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Feed Materials Production Center – Occupational Environmental Dose

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

☐ New ☐ Total Rewrite ☒ Revision ☐ Page Change

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
04/06/2004	00	New Technical Basis Document for the Fernald Environmental Management Project – Occupational Environmental Dose. First approved issue. Initiated by Samuel L.T. Chu.
02/07/2006	01	Approved issue of Revision 01 to add NIOSH "boilerplate" language. Added totals to tables 4.2a and b, revised tables 4.9a and b to include ^{222}Rn, table 4-10a to include intakes uranium and nonuranium radionuclides and table 4-10b to include site wide intakes of ^{222}Rn, added ^{222}Rn concentrations in Table 4A in Appendix A. Constitutes a total rewrite of the document. Initiated by Melton H. Chew.
03/13/2014	02	Revision initiated to update information. Changed site name to Feed Materials Production Center. Incorporated Sanford Cohen & Associates matrix issues. Added Special Exposure Cohort information and includes additional references, updates, and personnel radon exposure data information. Incorporated an auditable set of calculations in a referenced spreadsheet. Combined all intake rate calculations into a single spreadsheet available in the Site Research Database and deleted Attachment E, which contained sample exposure calculations and the detailed calculation results, and Attachment F, which contained the exposure rate calculation results. After 1988, operations ceased, environmental monitoring was increased, and so the Exposure Areas (EA) used to define intakes prior to 1989 is replaced by the standard approach defined in ORAUT-PROC-0031. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Karen S. Kent.
12/03/2015	03	Revision initiated to modify the K-65 radon emissions from the median value to the 95th-percentile value, to include Am-241 in the list of radionuclides that contaminate recycled uranium, and to address S. Cohen & Associates comments. These revisions were made to an auditable set of calculations in an updated referenced spreadsheet. Incorporates minor changes made for clarification purposes. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Karen S. Kent.

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
09/22/2016	04	Revision initiated to correct a unit conversion error in the calculational excel spreadsheet. The value 27.027 was incorrectly assigned to a spreadsheet variable named “BqPERpCi” that was used in several calculations. It was replaced by the correct value of 0.037. The variable was used only in the calculation of intake rates and air concentration after 1988. Corrections are made to the effected values in Tables 4-2, 4-17 and 4-18. The correction has the effect of reducing the intake rate during 1989 and subsequent years by a factor of 0.00137 for all radionuclides, except radon. The correction also reduces the uranium, Th-230 and radon air concentrations presented in Table 4-17 and 4-18 by a factor of 0.00137. Additionally, the specific activity values defined in ORAUT-TKBS-0017-5 for 1% and 2% enriched uranium were used in the revised calculation spreadsheet. The recycled uranium radionuclides and contaminant concentrations specified were applied. The result is a modified Table 4-2. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Objective Manager. Initiated by Karen S. Kent.

TABLE OF CONTENTS

LIST OF TABLES

TABLE **TITLE** **PAGE**

LIST OF FIGURES

FIGURE **TITLE** **PAGE**

ACRONYMS AND ABBREVIATIONS

AMAD	activity median aerodynamic diameter
AMS	air monitoring station
AWE	Atomic Weapons Employer
Bq	becquerel
Ci	curie
d	day
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
DCF	dose conversion factor
DR	dose reconstructor
EA	Exposure Area
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
FEMP	Fernald Environmental Management Project
FMPC	Feed Materials Production Center
ft	foot
g	gram
hr	hour
ICRP	International Commission on Radiological Protection
IREP	Interactive RadioEpidemiological Program
kg	kilogram
L	liter
m	meter
mg	milligram
min	minute
mR	milliroentgen
mrad	millirad
mrem	millirem
MTU	metric ton of uranium
NA	not available
NIOSH	National Institute for Occupational Safety and Health
ORAU	Oak Ridge Associated Universities
pCi	picocurie
ppb	parts per billion
RU	recycled uranium
s	second
SRDB Ref ID	Site Research Database Reference Identification (number)

SEC	Special Exposure Cohort
TBD	technical basis document
TLD	thermoluminescent dosimeter
U.S.C.	United States Code
WL	working level
WLM	working level month
yr	year
µBq	microbecquerel
µm	micrometer
µR	microroentgen
§	section or sections

4.1 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 Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies). These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of the individual work required for each dose reconstruction.

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

Accordingly, a Part B claim for benefits must be based on an energy employee's eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility's designated time period and location (i.e., covered employee). After DOL determines that a claim meets the eligibility requirements under EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and the types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. Unlike the abovementioned statutory provisions on DOE facility definitions that contain specific descriptions or exclusions on facility designation, the statutory provision governing types of exposure to be included in dose reconstructions for DOE covered employees only requires that such exposures be incurred in the performance of duty. As such, NIOSH broadly construes radiation exposures incurred in the performance of duty to include all radiation exposures received as a condition of employment at covered DOE facilities in its dose reconstructions for covered employees. For covered employees at DOE facilities, individual dose reconstructions may also include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction.

NIOSH does not consider the following types of exposure as those incurred in the performance of duty as a condition of employment at a DOE facility. Therefore these exposures are not included in dose reconstructions for covered employees (NIOSH 2010):

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

4.1.1 Purpose

This technical basis document (TBD) is part of the overall Feed Materials Production Center (FMPC) site profile, which describes plant facilities and processes, historical information, and environmental data in relation to dose reconstruction for FMPC workers. This TBD provides a technical basis for evaluating the occupational environmental dose for EEOICPA claimants who were employed at FMPC. Since operations ended in 1988, the Fernald site has been variously named the Fernald Environmental Management Project (FEMP), the Fernald Closure Project, and is now the Fernald Preserve. Occupational environmental dose refers to radiation exposures workers received from radioactive material discharges to the atmosphere from the facilities, while outside the facilities. In addition, environmental dose includes ambient external radiation that originated in the facilities but is outside the buildings or facilities. FMPC employees who did not wear external dosimetry or who were not fully monitored for external exposures are assigned environmental doses.

4.1.2 Scope

This TBD describes the estimated annual intakes for inhalation exposure and estimated radiation doses from ambient external exposures at FMPC to provide the basis for estimating the environmental dose to unmonitored workers or for years and radionuclides that have not already been assessed for dose based on positive bioassay results or missed dose, or whose records were incomplete or unavailable.

For the operational years (1951 to 1988) when stack monitoring efforts were extensive and when environmental monitoring systems were not comprehensive, air concentrations were derived from facility radionuclide emissions. Environmental measurements were used to establish intake rates for years after 1988.

This TBD provides annual environmental intakes and ambient external doses for 1951 to present. Intake rates and exposure after 2009 should be assigned using the values established for 2009.

Exposures to the skin, including localized doses from direct deposition should be reconstructed in accordance with ORAUT-OTIB-0017, *Interpretation of Dosimetry Data for the Assignment of Shallow Dose* (ORAUT 2005), and other applicable Project guidance.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 4.6.

4.1.3 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 SEC:

Thorium, 1968 through 1978

> *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* (Sebelius 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, to which employees at this facility may have been subjected.

Thorium, 1954 through 1967

> *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* (Sebelius 2013a).

It was determined that NIOSH lacked 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.

Uranium, 1951 through 1983 [applies only to subcontractors]

> *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* (Sebelius 2013b)*.*

It was determined that NIOSH lacked sufficient information to allow it to estimate with sufficient accuracy the potential internal doses from exposure to uranium, to which employees of subcontractors who worked at the FMPC may have been subjected.

This dose reconstruction infeasibility for the period from 1951 through 1983 applies only to subcontractors. NIOSH has access to an electronic dataset that contains the results of the uranium urinalysis bioassay program for all 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 dataset. However, the dataset 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 employee bioassay data were retained by the site or captured by data capture efforts. In addition, 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 2013b).

Dose reconstruction guidance in this document for the period from January 1, 1954, to December 31, 1983, 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 less than 250 days). Although it is not possible to completely reconstruct internal radiation doses for all workers for this period, NIOSH has determined, and the U.S. Department of Health and Human Services has concurred, that it is feasible to reconstruct external radiation doses for all FMPC workers for this period (Sebelius 2013b).

4.2 SITE BACKGROUND AND HISTORY

FMPC began limited operations in 1951 and was fully operational by the end of 1954. Its primary function was to convert uranium ore concentrates and recycled materials to either uranium oxides or highly purified uranium ingots and billets for machining or extrusion into tubular forms of standard isotopic assays. These products were for use as production reactor fuel cores and target fuel element fabrication. In addition, FMPC processed relatively small amounts of thorium.

Uranium metal deliveries peaked in 1960 at approximately 10,000 metric tons of uranium (MTU) (Mead, Savage, and Fugate ca. 1985, p. 19). Deliveries began to decline in 1964 and reached a low of approximately 1,230 MTU in 1975. During the 1970s, DOE predecessor agencies considered closing FMPC. The staffing level, which peaked at 2,891 in 1956, slowly declined from 662 in 1972 to 538 in 1979. Starting in 1981, production output increased to three times the 1979 level and employment increased from 538 to more than 1,000 (ASI ca. 1986, p. 4).

The production of uranium metal at FMPC ended in July 1988, and site personnel began to concentrate on cleanup efforts. Current operations are limited to long-term surveillance and maintenance.

Production area facilities included nine separate plants, the Pilot Plant, ancillary buildings, and administrative buildings that were connected by a network of roadways. These facilities, along with concrete storage pads, gravel ground cover, railroad access, sanitary landfill, and metal scrap piles, were surrounded by security fencing (as are the remaining facilities). The Waste Storage Area outside the 136-acre fenced production area included six low-level radioactive waste storage pits, two earthen-bermed concrete silos that contained K-65 (high-specific-activity, low-level, radium-bearing) residues, one concrete silo that contained metal oxides, and all affected adjoining areas. This area included two fly-ash piles about 3,000 ft south-southwest of the waste storage area and a burn pit between Waste Pits 3 and 4. Production operations occurred in Plants 1 to 9 and the Pilot Plant.

In 2005, after remedial actions were completed, responsibility for the site was assigned to DOE Legacy Management. As a result of remedial actions, occupational environmental dose at the site is minimal and is expected to continue to be so in the future.

4.3 SITEWIDE OCCUPATIONAL ENVIRONMENTAL DOSE SUMMARY

Airborne emissions and external gamma radiation data indicate that the radioactive aerosol concentration and gamma dose rates have varied significantly from facility to facility and in various sections of the site. The location where maximum exposure to thorium occurred was not the same location where maximum exposure to uranium occurred. The same is true for radium and radon progeny. This difficulty is addressed by dividing FMPC into smaller areas, which are designated Exposure Areas (EAs), on a grid that provides a realistic estimated representation of radiological conditions where employees might have worked. Figure 4-1 shows the FMPC site plan with the grid. Table 4-1 lists major facilities in each EA. Airborne radionuclide intake rates are presented in Table 4-2. Section 4.4 describes how Table 4-2 was derived. A spreadsheet (ORAUT 2016a) documents the various calculations that were used to develop annual exposures and intake rates for each source, EA, and year combination. The maximum sitewide annual intakes are assigned in the Interactive RadioEpidemiological Program (IREP) as a lognormal distribution with a geometric standard deviation of 3.0.

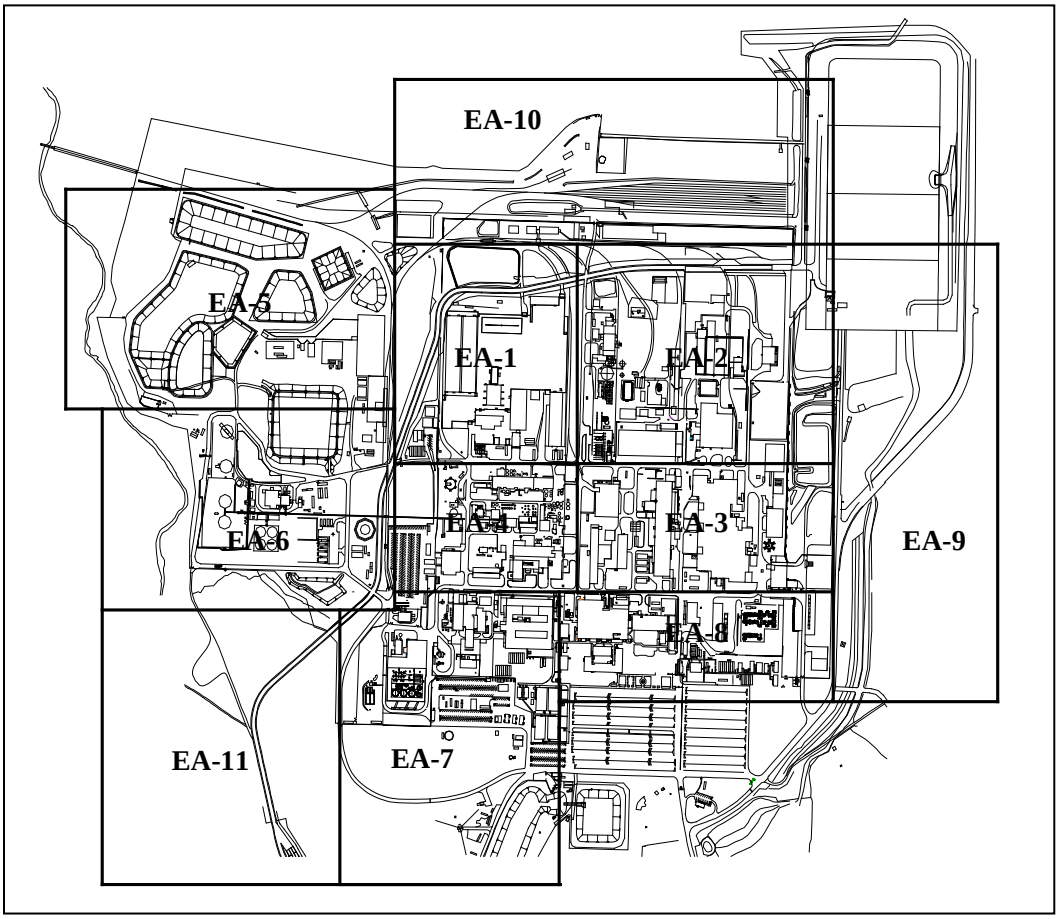

Figure 4-1. EAs for 1951 to 1988 as used in this analysis.

Table 4-1. Facilities and buildings in each EA.

EA	Facilities and buildings
EA-1	Plant 1, Buildings 67 and 68
EA-2	Plant 9, Buildings 64 and 65
EA-3	Plants 4, 5, 6, and 7
EA-4	Plant 2/3, Plant 8, and Incinerator Building
EA-5	Waste Pit Area
EA-6	K-65 silos
EA-7	Pilot Plant, Building 69, Laboratory Building Waste Water Treatment Facility, Storm Water Retention Basins, Administrative Area (expanded into this EA in the 1990s)
EA-8	Administrative Area, Industrial Relations, and Security Buildings
EA-9	Onsite Disposal Facility and Sewage Treatment Plant
EA-10	Railroad
EA-11	Waste Haul Road

The external gamma dose rate sitewide exposure rates by the EAs are presented in ORAUT (2016a) and summarized in Table 4-3.

These doses should be assigned based on work location as defined by an EA. If the work location is unknown or an individual worked in various locations throughout the site in a given year, the annual maximum gamma doses should be assigned for overestimate claims with a constant distribution in IREP. For best-estimate claims or those with a probability of causation greater than 50%, the annual site average gamma doses should be used and assigned as a normal distribution in IREP with a standard deviation of 30%.

Table 4-2. Maximum sitewide average (1951 to 1988) or median (1989 to present) annual intake rates. Intake rates are based on inhalation and in units of WLM/yr for ^{222}Rn and Bq/yr for all other radionuclides.[a,b]

Year	Total U[d] (Bq/yr)	Th-228 (Bq/yr)	Th-230 and Ra-226[c,e] (Bq/yr)	Th-232 (Bq/yr)	Pu-239 (Bq/yr)	Pu-241 (Bq/yr)	Np-237 (Bq/yr)	Tc-99 (Bq/yr)	Ru-103/106[f] (Bq/yr)	Zr/Nb-95[g] (Bq/yr)	Sr-90 (Bq/yr)	Am-241 (Bq/yr)	Rn-222[i] (WLM/yr)
1951	7.32E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.11E-01
1952	4.04E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.25E+01
1953	1.96E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.98E+01
1954	1.43E+03	4.83E+00	0.00E+00	4.83E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.98E+01
1955	3.53E+03	5.52E+00	0.00E+00	5.52E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.73E+01
1956	1.47E+03	5.18E+00	0.00E+00	5.18E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.73E+01
1957	1.04E+03	2.55E-03	0.00E+00	2.55E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.52E+01
1958	1.14E+03	5.83E-03	0.00E+00	5.83E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.77E+01
1959	1.32E+03	5.83E-03	0.00E+00	5.83E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.82E+00
1960	6.92E+03	7.51E-02	0.00E+00	7.51E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.82E+00
1961	1.64E+03	1.46E-01	0.00E+00	1.46E-01	1.35E+00	1.39E+01	4.66E-01	1.72E+02	1.85E+02	2.79E+01	7.44E+00	8.95E-04	1.82E+00
1962	1.17E+03	2.86E-01	0.00E+00	2.86E-01	9.60E-01	9.91E+00	3.32E-01	1.23E+02	1.32E+02	1.99E+01	5.30E+00	6.37E-04	1.82E+00
1963	1.25E+03	3.20E-01	0.00E+00	3.21E-01	1.03E+00	1.06E+01	3.56E-01	1.32E+02	1.42E+02	2.13E+01	5.67E+00	6.83E-04	1.82E+00
1964	9.29E+02	2.46E+00	0.00E+00	2.46E+00	7.63E-01	7.87E+00	2.64E-01	9.76E+01	1.05E+02	1.58E+01	4.21E+00	5.06E-04	1.82E+00
1965	5.40E+02	2.46E+00	0.00E+00	2.46E+00	2.67E-01	2.76E+00	9.24E-02	3.41E+01	3.69E+01	5.51E+00	1.48E+00	1.77E-04	1.82E+00
1966	5.05E+04	3.12E+00	0.00E+00	3.12E+00	2.47E+01	1.59E+02	8.63E+00	3.18E+03	3.44E+03	5.15E+02	1.38E+02	3.25E+00	1.82E+00
1967	4.53E+02	2.46E+00	0.00E+00	2.46E+00	2.22E-01	1.43E+00	7.74E-02	2.86E+01	3.09E+01	4.62E+00	1.24E+00	2.92E-02	1.82E+00
1968	7.94E+02	2.47E+00	0.00E+00	2.47E+00	3.89E-01	2.50E+00	1.36E-01	5.01E+01	5.42E+01	8.10E+00	2.17E+00	5.12E-02	1.82E+00
1969	3.15E+02	1.43E+01	0.00E+00	1.43E+01	1.54E-01	9.93E-01	5.39E-02	1.99E+01	2.15E+01	3.21E+00	8.60E-01	2.03E-02	1.82E+00
1970	2.87E+02	1.05E+01	0.00E+00	1.05E+01	1.41E-01	9.05E-01	4.91E-02	1.81E+01	1.96E+01	2.93E+00	7.84E-01	1.85E-02	1.82E+00
1971	1.73E+02	5.56E+00	0.00E+00	5.56E+00	8.50E-02	5.46E-01	2.97E-02	1.09E+01	1.18E+01	1.77E+00	4.73E-01	1.12E-02	1.82E+00
1972	3.23E+02	1.03E+00	0.00E+00	1.03E+00	1.58E-01	1.02E+00	5.52E-02	2.04E+01	2.20E+01	3.29E+00	8.81E-01	2.08E-02	1.82E+00
1973	4.52E+02	5.69E-01	0.00E+00	5.69E-01	8.86E+00	5.69E+01	2.12E+00	9.49E+01	3.08E+01	4.61E+00	1.23E+00	1.16E+00	1.82E+00
1974	5.04E+02	7.44E-01	0.00E+00	7.44E-01	9.88E+00	6.35E+01	2.37E+00	1.06E+02	3.44E+01	5.14E+00	1.38E+00	1.30E+00	1.82E+00
1975	5.90E+02	5.41E-01	0.00E+00	5.41E-01	1.16E+01	7.44E+01	2.77E+00	1.24E+02	4.03E+01	6.02E+00	1.61E+00	1.52E+00	1.82E+00
1976	4.92E+02	6.63E-01	0.00E+00	6.63E-01	9.54E+00	3.84E+01	2.31E+00	1.03E+02	3.35E+01	5.02E+00	1.34E+00	2.02E+00	1.82E+00
1977	2.12E+02	2.94E+00	0.00E+00	2.94E+00	4.12E+00	1.66E+01	9.97E-01	4.46E+01	1.45E+01	2.16E+00	5.79E-01	8.72E-01	1.82E+00
1978	1.10E+02	2.94E+00	0.00E+00	2.94E+00	2.13E+00	8.55E+00	5.15E-01	2.30E+01	7.47E+00	1.12E+00	2.99E-01	4.50E-01	1.82E+00
1979	2.03E+02	2.94E+00	0.00E+00	2.94E+00	3.94E+00	1.59E+01	9.55E-01	4.27E+01	1.39E+01	2.07E+00	5.55E-01	8.36E-01	1.82E+00
1980	7.79E+01	7.51E-01	0.00E+00	7.51E-01	1.51E+00	6.07E+00	3.66E-01	1.64E+01	5.31E+00	7.94E-01	2.13E-01	3.20E-01	3.67E-01
1981	9.28E+01	7.51E-01	0.00E+00	7.51E-01	1.80E+00	7.24E+00	4.36E-01	1.95E+01	6.33E+00	9.46E-01	2.53E-01	3.81E-01	3.67E-01
1982	7.98E+01	7.51E-01	0.00E+00	7.51E-01	1.55E+00	6.23E+00	3.75E-01	1.68E+01	5.44E+00	8.14E-01	2.18E-01	3.28E-01	3.67E-01
1983	1.34E+02	7.51E-01	0.00E+00	7.51E-01	2.61E+00	1.05E+01	6.31E-01	2.82E+01	9.16E+00	1.37E+00	3.67E-01	5.52E-01	3.67E-01
1984	9.45E+01	7.51E-01	0.00E+00	7.51E-01	1.83E+00	7.37E+00	4.44E-01	1.99E+01	6.45E+00	9.64E-01	2.58E-01	3.89E-01	3.67E-01
1985	7.52E+01	7.51E-01	0.00E+00	7.51E-01	1.46E+00	5.86E+00	3.53E-01	1.58E+01	5.13E+00	7.67E-01	2.05E-01	3.09E-01	3.67E-01
1986	7.66E+01	7.51E-01	0.00E+00	7.51E-01	1.48E+00	3.70E+00	3.60E-01	1.61E+01	5.23E+00	7.82E-01	2.09E-01	3.85E-01	3.67E-01
1987	7.31E+01	7.85E-01	0.00E+00	7.85E-01	1.41E+00	3.53E+00	3.44E-01	1.54E+01	4.99E+00	7.46E-01	2.00E-01	3.67E-01	3.67E-01
1988	7.11E+01	8.14E-01	0.00E+00	8.14E-01	1.37E+00	3.43E+00	3.34E-01	1.49E+01	4.85E+00	7.25E-01	1.94E-01	3.57E-01	2.17E-01
1989	9.19E-02	4.37E-08	0.00E+00	6.26E-08	1.77E-03	4.44E-03	4.32E-04	1.93E-02	6.27E-03	9.37E-04	2.51E-04	4.61E-04	9.64E-02
1990	3.26E-02	1.55E-08	0.00E+00	2.22E-08	6.30E-04	1.58E-03	1.53E-04	6.85E-03	2.23E-03	3.33E-04	8.91E-05	1.64E-04	9.64E-02
1991	4.20E-02	2.00E-08	0.00E+00	2.86E-08	8.10E-04	2.03E-03	1.97E-04	8.81E-03	2.86E-03	4.28E-04	1.15E-04	2.11E-04	1.05E-01

