovaries, uterus, urinary bladder, and prostate.

The geometric correction factor that was calculated for application to the general population of workers is 2.1 and should be applied as a constant in IREP as determined in DCAS-TIB-0013, *Selected Geometric Exposure Scenario Considerations for External Dose Reconstruction at Uranium Facilities* (NIOSH 2010b). This geometric correction factor should be applied to the measured and missed photon doses for operators, material handlers, and trade workers including chemical operators, production operators, uranium material handlers, pipefitters, carpenters, welders, sheet metal workers, electricians, foremen, etc. The guidance in DCAS-TIB-0013 should be used to adjust the measured and missed photon dose to Fernald workers. Because the factor in DCAS-TIB-0013 was determined by an analysis of photon fluence – not dose computation – the recommendations of that document would apply to workers exposed to thorium as well.

6.12 ATTRIBUTIONS AND ANNOTATIONS

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

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GLOSSARY

accreditation
For external dosimetry, the assessment of whether or not a personnel dosimetry system meets specific criteria. The assessment includes dosimeter performance and the associated quality assurance and calibration programs.

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.

algorithm
Set of rules or steps for solving a problem, especially for calculating a value.

alpha particle (α)
See *alpha radiation*.

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.

beta particle (β)
See *beta radiation*.

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.

deep dose equivalent [*Hp(10)*]
The dose equivalent at the respective depth of 1.0 cm in tissue.

dose equivalent (*H*)
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.

DOE Laboratory Accreditation Program (DOELAP)
Program for accreditation by DOE of DOE site personnel dosimetry and radiobioassay programs based on performance testing and the evaluation of associated quality assurance, records, and calibration programs.

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 *film dosimeter, pocket ionization chamber, and thermoluminescent dosimeter*.

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.

DuPont 552 film packet
Film packet containing DuPont 502 sensitive film and DuPont 510 insensitive film.

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.

extremity
Portion of the arm from and including the elbow through the fingertips and the portion of the leg from and including the knee and patella through the toes.

favorable to claimant
In relation to dose reconstruction for probability of causation analysis, having the property of ensuring that there is no underestimation of potential dose, which often means the assumption of a value that indicates a higher dose than is likely to have actually occurred in the absence of more accurate information. See *probability of causation*.

film
In the context of external dosimetry, radiation-sensitive photographic film in a light-tight wrapping. See *film dosimeter*.

film density
See *optical density*.

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.

filter
Material used (1) in a dosimeter to adjust radiation response to provide an improved tissue equivalent or dose response and (2) in an X-ray machine to selectively absorb photons from the beam to reduce unnecessary exposure of individuals or to improve radiographic quality.

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.

gamma ray, particle, or photon (γ)
See *gamma radiation*.

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*.

luminescence
Emission of light from a material as a result of some excitation. See *thermoluminescence*.

neutron
Basic nucleic particle that is electrically neutral with mass slightly greater than that of a proton. There are neutrons in the nuclei of every atom heavier than normal hydrogen.

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. See *neutron*.

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.

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

optical density
Measure of the degree of opacity of photographic or radiographic film defined as $OD = \log_{10} (I_0/I)$, the base-10 logarithm of the ratio of the reference light intensity I_0 (without film) to the transmitted light intensity (through the film). Also called film density and density reading.

pencil dosimeters
See *pocket ionization chamber*.

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

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 millimeters (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. The International Commission on Radiological Measurement and Units recommended *Hp(d)* in 1993 as dose quantity for radiological protection.

photon
A unit or "particle" of electromagnetic radiation consisting of X- and/or gamma rays.

pocket ionization chamber (PIC)
Cylindrical monitoring device commonly clipped to the outer clothing of an individual to measure ionizing radiation. A PIC may be self-reading or require the use of a outside device

to be able to read the dosimeter. Also called pencil, pocket pencil, pencil dosimeter, and pocket dosimeter.

rad

Traditional unit for expressing absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium. A dose of 1 rad is equivalent to the absorption of 100 ergs per gram (0.01 joules per kilogram) of absorbing tissue. The rad has been replaced by the gray in the International System of Units (100 rad = 1 gray). The word derives from radiation absorbed dose; rad is also the plural.

radiation

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

radioactivity

Property possessed by some elements (e.g., uranium) or isotopes (e.g., ^{14}C) of spontaneously emitting energetic particles (electrons or alpha particles) by the disintegration of their atomic nuclei.

