



ORAU TEAM Dose Reconstruction Project for NIOSH

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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 non uranium 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.

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ACRONYMS AND ABBREVIATIONS

AMS	air monitoring station
Bq	becquerel
CFR	Code of Federal Regulations
Ci	curie
d	day
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
DCF	Dose Conversion Factor
EA	Exposure Area
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
FR	Federal Register
FEMP	Fernald Environmental Management Project
FMPC	Feed Materials Production Center
ft	foot
g	gram
GSD	geometric standard deviation
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
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
ORAU	Oak Ridge Associated Universities
pCi	picocurie
POC	probability of causation
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
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 sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

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

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled “Exposure in the Performance of Duty.” That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer “shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer ... was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation¹] guidelines established under subsection (c) ...” [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation, 42 C.F.R. Pt. 82) restrict the “performance of duty” referred to in 42 U.S.C. § 7384n(b) to nuclear weapons work (NIOSH 2010).

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

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

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

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 while on the site but outside the FMPC buildings or facilities from radioactive material discharges to the atmosphere from the facilities and from ambient external radiation that originated in the 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 workers who were unmonitored 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 emission nuclide quantities. Environmental measurements were used to establish intake rates for periods after 1988.

This TBD provides annual environmental intakes and ambient external doses for 1951 to 2012. Intake rates and exposure after 2012 should be assigned using the values established for 2012.

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 Special Exposure Cohort (SEC):

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

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

All employees of the DOE, its predecessor agencies, and their contractors and subcontractors who worked at the Feed Materials Production Center in Fernald, Ohio, from January 1, 1954, through December 31, 1967, for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees included in the Special Exposure Cohort (78 FR 229; November 27, 2013).

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

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

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

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

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

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 to 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 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 in this TBD 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. This TBD presents airborne radionuclide intake rates in Table 4-2. A spreadsheet (ORAUT 2014) documents the various calculations that were used to develop annual exposures and intake rates for each source, EA, and year combination. The maximum site wide annual intakes are assigned in the Interactive RadioEpidemiological Program (IREP) as a lognormal distribution with a geometric standard deviation (GSD) of 3.0.

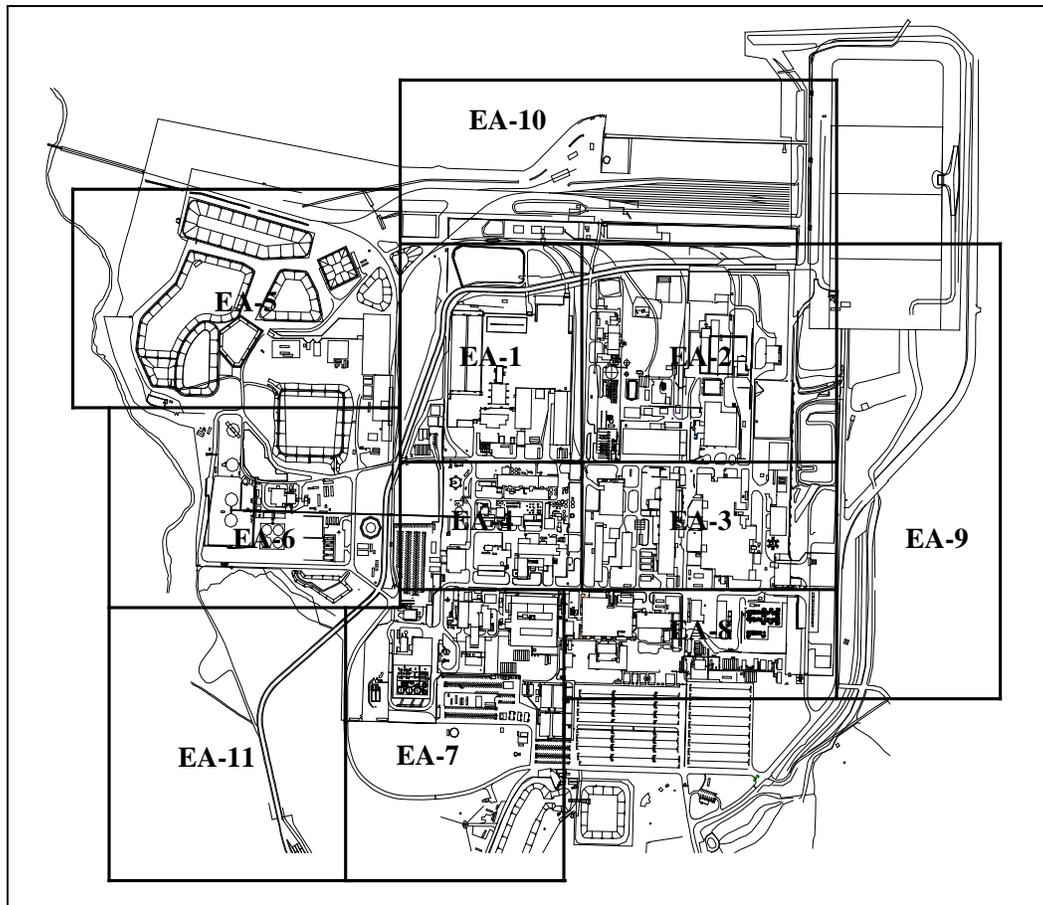


Figure 4-1. EAs for 1951 to 1988 as used in this analysis.

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, 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

Table 4-2. Maximum site wide average (1951 to 1988) or median (1989 to 2012) annual intake rates (WLM/yr for ²²²Rn and Bq/yr for all other radionuclides) via inhalation.^{a,c,d}

Year	Total U ^b (Bq/yr)	Th-228 (Bq/yr)	Th-232 (Bq/yr)	Pu-239 (Bq/yr)	Np-237 (Bq/yr)	Tc-99 (Bq/yr)	Ru-103/106 (Bq/yr)	Zr/Nb-95 (Bq/yr)	(Sr-90) (Bq/yr)	Rn-222 (WLM)
1951	5.87E-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	6.11E-01
1952	3.25E+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	2.20E+01
1953	1.57E+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	3.95E+01
1954	1.14E+03	4.83E+00	4.83E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.93E+01
1955	2.84E+03	5.52E+00	5.52E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.68E+01
1956	1.18E+03	5.18E+00	5.18E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.68E+01
1957	8.35E+02	2.55E-03	2.55E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.47E+01
1958	9.19E+02	5.83E-03	5.83E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.72E+01
1959	1.06E+03	5.83E-03	5.83E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.55E+00
1960	5.56E+03	7.51E-02	7.51E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.55E+00
1961	1.32E+03	1.46E-01	1.46E-01	1.06E+01	4.21E+00	2.59E+02	1.85E+02	5.56E+01	7.41E+00	1.55E+00
1962	9.39E+02	2.86E-01	2.86E-01	7.54E+00	3.00E+00	1.85E+02	1.32E+02	3.96E+01	5.28E+00	1.55E+00
1963	1.01E+03	3.20E-01	3.21E-01	8.07E+00	3.21E+00	1.98E+02	1.41E+02	4.24E+01	5.66E+00	1.55E+00
1964	7.46E+02	2.46E+00	2.46E+00	5.99E+00	2.38E+00	1.47E+02	1.05E+02	3.15E+01	4.20E+00	1.55E+00
1965	3.34E+02	2.46E+00	2.46E+00	2.10E+00	8.36E-01	5.15E+01	3.68E+01	1.10E+01	1.47E+00	1.55E+00
1966	3.12E+04	3.12E+00	3.12E+00	1.96E+02	7.81E+01	4.81E+03	3.44E+03	1.03E+03	1.38E+02	1.55E+00
1967	2.80E+02	2.46E+00	2.46E+00	1.76E+00	7.00E-01	4.31E+01	3.09E+01	9.25E+00	1.23E+00	1.55E+00
1968	4.92E+02	2.47E+00	2.47E+00	3.09E+00	1.23E+00	7.57E+01	5.41E+01	1.62E+01	2.17E+00	1.55E+00
1969	1.95E+02	1.43E+01	1.43E+01	1.23E+00	4.88E-01	3.00E+01	2.15E+01	6.44E+00	8.59E-01	1.55E+00
1970	1.78E+02	1.05E+01	1.05E+01	1.12E+00	4.44E-01	2.74E+01	1.96E+01	5.87E+00	7.83E-01	1.55E+00
1971	1.07E+02	5.56E+00	5.56E+00	6.75E-01	2.68E-01	1.65E+01	1.18E+01	3.54E+00	4.73E-01	1.55E+00
1972	2.00E+02	1.03E+00	1.03E+00	1.26E+00	4.99E-01	3.07E+01	2.20E+01	6.59E+00	8.80E-01	1.55E+00
1973	2.80E+02	5.69E-01	5.69E-01	7.04E+00	2.20E+00	1.51E+02	3.08E+01	9.23E+00	1.23E+00	1.55E+00
1974	3.12E+02	7.44E-01	7.44E-01	7.85E+00	2.45E+00	1.69E+02	3.44E+01	1.03E+01	1.37E+00	1.55E+00
1975	3.65E+02	5.41E-01	5.41E-01	9.19E+00	2.87E+00	1.97E+02	4.02E+01	1.21E+01	1.61E+00	1.55E+00
1976	3.04E+02	6.63E-01	6.63E-01	7.66E+00	2.39E+00	1.65E+02	3.35E+01	1.01E+01	1.34E+00	1.55E+00
1977	1.31E+02	2.94E+00	2.94E+00	3.30E+00	1.03E+00	7.10E+01	1.45E+01	4.34E+00	5.78E-01	1.55E+00
1978	6.78E+01	2.94E+00	2.94E+00	1.71E+00	5.33E-01	3.67E+01	7.47E+00	2.24E+00	2.99E-01	1.55E+00
1979	1.26E+02	2.94E+00	2.94E+00	3.17E+00	9.89E-01	6.80E+01	1.39E+01	4.15E+00	5.54E-01	1.55E+00
1980	4.82E+01	7.51E-01	7.51E-01	1.21E+00	3.79E-01	2.60E+01	5.31E+00	1.59E+00	2.12E-01	3.41E-01
1981	5.74E+01	7.51E-01	7.51E-01	1.44E+00	4.51E-01	3.10E+01	6.32E+00	1.90E+00	2.53E-01	3.41E-01
1982	4.94E+01	7.51E-01	7.51E-01	1.24E+00	3.88E-01	2.67E+01	5.44E+00	1.63E+00	2.18E-01	2.03E-01
1983	8.31E+01	7.51E-01	7.51E-01	2.09E+00	6.54E-01	4.49E+01	9.15E+00	2.74E+00	3.66E-01	2.03E-01
1984	5.85E+01	7.51E-01	7.51E-01	1.47E+00	4.60E-01	3.16E+01	6.44E+00	1.93E+00	2.58E-01	2.03E-01
1985	4.65E+01	7.51E-01	7.51E-01	1.17E+00	3.66E-01	2.51E+01	5.12E+00	1.54E+00	2.05E-01	2.03E-01
1986	4.74E+01	7.51E-01	7.51E-01	1.19E+00	3.73E-01	2.56E+01	5.22E+00	1.57E+00	2.09E-01	2.03E-01

