DOE Review Release 07/25/2011

Document Title: **An Exposure Matrix for Linde Ceramics Plant (Including Tonawanda Laboratory)**	Document Number: ORAUT-TKBS-0025 Revision: 02 Effective Date: 07/15/2011 Type of Document: TBD Supersedes: Revision 01 PC-1

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☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

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

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
05/31/2005	00	New Technical Basis Document: Basis for the Development of an Exposure Matrix for Linde Air Products, Tonawanda, New York and Buffalo, New York. First approved issue. Initiated by Cindy Bloom.
01/19/2006	00 PC-1	Modified to incorporate comments from OCAS, a Worker Outreach meeting comment and minor editing. The Introduction on page 10 was updated. Dates were changed in Section 3.0 to incorporate the decision to designate Linde Ceramics workers, who received internal exposures prior to 11/01/1947, as a Special Exposure Cohort. Moved Table 12, an internal exposure table out of Section 4.0. Changes were made to page 23 in Section 3, page 34 in Table 7, page 35 in Table 8, page 38 in Table 11, and Table 12 on page 39. Approved page change revision. No sections were deleted. Training required: As determined by the Task Manager. Initiated by Cindy W. Bloom. Signature on File 01/18/2006 Cindy W. Bloom, TBD Team Leader Signature on File 01/18/2006 Judson L. Kenoyer, Task 3 Manager Signature on File 01/18/2006 Kate Kimpan, Project Director Signature on File 01/19/2006 James W. Neton, Associate Director for Science
11/04/2008	01	Approved revision to change from a page change revision (Rev 00 PC-2-B) to a total rewrite (Rev 01-A) as a result of formal NIOSH review. Revised to incorporate: (1) change in facility designation, (2) DOL interpretation of applicability of residual period to Ceramics Plant, (3) and resolution of Advisory Board Working Group comments, (4) clarified the implementation instructions for SEC00044 for the period October 1, 1942 through October 31, 1947. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Task Manager. Initiated by Joseph S. Guido.

11/10/2009	01 PC-1	Approved page change revision to incorporate additional guidance on the use of Attachment E of the document. This change constitutes the resolution of Advisory Board Working Group concern and clarifies how to proceed in the event of credible indication of potential exposure to contaminated burlap bags that were used for transport of ore at Linde Ceramics. Changes were made to page 37 in Section 4.0. No sections were deleted. No changes occurred as a result of formal internal review. Incorporates formal NIOSH review comments. Training required: As determined by the Objective Manager. Initiated by Mutty Sharfi. Signature on File 11/05/2009 Mutty Sharfi, Document Owner Signature on File 11/04/2009 John M. Byrne, Objective 1 Manager Signature on File 11/04/2009 Edward F. Maher, Objective 3 Manager Signature on File 11/04/2009 Kate Kimpan, Project Director Signature on File 11/10/2009 James W. Neton, Associate Director for Science
07/15/2011	02	Revision to incorporate changes associated with SEC-0107. This change constitutes the resolution of SEC-0107. This includes the exposure scenarios for the tunnels, the external for the renovation period and the post-renovation period. An SEC was approved covering internal exposures during the renovation period (1954 to 1969). Changes were made to Sections 6.0, 6.1, 6.2, and the references. In addition, Sections 1.3, 2.3.5, and 6.4 were added. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Mutty M. Sharfi.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
d	day
DCF	dose conversion factor
DHHS	U.S. Department of Health and Human Services
DOE	U.S. Department of Energy
dpm	disintegrations per minute
D&D	decontamination and decommissioning
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ft	foot
FUSRAP	Formerly Utilized Sites Remedial Action Program
GM	geometric mean
GSD	geometric standard deviation
hr	hour
ICRP	International Commission on Radiological Protection
keV	kiloelectron-volt, 1,000 electron-volts
L	liter
lb	pound
LAPC	Linde Air Products Company
LOD	limit of detection
LOOW	Lake Ontario Ordnance Works
MAC	maximum allowable concentration
m	meter
MED	Manhattan Engineer District
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
mL	milliliter
mm	millimeter
mr	radiation exposure unit used by MED in the 1940s; interpreted here as millirad for photons and millirem for beta radiation
mR	milliroentgen
mrad	millirad
mrem	millirem
mrep	millirep
MT	metric ton (1,000 kg)
NIOSH	National Institute for Occupational Safety and Health
NYOO	New York Operations Office

OCAS	Office of Compensation Analysis and Support
ORNL	Oak Ridge National Laboratory
ORAUT	Oak Ridge Associated Universities Team
PAEC	potential alpha energy concentration (a measure of concentration of radon progeny)
PAEE	potential alpha energy exposure (a measure of exposure to radon progeny)
pCi	picocuries
PL	preferred level
POC	probability of causation
R	roentgen
R&D	research and development
s	second
SC&A	Sanford Cohen & Associates
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
t	ton (2,000 lb)
U.S.C.	United States Code
USACE	U.S. Army Corps of Engineers
USBM	U.S. Bureau of Mines
wd	workday
wk	week
WL	working level
WLM	working level-month
yr	year
α	alpha particle
µg	microgram
µm	micrometer
µR	microroentgen
°C	degrees Celsius
°F	degrees Fahrenheit
§	section or sections

1.0 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historic background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

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

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

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 historic 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 [AWE] facility" or a "Department of Energy [DOE] facility" as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 [EEOICPA; 42 U.S.C. § 7384I(5) and (12)]. EEOICPA, as amended, provides for employees who worked at an AWE facility during the contract period and/or during the residual period.

Under EEOICPA, employment at an AWE facility is categorized as either (1) during the DOE contract period (i.e., when the AWE was processing or producing material that emitted radiation and was used in the production of an atomic weapon), or (2) during the residual contamination period (i.e., periods that NIOSH has determined there is the potential for significant residual contamination after the period in which weapons-related production occurred). For contract period employment, all occupationally derived radiation exposures at covered facilities must be included in dose reconstructions. This includes radiation exposure related to the Naval Nuclear Propulsion Program and any radiation exposure received from the production of commercial radioactive products that were concurrently manufactured by the AWE facility during the covered period. NIOSH 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

For employment during the residual contamination period, only the radiation exposures defined in 42 U.S.C. § 7384n(c)(4) [i.e., radiation doses received from DOE-related work] must be included in dose reconstructions. Doses from medical X-rays are not reconstructed during the residual contamination period (NIOSH 2010). It should be noted that under subparagraph A of 42 U.S.C. § 7384n(c)(4), radiation associated with the Naval Nuclear Propulsion Program is specifically excluded from the employee's radiation dose. This exclusion only applies to those AWE employees who worked during the residual contamination period. Also, under subparagraph B of 42 U.S.C. § 7384n(c)(4), radiation from a source not covered by subparagraph A that is not distinguishable through reliable documentation from radiation that is covered by subparagraph A is considered part of the employee's radiation dose. This site profile covers only exposures resulting from nuclear weapons-related work. Exposures resulting from non-weapons-related work, if applicable, will be covered elsewhere.

1.1 PURPOSE

This site profile document provides an exposure matrix for workers at the Tonawanda Laboratory and Linde Ceramics Plant facilities of the Linde Air Products Company (LAPC) in Tonawanda, New York.

1.2 SCOPE

DOL has designated the Linde Ceramics Plant a DOE facility and the Tonawanda Laboratory an AWE facility (Turcic 2007). This document covers both facilities. The information in this site profile supports the assumed operational and residual contamination periods listed below.

Facility	Operational period	Residual contamination period[a]
Linde Ceramics Plant and Tonawanda Laboratory	10/01/1942–10/31/1953 1988–1992; 1996 (remediation)	January 1, 1954 to July 2006[b]

a. Also called the postcleanup period in this document.
b. Excluding 1988 to 1992 and 1996.

Section 2.0 describes the site and its operational history. Sections 3.0 and 4.0 describe estimation of internal and external exposure from 1942 to December 31, 1953, respectively. Section 5.0 describes occupational medical exposure. Section 6.0 provides information on exposures during the residual contamination period after 1953. Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 7.0.

Attachment A contains data that was used in analyzing exposures of workers to beta radiation. Attachment B lists codes and special terminology in the LAPC records. Attachment C shows data sources on uranium progeny concentrations, and Attachment D provides a uranium coworker assessment for November 1947 to January 1950. Attachment E provides an assessment of dose consequences from uranium ore bag that were stored on the site during the postoperations period.

1.3 SPECIAL EXPOSURE COHORT PETITION INFORMATION FOR LINDE

October 1, 1942, through October 31, 1947
NIOSH has determined, and the Secretary of the U.S. Department of Health and Human Services (DHHS) has concurred, that it is not feasible to reconstruct internal radiation dose for (Leavitt 2005):

> *Atomic Weapons Employees who worked at the Linde Ceramics Plant from October 1, 1942, through October 31, 1947, and who were employed for a number of work days aggregating at least 250 work days either solely under this employment or in combination with work days occurring within the parameters (excluding aggregate work day requirements) established for other classes of employees included in the SEC* [Special Exposure Cohort].

Subsequent correspondence (Elliott 2006) confirmed that the Tonawanda Laboratory (as well as all other buildings on the Linde site) are included in this SEC class designation. Reconstruction of external exposure (including medical X-ray examinations) has been determined to be feasible (Leavitt 2005).

For any claim referred to NIOSH regarding an employee,

- who was employed during the SEC class period but because of limited employment during this period is not a member of the SEC, or
- who is a member of the SEC class and whose cancer is not defined as a specified cancer under EEOICPA (and so is not eligible for compensation under EEOICPA without a dose reconstruction),

NIOSH will continue to attempt to complete a dose reconstruction for the exposure period based solely on external and medical X-ray radiation sources. However, because of the SEC class determination that it is infeasible to adequately reconstruct internal dose during the period from October 1, 1942, through October 31, 1947 (Leavitt 2005), dose estimates for this period are considered partial dose estimates.

January 1, 1954, through December 31, 1969
NIOSH has determined, with concurrence from the Secretary of DHHS (Sebelius 2011), that internal doses at the Linde Ceramic Plant cannot be reconstructed with sufficient accuracy from the beginning of 1954 through the end of 1969. For this reason, the following class of Linde employees has been added to the SEC (Sebelius 2011):

All Atomic Weapons Employees who worked at the Linde Ceramics Plant in Tonawanda, New York, from January 1, 1954, through December 31, 1969, 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 in the Special Exposure Cohort.

This SEC class includes all workers during the SEC class period. Because of the identified dose reconstruction infeasibility, all dose reconstructions for all workers having employment during the SEC class period are considered partial dose reconstructions. If monitoring data are available for workers included in the SEC class, dose is to be assigned as appropriate based on such data; however, such dose reconstructions are still considered partial dose reconstructions because NIOSH has determined that internal exposures during the SEC class period cannot be bounded.

2.0 SITE DESCRIPTION AND OPERATIONAL HISTORY

When the U.S. government and its contractors first became interested in uranium, LAPC, then a division of Union Carbide and Carbon Corporation, operated Tonawanda Laboratory, which had been producing U_3O_8 that was sold as a coloring agent for ceramics. Because of the great interest in obtaining uranium that could be used to create experimental uranium piles, Linde was contracted to develop uranium chemical processes and build a facility that could process large amounts of uranium ore. This commissioned facility was called the Linde Ceramics Plant. The plant worked with two types of radioactive material:

- Refined uranium materials and preprocessed (partially refined) domestic ores, and
- Unprocessed African ores.

Processing of the African ores involved exposures to significantly higher levels of ^{230}Th, ^{226}Ra, and ^{222}Rn. In the other materials, the initial refining process probably removed most nonuranium radionuclides. Due to the long half-lives of ^{230}Th and ^{231}Pa, significant ingrowth of these nuclides and their progeny in the ^{238}U and ^{235}U decay chains did not occur in the interval between the refining of the materials and their processing at Linde.

LAPC performed work for the Manhattan Engineer District (MED) and its successor the U.S. Atomic Energy Commission (AEC) between 1942 and 1949. Tonawanda Laboratory performed research and development (R&D) on uranium processing for the MED beginning in late 1942 and probably ending in 1946. In the early years, it operated pilot plants to develop procedures for the Ceramics Plant. The Ceramics Plant produced uranium materials for the MED and AEC from 1943 to 1946 and from 1947 to 1949. From 1947 to 1949 (and perhaps earlier), Linde received UO_2 for processing from Mallinckrodt Chemical Works (AEC 1949a). The plant also produced nickel material for the K-25 diffusion barrier. The Ceramics Plant was in standby from mid-1946 to late 1947. The end of production in 1949 was followed by cleanup and decontamination and then turnover of the production facilities back to Linde. This turnover probably occurred in 1954.

Uranium processing at the Ceramics Plant involved three steps:

- Step I, conversion of ore to U_3O_8 (black oxide);
- Step II, conversion of U_3O_8 to UO_3 (orange oxide) as an intermediary product and then to UO_2 (brown oxide);
- Step III, conversion of UO_2 to UF_4 (green salt).

The following sections provide more detailed histories of Linde facilities, radiation sources, and processes that related to MED/AEC radiation exposures. Key dates are noted in Table 2-1.

2.1 FACILITIES

The Linde Ceramics Plant and Tonawanda Laboratory were on land that was owned by Union Carbide at East Park Drive and Woodward Avenue in Tonawanda, New York (ORNL 1978, Fig. 3). The site is near the intersection of Riverview Boulevard and Woodward Avenue. It is north of Woodward Avenue, east of East Park Drive, and west of the Conrail railroad tracks.

Tonawanda site buildings that were involved in MED/AEC work are shown in Figure 2-1. The Tonawanda Laboratory occupied Building 14, and the Ceramics Plant used Buildings 30, 31, 37, 38, and A. Building B contained MED offices. Ownership of the Ceramics Plant buildings was transferred to LAPC after the site cleanup that began with the shutdown of production in 1949. The transfer probably was completed in 1954. In the 1990s, the site was acquired by Praxair (USACE 2003).

2.1.1 Buildings A and B

Building A was the Linde Ceramics Plant office building for administrative and support personnel. During the MED years, Building B housed the Tonawanda Area Engineer office of the MED (Dupree 1983a) and probably was later used by AEC personnel. The designations A and B were adopted for this document and might be different from the official Linde or MED designations. Buildings A and B were no longer standing in 1978, when the results of a 1976 radiological survey were issued by Oak Ridge National Laboratory (ORNL 1978).

Figure 2-1 is based on Figures 3-1 and B-11 of BNI (1982), Figure 3 of Frame et al. (1981), and LAPC (1945a).

A shelter in which workers were allowed to smoke was at the south end of Building B (Dupree 1983a). This was probably for use only by office workers. A separate smoking shelter was provided for production workers (Klevin 1949a, data sheets 543 and 546).

Table 2-1. Key dates.