Year	Total U[d] (Bq/yr)	Th-228 (Bq/yr)	Th-230 and Ra-226[c,e] (Bq/yr)	Th-232 (Bq/yr)	Pu-239 (Bq/yr)	Pu-241 (Bq/yr)	Np-237 (Bq/yr)	Tc-99 (Bq/yr)	Ru-103/106[f] (Bq/yr)	Zr/Nb-95[g] (Bq/yr)	Sr-90 (Bq/yr)	Am-241 (Bq/yr)	Rn-222[i] (WLM/yr)
1992	5.46E-02	2.60E-08	0.00E+00	3.72E-08	1.05E-03	2.64E-03	2.57E-04	1.15E-02	3.72E-03	5.57E-04	1.49E-04	2.74E-04	7.99E-02
1993	6.09E-02	2.90E-08	0.00E+00	4.15E-08	1.18E-03	2.94E-03	2.86E-04	1.28E-02	4.16E-03	6.22E-04	1.66E-04	3.06E-04	1.25E-01
1994	2.13E-02	1.01E-08	0.00E+00	1.45E-08	4.11E-04	1.03E-03	1.00E-04	4.48E-03	1.45E-03	2.17E-04	5.82E-05	1.07E-04	1.29E-01
1995	1.53E-02	7.29E-09	0.00E+00	1.04E-08	2.96E-04	7.40E-04	7.20E-05	3.22E-03	1.04E-03	1.56E-04	4.18E-05	7.69E-05	1.21E-01
1996	3.46E-02	1.65E-08	0.00E+00	2.36E-08	6.68E-04	1.67E-03	1.63E-04	7.27E-03	2.36E-03	3.53E-04	9.45E-05	1.74E-04	1.58E-01
1997	6.83E-02	3.25E-08	0.00E+00	4.65E-08	1.32E-03	3.30E-03	3.21E-04	1.43E-02	4.66E-03	6.96E-04	1.86E-04	3.43E-04	1.70E-01
1998	5.06E-02	2.41E-08	0.00E+00	3.45E-08	9.77E-04	2.44E-03	2.38E-04	1.06E-02	3.45E-03	5.16E-04	1.38E-04	2.54E-04	1.13E-01
1999	7.33E-02	3.49E-08	1.67E-03	4.99E-08	1.41E-03	3.54E-03	3.44E-04	1.54E-02	5.00E-03	7.47E-04	2.00E-04	3.68E-04	1.21E-01
2000	6.59E-02	3.14E-08	1.27E-02	4.49E-08	1.27E-03	3.18E-03	3.10E-04	1.38E-02	4.50E-03	6.73E-04	1.80E-04	3.31E-04	1.05E-01
2001	6.59E-02	3.14E-08	4.93E-02	4.49E-08	1.27E-03	3.18E-03	3.10E-04	1.38E-02	4.50E-03	6.73E-04	1.80E-04	3.31E-04	1.05E-01
2002	1.84E-02	8.78E-09	3.86E-02	1.26E-08	3.56E-04	8.91E-04	8.67E-05	3.87E-03	1.26E-03	1.88E-04	5.04E-05	9.26E-05	1.21E-01
2003	1.53E-01	7.29E-08	1.40E-02	1.04E-07	2.96E-03	7.40E-03	7.20E-04	3.22E-02	1.04E-02	1.56E-03	4.18E-04	7.69E-04	1.05E-01
2004	8.66E-01	4.12E-07	2.80E-02	5.90E-07	1.67E-02	4.18E-02	4.07E-03	1.82E-01	5.90E-02	8.83E-03	2.36E-03	4.35E-03	1.05E-01
2005	2.33E-02	1.11E-08	5.19E-03	1.59E-08	4.50E-04	1.13E-03	1.10E-04	4.90E-03	1.59E-03	2.38E-04	6.36E-05	1.17E-04	1.00E-01
2006	4.36E-03	2.08E-09	0.00E+00	2.97E-09	8.42E-05	2.11E-04	2.05E-05	9.16E-04	2.98E-04	4.45E-05	1.19E-05	2.19E-05	1.13E-01
2007	1.73E-03	8.24E-10	0.00E+00	1.18E-09	3.34E-05	8.36E-05	8.14E-06	3.64E-04	1.18E-04	1.77E-05	4.73E-06	8.69E-06	9.64E-02
2008	1.20E-03	5.71E-10	0.00E+00	8.16E-10	2.31E-05	5.79E-05	5.63E-06	2.52E-04	8.18E-05	1.22E-05	3.27E-06	6.02E-06	1.13E-01
2009-present[h]	9.32E-04	4.44E-10	0.00E+00	6.35E-10	1.80E-05	4.50E-05	4.38E-06	1.96E-04	6.36E-05	9.51E-06	2.55E-06	4.68E-06	1.13E-01

a. The material type (F, M, and S) to be used for dose reconstruction should be that which is favorable to the claimant and consistent with the specific claim and guidance in ICRP Publication 68 (ICRP 1995).
b. Total uranium is interpreted as U-234 for dose reconstruction purposes.
c. Intake rates for Th-230 and Ra-226 are equal.
d. Three short-duration episodic emissions (Table 4-6) were modeled using Equation 4-7 and averaged into the annual intake rates.
e. These intake rates are associated with the remediation of waste pits that occurred from 1999 through 2005. Intake rates for Th-230 and Ra-226 for other years are negligible.
f. Ru-103/106 is assumed to be 100% Ru-106, due to its longer half-life.
g. Zr/Nb-95 pair is assumed to be in equilibrium, with equal activities totaling the intake rate indicated in the table.
h. Results for 2010 to present based on 2009, the last year of monitoring.
i. The ^{222}Rn intake rate for 1952 through 1988 is based on the 95th-percentile value of the K-65 silo emission rates; 1951 is based on median value of Plant 1 pad drum storage emission rate.

Table 4-3. External gamma radiation (mrem/yr).[a,b,c]

Year	EA-1	EA-2	EA-3	EA-4	EA-5	EA-6	EA-7	EA-8	EA-9	EA-10	EA-11	Site average	Standard deviation	Max.
1952	241.2	169.3	169.2	241.1	14.7	34.9	12.6	20.7	15.4	13.5	9.8	85.7	97.7	241
1953	294.4	206.5	206.4	294.1	19.6	60.0	15.4	25.3	18.8	16.5	12.0	106.3	118.2	294
1954	294.6	206.6	206.5	294.3	21.7	82.3	15.5	25.4	18.9	16.6	12.1	108.6	117.4	295
1955	490.8	344.2	344.1	490.3	34.0	114.8	25.8	42.2	31.4	27.6	20.1	178.7	196.4	491
1956	980.6	688.1	687.9	980.1	61.3	140.5	51.2	84.3	62.7	55.0	39.7	348.3	397.0	981
1957	980.3	688.0	687.8	980.0	56.8	140.0	51.2	84.2	62.6	54.9	39.7	347.8	397.3	980
1958	980.3	688.0	687.8	980.0	57.2	140.0	51.2	84.2	62.7	55.0	39.7	347.8	397.3	980
1959	980.4	688.0	687.9	980.0	58.3	140.1	51.2	84.2	62.7	55.0	39.7	348.0	397.2	980
1960	980.6	688.0	687.9	980.1	60.5	140.4	51.2	84.2	62.7	55.0	39.7	348.2	397.1	981
1961	980.7	688.1	687.9	980.1	62.5	140.6	51.2	84.3	62.7	55.1	39.7	348.4	396.9	981
1962	980.8	688.1	688.0	980.2	64.5	140.8	51.3	84.3	62.7	55.1	39.8	348.7	396.8	981
1963	980.9	688.2	688.0	980.2	66.5	141.0	51.3	84.3	62.7	55.1	39.8	348.9	396.7	981
1964	981.4	688.4	688.1	980.5	69.4	202.0	51.5	84.4	62.8	55.2	40.1	354.9	393.8	981
1965	981.9	688.6	688.2	980.8	72.0	263.0	51.7	84.5	62.9	55.3	40.3	360.8	391.7	982
1966	982.4	688.8	688.4	981.1	74.7	324.0	51.9	84.6	62.9	55.4	40.6	366.8	390.5	982
1967	982.9	689.0	688.5	981.4	77.1	384.9	52.1	84.7	63.0	55.5	40.9	372.7	390.2	983
1968	983.2	689.1	688.6	981.6	76.6	445.6	52.2	84.8	63.0	55.5	41.1	378.3	390.9	983
1969	983.7	689.3	688.7	981.9	79.1	506.5	52.4	84.9	63.1	55.6	41.4	384.2	392.3	984
1970	984.2	689.5	688.8	982.2	81.6	567.5	52.6	85.0	63.2	55.7	41.6	390.2	394.5	984
1971	847.5	593.5	592.6	845.4	76.9	621.3	45.7	73.4	54.5	48.1	36.4	348.7	347.2	848
1972	710.9	497.4	496.5	708.5	72.4	675.2	38.7	61.7	45.8	40.5	31.2	307.2	305.7	711
1973	564.4	394.5	393.5	561.9	67.3	728.5	31.3	49.2	36.5	32.4	25.6	262.3	270.1	729
1974	427.8	298.4	297.3	425.1	62.7	782.3	24.4	37.6	27.8	24.8	20.3	220.8	250.0	782
1975	310.7	216.1	214.9	307.8	59.1	837.2	18.5	27.6	20.4	18.3	15.9	186.0	247.0	837
1976	302.8	214.3	213.0	299.7	73.5	910.0	30.5	38.8	32.0	30.2	28.3	197.6	261.2	910
1977	351.4	247.9	247.1	349.0	68.5	558.1	29.0	39.6	31.9	29.6	25.7	179.8	182.6	558
1978	330.8	233.3	232.3	328.0	72.1	733.4	28.9	38.6	31.2	29.1	26.1	189.4	218.8	733
1979	443.3	310.1	308.0	438.5	86.5	1,615.0	34.4	46.5	36.2	33.2	31.3	307.5	465.0	1,615
1980	419.0	293.6	290.8	413.0	99.9	2,145.6	39.0	49.2	39.3	36.6	37.0	351.2	615.0	2,146
1981	740.5	515.5	512.8	734.4	106.5	2,150.8	44.1	65.6	48.4	43.0	38.1	454.5	631.9	2,151
1982	754.1	526.4	524.3	749.2	101.7	1,626.1	45.0	67.8	50.5	45.1	38.2	411.7	498.5	1,626
1983	766.2	534.5	532.2	760.9	106.4	1,802.3	45.8	68.7	51.1	45.6	39.1	432.1	543.5	1,802
1984	361.0	253.9	252.1	356.7	91.5	1,281.9	33.2	42.9	34.7	32.4	30.8	251.9	366.8	1,282
1985	294.3	208.6	206.8	289.9	94.9	1,318.8	35.0	42.2	35.5	33.9	33.4	235.8	374.2	1,319
1986	516.1	361.2	359.7	512.2	92.0	1,091.1	35.1	50.6	38.8	35.4	30.5	283.9	332.9	1,091
1987	60.6	46.1	44.8	57.0	79.9	923.3	25.1	24.9	23.6	23.8	25.9	121.4	266.6	923
1988	252.2	181.0	179.0	247.3	103.4	1,517.4	40.8	45.9	40.2	39.0	39.5	244.2	430.7	1,517
1989	196.6	141.4	139.7	192.3	93.1	1,247.7	34.0	37.9	33.4	32.6	32.9	198.3	354.1	1,248
1990	87.4	62.7	61.3	83.7	75.7	988.4	20.0	20.9	18.8	18.7	20.2	132.5	285.2	988
1991	279.9	195.9	194.3	289.8	81.6	545.7	24.5	31.9	25.4	23.8	22.4	155.9	167.0	546
1992	184.7	132.2	131.9	145.4	65.6	66.8	21.8	27.4	23.5	22.6	19.4	76.5	61.0	185

Year	EA-1	EA-2	EA-3	EA-4	EA-5	EA-6	EA-7	EA-8	EA-9	EA-10	EA-11	Site average	Standard deviation	Max.
1993	269.4	190.0	189.6	277.1	66.3	73.2	21.6	30.1	24.2	22.7	18.3	107.5	103.4	277
1994	323.0	227.2	226.7	325.2	68.4	74.7	23.2	33.5	26.4	24.4	19.3	124.7	124.7	325
1995	306.0	215.7	215.3	334.0	69.1	111.6	23.9	33.6	26.9	25.0	20.2	125.6	120.6	334
1996	21.2	18.9	18.5	19.4	61.5	180.0	18.6	18.3	18.1	18.6	18.8	37.4	49.0	180
1997	21.3	18.8	18.4	19.4	62.0	224.0	18.5	18.2	18.0	18.4	18.7	41.4	61.9	224
1998	19.8	17.3	16.8	17.8	60.7	236.4	16.9	16.6	16.4	16.8	17.2	41.1	66.1	236
1999	19.7	17.2	16.8	17.8	60.5	226.4	16.9	16.6	16.4	16.8	17.2	40.2	63.1	226
2000	20.0	17.3	16.8	18.0	61.2	268.4	17.0	16.7	16.4	16.9	17.3	44.2	75.5	268
2001	20.9	18.2	17.7	18.9	62.4	295.2	17.9	17.5	17.2	17.7	18.2	47.4	83.2	295
2002	21.8	19.0	18.5	19.7	63.3	300.0	18.7	18.3	18.0	18.5	19.0	48.6	84.4	300
2003	18.2	16.1	15.8	16.6	58.0	121.2	15.8	15.7	15.5	15.9	15.9	29.5	32.9	121
2004	18.7	16.8	16.5	17.1	57.9	69.8	16.5	16.4	16.2	16.7	16.5	25.4	19.2	70
2005	26.1	23.8	23.3	24.3	66.6	205.4	23.4	23.2	22.9	23.4	23.6	44.2	55.0	205
2006	26.1	23.8	23.3	24.3	66.6	205.4	23.4	23.2	22.9	23.4	23.6	44.2	55.0	205
2007	26.1	23.8	23.3	24.3	66.6	205.4	23.4	23.2	22.9	23.4	23.6	44.2	55.0	205
2008	26.1	23.8	23.3	24.3	66.6	205.4	23.4	23.2	22.9	23.4	23.6	44.2	55.0	205
2009–present[d]	26.1	23.8	23.3	24.3	66.6	205.4	23.4	23.2	22.9	23.4	23.6	44.2	55.0	205

a. External exposures include ambient doses because background doses were not available for all of the dose components.
b. Dose reconstructors should use ambient (*H*10*) dose conversion factors (DCFs) for 1952 to 1976 (based on instrument readings) and exposure (*R*) DCFs for 1977 to the present (based on readings from a TLD on a fence line) (NIOSH 2007a, Section 4.1.2).
c. Ambient gamma radiation doses are based on 2,000 hr/yr.
d. Results for 2010 to the present based on 2009, the last year of monitoring.

4.4 INTERNAL DOSES FROM ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

This section describes the derivation of radionuclide concentrations in the general areas for each year for the unmonitored worker dose reconstruction effort using available documented technical information and data from the system of air monitoring stations (AMSs). Table 4-4 indicates the types of information and data available from the beginning of operations at FMPC.

Table 4-4. Historical information and data available for environmental dose reconstruction.

Period	Available historic information and data description	Source document
1951–1988	Airborne uranium, thorium, and radon emissions including unmonitored and episodic releases.	*Radiation Doses and Risk to Residents from FMPC Operations from 1951–1988, Volumes I and II* (RAC 1998a, 1998b). *History of FEMP Radionuclide Discharges* (Boback et al. 1987) *Addendum to FMPC-2082, "History of FEMP Radionuclide Discharges"* (Dolan and Hill 1988) *The Fernald Dosimetry Reconstruction Project, Tasks 2 and 3, Radionuclide Source Terms and Uncertainties* (Voillequé et al. 1995).
1989–2005	Estimated airborne emissions for FMPC for radionuclides other than radon reported in the annual site environmental reports.	Environmental reports for 1989 to 2005 (Dugan et al. 1990; Byrne et al. 1991; WEMCO 1992; FERMCO 1993 to 1996; FDF 1997 to 1999; Fluor Fernald 2000 to 2006.)
1989–1996	Radon in ambient air from continuous, high-volume AMSs at the site fence line locations. Results for onsite locations not reported until 1997.	Environmental reports for 1989 to 1996 (Dugan et al. 1990; Byrne et al. 1991; WEMCO 1992; FERMCO 1993 to 1996; FDF 1997)
1997–2005	Radon in ambient air data from the Radon Monitoring Program that included AMSs at onsite locations.	Environmental reports for 1997 to 2005 (FDF 1998, 1999; Fluor Fernald 2000 to 2006.)
2006–present[a]	Ambient particulate and radon air monitoring data collected by DOE Legacy Management. Particulate through 2009 and radon through 2008.	Environmental reports for 2006 to 2012 (Stoller 2007; DOE 2008 to 2013)

a. For dose reconstruction purposes, the last year that ambient air was monitored and reported at the site was 2009.

It is necessary to develop appropriate methodologies for determining the atmospheric radionuclide concentrations specific to the type of information available for the various periods in Table 4-4. Therefore, this TBD presents the derivation of the internal occupational environmental dose in separate periods as listed in the first column of Table 4-4.

Note that beginning in 1989, the kind of data available to determine intake rates was different than the kind of data available in earlier years. Before 1989, point source radionuclide emission rates were available. Beginning in 1989, ambient environmental radionuclide air concentrations were routinely measured and reported. Different kinds of data required different calculation methods. Before 1989, the calculation method involved atmospheric dispersion of radionuclides from elevated and ground level emission point sources. These calculations included assumptions that were favorable to the claimant. Beginning in 1989, intakes were calculated as the product of the median respirable air concentration, breathing rate, and duration of occupancy. As a result of the relatively high radionuclide emission rates prior to 1989 and the calculation method, intake rates were higher before 1989 than they were starting in 1989.

4.4.1 Radioactive Materials Air Concentrations and Intakes, 1951 to 1988

4.4.1.1 Radionuclide Discharges and Sources of Emissions

During FMPC production operations, approximately 308,000 kg of uranium and an estimated 14,500 kg of thorium were released to the atmosphere from various emission sources during routine operations. *History of FEMP Radionuclide Discharges* (Boback et al. 1987) and *The Fernald Dosimetry Reconstruction Project, Tasks 2 and 3, Radionuclide Source Terms and Uncertainties* (Voillequé et al. 1995) provide the most recent evaluations of radioactive emissions from the FMPC operating years (1951 through 1988). Dolan and Hill (1988), an addendum to Boback et al. (1987), provides the initial assessments of the emissions from site operations. For the purpose of environmental dose reconstruction, this TBD uses the larger of the emissions amounts from either Voillequé et al. (1995) or Boback et al. (1987). The values for routine and nonroutine uranium discharges from operations and episodic releases during the operating years are from Voillequé et al (1995). The thorium discharges are from Dolan and Hill (1988). The values for releases of ^{222}Rn and its progeny are from Voillequé et al. (1995) for the K-65 silo releases and drummed K-65 materials at the Plant 1 Pad. Radon releases from the material in Q-11 silos were derived from Pinney et al. (2004).

Table 4-5 lists annual airborne uranium emissions due to FMPC operations throughout the operating history (1951 through 1988) from each production plant and processing facility from Voillequé et al. (1995, p. 643).