Year	Total U ^b (Bq/yr)	Th-228 (Bq/yr)	Th-232 (Bq/yr)	Pu-239 (Bq/yr)	Np-237 (Bq/yr)	Tc-99 (Bq/yr)	Ru-103/106 (Bq/yr)	Zr/Nb-95 (Bq/yr)	(Sr-90) (Bq/yr)	Rn-222 (WLM)
1987	4.53E+01	7.85E-01	7.85E-01	1.14E+00	3.56E-01	2.45E+01	4.98E+00	1.49E+00	1.99E-01	2.03E-01
1988	4.40E+01	8.14E-01	8.14E-01	1.11E+00	3.46E-01	2.38E+01	4.84E+00	1.45E+00	1.94E-01	1.07E-01
1989	3.65E+01	2.76E-05	3.94E-05	9.18E-01	2.87E-01	1.97E+01	4.02E+00	1.21E+00	1.61E-01	1.00E-01
1990	1.97E+01	1.49E-05	2.13E-05	4.96E-01	1.55E-01	1.07E+01	2.17E+00	6.51E-01	8.68E-02	9.22E-02
1991	3.06E+01	2.32E-05	3.31E-05	7.71E-01	2.41E-01	1.66E+01	3.38E+00	1.01E+00	1.35E-01	1.05E-01
1992	1.65E+01	1.25E-05	1.79E-05	4.16E-01	1.30E-01	8.94E+00	1.82E+00	5.46E-01	7.29E-02	1.05E-01
1993	1.70E+01	1.29E-05	1.84E-05	4.28E-01	1.34E-01	9.20E+00	1.88E+00	5.62E-01	7.50E-02	1.21E-01
1994	9.49E+00	7.18E-06	1.03E-05	2.39E-01	7.46E-02	5.13E+00	1.04E+00	3.13E-01	4.18E-02	1.21E-01
1995	1.46E-01	1.10E-07	1.58E-07	3.67E-03	1.15E-03	7.89E-02	1.61E-02	4.82E-03	6.43E-04	1.13E-01
1996	2.04E+01	1.55E-05	2.21E-05	5.14E-01	1.61E-01	1.10E+01	2.25E+00	6.75E-01	9.00E-02	1.54E-01
1997	2.34E+01	1.77E-05	2.52E-05	5.88E-01	1.84E-01	1.26E+01	2.57E+00	7.71E-01	1.03E-01	1.70E-01
1998	2.12E+01	1.60E-05	2.29E-05	5.32E-01	1.66E-01	1.14E+01	2.33E+00	6.99E-01	9.32E-02	5.52E-02
1999	2.85E+01	2.16E-05	3.08E-05	7.17E-01	2.24E-01	1.54E+01	3.14E+00	9.41E-01	1.25E-01	1.21E-01
2000	2.75E+01	2.08E-05	2.97E-05	6.92E-01	2.16E-01	1.49E+01	3.03E+00	9.08E-01	1.21E-01	1.05E-01
2001	2.54E+01	1.93E-05	2.75E-05	6.40E-01	2.00E-01	1.38E+01	2.80E+00	8.40E-01	1.12E-01	1.05E-01
2002	1.35E+01	1.02E-05	1.46E-05	3.39E-01	1.06E-01	7.28E+00	1.48E+00	4.45E-01	5.94E-02	1.21E-01
2003	5.69E+01	4.31E-05	6.15E-05	1.43E+00	4.48E-01	3.08E+01	6.27E+00	1.88E+00	2.51E-01	1.05E-01
2004	3.19E+02	2.41E-04	3.45E-04	8.02E+00	2.51E+00	1.72E+02	3.51E+01	1.05E+01	1.40E+00	1.05E-01
2005	9.07E+00	6.87E-06	9.81E-06	2.28E-01	7.14E-02	4.90E+00	9.99E-01	3.00E-01	4.00E-02	9.64E-02
2006	3.16E+00	2.39E-06	3.42E-06	7.96E-02	2.49E-02	1.71E+00	3.48E-01	1.04E-01	1.39E-02	1.09E-01
2007	1.22E+00	9.20E-07	1.31E-06	3.06E-02	9.57E-03	6.57E-01	1.34E-01	4.02E-02	5.36E-03	9.64E-02
2008	8.03E-01	6.07E-07	8.68E-07	2.02E-02	6.31E-03	4.34E-01	8.84E-02	2.65E-02	3.54E-03	1.09E-01
2009	6.57E-01	4.97E-07	7.10E-07	1.65E-02	5.17E-03	3.55E-01	7.23E-02	2.17E-02	2.89E-03	1.09E-01
2010	6.63E-01	5.01E-07	7.16E-07	1.67E-02	5.21E-03	3.58E-01	7.30E-02	2.19E-02	2.92E-03	1.09E-01
2011	6.63E-01	5.01E-07	7.16E-07	1.67E-02	5.21E-03	3.58E-01	7.30E-02	2.19E-02	2.92E-03	1.09E-01
2012	6.63E-01	5.01E-07	7.16E-07	1.67E-02	5.21E-03	3.58E-01	7.30E-02	2.19E-02	2.92E-03	1.09E-01

- 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).
- Total uranium is interpreted as U-234 for dose reconstruction purposes.
- Three short-duration episodic emissions (Table 4-6) were modeled using Equation 4-7 and averaged into the annual intake rates.
- Intake rates for Th-230 and Ra-226 were calculated to be in the range of a few $\mu\text{Bq/yr}$ but are not shown because they are judged to be negligibly small.

The external gamma dose rate site wide exposure rates by the EAs are presented in ORAUT (2014) 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, then the annual maximum gamma doses should be assigned for overestimate claims with a constant distribution in IREP. For best-estimate and compensable claims, the annual site average gamma doses should be used with a lognormal distribution in IREP with a GSD of 1.3.

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 4-3. 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

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.
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
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	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
2010	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
2011	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
2012	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

- External exposures include ambient doses since background doses were not available for all of the dose components.
- Dose Reconstructors should use Ambient (H*10) DCFs for 1952-1976 (based on instrument readings) and Exposure (R) DCFs for 1977-2012 (TLD on a fenceline based), (Section 4.1.2 of NIOSH 2007a).
- Ambient gamma radiation doses are based on 2000 hours/year.

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

Period	Available historic information and data description	Source document ^a
1951–1988	Airborne uranium, thorium, and radon emissions including unmonitored and episodic releases.	<i>History of FMPC Radionuclide Discharges</i> (Boback et al. 1987) <i>Addendum to FMPC-2082, “History of FMPC Radionuclide Discharges”</i> (Dolan and Hill 1988) <i>The Fernald Dosimetry Reconstruction Project, Tasks 2 and 3, Radionuclide Source Terms and Uncertainties</i> (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
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
1997–2005	Radon in ambient air data from the Radon Monitoring Program that included AMSs at onsite locations.	Environmental Reports for 1997 to 2005
2006–2012 ^b	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

- a. For convenience, the annual environmental reports for 1975 to 2012 are listed sequentially in the Environmental Reports section at the end of the Reference list.
- b. 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 time spans as listed in the first column of Table 4-4.

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 FMPC 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 (from 1951 through 1988). Dolan and Hill (1988), an addendum to Boback et al., 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 quantities 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. The thorium discharges are from Dolan and Hill (1988). The values for releases of ²²²Rn and its progeny are from Voillequé et al. 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).

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
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	9.2e	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	32e	0.6	0	3.0	0.0	133	0	1.0	0.5	25e	1.5	197

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
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 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.

The values in Table 4-5 include:

- Monitored source emissions from dust collectors,
- Emissions from Plant 2/3 UO₃ gulping,
- Emissions from Plant 8 scrubbers,
- Building exhaust emissions due to normal operating conditions,
- Fugitive emissions from waste pits,
- Nonroutine uranium fires and UF₆ 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) to show a summary of six episodic releases of uranium that were identified from the incident reports and air monitoring.

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 summarizes annual fugitive thorium emissions due to wind erosion from 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).

ORAUT (2011) provides a background discussion and summary NIOSH position on the activity of ²²²Rn that was emitted from the K-65 silos, which is between 5,000 and 6,000 Ci/yr during years when the silos were filled. Appendix J of Voillequé et al. (1995) estimated emissions of ²²²Rn and radon progeny primarily from the K-65 silos and from other stored K-65 materials, including the drummed material stored at Plant 1, which would eventually be placed in the K-65 silos.

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 obtained from the African Metals Corporation before processing. This ore had high ²²⁶Ra content which decays to ²²²Rn. 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.

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

Year	Plant 8	Plant 9	Pilot Plant	Yearly total
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

Figure 13 in Pinney et al. (2004) reported the yearly mean and maximum exposure in working level month (WLM) 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. is reproduced in Figure A-1 in Attachment A. The yearly radon emissions from the Q-11 silos were estimated based a re-arrangement 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.

Combining each of these emission sources, Figure 4-2 shows a graphic representation of the ²²²Rn emission quantity estimates and Table 4-9 summarizes the annual median radon release estimates from (Voillequé et al. 1995) and maximized radon emission rates derived from the data in Pinney et al. (2004). Waste Pits were also a source of radon at FMPC but the quantities released were negligible compared to those in Table 4-9. To ensure 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 background radon concentrations measured from 1989 to 2000 (Environmental Report Fluor Fernald 2001). Attachment C shows background concentration measurements.

4.4.1.2 Exposure Site Locations

Airborne emissions data indicate that the quantity 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 in the previous section 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 (2014).

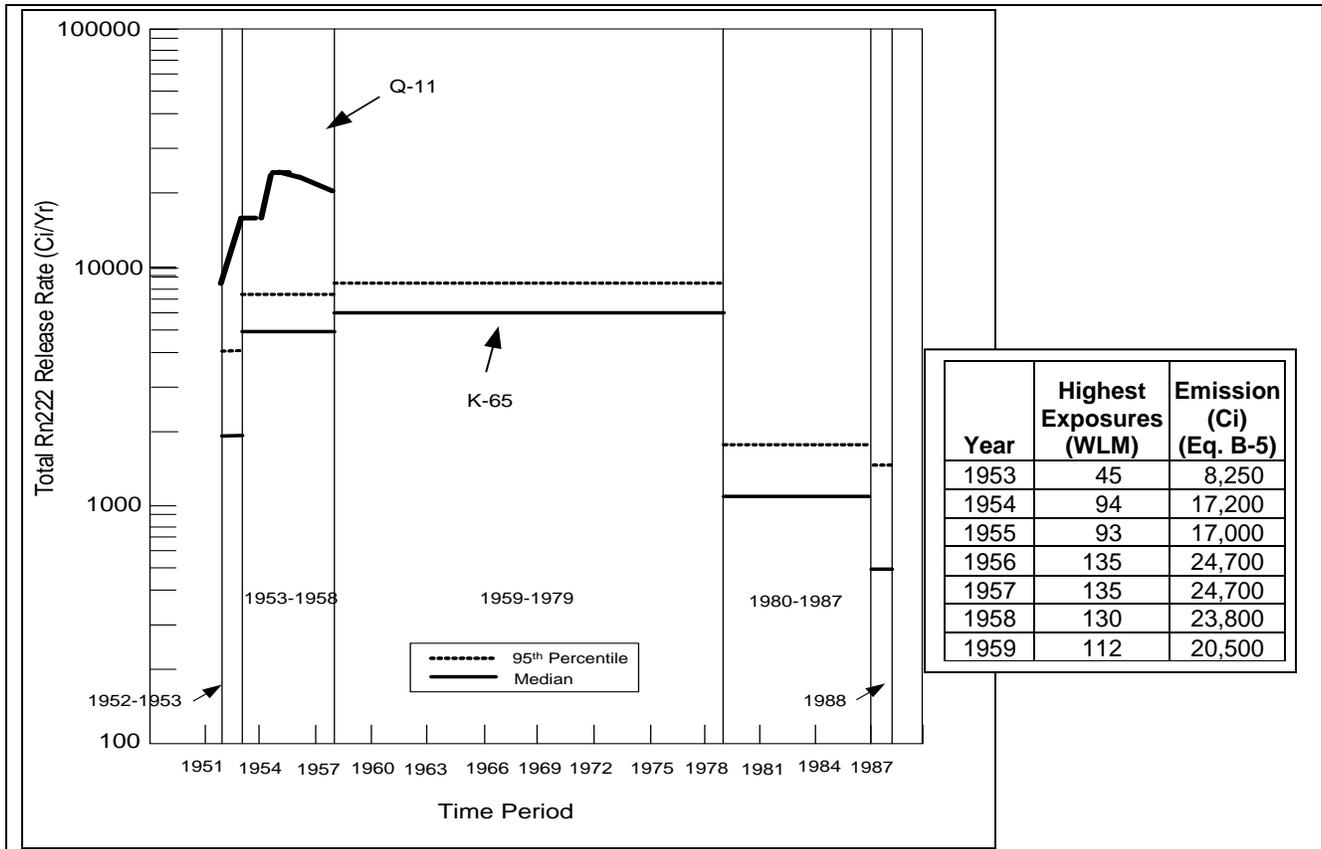


Figure 4-2. Estimate of total ²²²Rn releases from K-65 (Voillequé et al. 1995, p. 45) and Q-11 silos (Pinney et al. 2004, p. 61).

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:

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

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.
- $(X/Q)_{di}$ = 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 [$<10 \mu m$ (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.