Assumed date or period[a]	Event or activity
Ceramics Plant	
06/01/1943–07/31/1946[b]	Step I production (U_3O_8 from ore or sludge)
06/01/1943–11/30/1943	Preprocessed ores (domestic and scrap)
12/01/1943–11/12/1944	African ores
11/13/1944–01/31/1946	Preprocessed domestic ores
02/01/1946–02/28/1946	African ores (48%) and preprocessed domestic ores (52%)
03/01/1946–06/30/1946	African ores
07/01/1946–07/31/1946	African ores and preprocessed ash
04/27/1943–03/08/1944[c]	Step II production (UO_2 from U_3O_8)
07/25/1943–06/26/1946[d]	Step III production (UF_4 from UO_2)
08/01/1946–09/14/1947	Standby
09/15/1947–10/31/1947[e]	Step III rehabilitation
11/01/1947–06/30/1949[f]	Step III production
10/01/1944–02/28/1946[g]	Production of nickel material for K-25 diffusion barrier
07/01/1949–03/31/1950[h]	Cleanup of Building 30
07/01/1949–12/31/1954[i]	Cleanup of Linde Ceramics Buildings
Before 1978	Demolition of Buildings A and B (office buildings on Figure 2-1)
After 1954	Postcleanup period
08/31/1981[J]	Demolition of Building 37 began
08/31/1996[J]	Demolition of Building 38 began
09/05/1998[k]	Demolition of Building 30 began
09/30/2000[J]	Soil remediation began (scheduled for completion in 2007)[J]
Tonawanda Laboratory	
10/01/1942–07/31/1946[l]	MED-related R&D operations
08/01/1946–12/31/1946[m]	Cleanup period

Assumed date or period[a]	Event or activity
After 1947	Postcleanup period
04/30/2004[j]	Demolition of Building 14 began

a. Unless more precise information was available, activities were assumed to begin on the first day of the start month and to end on the last day of the completion month in the data source, and demolition activities were assumed to begin on the last day of the start month in the data source.
b. All Step I dates and data are based on Table B-1 of Wallo, Vierzba, and Roberts (1981) except for the November 1944 transition date from African to domestic ore. The date was determined from analysis of Step I film badge data (see Section 4.1.3.2).
c. LAPC (1946a, p. 40).
d. LAPC (1946b, p. 42).
e. Start date based on Kent (1947, p. 1).
f. Start date based on Rennich (1947). End date based on Heatherton (1948a).
g. Hickey, Crawford-Brown, and Tankersley (1988, Figure 13, p. 2).
h. Start date assumed to be day after end of Step III production; end date estimated based on Eisenbud (1950).
i. Start date assumed to be day after end of Step III production; end date estimated based on Harris (1954).
j. Pilon (2004).
k. USACE (1998).
l. Start date based on Jenness and Ewing (1943). End date assumed to be the end date of 1943 to 1946 production at the Ceramics Plant.
m. Start date based on estimated end date of R&D period; end date estimated.

2.1.2 Building 14 (Tonawanda Laboratory)

Building 14 housed Tonawanda Laboratory. The U.S. Army Corps of Engineers (USACE) has described its use as follows (USACE 2003):

> *Building 14 was used for laboratory and pilot plant studies for uranium separation in the early part of MED operations. Historical drawings indicate that the MED laboratory and pilot plant studies were initially confined to the south part of the building. It is unclear how extensively the remainder of the building was used for MED operations. However, documents indicate that laboratory and pilot plant operations were continued for the*

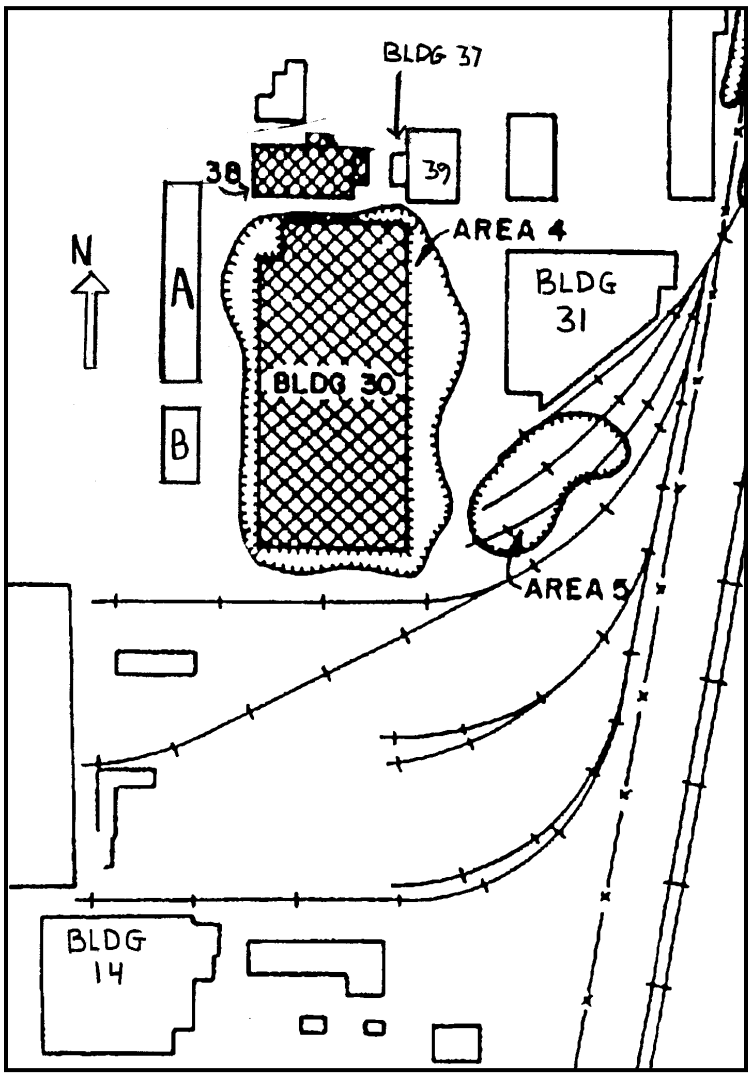

Figure 2-1. Linde Ceramics Plant and Tonawanda Laboratory buildings on LAPC Tonawanda site.

> *purpose of experimenting and developing more efficient processing methods, and operations appear to have been expanded into most of the building, possibly to support larger pilot studies. The available records do not indicate whether or not the use of Building 14 ceased before the MED/Atomic Energy Commission (AEC) operations were discontinued at Linde.*

A 1942 Tonawanda Laboratory directory listed approximately 200 employees (LAPC 1942). The locations of most were designated by room numbers. Some employees were said to be in the Eng. Lab and others in the Proving Lab. A 1944 employee list reported some to be in the Pilot Plant (LAPC 1944a). The terms Proving Lab and Pilot Plant appear to refer to the same facility.

Tonawanda Laboratory is also referred to by other names such as Linde Research Laboratory (Dupree 1983a) and The Laboratory of The Linde Air Products Company (Jenness and Ewing 1943).

Contamination in Building 14 was found in a 1976 radiological survey (ORNL 1978). At the time, Building 14 housed fabrication facilities, research facilities, and offices generally used by 20 to 30 employees. Demolition of Building 14 began in April 2004 (Pilon 2004).

2.1.3 Building 30 (Ceramics Plant)

Building 30—also called the uranium refinery—was used for Steps I and II. Figures 2-2 and 2-3 show the layout. It had two floors. Large pieces of equipment such as digestion tanks were on the ground floor. Process workers operated equipment from the second floor (Dupree 1983a,b). The 1949 to 1950 decontamination work by LAPC and the results are described in Heatherton (1950). As of 1976, the building was used as a shipping and receiving warehouse and occupied by about 20 to 30 employees. Radioactive contamination was found during a 1976 survey (ORNL 1978). Building 30 was demolished during the period from September 5 to 19, 1998 (USACE 1998).

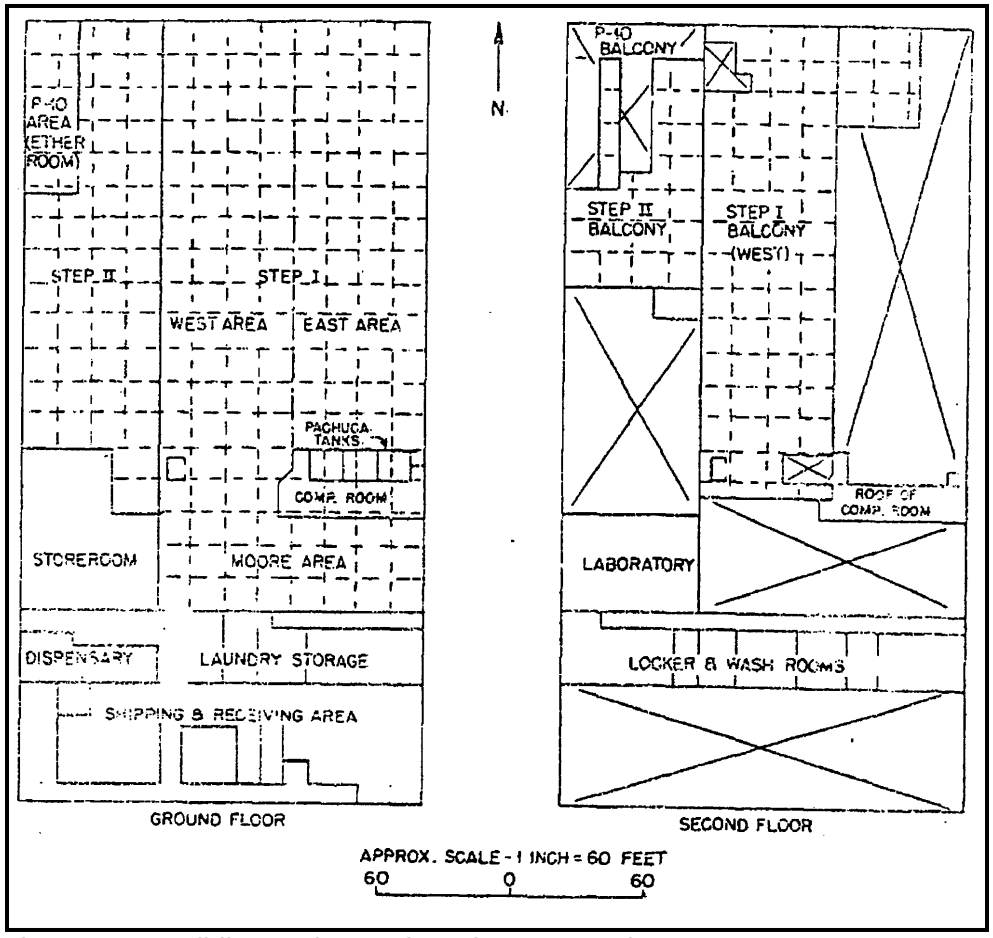

Figure 2-2. Building 30 layout (Heatherton 1950).

2.1.4 Building 31 (Ceramics Plant)

Building 31 is described as "used in uranium separation process during MED operations" (BNI 1993, p. 1-63). A partially readable 1945 plant drawing indicates that it contained shipping and storage areas, maintenance facilities (including a welding shop, a machine shop, a gauge test room, and a carpentry area), and a small number of associated offices (LAPC 1945a). Building 31 might have been the final location of the Linde Tonawanda nickel processing operations.

As of 1976, the building was described as housing fabrication facilities, offices, and storage areas, and it was occupied by about 12 to 15 employees. Radioactive contamination was found during a 1976 survey (ORNL 1978). Building 31 was decontaminated in 1997 (USACE 2002a). As of 2004, it was still standing and in use (Pilon 2004).

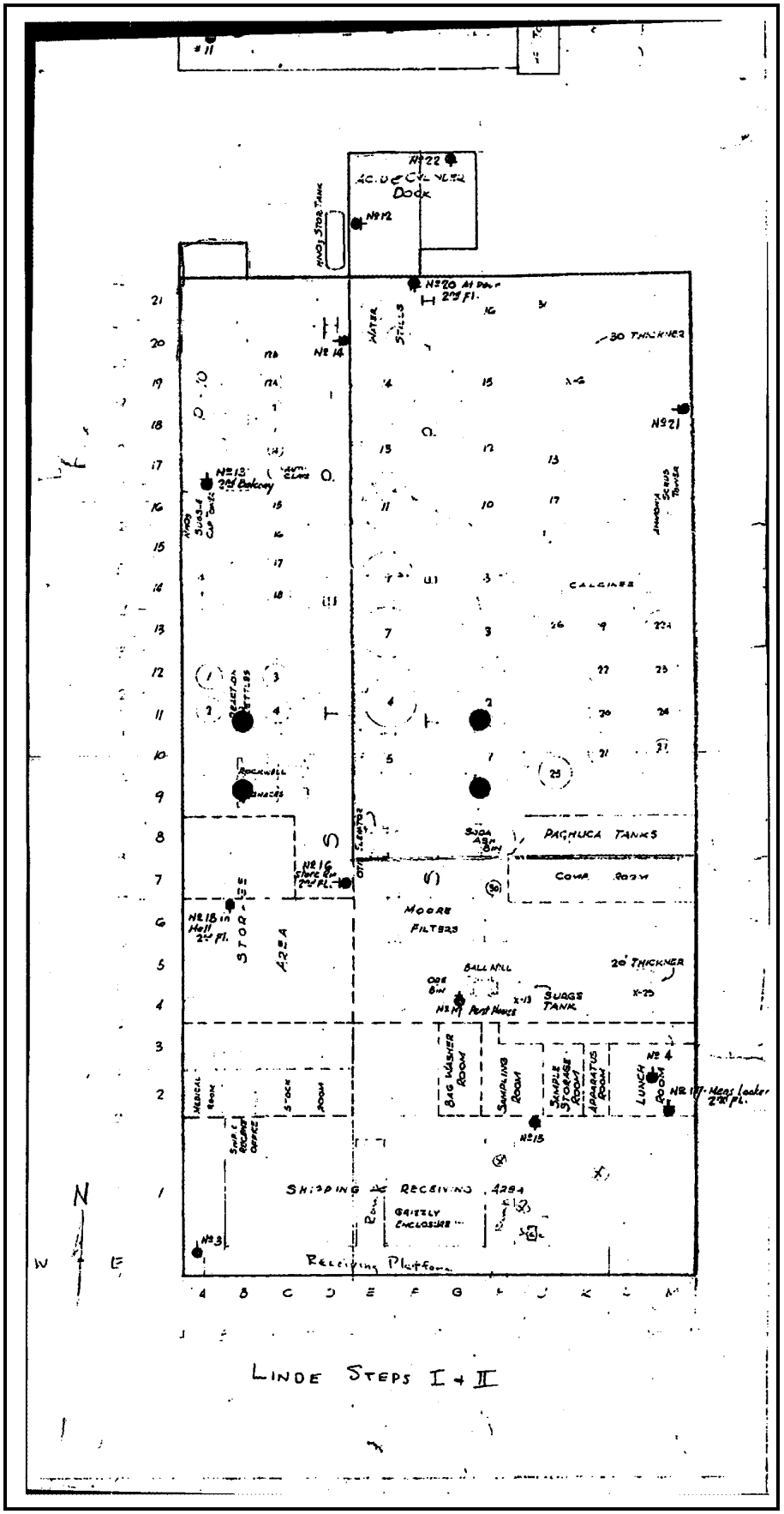

Figure 2-3. Building 30 layout (LAPC undated a).

2.1.5 Building 37 (Ceramics Plant)

Building 37 was a small appendage to Building 39, which was east of Building 38 (Figure 2-1). According to BNI 1993 (p. 1-65), Building 37 was used for Step III. However, no details of this use have been found and its small size, approximately 16 by 36 ft (BNI 1982, Figure B-11), indicates at most a minor role. The 1945 plant drawing labeled it Test Bldg (LAPC 1945a). Radioactive contamination was found in 1976 (ORNL 1978). Linde demolished Building 37 in 1981 (Pilon 2004).

2.1.6 Building 38 (Ceramics Plant)

Building 38 was used for Step III. Figure 2-4 shows its layout. Radioactive contamination was found in 1976 (ORNL 1978). Building 38 was demolished in 1996 (Pilon 2004).