The values in Table 4-5 include:

- Monitored source emissions from dust collectors,
- Emissions from Plant 2/3 UO_3 gulping,
- Emissions from Plant 8 scrubbers,
- Building exhaust emissions due to normal operating conditions,
- Fugitive emissions from waste pits,
- Nonroutine uranium fires and UF_6 gas releases,
- Incinerator and burner emissions, and
- Six episodic releases of uranium documented in incident reports.

Table 4-6 reproduces Table K-43 of Voillequé et al. (1995, p. 643) to show a summary of six episodic releases of uranium that were identified from the incident reports and air monitoring.

Table 4-7 summarizes annual fugitive thorium emissions due to wind erosion from four of the six waste pits in the waste storage area (Dolan and Hill 1988). Only Waste Pits 2, 3, 4, and 5 released thorium. Data on uranium and radon emissions from these waste pits are also published in Dolan and Hill (1988). The uranium data are not used in this analysis because the values from this source are lower than the values Voillequé et al. (1995) subsequently estimated (Table 4-5). Radon effluents from the waste pits are not included because they contributed a negligible fraction of the total site annual radon emission rate, which in many years was in the kilocurie range.

Table 4-8 lists annual airborne emissions from thorium operations at Plants 8 and 9 and the Pilot Plant from 1951 through 1988 from Dolan and Hill (1988).

RAC (1998b) provides estimates of the annual emission of ^{222}Rn from the K-65 silos for the period from 1952 through 1988. The 95th-percentile values of these estimates, which define upper estimates of emission rates, have been used in this analysis. Median estimates of the emission rates (Voillequé et al. 1995, p. 435) were applied to materials stored at Plant 1 for 1951. Those materials stored at Plant 1 were eventually placed in the K-65 silos.

Table 4-5. Uranium emissions summary (kg) (Voillequé et al. 1995).

Year	Plant 1	Plant 2/3[a]	Plant 4	Plant 5	Plant 6[b]	Plant 7[c]	Plant 8[d]	Plant 9[e]	Pilot Plant[f]	Building exhaust[g]	Waste pits	UF_6 release[h]	Year total
1951	0.0	0.0	0.00.0	0.0	2.9	0.0	0.0	0.0	11	0.0	0.0	0.0	14
1952	0.0	0.0	0.0	0.0	19	0.0	0.0	0.0	300	16	0.0	44	379
1953	5.5	245	2,500	2,000	62	0	245	0	500	16	0.0	44	5,617
1954	60	1,547	7,000	7,000	121	5,500	3,192	0	520	171	0.0	44	25,155
1955	59	3,429	14,000	14,000	153	25,000	6,015	0	800	236	0.0	44	63,736
1956	55	4,311	6,400	4,500	174	4,200	8,082	0	35	206	0.0	44	28,007
1957	80	5,911	1,200	4,800	174	0.0	6,062	265	20	256	125	0.0	18,893
1958	300	5,360	700	850	217	0.0	7,531	712	30	291	230	44	16,265
1959	55	5,534	2,000	600	164	0.0	9,019	458	40	326	230	45	18,470
1960	35	5,601	220	250	346	0.0	9,864	254	1,470	381	285	45	18,750
1961	45	4,898	550	150	174	0.0	12,568	81	120	366	340	45	19,337
1962	40	2,096	1,000	490	105	0.0	9,157	163	130	323	345	45	13,894
1963	85	2.0	2,000	900	217	0.0	10,447	183	55	306	320	45	14,560
1964	28	193	650	400	38	0.0	7,794	275	15	308	310	45	10,056
1965	5.2	685	370	300	62	0.0	2,134	71	11	282	165	45	4,130
1966	21	1,196	300	90	16	0.0	1,073	50	21	167	125	45	3,104
1967	30	1,831	300	180	6.2	0.0	1,521	81	15	181	20	44	4,210
1968	0.8	1,927	300	110	46	0.0	3,481	122	4.0	146	20	0.0	6,157
1969	45	1,057	53	130	5.3	0.0	1,260	13	4.0	92	20	0.0	2,679
1970	7.0	660	42	89	8.8	0.0	1,268	15	1.0	108	20	0.0	2,219
1971	15	1,251	1.3	1.0	5.0	0.0	276	1.0	1.0	23	20	0.0	1,594
1972	70	2,719	20	45	4.0	0.0	100	25	1.0	23	20	0.0	3,027
1973	2.0	3,860	75	102	4.0	0.0	120	15	1.0	24	20	0.0	4,223
1974	1.0	4,344	30	40	4.0	0.0	146	41	1.0	25	20	0.0	4,652
1975	7.3	4,733	150	26	4.0	0.0	468	1.0	1.0	31	23	0.0	5,445
1976	3.0	4,114	30	17	5.3	0.0	276	3.1	1.0	20	24	0.0	4,493
1977	0.8	1,227	16	75	4.0	0.0	624	1.0	12	16	24	0.0	2,000
1978	2.0	2.0	16	40	4.0	0.0	544	79	2.0	11	24	0.0	724
1979	1.0	2.0	100	17	4.0	0.0	1,077	2.0	1.0	10	25	0.0	1,239
1980	18	14	200	130	4.0	0.0	125	1.0	2.2	16	25	0.0	535
1981	1.0	56	530	160	4.0	0.0	153	1.0	1.0	21	25	0.0	952
1982	2.0	89	27	140	5.0	0.0	279	5.1	1.0	25	25	0.0	599
1983	8.0	172	60	60	6.5	0.0	628	1.0	1.0	28	25	0.0	990
1984	13	505	42	100	4.0	0.0	204	204	3.2	23	25	0.0	1,123

Year	Plant 1	Plant 2/3[a]	Plant 4	Plant 5	Plant 6[b]	Plant 7[c]	Plant 8[d]	Plant 9[e]	Pilot Plant[f]	Building exhaust[g]	Waste pits	UF_6 release[h]	Year total
1985	2.0	48	55	40	3.9	0.0	160	18	0.6	23	25	1.5	378
1986	1.0	19	32	20	3.3	0.0	266	9.2[e]	0.3	20	25	7.5	403
1987	0.0	65	20	15	3.2	0.0	153	6.1	0.2	2.6	25	1.5	292
1988	0.0	32[e]	0.6	0	3.0	0.0	133	0	1.0	0.5	25[e]	1.5	197
Total	**1,104**	**69,736**	**40,990**	**37,867**	**2,184**	**34,700**	**106,444**	**3,158**	**4,134**	**4,519**	**2,980**	**680**	**308,495**

a. Emissions from Plant 2/3 include dust collectors, scrubbers, and unmonitored releases.
b. Emissions from Plant 6 include dust collectors, unmonitored releases, and cooling tower releases due to their proximity to Plant 6.
c. Emissions from Plant 7 include dust collectors and episodic releases.
d. Emissions from Plant 8 include dust collectors, scrubbers, and unknown episodic and incinerator releases due to their proximity to Plant 8.
e. Emissions from Plant 9 include dust collectors and unmonitored releases.
f. Emissions from the Pilot Plant include dust collectors and episodic releases.
g. Particulate emissions from the building exhaust and nonroutine fire releases are combined and assumed to be released at the center of the FMPC operational areas. Plant 4 is near the center of the operational area, so the Plant 4 stack location is assumed, however, the release height is assumed to be ground level.
h. Emissions from nonroutine UF6 releases are assumed to be ground-level gaseous releases from the location of the Pilot Plant.

Table 4-6. Summary of six episodic releases of uranium from incident reports and air monitoring (Voillequé et al. 1995, p. 643).

Event	Start	Description	Uranium release (kg)	Duration of release
1	11/07/1953	Release of UF_6 from defective cylinder in Pilot Plant	45	15 min
2	11/12/1960	Dust loss from dust collector bags in Pilot Plant	310	1 or 3 d
3	02/14/1966	Release of UF_6 from cylinder in Pilot Plant due to operator error	750	1 hr
4	09/21/1978	Unknown	20–370[a]	7 d
5	02/01/1979	Unknown	60–680[a]	7 d
6	09/13/1983	Unknown	290–360[a]	7 d

a. Range of values is based on results from several ambient AMSs.

Table 4-7. Fugitive thorium emissions (kg) from wind erosion of Waste Pits 2, 3, 4, and 5 (Dolan and Hill 1988).

Year	Pit 2	Pit 3	Pit 4	Pit 5	Total
1957	0.014	0.0	0.0	0.0	0.014
1958	0.032	0.0	0.0	0.0	0.032
1959	0.032	0.0	0.0	0.0	0.032
1960	0.032	0.0	0.38	0.0	0.412
1961	0.036	0.0	0.76	0.0	0.796
1962	0.036	0.0	1.53	0.0	1.566
1963	0.036	0.0	1.72	0.0	1.756
1964	0.036	0.01	1.91	0.0	1.956
1965	0.018	0.014	1.91	0.0	1.942
1966	0.01	0.014	2.1	0.0	2.124
1967	0.0	0.0	2.1	0.0	2.1
1968	0.0	0.0	2.3	0.0	2.3
1969	0.0	0.005	2.3	0.0	2.305
1970	0.0	0.01	2.3	0.0	2.31
1971	0.0	0.01	2.3	0.0	2.31
1972	0.0	0.005	2.49	0.0	2.495
1973	0.0	0.005	2.49	0.0	2.495
1974	0.0	0.0	2.49	0.1	2.59
1975	0.0	0.01	2.68	0.24	2.93
1976	0.0	0.01	2.87	0.76	3.64
1977	0.0	0.01	3.25	0.29	3.55
1978	0.0	0.01	3.64	0.29	3.94
1979	0.0	0.01	3.83	0.29	4.13
1980	0.0	0.005	3.83	0.29	4.125
1981	0.0	0.005	3.83	0.29	4.125
1982	0.0	0.0	3.83	0.29	4.12
1983	0.0	0.0	3.83	0.29	4.12
1984	0.0	0.0	3.83	0.29	4.12
1985	0.0	0.0	3.83	0.29	4.12
1986	0.0	0.0	3.83	0.29	4.12
1987	0.0	0.0	3.83	0.48	4.31
1988	0.0	0.0	3.83	0.64	4.47
Total	**0.282**	**0.133**	**79.8**	**5.1**	**85.4**

Table 4-8. Thorium emissions data by year (kg) (Dolan and Hill 1988).

Year	Plant 8	Plant 9	Pilot Plant	Yearly total
1953	0	0	0	0
1954	0	1,028	0	1,028
1955	0	1,176	0	1,176
1956	0	0	0	0
1957	0	0	0	0
1958	0	0	0	0
1959	0	0	0	0
1960	0	0	0	0
1961	0	0	0	0
1962	0	0	0	0
1963	0	0	0	0
1964	0	0	344	344
1965	0	0	344	344
1966	118	0	344	462
1967	0	0	344	344
1968	0	0	344	344
1969	1,040	0	394	1,434
1970	699	0	499	1,198
1971	430	0	62	492
1972	0	0	141	141
1973	0	0	50	50
1974	0	0	100	100
1975	0	0	3	3
1976	0	0	0	0
1977	0	0	408	408
1978	0	0	408	408
1979	0	0	408	408
1980	0	0	0	0
1981	0	0	0	0
1982	0	0	0	0
1983	0	0	0	0
1984	0	0	0	0
1985	0	0	0	0
1986	0	0	0	0
1987	0	0	0	0
1988	0	0	0	0
Total	**2,287**	**2,204**	**4,193**	**8,684**

Pinney et al. (2004) identified another large radon emission source, the Q-11 ore silos near Plant 2/3, which provided a relatively larger exposure to workers in its vicinity from 1952 to 1958 when the silos were filled with highly radioactive ore. Hornung et al. (2008, p. 9) also provided information on this topic. Interviews with former employees revealed that six smaller silos in the production area had been used in the 1950s to store high-grade ore from the African Metals Corporation before processing. This ore had high ^{226}Ra content, which decays to ^{222}Rn. Although the amount of material was considerably less than that stored in the K-65 silos, the Q-11 silos were much closer to the production area and were not sealed.

Figure 13 in Pinney et al. (2004) reported the yearly mean and maximum exposure in working level months (WLMs) to workers due to the Q-11 emissions during the years when these silos served as radon sources (1952 to 1958). Figure 13 from Pinney et al. (2004) is reproduced in Figure A-1 in Attachment A. The yearly radon emissions from the Q-11 silos were estimated based upon a rearrangement of the atmospheric dilution factor (χ/Q) equation and the maximum exposures (in WLM) from Figure 13. A detailed description of the treatment of adding the Q-11 silos to the overall radon release source terms for those years the silos were active is provided in Attachment A.

Each of these radon emission sources are combined and summarized in Table 4-9. Waste pits were also a source of radon at FMPC, but the released amounts were negligible in comparison with those

in Table 4-9. To ensure the radon intake rate is not underestimated, the estimated naturally occurring background radon levels are included in this evaluation. An ambient background radon concentration of 0.47 pCi/L is assumed. This value is based on the average of the measured background radon concentrations from 1989 to 2000 (Environmental Report Fluor Fernald 2001). Attachment C shows background concentration measurements.

Table 4-9. Releases (Bq) of ^{222}Rn used in this analysis for 1952 through 1988 (RAC 1998b; Voillequé et al. 1995, p. 435; Pinney et al. 2004).

Year	K-65 silos	Q-11 silo	Plant 1 pad, drum storage	Annual total
1951	0.00E+00	0.00E+00	1.30E+12	1.30E+12
1952	1.96E+14	3.05E+14	2.15E+13	5.23E+14
1953	1.96E+14	6.37E+14	3.52E+12	8.37E+14
1954	3.70E+14	6.31E+14	0.00E+00	1.00E+15
1955	3.70E+14	9.15E+14	0.00E+00	1.29E+15
1956	3.70E+14	9.15E+14	0.00E+00	1.29E+15
1957	3.70E+14	8.81E+14	0.00E+00	1.25E+15
1958	3.70E+14	7.59E+14	0.00E+00	1.13E+15
1959	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1960	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1961	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1962	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1963	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1964	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1965	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1966	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1967	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1968	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1969	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1970	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1971	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1972	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1973	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1974	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1975	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1976	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1977	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1978	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1979	4.81E+14	0.00E+00	0.00E+00	4.81E+14
1980	8.88E+13	0.00E+00	0.00E+00	8.88E+13
1981	8.88E+13	0.00E+00	0.00E+00	8.88E+13
1982	8.88E+13	0.00E+00	0.00E+00	8.88E+13
1983	8.88E+13	0.00E+00	0.00E+00	8.88E+13
1984	8.88E+13	0.00E+00	0.00E+00	8.88E+13
1985	8.88E+13	0.00E+00	0.00E+00	8.88E+13
1986	8.88E+13	0.00E+00	0.00E+00	8.88E+13
1987	8.88E+13	0.00E+00	0.00E+00	8.88E+13
1988	4.81E+13	0.00E+00	0.00E+00	4.81E+13

4.4.1.2 Exposure Site Locations

Airborne emissions data indicate that the amount of normal releases at FMPC has varied significantly from facility to facility. This TBD analysis assumed that the radioactive aerosol concentration varied significantly in various sections of the site. The dose a worker received is highly dependent on the amount of time spent in specific work areas. To provide supportive information for dose

reconstruction, this TBD has divided FMPC into smaller areas (designated as EAs) on a grid that provides a realistic estimated representation of radiological conditions where employees might have worked. Figure 4-1 and Table 4-1 above show the FMPC site plan with the grid and the major facilities in each EA, respectively. This TBD presents airborne radionuclide intakes from operations for the 11 EAs. These calculations are documented in ORAUT (2016a).

4.4.1.3 Airborne Radionuclide Concentration Derivation

The radioactive aerosol concentrations at various distances from a continuous source in each EA can be obtained by:

$$C_{EA} = \sum_i R_i \left(\frac{\chi}{Q} \right)_{di} f_i(RF) \qquad (4\text{-}1)$$

where

C_{EA} = radionuclide concentration in EA (g/m^3)
R_i = radionuclide emission rate from source *i* (g/s) where *i* can be Plant 1, 2/3, 4, etc.
$(\chi/Q)_{di}$ = atmospheric dilution factor at distance *d* from source *i* (s/m^3)
f_i = fraction of time that the wind blew toward the EA from source *i*
RF = respirable fraction of the source aerosol [<10 µm (DOE 1994b)]

Airborne radionuclide concentrations for each EA were calculated as the sum of the contributions of all emission sources to the most densely occupied area in the EA. In EAs that contained multiple emission sources (i.e., EA-3 and EA-4), the radionuclide concentrations were determined at the vicinity of the highest emission source in the EA.

4.4.1.3.1 Radionuclide Emission Rates, $R_i \times RF$

The average annualized emission rates were derived by dividing emission amounts in Tables 4-5, 4-8, and 4-9 by 3.156×10^7 seconds (an average year). In 1951 and 1953, ^{222}Rn and radon progeny were released from stored drums of K-65 materials at the Plant 1 pad over 3 months in one case and 6 months in the other (Table 4-9). However, their release rates were annualized as if the total emission amount were released over a 1-year period.

For internal dose assessment, the particle sizes of interest are those below 10 µm, which are respirable and sufficiently small to remain airborne during transport. Spenceley (1985) presents 16 datasets on emission stack particle size distributions for 15 different stacks in three different plants (Plants 4, 5, and 9). The activity median aerodynamic diameter (AMAD) and the 1-sigma uncertainty were reported for each determination. The range of the measured AMAD values was 0.72 to 9.8 µm. The stack with the lowest value was apparently resampled with a result of 9.0 µm. Figure 4-2 shows the measured particle size distributions from the stacks. The average value of the AMAD was 6.6 µm.

The measurements were made using a multi-stage particle size separator, so a cumulative distribution of the "percent greater than stated size" was reported. The scale of the "stated size" was not consistent with each determination, but a value of about 10 µm (the nominal cutoff value associated with respirable particles) was available in each case. Typically (in 13 out of 16 instances) the cutoff value was greater than 11 µm, so the percentage of respirable particles tends to be slightly overestimated, which results in a bias that is favorable to the claimant for dose reconstruction purposes. Figure 4-3 shows the percentage of the emissions from the various stacks that were less than nominally 10 µm. The average of the values represented in Figure 4-3 is 74%.

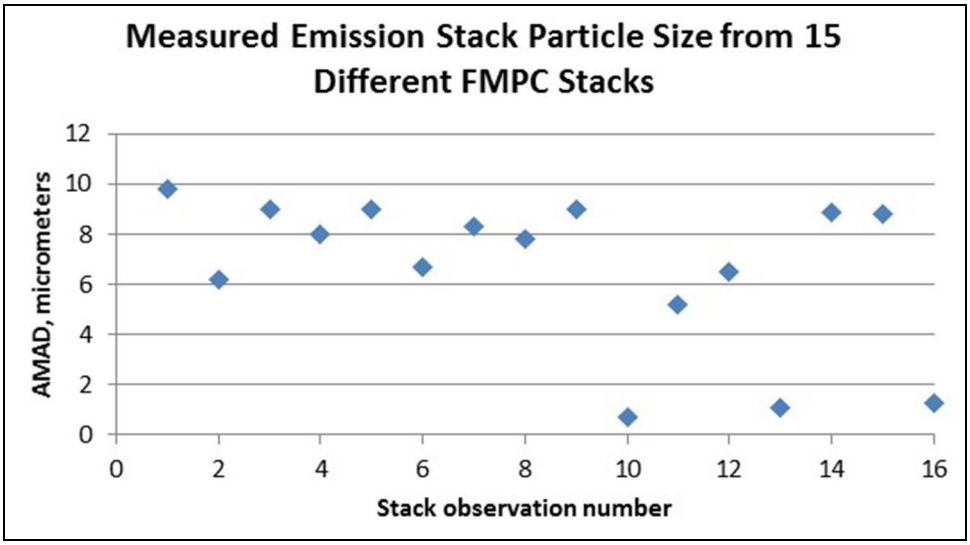

Figure 4-2. Measured emission stack particle size from 15 different FMPC stacks (Spenceley 1985).

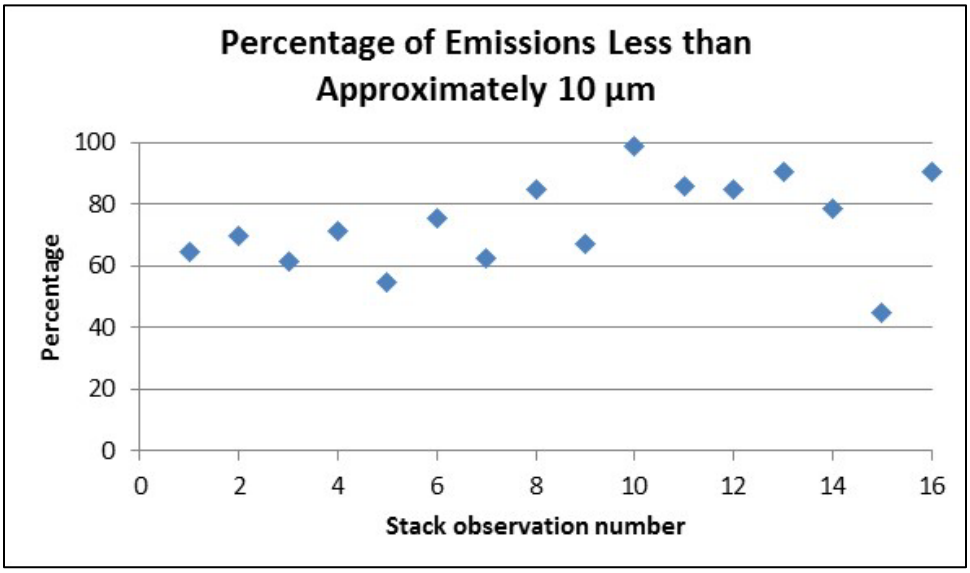

Figure 4-3. Percentage of stack emission less than approximately 10 µm AMAD (Spenceley 1985).