Table 4-9. Releases (Bq) of ²²²Rn used in this analysis (Voillequé et al. 1995).

Year	K-65 silos	Q-11 silo	Plant 1 Pad, Drum Storage	Annual total
1951	0.0E+00	0.00E+00	1.30E+12	1.30E+12
1952	7.4E+13	3.05E+14	2.15E+13	4.01E+14
1953	7.4E+13	6.37E+14	3.52E+12	7.15E+14
1954	1.9E+14	6.31E+14	0.00E+00	8.21E+14
1955	1.9E+14	9.15E+14	0.00E+00	1.11E+15
1956	1.9E+14	9.15E+14	0.00E+00	1.11E+15
1957	1.9E+14	8.81E+14	0.00E+00	1.07E+15
1958	1.9E+14	7.59E+14	0.00E+00	9.49E+14
1959	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1960	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1961	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1962	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1963	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1964	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1965	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1966	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1967	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1968	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1969	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1970	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1971	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1972	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1973	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1974	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1975	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1976	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1977	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1978	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1979	2.2E+14	0.00E+00	0.00E+00	2.20E+14
1980	4.4E+13	0.00E+00	0.00E+00	4.40E+13
1981	4.4E+13	0.00E+00	0.00E+00	4.40E+13
1982	4.4E+13	0.00E+00	0.00E+00	4.40E+13
1983	4.4E+13	0.00E+00	0.00E+00	4.40E+13
1984	4.4E+13	0.00E+00	0.00E+00	4.40E+13
1985	4.4E+13	0.00E+00	0.00E+00	4.40E+13
1986	4.4E+13	0.00E+00	0.00E+00	4.40E+13
1987	4.4E+13	0.00E+00	0.00E+00	4.40E+13
1988	1.9E+13	0.00E+00	0.00E+00	1.90E+13
Total	6.11E+15	5.04E+15	2.63E+13	1.12E+16

4.4.1.3.1 Radionuclide Emission Rates, $R_i \times RF$

The average annualized emission rates were derived by dividing emission quantities in Tables 4-5, 4-8, and 4-9 by 3.156×10^7 seconds (an average year). In 1951 and 1953, ²²²Rn 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 quantities 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 data sets 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 re-sampled with a result of 9.0 μm . Figure 4-3 shows the measured particle size distributions from the stacks. The average value of the AMAD was 6.6 μm .

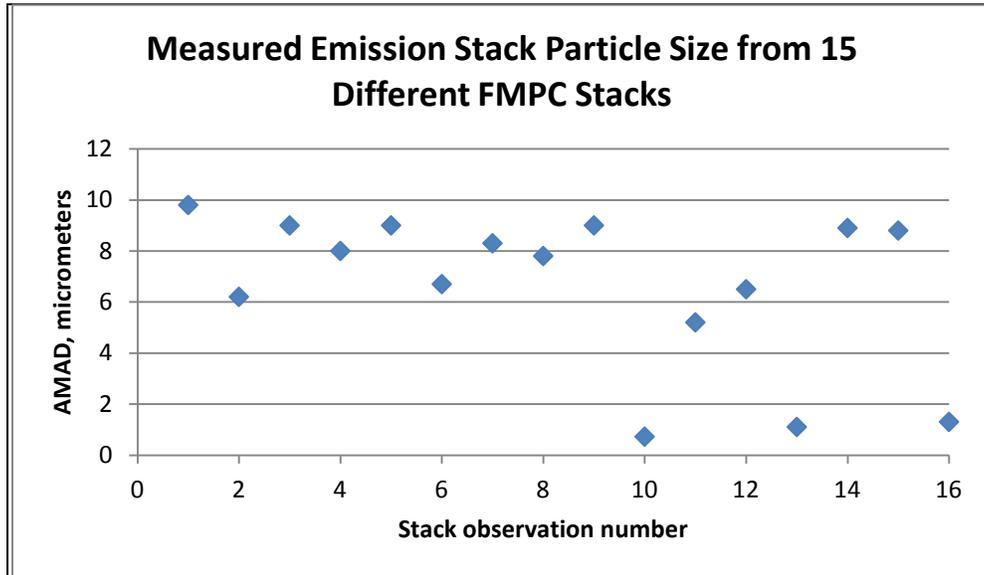


Figure 4-3. Measured emission stack particle size from 15 different FMPC stacks (Spenceley 1985).

The measurements were made using a multi-stage particle size separator and, 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 in the neighborhood of 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 and this bias is favorable to the claimant for dose reconstruction purposes. Figure 4-4 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 4-4 is 74%.

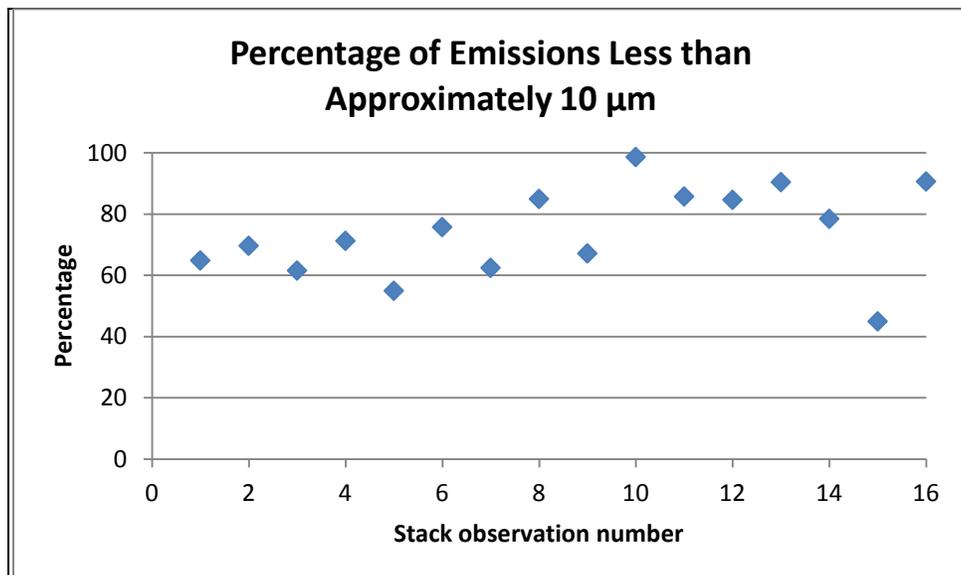


Figure 4-4. Percentage of stack emission less than approximately 10 μm AMAD (Spenceley 1985).

Appendix E of *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. 184):

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

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 4-5 and 4-6, respectively
- u = mean wind speed at level of plume centerline (m/s)
- H = effective release height (m)

For particulates, the χ/Q equation becomes:

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

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 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 4-5 and 4-6, 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 first encountered occupied location in each EA. These distances are the shortest distance between the emission sources and the occupied locations. For EAs that contained

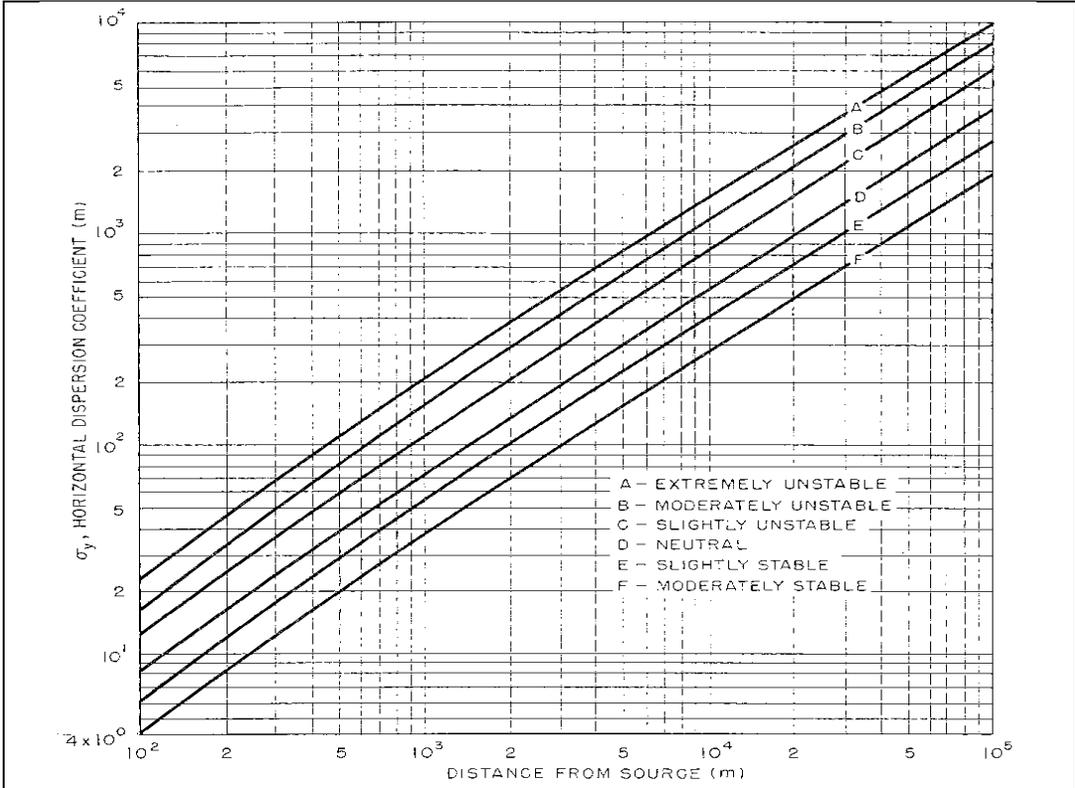


Figure 4-5. Horizontal diffusion σ_y versus downwind distance from point source for Pasquill's stability categories (Slade 1968, p. 115).

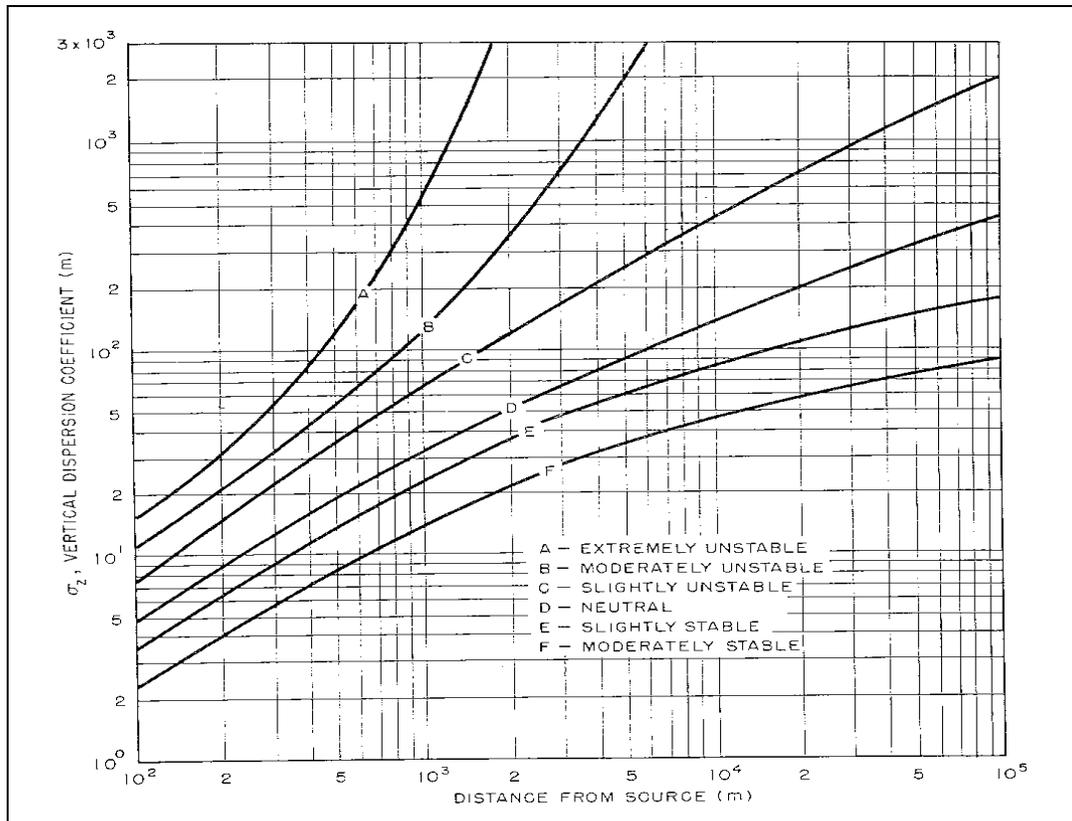


Figure 4-6. Vertical diffusion σ_z versus downwind distance from continuously emitting point source for Pasquill's stability categories (Slade 1968, p. 116).