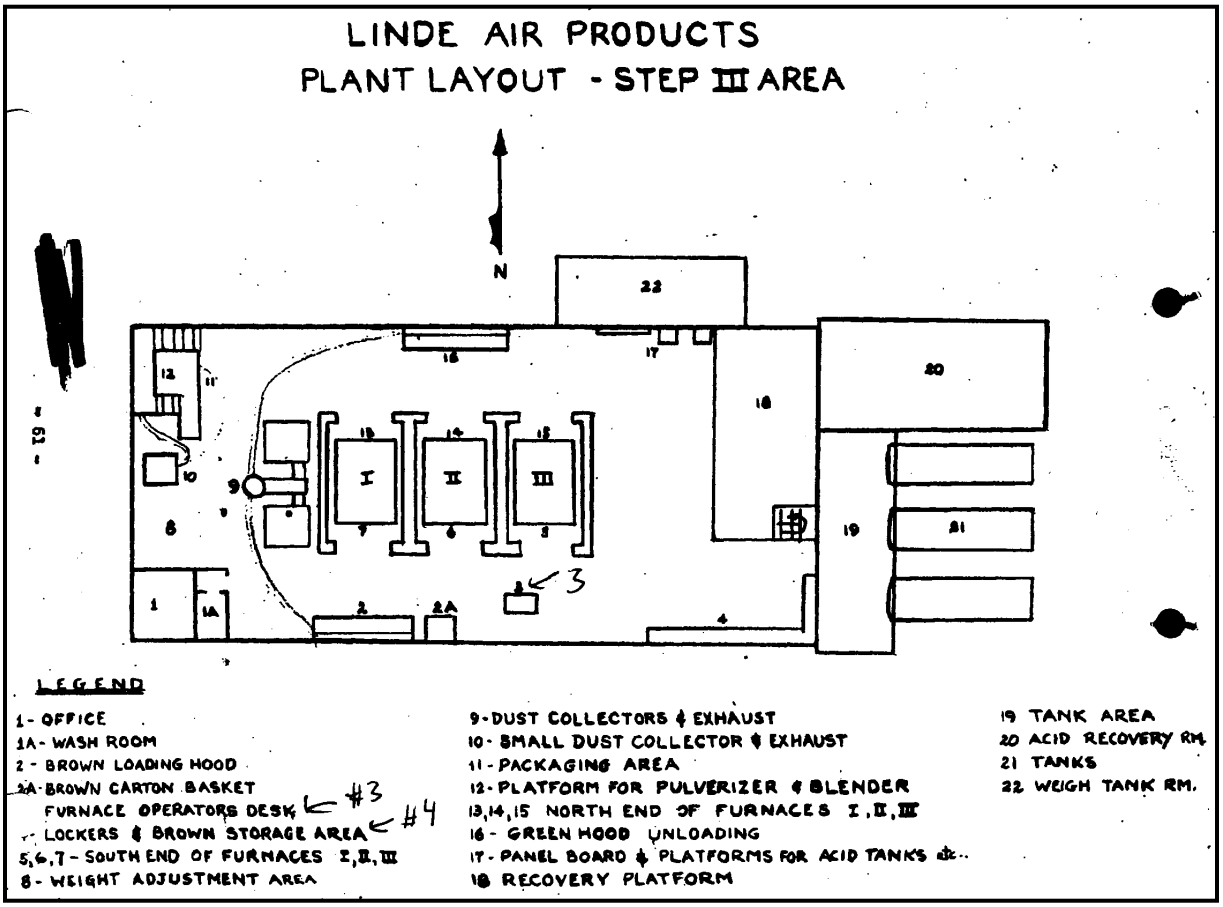

Figure 2-4. Building 38 layout (Klevin 1949a).

2.2 SOURCE TERMS FOR AFRICAN AND DOMESTIC URANIUM ORES

Linde used two types of starting materials for Step I (ore to U_3O_8) processing. These are generally referred to as African ore and domestic ore in Linde-related literature. Most of the African ore was pitchblende; some was torbernite (Wallo, Vierzba, and Roberts 1981, p. 1). Neither African ore underwent chemical preprocessing before being shipped to Linde. Therefore, they contained not only uranium but also all members of the uranium decay series including radium and its progeny.

The domestic ore category included preprocessed domestic ores and byproducts of other uranium conversion processes. The preprocessed domestic ores derived from tailings from vanadium processing. The preprocessing concentrated the uranium and had the effect of removing "the major portion of the radium" in the original ore (Wallo, Vierzba, and Roberts 1981, p. 5). The preprocessing might have also separated uranium from other radionuclides in the ore. If the separation of uranium

had been complete, then the radioactivity in the preprocessed ore delivered to Linde would have consisted primarily of ^{234}U, ^{235}U, and ^{238}U (in their natural abundances) and their short-lived progeny that had grown in after the extraction process. The completeness of the separation of uranium from other radioactivity in the vanadium tailings Linde used is not known. A review of the chemical processing concluded only that "the radium in the domestic ore was significantly less than that in the African ores" (Wallo, Vierzba, and Roberts 1981, p. B-1).

The distinction between the two types of ore is important. The primary radiological hazards from an ore that contained only uranium and its short-lived progeny would be due to alpha and beta emissions. In contrast, radium and other progeny in the African ores would produce, in addition to the alpha and beta emissions, significant gamma emissions and elevated levels of radon. In this document, the term "domestic ore" always means *preprocessed* domestic ore, and is sometimes referred to as "refined ore." "African ore" always means ore that has not undergone preprocessing.

Enriched uranium (up to 35% or less by mass) might have been used on a limited basis for K-25 barrier testing and R&D at Tonawanda Laboratory. Because of the very limited quantities in comparison to the ore, only natural-abundance uranium is explicitly considered for this site.

2.3 ACTIVITIES

LAPC is reported to have begun research into the processing of uranium ore in the late 1930s before it had a contractual relationship with the MED. At the time, uranium was used to make colored glazes for ceramic dinnerware. A former MED employee interviewed in 1983 (Dupree 1983a) said that Linde Research Laboratory in Tonawanda, New York, developed a process to obtain purified U_3O_8 from carnotite ore and produced 80 t of U_3O_8. Linde's capability to provide commercial quantities of U_3O_8 as of 1941 was confirmed in a letter to atomic weapons physicist Leo Szilard (Dana 1941).

On November 16, 1942, LAPC entered into contract W-7401 eng-14 with the MED to design and construct facilities at the Ceramics Plant for processing uranium ores and to produce U_3O_8, UO_2, and UF_4 (Cornell 1942; Marshall 1943). The contract also called for R&D into processes that would be used in the Ceramics Plant facilities.

2.3.1 Uranium Research and Development

Available reports indicated two main areas of Tonawanda Laboratory MED uranium work. The first was development of methodologies for Ceramics Plant operations. A report on the first year of work covered the period from October 1942 to November 1943 (Jenness and Ewing 1943). Most of the work was devoted to study of the problems in the design, operation, and improvement of the Ceramics Plant.

Three different large-scale pilot plant programs were described. As of November 20, 1942, one was producing 2 to 3 t of U_3O_8 per week (Bonsib 1942). The third program, which involved African ore, had begun in October 1943 and was expected to conclude in December 1943.

Available documentation from the Laboratory includes three research reports (Brimm 1943a; Skinner 1944; Wiesendanger 1944) and a final report (LAPC 1946c). The final report discussed work on R-10, a low-grade African ore, and Q-20, a natural (i.e., not preprocessed) torbernite ore. No dates for this work were provided, but because R-10 and Q-20 processing did not begin at the Ceramics Plant until February 1946, the R-10 and Q-20 studies at Tonawanda Laboratory probably were performed in late 1945.

The second area of Tonawanda Laboratory work related to conversion of UF_6 process gas to UO_3. Two research reports (Brimm and Schubert 1945a,b) and a final report (LAPC 1946d) indicated that the associated pilot plant operations used pounds rather than tons of material.

A later Linde research report on uranium processing from September 29, 1948 was prepared by Ceramics Plant rather than Tonawanda Laboratory personnel (Chapman et al 1948). The absence of MED or AEC reports after May 1946 suggests that MED work by Tonawanda Laboratory ended then and that there was no later AEC work. For dose reconstruction, it is assumed that there was no Tonawanda Laboratory work for the MED or AEC after July 31, 1946, the end date of the initial production period at the Ceramics Plant.

2.3.2 Uranium Production

The Linde Ceramics Plant engaged in three different uranium production activities — Steps I, II, and III. Step I (production of U_3O_8) typically involved ores that contained 3% to 20% U_3O_8 by weight. Approximately 70% of the ore was African and 30% domestic by weight. About 26,000 MT of ore were processed, and about 2,300 MT of U_3O_8 were produced (Wallo, Vierzba, and Roberts 1981, Table B-1).

Step I (production of U_3O_8) took place in Building 30 from June 1943 through July 1946. Step II (production of UO_2) took place in Building 30 from April 1943 through March 1944. Step III (production of UF_4) took place in Building 38 from July 1943 to June 1946. Linde sent the U_3O_8 from Step I to DuPont and Mallinckrodt after Step II production ceased (Gates 1946). Linde received UO_2 from other companies for Step III processing (AEC 1949a). Tonawanda Laboratory might have investigated all of these processing steps in its pilot plant. Uranium production was in standby after July 1946. Preparations for Step III resumption began in September 1945. Step III operations were resumed under contract AT-30-1-GEN-165 in November 1947 (Rennich 1947). Linde ended uranium production for the AEC in June 1949.

A MED record-of-negotiations memorandum (Dreveskracht 1945) noted that Linde had offered for sale 36,000 lb of uranium ore concentrates from Colorado and 8,900 lb of residue, which were the "property of Contractor as a result of operations before his present CPFF [cost plus fixed-fee] work for the MED (Contract No. W-7401 eng-14)." The transfer was made under contract W-17-028 eng-29 on May 4, 1945 (USEO 1945). Correspondence indicates that Linde transferred ownership of the uranium concentrates and residues to the government on August 1, 1945 (Martin 1945), and the material was subsequently processed in the Ceramics Plant for MED (Dupree 1983a, Robinson 1945).

Detailed descriptions of the typical Ceramics Plant uranium processing methods are presented below. These include descriptions of chemical and operational processes.

2.3.2.1 Step I (Uranium Ore to U_3O_8)

Step I extracted purified U_3O_8 (black oxide) from uranium ore. The Step I process varied with time as the nature of the ore changed and as Linde gained experience with the procedure.

Domestic ore was ground and mixed with water to form a slurry. Sulfuric acid and other chemicals were added, and the mixture was digested at 90°C for 2 to 3 hours and then cooled to 60°C. This left the uranium in solution as uranyl sulfate along with some of the impurities. Soda ash and sodium bicarbonate were added to make the solution basic. This precipitated most of the remaining impurities and left the uranium in solution as sodium uranyl tricarbonate. The slurry was filtered in Moore filters (described in the next section). The liquors contained the uranium and some objectionable impurities. Vanadium and phosphorous impurities were precipitated by the addition of ferrous and ferric sulfates and then removed in a second filtering in plate and frame presses, which produced "iron cake." The liquors were treated with caustic soda to precipitate the uranium as sodium diuranate, which was removed as cake in a third filtering process. The sodium diuranate cake was treated with sulfuric acid and ammonium sulfate to convert it to an ammonium uranyl sulfate complex.

This was removed in a fourth filtering process. The resulting "acid leach cake" was fed to a calciner to drive out the ammonia, sulfur dioxide and trioxide, and water to leave the black oxide (U_3O_8).

A modification of the above procedure was used to treat African ore. The filter cake from the Moore filters (first precipitation step) contained the radium in the ore. In the second step, instead of iron sulfates, sodium sulfide was added to remove lead. Some of the other chemical additives that were used at various points were different (LAPC 1946e; Aerospace 1981, p. A-1).

Step I Operations

Ore was delivered by rail in boxcars. African L-30 pitchblende arrived in burlap bags inside paper bags. Each ore bag weighed 50 to 80 lb. For African ores, the car doors were left open for 12 to 24 hours before unloading to reduce radon concentrations in the car. Either ore bags or bulk ore (the practice varied) were manually loaded onto wheeled carts (buggies) by workers called loaders or movemen and transported to a storage or processing area (Olevitch 1944; LAPC 1946e; Cranch 1944a).

For dry ore, processing began in the dumping room on the receiving platform. Ore was dumped through a grate onto a conveyer belt that carried it to the ore storage bin. The storage bin held about 75 t of ore. Ore from the bin was fed to a ball mill by a Feedoweight (a belt conveyor that weighed the ore). After milling, the ore was fed to the digestion process.

Wet ore was transported to the second floor using the freight elevator and either dumped into the scoop box of the ball mill or directly into digest tanks (skipping the ball mill step). Loaders did not ride on the elevator with the ore.

After the dumping, about a 0.5 lb of ore remained in each bag. To recover it, the bags were shaken in a bag shaker over a conveyor belt onto which freed ore dropped. The bags were then washed, centrifuged, and burned in an incinerator. The washed bags contained about 10% to 30% of the remnant ore. At times, washed bags that were awaiting incineration were stored outdoors on the site. A July 1944 report noted that 19,000 bags were currently being stored (Olevitch 1944, Appendix A).

For digestion, chemicals appropriate to the ore type were added to the digest tanks, and the mixture was heated and cooled to precipitate components. Precipitates were separated from solutions using various filter assemblies. The first filtration in the Step I processing used the Moore filters. A Moore filter consisted of a large basket that contained 24 rectangular leaves with each leaf approximately 35 ft^2 in area. Each leaf was formed by a rectangular perforated pipe frame over which a bag was stretched. The filter basket was placed by crane into a neutralized digest tank and held there while suction was applied to the pipe. Gelatinous cakes about 1-in. thick formed on the leaves. The basket was removed from the digest tank, placed in a wash tank, washed, and then moved to a hopper. The suction was then reversed to inflate the bags and cause the cake to fall into the hopper. From there the cake was transferred to an open-top truck below the hopper for transport to a storage location. After sampling, the filter cake was hauled away by truck. Radium-containing filter cake from the African ores was shipped to the Lake Ontario Ordnance Works (LOOW) in Lewiston, New York, for storage. Filter cake from the domestic ores went to a tailings pile. From some filter presses, the filter cake was manually dumped by the operators into buggies for transport to the next step. From other presses, the cake could be fed automatically to the next step (MED undated, page 3; Olevitch 1944; LAPC 1946e; Aerospace 1981, p. A-1).

The tanks used in the process were large: 18 to 31 ft in diameter and one story high (Dupree 1983a,b).

2.3.2.2 Step II (U_3O_8 to UO_2)

Step II converted U_3O_8 to UO_2 (brown oxide) with UO_3 (orange oxide) as an intermediary product.

Black oxide (U_3O_8) was dissolved in nitric acid. Some insoluble impurities were filtered off, and the solution was evaporated to molten uranyl nitrate hexahydrate. The next step was based on ether extraction of the uranium. The extraction took advantage of the fact that metallic salts are more soluble in water than in ether, but the uranyl nitrate is nearly equally soluble in either and can be shifted from one medium to the other by changes in volume and concentration ratios. The uranyl nitrate hexahydrate was added to cold ether, and the solution was washed with water. The impurities went to the water layer and the uranyl nitrate to the ether layer. The purified uranyl nitrate was denitrated to powdered uranium trioxide (UO_3) by heating in a stainless-steel pot furnace. The trioxide powder was collected mechanically and flowed by gravity through a micropulverizer into a rotary tube reduction furnace. There the powder was reduced to the desired uranium dioxide (UO_2) using gaseous hydrogen as a reducing agent. The finished UO_2 product was packaged, sampled, sealed, and stored (LAPC 1946a; Aerospace 1981, p. A-1).

