Site Profile for General Electric Co. (Ohio)

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ACRONYMS AND ABBREVIATIONS

This document uses a form of scientific notation (e.g., 1.2E−3, meaning 1.2 times 10 to the negative third power).

AEC	U.S. Atomic Energy Commission
ANP	Aircraft Nuclear Propulsion
ANPD	Aircraft Nuclear Propulsion Department
ATR 1	Advanced Test Reactor 1
AWE	Atomic Weapons Employer
Bq	becquerel
cm	centimeter
Cs	cesium
d	day
DCC	dose conversion coefficient
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
FMPC	Feed Materials Production Center
ft	foot
g	gram
gal	gallon
GE	General Electric Company
HEPA	high-efficiency particulate air
hr	hour
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
in.	inch
keV	kiloelectron-volt, 1,000 electron-volts
L	liter
LOD	limit of detection
m	meter
M	moderate (absorption type)
MDC	minimum detectable concentration
MFPs	mixed fission products
mg	milligram
mi	mile
mL	milliliter
mR	milliroentgen
mrem	millirem

NIOSH	National Institute for Occupational Safety and Health
Np	neptunium
NTA	nuclear track emulsion, Type A
ORAU	Oak Ridge Associated Universities
ORAUT	ORAU Team
PIC	pocket ionization chamber
Pu	plutonium
Ra	radium
S	slow (absorption type)
SEC	Special Exposure Cohort
Sr	strontium
SRDB Ref ID	Site Research Database Reference Identification (number)
t	ton
TBD	technical basis document
Tc	technetium
Th	thorium
U	uranium
U.S.C.	*United States Code*
USAF	U.S. Air Force
wk	week
yr	year
µg	microgram
§	section or sections

1.0 INTRODUCTION

Technical basis documents (TBDs) 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 U.S. Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies), such as changing scientific understanding of operations, processes, or procedures involving radioactive materials. These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of individual dose reconstructions under Part B of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA).

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

Under EEOICPA, a Part B cancer claim for benefits must be based on an energy employee’s eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility’s designated time period and location (i.e., a “covered employee with cancer”). After DOL determines that a claim meets the eligibility requirements under Part B of EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and the types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. 42 U.S.C. § 7384l(11). Also, under EEOICPA, the types of exposure to be included in dose reconstructions for DOE employees are those radiation exposures incurred in the performance of duty. As such, NIOSH includes all radiation exposures received as a condition of employment at DOE facilities in its dose reconstructions for covered employees, which may include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. This is because NIOSH does not determine the fraction of total measured radiation exposure at a DOE facility that is contributed by the Naval Nuclear Propulsion Program at the DOE facility during a specified period of time for inclusion in dose reconstruction.

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

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

1.1 PURPOSE

This site profile provides methods for dose reconstruction for the General Electric Co. (GE) site in Evendale, Ohio. The site has had many names as detailed below; for convenience this document uses GE Ohio.

1.2 SCOPE

Section 1.3 provides information on classes of GE Ohio workers that have been added to the SEC. Section 2.0 describes facilities and operations at the site. Sections 3.0 to 6.0 discuss occupational medical, environmental, internal, and external dose, respectively. Sections 7.0 and 8.0 discuss internal dosimetry co-exposure data and external dosimetry co-exposure data, respectively.

1.3 SPECIAL EXPOSURE COHORT

The Secretary of the U.S. Department of Health and Human Services has designated a class of GE Ohio workers to the SEC based on the findings and recommendations of NIOSH and the Advisory Board on Radiation and Worker Health:

January 1, 1961, through June 30, 1970

> *All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at General Electric Co. in Evendale, Ohio, from January 1, 1961 through June 30, 1970, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees included in the Special Exposure Cohort* [Sebelius 2011, p. 3].
>
> *NIOSH lacks sufficient information, which includes biological monitoring data, sufficient air monitoring information, and sufficient process and radiological source information, to allow it to estimate with sufficient accuracy the potential internal and external exposures to uranium, thorium, and fission product radionuclides for individuals employed at General Electric Company (Ohio) during the period from January 1, 1961 through June 30, 1970* [Sebelius 2011, p. 4].

Although NIOSH found that it is not possible to completely reconstruct radiation doses for the proposed class, NIOSH intends to use any internal and external monitoring data that might become available (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures) for an individual claim. Therefore, dose reconstructions for workers at GE Ohio who do not qualify for inclusion in the SEC during the SEC period from January 1, 1961, through June 30, 1970, may be performed using these data as appropriate [NIOSH 2011a].

2.0 SITE DESCRIPTION

The GE Ohio site was originally called the “GE Lockland” plant because much of the property was in the town of Lockland, Ohio, but it gradually became known as the “GE Evendale” plant after the village of Evendale incorporated in 1952 [Kennedy 2019, p. 5]. This site has also been known as GE Cincinnati and Air Force Plant 36 [NIOSH 2011a, p. 32]. The U.S. Air Force (USAF) owned the 68-acre Air Force Plant 36 property from October 1948 to June 1989 [Ohio Environmental Protection Agency, no date]. The GE Ohio site was contiguous on its north side with the facilities of the GE Aircraft Engine Business Group where commercial and military jet engines were manufactured. This 1.5-mi-long industrial complex is adjacent to Interstate 75 and is 12 mi north of downtown Cincinnati [GE 1986a, pp. 7, 10]. Figure 2-1 is an aerial view of the GE site in Evendale, Ohio. The Aircraft

Nuclear Propulsion (ANP) program was housed in Air Force Plant 36, delineated by the area bounded by the dashed line. The GE Ohio site (USAF owned part of the facility) is the DOE covered facility from January 1, 1961, through June 30, 1970. The major mission of the GE Ohio site was to build aircraft engines. The U.S. Atomic Energy Commission (AEC) used this facility to work with a variety of radioactive materials including uranium and thorium. This facility was also involved in the refining or fabrication of beryllium or beryllium oxide. The GE Ohio site covered period ends on June 30, 1970 [DOE, no date a].

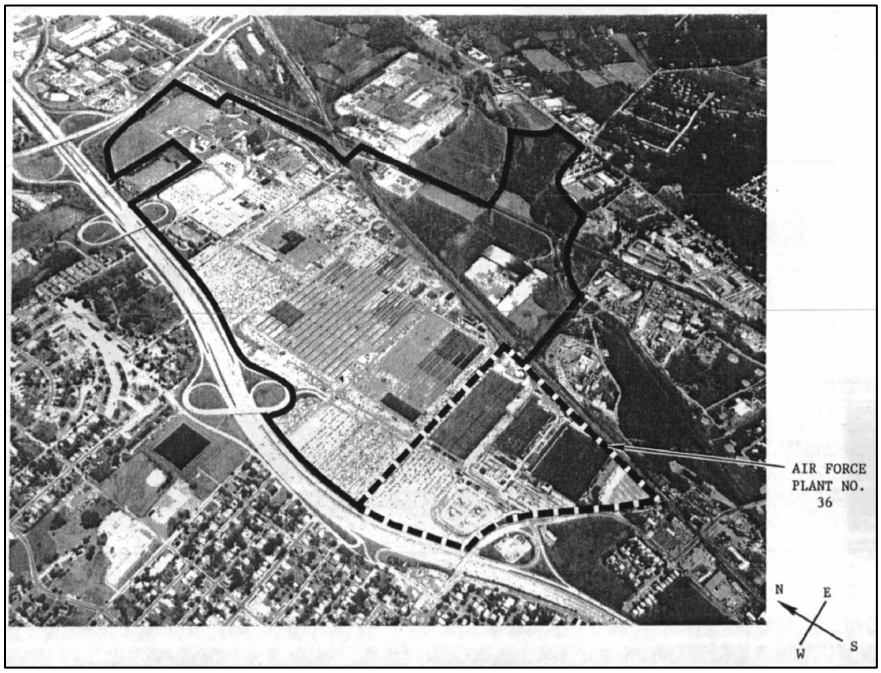

Figure 2-1. Aerial view of the GE Ohio site [GE 1984, p. 10].

DOE considered the GE Ohio site for inclusion under the Formerly Utilized Sites Remedial Action Program, but it was eliminated from consideration [DOE, no date b].

The GE Ohio plant had three primary buildings, B, C, and D. Figure 2-2 shows the layout of these buildings with labels for the C-West portion of Building C and the Building D laboratory area.