Appendix E of *The Fernald Dosimetry Reconstruction Project* (Voillequé et al. 1995, p. 291) indicates that, in general, the size distribution of emissions from uranium processes involved particle sizes below 10 µm in the 60th to 70th percentile, weighted toward the 60th percentile. In particular, Table E-8 of Voillequé et al. (1995, p. 291) indicates that the composite of U_3O_8 releases from the dust collectors that showed more than 75% of the emission had a particle size less than 10 µm. Therefore, for particulate emissions, a value of 75% was used for *RF* in Equation 4-1. For gaseous releases – including UF_6, radon, and radon progeny – a value of 100% was used.

4.4.1.3.2 Far-Field Atmospheric Dilution Factor, *χ/Q*

The atmospheric dilution factor *χ/Q* can be calculated using the Pasquill-Gifford equation for the dispersion of gas from a continuous source (Cember 1983, p. 347):

$$\frac{\chi(x,y)}{Q} = \frac{1}{\pi\sigma_y\sigma_2 u} exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2}+\frac{H^2}{\sigma_2^2}\right)\right] \quad (4\text{-}2)$$

where

$\chi(x,y)$ = ground-level concentration at point x,y (g/m^3)
x = downwind distance on plume centerline (m)
y = crosswind distance (m) (in all cases y = 0)
Q = emission rate (g/s)
σ_y, σ_z = horizontal and vertical standard deviations of contaminant concentration in the plume (m), Figures 4-4 and 4-5, respectively
u = mean wind speed at level of plume centerline (m/s)
H = effective release height (m)

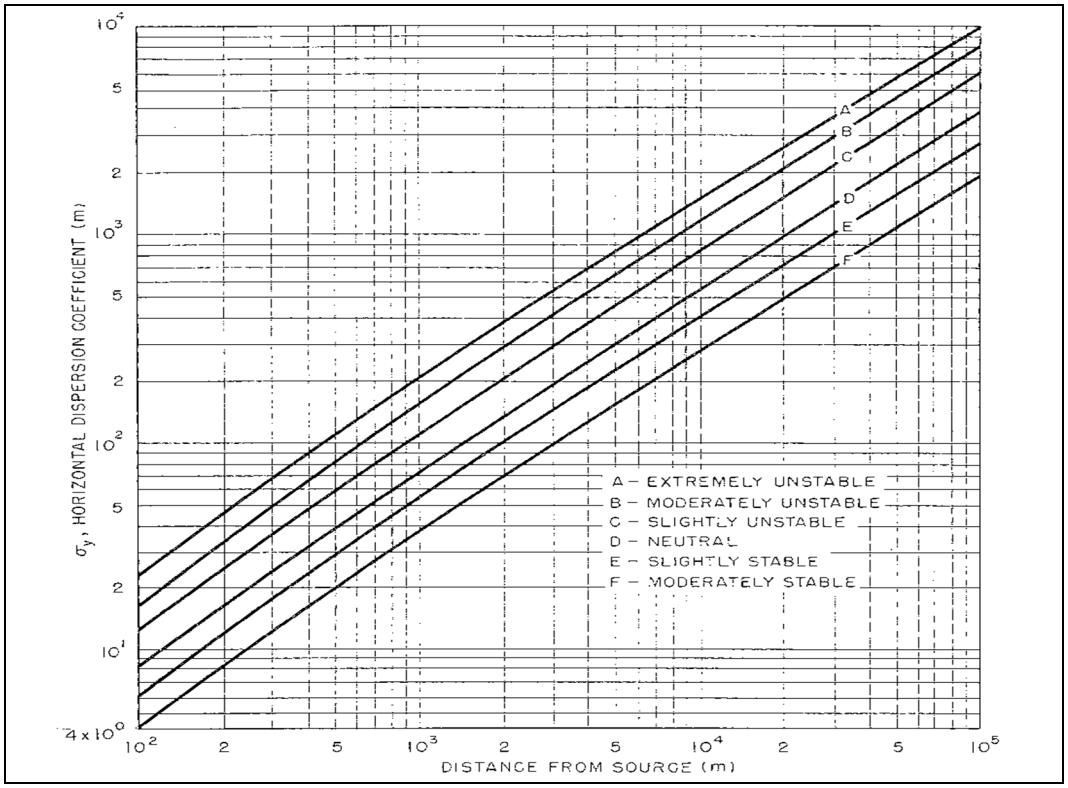

Figure 4-4. Horizontal diffusion σ_y versus downwind distance from point source for Pasquill's stability categories (Slade 1968, p. 115).

For particulates, the χ/Q equation becomes:

$$\frac{\chi(x,y)}{Q} = \frac{1}{2\pi\sigma_y\sigma_2 u} \exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2}+\frac{H^2}{\sigma_2^2}\right)\right] \quad (4\text{-}3)$$

Equation 4-2 yields the ground-level concentration of a gas that is continuously emitted from a point source and is based on total reflection of the gas by the ground and an elevated mixing lid (inversion) that keeps the contaminant within the bounds of the ground and lid. If the pollutant in the plume is retained on the ground, as in the case of particulates (Equation 4-3), the ground-level concentration is approximately half of that in Equation 4-2. Note that these equations do not assume depletion of the

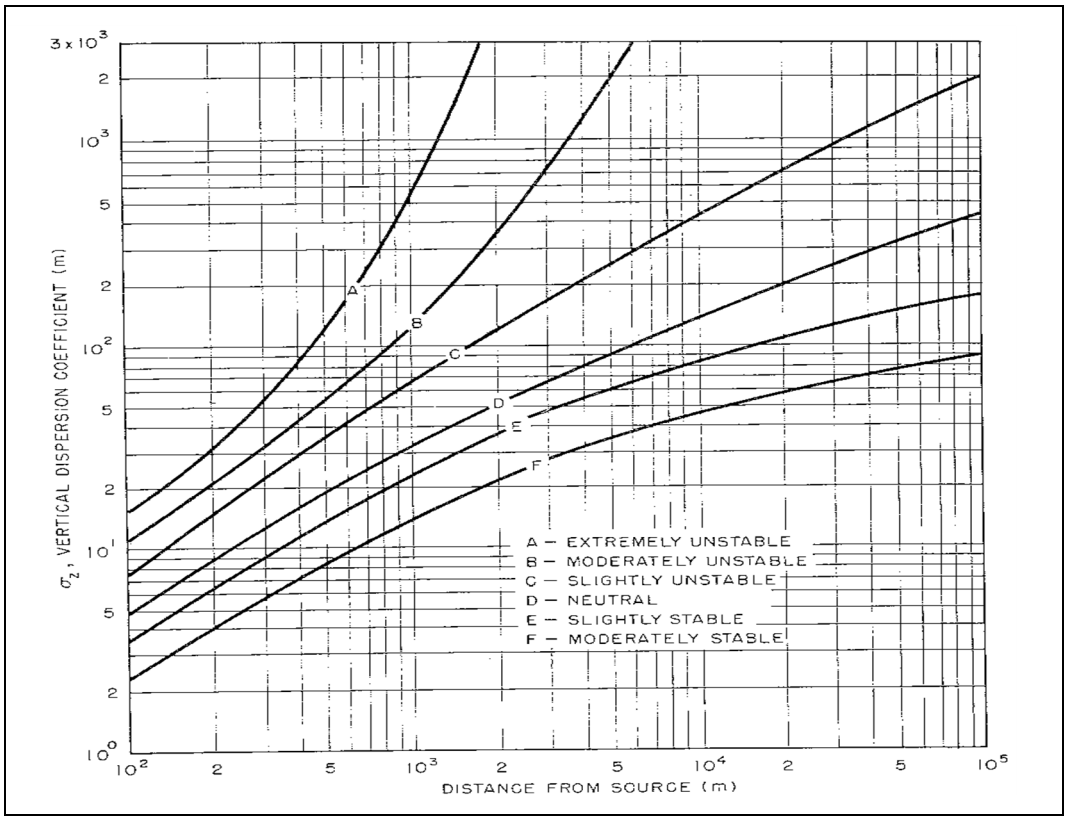

Figure 4-5. Vertical diffusion σ_z versus downwind distance from continuously emitting point source for Pasquill's stability categories (Slade 1968, p. 116).

plume by such mechanisms as dry or wet deposition or gravitational settling, which are known to occur and would act to reduce plume concentrations.

Instead of manually interpreting the values of σ_y and σ_z from Figures 4-4 and 4-5, an alternate method (Napier and Ramsdell 2005) can be applied. This method results in the same values for σ_y and σ_z when the limitations of graphical interpretation are considered.

4.4.1.3.3 Receptor Downwind Distances

The horizontal and vertical standard deviations of radionuclide concentration in the plume (σ_y, and σ_z) are functions of the downwind receptor distances from the emission sources. Table 4-10 lists the EA receptor downwind distance estimates from each contributing emission source that were used for radioactivity concentration calculations. The estimated distances were based on the line-of-sight from the emission sources to the closest occupied location in each EA. These distances are the shortest distance between the emission sources and the occupied locations.

For EAs that contained multiple emission sources (i.e., EA-3 and EA-4), the receptor locations were placed at the vicinity (within 50 m) of the highest emission source. Pasquill's stability category F was used for the σ_y and σ_z determination because it is the most favorable to claimants. In accordance with Figures 4-4 and 4 5, the F stability category represents a moderately stable condition that has the lowest σ_y and σ_z, which results in a higher material concentration downwind from the point of release. Ground releases and elevated releases were assumed as appropriate to the configuration of the emission sources. The elevated releases assumed an average stack height of 10 m from Table 2 of Boback et al. (1987, p. 54), which is also favorable to claimants because Table 2 shows that the overwhelming majority of the FMPC stacks have been well over 30 m above ground level (not accounting for stack exit velocity or plume buoyancy that in both cases would increase the effective

Table 4-10. Receptor downwind estimated distances from contributing emission sources (m).

EA	Plant 1	Plant 2/3	Plant 4	Plant 5	Plant 6	Plant 7	Plant 8	Plant 9	Pilot Plant	Waste pits	K-65 silos
1	50	150	150	250	400	250	250	350	350	450	500
2	300	250	150	150	150	200	300	25	500	800	900
3[a]	250	150	50	100	100	100	150	150	300	750	750
4[a]	100	100	100	200	300	100	50	350	150	550	550
5	550	350	550	650	800	550	450	800	500	100	400
6	450	550	700	850	950	700	550	1,000	450	300	50
7	350	200	250	300	400	150	100	550	50	700	500
8	450	250	200	200	150	100	200	400	300	950	800
9	800	700	500	400	250	550	700	250	900	1,000	1,500
10	400	600	550	600	600	650	650	450	800	250	700
11	600	450	750	800	1,000	650	500	1,000	300	500	200

a. The receptor was located at the vicinity of the highest emission source.

stack height). The effects of building wake were taken into account in estimating close-in concentrations (below 200 m) from an elevated release. In 1963, lateral support berms were added to K-65 silos. As a consequence, the K-65 release height is estimated to be 10 m through 1962 and 0 m beginning in 1963. The method for modifying *χ/Q* to account for building wake effects is presented in Section 4.4.1.3.5. Because uranium and thorium emissions from the production plants were particulates that were released through the stacks, an elevated release *χ/Q* was used for uranium and thorium releases. A ground-level release *χ/Q* was used for uranium and thorium fugitive emissions from the waste pits. Radon-222 was treated as a ground-level gaseous release primarily from the headspace of the K-65 and Q-11 silos and from drummed material storage near Plant 1.

4.4.1.3.4 Effects of Wind Direction, *fi*

Wind direction plays a key role in predicting the amount of aerosol that arrived at an EA. In Equation 4-2, the mean wind speed *u* at the level of the plume centerline is dependent on the general wind direction and expressed as wind rose data. The wind rose data include the percentage of time that the wind blew in a particular direction. The site environmental reports for 1991 to 2001 (WEMCO 1992; FERMCO 1993 to 1996; FDF 1997 to 1999; Fluor Fernald 2000 to 2002) provided FMPC meteorological conditions for climatic condition evaluations and atmospheric modeling. Only 9 years of the data (1991 to 1997 and 2000 to 2001) were sufficiently legible for use. This TBD used the 95th-percentile values of this wind rose data (Figure 4-6) to determine the wind direction frequency for each of the compass points. Table 4-11 presents the summary wind rose data that were used in this analysis. Attachment B contains more detail.

4.4.1.3.5 Concentration Fields Near Buildings, Modified *χ/Q* for Building Wake

When the assumed height of the release is within 2.5 times the height of the surrounding structures, estimation of the aerosol concentrations near buildings needs to consider the effect of the building wake that entrains some of the effluent and brings it to ground level. The building wake effect can be accounted for by calculating the modified diffusion factors σ_y' and σ_z' using the method in Randerson (1984). The modified expressions for σ_y' and σ_z' for an elevated release are:

$$\sigma_y' = 0.7(W/2) + 0.0967(x - 3H_b) \quad (4\text{-}4)$$

$$\sigma_z' = 0.7H_b + 0.067(x - 3H_b) \quad (4\text{-}5)$$

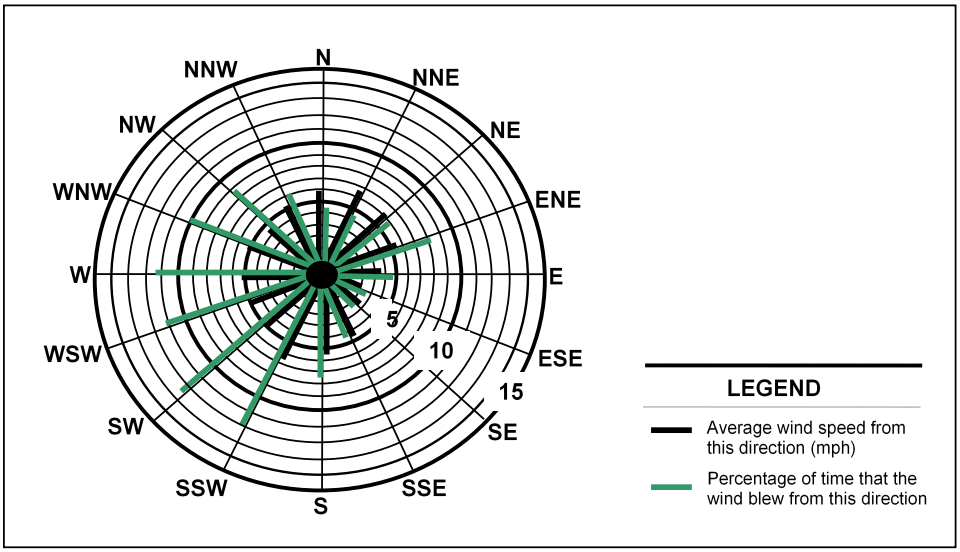

Figure 4-6. 95th-percentile wind rose data, 10-m height, 1991 to 1997 and 2000 to 2001 (FDF 1998, p. 35; Fluor Fernald 2002, p. 28).

Table 4-11. 95th-percentile values from wind rose data

Wind from the …	Frequency	Speed (m/s)
N	0.0456	2.70
NNE	0.0468	2.80
NE	0.058	2.60
ENE	0.083	2.60
E	0.0472	1.77
ESE	0.028	1.42
SE	0.028	1.67
SSE	0.0426	2.12
S	0.0756	2.62
SSW	0.125	2.94
SW	0.1296	2.24
WSW	0.113	2.24
W	0.1156	2.36
WNW	0.103	2.41
NW	0.0892	2.20
NNW	0.0712	2.41

and

$$\frac{\chi}{Q} = \frac{1}{\pi \sigma'_y \sigma'_z u} \exp\left[-\frac{1}{2}\left(\frac{h_e^2}{\sigma'^2_z} \right) \right] \qquad (4\text{-}6)$$

where

H_b = building height (m)
W = building width (m)
x = distance to the receptor (m)
h_e = effective release height (m)
u = mean wind speed (m/s)

For particulates, multiply Equation 4-6 by 0.5. Information on building dimensions (i.e., width and roof peak height) from Boback et al. (1987, p. 54) and from Figure 1 of ORAUT (2011, p. 15) for the K-65

silos were used for the near-building χ/Q calculations. Table 4-12 lists the building or structure dimensions that were used to solve Equations 4-4 and 4-5.

Table 4-12. Building and K-65 silo dimensions for building wake χ/Q calculations.

Structures	Width ft (m)	Roof peak height ft (m)
Plant 1	100 (30.5)	60 (18.3)
Plant 2/3	60 (18.3)	67 (20.4)
Plant 4	165 (50.3)	94 (28.7)
Plant 5	100 (30.5)	52 (15.8)
Plant 6	200 (61.0)	39 (11.9)
Plant 7	80 (24.4)	114 (34.7)
Plant 8	60 (18.3)	48 (14.6)
Plant 9	225 (68.6)	40 (12.2)
Pilot Plant	210 (64.0)	54 (16.5)
K-65 silos	80 (24.4)	36 (11.0)

4.4.1.3.6 Episodic Releases

Table 4-6 summarizes six significant episodic releases of uranium that were identified from incident reports and air monitoring data with a total median release estimate of 1,700 kg of uranium. For comparison purposes, the total estimated uranium emissions to the atmosphere from nonroutine events between 1952 and 1988 was 2,784 kg (Voillequé et al. 1995, p. 40), which is 2% of the approximately 135,000 kg of uranium that was previously reported by Boback et al. (1987) for 1951 through 1984 (Voillequé et al. 1995, p. 616). Releases that had occurred in a period of less than a day (i.e., Events 1, 2, and 3) were evaluated as quasi-instantaneous (puff) releases. The remaining episodic releases (i.e., Events 4, 5, and 6), which occurred over about a week, were treated as continuous releases. The exposures from the puff releases were considered as additions to the annual environmental doses, which also include all of the episodic uranium releases from Table 4-6.

The exposures from these events would have been received during the months when the releases occurred, but for dose reconstruction are averaged into the annual intake rate.

The atmospheric diffusion equation for puff release (Slade 1968) is:

$$\frac{\psi}{Q} = \frac{1}{\pi \sigma_{yl} \sigma_{zl} u} \exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_{yl}^2} + \frac{h^2}{\sigma_{zl}^2} \right) \right] \quad (4\text{-}7)$$

where

ψ = consequence or exposure to a receptor expressed in terms of the integrated concentration (g/m^3)
Q = emission rate (g/s)
σ_{yl}, σ_{zl} = horizontal and vertical standard deviations of puff concentration (m)
u = mean wind speed at level of plume centerline (m/s)
y = crosswind distance (m) (in all cases $y = 0$)
h = effective release height (m)

For particulates, multiply Equation 4-7 by 0.5. Slade (1968, p. 419) presented equations to calculate values of σ_{yl} and σ_{zl} as a function of downwind distance x in meters. To apply these equations, this analysis assumed that the default Class F atmospheric stability condition corresponds to "very stable" conditions. The value of σ_{yl} is calculated as:

$$\sigma_{yl} = 0.14 \times x^{0.93} \quad (4\text{-}8)$$

where

x = downwind distance (m)

The value of σ_{zl} is calculated as:

$$\sigma_{zl} = 0.05 \times x^{0.61} \quad (4\text{-}9)$$

The equation for the radionuclide intakes using ψ/Q can be written:

$$I = q \times \left(\frac{\psi}{Q}\right) \times f \times Br \times RF \quad (4\text{-}10)$$

where

I = radionuclide intake (g)
q = episodic release amount (g)
(ψ/Q) = atmospheric dilution factor for puff release (s/m^3) (see Attachment C)
f = fraction of time that the wind blew toward the EA (Figure 4-7)
Br = breathing rate (m^3/s)
RF = respirable fraction of the source aerosol (<10 µm)

4.4.1.4 Radionuclide Intake Derivation

The intake calculation assumed a breathing rate of 2,400 m^3/yr. The isotopic content in the uranium emissions varied based on the type and form of uranium that was processed in various plants and during different periods.

For purposes of dose reconstruction, the assumption was made that, through 1964, all uranium processed at FMPC was enriched in ^{235}U to a level of 1%. After 1964, all uranium was assumed to be 2% enrichment (NIOSH 2007b, p. 48). Table 4-13 lists the primary assumptions for FMPC uranium enrichments and the isotopes that are associated with these enrichments.

Table 4-13. Uranium enrichment isotopes (ORAUT 2016b).

Isotope	Uranium enrichment	Specific activity (Bq/mg U)	Activity fraction
U-234	1% enriched	36.1	0.637
U-235	1% enriched	36.1	0.021
U-238	1% enriched	36.1	0.342
U-234	2% enriched	59.8	0.769
U-235	2% enriched	59.8	0.0268
U-238	2% enriched	59.8	0.204

RU was first processed in significant amounts at FMPC beginning in 1961, so none of the RU-associated contaminants were present before 1961. Most of the RU came from Hanford and was recycled from weapons-grade plutonium. Therefore, a 6% weapons-grade mixture that had been

chemically purified in 1961 was selected for bounding calculations. For dose reconstruction, the plutonium alpha dose from the plutonium mixtures in RU is assumed to be from 100% ^{239}Pu. Plutonium mixture ratios for fresh, 10-, 20-, and 30-year aged intervals after purification were used to estimate the ^{241}Pu and ^{241}Am contaminant levels (ORAUT 2016b).

Table 4-14 presents the RU contaminant intakes per unit activity of uranium.

Table 4-14. RU contaminant intakes per unit activity of uranium (ORAUT 2016b).