Step II Operations
U_3O_8 in buggies from Step I was weighed and hoisted to the second floor. It was then fed into nitric acid in digest tanks with scoops. The digest tanks were heated to 90°C. After digestion, insoluble impurities were filtered out using a filter press. Liquids were piped from one vessel to another. Frequent chemical analyses of samples were required in this process to determine the progress of chemical reactions. The processing produced various liquors and some associated cakes that were designated “OK” or “NG” (LAPC 1946a).

2.3.2.3 Step III (UO_2 to UF_4)

Step III converted UO_2 to UF_4 (green salt) using the chemical reaction

$$UO_2 + 4HF \rightarrow UF_4 + 2H_2O$$

The reaction was carried out at 1,000°F. Gaseous anhydrous HF gas was flowed over the UO_2, which was placed on magnesium trays stacked on spacer bars inside 9-in.-square tubes 10 ft long that were inside a furnace (LAPC 1946b; Hickey, Crawford-Brown, and Tankersley 1988, p. 8).

Step III Operations
UO_2 was weighed and hand-troweled into shallow trays inside a hood with a dust collector operating. Twelve pounds of UO_2 were placed on each tray. The oxide was furrowed when loaded to maximize surface area. The trays were transported on buggies and inserted into the furnace.

After loading, the furnace was sealed, purged of air, and heated to the starting temperature. Then the flow of HF gas was started. After the required number of hours, the heat and gas flow were stopped, and the furnace was purged of HF gas, cooled, and opened.

The trays were removed from the furnace and hand-placed onto buggies for transport to and placement inside an unloading hood with a dust collector. Good material (light green) was loaded into hoppers. Bad material (dark green) was placed in fiber-pack drums for later retreatment. Good material was pulverized, blended, sampled, packaged, weighed, and shipped (LAPC 1946b; Hickey, Crawford-Brown, and Tankersley 1988, p. 8).

2.3.3 Nickel Production

After the shutdown of Step II in 1944, the Step II equipment was converted for use in production of nickel powder, a nonradioactive process, which occurred from October 1944 through February 1946 (Hickey, Crawford-Brown and Tankersley 1988, pp. 9 and 13). The nickel was for use in the K-25 diffusion barrier (Hunter 1949; Dupree 1983b, p. 3).

There is discrepant evidence on the location of the nickel processing work. A contractual record (USA 1945) states that Linde was contracted on April 7, 1944, via letter contract W-7418 eng-51 to construct and equip a plant to produce 80 t of special nickel oxide per month and that the plant was to be constructed in Tonawanda on the same premises and adjacent to the buildings that were used in connection with contract W-7401 eng-14 (the contract for the uranium refining work). A drawing that was attached to a former Tonawanda Area Engineer Office employee interview record (Dupree 1983a) shows the location of the nickel processing to be in a building east of Building 30. Therefore, nickel processing might have occurred in Building 31 (see Figure 2-1). However, another record of interview (Dupree 1983b) states that when Step II operations in Building 30 ended, the Step II area was used to refine nickel. Both recollections could be correct. It could be that nickel processing initially occurred in the Step II area and later was moved to Building 31.

Evidence that supports Building 31 as at least the ultimate location for Linde's Tonawanda nickel processing is found in a 1945 plant drawing (LAPC 1945a). The visible portions of a partially obscured label on the drawing are consistent with the interpretation that it says “Nickel Plant.” In addition, the drawing shows an ammonia cracking area in Building 31, and other records indicate that ammonia was used in the nickel processing [chemical operators in department C2P used ammonia according to LAPC (1945b, p. 17), and department C2P engaged in nickel processing according to Hickey, Crawford-Brown, and Tankersley (1988, p. 28)].

Nickel processing at the Ceramics Plant has been described as follows (Hickey, Crawford-Brown, and Tankersley 1988, p. 9):

> *Metallic Ni slugs (small ingots) were oxidized with $HN0_3$, filtered, and heated to 1200 C to produce a nickel oxide (NiO). This was pulverized, mixed with ammonium chloride (NH_4Cl), and heated to form a porous mass of the oxide, which was reduced to powdered nickel by exposure to H_2. The powder was pulverized and packaged. … It is not known whether both NiO and powdered Ni or only NiO were produced in quantity at this plant.*

2.3.4 Other Radiological Activities

Early on, radioactive liquid wastes were discharged to the Tonawanda sanitary sewage system. Due to the nature of the liquids, this became a problem, and Linde began to dispose of liquid wastes in onsite wells that sometimes overflowed. Later still, liquid wastes were discharged to a drainage ditch that led to a sewer conduit (BNI 1993, pp. 1-9 to 1-15).

After MED work began at the Tonawanda site, there was potential exposure of workers to radiation and radioactivity when outdoors. Portions of the site are known to have been contaminated with radioactivity (Heatherton 1948b; ORNL 1978; BNI 1982); resuspension would have produced airborne radioactivity. One source of ground contamination and airborne radioactivity was the ore unloading process, which involved transporting ore in buggies – sometimes in bulk and sometimes in bags – from boxcars to Building 30 (see Step I Operations in Section 2.3.2.1). Outdoor areas of the site were sometimes used for storage of radioactive materials. Olevitch (1944) reports the outdoor storage of contaminated ore bags that at times numbered in the thousands. In 1948, 1 mR/hr gamma and 3,000 α dpm/100 cm^2 were measured from the soil in an area formerly used for storage of radioactive materials (Heatherton 1948b).

An additional source was the release of liquid effluents either to onsite wells that sometimes overflowed, or to an onsite drainage ditch (BNI 1993, pp. 1-9 to 1-15). Airborne effluents from the plant were an additional source of outdoor radioactivity.

In 1949, Linde workers unloaded drums of K-65 product that was shipped by rail from Mallinckrodt in St. Louis to the LOOW at Modeltown, New York (Wolf 1949a; Heatherton 1949a; Kelley 1949). Linde film badges were worn during this work.

At the April 18, 2005, Worker Outreach Meeting, strontium was mentioned. At the June 27, 2005, Worker Outreach Meeting, it was clarified that the strontium was in the form of a ^{90}Sr sealed source. This source was reportedly brought to the site in the late 1950s. The estimates of uranium exposure based on estimates of exposure periods and source term, which were based on worst-case assumptions when a parameter was not well supported by available information, would be sufficiently bounding to account for small amounts of radioactive strontium.

2.3.5 Utility Tunnel Complex

Description

The Linde site had a system of utility tunnels, constructed at different times, to carry steam, electricity, water, telephone lines, and other utilities from one part of the plant to another (LAPC ca. 1937, 1957, 1961; Coutts 2005). Documentary evidence shows that the first tunnel section was built in 1937 and ran from the powerhouse (Building 8) past the Tonawanda Laboratory (Building 14, also called the Proving Laboratory) to Building 10. This part of the tunnel complex was not found to be contaminated with radium because the pre-World War II uranium-refining operations in Building 14 used preprocessed ores from the Colorado Plateau vanadium deposits, which did not contain radium. Borehole data from the site cited in the 1978 and 1982 Formerly Utilized Sites Remedial Action Program (FUSRAP) surveys confirmed that no excess radium was found in the soils near this section of tunnel (ORNL 1978; BNI 1982). Another section of tunnel was constructed in 1957 near Buildings 57, 58, and 31 in the northeastern area of the Ceramics Plant, and an extensive addition to the tunnel system was done in 1961 when the 1937 and 1957 tunnels were linked by new tunnels that ran between Buildings 30 and 31 and then branched south to Building 8 and west past Buildings 70, 2, and 2A. The 1957 and 1961 tunnel sections ran through areas of soil that were contaminated by radium-bearing ore and were subject to radon infiltration from this source.

The underground tunnels housed gas, electrical, water, steam, telephone, and other utility lines. They had a maximum dimension of about 8 to 10 ft high and 15 ft wide. They also had exhaust fans (approximately 6 ft in diameter) in some sections of the tunnel that pulled air through the tunnels (Murray 2010a,b,c), and the average air exchange rate in any particular tunnel section was estimated by the USACE as approximately once every 10 hours. The tunnels had few stairway access points and several locations where covered manways could be used to access the tunnel (USACE 2002b). These tunnels were routinely inspected and accessed for maintenance activities. However, no uranium process piping was ever installed in these tunnels (USACE 2002b).

The processing of uranium ore in the 1940s resulted in contamination of the site building surfaces, soils, building sumps, and sanitary sewer system. In addition, approximately 55 million gal of liquid waste effluent was injected into seven wells that were used for liquid effluent disposal from February 27, 1945, through mid-July 1946. The wells were in two general areas: three near Building 8 and four in the area of Buildings 30 and 38. The wells were drilled approximately 150 ft (40 m) deep through a clay formation into a gravel and sand layer and the bedrock layer. The liquid effluent was injected down into the wells between 27 to 46 m (90 to 150 ft) below the surface at a temperature of approximately 60°C (140°F). The effluents primarily contained ions of sodium, sulfate, sodium carbonate, sodium bicarbonate, and chloride. Minor concentrations of vanadium, cobalt, nickel, molybdenum, uranium, and radium were also present. This liquid had a high pH (greater than 10) and a high concentration of total dissolved solids (greater than 20,000 ppm). Problems with clogging of the wells and fluid refusal that were reported during the injection process were most likely to have been caused by the formation of precipitates in the bedrock fractures and pore spaces of the sediments. The precipitates were formed when the high-temperature high-pH effluent containing dissolved solids became mixed with the cooler-temperature, natural formation water, which resulted in

the precipitation of most of the dissolved metals (including uranium and radium) from the solution. Under high-pH conditions, thorium hydroxide, barium sulfate, calcium carbonate, and calcium sulfate would also have precipitated. These precipitates contributed to the pores becoming clogged during effluent injection. It has been suggested that runoff water contaminated from these sources entered the service tunnels throughout the lifetime of the site and resulted in a constant contamination level over time.

Limited radiological characterization of these areas was conducted during the 1978 and 1981 FUSRAP surveys of the Linde site (ORNL 1978 and BNI 1982). A detailed characterization of these tunnels was performed in the summer of 2001 (USACE 2002b).

Area Geology and Hydrology

Sections 3.4 and 3.5 of the 1993 remedial investigation report for the Tonawanda site (BNI 1993, pp. 3-21–3-42) present information on the Linde site geology including structural features, physical and hydrogeologic properties, and the geochemical properties. Section 3.5.5 of the report describes, in some detail, the soil interval between the ground surface and the water table, known as the vadose zone, and how flow occurs through this layer. The report made a conservative estimate of vertical groundwater velocity and applied a retardation factor to account for the slower movement of contaminants through the vadose zone due to interactions between the contaminants and the soil particles. The report states the uranium contamination in groundwater would migrate 47 times slower than the estimated vertical groundwater velocity of 0.09 m/yr (0.3 ft/yr). Because the retardation factors for thorium and radium are typically greater than for uranium, thorium and radium would migrate at a slower rate.

The Aerospace report (Wallo, Vierzba, and Roberts 1981) indicates the convoluted rock beds in the area have a slight southerly slope, about 28 ft/mi. Due to the composition of the formations in the area, it is very difficult to predict the direction and flow rate of water at any given point in the formations. In carbonate rock, the solution openings that are caused by water dissolving the carbonate rock are irregularly distributed both horizontally and vertically.

Tunnel Access

The utility tunnels would have been routinely accessed by inspectors and maintenance personnel throughout the site history. There is only limited information on the number of hours a worker might have worked in these tunnels but, according to USACE (2002b), approximately 2 months a year are currently needed for annual maintenance (sump pumps replacement, condensate pump repair, lighting repairs, etc.). The report also provides an estimate of effort required for tunnel inspection each year. This estimate is supported by worker interviews (Murray 2010a,b,c). There were stairwells from the tunnel up to Buildings 2, 8, 14, and 30 (Building 30's access was by means of a ladder at Junction Box 6), but it was not general practice for employees to use them to get from one building to the other, and the practice was not condoned by the company (Murray 2010c). However as a favorable to claimant assumption, it is assumed that all other personnel may have used the tunnels for transit. A transit occupancy time of 10 minutes per workday (wd) is assumed.

Uranium in Liquid Effluent

Uranium losses in the effluents were monitored by Linde. From mid-1944 until 1946, the losses were reported on a regular basis in the weekly progress reports and, before that, in memoranda or occasional progress reports. The weekly averages of analyzed uranium oxide concentrations in the effluents from April 1944 to July 1946 (from progress reports) ranged between 0.011 and 0.064 g of uranium oxide per liter of effluent, with the average being about 0.026 g/L (Wallo, Vierzba, and Roberts 1981, p. 11). This would equate to an average process loss of about 26 kg of uranium oxide per million L (220 lb per million gal) of effluent during the entire period when the wells and storm sewer were being used for disposal (February 27, 1945, to mid-July 1946).

Concentrations of uranium oxide in the effluent were somewhat higher during the period when the sanitary sewer was used for disposal of the effluent. It is estimated that the concentrations averaged 0.15 g/L in 1943 and 0.03 g/L during the first 3 months of 1944, or about 1,200 and 250 lb of uranium oxide per million gallons, respectively.

Assuming these loss rates and the volume of effluent that was disposed of in the wells (55 million gal), approximately 5,400 kg (12,000 lb) of uranium oxide (about 3 Ci of natural uranium) were discharged to the wells. The remainder of the process effluents that were discharged to the storm sewer during this period (approximately 212 million L or 56 million gal) would have contained approximately 5,600 kg (12,000 lb) of uranium oxide (again, about 3 Ci of natural uranium). These loss rates are consistent with the assumption that Linde maintained uranium losses in the effluents to about 0.5% of the total uranium present. If, as reported, 1,812 MT (1990 t) of uranium oxide (at about 97% extraction efficiency) were produced from April 1944 to July 1946, then about 9.3 MT (about 10 t) of uranium oxide (5 Ci of natural uranium) were lost in the effluents, which is in good agreement with the values estimated by using the effluent concentration data. The 1981 survey (Frame et al. 1981) estimates the maximum concentration of uranium in the effluent released to the disposal wells and storm sewers at 4×10^{-5} µCi/mL.

Radium in the Liquid Effluent

Estimates of the maximum amount of radium that was discharged during the processing of the L-30 and L-50 ores can be made based on the contractually specified return of at least 95% of the radium from the processed ore to African Metals. Processing operations reportedly held losses to less than 3%, meaning 97% of the radium remained in the residues. Given a total of 986 MT (1,080 t) of U_3O_8 that was produced from the L-30 and L-50 ores and a uranium extraction efficiency of about 97%, then 862 MT (950 t) of uranium, or less than 595 Ci of natural uranium, were in the ore. This equates to an estimate of 290 Ci of ^{238}U in the ore. Assuming the ^{226}Ra was in equilibrium with the ^{238}U, the maximum effluent losses were calculated to be 8.5 Ci of ^{226}Ra. A similar analysis for the R-10 ore, but assuming a 95% extraction efficiency, suggested that a maximum of 2.7 Ci of radium was lost during the processing of this ore (Wallo, Vierzba, and Roberts 1981, p. 14).