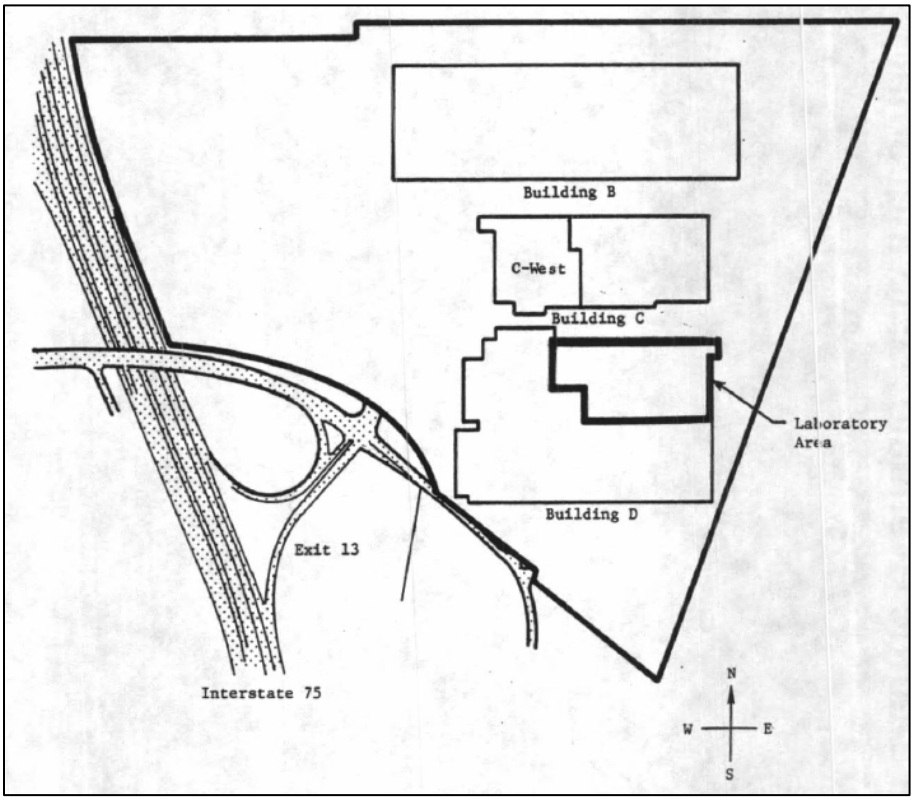

Figure 2-2 Site layout [GE 1986a, p. 12].

The Defense Mobilization Board built Building D during World War II (around 1941) as an aluminum foundry for the Wright Aeronautical Engine Plant. After the war, Building D remained idle until 1951 [GE 1986a, p. 10]. In 1951, GE became a prime contractor to the AEC and participated in the ANP program. This program was a joint venture between USAF and the AEC to develop a nuclear-powered aircraft. The AEC selected Building D as the operational building for this program [Hoffer 1979, p. 4]. Before anyone moved into Building D, GE gutted the interior then redesigned and reconstructed it to handle radioactive materials safely and to meet the needs of the ANP program. The AEC established the Lockland Area Office and moved into a series of offices in Building D to administer the ANP contract. Figure 2-3 shows a layout of Building D [GE 1986a, p. 14].

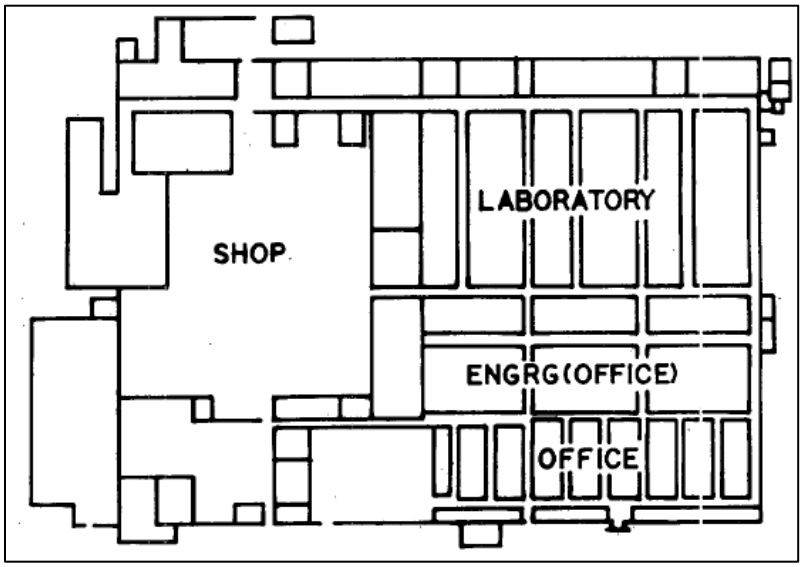

Figure 2-3. Main floor layout, Building D [GE 1986a, p. 14].

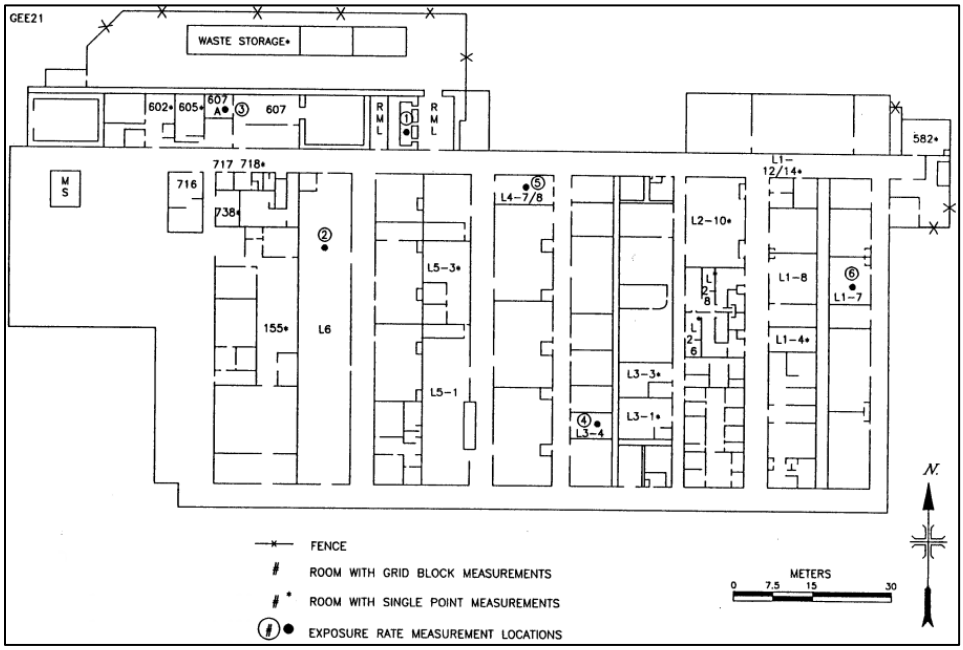

Figure 2-4. Main floor layout, Building D indicating locations in which radioactive materials were processed or stored [Murphy and Luck 1988, p. 22].

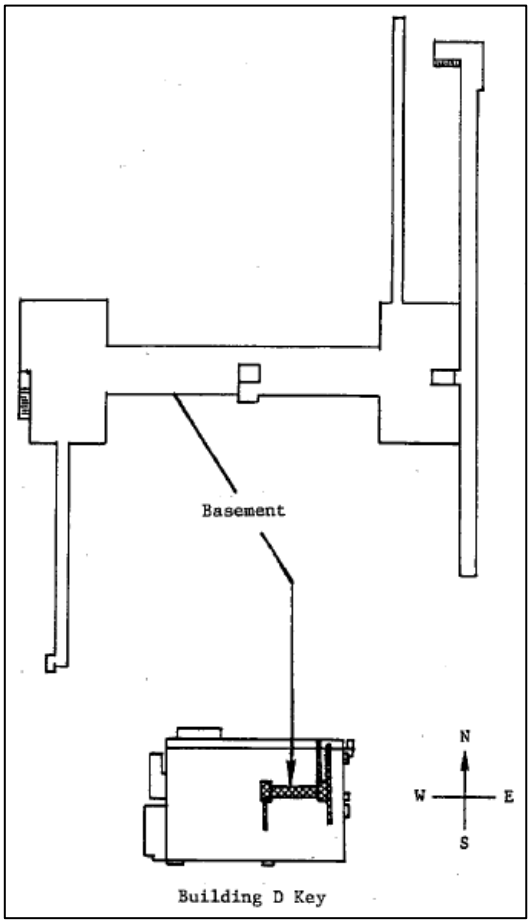

Figure 2-5. Basement layout, Building D [GE 1986a, p. 15].

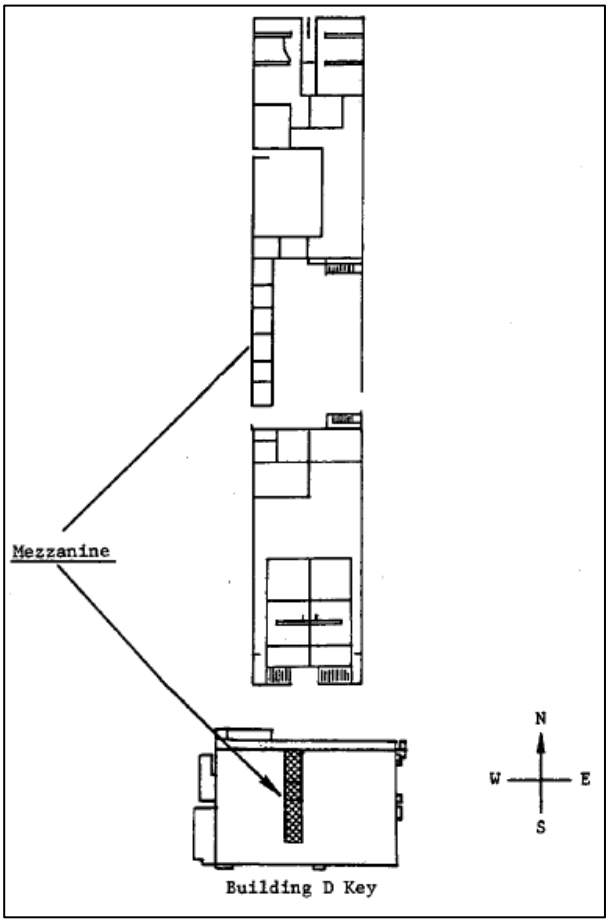

Figure 2-6. Mezzanine layout, Building D [GE 1986a, p. 16].