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

EA	Contributing emission sources										
	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.

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 4-5 and 4-6, the "F" stability category represents a moderately stable condition that has the lowest σ_y and σ_z , which would result 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, which is also favorable to claimants, from Table 2 of Boback et al. (1987, p. 54), which 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 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 the 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, f_i

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 annual environmental reports for 1991 to 2001 (see the end of the Reference list) 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 4-7) to determine the wind direction frequency for each of the compass points. Table 4-11 presents the summary wind rose data that was used in this analysis. Attachment B contains more detail.

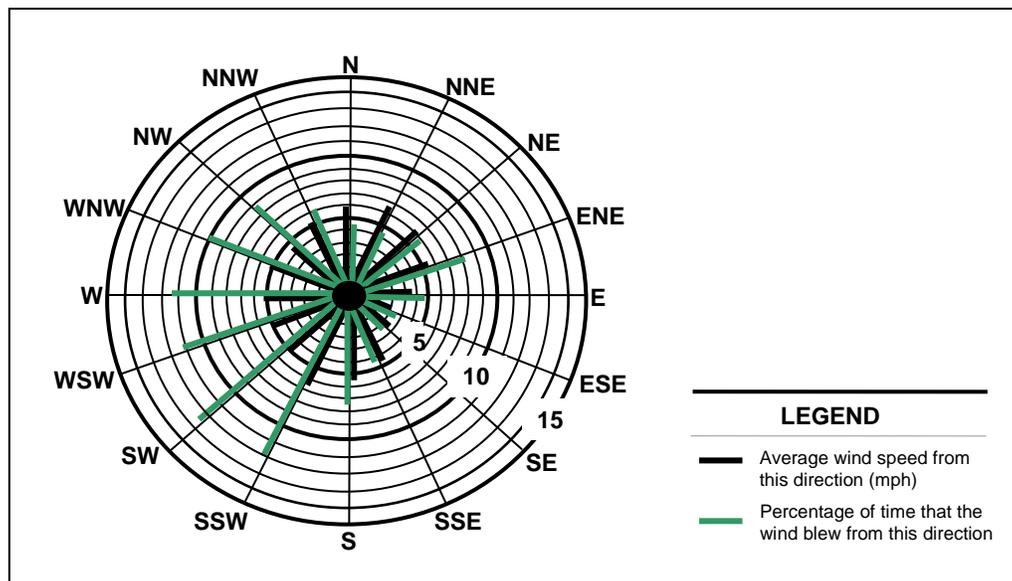


Figure 4-7. 95th-percentile wind rose data, 10-m height, 1991 to 1997 and 2000 to 2001 (FDF 1998, p. 35; Fluor Fernald 2002, p. 28).

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

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 methodology in Randerson (1984). The modified expressions for σ_y' and σ_z' for an elevated release are:

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

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

and

$$\frac{\chi}{Q} = \frac{1}{\pi\sigma_y'\sigma_z'u} \exp\left[-\frac{1}{2}\left(\frac{h_e^2}{\sigma_z'^2}\right)\right] \quad (4-6)$$

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)

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. 39) 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, such as those from Events 1, 2, and 3, were evaluated as quasi-instantaneous (puff) releases. The remaining episodic releases (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:

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

where

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

For particulates, multiply Equation 4-7 by one-half. Slade (1968 p. 419) presented equations to calculate values of σ_{y_l} and σ_{z_l} 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_{y_l} is calculated as:

$$\sigma_{y1} = 0.14 \times X^{0.93} \quad (4-8)$$

The value of σ_{z1} is calculated as:

$$\sigma_{x_{z1}} = 0.05 \times X^{0.61} \quad (4-9)$$

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

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

where

- I = radionuclide intake (g)
- q = episodic release quantity (g)
- (ψ/Q) = atmospheric dilution factor for puff release (s/m³) (see Attachment C)
- f = fraction of time that the wind blew toward the EA (Figure 4-8)
- Br = breathing rate (m³/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 m³/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 that was processed at FMPC was enriched in ²³⁵U 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 2004, p.10).

Uranium enrichment	Specific activity (Bq/mg U)	Isotope	Activity fraction
1% enriched	29	U-234	0.637
		U-235	0.021
		U-238	0.342
2% enriched	37	U-234	0.769
		U-235	0.0268
		U-238	0.204

ORAUT-RPRT-0052, *Feed Materials Production Center Internal Dose Topics* (ORAUT 2011), documented the resolution of several internal dosimetry issues including the basis for recycled uranium (RU) default values in Table 4-14. RU was first processed in significant quantity at FMPC beginning in 1961, so none of the RU-associated contaminants was present before 1961. Between 1961 and 1972, RU was present in the process stream at concentrations shown in Table 4-14. Beginning in 1973, some RU contaminants were present in higher concentrations.

Table 4-14. RU contaminant radionuclides (Bq/g U) (ORAUT 2011, pg. 44).

Period	Pu-239	Np-237	Tc-99	Th-232	Th-228	Ru-103/106	Zr/Nb-95	Sr-90
Before 1961	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1961–1972	2.33E+02	9.25E+01	5.70E+03	4.00E-02	2.80E-02	4.08E+03	1.22E+03	1.63E+02
After 1972	9.31E+02	2.91E+02	2.00E+04	4.00E-02	2.80E-02	4.08E+03	1.22E+03	1.63E+02

Plant 2/3, the Refinery, was operated to convert the U.S. Government stockpile of natural uranium ore concentrates to UO₃. Ore process operation in Plant 2/3 started in 1953 and continued until 1962 when UO₃ production was transferred to the Mallinckrodt Chemical Company at the Weldon Spring Plant. In 1966, the UO₃ refinery at the Weldon Spring Plant was shut down; the refinery at Fernald was reactivated in April 1966 and continued operating 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, ²²⁶Ra and ²³⁰Th. 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 ²²⁶Ra and ²³⁰Th 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 ²²⁶Ra and ²³⁰Th activity to uranium mass. The uranium effluent in kilograms from Plant 2/3 was multiplied by these ratios to estimate the activity of ²²⁶Ra and ²³⁰Th in the Plant 2/3 effluent.

Table 4-15. Ratio of ²²⁶Ra and ²³⁰Th 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 (2014) for derivation.

Radon

Radon-222 intake levels are expressed in units of working level months (WLM). For ²²²Rn, 1 working level (WL) is equal to 100 pCi/L of ²²²Rn in 100% equilibrium with its short-lived decay products. A month is defined as a period of 170 work hours in a month. 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,

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

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.

Radon Exposure Records

Some claimant files may contain radon exposure records in units of WLMs. These records typically appear in “Personnel Exposure” records, separate from the “DOE Response file.” Dose Reconstructors 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 ²²²Rn exposures, and when individual radon exposure data exist, both the environmental ²²²Rn and the individual ²²²Rn data need to be evaluated. However, Dose Reconstructors should assign only the highest dose determined from this comparison, not both. The radon intakes from the “Personnel Exposure Records” are assumed to be a chronic exposure and assigned as a constant distribution in IREP for overestimates. For best-estimate or compensable claims, because the dose is almost instantaneous, Dose Reconstructors 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 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 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 Dose Reconstructors on what the fields represent.

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

Field	Content
SSN	Social Security Number
SURVEY_TOT_EXP	Yearly K65 + Q11 radon exposure (WLM) from survey data
SURVEY_YRLY_K65	Yearly K65 radon exposure (WLM) from survey data
SURVEY_YRLY_Q11	Yearly Q11 radon exposure (WLM) from survey data
TOTAL_EXP	Yearly K65+Q11 radon exposure (WLM) from estimated shift
YR	Start date year
YRLY_K65	Yearly K65 radon exposure (WLM) from estimated shift
YRLY_Q11	Yearly Q11 radon exposure (WLM) from estimated shift

4.4.1.5 Results

ORAUT (2014) 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 are values summarized and, for each radionuclide of interest, the intake rate in the EA with the highest value is assigned for site wide application. Table 4-2 presents these results in summary form.

4.4.2 Nonradon Radionuclide Air Concentrations and Intakes, 1989 to 2012

Starting in 1989, annual environmental reports to DOE reported estimates of the amount of radioactive airborne material that was released from the site. FMPC used those amounts to estimate how much radiation exposure to which workers, nearby residents, and the population in general might have received as the result of radioactive airborne emissions from the facilities at FMPC. The annual airborne uranium concentrations from the onsite and perimeter air monitoring stations were used with the model in ORAUT-PROC-0031, *Site Profile and Technical Basis Document Development* (ORAUT 2012b), to determine the onsite air concentrations and intakes for environmental dose to workers for 1989 to 2012. The date range for this section suggests that the period of time of analyses of data occurred through 2012. 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 2012.

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 are monitored by taking continuous isokinetic particulate samples, which are 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 EPA 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 air in the plants is significantly lower than the calculated average of the continuous air monitor results.

4.4.2.1.5 Fugitive Emissions

These estimates were based upon 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.

Table 4-17 presents the median of maximum values of all uranium air samples that were collected after 1988.

Table 4-17. Medians of the maximum values of all uranium air concentrations (Bq/m³) samples by year (ORAUT 2014).

Year	Uranium air concentration
1989	2.03E-02
1990	1.09E-02
1991	1.70E-02
1992	9.19E-03
1993	9.46E-03
1994	5.27E-03
1995	8.11E-05
1996	1.14E-02
1997	1.30E-02
1998	1.18E-02
1999	1.58E-02
2000	1.53E-02
2001	1.41E-02
2002	7.49E-03
2003	3.16E-02
2004	1.77E-01
2005	5.04E-03
2006	1.76E-03
2007	6.76E-04
2008	4.46E-04
2009	3.65E-04
2010	3.65E-04 ^a
2011	3.65E-04 ^a
2012	3.65E-04 ^a

a. Results based on 2009 air sampling data when particulate uranium site sampling ended.

Results

Uranium intake rates and intakes of associated trace contaminants are reported in Table 4-2. Calculation of the intake rates is presented in ORAUT (2014).

4.4.3 Radon Air Concentrations and Intake Estimates, 1989 to 2012

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 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 are changed quarterly and sent to the supplier for analysis. At year's end, 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 4-8 and 4-9 show monitoring locations in 1994 timeframe for:

- 11 locations offsite,

- 21 locations along the fence line,
- 4 locations onsite 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 radon air samples that were collected after 1988 through 2008. The data in the table from 2009 to 2012 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.

Table 4-18. Medians of the maximum values of all radon air concentrations (Bq/m³) samples by year (ORAUT 2014).

Year	Radon air concentration
1989	7.50E-01
1990	6.50E-01
1991	8.00E-01
1992	8.00E-01
1993	1.00E+00
1994	1.00E+00
1995	9.00E-01
1996	1.40E+00
1997	1.60E+00
1998	2.00E-01
1999	1.00E+00
2000	8.00E-01
2001	8.00E-01
2002	1.00E+00
2003	8.00E-01
2004	8.00E-01
2005	7.00E-01
2006	8.50E-01
2007	7.00E-01
2008	8.50E-01
2009	8.50E-01 ^a
2010	8.50E-01 ^a
2011	8.50E-01 ^a
2012	8.50E-01 ^a

a. Results based on 2008 air sampling data, when radon site sampling ended.

4.5 ONSITE EXTERNAL DOSES FROM AMBIENT RADIATION

External environmental radiation dose at FMPC has been primarily 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 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.