Based on data in the progress reports, approximately 169 to 173 million L (about 45 million gal) of effluent were released during L-30 and L-50 processing, and about 77 million L (about 20 million gal) were released during R-10 processing. Based on this volume, the maximum concentration of radium in the effluent would be about 4.6×10^{-5} µCi/mL. This value represents the maximum radium content in the uranium carbonate solutions and was based on the amount of radium that was not precipitated on the Moore filters after the digest circuit. Of the unprecipitated radium at this stage, a portion probably precipitated during the liquor purification (lead, vanadium, and /or phosphorus removal step), but the largest fraction of the remaining radium would be expected to have precipitated with the uranium after the addition of the caustic during the soda salt precipitation step. The remainder would be discharged with the effluents.

A letter from the University of Rochester to the MED suggested that the radium content of the filtrates from the processing of R-10 ore was “on the order of 0.25×10^{-8} grams/liter” (2.5×10^{-6} µCi/mL). This value is less than 1/18 of the maximum value that was estimated in the analysis above, which supports the theory that much of the radium that escaped the Moore filters could have been precipitated during the liquor purification or the uranium precipitation and therefore was not discharged to the wells or storm sewer. The implications are that the radium that was released in the effluents from the African ore processing could be on the order of 0.6 Ci rather than 11.2 Ci (Wallo, Vierzba, and Roberts 1981, p. 15).

Soil Analysis near Injection Wells

During the remedial investigation (BNI 1993, Section 4) data was collected to determine the nature and extent of contamination at the Linde Ceramics facility. Borehole samples were analyzed, including borehole B29R23 just to the northwest of Building 30 near a former injection well that was

found just inside the building. This borehole was sampled at three intervals (depths), and the samples showed low-level radium-226 concentration in the first interval (BNI 1993, pp. 4-24 - 4-25), possibly as a result of spilled effluent at the site of the injection well. The concentration decreased at the deeper sample interval.

2.4 PROTECTIVE MEASURES

Steps were taken to protect against radiological and chemical hazards during processing (LAPC 1945, 1946a,b,e; Dupree 1983a,b). Although some protective measures were included early in the Linde program, protective measures were upgraded over time in response to both survey findings and newly available information about biological risks of exposure. These measures included limiting the time a worker was allowed to spend in a radiologically hazardous activity, provision and upgrading of ventilation to reduce concentrations of radon and radioactive dusts in the air, and good housekeeping to reduce radioactive dust on the floor. Respirators to reduce dust and chemical exposure and gloves to reduce beta exposure to the hands were required for some of the more hazardous tasks, but these safety devices were not used in all process areas by all workers. Some workers changed into company-issued clothing when they arrived at work and changed back to their own clothing at the end of a shift. Showers were available. At some point in Linde operations, showering at the end of a shift became mandatory for some workers, and they were "wanded" by a health physicist to check for radioactivity when they left the shower.

Chemical protective measures included goggles, respirators, rubber gloves, face shields, rubber or plastic-coated aprons and sleeves, and head caps. There were more substantial protective measures for operators who worked with hydrofluoric acid.

According to interviewees, use of respirators was often at the worker's discretion, but they were "usually used" (Dupree 1983b). Inspectors reported mixed compliance with recommendations to wear respirators. The author of a June 1944 inspection report commented, "The wearing of respirators remains difficult to enforce" (Ferry 1944a). The author of an October 1944 inspection report observed, "Respirator discipline was about 80% efficient in the dumping room" (Tybout 1944a). In 1945, the same inspector witnessed the collection of dust samples by Mr. H. Seemann, Assistant to the Safety Engineer. In his report, the inspector noted, "The men wore respirators, but Mr. Seemann stated that this was the case only because he was present" (Tybout 1945a). A 1948 dust sample collector commented that when samples were collected, individuals on the day shift wore respirators but those on the evening shift did not (Hayden 1948).

No credit for safety gear (including respirators) is taken in this document.

2.5 PERSONNEL, JOB CATEGORIES, AND WORK HOURS

For Tonawanda Laboratory, two listings of employees might aid in determining whether a claimant worked there. A September 1942 Laboratory directory (LAPC 1942) lists approximately 200 employees. Several employees are linked to the Proving Lab, which might be an indicator of higher exposures once the uranium work began. A 1944 employee list (LAPC 1944a) contains approximately 120 names and similar information. It is not clear whether this list includes all employees.

For the Ceramics Plant, an April 1944 employee list contains nearly 400 names (Neuman 1944) and specifies the workers' jobs. A December 1945 job description specifies the duties of 51 categories of workers and lists a department code for each position (LAPC 1945b). Because the same title (e.g., chemical operator) was sometimes used in different departments in which the nature of the work was very different (e.g., Step I and nickel processing), knowing the department might help identify the type of activity in which a worker was involved. Table 2-2 shows Ceramics Plant department codes that were associated with various worker activities as determined in a 1988 study of worker hazards

(Hickey, Crawford-Brown, and Tankersley, 1988). A 1949 study reported that there were 139 Step III personnel and listed the number in each job. Names of some of the personnel were provided (Klevin 1949a, pp. 6, 80).

Table 2-2. Department codes for various worker activities.

Department	Department code[a]
Administrative services	AB, AD, AL, AM, BA, BB, BT, A/M
Chemical control, R&D	BL, HL
Engineering and development	AW
Maintenance	AN
Process operations: Step I	CL
Step II	CM
Step III	CN
Nickel	CP, CR, C2M, C2P
Loading: general	AN, AH, LOOW
Janitors	AA, GE
Stores and supplies	AS, ASA, A8A, AT
Safety and security	AJ, A9A, A2J
Undetermined activity	CA, CS
Untitled jobs and departments	(b)

a. From Hickey, Crawford-Brown, and Tankersley (1988, p. 27). Based in part on a plant document identifying some of the codes and in part on inference from less direct information in plant records.
b. Not available.

The term “day” in this document refers to a calendar day. The term “workday” (wd) is used to describe a day at work. Default assumptions are 250 wd/yr and 8 hr/wd: this results in 2,000 hours of work per year. The distinction between workday and calendar day is especially important when considering internal dose rates for use in calculating organ doses, because intakes actually occur during the workday, but dosimetry calculations are usually based on integration over calendar days.

The Ceramics Plant work schedule during the production period (1943 to 1946) is described as involving 8-hour shifts 6 d/wk (Dupree 1983a,b,c). A 9-hour workday with a lunch period included is assumed because there is evidence that many employees worked overtime (Dupree 1983b, p. 4; MED undated, p.1). An 8-hour workday plus a half-hour for lunch is known to have been in effect for many of the workers in late 1948 (Klevin 1949a). The transition from a 6-day workweek to a 5-day workweek was assumed to have occurred on January 1, 1951, based on a report by interviewees that the 48-hour workweek lasted until 1950 (Dupree 1983b, p. 4).

Table 2-3 provides a list of some Linde references that include worker names and sometimes additional job information.

Table 2-3. Worker information.

Item	Reference
Tonawanda Laboratory employees and room numbers, September 2, 1942. Job titles are not stated.	LAPC 1942
Linde employees associated with contract W-7401-eng-14, July 20, 1943. Some are Tonawanda Laboratory personnel. Some might be Ceramics Plant personnel. Job titles are included.	Pew 1943
Ceramics Plant personnel with job titles, April 25, 1944.	Neuman 1944
Ceramics Plant Step III process operators, June 14, 1948.	Heatherton 1948c
Ceramics Plant Step III process area personnel (including operators, millwrights, and stores attendants), January 25, 1949.	Klevin 1949a, p. 50
Film badge exposure report listing 23 Step III personnel by name and job title, for week beginning March 28, 1949.	AEC 1949b
List of Ceramics Plant Step III process operators, August 26, 1949.	Klevin 1949b, p. 11

Based on the above, the periods and work schedule at the Ceramics Plant from the onset of MED operations through December 31, 1953, were assumed to have been as shown in Table 2-4.

Table 2-4. Ceramics Plant assumed work schedule (including lunch and breaks), 1942 to December 31, 1953.

Period	Start	End	hr/wd	wd/wk	wk/yr
Preproduction	10/01/1942	04/26/1943	9.0	6	50
Production	04/27/1943	07/31/1946	9.0	6	50
Standby	08/01/1946	09/14/1947	8.5	6	50
Rehabilitation and production	09/15/1947	06/30/1949	8.5	6	50
Cleanup	07/01/1949	12/31/1950	8.5	6	50
Cleanup[a]	01/01/1951	12/31/953	8.5	5	50

a. Assumed date of transition from 6- to 5-day workweek (based on Dupree 1983b, p. 4).

2.6 DECONTAMINATION DURING THE MED/AEC CONTRACT PERIOD

Cleanup and decontamination activities at the Ceramics Plant facilities during the MED/AEC contract period are discussed below. No information was found on cleanup of Tonawanda Laboratory facilities, but it was noted at the April 18, 2005, Worker Outreach Meeting that Building 14 was contaminated, and later surveys indicated that residual contamination remained after AEC work (BNI 1993).

Cleanup of the Ceramics Plant began before the shutdown of Step III production, which occurred on June 30, 1949. Some Step II equipment in Building 30 was removed in March and April of 1948 by or with the assistance of a contractor (H. K. Ferguson) (Heatherton 1948d,e,f). Dismantling of Step I and remaining Step II equipment in Building 30 was under way in May 1949 (Heatherton 1949b). A contractor (Kulp Waco) assisted in some of the work (Heatherton 1949c,d).

Shortly after the shutdown, a comprehensive cleanup effort was undertaken to reduce levels of radioactivity in Building 30 to enable its release to Linde for unrestricted use (Heatherton 1950). After removal of the bulk of the process equipment, the entire building was vacuum-cleaned and flushed with water. A systematic radiation survey was conducted to identify areas of contamination. Decontamination was accomplished primarily by removing contaminated parts of the building (such as portions of the second floor balcony on which process operators had been stationed) and by abrading surfaces (mostly by sandblasting, although oxygen acetylene torches were also used).

After each area was decontaminated, it was again cleaned and flushed, and then a final radiation survey was made.

Workers who performed surface abrasion activities were provided with U.S. Bureau of Mines (USBM)-approved rubber hoods that protected the face, neck, and shoulders and had a supplied air respirator to permit breathing of uncontaminated air. Other personnel in the area wore demand respirators or, for short periods, USBM respirators approved for all dusts.

An AEC internal memorandum from March 29, 1950 reported that the decontamination of Building 30 had been completed and indicated that a decision on release of the building to Linde was to be made by April 1, 1950 (Eisenbud 1950).

Decontamination of Building 38 began with a radiation survey in November 1952 (Harris 1952). An AEC memorandum from April 1954 (Harris 1954) reported levels after decontamination. Average readings on categories of equipment to be left at the plant ranged from 0.05 to 0.6 mrep/hr beta plus gamma with the gamma component either undetectable (which meant less than twice background) or less than 5% of the total. The memorandum reported “overall” floor and wall levels to be 1.01 “mrep/hr/ft^2” beta plus gamma. This was considered excessive. The memorandum cited data

that indicated this could be reduced to 0.065 mrep/hr/ft^2 by covering contaminated floor areas with asphalt tile. It recommended release of the building to Linde once this occurred.

No information was found on cleanup and decontamination activities in Buildings 31 and 37.

2.7 POST-MED/AEC OPERATIONS

FUSRAP began in 1976. ORNL surveyed the Linde Tonawanda site from October 18 through November 5, 1976, to determine if remediation would be required (ORNL 1978). Radiation and radioactive contamination measurements were made inside Buildings 14, 30, 31, 37, and 38; on the Tonawanda property outside the buildings; and at nearby offsite locations. Linde employees noted that Building 30 renovation occurred in the 1960s and could have resulted in elevated employee radiation exposures. Notes from the Worker Outreach Meeting on April 18, 2005, mention contamination in association with Building 57, and an additional review of the BNI (1993) remedial investigation report shows areas of residual radioactive contamination were associated with areas in or near Buildings 57, 58, and 90. The highest indoor radiation levels were found in the principal production buildings, Buildings 30 and 38. Linde was designated as a FUSRAP site in 1980. Additional radiological surveys and decontamination efforts followed (BNI 1993). These led eventually to demolition of Building 14 and all of the Linde Ceramics Plant buildings that were involved in MED/AEC work except Building 31. Table 2-1 shows demolition dates. As of 2004, Building 31 remained in use and onsite soil remediation was in progress with completion scheduled for 2007 (Pilon 2004).

3.0 ESTIMATION OF INTERNAL EXPOSURE, NOVEMBER 1, 1947, TO DECEMBER 31, 1953

This section develops parameters for reconstruction of doses due to internal exposures from November 1, 1947, until December 31, 1953. DHHS has determined, and NIOSH has concurred, that it is not feasible to reconstruct internal exposure that occurred before November 1, 1947 (Leavitt 2005).

This section describes information related to internal dose reconstruction including Linde's uranium bioassays, airborne radioactivity concentration measurements, radon breath analyses for determination of radium body burdens, and radon air concentrations. To expedite preparation of this document, the Linde information is considered in conjunction with information from facilities that did similar types of uranium processing to establish preliminary estimates of internal intakes and exposures. These estimates are considered best estimates, until data can be further considered. It is believed that additional analysis of the data will lower at least some of the intakes and exposures that are estimated in this section.

This document assumes that Tonawanda Laboratory primary internal exposures occurred from October 1, 1942, to December 31, 1946. At the Ceramics Plant, four periods were assumed: October 1, 1942, to July 31, 1946 (operations), August 1, 1946, to September 14, 1947 (standby), September 15, 1947, to October 31, 1947 (rehabilitation), and November 1, 1947, to December 31, 1953 (operations and cleanup). From October 1, 1942, into 1946, both African and domestic ore were used for research, development, and production. Tonawanda Laboratory AEC work is assumed to have stopped radiation work after December 31, 1946, although some workers might have visited the Ceramics Plant buildings. The exact levels of contamination that remained at the Ceramics Plant and the nature of worker activities in areas of residual activity are unknown for the standby and rehabilitation periods. Exposures to uranium progeny and radon would have decreased at the Ceramics Plant after August 31, 1946, because ore was no longer processed during this period. Continued lower level exposures to uranium progeny and to radon were assumed because some radioactive waste was disposed of on the site and because initial cleanup was not completed until the

end of 1954; however, for the Ceramics Plant the uranium exposures would have dominated during the post-1946 period.

The primary sources of internal radiation exposure at Linde due to MED and AEC work were natural uranium and its progeny.

3.1 ESTIMATION OF PARTICULATE INTAKES

During World War II and the time of the MED, the permissible level for natural uranium air dust concentration was set at 500 μg/m^3 for insoluble uranium compounds and 150 μg/m^3 for soluble uranium compounds. After the war, the University of Rochester in Rochester, New York, recommended lowering the permissible level to 50 μg/m^3 for all uranium compounds on the basis of chemical toxicity. This level was also stated as 70 dpm/m^3 for natural uranium and was based primarily on animal studies. The Medical Division of the AEC NYOO thought that the maximum permissible level should be based on human data and was thus unknown. Therefore, the level of 50 μg/m^3 was generally referred to as the preferred level (PL) (AEC 1949a). Contractors of the AEC often used the term maximum allowable (air) concentration (MAC) interchangeably with PL and reported air-sampling results as multiples of the MAC (Stewart 1952; AEC 1953). When considering air concentrations that were reported in multiples of the PL or MAC, the actual assumed value of the PL or MAC should be verified.