The Building D laboratory had many elaborate engineering and safety features to ensure safe handling of radioactive materials. It was virtually isolated from the remainder of the building with its own utility services and ventilation system. The most prominent of the safety features was the central exhaust system that maintained negative differential air pressure relative to the outside of the building. Work areas where radioactive materials were handled were kept at a negative differential air pressure with respect to the interior of the building. Gloveboxes were used for mixing and handling materials until the physical state was such that there would be no potential for the spread of contamination. Gloveboxes were maintained at a negative differential pressure relative to the work area. The ventilation systems in all laboratories, rooms, or radioactive and toxic materials handling areas used high-efficiency particulate air (HEPA) filters [GE 1986a, pp. 13, 18].

The liquid waste drain system for the laboratory area was an extensive collecting system that flowed into retention tanks in the basement of the laboratory. This waste was assayed for radioactive content and, if within permissible limits, was discharged into the sanitary system outside the northeast corner of Building D. The laboratory design focused on maximizing safety to workers and the environment. Criticality safety was an overriding factor in many operations because moderator and fissionable materials were mixed in small batches. It was necessary to use engineering and administrative control measures to ensure safe operating conditions [GE 1986a, p. 18].

Building C began as a foundry around 1941 for the Defense Mobilization Board as part of the Wright Aeronautical Engine Plant. After World War II, Building C remained idle until 1954 when GE partially refurbished and used it for storage, manufacturing, and other purposes for jet engine production. Building C is approximately 660 ft long by 280 ft wide. The building had a structural steel frame supporting a concrete roof. The walls were masonry. The main floor was poured concrete over earth and a partial poured concrete basement [GE 1984, p. 9]. Figure 2-7 shows the main floor of Building C-West.

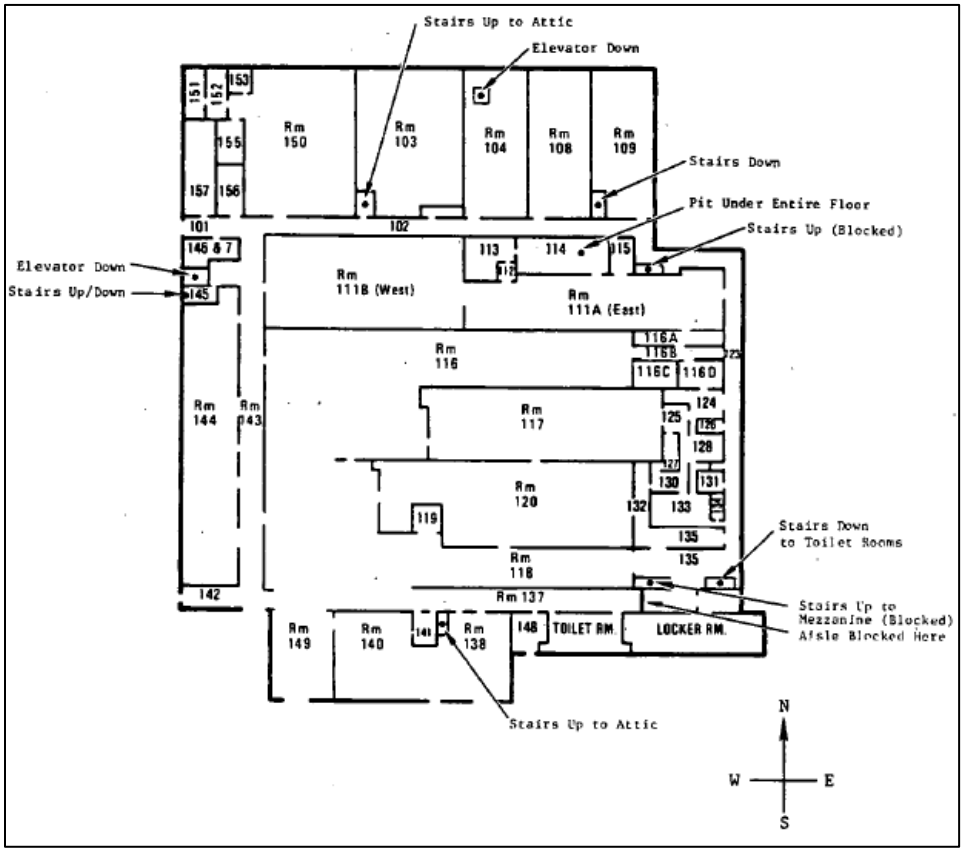

Figure 2-7. Main floor layout, Building C-West [GE 1984, p. 12].

In 1956, the ANP program needed more space to support operations in Building D, and GE allocated 42% of the west end of Building C and named it Building C-West. GE divided this part of the building with a floor-to-ceiling wall and completely renovated the area. It housed a pilot production line for ceramic nuclear fuel elements. These facilities were for handling large quantities of radioactive materials with safety to workers and the environment a critical part of design [Hoffer 1979, p. 4; GE 1984, p. 15]. Building C-West was approximately 280 ft long by 280 ft wide [GE 1984, p. 9]. The interior of Building C-West was kept at a negative differential pressure relative to the outside, and gloveboxes had negative differential pressure relative to the work areas. The ventilation air system was single-pass, HEPA-filtered, and exhausted to the outside. The gloveboxes were for mixing and handling materials until the physical state eliminated the potential spread of contamination. Criticality safety was the overriding factor because a moderator and fissionable material were mixed in small batches [GE 1986a, pp. 13, 18].

Figure 2-8 shows the basement of Building C-West with a vault, a storage area, a transfer room, and west and north basement corridors, and Figure 2-9 shows the mezzanine of Building C-West with an office area and a quality control area.

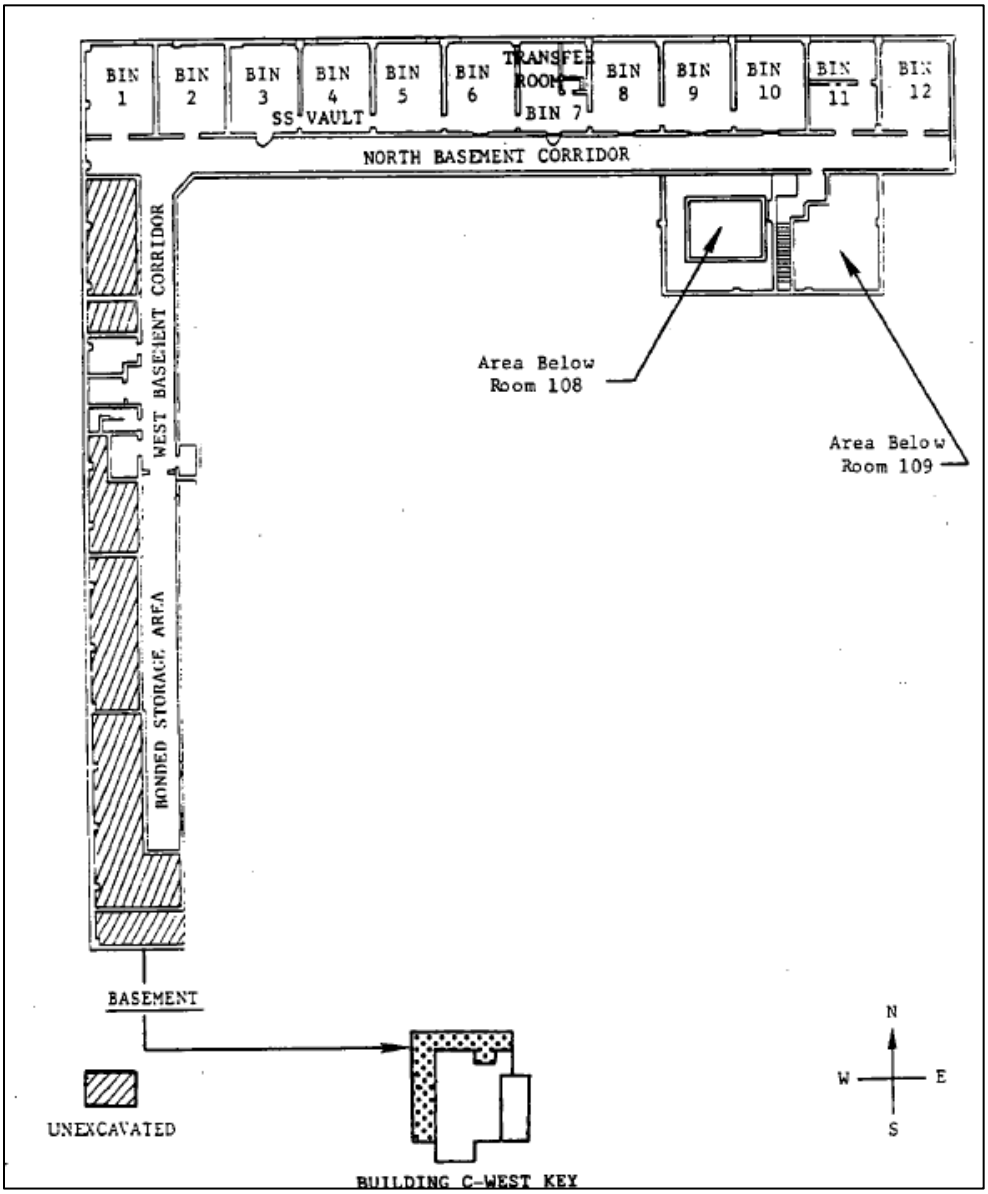

Figure 2-8. Basement layout, Building C-West [GE 1984, p. 13].