rem

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

rep

Historical quantity of radiation (usually other than X-ray or gamma radiation) originally defined as 93 ergs absorbed per gram in the body and redefined in the 1940s or early 1950s as the amount that would liberate the same amount of energy (93 ergs per gram) as 1 roentgen of X- or gamma rays. Replaced by the gray in the International System of Units; 1 rep is approximately equal to 9.3 milligray. The word derives from roentgen equivalent physical; rep is also the plural.

roentgen

Unit of photon (gamma or X-ray) exposure for which the resultant ionization liberates a positive or negative charge equal to 2.58×10^{-4} coulombs per kilogram (or 1 electrostatic unit of electricity per cubic centimeter) of dry air at 0 degrees Celsius and standard atmospheric pressure. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher energy photons (generally greater than 100 kiloelectron-volts).

shallow dose equivalent [*Hp(0.07)*]

Dose equivalent in units of rem or sievert at a depth of 0.07 millimeters (7 milligrams per square centimeter) in tissue equal to the sum of the penetrating and nonpenetrating doses.

sievert

International System unit for dose equivalent, which indicates the biological damage caused by radiation. The unit is the radiation value in gray (equal to 1 joule per kilogram) multiplied by a weighting factor for the type of radiation and a weighting factor for the tissue; 1 sievert equals 100 rem.

skin dose

See *shallow dose equivalent*.

tissue equivalent

Substance with response to radiation equivalent to tissue. A tissue-equivalent response is an important consideration in the design and fabrication of radiation measuring instruments and dosimeters.

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.

U.S. Atomic Energy Commission

Federal agency created in 1946 to assume the responsibilities of the Manhattan Engineer District (nuclear weapons) and to manage the development, use, and control of nuclear energy for military and civilian applications. The U.S. Energy Research and Development Administration and the U.S. Nuclear Regulatory Commission assumed separate duties from the AEC in 1974. The U.S. Department of Energy succeeded the U.S. Energy Research and Development Administration in 1979.

whole-body 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.

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
FERNALD COWORKER DOSE ASSIGNMENT

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ATTACHMENT A
FERNALD COWORKER DOSE ASSIGNMENT (continued)

A.1 PURPOSE

The purpose of this attachment is to provide information to allow dose reconstructors to assign doses that are based on site coworker data to FMPC workers who have no or limited monitoring data. In addition, the data in this attachment should be used to assign doses for gaps in dosimetry records. The data are to be used in conjunction with ORAUT-OTIB-0020, *Use of Coworker Dosimetry Data for External Dose Assignment* (ORAUT 2011a).

A.2 BACKGROUND

The Oak Ridge Associated Universities (ORAU) Team is conducting a series of coworker data studies to permit dose reconstructors to complete certain cases for which external or internal monitoring data are unavailable or incomplete. Cases that do not have complete monitoring data could fall into one of several categories:

- The worker was unmonitored and, even by today's standards, did not need to be monitored (e.g., a nonradiological worker).
- The worker was unmonitored but, by today's standards, would have been monitored.
- The worker might have been monitored, but the data are not available to the dose reconstructor.
- Partial information is available, but it is insufficient to facilitate a dose reconstruction.

As described in ORAUT-OTIB-0020 (ORAUT 2011a), some cases without complete monitoring data can be processed based on assumptions and methodologies that do not involve coworker data. For example, many cases in the first category can be processed by the assignment of ambient external and internal doses based on information in the relevant site TBDs.

As described in the main text, operations at the site began in 1951. FMPC used film dosimeters between 1951 and 1985. Use of a multielement TLD was implemented in 1985. Exchange frequencies varied from quarterly to weekly dependent on the era. There does not appear to have been any significant administrative practice that would have jeopardized the integrity of the dose of record.

A.3 GENERAL APPROACH

As described in ORAUT-OTIB-0020 (ORAUT 2011a), the general approach to the development of coworker data for cases without external monitoring data is to assign either 50th- or 95th-percentile doses with the intent that the assigned doses represent, but do not underestimate, the doses that would have been assigned had the worker been monitored.

A.4 APPLICATIONS AND LIMITATIONS

Some FMPC workers could have worked at one or more other major sites in the DOE complex during their employment histories. Therefore, the data in this attachment must be used with caution to ensure that, for likely noncompensable cases, unmonitored external doses from multiple site employments have been overestimated. This typically requires the availability of the recorded doses or external coworker dosimetry data for all relevant sites.

ATTACHMENT A
FERNALD COWORKER DOSE ASSIGNMENT (continued)

The data in this attachment address penetrating gamma radiation and nonpenetrating electron and/or low-energy photon radiation. Neutron data are not presented in detail; methods for determining neutron dose are discussed in Section A.7.