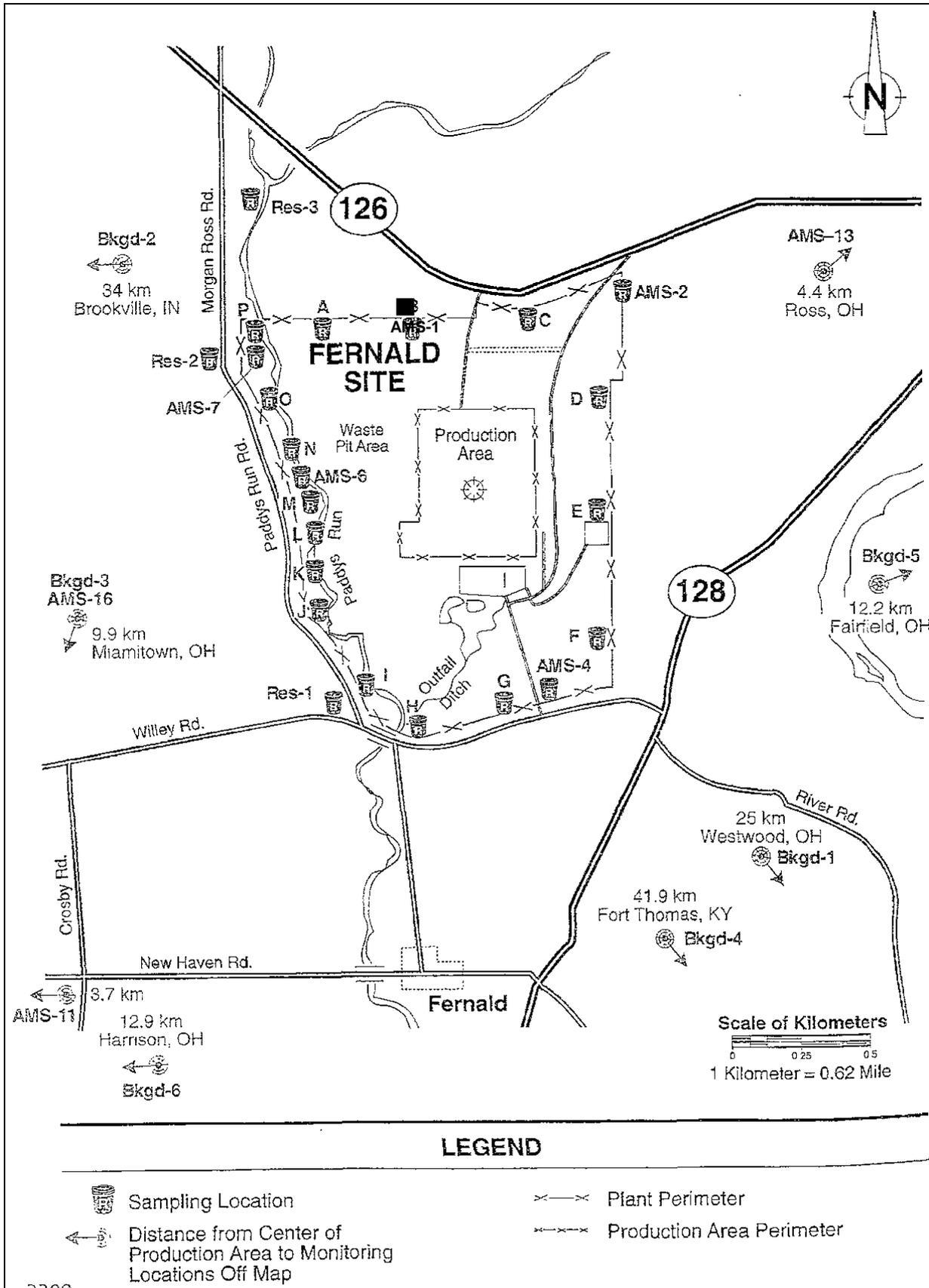


Figure 4-8. Offsite and fenceline radon monitoring locations (Environmental Report FERMCO 1995, p. 156).

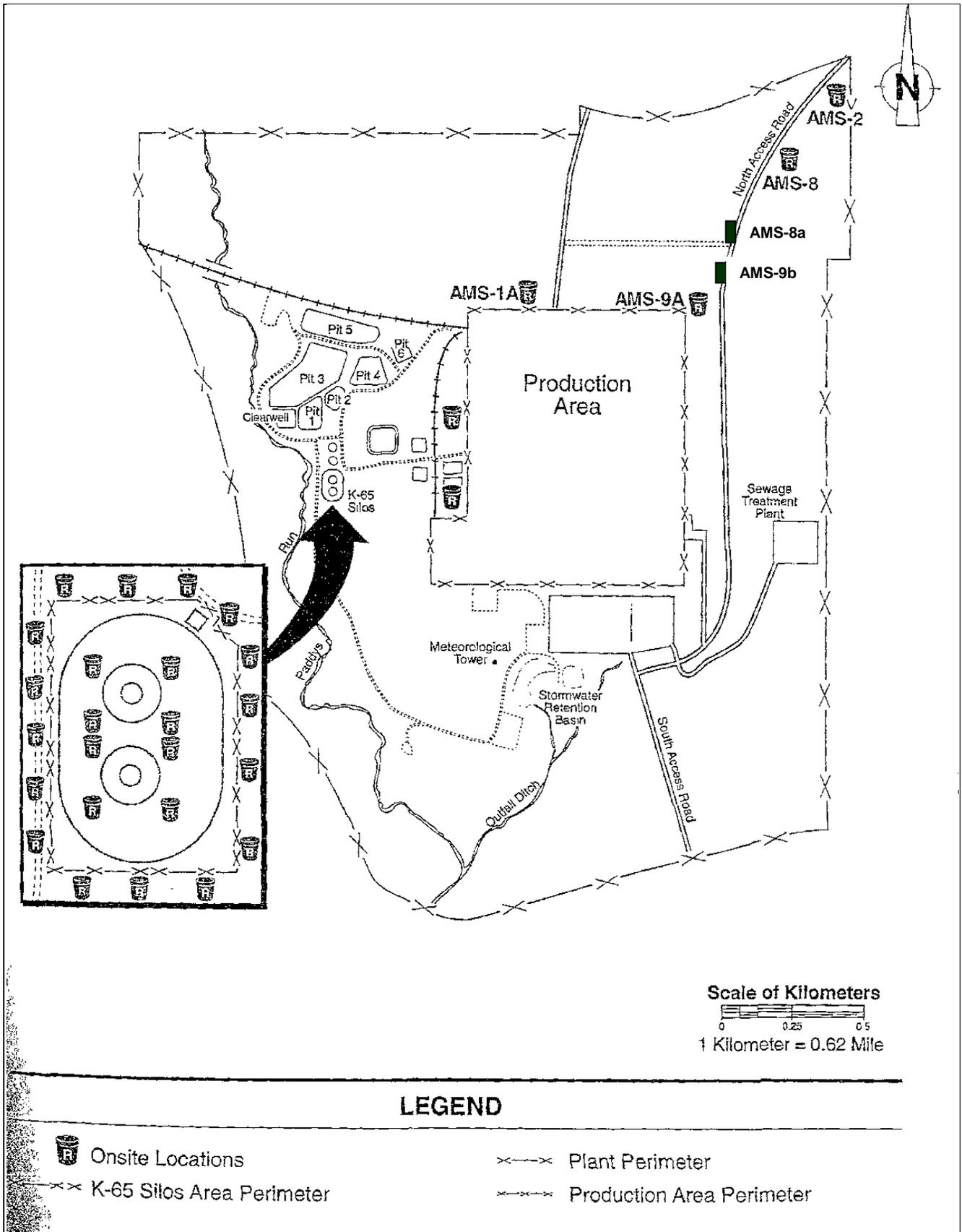


Figure 4-9. Onsite radon monitoring locations (Environmental Report FERMCO 1995, p. 157).

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. 11). 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 one order of magnitude occurred as a result of the bentonite. However, the bentonite had provided only a temporary reduction. Dose rates slowly increased again, and in 2002 monitoring showed an increase of about a factor of four in external radiation.

The only data found from before 1975 (Klein 1963, p. 2) 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, p. 2) 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 (2014) 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 (1976 Environmental Report Boback, Ross, and Fuchs 1977, p. 62). Table 4-19 is a summary of the maximum measured dose rates at the monitoring stations for 1976 through 1995. Figure 4-10 shows the locations of the fence line monitoring stations.

Since 1996, FMPC expanded its direct radiation monitoring system to measure 32 locations continuously including several onsite monitoring locations. Table 4-20 provides summary level information about direct radiation measurements for 1996 through 2005. Figure 4-11 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.

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

Year	Maximum measured dose at TLD stations										
	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.

b. NA = not available.

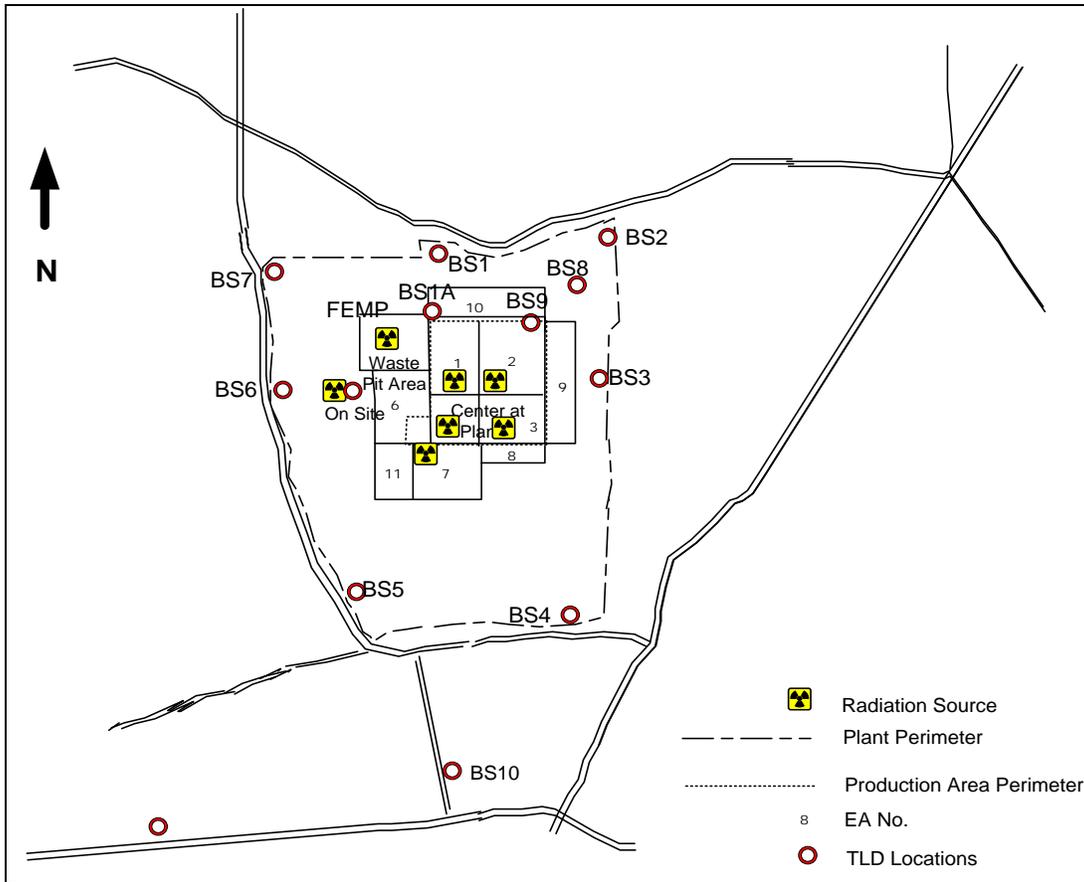


Figure 4-10 Fenceline TLD locations, 1976 to 1995 (Environmental Reports FERMCO 1993 to 1996).

Table 4-20. TLD measurement summary (mrem/hr).^a

Year	TLD monitoring results						
	Fence line		Onsite		Offsite		Background average
	Min.	Max.	Min. (Health and Safety Bldg)	Max. (K-65 silos fence line)	Min.	Max.	
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.