In 1949, the Medical Division of the NYOO published a report on the health hazards at seven facilities that produced and/or processed uranium for the AEC. These facilities included Mallinckrodt Chemical Works, Harshaw Chemical Company, LAPC, Electro Metallurgical Company, and Vitro Manufacturing Company. The AEC used the information on work tasks with measured air concentrations in breathing zones, general areas, and process areas to determine average air concentrations that were weighted by exposure times and summed these time-weighted air concentrations to determine daily time-weighted average air concentrations by job category. Up until the time of the 1949 AEC report (AEC 1949a), surveys by the NYOO indicated that out of 648 exposed workers at these plants, 9% were exposed to uranium air concentrations greater than 125 MAC (greater than 6,250 μg/m^3), 9% were exposed at 25 to 125 MAC (1,250 to 6,250 μg/m^3), and 82% were exposed to less than 25 MAC (less than 1,250 μg/m^3). Linde's maximum time-weighted exposure during this period was 33 or 32 MAC (the data in the report's text and its graph differed). As a result of the NYOO report, significant improvements were made in operational conditions such as redesign of ventilation systems, enclosing some processes, and using remote controls (AEC 1949a). By the end of 1949, exposure levels were significantly reduced at these larger plants even though production levels increased (Mason 1959).

Indications are that some of the higher routine (versus episodic) exposures occurred at the uranium ore processing facilities. The AEC did not exist until January 1, 1947, and its report (AEC 1949a) did not elaborate on internal exposures before its tenure. A general review of air concentration data, safety reports, and production and progress reports from the early 1940s through 1946 indicates that there were significant reductions in exposures due to improved engineering, process, workplace, and administrative controls brought into effect before the AEC report. This indicates that the exposures at the ore and waste processing plants in the earliest years of operations would likely have been higher than the exposures during AEC tenure. Indeed, the "tolerance" air concentration was 10 times higher for insoluble compounds during the early years and 3 times higher for soluble compounds.

After the ore processing, Linde began a standby period. It was assumed that exposures decreased to 0.1 MAC at the Tonawanda Laboratory after cleanup in 1946 until December 31, 1953. Based on reviews of later air concentrations at Linde and reviews of air concentration data from other sites, most workers' exposures would have been much lower during these periods.

The standby period at Linde Ceramics was assumed to end on September 14, 1947. Rehabilitation of the Step III process was assumed to begin on September 15, 1947, and continue through October 31, 1947.

3.2 URANIUM

During uranium research and the processing of uranium ores at Tonawanda Laboratory and the Ceramics Plant, workers might have been exposed to a variety of uranium chemical forms that encompass all lung absorption types: F, M, and S. Although some process steps might have had more or less exposure to a given lung absorption type, it is not clear how well separated these areas were before standby. In addition, workers could have worked in multiple uranium process areas. This analysis assumed that types F, M, and S were available before standby. After operations ceased in 1946, it is more likely that remaining uranium contamination would have been oxidizing and becoming less soluble. Beginning with the rehabilitation period for restart of Step III operations in 1947, it is assumed that only type M and S materials would be available. The selection of absorption type should depend on the organ of interest.

3.2.1 Uranium Urinalysis Data

Medical urinalyses for specific gravity, proteins, sugars, etc. were a requirement at Linde from the beginning of MED work, but there is no evidence that urine was analyzed for radioactivity until 1947. MED noted that tentative arrangements had been made for uranium urinalyses of certain groups of employees at Linde (Ferry 1944b), but no records are available to indicate the analyses were done. A page titled, "Schedule of Examinations, Contract AT-30-1-GEN-165" (Author unknown ca.1947), which commenced sometime in 1947 (Rennich 1947), indicates that 60 cm^3 of urine from each employee was to be sent monthly to the University of Rochester, the location of one of the earliest bioassay analysis programs. Individual uranium urinalysis data from November 1947 to 1950 are available for some Linde Ceramics workers (LAPC 1947–1950). The period covers both Step III production in Building 38 and decontamination in Building 30. Worker radiation exposures during this period were likely to be lower than during the production period before standby because African ore was no longer being processed, Step I and II processes had ceased, and engineered safety controls had been improved.

The reported results of Linde uranium urinalyses during Step III production and later cleanup ranged from 0 to 3.10 mg/L, with many of the higher results reported in the November to December 1947 "preemployment urines." Because cleanup/preparation activities occurred before collection of these samples, it is possible the elevated urinalysis results represented exposures that were received either during rehabilitation of the Step III equipment or perhaps from the less soluble component of uranium that was taken in during earlier operations.

In November 1948, two uranium urinalysis results, 3.10 and 0.33 mg/L, were noted as high, and Linde was requested to consider the possibility of contamination. A fluoride result from the individual with the highest uranium results was also elevated. No information was found to indicate whether these were contaminated samples, but the results are consistent with intakes of soluble uranium.

Some samples might have been collected over the weekend: one set of samples indicated that collection bottles were taken home on a Friday because there was no work for 2 days, and the bottles were returned on Monday morning at 8:00 a.m. A note on the January 1950 uranium urinalysis results states, "Please add approx[imately] 1% of conc[entrated] HCl to urine as a preservative instead of toluene."

It appears that the University of Rochester and the AEC NYOO were performing the uranium urinalyses for Linde (Wolf 1948). The uranium fusion photofluorimetry urinalyses that were performed by the University of Rochester and the AEC NYOO were similar to those at other AEC facilities. The

default detection threshold for uranium urinalysis was assumed to be 10 µg/L based on a reported sensitivity of 5 to 10 µg/L for uranium fluorimetry urinalysis in the early years (Wilson 1958). However, the January 1948 uranium urinalysis results AEC transmitted to Linde included a note that indicated the minimum detectable amount might have been much higher (Wolf 1948):

> *The uranium content of those samples listed as being 0.1 mg U/l or less is below the limit of accurate quantitative determination by the photofluorometric method of analysis.*

February, May, June, August, and November 1948 uranium urinalysis results included a similar note, but 0.01 mg U/L was reported as the limit of accurate quantitative determination or reliable determination for photofluorometric analysis. The January 1948 determination level of 0.1 mg/L is assumed to be a typographical error because this is the same as the determination level reported for (nonradioactive) fluoride urinalysis and because there seems to be no change in the format of the reported numbers.

The AEC requested "check samples" on all uranium urinalyses that exceeded 0.1 mg/L, and noted in February 1948 that it would be appropriate to resample in about 3 months (Wolf 1948). Linde's reply requested clarification because it had previously understood that resampling was to occur monthly (Heatherton 1948a).

ORAUT-OTIB-0019, *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005a), describes the general process for analyzing bioassay data for assignment of doses to individuals based on coworker results. The above-described bioassay results were analyzed in accordance with Attachment D of that document. The results of the analysis are presented in Tables 3-1 and 3-2. Individual uranium urinalysis results should be used to determine internal exposure to the individual when they are available. Where individual results are not available, the coworker data included in Attachment D of ORAUT-OTIB-0019 and summarized in Tables 3-1 and 3-2 are to be used to estimate internal exposures that are favorable to claimants.

Table 3-1. Chronic intake rate for Type M uranium (pCi/d).

Start	End	50th-percentile value	GSD
11/01/1947	12/31/1953	74	4.0

Table 3-2. Chronic intake rate for Type S uranium (pCi/d).

Start	End	50th-percentile value	GSD
11/01/1947	12/31/1953	1884	4.3

3.3 RADIUM

DHHS has determined, and NIOSH has concurred, that it is not feasible to reconstruct internal exposure that occurred before November 1, 1947 (Leavitt 2005). Information on radon exposure before November 1, 1947, is provided only as a basis for extrapolation of later exposures and is not intended to be used for this period in which reconstruction of internal dose has been determined to be infeasible.

All radium compounds are lung absorption type M. Radon breath analyses have been used to provide information on the amount of radium in the body and are available for some Linde workers.

Radon Breath Data

The level of radon in exhaled breath can provide information on the amount of radium in the body. Radon breath data were found for 45 samples from Linde workers between June 12, 1944, and January 17, 1945 (LAPC 1944–1945). The records indicate that Dr. R. D. Evans in Cambridge, Massachusetts, analyzed some of the samples. Room background concentrations of radon were sometimes subtracted from the results and sometimes not. In July and August 1944, room

background concentrations were reported respectively as 0.6 and 0.3 pCi/L (Tybout 1944b). The measured radon levels were given for some but not all of the samples. Reported radon breath results ranged up to 2.2 pCi/L. The method of deriving the ^{226}Ra level was not stated, but in many cases, the reported burden of radium in micrograms was numerically equal to 10% of the breath radon concentration in picocuries. Radon breath results are the starting point for dose reconstruction, so it might be necessary to back-calculate breath results from either ^{226}Ra burdens or tolerance levels. The original records might have sufficient information for this determination. This information has not yet been tabulated for inclusion in this site profile.

3.4 URANIUM PROGENY

Ceramics Plant 1943 to 1946 Production, and Tonawanda Laboratories
DHHS has determined, and NIOSH has concurred, that it is not feasible to reconstruct internal exposure that occurred before November 1, 1947 (Leavitt 2005).

Ceramics Plant 1947 to 1949 Step III Production and Subsequent Initial Cleanup
During this period, refined uranium materials were handled. None of the uranium progeny would have been present in significant quantities in the refined uranium materials but, to account for uranium progeny potentially present from past activities and resuspended during decontamination and decommissioning (D&D) activities, data from the postoperations period was reviewed to determine bounding activity ratios (Attachment E). Table 3-3 presents bounding indoor uranium progeny ratios for use for dose reconstruction for the period from November 1, 1947, through December 31, 1953. The values in this table were the highest observed values from the indoor and storm sewer sampling locations.

3.5 RADON

DHHS has determined, and NIOSH has concurred, that it is not feasible to reconstruct internal exposure that occurred before November 1, 1947 (Leavitt 2005). Information on radon exposure before November 1, 1947, is provided only as a basis for extrapolation of later exposures and is not intended to be used for this period in which reconstruction of internal dose has been determined to be infeasible.

Table 3-3. Progeny to uranium ratios.

Progeny/U (total)	Ratio to uranium
Th-230/U	0.26
Ra-226/U	0.21
Po-210/U[a]	0.21
Ac-227/U	0.29
Pa-231/U	0.01

a. Po-210 activity not reported, assumed to be the same as parent (Ra-226)

More than 200 measurements of radon concentrations were made during 1942 to 1946 pilot plant and production processing of African ore at Linde (LAPC 1944–1946). The early measurements were made using liter glass bulbs that were evacuated immediately before sampling to below 50 µm (Skinner 1944). It is likely that the later measurements during operations were made in the same way. Early samples were analyzed at the Massachusetts Institute of Technology; later analyses appear to have been performed at the University of Rochester. The tolerance value for radon, sometimes abbreviated as TV in the old records, was 100 pCi/L.

To determine exposure due to a specified radon concentration (assumed to be ^{222}Rn), the concentration *C* in picocuries per liter was converted to potential alpha energy concentration (PAEC) in units of working level (WL) using:

$$PAEC = C \times F \div (100 \text{ pCi/L per WL}) \quad (3\text{-}1)$$

where the equilibrium factor *F* was taken as 0.4, a value that is recommended by International Commission on Radiological Protection (ICRP) Publication 32 (ICRP 1981) and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR 1993). Potential alpha energy exposure (PAEE) in units of working level-months (WLMs) was obtained from the PAEC and the number of months of exposure *M* using:

$$PAEE = PAEC \times M \quad (3\text{-}2)$$

where a month is assumed to be 170 work hours.

3.5.1 Ceramics Plant Radon Exposures

During Ceramics Plant preproduction and initial production (which involved only domestic ore processing), the only source of radon was African ore processing at Tonawanda Laboratory. The indoor and outdoor radon concentrations to which Ceramics Plant workers were exposed were assumed to equal the outdoor concentration from Tonawanda Laboratory work. No direct measurement of this was available. An estimate was made based on the lowest measured indoor concentrations at the Ceramics Plant during African ore processing. These were viewed as indicating the upper limit to the outdoor concentration because outdoor air was drawn indoors for ventilation. Approximately 20% of the measurements in the Ceramics Plant ore processing building yielded results of 10 pCi/L or less, with most of these results at or near 10 pCi/L. Therefore, 10 pCi/L was taken as the estimated outdoor concentration.

Measured radon concentrations during processing of African ore at the Ceramics Plant are summarized in Table 3-4. Based on geometric means (GMs), locations were grouped into three categories – high, medium, and low – and the GM and geometric standard deviation (GSD) of all results in each category were determined. Some results were reported as less-than values and these are denoted as less than the limit of detection (<LOD).

Table 3-4. Measured radon concentrations during Ceramics Plant African ore processing.

Location category/location	# of samples	GM (pCi/L)[a]	GSD	GM (pCi/L)[a]	GSD
High				99.3	3.43
Scales	5	153	2.50		
Ore box car	17	125	3.92		
Ore sampling room	24	100	3.11		
Ore dumping grill	18	88.6	2.91		
Ore storeroom	26	88.5	4.49		
Ore bin/conveyor	7	83.3	9.37		
Medium				42.5	2.72
Ore bag tumbler	17	60.2	2.53		
Ore bag wash room	8	45.1	2.78		
ball mill & classifier	24	32.5	2.74		
Low				22.4	3.17
Step 1 digestion & filtration	55	23.7	3.17		
Loading dock	1	18.0	(b)		
Receiving desk	5	12.9	2.43		

a. In computing the GM, a <LOD value was assumed to equal the LOD.
b. GSD could not be determined.

After the end of African ore processing, concentration in the main ore processing building, Building 30, was assumed to remain at the 10-pCi/L level that was measured during the second period of domestic ore processing until the end of cleanup of Building 30. Concentrations in other Ceramics Plant buildings were also assumed to be 10 pCi/L until the end of cleanup in those buildings. Because the locations of many workers are likely to be unknown, it was assumed that all workers were exposed to 10 pCi/L of radon from November 1, 1947, through December 31, 1953. Table 3-5 lists concentrations and exposure rates.

Table 3-5. Ceramics Plant worker radon exposure rates, 1947 to 1953.

Period/work location	Time-weighted concentration (pCi/L)	Exposure rate (WLM/yr)
11/01/1947–12/31/1953		
All workers	10.0	0.480

3.5.2 Tonawanda Laboratory Radon Exposures

Few measurements were available for the Tonawanda Laboratory, so estimates of radon concentrations were based on Ceramics Plant data. African ore work at Tonawanda Laboratory appears to have occurred just before each period of African ore processing at the Ceramics Plant. The precise dates of African ore research at Tonawanda Laboratory are not known. Based on the available reports, it appears to have occurred only about half of the time that the Laboratory was engaged in MED work, with most of the African ore work concentrated in the periods immediately before the beginnings of the two African ore processing periods at the Ceramics Plant. For dose reconstruction, it was assumed that African ore processing at Tonawanda Laboratory occurred during the entire MED period, but that the peak concentrations of radon were equal to the medium values at the Ceramics Plant. It was assumed that after MED research and initial cleanup at the Laboratory ended, the radon concentration dropped to 10 pCi/L (the level in the Ceramics Plant after it switched from African ore processing to domestic ore processing) and remained there until the end of the cleanup at the Laboratory. After the end of cleanup, radon exposure in the Laboratory was based on the highest GM PAEC that was determined for a Tonawanda site building from measurements in 1981. That value was 1.68×10^{-02} WL for Building 31 based on analysis of data in BNI (1982, Table B-3). Table 3-6 summarizes the assumed radon concentrations and resultant exposures.

Table 3-6. Tonawanda Laboratory radon exposure rates, 1947 to 1953.