Isotope	1961–1964 (Bq/Bq U)	1965 (Bq/Bq U)	1966–1972 (Bq/Bq U)	1973–1975 (Bq/Bq U)	1976–1985 (Bq/Bq U)	1986 on (Bq/Bq U)
U	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Pu-alpha[a]	8.21E–04	4.94E–04	4.90E–04	1.96E–02	1.94E–02	1.93E–02
Pu-241	8.47E–03	5.10E–03	3.15E–03	1.26E–01	7.80E–02	4.83E–02
Am-241	5.45E–07	3.28E–07	6.44E–05	2.57E–03	4.11E–03	5.02E–03
Np-237	2.84E–04	1.71E–04	1.71E–04	4.70E–03	4.70E–03	4.70E–03
Tc-99	1.05E–01	6.31E-02	6.31E-02	2.10E–01	2.10E–01	2.10E–01
Th-232	1.13E–06	6.81E-07	6.81E-07	6.81E–07	6.81E–07	6.81E–07
Th-228	7.91E–07	4.76E–07	4.76E–07	4.76E–07	4.76E–07	4.76E–07
Ru-103/106[b]	1.13E–01	6.82E-02	6.82E-02	6.82E-02	6.82E-02	6.82E–02
Zr-95	1.70E–02	1.02E–02	1.02E–02	1.02E–02	1.02E–02	1.02E–02
Nb-95	1.70E–02	1.02E–02	1.02E–02	1.02E–02	1.02E–02	1.02E–02
Sr-90	4.53E–03	2.73E–03	2.73E–03	2.73E–03	2.73E–03	2.73E–03

a. The plutonium alpha mixture is assessed as 100% Pu-239. Am-241 and Pu-241 are assessed based on 6% weapons-grade plutonium mixture.
b. Ru-103/106 is assumed to be 100% Ru-106, due to its longer half-life.

Plant 2/3, the Refinery, was operated to convert the U.S. government stockpile of natural uranium ore concentrates to UO_3. Ore process operation in Plant 2/3 started in 1953 and continued until 1962 when UO_3 production was transferred to the Mallinckrodt Chemical Company at the Weldon Spring Plant. In 1966, the UO_3 refinery at the Weldon Spring Plant was shut down; the refinery at Fernald was reactivated in April 1966 and continued operations through 1977 when it was placed in standby (Mead, Savage, and Fugate ca. 1985, pp. 20–22). In 1988, the refinery operated intermittently in a sequential campaign to convert enriched recycled materials to oxide. The natural uranium ore concentrates (often called yellowcake) include the decay products of uranium, ^{226}Ra, and ^{230}Th. To a large extent these impurities were removed during the milling process, but some of the decay products remained in the ore concentrates. These decay products could have been emitted from Plant 2/3 and therefore could have contributed nontrivial environmental doses. This TBD evaluates the environmental impacts of these uranium decay products in the Plant 2/3 uranium emissions from 1953 through 1988. A wide range of ratios for ^{226}Ra and ^{230}Th to uranium existed in Plant 2/3 depending on the origins of the uranium ore. This evaluation assumed none of the decay products were removed in the milling process and that the decay products were in equilibrium with the activity of natural uranium in the ore. This assumption is favorable to claimants. Based on this assumption, Table 4-15 provides the ratios of ^{226}Ra and ^{230}Th activity to uranium mass. The uranium effluent in kilograms from Plant 2/3 was multiplied by these ratios to estimate the activity of ^{226}Ra and ^{230}Th in the Plant 2/3 effluent. The Table 4-15 information is reflected in Table 4-2.

Table 4-15. Ratio of ^{226}Ra and ^{230}Th to uranium.

Radioisotopes	Progeny/U ratio
Ra-226	3.47E-01 Bq Ra-226/kg U
Th-230	3.47E-01 Bq Th-230/kg U

a. See ORAUT (2016a) for derivation.

4.4.1.4.1 Radon

Radon-222 intake levels are expressed in units of WLM. For ^{222}Rn, 1 working level (WL) is equal to 100 pCi/L of ^{222}Rn in 100% equilibrium with its short-lived decay products. A month is defined as a period of 170 work hours. The derivation of the number of WLMs in a year was based on an intake period of 2,000 hours for the year and an assumed environmental radon progeny equilibrium factor of 70%. This ratio for ambient outside air is in accordance with widespread sampling throughout the United States from National Council on Radiation Protection and Measurements Report 78 (NCRP 1984).

Therefore,

$$\begin{aligned} {}^{222}Rn\ WLM\ in\ a\ year &= \frac{0.7 \times 2000}{100 \times 170} \times Concentration\left(\frac{\text{pCi}}{\text{L}}\right) \\ &= 8.24 \times 10^{-2} \times Concentration\left(\frac{\text{pCi}}{\text{L}}\right) \end{aligned} \quad (4\text{-}11)$$

The average concentration of radon emissions from the waste pits (EA-5) was typically an order of magnitude lower than those from the K-65 silos. Therefore, the radon exposure results for the EAs do not include the radon emissions from the waste pits.

4.4.1.4.2 Radon Exposure Records

Some claimant files might contain radon exposure records in units of WLMs. These records typically appear in "Personnel Exposure Records," separate from the "DOE Response" file. DRs should follow the guidance in DCAS-TIB-011, *Lung Dose Conversion Factor for Thoron WLM*, when evaluating these records (NIOSH 2013a). For dose reconstructions that require assignment of ^{222}Rn exposures, and when individual radon exposure data exist, both the environmental ^{222}Rn and the individual ^{222}Rn data need to be evaluated. However, DRs should assign only the highest dose determined from this comparison, not both. The radon intakes from the Personnel Exposure Records should be assumed to be a chronic exposure and should be assigned as a constant distribution in IREP for overestimates. For best-estimate or compensable claims, DRs must prorate the radon doses to actual periods of employment.

Based on information from Pinney et al. (2004), the total radon exposure is the summation of the radon exposure from the K-65 silos and Q-11 silos. Worker radon data were obtained from two files, NIOSH (ca. 2013b) and NIOSH (ca. 2013c). The total K-65 radon exposure is equivalent to the sum of the individual radon results from (NIOSH ca. 2013b) in the YRLY_K65 field to the results in the K65_TOTAL field of the second file (NIOSH ca. 2013c). The total Q-11 radon exposure is equivalent to the sum of the individual radon results from (NIOSH ca. 2013b) in the YRLY_Q11 field to the results in the Q11_TOTAL field of the second file (NIOSH ca. 2013b). The fields in NIOSH (ca. 2013b) are YR, YRLY_Q11, YRLY_K65, TOTAL_EXP, SURVEY_YRLY_Q11, SURVEY_YRLY_K65, SURVEY_TOTAL_EXP, and SSN. The fields in NIOSH (ca. 2013c) are SSN, K65_TOTAL, Q11_TOTAL, and TOTAL_EXP. The alphabetical listing of fields and their content is shown in Table 4-16, which provides information to DRs on what the fields represent.

4.4.1.5 Results

ORAUT (2016a) documents each calculation and lists estimated radionuclide intake rates in becquerels per year or WLM per year for unmonitored workers who spent a year in an EA between 1951 and 1988. These values are summarized and, for each radionuclide of interest, the intake rate

Table 4-16. Alphabetical list of fields (Pinney et al. 2004, p. 67).

Field	Content
SSN	Social Security Number
SURVEY_TOT_EXP	Yearly K-65 + Q-11 radon exposure (WLM) from survey data
SURVEY_YRLY_K65	Yearly K-65 radon exposure (WLM) from survey data
SURVEY_YRLY_Q11	Yearly Q-11 radon exposure (WLM) from survey data
TOTAL_EXP	Yearly K-65 + Q-11 radon exposure (WLM) from estimated shift
YR	Start date year
YRLY_K65	Yearly K-65 radon exposure (WLM) from estimated shift
YRLY_Q11	Yearly Q-11 radon exposure (WLM) from estimated shift

in the EA with the highest value is assigned for sitewide application. Table 4-2 presents these results in summary form.

4.4.2 Nonradon Radionuclide Air Concentrations and Intakes, 1989 to Present

Starting in 1989, the operational and effluent data needed to calculate environmental intake rates in each Exposure Area were no longer available. Instead air sample data at fence line and on-site locations published in the annual environmental reports were used as input to the intake rate calculations. Citations to the individual annual reports are organized by year in the reference section of this document. When full data sets were published, the maximum air sample result for each sample location was tabulated and the median of each of these maximum values was assumed as the concentration for that year. When full data sets were not available, the maximum published value was assumed. No ambient offsite background air concentration was subtracted. The annual airborne uranium concentrations from the onsite and perimeter AMSs were used with the model in ORAUT-PROC-0031, *Site Profile and Technical Basis Document Development* (ORAUT 2012), to determine the onsite air concentrations and intakes for environmental dose to workers for 1989 to present. Recycled uranium contaminant levels shown in Table 4-14 were also applied. The date range for this section suggests that the period of time of analyses of data occurred through present. The empirical uranium air sampling data from the environmental reports only extend through 2009 because DOE ended the collection programs because the data from previous years indicated levels had reached *de minimis* levels. For years after 2008 (for uranium) and 2009 (for radon) when no air monitoring occurred, the data from the last year that monitoring occurred were used to extend the intakes to present.

4.4.2.1 Annual Air Emissions for Radionuclides Other than Radon

The uranium air emissions in the FMPC environmental reports contain measured air concentrations from onsite and perimeter AMSs. The air concentrations are attributed to uranium losses from:

- Monitored stacks,
- Unmonitored stacks,
- Scrubbers,
- Building vents, and
- Fugitive emissions from the waste pits.

4.4.2.1.1 Monitored Stacks

Stack discharges were monitored by taking continuous isokinetic particulate samples, which were routinely analyzed for uranium and other radionuclides.

4.4.2.1.2 Unmonitored Stacks

Facilities that operated on a limited basis were unmonitored. This included losses from the cooling towers. Estimates were derived by engineering calculations, periodic stack emission measurements, or a combination of both. In general, a detailed analysis was made of the processes, the characteristics of emitted materials, and the operating time for each system. This information was used to derive emission factors that, when multiplied by amounts of material processed over a period of time, provided estimates for emissions of unmonitored processes.

4.4.2.1.3 Scrubbers

A scrubber is a pollution control device that uses a liquid to remove contaminants from air. Losses were based on an emission factor (per hour) times the actual number of hours these sources operated. The emission factors were derived from stack emission tests that were conducted in accordance with Environmental Protection Agency methods.

4.4.2.1.4 Building Vents

Building ventilation emissions were calculated based on data from continuous air monitors in each plant near processes that were thought to produce radionuclide emissions. The yearly emissions for each plant were estimated from the average annual concentrations from the plant continuous air monitors, the estimated hours each fan operated, and a 0.1 dilution factor. Because the emissions were measured only in the areas of highest suspected concentrations, the actual average uranium concentration in the air in the plants would have been significantly lower than the calculated average of the continuous air monitor results.

4.4.2.1.5 Fugitive Emissions

Estimates of fugitive uranium were based on the average number of samples that were processed per year in each hood, the probable loss per sample, and the estimated uranium concentration of the sample.

Waste pits 1 through 6, the Clearwell, and the Burn Pit are described in ORAUT-TBKS-0017-2. These structures were remediated over the period 1999 through 2005. Maximum fence line ^{230}Th air concentrations observed during each of those years (Fluor-Fernald, 2000, 2001, 2002, 2003, 2004, 2005, 2006) were used to calculate the annual intake rate assuming full-time work year occupancy. Results are presented in Table 4-2.

Table 4-17 presents the median of maximum values of all uranium air samples that were collected after 1988. It also show the maximum fence line concentrations of ^{230}Th measured during waste pit remediation activities.

4.4.2.2 Results

Intake rates for uranium and intakes of associated trace contaminants and ^{230}Th are reported in Table 4-2. Calculation of the intake rates is presented in ORAUT (2016a).

4.4.3 Radon Air Concentrations and Intake Estimates, 1989 to present

Radon was monitored as a separate component of the air pathway at FMPC. FMPC used alpha-track radon detectors in weatherproof housings to determine radon concentrations in the environment (Environmental Report Dugan et al. 1990, p. 87). From 1989 to 1998, there were alpha-track radon detectors at each of the locations. Alpha-track monitoring ended in 1998. An alpha-track radon

Table 4-17. Medians of the maximum values of all uranium air concentration (Bq/m^3) samples by year (ORAUT 2016a) and maximum fence line values of ^{230}Th (Bq/m^3) during waste pit remediation activities.

Year	Uranium air concentration	Th-230 Emissions from Waste Pits
1989	5.11E-05	NA
1990	1.81E-05	NA
1991	2.33E-05	NA
1992	3.03E-05	NA
1993	3.39E-05	NA
1994	1.18E-05	NA
1995	8.51E-06	NA
1996	1.92E-05	NA
1997	3.79E-05	NA
1998	2.81E-05	NA
1999	4.07E-05	9.25E-07
2000	3.66E-05	7.03E-06
2001	3.66E-05	2.74E-05
2002	1.02E-05	2.15E-05
2003	8.51E-05	7.77E-06
2004	4.81E-04	1.55E-05
2005	1.30E-05	2.89E-06
2006	2.42E-06	NA
2007	9.62E-07	NA
2008	6.66E-07	NA
2009–present	5.18E-07[a]	NA

a. Results for 2010 to present based on 2009 air sampling data when particulate uranium site sampling ended.

detector is a passive, long-term device for integrating radon concentrations in air by permanently recording the tracks of alpha particles from radon and its decay products. The detectors were changed quarterly and sent to the supplier for analysis. At the end of the year, the averages of the four quarterly concentrations were computed and documented in the site's annual environmental reports. Beginning in 1999, the radon monitoring continued through the radon continuous monitoring program using alpha scintillation monitoring cells. These cells allowed the air to continuously diffuse into the detector cell through a foam barrier, without the aid of a pump, and radon and its progeny would alpha-decay in the cell and be measured by the scintillator. The system was used until radon monitoring at the site was discontinued in 2008. These data were analyzed using the method in ORAUT-PROC-0031 (ORAUT 2012) to develop intakes to workers, which are presented in Table 4-2.

The monitoring locations changed over time as remedial actions occurred. Figures 4-7 and 4-8 show monitoring locations in the 1994 timeframe for 11 locations off site, 21 locations along the fence line, four locations on site at various distances from the silos, and 16 locations immediately adjacent to the K-65 silos.

Table 4-18 presents the median of maximum values of all on-site and fence line radon air samples that were collected after 1988 through 2008. The data in the table from 2009 to present are an extension of the data from 2008. These data were used to estimate the radon and radon progeny intake rate, which are reported in Table 4-2.

4.5 ONSITE EXTERNAL DOSES FROM AMBIENT RADIATION

External environmental radiation dose at FMPC has primarily been the result of photon radiation (i.e., gamma and X-rays from radionuclides that were stored on the site). For unmonitored personnel who worked in the administration areas, the largest source of direct radiation was the waste in the

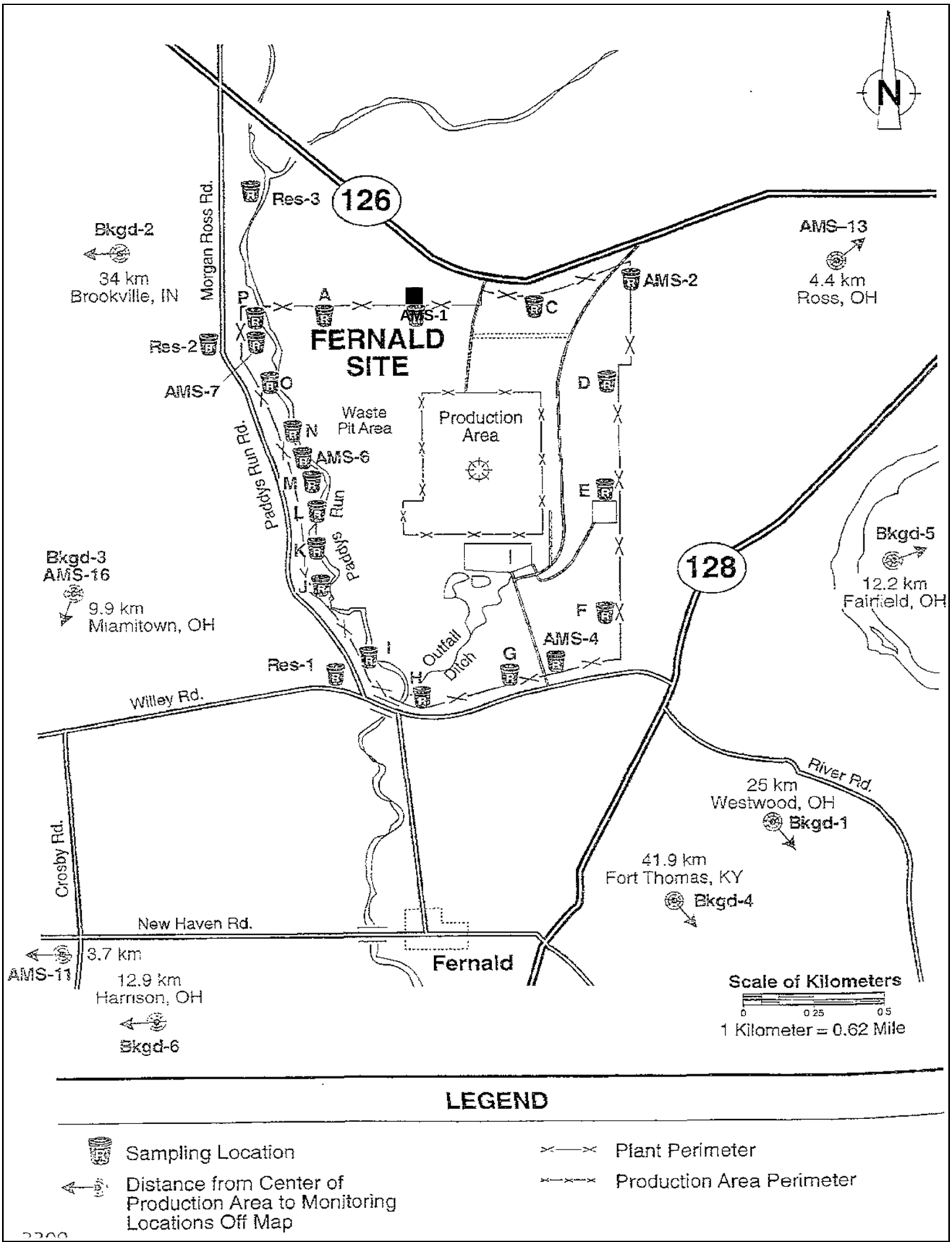

Figure 4-7. Offsite and fence line radon monitoring locations (Environmental Report FERMCO 1995, p. 171).

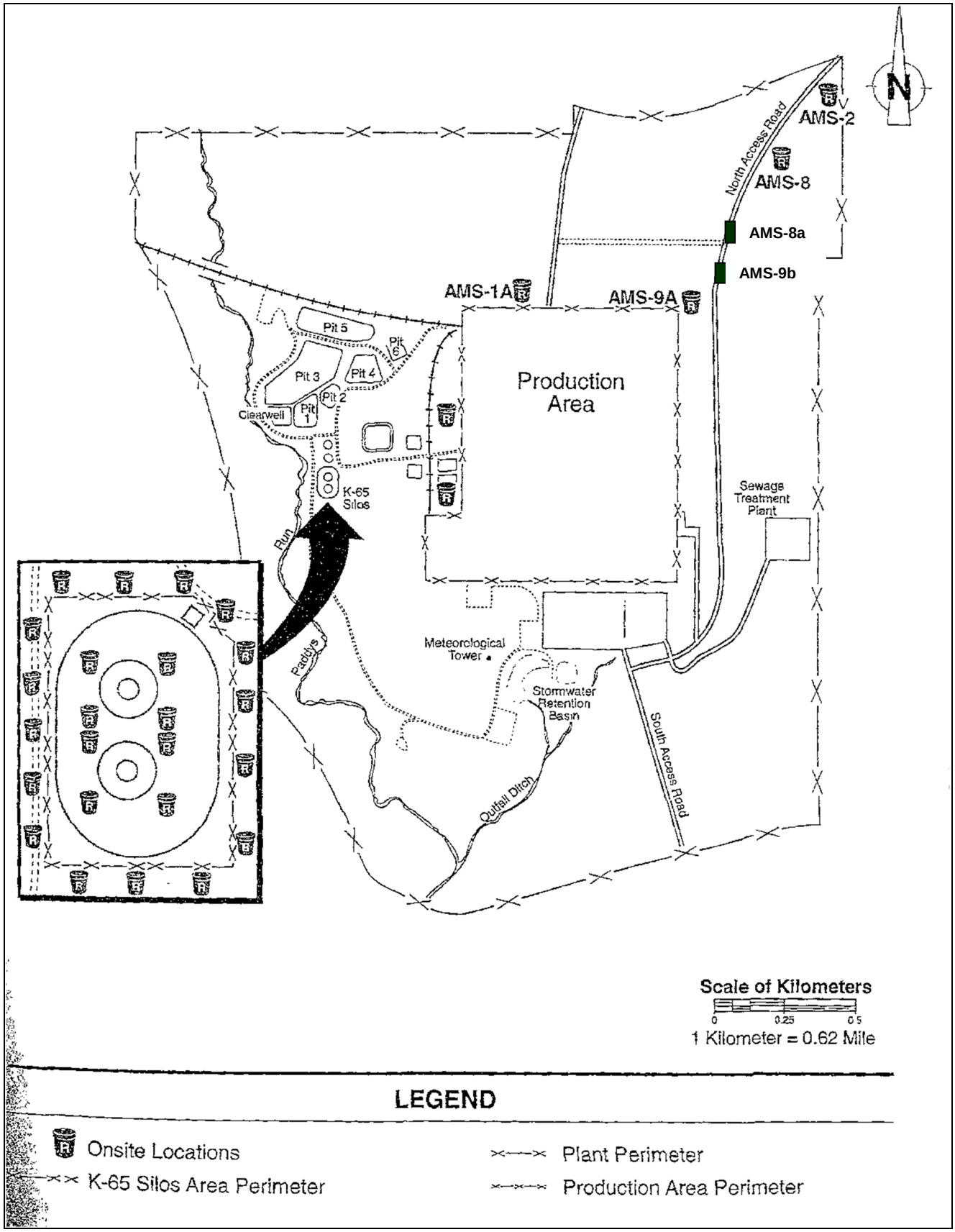

Figure 4-8. Onsite radon monitoring locations (Environmental Report FERMCO 1995, p. 172).