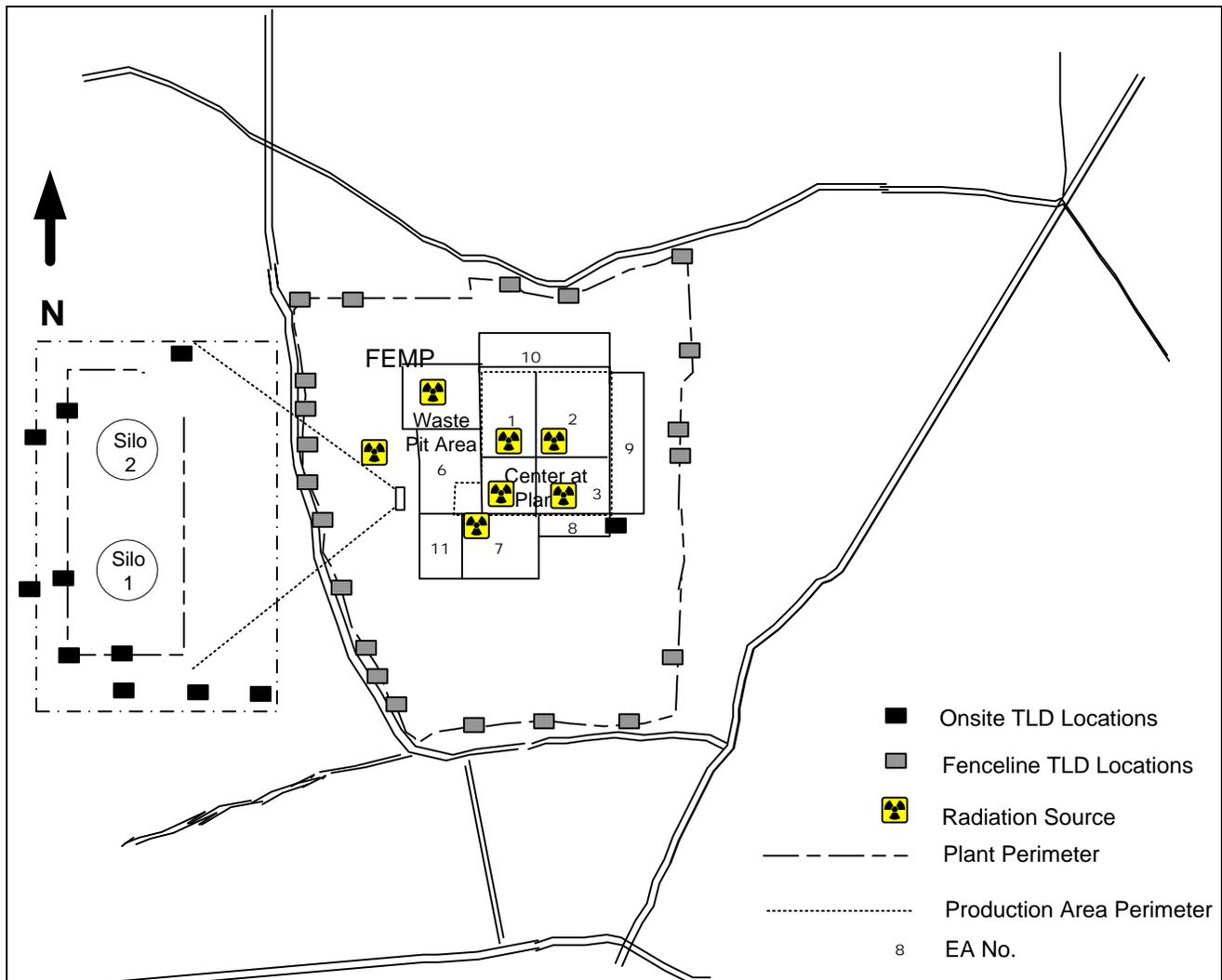


Figure 4-11. TLD monitoring locations, 1996 to 2005 (Environmental Report Fluor Fernald 2004).

4.5.2 External Dose Contribution from Waste Pits

Figure 4-12 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 4-13 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 is 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 typically is 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 waste pits) using the measured dose rate data in Tables 4-19 and 4-20 and the estimated dose rate profile in Figure 4-13.

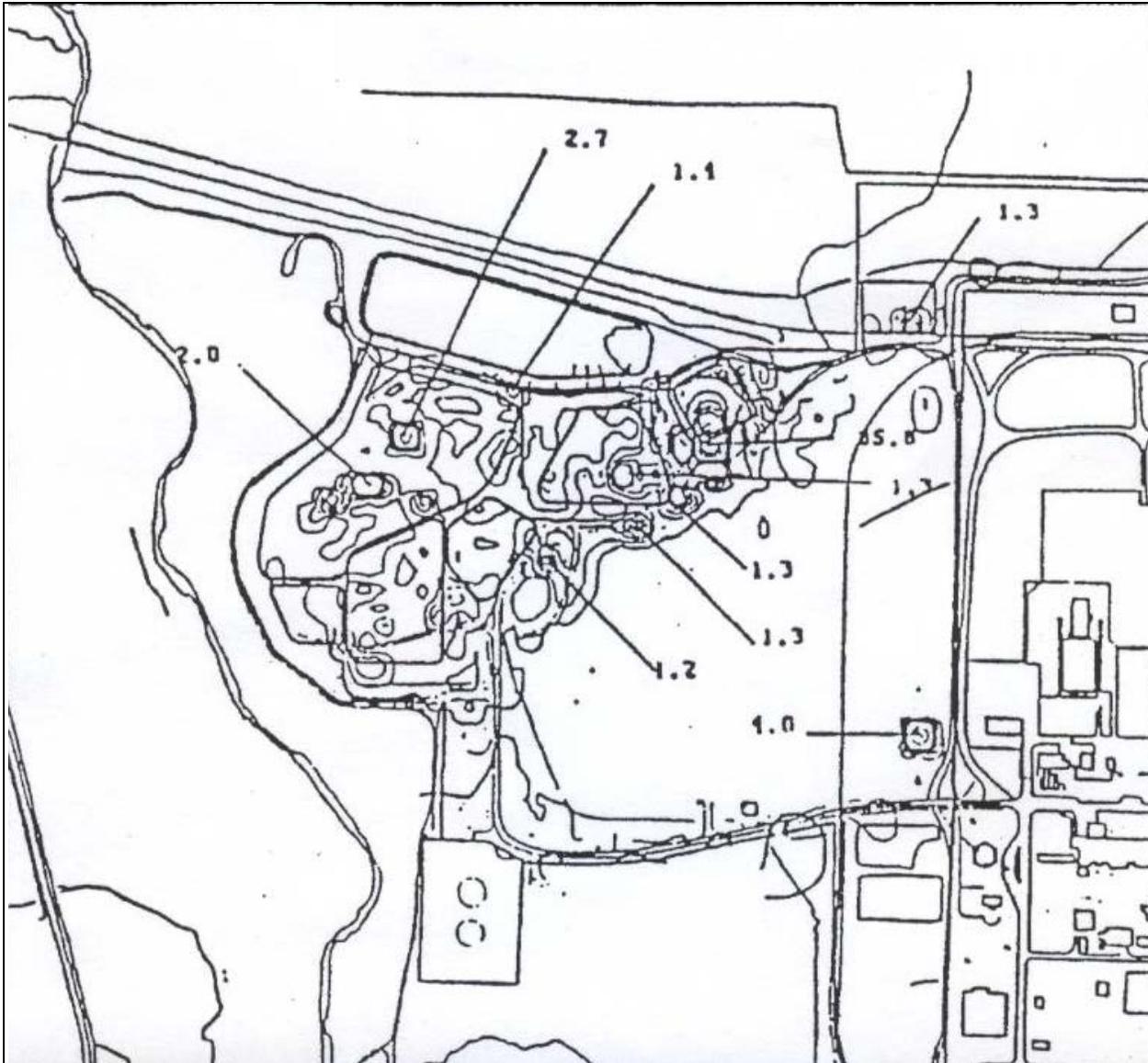


Figure 4-12. Beta/gamma dose rate (mrad/hr) contours in waste pit area (DOE 1994a, SRDB Ref ID 3574, p. 1213).

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 NCRP 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, SRDB Ref ID 3574, p. 23).

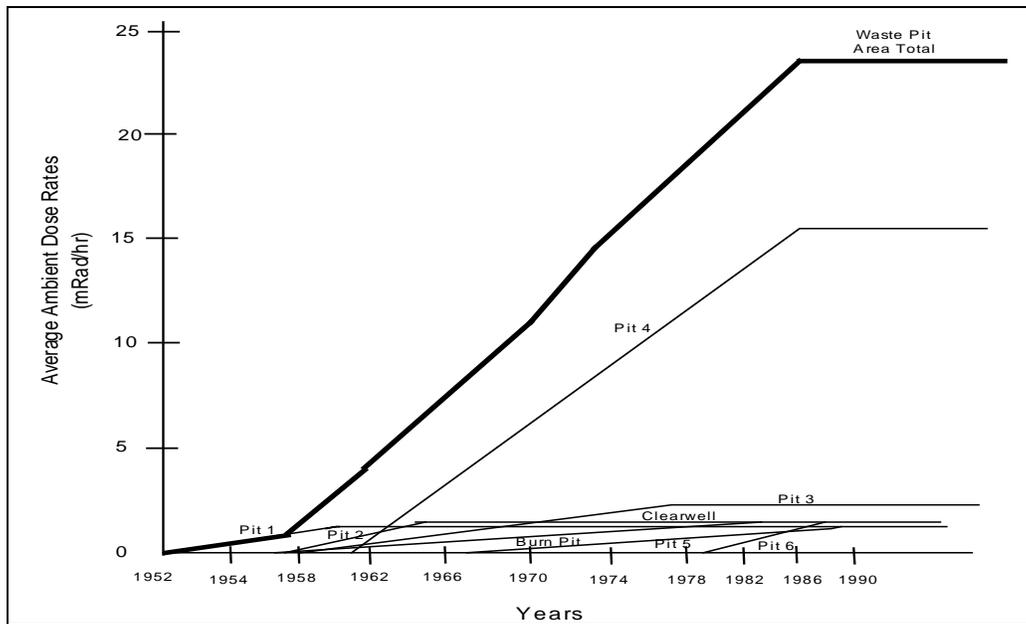


Figure 4-13. Waste pit dose rate profiles.

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

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

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_{EAi} = distance from K-65 silos to EA_i

BSj = measured dose rate at fence line TLD stations [j is 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.]

d_{BSj} = distance from TLD stations to production plants

d_{EAj} = 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_{EAk} = distance from waste pit area to EA_i

ORAUT (2014) documents Equation 4-12 results. These results were compared to the direct dose rate contour maps (Figures 4-14 and 4-15) 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.