External onsite ambient dose should be applied as specified in the latest revision of ORAUT-PROC-0060*, Occupational Onsite Ambient Dose Reconstruction for DOE Sites* (ORAUT 2006b).

A.5 COWORKER DATA DEVELOPMENT

The FMPC HIS-20 database served as the data resource for coworker data analysis. The following items describe assumptions that were made for the analysis:

- Different types of dosimeters were available in the HIS-20 database. Only records with a wear location of "chest" were used in the analysis.
- If a record had a blank Social Security Number, then the record was excluded.
- Deep and shallow dose values that were "null" were excluded and not treated as zeros. The fact that "null" records were not zero values was confirmed by comparison with DOE records from several claimants with the database.
- Dose records from 10 EEOICPA claims were compared to the database, and in all cases the results from HIS-20 data exactly matched the data from DOE under EEOICPA.
- The amount of time a badge was worn by an employee is determined in a calculated field called "WearLength" that is equal the number of days between the End_Date and the Begin_Date in the database.
  - Only records with a WearLength more than zero and less than 400 days were considered in the analysis. Relatively few records were removed from the analysis by filtering the records in this way. Of 35,795 records, 35,251 met the filter criteria of a WearLength more than zero and less than 400 days. Only 544 records did not meet the filtering criteria.
  - If the WearLength was more than 29 days then the dose was adjusted upward to represent 365 days of continuous wear. Dose values were used "as is" in the analysis if the WearLength was less than or equal to 29 days.
- The shallow dose represents the sum of the shielded and open window portions of the dosimeter as discussed in Section 6.4 of this document.

A.6 ADJUSTMENT FOR MISSED DOSE

According to OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* (NIOSH 2007), missed doses are assigned for reported zero readings for each monitoring cycle to account for the possibility that doses were received but either not recorded by the dosimeter or not reported by the site. In addition, reported dose values less than one-half the applicable MDLs are assigned as missed dose. Annual maximum potential missed doses are calculated by multiplying the number of zero or unrecorded badge readings by the reported dosimeter LOD and summing the results. These values are used as the 95th-percentile values of a lognormal distribution to calculate the POC, which is determined by DOL. Therefore, in IREP, Parameter 1 is equal to the calculated maximum annual

ATTACHMENT A
FERNALD COWORKER DOSE ASSIGNMENT (continued)

missed doses multiplied by 0.5 and Parameter 2 is equal to 1.52. These values represent the geometric mean and geometric standard deviation, respectively, for each year of analysis.

The assignment of maximum potential missed doses for monitored workers is particularly significant for FMPC workers from 1952 to 1953 when they could have been monitored weekly. Table A-1 lists the maximum annual missed dose by monitoring period based on information in the main text.

Table A-1. Missed external doses (rem).

Monitoring period	Penetrating and nonpenetrating LOD	Exchange frequency	Maximum potential annual missed penetrating and nonpenetrating dose
1952–1953	0.04	Weekly	2.080
1954–1958	0.03	Biweekly	0.780
1959–1984	0.03	Monthly	0.360
1985–2006	0.02	Quarterly	0.080

A.7 COWORKER ANNUAL DOSE SUMMARIES

Based on the described information and approaches, FMPC coworker annual external dosimetry summaries were developed for use in the evaluation of external penetrating and nonpenetrating dose for certain workers who were potentially exposed to workplace radiation but for whom there is no or limited monitoring data from DOE. These summaries were developed using the following steps:

Step 1. As described in Section A.6, the reported penetrating dose was modified for each worker to account for partial years of employment. This permits the dose reconstructor to assign an appropriate prorated dose to account for partial years of employment or potential exposure.

Step 2. One-half of the maximum potential annual missed doses in Table A-1 were added to the reported annual doses from Step 1 (with the exception of reported positive doses, in which case the maximum missed dose was reduced by the dose that corresponded to one badge exchange because it is not possible that all individual badge results were zero if a positive annual dose was reported).

Step 3. The 50th- and 95th-percentile annual coworker gamma doses were derived from the doses from Step 2 by ranking the data into cumulative probability curves and extracting the 50th- and 95th-percentile doses for each year.