Table 4-18. Medians of the maximum values of all on-site and fence line radon air concentrations (Bq/m^3) samples by year (ORAUT 2016a).

Year	Radon air concentration
1989	2.59E+01
1990	2.59E+01
1991	2.96E+01
1992	1.85E+01
1993	3.89E+01
1994	4.07E+01
1995	3.70E+01
1996	5.37E+01
1997	5.92E+01
1998	3.33E+01
1999	3.70E+01
2000	2.96E+01
2001	2.96E+01
2002	3.70E+01
2003	2.96E+01
2004	2.96E+01
2005	2.78E+01
2006	3.33E+01
2007	2.59E+01
2008–present	3.33E+01[a]

a. Results for 2009 to present based on 2008 air sampling data when radon site sampling ended.

K-65 silos, thorium residues in various warehouses throughout the site, and the radionuclides in the Production Plants and other onsite facilities. To a lesser extent, the radioactive Waste Pit Area was another direct radiation source for unmonitored personnel who worked in that area.

4.5.1 K-65 Silos and Production Plants External Dose Contribution

There were three significant external radiation sources at FMPC, but by far the largest was the waste in the K-65 silos. The ambient dose rates started to increase to measurable levels in 1955 with the relocation of wastes from Niagara Falls, New York, to the silos. This elevation in ambient dose was due to the radium content of these wastes. Direct radiation occurred not only from radium but also from the radon progeny that were in secular equilibrium within the headspace of each of the silos. This condition prevailed until the silos became such a significant source of radiation that DOE found it necessary to construct reinforcement berms and coat the exterior surfaces in an attempt to strengthen the structures and reduce the radon emissions. While these actions provided some benefit, the radon dose contribution continued to increase as gaseous radionuclides emanated and accumulated in the void space under the domes and became an increasing source of external exposure both on and off the site. The source strength increased to the point that radiation levels required additional shielding to mitigate the direct radiation dose to personnel. In 1991, a bentonite layer was added to the surface of the stored material in an attempt to contain the radioactive gases (Environmental Report Fluor Fernald 2001, p. 13). This was effective, as indicated by an immediate reduction in direct radiation dose rates due to the shielding by the solid materials in the silos. An immediate dose rate reduction of approximately 1 order of magnitude occurred as a result of the bentonite. However, the bentonite provided only a temporary reduction. Dose rates slowly increased again and monitoring in 2002 showed an increase of about a factor of 4 in external radiation.

The only data found from before 1975 (Klein 1963) consisted of contact radiation readings using a Juno survey meter for measurements of exposure rates in the vicinity of the North and South Tanks

(i.e., the K-65 silos). The dose rates ranged from 10 to 20 mR/hr and 10 to 30 mR/hr, respectively. Later data (Boback 1978) provided readings from thermoluminescent dosimeters (TLDs) inside the South Tank that ranged from 590 mR/hr at the surface of the residue to 650 mR/hr at 10 ft and 470 mR/hr at 13 ft above the surface of the residue. These data are indicative of the magnitude of the K-65 silos as sources of ambient radiation, and they were used to project the radiation environment of the site in its early operating years (1952 to 1975). ORAUT (2016a) documents the details of this calculation.

Starting in 1976, gamma radiation at the site boundary (fence line) sampling stations was measured with TLDs that were changed and processed every 3 months (Environmental Report Boback et al. 1977, p. 62). Table 4-19 is a summary of the maximum measured dose rates at the monitoring stations for 1976 through 1995. Figure 4-9 shows the locations of the fence line monitoring stations.

In 1996, FMPC expanded its direct radiation monitoring system to continuously measure 32 locations, including several onsite monitoring locations. Table 4-20 provides summary level information about direct radiation measurements for 1996 through 2005. Figure 4-10 shows the locations of the TLD stations.

Tables 4-19 and 4-20 list the external radiation information for 1976 to 2005 with source information. The collection of these TLD measurement data became possible with advances in the development of the TLD as a useful dose measurement device. These technology developments made it possible to measure radiation doses over long periods, in the exposure environment, and without power. TLDs became a popular and accurate tool for measuring doses in the environment over extended periods.

4.5.2 External Dose Contribution from Waste Pits

Figure 4-11 shows the beta/gamma dose rate contours in the Waste Pit Area, which is in EA-5 (DOE 1994a). Table 4-21 lists operational periods and the direct dose rate for each pit (DOE 1994a). Figure 4-12 shows the radiation profile of the Waste Pit Area, which accounts for the development of each pit (i.e., increasing dose rates as the pits were filled with wastes).

The beta and gamma radiation at the Waste Pit Area was emitted from radionuclides that were buried in the ground. Gamma radiation is emitted at ground level into the atmosphere, which then scatters the radiation to receptor locations some distance from the area; this is the skyshine phenomenon. Radiation from skyshine at the receptor location is typically about 1% of the dose rate at the source.

4.5.3 Onsite Ambient Dose Rate Estimates, 1976 to 2005

From 1976 to 2005, when TLD stations were deployed to monitor the environmental radiation levels, the dose rate at each EA was estimated by summing the dose rates from the major direct dose contributors (i.e., K-65 silos, storage and production facilities, and the waste pits) using the measured dose rate data in Tables 4-19 and 4-20 and the estimated dose rate profile in Figure 4-12.

Table 4-19. Radiation dose rates at site boundary (mR/hr or mrem/hr).[a,b]

	Maximum measured dose at TLD stations										
Year	**BS1/1A**	**BS2**	**BS3**	**BS4**	**BS5**	**BS6**	**BS7**	**BS8**	**BS9**	**Onsite**	**Background**
1976	0.010	0.012	0.012	0.011	0.012	0.015	NA	NA	NA	NA	0.010
1977	0.011	0.010	0.011	0.011	0.011	0.012	NA	NA	NA	NA	0.009
1978	0.010	0.011	0.011	0.010	0.011	0.013	NA	NA	NA	NA	0.009
1979	0.012	0.012	0.010	0.010	0.011	0.018	NA	NA	NA	NA	0.009
1980	0.012	0.014	0.014	0.013	0.012	0.023	NA	NA	NA	NA	0.011
1981	0.010	0.012	0.012	0.012	0.012	0.021	0.013	NA	NA	NA	0.009
1982	0.011	0.014	0.013	0.013	0.014	0.019	0.013	NA	NA	NA	0.010
1983	0.012	0.013	0.013	0.013	0.013	0.020	0.014	NA	NA	NA	0.010
1984	0.010	0.0113	0.0114	0.011	0.0115	0.0171	0.0123	NA	NA	NA	0.010
1985	0.0123	0.0129	0.0125	0.0122	0.0127	0.0191	0.0138	NA	NA	NA	0.0118
1986	0.0127	0.011	0.0111	0.0111	0.0116	0.015	0.0104	NA	NA	NA	0.009
1987	0.0096	0.0107	0.0103	0.0104	0.010	0.0157	0.0108	NA	NA	NA	0.0106
1988	0.0144	0.0164	0.0169	0.0155	0.015	0.0235	0.0158	0.0148	0.0186	NA	0.0151
1989	0.0135	0.0131	0.0161	0.0126	0.0127	0.0197	0.0132	0.0116	0.0175	NA	0.0128
1990	0.0079	0.0079	0.0076	0.0078	0.0073	0.013	0.0075	0.0074	0.010	NA	0.0075
1991	0.0081	0.0085	0.0077	0.0082	0.0077	0.013	0.008	0.0078	0.010	0.258	0.0069
1992	0.0080	0.0094	0.009	0.009	0.0091	0.0087	0.0083	0.0084	0.0115	0.0205	0.0082
1993	0.0089	0.0084	0.0077	0.0078	0.0077	0.0077	0.0075	0.0079	0.0107	0.0239	0.0068
1994	0.0128	0.0082	0.0071	0.0074	0.0081	0.0079	0.0072	0.0079	0.0107	0.0239	0.0068
1995	0.0157	0.0085	0.0075	0.0074	0.0079	0.0085	0.0077	0.0077	0.0108	0.0420	0.0074

a. Sources: 1976 to 1995 environmental reports (Boback et al. 1977, 1978; Boback and Ross 1980, 1981; Fleming et al. 1982 to 1984; Facemire et al. 1985; Aas et al. 1986, 1987; Gels and Lojeck 1988; Gels et al. 1989; Dugan et al. 1990; Byrne et al. 1991; WEMCO 1992; FERMCO 1993 to 1996).

b. NA = not available.

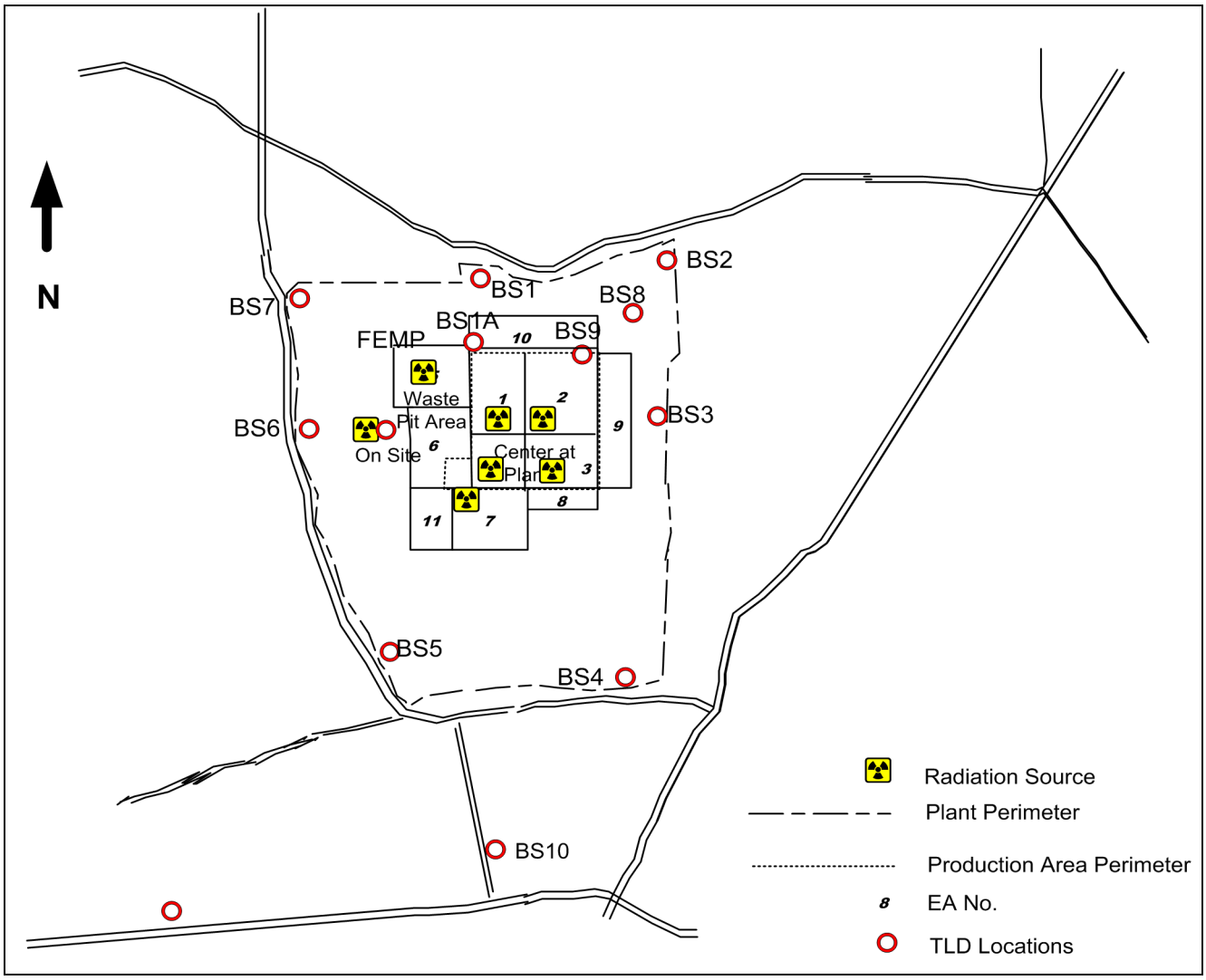

Figure 4-9. Fence line TLD locations, 1976 to 1995 (Environmental Reports FERMCO 1993 to 1996).

Table 4-20. TLD measurement summary (mrem/hr).[a,b]

Year	Fence line min.	Fence line max.	Onsite min. (Health and Safety Bldg.)	Onsite max. (K-65 silos fence line)	Offsite min.	Offsite max.	Background average
1996	0.008	0.0104	0.0069	0.0789	0.0069	0.0084	0.0088
1997	0.0082	0.0103	0.0070	0.101	0.0068	0.0085	0.0087
1998	0.0072	0.0096	0.0063	0.108	NA	NA	0.0079
1999	0.0072	0.0093	0.0063	0.103	NA	NA	0.0079
2000	0.0074	0.0097	0.0066	0.124	NA	NA	0.0079
2001	0.0079	0.0103	0.0066	0.137	NA	NA	0.0083
2002	0.0081	0.0111	0.0064	0.139	NA	NA	0.0087
2003	0.0073	0.0087	0.0064	0.0508	NA	NA	0.0075
2004	0.0073	0.0093	0.0059	0.0247	NA	NA	0.0079
2005	0.0104	0.0126	0.0097	0.0892	NA	NA	0.0112

a. 1996 to 2005 environmental reports (FDF 1997 to 1999; Fluor Fernald 2000 to 2006).
b. NA = not available.

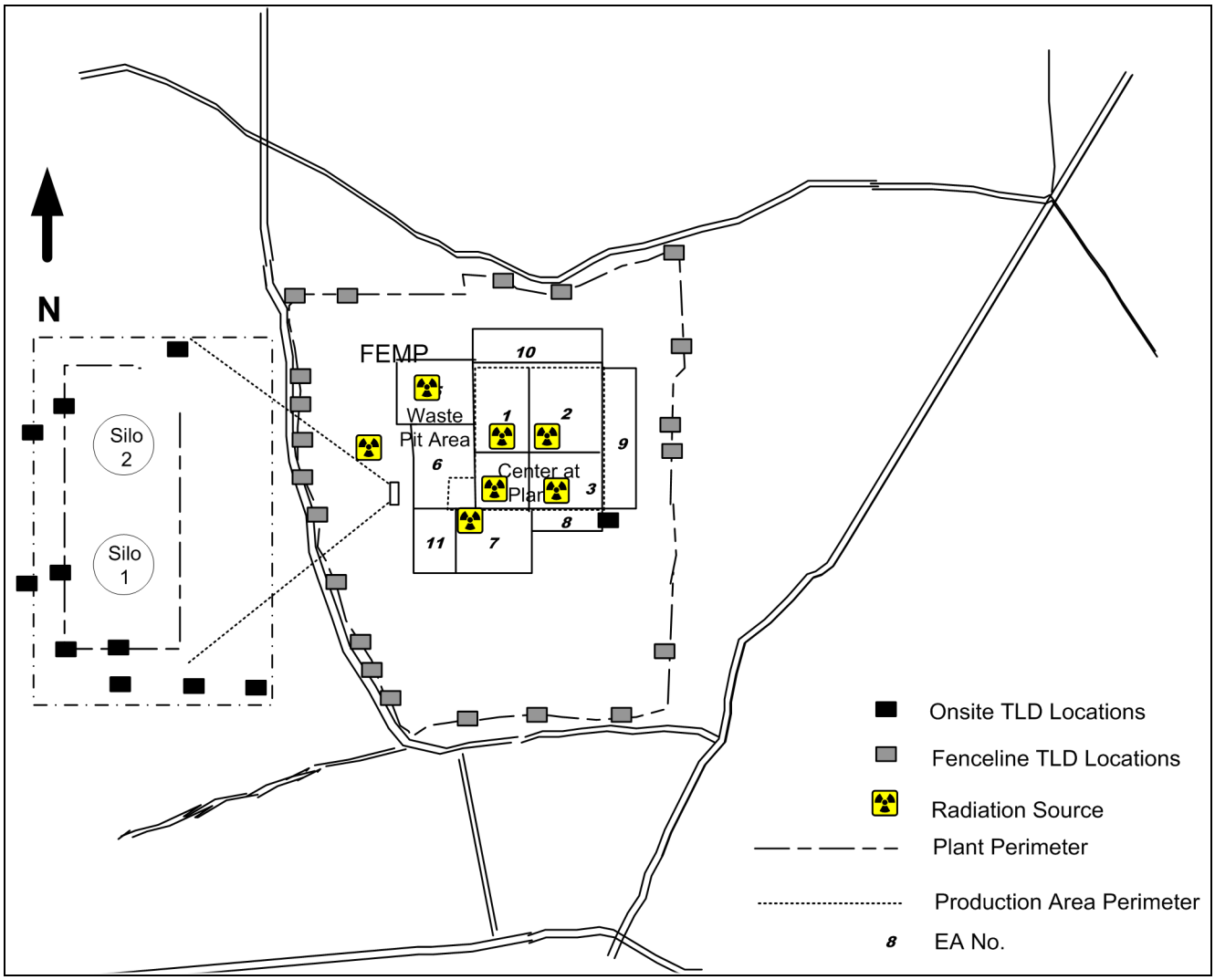

Figure 4-10. TLD monitoring locations, 1996 to 2005 (Environmental Report Fluor Fernald 2004).

The dose rates can be represented by Equation 4-12:

$$D_i = \left(BSs - D_{bkg}\right) \times \left(\frac{d_{BSs}}{d_{EAsi}}\right)^2 + Average\left[\left(BSj - D_{bkg}\right) \times \left(\frac{d_{BSj}}{d_{EAji}}\right)^2\right] + 0.01\left[D_{pit} \times \left(\frac{d_k}{d_{EAki}}\right)^2\right] + D_{bkg} \quad (4\text{-}12)$$

where

D_i = estimated dose rate at EA*i*
BSs = measured dose rate at the fence line TLD station closest to the K-65 silos (BS6) (Starting in 1996, the K-65 silo fence line values in Table 4-19 were used in the equation and as EA-6 ambient dose rate.)
D_{bkg} = background dose rate
d_{BSs} = distance from TLD stations to the K-65 silos
d_{EAsi} = distance from K-65 silos to EA*i*
BSj = measured dose rate at fence line TLD stations, where *j* represents the combination of BS1, 2, 3, 4, 5, 7, and 9; starting in 1996, *BSj* values were assigned the average background values because the onsite TLD station BS32 (Health and Safety Building) readings in Table 4-20 were below background

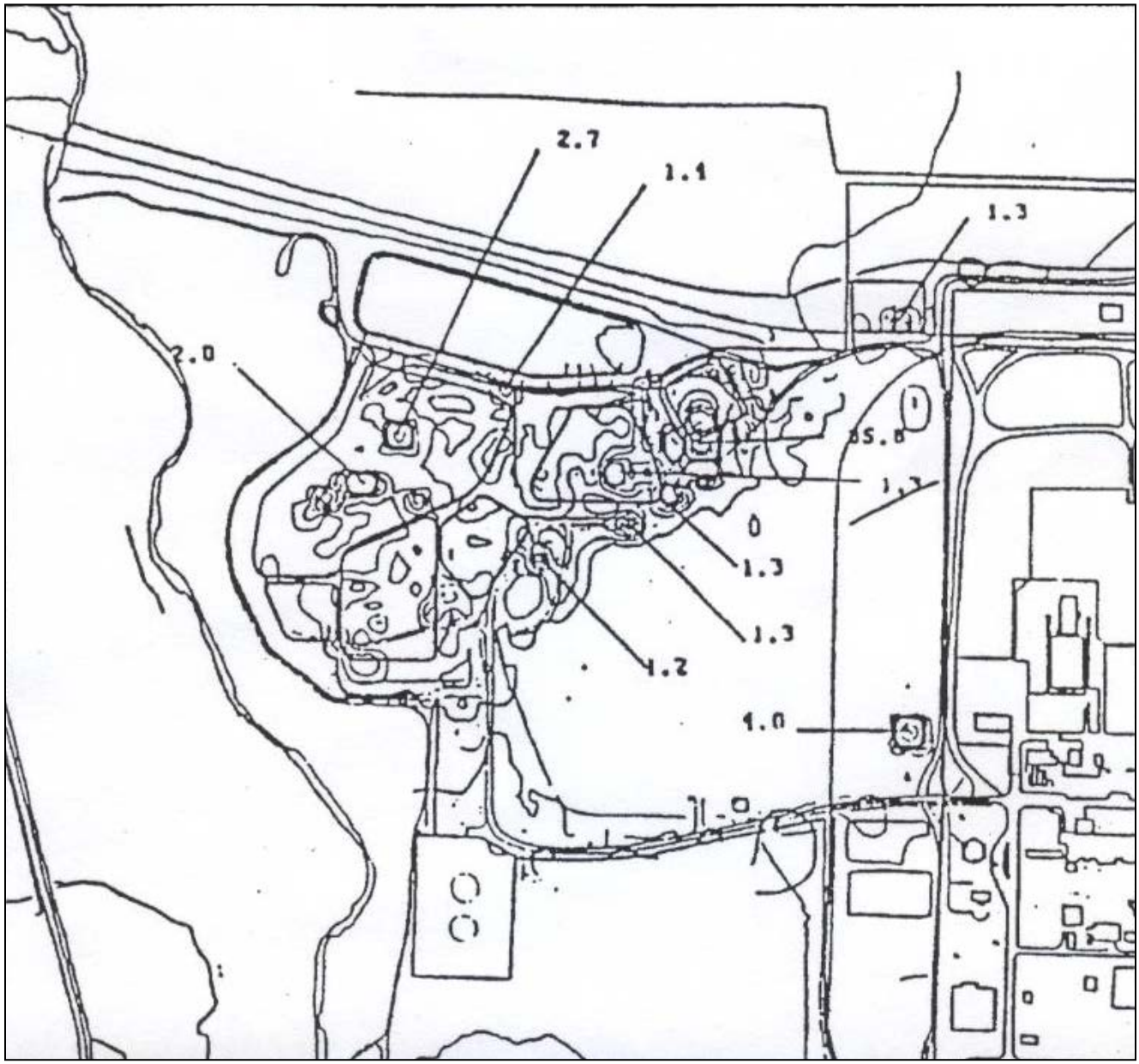

Figure 4-11. Beta/gamma dose rate (mrad/hr) contours in Waste Pit Area (DOE 1994a, p. 1213).

d_{BSj} = distance from TLD stations to production plants
d_{EAji} = distance from production plants to EA*i*
D_{pit} = average dose rate at the Waste Pit Area (the waste pit component of the equation was applied to all operating years from 1952 to 2002)
d_k = distance from center of Waste Pit Area to edge of Waste Pit Area
d_{EAki} = distance from Waste Pit Area to EA*i*

ORAUT (2016a) documents Equation 4-12 results. These results were compared to the direct dose rate contour maps (Figures 4-13 and 4-14) from the EG&G aerial surveys in 1976 and 1985 (see Section 4.5.5). The comparisons have resulted in closely matched dose rates in each EA, as shown in Table 4-22.