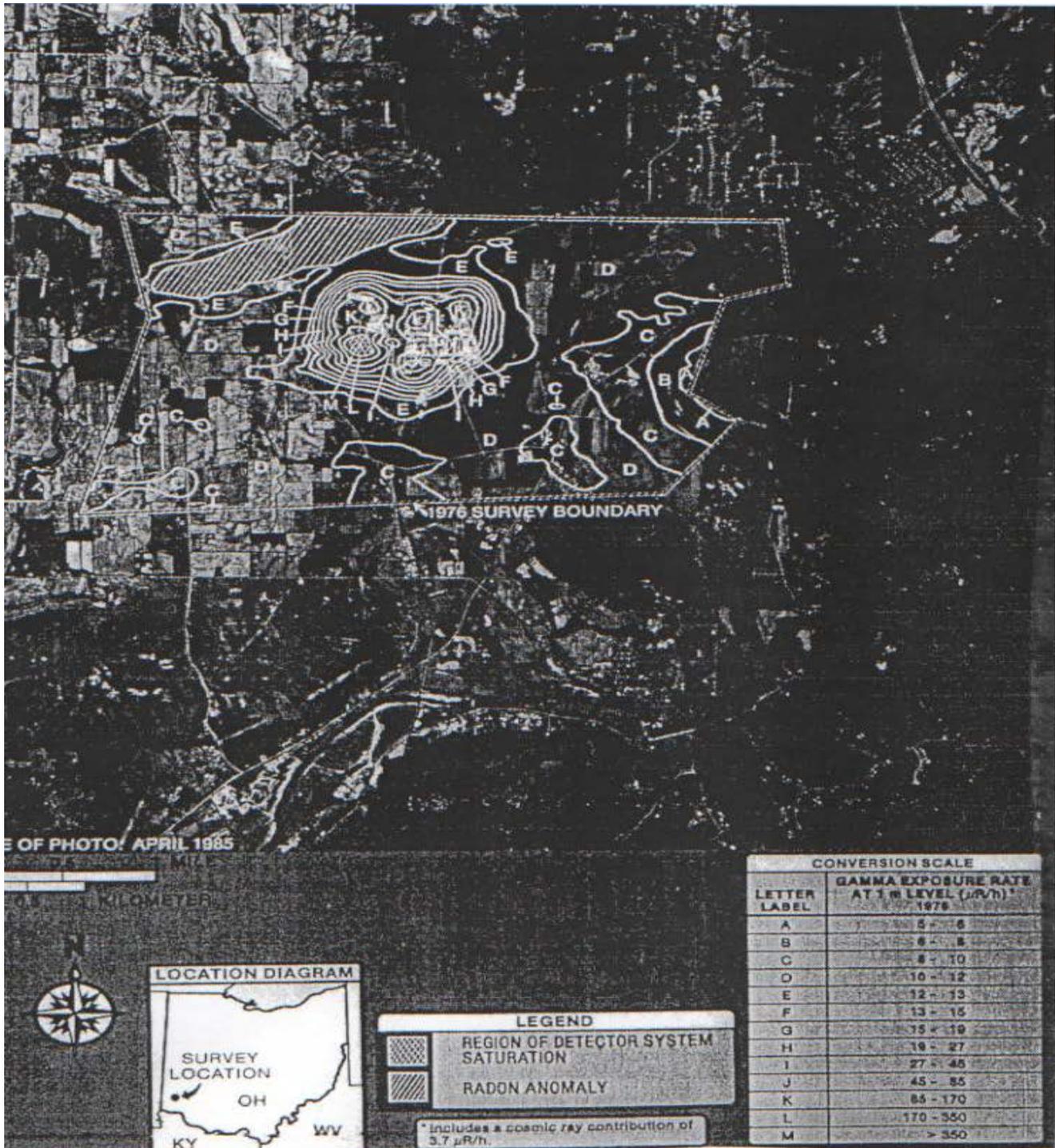


Figure 4-14. 1976 exposure rate ($\mu\text{R/hr}$) contour map of FMPC survey area (Shipman 1985, p. 18).

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.

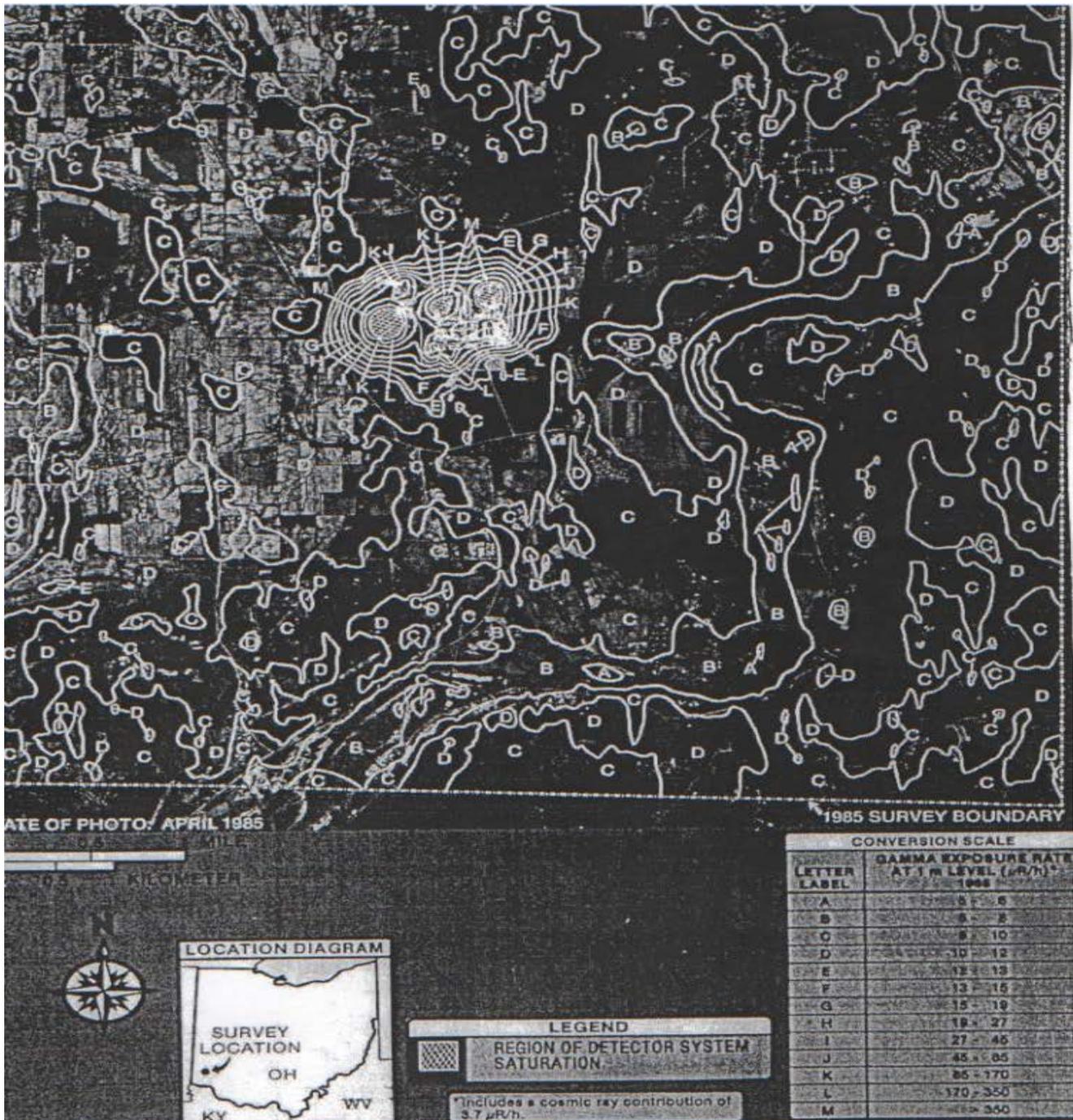


Figure 4-15. 1985 exposure rate ($\mu\text{R/hr}$) contour map of FMPC survey area (Shipman 1985, p. 15).

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

Figure 4-16 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 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, p. 2). The source strength increased to

Table 4-22. Comparison of estimated dose rates in EAs and aerial survey results.

Exposure areas	Estimated dose rates compared to aerial surveys (mrem/hr)					
	1976		1977	1985		1986
	Eq. 4-12	Aerial survey ^a	Eq. 4-12	Eq. 4-12	Aerial survey ^a	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 4-14 and 4-15.

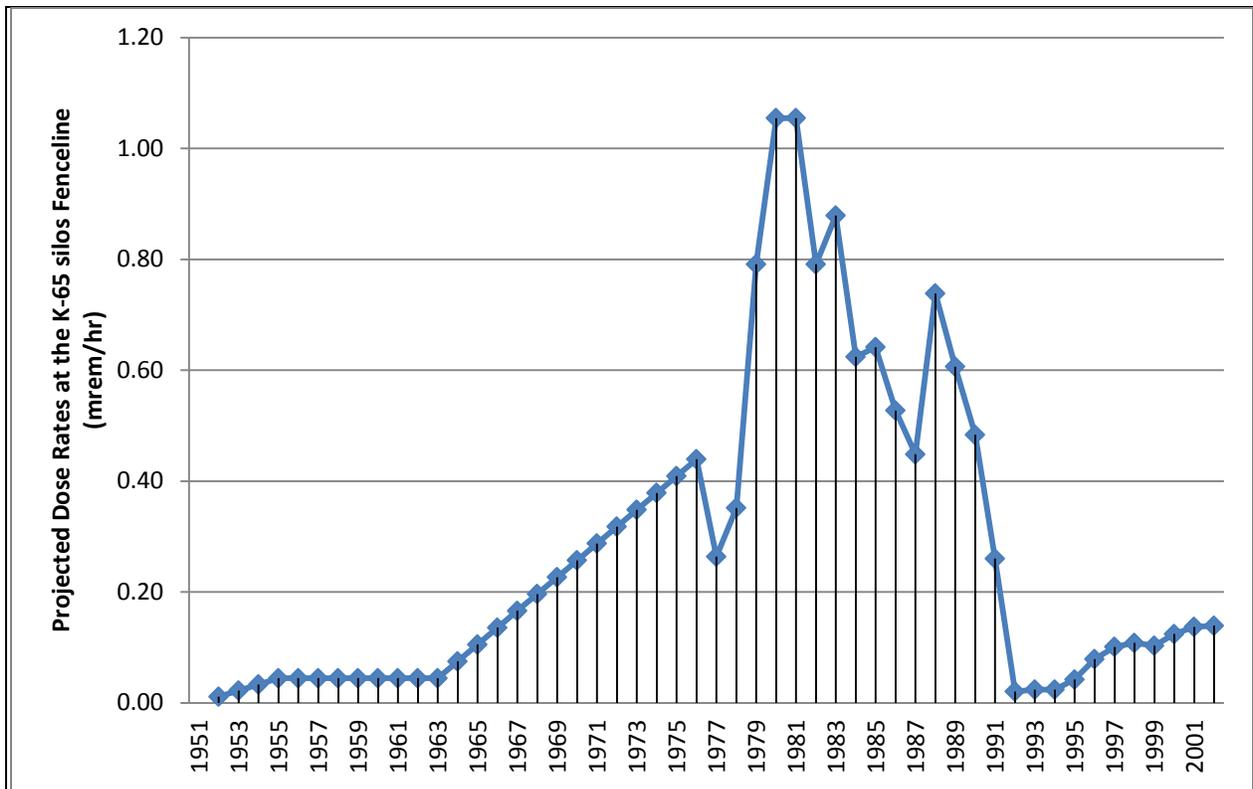


Figure 4-16. Projected K-65 silos fenceline dose rates histogram.

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. 11). 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 4-16 shows that the dose rates progressively increased from 1952 to a “measurable” level in 1955. The “measurable” level as measured by Juno survey meter in 1963 was interpreted to be an 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 silo 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 until recently. Figure 4-16 forms the basis for the estimation of the radiation levels in each EA due to the K-65 silos for 1952 to 1976. Since then, the 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 (2014).

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 quantities 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, and 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% site systems capacity. The site production capacity 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 site production rate increased to 3,700 MTU (3 times the 1979 level), which is about 74% of full production capacity. The onsite ambient direct radiation dose rates for the years before 1976 were estimated by the ratio of the systems capacity to the corresponding measured dose rate in the 1980s. Table 4-23 lists the resulting onsite ambient dose rates. Figure 4-17 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 (2014).

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.

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 4-14 and 4-15, 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 is addressed via references integrated into the References section of this document.