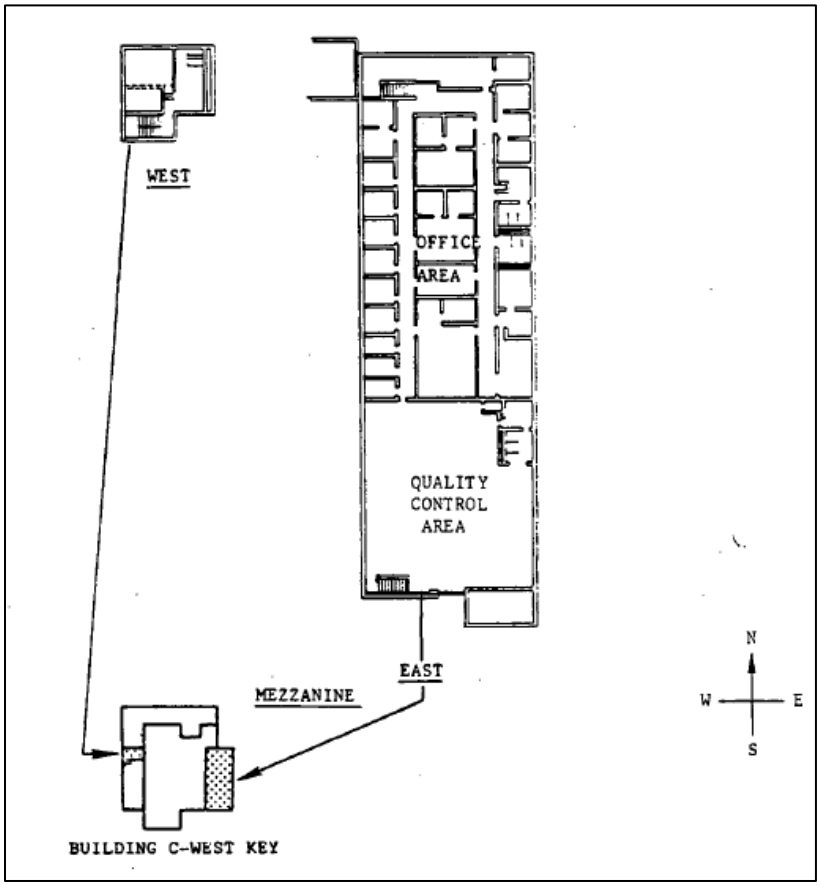

Figure 2-9. Mezzanine layout, Building C-West [GE 1984, p. 14].

In 1957, GE built radioactive waste storage facilities to upgrade the centralized storage and control of radioactive waste before shipment to an approved burial site [GE 1986b, pp. 7, 10]. These facilities were on the north end of Building D and consisted of a fenced concrete storage pad on the west next to a decontamination building in the middle and a clean drum storage area east of the pad [Murphy and Luck 1988, p. 39; GE 1986b, pp. 7, 10]. Figure 2-10 shows the waste storage area. A heated building, mainly for storage of liquid wastes during low temperatures, was next to the clean drum storage on the east pad. The west pad was an open area for storage of 55-gal drums filled with contaminated wastes. On a few occasions (dates unknown), thorium liquid leakage from the drums contaminated the concrete floor [GE 1986b, pp. 7, 10].

In 1961, the ANP Program was cancelled due to competition for funding and scientific manpower with other defense research projects including the intercontinental ballistic missile. However, the AEC leased Building C-West and D from the USAF, and GE continued research as a prime contractor under the Nuclear Materials and Propulsion Operation investigating the use of high-temperature fuel elements and reactor materials [GE 1991, p. 7]. The AEC leased the buildings from USAF [GE 1986a, p. 18]. From 1961 to June 30, 1970, the AEC contract work occupied Buildings C and D and certain other smaller auxiliary structures under a use permit from USAF. The use permit expired, and USAF took back custody on June 30, 1970 [Air Force Plant, no date; GE 1986a, p. 18].

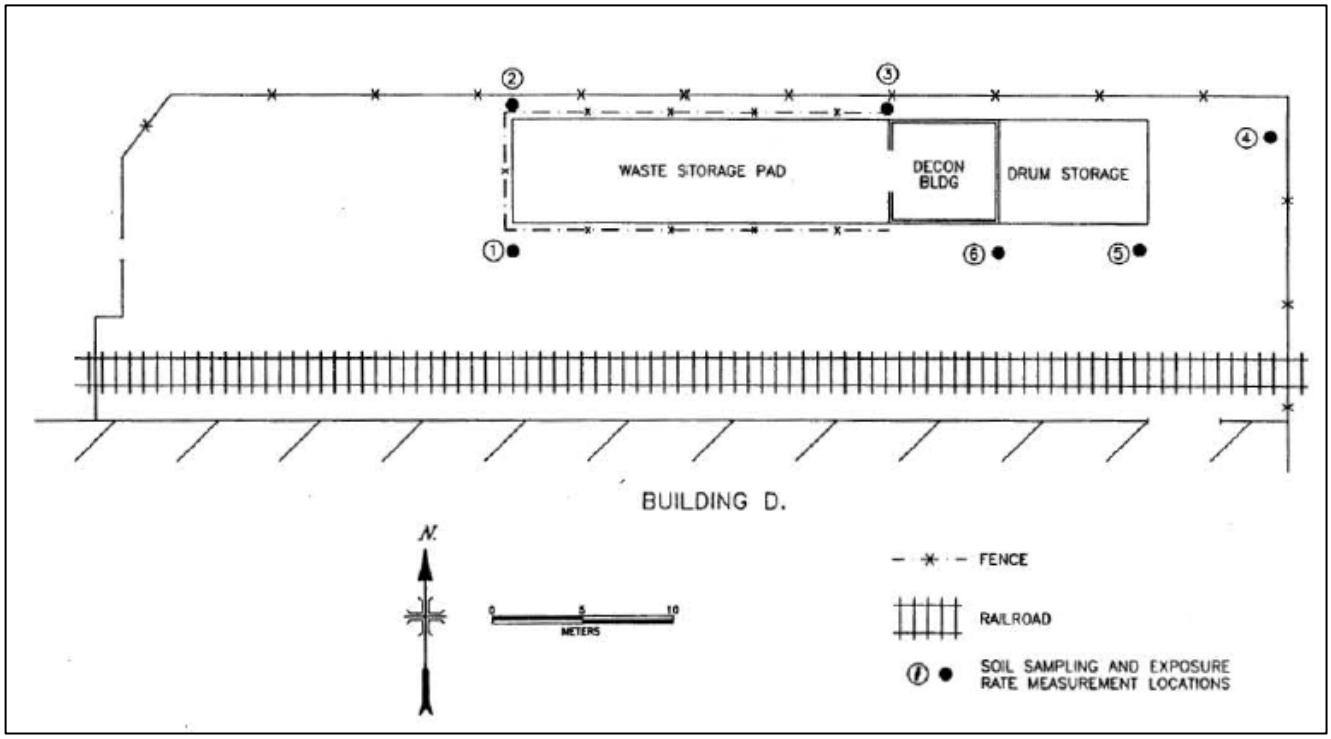

Figure 2-10. Waste storage area, Building D [Murphy and Luck 1988, p. 39].

From 1961 to 1968, the Building D laboratory and Building C-West housed experimental work to support fuel element production for the AEC 710 and 630A reactor development programs. The 710 Program dealt with developing a compact fast-spectrum reactor for use either as an open-cycle hydrogen-cooled system in a nuclear rocket engine or as an inert gas closed-cycle powerplant for space or terrestrial applications. The highly enriched fuel elements for this reactor concept were fabricated in Building C-West. The aim of the 630A Program was to use the ANP reactor technology to develop a gas- or air-cooled powerplant for commercial merchant ships. Some fuel element development work for this project occurred in Building C-West [GE 1986a, p. 18; GE 1984, pp. 15–16].

In a 1962 memorandum, GE submitted a nuclear material draft for 2,939 g of 30% enriched uranium dioxide (UO_2) for the fabrication of in-pile test specimens in the High Temperature Materials Program under Contract AT (40-1)-2847. The memorandum states that the 30% enrichment was particularly suited to the requirements of the proposed experiments [Karl 1962]. Details about unclassified research projects covered under Contract AT (40-1)-2847 are in Rice [1966, pp. 110–130]. However, during the covered period, workers could have handled uranium with varying levels of enrichment from depleted to 93% [GE 1968a; Roth 1962; Oak Ridge Associated Universities (ORAU) Team (ORAUT) 2009].

Facilities for producing 1 t/d of thorium as light oxide (ThO_2) were installed in the Feed Materials Production Center (FMPC) Pilot Plant for startup in 1967. Purified thorium nitrate tetrahydrate ($Th(NO_3)_4$) was converted to light oxide via the sol-gel process. A joint venture with GE Ohio enabled GE Ohio facilities to be used for densification of the light oxide and water fabrication of dense thoria [Kispert 1997, p. 3]. Beginning in 1967, Room 120 of Building C-West housed calcining equipment for high-purity thoria (ThO_2) for the thermal breeder reactor program [GE 1984, p. 16]. About 300 t of thorium were produced as dense thoria for Hanford until 1969 when the program was completed [Kispert 1997, p. 3].