Table 4-21. Waste pit dose rate data (DOE 1994a).

Waste Pit	Operation period	Average ambient dose rate (mrad/hr)
1	1952–1959	1.0
2	1957–1964	1.2
3	1958–1977	2.0
4	1960–1986	16.0[a]
5	1968–1987	1.0
6	1979–1985	1.3
Clearwell	1960s–1970s	1.0
Burn pit	1957–1984	1.3

a. Waste Pit 4 dose rates are much higher than others due to having the highest uranium waste inventory. Calculations using National Council on Radiation Protection and Measurements methodology show that releases of uranium by far are the most important contributors to the potential doses from releases to the atmosphere at the FMPC (DOE 1994a).

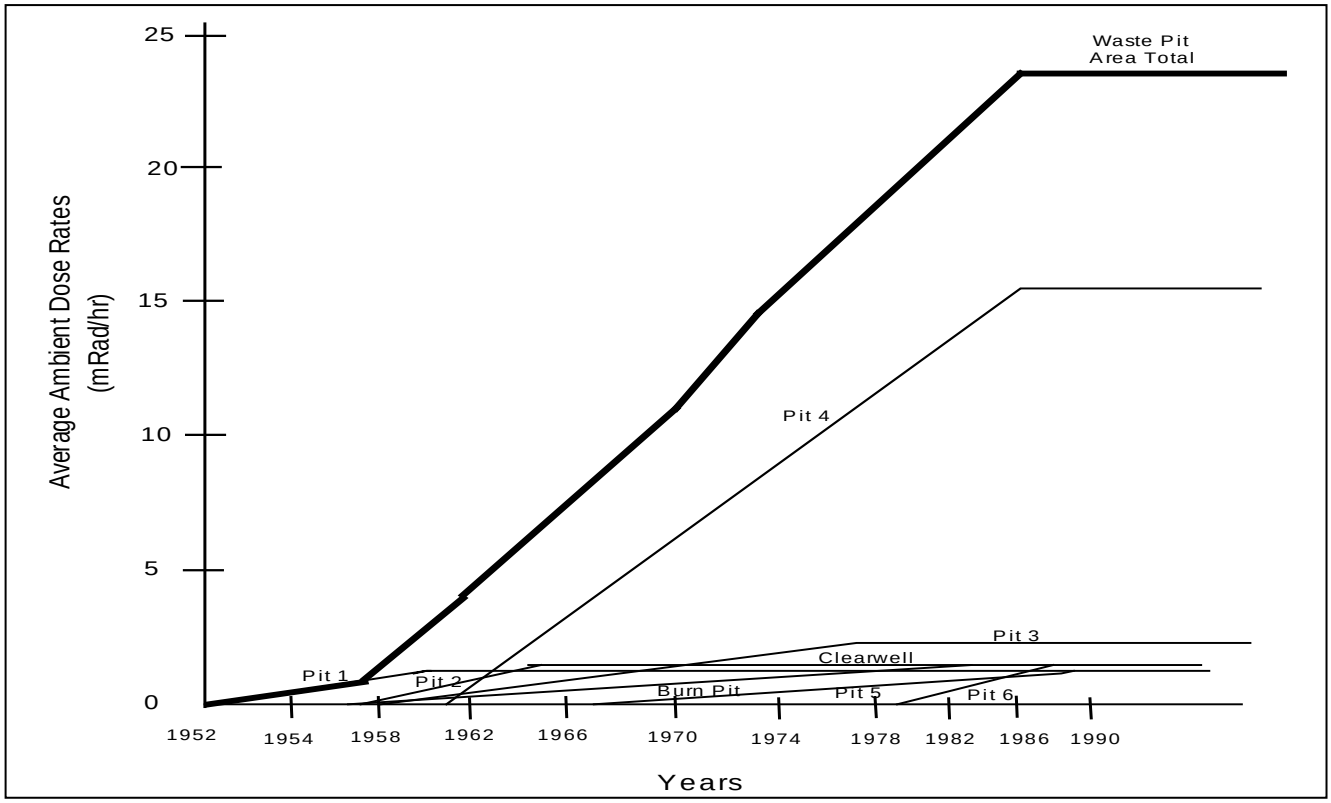

Figure 4-12. Waste pit dose rate profiles.

4.5.4 Onsite Ambient Dose Rate Estimates, 1952 to 1975

Because no usable direct radiation measurements were found in the available reference materials for the period before 1976, it was necessary to estimate the ambient dose rates on the site for those years based on extrapolation from known data or correlation to historical events. The following sections present the methodologies that were used to extrapolate onsite dose rates for those years that predate the implementation of mature radiation monitoring programs at FMPC.

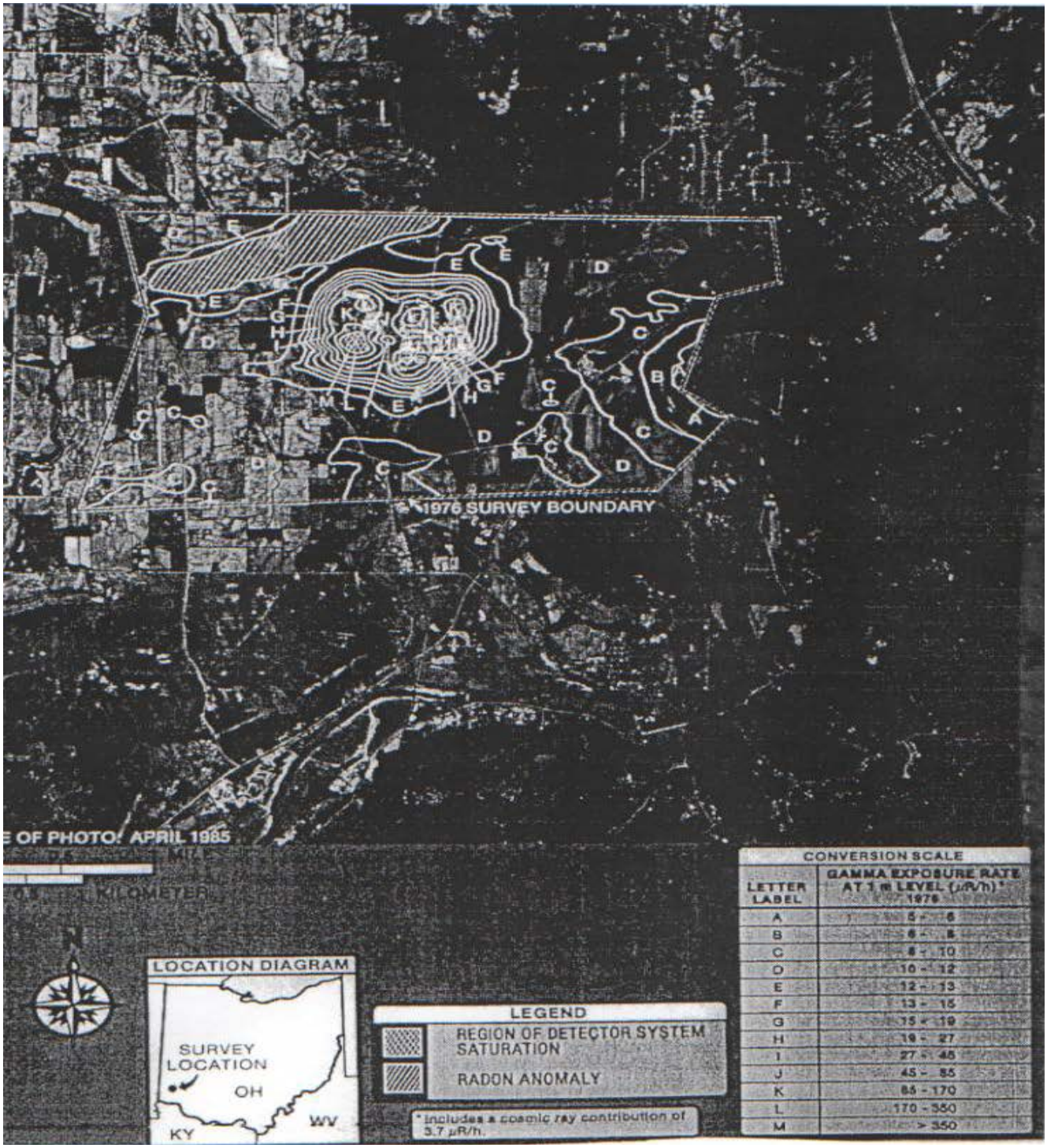

Figure 4-13. 1976 exposure rate (µR/hr) contour map of FMPC survey area (Shipman 1985, p. 18).

4.5.4.1 Dose Rate Due to K-65 Silos, Before 1976

Figure 4-15 is the histogram of the average direct dose rates at the K-65 silo fence line that was developed by a combination of interpreting the historic description of the K-65 silo radiation levels and the application of measured dose rate values.

As noted in Section 4.5.1, the ambient dose rates started to increase to measurable levels in 1955 with the relocation of wastes from Niagara Falls, New York, to the silos. Before 1975, the dose rates

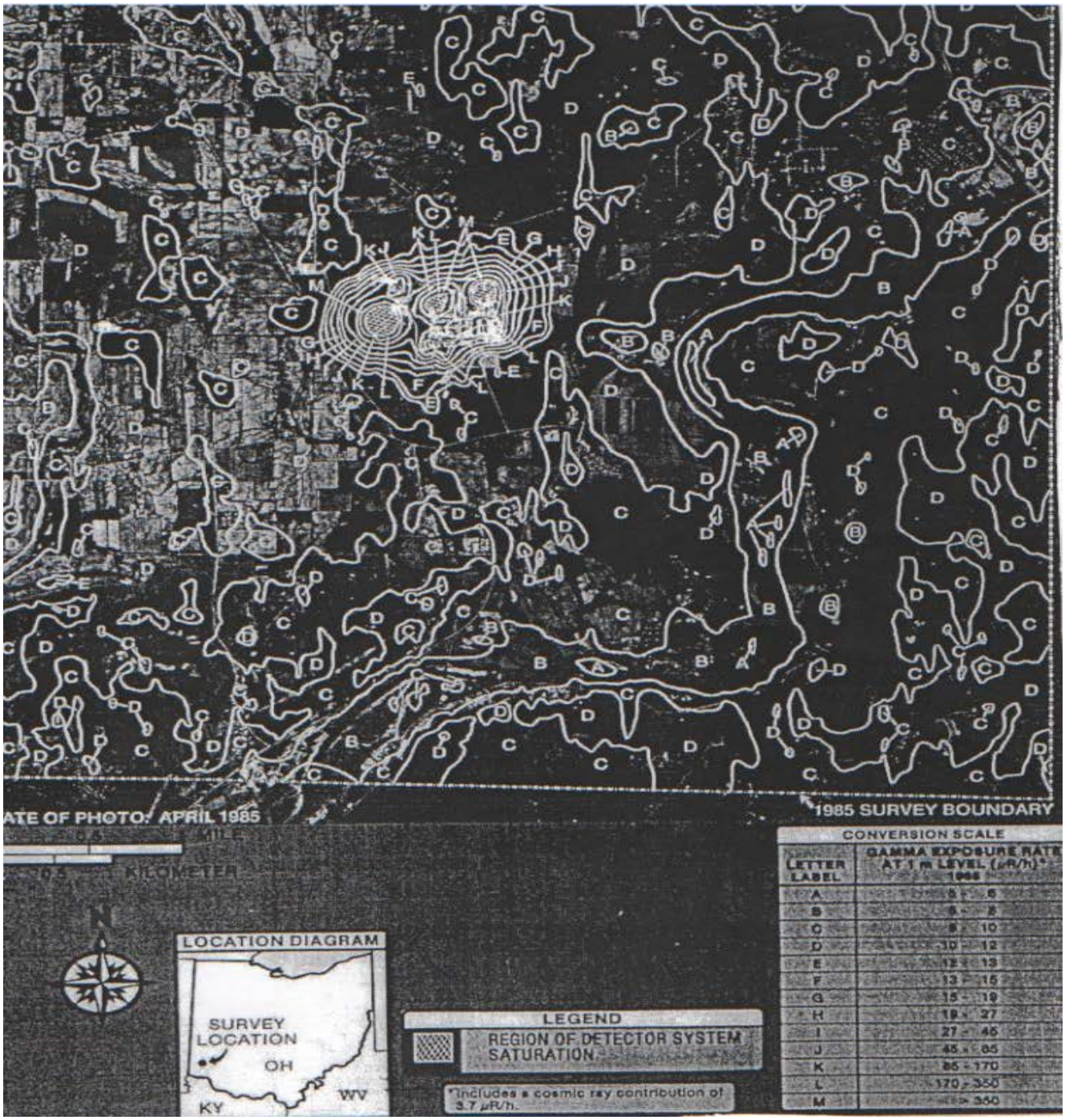

Figure 4-14. 1985 exposure rate (µR/hr) contour map of FMPC survey area (Shipman 1985, p. 15).

in the vicinity of the North and South Tanks (i.e., the K-65 silos) that were measured by a Juno survey meter ranged from 10 to 30 mR/hr, respectively (Klein 1963). The source strength increased to the point that radiation levels require additional shielding to mitigate the direct radiation dose to personnel.

In 1991, a bentonite layer was added (Environmental Report Fluor Fernald 2001, p. 13). An immediate dose rate reduction of approximately 1 order of magnitude occurred. Dose rates slowly increased again, and in 2002 monitoring showed an increase of about a factor of 4 in external radiation. Figure 4-15 shows that the dose rates progressively increased from 1952 to a “measurable” level in 1955. This level, as measured by a Juno survey meter in 1963, was interpreted to be an

Table 4-22. Comparison of estimated dose rates in EAs and aerial survey results (mrem/hr).

Exposure areas	1976 Eq. 4-12	1976 Aerial survey[a]	1977 Eq. 4-12	1985 Eq. 4-12	1985 Aerial survey[a]	1986 Eq. 4-12
EA-1	0.151	0.045–0.085 (J)	0.176	0.147	0.170–0.350 (L)	0.258
EA-2	0.107	0.045–0.085 (J)	0.124	0.104	0.170–0.350 (L)	0.181
EA-3	0.107	0.045–0.085 (J)	0.124	0.103	0.170–0.350 (L)	0.180
EA-4	0.0150	0.045–0.085 (J)	0.175	0.145	0.170–0.350 (L)	0.256
EA-5	0.0367	0.045–0.085 (J)	0.0342	0.047	0.045–0.65 (J)	0.046
EA-6	0.455	>0.350 (M)	0.279	0.659	>0.350 (M)	0.546
EA-7	0.0153	0.013–0.015 (F)	0.0145	0.0175	0.019–0.027 (H)	0.0176
EA-8	0.0194	0.013–0.015 (F)	0.0198	0.0211	0.027–0.045 (I)	0.0253
EA-9	0.0160	0.012–0.013 (E)	0.0159	0.0177	0.019–0.027 (H)	0.0194
EA-10	0.0151	0.012–0.013 (E)	0.0148	0.0169	0.019–0.027 (H)	0.0177
EA-11	0.0141	0.012–0.013 (E)	0.0128	0.0167	0.015–0.019 (G)	0.0152

a. See Figures 4-13 and 4-14.

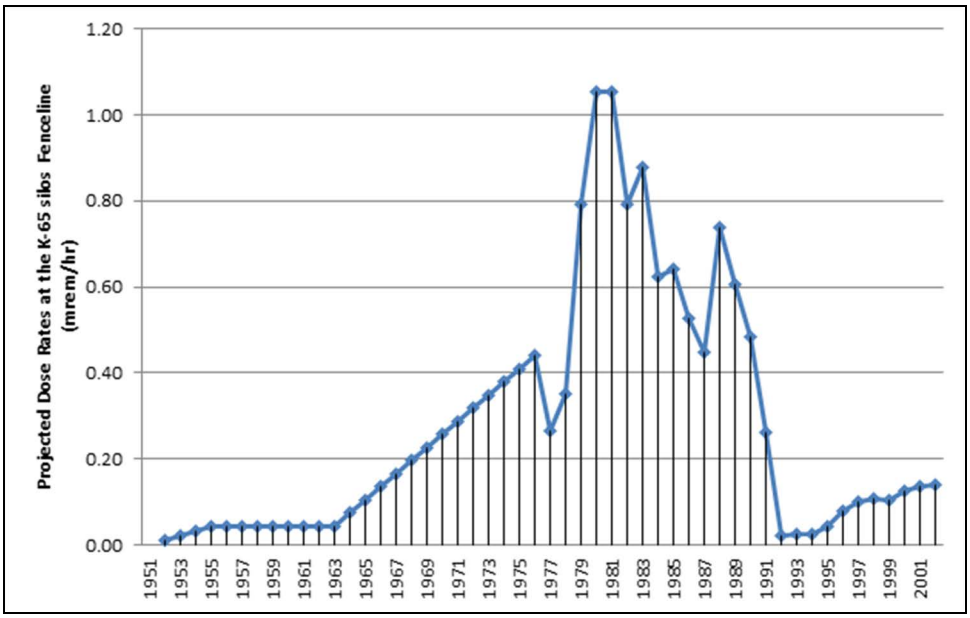

Figure 4-15. Projected K-65 silos fence line dose rates histogram.

exposure rate of 30 mR/hr at 3 ft (1 m) from each (North and South) silo, a total of 60 mR/hr at 3 ft from the tanks. The K-65 fence line dose rate was calculated to be 0.0443 mrem/hr based on this measurement. The histogram shows that this radiation level was assumed to remain steady from 1955 to 1963. Starting in 1964, the K-65 fence line dose rates once again incrementally increased each year to the 1976 level of 0.44 mrem/hr, which is the first year with TLD measurements. From 1976 on, the histogram reflects that the radiation level increased until the early 1980s. Since then the dose rates built up again. Figure 4-15 forms the basis for the estimation of the radiation levels in each EA due to the K-65 silos for 1952 to 1976. For 1977 to the present, environmental TLD station measurements have provided the data for estimation of the EA dose rate levels. The resulting dose rates and doses for dose reconstruction are listed in ORAUT (2016a).

4.5.4.2 Dose Rate Due to Production and Storage Facilities, Before 1976

The onsite direct radiation levels due to radioactive materials that were processed, handled, and stored at site facilities before 1976 can be extrapolated by the site production rate and the capacity of the processing systems. This extrapolation is further supported by the fact that large amounts of thorium were not present in storage until 1976, when measurements from the environmental TLD

stations (which were used to estimate the EA dose rate levels) were made available in the annual site environmental reports.

As production rates went up, the amount of radioactive materials in process equipment, in-process storage, material staging areas, and product storage also increased and thus caused the ambient radiation to increase. However, the radioactive materials increase was limited to the capacity of the site systems. When the systems were at their maximum capacities, the onsite ambient radiation was at its maximum level. Assuming that the site required two shifts that operated at 100% systems capacity to meet the peak production rate of 10,000 MTU/yr (Mead, Savage, and Fugate ca. 1985, p. 19), the sources of radiation were limited to full systems capacity of 5,000 MTU/yr. The peak production rate occurred from about 1954, when the production plants became operational, to 1964. In 1964, the production rates began to decline to a low in 1975 of approximately 1,230 MTU, which was about 25% of capacity. The production rate remained at 25% until 1981 when direction to restore the site facilities to accommodate projected product requirements close to the original production capacity was received. Thereafter, until production of uranium ended in 1988, the production rate increased to 3,700 MTU (3 times the 1979 level), which is about 74% of full capacity. The onsite ambient direct radiation dose rates for the years before 1976 were estimated by the ratio of the system's capacity to the corresponding measured dose rate in the 1980s. Table 4-23 lists the resulting onsite ambient dose rates. Figure 4-16 overlays the projected dose rate on the FMPC metal production rate profile.