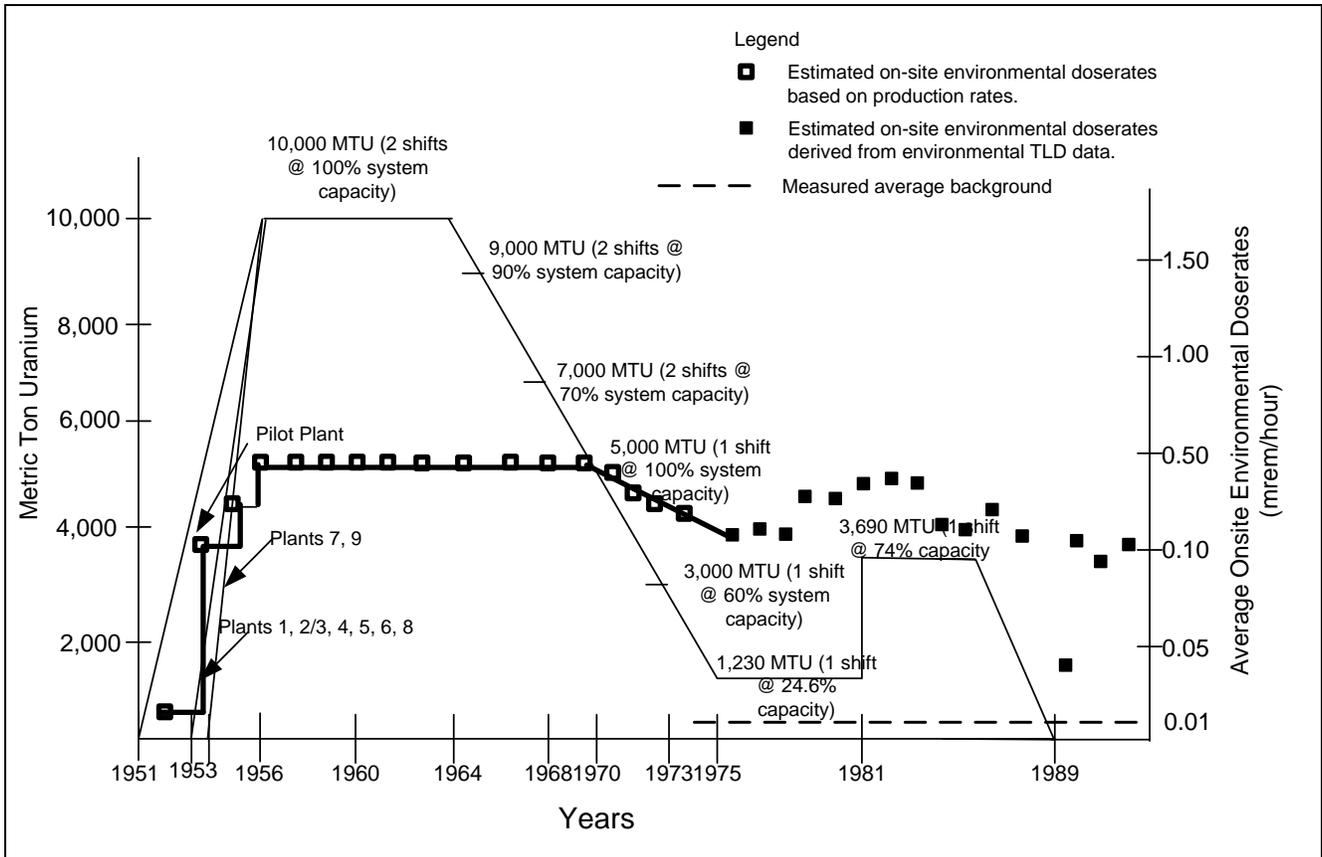


Figure 4-17. Projected site wide direct radiation dose rates based on metal production rate.

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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

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 reports the yearly maximum and mean exposures in terms of 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 Q-11 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.

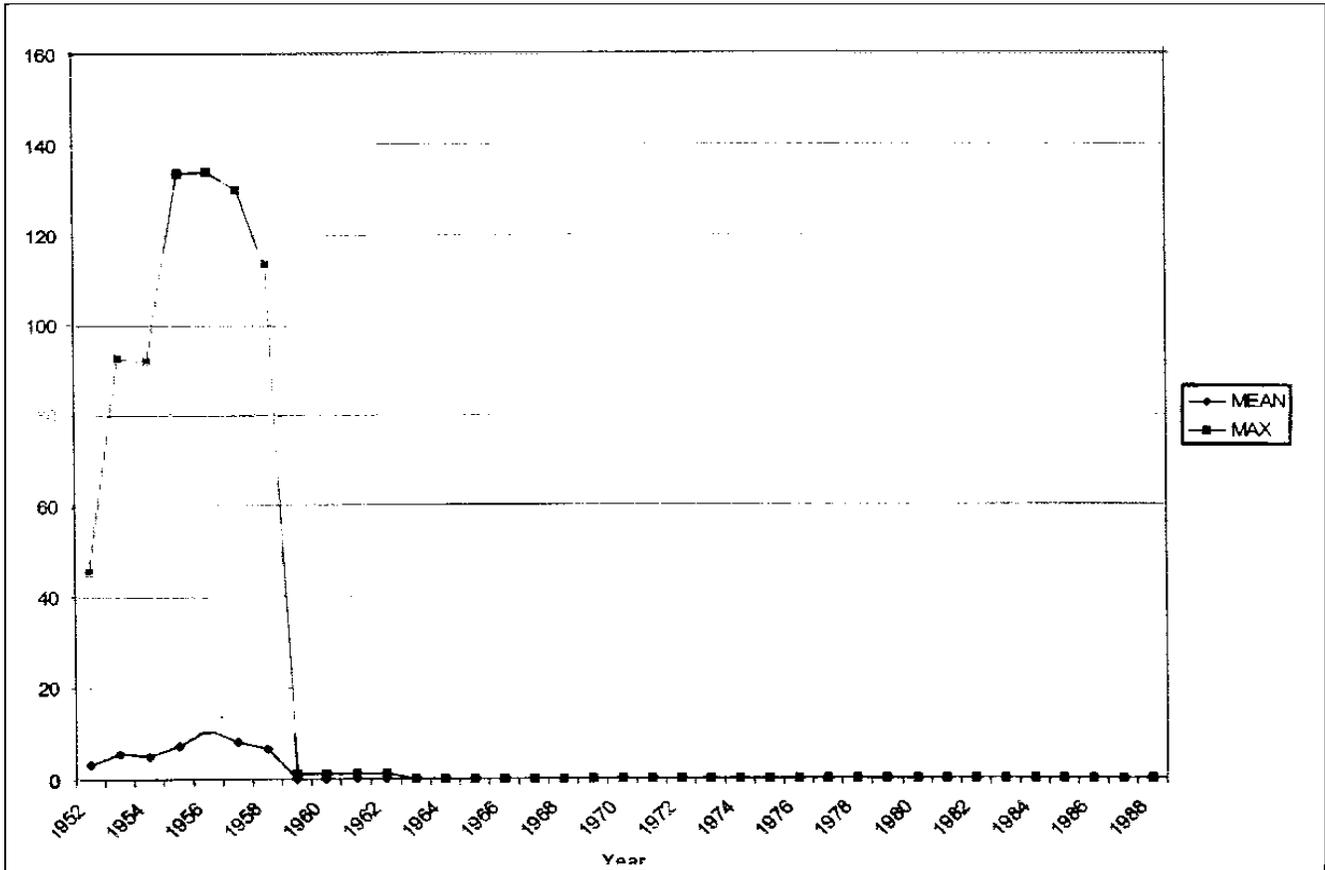


Figure A-1. Mean and highest yearly worker radon exposure estimates due to Q-11 silos (Pinney et al. 2004, p. 61, Figure 13).

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

$$C_{EA} = R_{Q-11} \left(\frac{X}{Q} \right)_d f(RF) \quad (A-1)$$

where

- C_{EA} = radionuclide concentration in EA (Ci/m^3)
- R_{Q-11} = radionuclide release rate from source Q-11 silos (Ci/yr)
- $(X/Q)_d$ = 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 ($<10 \mu\text{m}$)

ATTACHMENT A
INCORPORATION OF Q-11 SILOS AS A RADON RELEASE SOURCE, 1952 TO 1958 (continued)

Rearrangement of Equation A-1 yields:

$$R_{Q-11} = C_{EA} \times \left[\left(\frac{X}{Q} \right)_d f(RF) \right]^{-1} \quad (A-2)$$

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:

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

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

$$R_{Q-11} = \frac{D}{0.0824} \times \left[\left(\frac{X}{Q} \right)_d f(RF) \right]^{-1} \quad (A-4)$$

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

$$R_{Q-11} = \frac{10^3 \text{ L/m}^3 \times (3.154 \times 10^7 \text{ s/yr})}{10^{12} \text{ pCi/Ci}} \times \frac{D}{0.0824} \text{ (pCi/L)} \times \left[\left(\frac{X}{Q} \right)_d \text{ (s/m}^3) f(RF) \right]^{-1} \quad (A-5)$$

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

$$R_{Q-11} = (0.383 \times D) \times \left[\left(\frac{X}{Q} \right)_d f(RF) \right]^{-1} \quad (A-6)$$

where

- R_{Q-11} = annual radon release rate (Ci/yr)
- D = dose (WLM)
- $(X/Q)_d$ = atmospheric dilution factor at distance d (m) from source (d is a close-in distance of 25 m, and the $(X/Q)_{25m} = 2.09 \times 10^{-3} \text{ s/m}^3$ 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_{Q-11} 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_{Q-11} from Equation A-6 was added to the K-65 source term as shown in Figure 4-5.

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

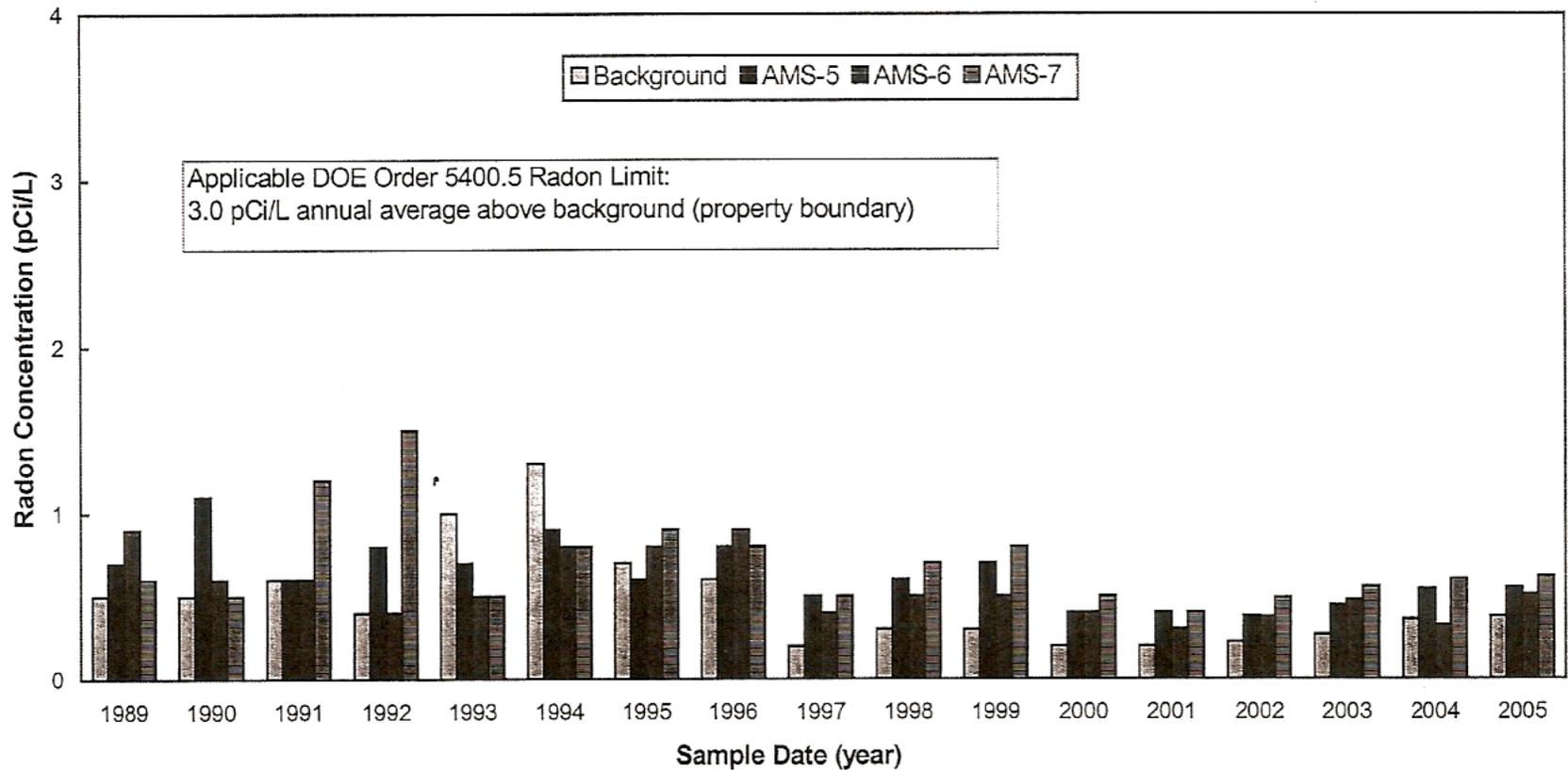
Table B-1. Wind rose data and derived 95th-percentile values for environmental radionuclide concentration estimates (1991 to 2001 Environmental Reports).

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



Note: The data from 1989 through 1996 are based on the alpha track-etch detectors, and the data from 1997 through 2005 are based on the continuous radon monitors.

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