Period		Time-weighted concentration (pCi/L)	Exposure rate (WLM/yr)
Start	End		
Postcleanup			
11/01/1947	12/31/1953	—	0.202

Radon and Radon Progeny
The radon exposure rates that are provided here are annual PAEE rates in WLM/yr. Each value is assumed to be the median of a lognormal distribution with a GSD of 3.43. This GSD is based on the location category having the highest GSD (3.43 per Table 3-4).

3.6 INHALATION INTAKE ESTIMATES OF RADIOACTIVE PARTICULATES AT THE TONAWANDA LABORATORY

The airborne concentrations in Section 3.1 can be used to determine uranium intakes for Tonawanda Laboratory employees from November 1, 1947, through December 31, 1953. This analysis assumed that a worker was chronically exposed to alpha airborne concentrations as stated in Section 3.1 and summarized in Table 3-7.

Table 3-7. Assumed airborne concentrations used to estimate intakes at the Tonawanda Laboratory.

Start	End	Activity description	# MAC	α dpm/m^3	Source
11/01/1947	12/31/1953	Postcleanup	0.1	7	Uranium and progeny

This analysis assumed Linde workers were exposed to the given air concentrations for 8 hr/d, 5 d/wk, and 50 wk/yr. Time-weighted exposure studies of even the larger plants like Mallinckrodt showed that the majority of workers were exposed at lower levels than were estimated for the early 1942 to 1946 operational period. No credit is taken for breaks or working at tasks where radioactive material intakes would be much lower or nonexistent, or for the use of facemasks; on the other hand no consideration was given to the longer workweeks and work hours in the early years. In addition, because workers were exposed to multiple absorption types, but the single absorption type that produces the larger dose is used to estimate organ dose, it is believed that this estimate adequately accounts for internal exposure.

Because the early air concentration measurements would have included detection of radiations from progeny, the air concentrations that are listed in Table 3-7 as including progeny are apportioned based on the assumption of full equilibrium among uranium and its progeny, using the alpha ratios of uranium and progeny of internal dosimetric importance.

For example, the annual uranium inhalation intake due to chronic exposure at 0.1 MAC was estimated by multiplying the air concentration of 7 dpm/m^3 by the alpha fraction of uranium (0.489), the ICRP Publication 66 (ICRP 1994) recommended breathing rate of 1.2 m^3/hr, and the assumed 2,000 work hours per calendar year. This results in an annual chronic inhalation intake of 8.215×10^3 dpm, which is equal to a daily intake rate of 22.5 dpm/d.

Inhalations of the uranium progeny were estimated by substituting the alpha fraction of the progeny for the alpha fraction of the uranium.

3.7 INGESTION INTAKE ESTIMATES AT THE TONAWANDA LABORATORY

In the case where inhalation intakes are calculated from air concentrations, ingestion intakes are also to be considered. NIOSH (2004) indicates that the ingestion rate, in terms of dpm for an 8-hour workday, can be estimated by multiplying the air concentration in dpm per cubic meter by a factor of 0.2, so the uranium ingestion rate based on an air concentration of 7 alpha dpm/m^3 would be 0.563 dpm/wd. To adjust this to ingestion intake per calendar day, 0.685 dpm/wd was multiplied by 250 wd/yr and divided by 365 d/yr, which equals 0.469 dpm/d. In accordance with NIOSH (2004), the f_1-value used for inhalation dose calculations is to be used for ingestion dose calculations.

3.8 OCCUPATIONAL INTERNAL DOSE RECONSTRUCTION ASSUMPTIONS AND SUMMARY

DHHS has determined, and NIOSH has concurred, that it is not feasible to reconstruct internal exposure that occurred before November 1, 1947 (Leavitt 2005).

Intake estimates and radon exposures that were estimated for Linde workers are shown in Tables 3-8, 3-9, 3-10, and 3-11. Uranium intakes are based on the coworker data in Tables 3-1 and 3-2. Uranium progeny intakes were determined by applying the isotopic ratios in Table 3-3. For the Ceramics Plant, either the values in Table 3-8 or 3-9 should be applied (not both), whichever is most favorable to claimants along with the radon intake in Table 3-10. Tonawanda Laboratory employees should be assigned the intakes in Table 3-11. For workers whose work location is considered indeterminate, intakes for the Ceramics Plant should be assumed.

Table 3-8. Internal exposure summary for the Ceramics Plant, November 1, 1947, to December 31, 1953 (based on Type M uranium).

Radionuclide	Start	End	Intake route	Absorption type	Intake or exposure	Units
U-234	11/01/1947	12/31/1953	Inhalation	M	1.64E+02[a]	dpm/d
Th-230	11/01/1947	12/31/1953	Inhalation	M	4.26E+01[a]	dpm/d
Ra-226	11/01/1947	12/31/1953	Inhalation	M	3.44E+01[a]	dpm/d

Po-210	11/01/1947	12/31/1953	Inhalation	M	3.44E+01[a]	dpm/d
Pa-231	11/01/1947	12/31/1953	Inhalation	M	1.64E+00[a]	dpm/d
Ac-227	11/01/1947	12/31/1953	Inhalation	M	4.76E+01[a]	dpm/d

a. Intake assuming type M uranium, value listed is GM of a lognormal distribution with GSD of 4.0.

Table 3-9. Internal exposure summary for the Ceramics Plant, November 1, 1947, to December 31, 1953 (based on Type S uranium).

Radionuclide	Start	End	Intake route	Absorption type	Intake or exposure	Units
U-234	11/01/1947	12/31/1953	Inhalation	S	4.18E+03[a]	dpm/d
Th-230	11/01/1947	12/31/1953	Inhalation	S	1.09E+03[a]	dpm/d
Ra-226	11/01/1947	12/31/1953	Inhalation	M	8.78E+02[a]	dpm/d
Po-210	11/01/1947	12/31/1953	Inhalation	M	8.78E+02[a]	dpm/d
Pa-231	11/01/1947	12/31/1953	Inhalation	S	4.18E+01[a]	dpm/d
Ac-227	11/01/1947	12/31/1953	Inhalation	S	1.21E+03[a]	dpm/d

a. Intake assuming type S uranium, value listed is GM of a lognormal distribution with GSD of 4.3.

Table 3-10. Internal exposure summary for the Ceramics Plant, November 1, 1947, to December 31, 1953.

Radionuclide	Start	End	Intake route	Absorption type	Intake or exposure	Units
Rn-222	11/01/1947	12/31/1953	Inhalation	(a)	0.48E+00	WLM/yr

a. Value shown is for full year, partial years should be prorated.

Uranium bioassays are available for some workers during the period from November 1947 through January 1950. If uranium bioassays are used to reconstruct an individual's dose (rather than the values in Tables 3-8 to 3-11), additional intakes from uranium progeny must be added using the activity ratios in Table 3-3.

Table 3-11. Internal exposure summary for the Tonawanda Laboratory, November 1, 1947, to December 31, 1953.

Radionuclide	Start	End	Intake route	Absorption type	Intake or exposure[b]	Units
U-234	11/01/1947	12/31/1953	Inhalation	M, S	2.25E+01	dpm/d
			Ingestion	(a)	4.69E-01	dpm/d
Th-230	11/01/1947	12/31/1953	Inhalation	M, S	1.10E+01	dpm/d
			Ingestion	(a)	2.29E-01	dpm/d
Ra-226	11/01/1947	12/31/1953	Inhalation	M	1.10E+01	dpm/d
			Ingestion	(a)	2.29E-01	dpm/d
Po-210	11/01/1947	12/31/1953	Inhalation	F, M	1.10E+01	dpm/d
			Ingestion	(a)	2.29E-01	dpm/d
Pa-231	11/01/1947	12/31/1953	Inhalation	M, S	5.13E-01	dpm/d
			Ingestion	(a)	1.07E-02	dpm/d
Ac-227	11/01/1947	12/31/1953	Inhalation	F, M, S	5.13E-01	dpm/d
			Ingestion	(a)	1.07E-02	dpm/d
Rn-222	11/01/1947	12/31/1953	Inhalation	-	2.021E-01	WLM/yr

a. Choose same f_1-value as used for inhalation per NIOSH (2004).
b. The dose distribution for particulate intakes is assumed to be constant. The dose distribution for radon is assumed to be lognormal with a GSD of 3.43.

4.0 ESTIMATION OF EXTERNAL EXPOSURE, 1942 TO DECEMBER 31, 1953

Because of the SEC determination (Leavitt 2005) that it is infeasible to adequately reconstruct internal dose during the period from October 1, 1942, through October 31, 1947, dose estimates for this period are considered partial dose estimates.

This section develops parameters for reconstruction of doses due to external exposures due to work activities from October 1, 1942, the assumed start date of MED work at Linde, until December 31, 1953. In some parts of this section, measurements and parameters for the period after 1953 are analyzed because they provide information used to estimate external exposures before 1954. Occupational medical exposures are treated in Section 5.0. Dose reconstruction for the period after 1953 is discussed in Section 6.0. Film badges were worn by some workers during some periods. For unmonitored workers and periods, doses are estimated from source term information, workplace measurements, and from available dosimetry results for workers.

Throughout this document, it is assumed that 1 mR = 1 mrad(air) = 1 mrad(tissue) = 1 mrem = 1 mrep.

Beta Radiation
In developing the beta dose rate estimates in this section, no attenuation due to shoes, apparel, or protective items (e.g., gloves, face shields) was considered. Attenuation due to apparel is variable and sometimes not very significant (see Table A-5). The types of protective items issued to workers varied with time, and the extent to which workers used them is uncertain.

Workers who frequently handled significant quantities of beta-emitting materials were assumed to have had higher beta doses to the hands and forearms than to the remainder of the body. The relationships assumed between contact dose rate and dose rates to other parts of the body were based in part on data from the Ceramics Plant (see Table A-6).

For the purpose of calculation of organ dose, all exposure geometries are assumed to be anterior-posterior (AP).

Burlap Bag Exposures
Based on the assessment in Attachment E of this TBD, NIOSH's review of historical records indicates that the period during which the burlap bags were staged and burned is within the current SEC period. NIOSH has found no other credible evidence, other than a former employee affidavit that indicated that two pallets of filled burlap bags were in Building 30 at the Ceramics Plant, to support a potential burlap bag exposure issue outside of the SEC period. If credible information is presented in a claimant interview or documentation related to an individual claim, to establish the worker's occupational proximity to the burlap bags, dose reconstructors should use the external dose estimates in Attachment E to assess this external exposure to support assessment of this external dose for the purpose of individual dose reconstruction.

4.1 CERAMICS PLANT BETA AND GAMMA EXPOSURE

4.1.1 Preproduction, 1942 to 1943

A systematic characterization of radiation levels in Building 30 was made in 1949 in conjunction with its decontamination (Heatherton 1950). A survey scheme called "restricted randomization" was used. The floor area was divided into 15- by 15-ft squares each of which in turn was divided into 100 small squares. In each large square, four of the small squares were selected at random and surveyed.

Total beta plus gamma radiation in millirep per hour was measured at contact at the square's center and corners and at 3 ft above the floor. A similar method was used to measure wall contamination. Before the initial survey, the building was vacuumed and flushed with water. After the initial survey, heavily contaminated portions of the building were removed or cleaned, mostly by sandblasting. Then the building was again vacuumed and flushed, and a final survey was taken. Heatherton (1950) reported the number of measurements and the lowest, mean, and highest values for each of several major plant areas. The middle columns of Table 4-1 display the results of a statistical analysis of the data.

Table 4-1. Floor and wall radiation in Building 30 before and after the 1949 to 1950 decontamination.

Time	Location	Measured beta plus gamma[a]		Estimated	
		Median (mrep/hr)	GSD	Beta (mrem/hr)	Gamma (mR/hr)
Before vacuum cleaning and flushing	Contact	(b)	(b)	2.03[c]	0.131[c]
	3 ft	(b)	(b)	0.676[c]	0.131[c]
Predecontamination (just before decontamination and after vacuum cleaning and flushing)	Contact	0.719	1.67	0.675[d]	0.0438[d]
	3 ft	0.240	1.67	0.225[d]	0.0438[e]
After decontamination	Contact	0.311	2.04	0.292[d]	0.0189[d]
	3 ft	0.0940	2.04	0.0883[d]	0.0189[e]

a. Based on data in Heatherton (1950, Tables III and IV). The data were assumed to be characterized by a lognormal distribution. For the contact measurements, the mean value and a percentile value were available (the percent of results less than 1 mrep/hr was stated). These were used with the computer program LOGNORM4 (Strom and Stansbury 2000) to estimate the median and GSD. For measurements at 3 ft, the mean was available but not the percentile value. The GSD determined from the contact data was assumed to apply and used to estimate the median.
b. Not available.
c. Assumed to be three times higher than the predecontamination values. The factor three is based on April 1949 data; see discussion in text.
d. Estimates based on beta and gamma percentages in Table 4-2.
e. Assumed to equal contact gamma exposure rate.

Another comprehensive survey of Building 30 was made in 1976 (ORNL 1978). Beta plus gamma levels in millirad per hour were measured at 1 cm from the surface, and gamma levels microroentgen per hour were measured at 1 m. Table 4-2 displays the results of a statistical analysis of the data. The measured levels were similar to the postdecontamination results of 1950, which indicated little change in conditions. Because gamma measurements were reported, the beta and gamma percentages of the radiation could be estimated. These were used to estimate the beta and gamma components in Table 4-1 for the pre- and postdecontamination results.

Table 4-2. Building 30 floor radiation levels in 1976.

Parameter	GM[a]		GSD[a]	Composition of radiation at 1 cm
	Value	Units		
Beta plus gamma at 1 cm	0.253	mrad/hr	3.47	(b)
Gamma at 1 m[c]	0.0154	mR/hr	1.69	(d)
Estimated gamma at 1 cm[e]	0.0154	mR/hr	(d)	6.09%
Estimated beta at 1 cm[f]	0.238	mrad/hr	(d)	93.9%

a. GM and GSD were calculated from data in Table 3 of ORNL (1978).
b. Estimate made in last two lines of the table.
c. Background was not subtracted from the external gamma values. Outdoor external background in the Tonawanda area was reported to be 8 to 15 µR/hr (ORNL 1978, p. 15).
d. Not applicable.
e. Estimated as equal to gamma at 1 m.
f. Estimated as 0.253 (beta plus gamma) less 0.0154 (gamma).

The measurements just before decontamination in 1949 and 1950 were made after the building had undergone vacuum cleaning and flushing. Brief, semiquantitative reports were available for two 1-day surveys in April 1949 before the vacuum cleaning and flushing (Blatz 1949; Wolf 1949b). Typical levels of measured gamma radiation at 3 ft on April 19 and at contact on April 22 were similar and averaged about 0.18 mR/hr, about 4 times higher than the estimated median contact gamma level before decontamination. Typical levels of measured beta at contact on April 22 averaged about 0.87 mrep/hr, about 1.3 times higher than the estimated median beta level before decontamination. The April results were the basis of the estimate in Table 4-1 that beta and gamma radiation levels before vacuum cleaning and flushing were 3 times higher than the values that were measured afterward.

In the 1976 survey, Building 30 was the most contaminated building on the site (ORNL 1978). The levels for Building 30 were assumed to apply to all buildings on the site.