In 1970, FMPC produced 7 t of thorium as dense thoria for Bettis Atomic Power Laboratory in support of the AEC Light Water Breeder Reactor Program. Full-scale operations began in 1972 and continued until completion in 1976. About 400 t Th were produced as thorium oxalate $Th(C_2O_4)$ from French-source $Th(NO_3)_4$ crystals at FMPC, which sent the oxalate product to GE Ohio for firing to dense thoria. GE returned the thoria to FMPC for certification and shipment [Kispert 1997, p. 3].

In 1970, the AEC ended its prime contract with the GE Nuclear Materials and Propulsion Operation. GE became the licensee and performed subcontract work with other AEC contractors. At this time, the AEC vacated the facility and returned it to USAF [Air Force Plant, no date; GE 1986a, p. 18].

2.1 RADIONUCLIDES OF CONCERN

Workers might have received radiation exposures from uranium, thorium, and mixed fission product (MFP) radionuclides during the covered period from January 1, 1961, to June 30, 1970 [NIOSH 2011a]. Subsequent sections discuss specific isotopes of concern.

3.0 OCCUPATIONAL MEDICAL DOSE

External doses from occupational X-ray screening procedures that were provided to the energy employee as a condition of employment and were performed at a covered facility must be included in dose reconstruction. X-rays performed for diagnostic or therapeutic reasons, however, are excluded. Screening X-rays are systematic examinations that are performed on asymptomatic people without history, complaint, physical findings, or physician evaluation. Diagnostic X-rays are careful examinations of people who already have suspicious signs or symptoms of a potential condition, and they are performed after physician evaluation [NIOSH 2010].

NIOSH has found limited information regarding occupational medical dose. According to a 1959 Radiological Health Services procedure, “X-ray pictures of the chest are usually taken as part of the routine medical examination performed by the ANPD [Aircraft Nuclear Propulsion Department]-Evendale Dispensary” [GE 1959, p. 2]. Based on a report of a final survey of Building D, the dispensary seems to have been located within this building [GE 1986a, Table 6-1, p. 68].

The following information will be considered when assigning occupational medical dose:

- For an overestimating approach to assigning X-ray dose, the dose reconstructor should assign annual posterior-anterior chest X-ray doses using ORAUT-OTIB-0006, *Dose Reconstruction from Occupational Medical X-Ray Procedures* [ORAUT 2019], ORAUT-OTIB-0079, *Guidance on Assigning Occupational X-Ray Dose under EEOICPA for X-Rays Administered Off Site* [ORAUT 2017a], and ORAUT-PROC-0061, *Occupational Medical X-Ray Dose Reconstruction* [ORAUT 2017b].

- For a best-estimate approach to assigning X-ray dose, use the information in the Individual Employee Dose Summaries as described in Section 6.3. If there are no Individual Employee Dose Summaries for the employee, then request the Individual Employee Dose Summaries. If the records are unavailable, contact the Principal Scientist for Medical Dosimetry.

- For an underestimate approach, assign no X-ray dose.

4.0 OCCUPATIONAL ENVIRONMENTAL DOSE

This section provides the basis for the internal environmental intakes and onsite ambient dose at GE Ohio. Dose reconstructors may use this information to supplement information in an individual’s dose record.

4.1 INTERNAL ENVIRONMENTAL INTAKES

GE Ohio had an environmental monitoring program as documented in its procedures [GE 1960]. Environmental surveillance activities, as described by procedure, included isokinetic stack effluent sampling, collection of environmental media (soil, water, and vegetation), fallout deposition samples, and air samples from the surrounding community. However, these data are largely unavailable in the appropriate quantity and quality for dose reconstruction. In a response to a set of U.S. Nuclear Regulatory Commission requests for additional information, GE stated that historical environmental effluents were low [Hoffer 1987]. Therefore, the analysis for dose reconstruction developed bounding intake values based on air concentration limits in the GE Ohio *Health and Hygiene Instruction Manual* [GE 1960].

The conceptual model for assessing environmental intakes is a worker submerged in a semi-infinite radioactive cloud with a concentration equal to the corresponding air limit for uranium, thorium, and MFPs. Assumptions include that 100% of the radioactive material was respirable and that the worker inhaled this material at a rate of 1.2 m^3/hr for 2,000 hr/yr. Annual environmental intakes were calculated as:

$$I_{\mathrm{Env},i} = C_{\mathrm{Lim},i} \times V_R \times OF_{Annual} \quad (4\text{-}1)$$

where

$I_{\mathrm{Env},i}$ is annual environmental intake for radionuclide *i* (Bq/yr)
$C_{\mathrm{Lim},i}$ is air concentration limit of radionuclide *i* (Bq/m^3)
V_R is breathing rate (1.2 m^3/hr)
OF_{Annual} is annual occupancy (2,000 hr/yr)

Intakes for uranium and MFPs were based on the community air limit from the GE Ohio *Health and Hygiene Instruction Manual* [GE 1960]. The community air limit was a concentration limit applicable to mobile air samples collected in the offsite area surrounding the plant. Mobile air sampling stations were deployed to locations where maximum downwind air concentrations were expected. A community air limit for thorium was not established. Therefore, thorium intakes were based on the in-plant air limit in the *Health and Hygiene Instruction Manual*. Using the in-plant air values for thorium to derive environmental intake is claimant favorable as the in-plant air limits are higher than the community air limits. Table 4-1 lists annual environmental intakes along with the corresponding air concentration limits.

Table 4-1. Annual environmental intakes.[a]

Nuclide type	Limit type	Limit (dpm/m^3)	Limit (Bq/m^3)	Intake (Bq/yr)
Uranium[b]	Community air[c]	8.900	0.148	356
Thorium[d]	In-plant air[e]	8.800	0.147	352
MFPs[f]	Community air[c]	222.00	3.70	8,880

a. Source: ORAUT [2026].
b. Assign uranium intakes as U-234 and include intakes of recycled uranium contaminants from Table 5-1.
c. Source: GE [1960, Table B].
d. Thorium intakes are assessed as the more favorable of natural or triple separated.
e. Source: GE [1960, Table A].
f. The MFP rate is the Sr-90 intake rate and should be assigned based on the Advanced Test Reactor 1 activity distribution in ORAUT-OTIB-0054 [ORAUT 2015].

MFP intakes are assessed using methodology described in ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses*.

Because the site used gross beta analyses to assess MFP air concentration, Sr-90 was selected as the indicator nuclide in accordance with the OTIB-0054 methodology. Activity fractions relative to the indicator nuclide for the Advanced Test Reactor 1 (ATR 1, Table 7-3a in ORAUT-OTIB-0054) are used to determine activity MFP fractions in gross beta air samples [ORAUT 2015]. ATR 1 was selected based on similarities in fuel characteristics and high burnup levels of the fuels examined as part of the 710 Program [GE 1968b]. Without specific claim information, base intakes on a 10-, 40-, 180-day, or 1-year fuel decay time, whichever is most favorable to the claimant.

Dose reconstructors should assign the annual environmental intakes in Table 4-1 using a constant distribution and use the material absorption types that maximize the dose. Environmental intakes are assigned to unmonitored workers. Environmental intakes are also assigned to monitored individuals if there are nuclides in the environmental suite that have not been assessed though other dose reconstruction methods per ORAUT-OTIB-0060, *Internal Dose Reconstruction* [ORAUT 2025]. Assign uranium intakes as U-234 and also include recycled uranium contaminants based on the activity ratios in Table 5-1. Because air filters were counted by gross alpha, the activity associated with the thorium intake would have been distributed between three long-lived alpha emitters (Th-232, Th-228, and Ra-224) in the decay chain. Unless there is claim-specific information, dose reconstructors should consider both natural and fresh triple-separated thorium as possibilities and assign the more favorable mixture. Natural thorium intakes are composed of equal activities of Th-232, Th-228, and Ra-224 (i.e., an intake of 117 Bq/yr for each alpha emitter). Fresh triple-separated thorium is assessed using activity ratios and guidance in ORAUT-OTIB-0076, *Guiding Reconstruction of Intakes of Thorium Resulting from Nuclear Weapons Programs* [ORAUT 2014].

The *Health and Hygiene Instruction Manual* states that the plant sewage system connected to a public City of Cincinnati system [GE 1960]. The manual instructs ANP Department employees not to dispose of radioactive waste via the Building D contaminated drain system because the system was not designed for disposal. Because Building D connected to this sanitary sewer system, it is reasonable to assume the drinking water came from a municipal source. Additionally, there was emphasis on personnel and environmental safety in the design of the buildings [GE 1984, p. 15; GE 1986a, p. 13] and the controlled liquid drain system was separate and distinct from the storm and sanitary drain systems [GE 1986a, p. 21]. Therefore, radioactive material ingestion via the drinking water pathway is not applicable and ingestion intakes were not assessed.