Step 4. Table A-2 lists the results of the coworker analysis. These percentile doses should be used for FMPC workers with no or limited monitoring data through the use of the methodologies in Section 6.0 of ORAUT-OTIB-0020 (ORAUT 2011a). In general, the 50th-percentile dose can be used as a best estimate of a worker's dose when professional judgment indicates that the worker was probably exposed to intermittent low levels of external radiation. The 50th-percentile dose should generally not be used for workers who were routinely exposed. For routinely exposed workers (i.e., workers who were expected to have been monitored and routinely exposed), the 95th-percentile dose should be applied. However, other options are available through the guidance in ORAUT-OTIB-0020. For instance, for cases in which routine monitoring data exist and coworker dose is used to supplement missing quarters or years, the percentile dose should be the one that is consistent with the recorded doses unless there is reason to believe that the worker's job or location in that year differed significantly from the job or location during the years dose was recorded. For workers who

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FERNALD COWORKER DOSE ASSIGNMENT (continued)

are unlikely to have been exposed, external onsite ambient dose should be used rather than coworker doses.

Step 5. Table A-3 lists penetrating dose values (as described in the steps above) that have been adjusted using the guidance in Section 8.0 of ORAUT-OTIB-0052, *Parameters to Consider When Processing Claims for Construction Trade Workers* (ORAUT 2011b). This guidance is applicable for construction trade workers who meet the criteria in Section 3.0 of that document. Because the document does not provide an adjustment factor for nonpenetrating dose, this dose component is not shown in this table.

Step 6. If needed, neutron dose should be calculated using the neutron-to-photon ratios in Table 6-10 of this document. These values would be applied to the penetrating dose values in Table A-2.

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FERNALD COWORKER DOSE ASSIGNMENT (continued)

Table A-2. Annual external coworker doses modified to account for missed dose (rem).

Year	Penetrating 95th percentile	Penetrating 50th percentile	Shallow 95th percentile	Shallow 50th percentile
1952	1,520	1,120	4,320	1,220
1953	1,315	1,040	5,332	1,221
1954	776	390	6,291	676
1955	1,177	390	5,088	676
1956	475	390	5,175	675
1957	475	390	5,099	776
1958	576	390	5,990	1,077
1959	366	180	6,482	1,268
1960	565	180	6,565	1,465
1961	867	180	5,881	867
1962	1,168	265	7,094	1,168
1963	1,769	466	9,591	1,870
1964	1,465	365	7,565	1,265
1965	1,484	265	6,798	1,168
1966	967	299	5,520	967
1967	967	265	6,927	1,168
1968	865	180	5,265	665
1969	867	180	5,179	666
1970	967	265	5,580	666
1971	466	180	3,038	366
1972	765	265	4,765	565
1973	571	265	4,176	505
1974	767	265	4,908	566
1975	666	265	3,760	494
1976	665	265	4,435	465
1977	767	179	4,321	369
1978	967	256	5,470	466
1979	867	198	5,078	366
1980	925	265	5,150	465
1981	698	180	4,906	366
1982	808	186	5,542	366
1983	566	172	7,100	274
1984	665	183	6,320	392
1985	431	35	4,843	113
1986	531	36	4,845	51
1987	485	39	3,758	42
1988	289	40	1,509	40
1989	124	40	170	40
1990	96	40	106	40
1991	111	40	115	40
1992	80	40	84	40
1993	62	40	70	40
1994	59	40	61	40
1995	57	40	64	40
1996	57	40	65	40
1997	53	40	54	40
1998	49	40	49	40
1999	43	40	43	40
2000	54	40	54	40
2001	66	40	67	40
2002	88	40	93	40
2003	82	40	83	40
2004	87	40	88	40
2005	206	40	212	40

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FERNALD COWORKER DOSE ASSIGNMENT (continued)

Table A-3. Annual external coworker doses modified in accordance with ORAUT-OTIB-0052 (rem).

Year	Penetrating 95th percentile	Penetrating 50th percentile
1952	1,720	1,160
1953	1,433	1,040
1954	937	390
1955	1,498	390
1956	515	390
1957	515	390
1958	656	390
1959	446	180
1960	725	180
1961	1,148	180
1962	1,569	305
1963	2,411	586
1964	1,985	445
1965	2,011	305
1966	1,288	353
1967	1,288	305
1968	1,145	180
1969	1,148	180
1970	1,288	305
1971	586	180
1972	1,005	305
1973	734	305
1974	1,007	305
1975	867	305
1976	865	305
1977	1,007	184
1978	1,288	293
1979	1,148	211
1980	1,229	305
1981	912	186
1982	1,065	194
1983	727	175
1984	865	190
1985	592	37
1986	732	39
1987	667	43
1988	392	40
1989	162	40
1990	122	40
1991	143	40
1992	100	40
1993	75	40
1994	71	40
1995	67	40
1996	68	40
1997	62	40
1998	57	40
1999	48	40
2000	64	40
2001	81	40
2002	112	40
2003	103	40
2004	110	40
2005	276	40