The dose rate values in Table 4-23 represent the ambient dose rate in EA-4 (center of the FMPC production area) and were input into the storage and production component of Equation 4-12 to estimate the dose rate distribution for each EA. The calculations for dose reconstruction are provided in ORAUT (2016a).

4.5.5 Calculated Dose Rates and Recorded Measurements Comparisons

In 1985, EG&G Energy Measurements conducted an aerial survey of FMPC and surrounding areas for radiation due to gamma emissions (Shipman 1985, p. 7). The survey resulted in dose rate contour maps of the site for 1976 and 1985 (Figures 4-13 and 4-14, respectively). The aerial survey data were compared with estimated dose rates for the EAs using site TLD measurements. The aerial survey data were particularly important because the 1976 data represent a low annual production rate of approximately 1,230 MTU, while those for 1985 represent a modest annual production rate of about 3,700 MTU. The peak FMPC production rate was 10,000 MTU in 1960 (Mead, Savage, and Fugate ca. 1985, p. 19). The comparisons resulted in generally comparable dose rates in the EAs, as Table 4-23 shows.

4.6 ATTRIBUTIONS AND ANNOTATIONS

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

Table 4-23. Projected onsite dose rates based on systems capacity.

Year	Systems capacity (%)	Fraction of 1981 values[a]	Dose rates (mrem/hr)
1951	0	0.00	0.00
1952	24.6	0.33	0.12
1953	30	0.41	0.15
1954	30	0.41	0.15
1955	50	0.68	0.24
1956	100	1.35	0.49
1957	100	1.35	0.49
1958	100	1.35	0.49
1959	100	1.35	0.49
1960	100	1.35	0.49
1961	100	1.35	0.49
1962	100	1.35	0.49
1963	100	1.35	0.49
1964	100	1.35	0.49
1965	100	1.35	0.49
1966	100	1.35	0.49
1967	100	1.35	0.49
1968	100	1.35	0.49
1969	100	1.35	0.49
1970	100	1.35	0.49
1971	86	1.16	0.42
1972	72	0.97	0.35
1973	57	0.77	0.28
1974	43	0.58	0.21
1975	31	0.42	0.15
1976	31	0.42	0.15

a. Systems capacity and the onsite dose rate in 1981 are 74% and 0.362 mrem/hr, respectively.

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GLOSSARY

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 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.

curie (Ci)
Traditional unit of radioactivity equal to 37 billion (3.7×10^{10}) becquerels, which is approximately equal to the activity of 1 gram of pure ^{226}Ra.

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 *thermoluminescent dosimeter*.

dosimetry
Measurement and calculation of internal and external radiation doses.

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

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

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.

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*.

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.

photon

Quantum of electromagnetic energy generally regarded as a discrete particle having zero rest mass, no electric charge, and an indefinitely long lifetime. The entire range of electromagnetic radiation that extends in frequency from 10^{23} cycles per second (hertz) to 0 hertz.

photon radiation

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

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.

roentgen (R)

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).

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.

working level (WL)

Unit of concentration in air of the short-lived decay products of ^{222}Rn (^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po) and ^{220}Rn (^{216}Po, ^{212}Pb, ^{212}Bi, ^{212}Po) defined as any combination of the short-lived radioactive progeny of radon or thoron in 1 liter of air, without regard to the degree of equilibrium, that results in the ultimate emission of 130,000 megaelectron-volts of alpha energy; 1 WL equals 2.083×10^{-5} joules per cubic meter.

working level month (WLM)

Unit of exposure to radon progeny defined as exposure for 1 working month (170 working hours) to a potential alpha energy concentration of 1 WL; 1 WLM equals 1 WL times 170 hours, which is 0.00354 joule-hours per cubic meter.

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
INCORPORATION OF Q-11 SILOS AS A RADON RELEASE SOURCE, 1952 TO 1958

Figure A-1 shows the yearly maximum and mean exposures in WLMs to workers in the area of the Q-11 silos during the years when these silos were radon sources (1952 to 1958). The yearly radon releases from the Q-11 silos were estimated based on the maximum exposures in Figure A-1, and the source term was back-extrapolated using Plant 2/3 as the point of release. The derivation of the equation for the source term estimation is shown below.

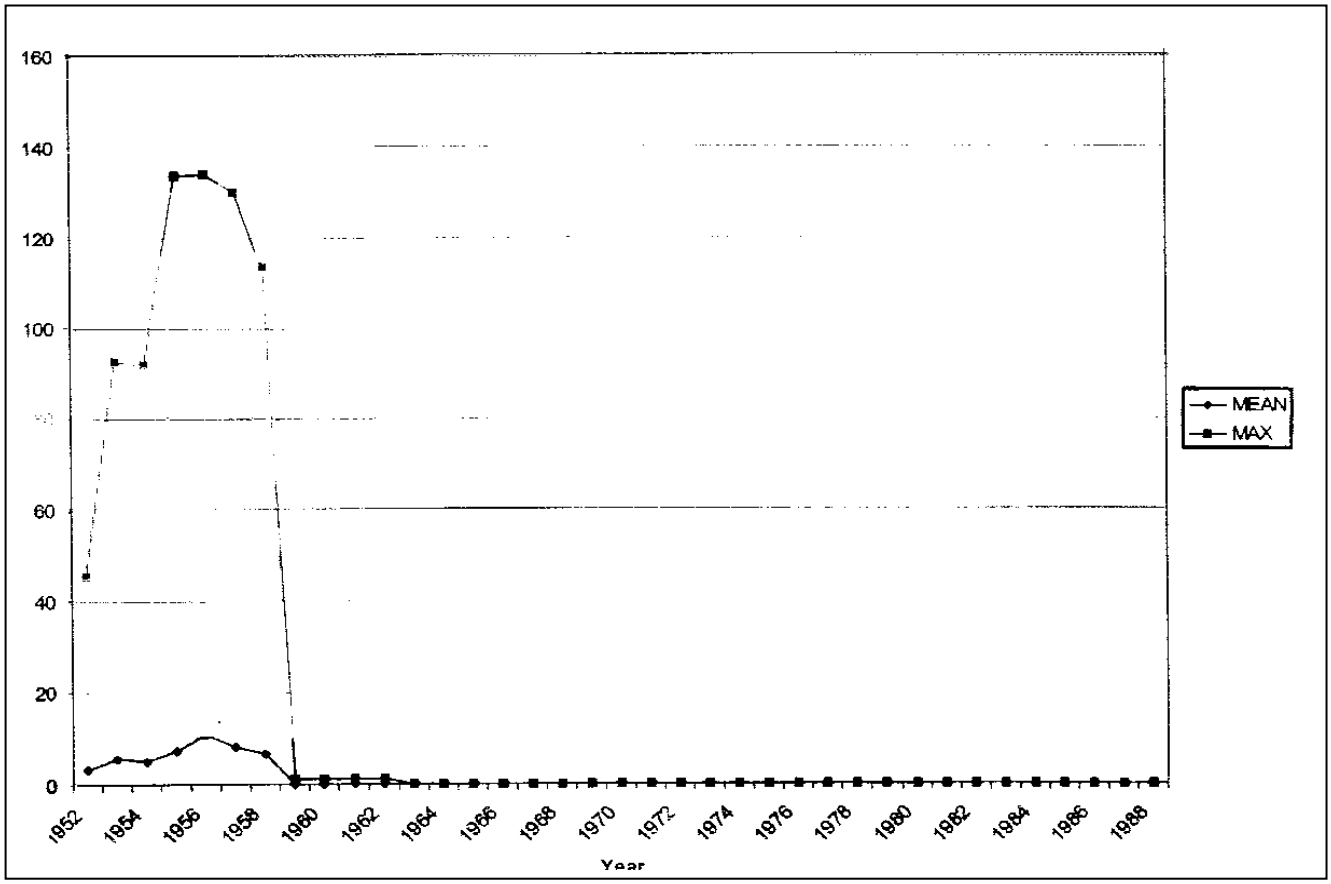

Figure A-1. Mean and highest yearly worker radon exposure estimates due to Q-11 silos (Pinney et al. 2004, p. 61, Figure 13).

The radon concentrations at distances from a continuous source were represented by modifying Equation A-1:

$$C_{EA} = R_{Q11}\left(\frac{\chi}{Q}\right)_d f(RF) \qquad \text{(A-1)}$$

where

C_{EA}	=	radionuclide concentration in EA (Ci/m^3)
R_{Q11}	=	radionuclide release rate from source Q-11 silos (Ci/yr)
$(\chi/Q)_d$	=	atmospheric dilution factor at distance *d* from Q-11 source (s/m^3)
f	=	fraction of time that the wind blew in the EA direction from source
RF	=	respirable fraction of the source aerosol (<10 µm)

Rearrangement of Equation A-1 yields:

$$R_{Q11} = C_{EA} \times \left[\left(\frac{\chi}{Q} \right)_d f\left(RF\right) \right]^{-1} \quad \text{(A-2)}$$

The radon concentration C_{EA} was expressed in terms of exposure D in WLM using Equation 4-11, and the rearranged Equation A-2 becomes:

$$C_{EA}\,(\text{pCi/L}) = \frac{D}{0.0824} \quad \text{(A-3)}$$

C_{EA} was substituted with Equation A-2 to yield:

$$R_{Q11} = \frac{D}{0.0824} \times \left[\left(\frac{\chi}{Q} \right)_d f\left(RF\right) \right]^{-1} \quad \text{(A-4)}$$

The radon release rate unit is converted to curies per year:

$$R_{Q11} = \frac{10^3\ \text{L/m}^3 \times (3.154 \times 10^7\ \text{s/yr})}{10^{12}\ \text{pCi/Ci}} \times \frac{D}{0.0824}(\text{pCi/L}) \times \left[\left(\frac{\chi}{Q} \right)_d (\text{s/m}^3) f\left(RF\right) \right]^{-1} \quad \text{(A-5)}$$

The Q-11 silo release rate R_{Q11} then reduced to:

$$R_{Q11} = (0.383 \times D) \times \left[\left(\frac{\chi}{Q} \right)_d f\left(RF\right) \right]^{-1} \quad \text{(A-6)}$$

where

R_{Q11}	=	annual radon release rate (Ci/yr)
D	=	dose (WLM)
$(\chi/Q)_d$	=	atmospheric dilution factor at distance d (m) from source, where d is a close-in distance of 25 m and $(\chi/Q)_{25m} = 2.09 \times 10^{-3}$ s/m^3 with due consideration of building wake effects
f	=	fraction of time the wind blows toward the receptor from Plant 2/3 (because the receptor is in the immediate vicinity of Q-11 silos, 100% of the time the wind blows toward the receptor and f =1.0)
RF	=	respirable fraction of the source aerosol (1.0)
0.383	=	conversion factor

Pinney et al. (2004) estimated the R_{Q11} exposures by building. The maximum dose was assumed to occur in the building nearest the Q-11 silos. The 25-m distance is a conservative assumption. The estimated annual radon release rate R_{Q11} from Equation A-6 was added to the K-65 source term as shown in Figure 4-4.

ATTACHMENT B
95TH-PERCENTILE WIND ROSE DATA FOR 16 COMPASS POINTS

Table B-1. Wind rose data and derived 95th-percentile values for environmental radionuclide concentration estimates (1991 to 2001 Environmental Reports: WEMCO 1992; FERMCO 1993 to 1996; FDF 1997 to 1999; Fluor Fernald 2000 to 2002.).

Years	N *f*	N *u* (mph)	N *u* (m/s)	NNE *f*	NNE *u* (mph)	NNE *u* (m/s)	NE *f*	NE *u* (mph)	NE *u* (m/s)	ENE *f*	ENE *u* (mph)	ENE *u* (m/s)
1991	0.040	6.10	2.73	0.034	5.80	2.60	0.049	5.40	2.42	0.073	6.00	2.69
1992	0.037	4.90	2.20	0.040	4.90	2.20	0.046	4.00	1.79	0.071	3.10	1.39
1993	0.030	5.50	2.46	0.040	5.00	2.24	0.048	4.50	2.02	0.070	4.40	1.97
1994	0.032	4.80	2.15	0.037	5.40	2.42	0.055	3.60	1.61	0.085	5.00	2.24
1995	0.045	5.70	2.55	0.034	5.80	2.60	0.054	5.50	2.46	0.080	5.00	2.24
1996	0.045	5.90	2.64	0.043	6.40	2.87	0.050	5.50	2.46	0.073	4.70	2.11
1997	0.046	5.20	2.33	0.048	6.00	2.69	0.050	6.00	2.69	0.080	5.50	2.46
2000	0.039	5.50	2.46	0.045	6.00	2.69	0.060	5.40	2.42	0.064	4.50	2.02
2001	0.030	5.50	2.46	0.032	5.80	2.60	0.053	4.80	2.15	0.063	4.50	2.02
95th percentile	**0.045**	**5.906**	**2.646**	**0.045**	**6.157**	**2.758**	**0.057**	**5.713**	**2.560**	**0.081**	**5.568**	**2.495**

Years	E *f*	E *u* (mph)	E *u* (m/s)	ESE *f*	ESE *u* (mph)	ESE *u* (m/s)	SE *f*	SE *u* (mph)	SE *u* (m/s)	SSE *f*	SSE *u* (mph)	SSE *u* (m/s)
1991	0.042	4.00	1.79	0.025	3.00	1.34	0.019	3.30	1.48	0.024	4.30	1.93
1992	0.030	3.00	1.34	0.020	2.50	1.12	0.025	3.00	1.34	0.031	3.90	1.75
1993	0.035	3.50	1.57	0.018	2.90	1.30	0.015	3.00	1.34	0.020	3.50	1.57
1994	0.045	3.80	1.70	0.015	1.50	0.67	0.018	2.50	1.12	0.018	2.50	1.12
1995	0.048	3.00	1.34	0.028	3.00	1.34	0.030	4.00	1.79	0.039	4.90	2.20
1996	0.045	3.30	1.48	0.028	2.80	1.25	0.025	3.30	1.48	0.045	4.50	2.02
1997	0.046	3.90	1.75	0.025	3.30	1.48	0.020	3.00	1.34	0.028	3.80	1.70
2000	0.040	3.60	1.61	0.020	3.00	1.34	0.020	3.20	1.43	0.030	4.00	1.79
2001	0.030	3.00	1.34	0.020	2.00	0.90	0.020	2.50	1.12	0.033	4.40	1.97
95th percentile	**0.047**	**3.891**	**1.743**	**0.027**	**3.135**	**1.405**	**0.026**	**3.594**	**1.610**	**0.039**	**4.637**	**2.078**

ATTACHMENT B
95TH-PERCENTILE WIND ROSE DATA FOR 16 COMPASS POINTS (continued)

Years	S *f*	S *u* (mph)	S *u* (m/s)	SSW *f*	SSW *u* (mph)	SSW *u* (m/s)	SW *f*	SW *u* (mph)	SW *u* (m/s)	WSW *f*	WSW *u* (mph)	WSW *u* (m/s)
1991	0.051	5.50	2.46	0.110	6.00	2.69	0.119	5.00	2.24	0.115	5.00	2.24
1992	0.061	4.80	2.15	0.103	4.70	2.11	0.115	4.70	2.11	0.090	3.70	1.66
1993	0.048	4.30	1.93	0.095	4.50	2.02	0.118	3.60	1.61	0.110	3.20	1.43
1994	0.018	4.00	1.79	0.050	4.30	1.93	0.090	4.80	2.15	0.018	4.70	2.11
1995	0.076	6.00	2.69	0.098	6.00	2.69	0.110	4.80	2.15	0.110	5.00	2.24
1996	0.075	5.50	2.46	0.108	5.40	2.42	0.100	4.50	2.02	0.090	4.60	2.06
1997	0.055	4.70	2.11	0.082	5.80	2.60	0.112	5.00	2.24	0.108	4.50	2.02
2000	0.066	5.60	2.51	0.125	6.50	2.91	0.129	4.60	2.06	0.109	4.00	1.79
2001	0.075	5.00	2.24	0.125	6.60	2.96	0.130	4.50	2.02	0.107	4.00	1.79
95th percentile	**0.074**	**5.751**	**2.577**	**0.120**	**6.395**	**2.865**	**0.126**	**4.941**	**2.213**	**0.112**	**4.898**	**2.194**

Years	W *f*	W *u* (mph)	W *u* (m/s)	WNW *f*	WNW *u* (mph)	WNW *u* (m/s)	SW *f*	SW *u* (mph)	SW *u* (m/s)	WSW *f*	WSW *u* (mph)	WSW *u* (m/s)
1991	0.104	5.00	2.24	0.081	5.20	2.33	0.064	4.20	1.88	0.050	5.20	2.33
1992	0.090	3.80	1.70	0.090	4.10	1.84	0.088	4.20	1.88	0.050	4.50	2.02
1993	0.112	3.40	1.52	0.105	3.50	1.57	0.075	3.50	1.57	0.040	4.80	2.15
1994	0.099	4.00	1.79	0.097	4.40	1.97	0.090	4.00	1.79	0.080	3.80	1.70
1995	0.112	4.80	2.15	0.100	5.00	2.24	0.070	4.00	1.79	0.058	5.00	2.24
1996	0.090	5.30	2.37	0.084	5.50	2.46	0.067	4.50	2.02	0.044	5.00	2.24
1997	0.118	5.20	2.33	0.088	5.00	2.24	0.075	4.30	1.93	0.047	4.70	2.11
2000	0.104	4.60	2.06	0.079	4.50	2.02	0.068	4.80	2.15	0.040	5.20	2.33
2001	0.095	4.00	1.79	0.080	4.50	2.02	0.075	5.00	2.24	0.038	5.50	2.46
95th percentile	**0.114**	**5.161**	**2.312**	**0.100**	**5.276**	**2.364**	**0.085**	**4.774**	**2.139**	**0.064**	**5.312**	**2.380**

ATTACHMENT C
ANNUAL AVERAGE RADON BACKGROUND CONCENTRATIONS AND CONCENTRATION AT SELECTED RADON LOCATIONS, 1989 TO 2005

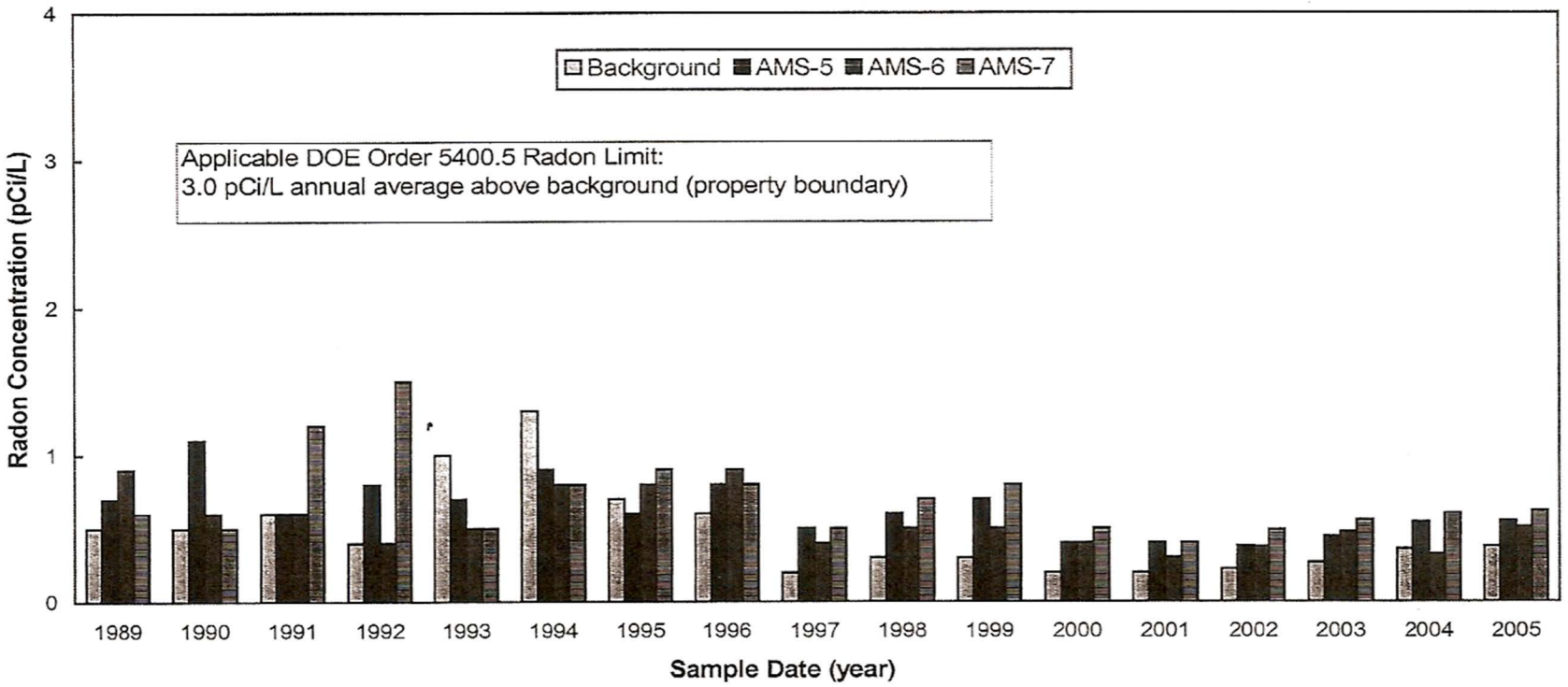

Figure C-1. Annual average radon background concentrations and concentration at selected radon locations, 1989 to 2005 (Environmental Report Fluor Fernald 2006).