4.2 ONSITE AMBIENT DOSE

In accordance with GE Ohio's procedures, the environmental monitoring program focused on assessing the airborne pathway via the collection of samples in the surrounding environs: air, water, soil, and deposition on vegetation. Traditional environmental measurements assessing external radiation (e.g., routine exposure rate surveys, fenceline film badges) were not part of this program. Because Buildings C and D were designed to contain and control radioactive material, the airborne pathway was of most concern. A worker would potentially receive external radiation exposure due to submersion in a radioactive cloud from the building air effluent and direct exposure from ground deposition of the same effluent. A screening calculation assessed exposure from the submersion and ground deposition pathways. Within the context of environmental monitoring, a screening calculation typically assesses the significance of an environmental pathway and determines if further monitoring is necessary. Screening uses conservative assumptions and environmental transport parameters. As a result, the estimated exposure is consistent with the bounding nature of the assessment approach in this site profile. The conceptual model for onsite ambient dose at the GE Ohio site was that a worker was standing in a semi-infinite cloud of radioactive material at a concentration equal to the corresponding limit in Table 4-1. The material would have deposited during the period of site operations and uniformly mixed in the top 3 cm of soil.

The annual onsite ambient dose due to submersion was calculated using Equation 4-2:

$$OSA_{\text{Sub,Ann}} = \sum_i F_i \times C_i \times DCC_i \times OF_{\text{Annual}} \quad (4\text{-}2)$$

where

$OSA_{\text{Sub,Ann}}$ is annual onsite ambient dose from the submersion pathway (mrem/yr)
F_i is activity fraction of radionuclide *i*
C_i is air concentration of radionuclide *i* (Bq/m^3)
DCC_i is International Commission on Radiological Protection (ICRP) Publication 144 submersion dose conversion coefficient (DCC) for radionuclide *i* (mrem/hr per Bq/m^3) [ICRP 2020]
OF_{Annual} is worker annual occupancy (2,000 hr/yr)

Three primary source terms for the onsite ambient dose screening calculation were 20% enriched uranium [NIOSH 2011a], thorium, and MFPs. Thorium consisted of Th-232 in full equilibrium with progeny, which is favorable to claimants. Rather than evaluate the fractional dose contribution from each fission product, the analysis selected Cs-137 and a 10-day fuel decay period with the maximum submersion DCC from ICRP Publication 144 [ICRP 2020]. Cesium-137 was assumed to be present at the MFP community air concentration limit.

Because of the bounding nature of the submersion pathway screening calculation, a 2,000 hr/yr occupancy was assumed. The total submersion dose for the three source terms was less than 1 mrem/yr [ORAUT 2026].

Modeling ground deposition and subsequent buildup in the soil is complex due to the large variation of environmental transport parameters. As with the assessment of the submersion pathway, a screening assessment evaluated the dose contribution from ground deposition using methods and equations in Section 6.3 of International Atomic Energy Agency (IAEA) [2001]. Equation 4-3 provides the method of calculating dose from ground deposition. Variables related to the environmental transport process, such as the total deposition rate, effective rate constant for soil activity depletion (λ_E), and effluent discharge time (t_b), can significantly influence the calculated dose rate.

$$OSA_{\text{DG,Ann}} = \sum_i F_i \times DCC_i \times OF_{\text{Annual}} \frac{\left[\left(V_{\text{d},i} + V_{\text{w},i}\right) C_i\right]\left[1 - \exp\left(-\lambda_{E,i} t_{b,i}\right)\right]}{\lambda_{\text{E},i}} \quad (4\text{-}3)$$

(Attachment A contains an extended description.)

where

$OSA_{\text{DG,Ann}}$ is annual onsite ambient dose due to direct ground exposure pathway (mrem/yr)
F_i is activity fraction of radionuclide *i*
C_i is air concentration of radionuclide *i* (Bq/m^3)
DCC_i is ICRP Publication 144 ground contamination DCC for radionuclide *i* (mrem/hr per Bq/m^2) [ICRP 2020]
OF_{Annual} is worker annual occupancy (2,000 hr/yr)
$V_{\text{d},i}$ is dry deposition coefficient for radionuclide *i* (m/d)
$V_{\text{w},i}$ is wet deposition coefficient for radionuclide *i* (m/d)

$\lambda_{E,i}$ is $\lambda_{R,i} + \lambda_{s,i}$, effective rate constant for reduction of activity in the top 10 to 20 cm of soil for radionuclide *i* (d^{-1}),

where

$\lambda_{R,i}$ is rate constant for decay of radionuclide *i* (d^{-1})
$\lambda_{s,i}$ is rate constant for activity reduction to processes other than decay of radionuclide *i* (d^{-1})

$t_{b,i}$ is duration of air effluent discharge for radionuclide *i* (d)

The effective rate constant for removal of activity from the soil accounts for physical processes such as radioactive decay and migration of the radionuclide deeper into the soil column, which depends on the radionuclide's solubility. For this screening assessment, the only activity removal considered is that due to radioactive decay; in other words, $\lambda_s = 0$. The deposition coefficient can vary widely depending on the physical and chemical form of the effluent, local meteorological conditions, and topological profile of the deposition surface. The effective deposition rate (sum of wet and dry deposition coefficient) was assumed to be 1,000 m/d for all radionuclides using guidance in Section 3.9 of IAEA [2001]. This selected deposition coefficient should be appropriate for a screening calculation and therefore bounding for this assessment. Finally, the effluent discharge time can significantly influence the expected ground concentration. The conceptual model for this assessment is that the facility discharges air effluent that produces a ground-level air concentration equal to the air concentration limits (Table 4-1) only during typical operating hours. It was assumed that the facility operated 8 hr/d for 250 d/yr. Therefore, for a year of operations, t_b = 83.3 days as shown below:

$$t_b = 1 \text{ operating year} \times \left(\frac{8 \text{ hr/d} \times 250 \text{ d/yr}}{8{,}760 \text{ hr/yr}} \right) \times 365 \text{ d/yr} = 83.3 \text{ d} \quad (4\text{-}4)$$

In other words, ground deposition occurs continually over nonconsecutive 83.3 total operating days in a calendar year. The dose rate at the end of the first calendar year applies to all years of the covered period. Assuming the annual deposition rate occurred over the entire coverage period would produce unrealistic concentrations (see later discussion in this section for a comparison of the screening values to survey data). Implementing Equation 4-3 for the complete suite of ORAUT-OTIB-0054 [ORAUT 2015] radionuclides is a complex calculation involving decay and ingrowth of parent and progeny. For simplification, Cs-137 was assumed to be the only radionuclide that deposits on the ground. Cesium-137 does not have the largest DCC of the ORAUT-OTIB-0054 radionuclide suite but, as discussed further below, this assumption produces a bounding estimate of the dose from the direct ground exposure pathway. ICRP Publication 144 volumetric DCCs representing uniform contamination of 5 cm^2/g of soil were used to assess direct ground exposure [ICRP 2020].

Table 4-2 is a summary of onsite ambient dose by exposure pathway which is assigned to unmonitored workers. When assigning onsite ambient dose, dose reconstructors should assign the total annual dose from each pathway in Table 4-2 (a total of 42 mrem/yr) multiplied by the appropriate organ dose conversion factor as photons with energy of 30 to 250 keV with a constant distribution using the guidance in Attachment B of ORAUT-OTIB-0088, *External Dose Reconstruction* [ORAUT 2021]. Onsite ambient dose should be assigned in this manner because the bounding value was calculated using the best available data. In addition, the bounding nature of the calculation justifies use of a constant distribution in the Interactive RadioEpidemiological Program.

Table 4-2. Onsite ambient dose by nuclide and pathway (mrem/yr).[a]

Nuclide	Submersion	Direct ground
Uranium[b]	0.0001	0.02
Thorium	0.02	5.35
MFPs	0.16	34.90
Total	0.18	40.27
Assignment[c]	**1**	**41**

a. Source: ORAUT [2026].
b. Includes recycled uranium contaminants from Table 5-1.
c. Annual doses from each pathway are assigned as onsite ambient dose. The values in the assignment row are the total values rounded up.

5.0 OCCUPATIONAL INTERNAL DOSE

5.1 INTERNAL SOURCES OF EXPOSURE

A majority of source term data obtained by NIOSH pertains to periods outside the 1961 to 1970 AEC covered period under evaluation. NIOSH has obtained source term information for some specific projects or experiments during the period from January 1, 1961, through June 30, 1970, but lacks the specific information by which to identify the operations with the highest exposure potential. Such information is required to enable NIOSH to bound potential exposures using source term data in the absence of personnel monitoring data. NIOSH has not identified sufficient documentation to define and quantify the total source term for GE Ohio during the DOE covered period. Available documentation indicates that GE Ohio worked with uranium, thorium, and MFP radionuclides throughout the entire DOE covered period. Without additional documentation, NIOSH can make no assumptions about the relative amounts of these materials that would have been encountered at the site during the period from January 1, 1961, through June 30, 1970. Therefore, there is insufficient source term information available to NIOSH to bound internal exposures to uranium, thorium, and MFPs for the period from January 1, 1961, through June 30, 1970 [NIOSH 2011a]. Workers might have received radiation exposures from uranium, thorium, and MFP radionuclides during the covered period from January 1, 1961, to June 30, 1970 [NIOSH 2011a].

> *Although NIOSH found that it is not possible to completely reconstruct radiation doses for the proposed class, NIOSH intends to use any internal… monitoring data that may become available (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures) for an individual claim. Therefore, dose reconstructions for individuals employed at General Electric Company (Ohio) during the period from January 1, 1961 through June 30, 1970, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate.* [Sebelius 2011, p. 5].

GE routinely analyzed urine samples from ANP Department workers for one or more of the following substances:

- *Normal (natural) uranium.* Most normal uranium compounds the department handled were insoluble [GE 1960]. Insoluble would generally refer to lung absorption Types M or S materials. Urinalysis of normal uranium was accomplished photofluorometrically by measuring the fluorescence in a sodium fluoride flux of ultraviolet light [GE 1960].

- *Enriched uranium*. Most enriched uranium compounds were insoluble.

- *Thorium from the calcination operations with thorium oxide*. Thorium compounds were either absorption Type M or S [Herries 1966, p. 5; Rennich 1965, p. 7; ORAUT 2008, p. 3].

- *Gross beta*. The urine samples from workers who handled or were near the handling of MFPs that can become airborne were analyzed for beta activity [GE 1960].

There might be radionuclides in the claim records that are not addressed in this document. If this is the case, contact the Principal Internal Dosimetry Scientist for assistance.

5.1.1 Uranium

In SEC-00161, NIOSH determined reconstruction of unmonitored internal doses during the SEC period from January 1, 1961, through June 30, 1970, is not feasible [NIOSH 2011a]. However, uranium bioassay data are available for some workers. Normal (natural) uranium bioassay samples were analyzed using photofluorometric measurement of urine samples by measuring the fluorescence of the sample in a NaF flux under ultraviolet light. Urinalysis for enriched uranium was performed via evaporation and plating onto a planchet and counting the alpha activity of the sample [GE 1960]. Dose reconstructors should evaluate absorption Types M and S and use the type that assigns the higher dose.

Urine sample results for workers who handled natural uranium were in mass units of milligrams per liter (mg/L) or uranium per milliliter (U/mL). Bioassay data that show less than an associated value (i.e., <0.001 mg/L) are considered below the minimum detectable concentration (MDC). The bioassay result is used as the MDC. If the bioassay data show less than an associated value, consider the result less than the MDC, otherwise consider the result positive. A specific activity of 1.56 disintegrations per minute per microgram (dpm/µg) should be used to convert the mass units from photofluorometric bioassay results to units of radioactivity [GE 1960].

Sample results from workers who handled enriched uranium were in activity units in dpm per liter (dpm/L) for uranium gross alpha [GE 1957]. Therefore, no conversion from mass to activity is necessary. Bioassay data that show no detectable (e.g., “nd”) activity are considered below the MDC. In this case, 5 dpm/L should be used as the MDC. Bioassay data that show less than an associated value (e.g., <5) are also considered below the MDC. In this case, the bioassay result is used as the MDC. If the bioassay data show less than an associated value, consider the result less than the MDC, otherwise consider the result positive.

To account for the possibility that recycled uranium was present, dose reconstructors should assign additional intakes of recycled uranium contaminants (Pu-239, Np-237, Tc-99, Th-232, and Th-228) when assigning a uranium intake. Table 5-1 lists the values to use from Battelle-TBD-6000, *Site Profiles for Atomic Weapons Employers That Worked Uranium Metals* [NIOSH 2011b].

Table 5-1. Recycled uranium contaminant activity fractions.[a]

Nuclide	Activity fraction
Pu-239	2.46E−03
Np-237	1.82E−03
Tc-99	3.79E−01
Th-232	2.73E−06
Th-228[b]	2.73E−06

a. Source: NIOSH [2011b, p. 19].
b. Assumes same activity as Th-232.

5.1.2 Thorium

Dose reconstructors should contact the Principal Internal Dosimetry Scientist for assistance in evaluating thorium bioassay.

5.1.3 Gross Beta

Gross beta bioassay data are available for some workers, but no detection limits are available for gross beta results. Given that no detection limits are available, it is assumed that all results that cannot be determined to have detectable activity (i.e., no reported detection limit or uncertainty) are assumed to be positive and will be assessed as a fitted exposure. When gross beta results are available, dose reconstructors assign intakes using ORAUT-OTIB-0054 and Sr-90 as the indicator radionuclide [ORAUT 2015]. As discussed in Section 4.1, ATR 1 is the applicable reactor type.

6.0 OCCUPATIONAL EXTERNAL DOSE

NIOSH has obtained individual external monitoring data for each year during the period 1961 through 1970. These data include over 3,500 film badge readings. Because NIOSH-identified deficiencies in internal monitoring data span the entire GE Ohio covered period, NIOSH has not evaluated the external film data on an individual worker basis, and has not determined whether sufficient data exist to generate adequate external co-exposure dose distributions for the assignment of unmonitored dose during partial dose reconstruction [NIOSH 2011a, p. 4].

> *Although NIOSH found that it is not possible to completely reconstruct radiation doses for the proposed class, NIOSH intends to use any… external monitoring data that may become available (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures) for an individual claim. Therefore, dose reconstructions for individuals employed at General Electric Company (Ohio) during the period from January 1, 1961 through June 30, 1970, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate* [Sebelius 2011, p. 5].

6.1 SOURCES OF EXTERNAL EXPOSURE

The potential for external radiation dose existed in Buildings C and D and surrounding support areas including a fenced and locked outdoor Radioactive Waste Storage Area north of Building D. Based on the site operations (Section 2.0), sources of exposure included:

- Natural, depleted, and highly enriched uranium [GE 1973–1996];
- MFPs from work with irradiated reactor fuel elements [GE 1973–1996; Rice 1966]; and
- Thorium from the calcination operations with thorium oxide [Herries 1966, p. 5; Rennich 1965, p. 7; ORAUT 2008, p. 3].

6.2 EXTERNAL MONITORING PROGRAM

The information in this section is primarily from the GE Ohio *Health and Hygiene Instruction Manual* [GE 1960].

Film Badges
GE dosimetry badges were a DuPont film Type 553 for beta/gamma and nuclear track emulsion, Type A (NTA) for neutron dose processed by Landauer [GE 1960].

Film badges and film packs were distributed for weekly or biweekly periods depending on the amount of radiation exposure an individual might receive. In general, individuals working in areas where little or no radiation exposure was likely received fresh film biweekly. Individuals working in areas with a potential radiation hazard greater than 75 mrem/wk received fresh film weekly. Individuals working in areas with a potential neutron hazard received fresh NTA film on a weekly or biweekly basis depending on the frequency of distribution for the beta/gamma film.

The film badge was a metal holder for dental size film packs used to measure radiation exposure to personnel. It was made of stainless steel with dimensions of 1.875 by 1.375 by 0.25 in. A finger clip fastener on the rear held the badge to clothing. The badge had a removable front section for loading of two dental size film packs. For identification, an opening at the top of the front section held a nameplate showing the wearer's name and security badge number. Because the metallic construction of the badge shielded out soft radiations (beta, low-energy X-rays, etc.), an open window (0.75 by 0.3125 in.) in the front section permitted entry and measurement of such radiation.

To shield out soft radiations that would pass through the upper opening, a cadmium filter (1.25 by 1.125 in. by 43 mils) was inserted in the back of the front section of the badge. A second filter was preinserted at the factory in the rear section to shield out unwanted radiation that might have entered through the back of the badge. To permit marking the film, the front filter was perforated with a number identical to the number on the nameplate.

Dental-size film packs (1.25 by 1.75 in.) were loaded in the film badges described above for personnel radiation dosimeters. Various types of film packs were produced to measure the different types and ranges of radiation. To meet the needs of the ANP Department, the following two types of film packs were used:

- DuPont Safety Film Pack, Type 553 for measuring beta, gamma, and X radiation. This type of film pack was identified by the green tab at the rear and the number "553" embossed on the tab. The Type 553 film pack contained the following three component films:
  - Type 502 ("sensitive film") with a practical useable range of 30 to 10,000 mR based on radium gamma calibrations,
  - Type 510 ("insensitive film") with a practical useable range of 500 to 20,000 mR based on radium gamma calibrations, and
  - Type 606 ("disaster film") with a range of 30,000 to 500,000 mR.
- Eastman Kodak Special Nuclear Safety Film, Type NTA, for detecting fast neutron exposure. This type of film pack can be identified by the name and the black tab found on one side of the pack.

Finger Rings

Finger rings were issued to personnel working in areas with a potential hand exposure greater than 200 mrem/wk. The distribution basis conformed to the film badge distribution basis. Radiation exposure indicated by film was part of the worker's permanent exposure record.

The finger ring measured radiation exposure to the hands. The finger ring for personnel monitoring in the ANP Department was made of plastic material and numbered on the back. Component parts of the ring included a plastic cap (thickness 0.14 cm) that sealed the film chamber, two cadmium washer-type shields (1.25 cm diameter) with a circular aperture (0.50 cm diameter), and a chamber that contained DuPont film type 555. Because the cadmium acted to shield out soft (beta, low energy

X-rays, etc.) radiations, the circular aperture in the cadmium washer permitted entry and measurement of such radiation.

Pocket Ionization Chambers (PICs)
PICs were issued daily to personnel regularly issued film badges and who worked in potential radiation fields of 5 mrem/hr or more. Radiation exposures indicated by PIC readings became part of the worker's permanent radiation exposure record.

The Landsverk Pocket Chamber Model L69 was an indirect-reading, fountain pen-type, polyethylene insulated ionization chamber that could be clipped to the breast pocket. In use, the electrode of the PIC was charged and, upon exposure to radiation, it discharged. A PIC could measure up to 200 mrem/wk of X-rays, gamma rays, and slow neutrons.

6.3 EXTERNAL MONITORING DATA

NIOSH has obtained film badge, PIC, and extremity monitoring results for GE Ohio workers for the period under evaluation [GE 1950–1974, 1957–1968; Sapirie 1965, p. 6; Clark 1961; GE 1961; Engel 1971, 1972–1973, 1977–1988]. NIOSH reviewed these data to identify work location information that might be useful in differentiating between exposed and unexposed worker populations or job descriptions. Less than 5% of the 4,451 external results had a noted work location. NIOSH obtained the following data for the covered period from January 1, 1961, to June 30, 1970:

- *1961*. 1,252 film, 124 PIC, and 47 finger/hand results;
- *1962*. 262 film and 18 finger/hand results;
- *1963*. 207 film and 38 finger/hand results;
- *1964*. 418 film and 42 finger/hand results;
- *1965*. 274 film and 28 finger/hand results;
- *1966*. 309 film and 250 finger/hand results;
- *1967*. 298 film and 252 finger/hand results;
- *1968*. 254 film and 49 finger/hand results;
- *1969*. 214 film and 3 finger/hand results; and
- *1970*. 107 film and 5 PIC results.

NIOSH has found limited information regarding occupational medical dose.

GE Ohio personnel usually receive chest X-rays as part of the routine medical examination. Each chest X-ray examination will be considered equivalent to 0.1 rem. This dose is included in the film badge totals [GE 1960]. Figure 6-1 shows an example of second quarter total film badge dose that includes an X-ray. The handwritten "X" after this total indicates that a chest X-ray exposure value of 0.1 rem was included.

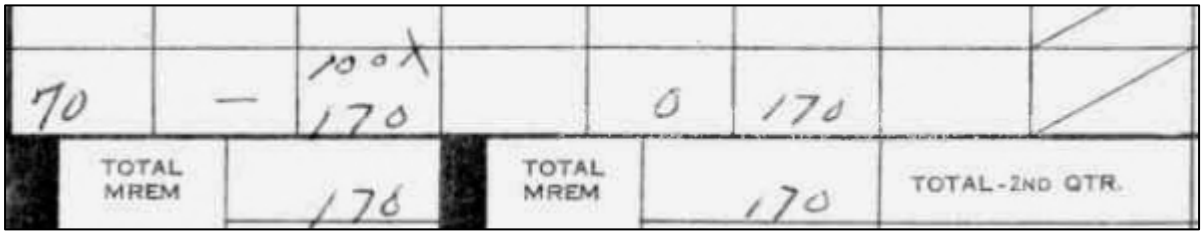

Figure 6-1. Example of second quarter total film badge dose that includes an X-ray [GE 1967–1986, p. 577].

Refer to Section 3.0 for information to assess X-ray dose.

The following information will be considered when assigning occupational external dose:

- For an overestimating approach to assigning external dose, use the records as they are.
- For a best-estimate approach to assigning external dose, use the information in the Individual Employee Dose Summaries. If there are no Individual Employee Dose Summaries for the employee, then request the Individual Employee Dose Summaries. If the records are not available, contact the Principal Scientist for External Dosimetry.
- For an underestimate approach, remove 0.1 rem from every year that exceeds 0.1 rem.

6.4 EXTERNAL DOSIMETRY GUIDELINES

6.4.1 Photon

Exposure to photons was possible during the handling of natural, depleted, and highly enriched uranium, thorium, and MFPs from work with irradiated reactor fuel elements.

Dose reconstructors should apply penetrating doses as 100% 30- to 250-keV photons as acute with exposure (roentgen)-to-organ (H_T) dose conversion factors for the anterior-posterior geometry from OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* [NIOSH 2007]. No information about dosimeter uncertainties is available. Therefore, apply an uncertainty of 30% to measured doses [Morgan 1961]. Table 6-1 shows the limit of detection (LOD) and exchange frequency for photon exposures.

Table 6-1. Photon LODs and exchange frequency.[a]

Period	Type	LOD	Exchange frequency
01/01/1961–06/30/1970	Film	30 mrem	Weekly

a. Source: GE [1960].

6.4.2 Beta

Evaluation of skin doses is complex because GE Ohio did not report doses as penetrating or nonpenetrating. The reporting scheme can be identified by evaluating the data. At GE Ohio, the reported beta (nonpenetrating) values are always equal to or higher than the reported gamma (penetrating) values. This indicates that the beta reading includes both penetrating and nonpenetrating radiation. Therefore, dose reconstructors should calculate nonpenetrating dose by subtracting the gamma dose from the beta dose.

Dose reconstructors should apply nonpenetrating doses as >15 keV electrons using acute exposure with a dose conversion factor from ORAUT-OTIB-0017, *Interpretation of Dosimetry Data for*

Assignment of Shallow Dose [ORAUT 2005]. Table 6-2 shows the LOD and exchange frequency for beta exposures.

Table 6-2. Beta LODs and exchange frequency.[a]

Period	Type	LOD	Exchange frequency
01/01/1961–06/30/1970	Film	30 mrem	Weekly

a. Source: GE [1960].

6.4.3 Neutron

Contact the Principal External Dosimetry Scientist if there are neutron dosimetry data.

7.0 INTERNAL DOSIMETRY CO-EXPOSURE DATA

NIOSH does not have access to sufficient personnel monitoring, workplace monitoring, or source term data to estimate potential internal exposures to uranium, thorium, or fission product radionuclides during the period of DOE operations at GE Ohio from January 1, 1961, through June 30, 1970 [NIOSH 2011a, p. 21]. Consequently, NIOSH finds it is not feasible to estimate unmonitored internal doses during the covered period and therefore, a co-exposure model cannot be developed.

8.0 EXTERNAL DOSIMETRY CO-EXPOSURE DATA

NIOSH has obtained individual external monitoring data for approximately 11% of the claims referred to NIOSH for dose reconstruction. NIOSH has found insufficient external monitoring data to allow for estimation of external dose for unmonitored energy employees [Sebelius 2011, p. 4].Consequently, NIOSH finds it is not feasible to estimate unmonitored external doses during the covered period and therefore, a co-exposure model cannot be developed.

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GLOSSARY

Atomic Weapons Employer (AWE) [42 U.S.C. § 7384l(5)]
Entity other than the United States, that—(A) processed or produced, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining and milling, and (B) is designated by the Secretary of Energy as an Atomic Weapons Employer for purposes of the [Energy Employees Occupational Illness] compensation program.

activation product
Radionuclide produced through neutron absorption.

becquerel (Bq)
International System unit of radioactivity equal to 1 disintegration per second; 1 curie equals 37 billion (3.7E+10) becquerels.

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.

centi-
Prefix that divides a unit by 100 (multiplies by 1E10−2).

co-exposure dose
Previously referred to as “coworker dose” on the Project, co-exposure dose is a representative dose typically assigned because (1) the worker was not monitored (unmonitored) but should have been or (2) the worker was monitored but the results are unavailable or unreliable.

curie
Traditional unit of radioactivity equal to 37 billion (3.7E+10) becquerels, which is approximately equal to the activity of 1 gram of pure Ra-226.

disintegrations per minute
Measure of radioactivity equal to the number of nuclear disintegrations in a mass per minute; 1 disintegration per minute equals 1/60 becquerel.

external dose
Dose received from radiation (e.g., photons, electrons, and neutrons) that originates outside the body including that from medical screening examinations.

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

gram
International System unit of mass equal to one-thousandth (1E−3) of a kilogram. The original weight-oriented definition of a gram was the weight of 1 cubic centimeter of water.

highly enriched uranium
Uranium enriched to at least 20% U-235 for use as fissile material in nuclear weapons components and some reactor fuels. Also called high-enriched uranium.

internal dose
Dose received from radioactive material in the body (e.g., plutonium or uranium) that was inhaled, ingested, absorbed, or injected through a wound.

kiloelectron-volt
Unit of particle energy equal to 1,000 (1E+3) electron-volts.

liter
International System unit of liquid volume approximately equal to 0.227 gallon.

milli-
Prefix that divides a unit by 1,000 (multiplies by 1E−3).

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

recycled uranium
Uranium first irradiated in a reactor then recovered through chemical separation and purification. Recycled uranium contains minor amounts of transuranic material (e.g., plutonium and neptunium) and MFPs (e.g., technetium) or uranium products (e.g., U-236) after purification.

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 *r*oentgen *e*quivalent in *m*an; rem is also the plural.

roentgen
Unit of photon (gamma or X-ray) exposure for which the resultant ionization liberates a positive or negative charge equal to 2.58E−4 coulomb 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 roentgen is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher energy photons (generally greater than 100 kiloelectron-volts).

ATTACHMENT A
EXTENDED DESCRIPTION OF EQUATION 4-3

This attachment contains an extended description for an equation that exceeds the character limit for inline alt text. This description is provided to enhance accessibility for screen reader users.

The annual onsite ambient dose due to ground exposure equals the summation of the activity fraction of radionuclide *i* times the ICRP 144 submersion DCC for radionuclide *i* times the receptor annual occupancy times the fraction where the numerator is the sum of the dry deposition coefficient for radionuclide *i* plus the wet deposition coefficient for radionuclide *i,* that sum times the air concentration of radionuclide *i* times the difference of 1 minus the exponential function to the minus effective rate constant for reduction of activity times the duration of air effluent discharge and the denominator is effective rate constant for reduction of activity.