For the preproduction period, it was assumed that there was no significant external beta or gamma exposure indoors for Ceramics Plant workers because the buildings were under construction or new. Some exposure while outdoors was possible because the transport of ore to Tonawanda Laboratory (Building 14) and the work there could have produced some site contamination. Resultant outdoor radiation levels are very likely to have been less than floor and wall radiation levels inside Building 14 because the indoor and outdoor sources would have been comparable but the outdoor sources could spread over a larger area and were subject to dispersal by the weather. The radiation levels before vacuuming and flushing that are estimated in Table 4-1 are taken as an estimate of the levels in Building 14 and also as an upper limit estimate of site outdoor radiation levels. Ceramics Plant workers were assumed to have been exposed to these levels for 0.5 hr/d. Table 4-3 shows estimated annual exposure rates.

Table 4-3. Preproduction beta and gamma radiation levels.

Beta (rem/yr)[a]	Gamma (R/yr)[a]	GSD[b]
1.01E-01	1.97E-02	3

a. Based on Table 4-1 beta and gamma levels at 3 ft before vacuum cleaning and flushing. Assumed exposure period: 0.5 hr/wd, 6 wd/wk, 50 wk/yr. The beta and gamma rates are for the whole body.
b. Estimated on the basis of the GSD values in Tables 4-1 and 4-2.

The beta dose rate was assumed to apply to the whole body with no added dose to the hands and forearms because it is unlikely that there was any significant handling of radioactive materials by Ceramics Plant workers in this period.

4.1.2 Production, 1943 to 1946

4.1.2.1 Beta

No beta film badge data were found for 1943 to 1946. There were beta film badge data for 1947 to 1949, but none of the data pertained to Steps I or II, as these processes were no longer in operation. For 1943 to 1946, one report of workplace beta exposure rates was found. Solid samples from the Ceramics Plant were sent on January 23, 1944, to laboratories of the Medical Section of the MED for measurement of beta radiation. Results were reported as “Beta Radiation Roentgens/8 Hr Day” (Ferry 1944c) and are reproduced here in the first five columns of Table 4-4. The roentgen is technically defined only for photon radiation; beta dose should be expressed in absorbed dose units such as rad or rem. However, because there is an approximate numerical equivalence between these units, it is assumed here that the roentgen unit used is equivalent to shallow dose at 0.07 mm in rem.

Table 4-4. Beta radiation exposure rates measured at Linde Ceramics, January 23, 1944.

From MED report of data during processing of L-30 African ore[a]					Estimated worker dose rate (rem/yr)[d]
Sample #	Sample location	Beta radiation (R/8-hr day)[b]	Allowable time of exposure per worker (hr/wk)[c]	Actual time of exposure per worker (hr/wk)	
13	L-30 Ore, Step I	0.1	150	50	31
9	L-30 Tailings, Step I	1.4	18	50	263
12	Barium Cake, Step I	0.8	30	0.33	2
14	A. L. Cake, Step I	1.3	18	9	73
15	PbS Cake, Step I	0.2	120	2	3
16	Soda Salt, Step I	0.5	48	9	28
10	Iron Cake, Step II	0.8	30	3	15
11	N.G. Cake, Step II	4.2	6	6	158

a. Results reported by the MED for measurements made on solid samples sent from Linde Ceramics January 23, 1944 (Ferry 1944c).
b. Units are those stated in the MED report.
c. The MED allowable time of exposure was based on the then-effective tolerance levels of 0.5 R/d and 3.0 R per 6-day week for beta radiation.
d. Except for Sample #9 (L-30 tailings), the dose rate is based on an assumed 50-week year and the dose rate and actual hours of exposure per week stated by the MED. For the worker handling L-30 tailings, the dose rate is estimated based on information in the MED letter (Thomas 1944) cited in the text. The dose rate when the worker is "within range of the radiation" is estimated as half of the contact dose rate. If the source is sufficiently large and there is no shielding, a distance of 1.6 m from the tailings would be required to reduce the dose rate by a factor of 2 (see Table A-4). The dose estimate calculation is as follows:

Location	(rem/8hr)	hr/wk	hr/yr	rem/yr
Hands in tailings	1.4	10	500	88
Within range of radiation	0.7	40	2,000	175
Total				263

From the fact that no distance values were reported with the measurements and from later discussion of the measurements in MED correspondence (Thomas 1944), it is apparent that contact exposure rates were reported. This is supported by the similarity of the results to contact dose rates for various natural uranium materials (see Table A-1).

The fourth and fifth columns of Table 4-4 contain the allowable times of exposure based on a 3.0-R/wk tolerance level and the actual hours of exposure per week as stated in the MED report. For the L-30 tailings sample, the actual exposure time was stated to be 50 hr/wk, much greater than the allowable exposure time of 18 hours. A MED memorandum (Hadlock 1944) questioned this discrepancy. It brought the following MED response (Thomas 1944):

> *No action has been taken on the Captain Ferry's recommendation that the exposure time to L-30 tailings be limited to 18 hours. The primary reason for this is the fact that the estimated fifty hours is based on the time which a man is actually within range of the radiation whereas the Medical Section's allowable exposure time is based on contact of the tailings directly with the hands.…*
>
> *The Linde operating and safety sections have estimated that the average time a worker spends with his hands in the tailings is six to eight hours one week. The maximum time is never over ten hours one week and even then heavy rubber gloves are worn.*

The last column of Table 4-4 shows worker doses derived from the data. These range from 2 to 263 rem/yr.

It is striking that the 1943 to 1946 dose rates in Table 4-4 are much higher than the 1947 to 1949 rates Section 4.1.4.1 discusses. For 1943 to 1946 Step I process operators, the beta dose rate was estimated as 263 rem/yr. For 1947 to 1949, the job with the highest beta dose rate was that of a Step III process operator, and film badge data indicate that the beta dose rate was only about 2 rem/yr (approximate value of annual $D_D + D_M$; see Table 4-16 later in this section). Despite the high estimates, the 1943 to 1946 data does not appear unreasonable. The dose rates in Table 4-4 are typical of contact dose rates for uranium materials (see Table A-1). The worker exposure times were debated and reviewed within the MED. In the MED's interpretation of the data, with allowances for the protective measures (e.g., gloves) the rates were within the limit in effect at the time, 3.0 R per 6-day week (Ferry 1944c) or 150 R/yr. Therefore, the estimates in Table 4-4 are considered a valid basis for dose reconstruction.

4.1.2.1.1 Estimation of Worker Doses

Beta doses for specific jobs are estimated below and summarized in Table 4-24 at the end of this section.

1943 to 1946 Loaders, Step I Process Operators, Step II Process Operators, Ball Mill Operators, and Weighmasters

The calculated dose in Table 4-4 for sample 13 was assumed to apply to a loader. The doses for samples 9 and 11 were assumed to be the doses to the most highly exposed Step I and Step II process operators, respectively. The dose for the most highly exposed process operator was assigned to all process operators in a step. A ball mill operator was assigned the same dose as a Step I process operator because the ball mill operators performed some of the tasks of process operators when not operating the ball mill (see job description for CL-3, Chemical Operator C on p. 13 of LAPC 1945b). Reductions for shielding by apparel and containers were not taken into account in estimating doses because of insufficient information. The dose rates in Table 4-4 are based on contact measurements, so they represent dose rates to the hands and forearms. For all but the Loader, dose rates to the remainder of the body were taken to be one-third of the contact dose rates. This reduction was based on data in Table A-6 (see Hands and Forearms Dose in Section 4.1.4.1). For the loader, dose rates for the remainder of the body were assumed to equal the contact dose rates because Loaders frequently hand carried ore bags, pushed carts of ore, and worked in the close vicinity of large ore piles.

The weighmaster was responsible for verifying weights of all materials (incoming, product, and byproduct; see p. 12 of LAPC 1945b) and so had frequent close contact with these materials. Furthermore, a radiological survey data sheet from March 1944 indicates that at that time the weighmaster was stationed for 8 hr/d at a location 4 ft from a pile of ore bags. The data in Table 4-4 and Table A-1 indicate that processed materials and byproducts had about 15 times the beta dose rate of ore. Therefore, the weighmaster was assigned the same beta exposure rate as a Step I process operator.

The dose rates in Table 4-4 were based on measurements on a particular batch of African L-30 ore. For a given type of ore, beta dose rates to ore handlers in Step I would be approximately proportional to ore grade (the weight percent of U_3O_8 in the ore). Ore type (African or domestic) would also matter. For the fully prerefined ore (which the domestic ore used at Linde might have approximated) the electron energy release per uranium decay is only 39% of that for African ore (compare Table A-2 and Table A-3). Beta dose rates in Steps II and III would be independent of ore grade. Dependence on ore type should have been weak because most radioactive impurities would have been removed in Step I.

To use the data in Table 4-4 to estimate time-averaged beta dose rates applicable to the entire 1943 to 1946 production period at Linde, the ratio of the average dose rates to those in Table 4-4 was estimated. Table 4-5 documents the determination of the ratio. The grade range of the L-30 ore that was used at Linde was 8% to 12% (Wallo, Vierzba, and Roberts 1981, Table B-1). To obtain the highest ratio, it was assumed that the measurements in Table 4-4 were made on the lowest L-30 ore grade, 8%. It was also assumed that beta dose was proportional to electron energy that is released per decay and that the worker doses were proportional to the mass of ore processed. With these assumptions, it was estimated that average doses would have been 0.84 of the doses that would be predicted by using the data in Table 4-4. Therefore, the estimated dose rates in Table 4-4 were multiplied by 0.84 to obtain time-averaged dose rates for 1943 to 1946 production. The results are in Table 4-6, which summarizes the results of all beta dose rate estimates in this section and divides the job categories into three groups (high, medium, and low).

Table 4-5. Dependence of beta dose rate on ore type.

Approximate processing period[a]	Ore ID[a]	Ore type[a]	Electron energy released per decay (MeV/nt)[b]	Average grade[a] (%)	Average beta dose rate relative to 8% L-30[c]	Mass processed (MT)[a]	Mass-weighted average beta dose rate relative to 8% L-30[d]

06/1943 to 11/1943	Not stated	Domestic & scrap	0.447	15.0	0.72	1,000	0.028
12/1943 to 10/1944	L-30	African	1.157	10.8	1.35	8,504	0.447
10/1944 to 11/1944	L-50	African	1.157	6.7	0.84	1,486	0.048
12/1944 to 02/1946	L-19	Domestic	0.447	14.0	0.68	6,102	0.161
02/1946 to 07/1946	R-10	African	1.157	3.5	0.44	8,492	0.145
06/1946 to 07/1946	Q-20	African	1.157	17.7	2.21	82	0.007
					Sum:	*25,666*	*0.84*

a. Source: Wallo, Vierzba, and Roberts (1981), Table B-1. MT = metric ton (1,000 kg).
b. Per Table A-2. nt = nuclear transformation.
c. (*Electron Energy Released per Decay*)*(*Average Grade*) ÷ 8%.
d. The value for each ore ID is (*Mass Processed*)*(*Average Beta Dose Rate Relative to 8% L-30*)/(*Total Mass Processed*). The sum of the values is the approximate ratio of the average 1943 to 1946 ore beta dose to the ore dose due to 8% L-30 ore.

1943 to 1946 Workers with Jobs Analogous to 1947 to 1949 Jobs

Workers in 1943 to 1946 who held jobs analogous to ones in 1947 to 1949 were assigned 3 times the 1947 to 1949 beta dose rates. The factor of 3 increase accounts for potential exposure to radiation from waste products from unrefined uranium ore and for the possibility that procedures in 1943 to 1946 did not involve as much radiological protection. (Tables A-2 and A-3 indicate unrefined uranium materials release approximately 2.6 times as much electron energy per uranium decay as refined uranium materials. In addition, the plant had a health physicist in 1947 to 1949 but not in 1943 to 1946.) Workers with jobs analogous to those assigned to the medium beta exposure category for

Table 4-6. Assigned beta dose rates, 1943 to 1946.

Category/job	Median dose rate (rem/yr)		GSD
	Hands and forearms	Remainder of body	
High	**Varies[a]**	**74**	
Ball mill operator	221	74	1.52
Step I process operator	221	74	1.52
Weighmaster	221	74	1.52
Step II process operator	158	53	1.52
Loader	26	26	1.52
Medium	**17.6**	**5.85**	
Chemist/lab technician	17.6	5.85	2.65
Engineer	17.6	5.85	2.65
Janitor	17.6	5.85	2.65
Laundry worker	17.6	5.85	2.65
Maintenance worker	17.6	5.85	2.65
Ore sampler	17.6	5.85	2.65
Seamster, seamstress	17.6	5.85	2.65
Step III process operator	17.6	5.85	2.65
Tool crib worker	17.6	5.85	2.65
Low	**3.00**	**3.00**	
Draftsman	3.00	3.00	2.65
Fire inspector	3.00	3.00	2.65
Guard	3.00	3.00	2.65
Nickel operator	3.00	3.00	2.65
Nurse	3.00	3.00	2.65
Office worker	3.00	3.00	2.65
Plant superintendent, asst. supt.	3.00	3.00	2.65
Shipping and receiving clerk	3.00	3.00	2.65
Storekeeper	3.00	3.00	2.65
Tank farm operator	3.00	3.00	2.65

a. See Section 4.1.2.1.3.

1947 to 1949 were assigned dose rates of 17.6 rem/yr to the hands and forearms and 5.85 rem/yr to the remainder of the body.

Workers with jobs analogous to those assigned to the low beta exposure category for 1947 to 1949 were assigned a beta dose rate of 3 rem/yr to the whole body. Table 4-6 shows the assignments.

1943 to 1946 Workers Ore Sampler
The ore sampler was considered to have an exposure potential similar to that of a chemist and was assigned dose rates of 17.6 rem/yr to the hands and forearms and 5.85 rem/yr to the remainder of the body.

4.1.2.1.2 Outdoor Dose Rate

The average outdoor beta dose rate to which workers would have been exposed during production was assumed to be at most equal to the indoor level in Building 30 based on the reasoning in Section 4.1.1. Therefore, the outdoor beta rate was estimated as 0.676 mrem/hr (based on beta dose rate at 3 ft before vacuum cleaning and flushing in Table 4-1). For 0.5 hr/wd exposure, 6 wd/wk, and 50 wk/yr, the average worker exposure would have been 0.1 rem/yr. This is negligible in comparison with the indoor doses, given the approximate nature of the estimates, and was ignored.

4.1.2.1.3 Categories

Table 4-6 summarizes assigned beta dose rates for the 1943 to 1946 production period. All are considered to be median values of a lognormal distribution. The dose rates that were derived from the 1943 to 1946 data in Table 4-4 were judged to be based on near-maximum exposure parameters. They were assigned a GSD of 1.52, which corresponds to a distribution in which the ratio of 95th-percentile value to median is equal to a factor of 2. The other dose rates were considered to be more uncertain. They were assigned a GSD of 2.65, which corresponds to a distribution in which the ratio of 95th-percentile value to median is equal to a factor of 5.

To simplify dose reconstruction and take into account the uncertainties of the estimates, jobs were grouped into three categories – high, medium, and low – based on beta dose to the "remainder of the body" (all parts of the body except the hands and forearms). For all jobs in each category, the beta dose to the remainder of the body was assumed to be the highest value among the jobs in the category.

Hands and Forearms Guidelines
If beta dose to the hands and forearms is needed for a particular dose reconstruction, it should be obtained as follows:

- For the low and medium categories, the values in Table 4-6 should be used.
- For the ball mill operator, Step I process operator, and weighmaster, the value in Table 4-6 should be used.
- For the Step II process operator, the dose to the hands and forearms should be taken as 3 times the remainder-of-the-body dose *for the high category* or 221 rem.
- For the Loader, the dose to the hands and forearms should be taken as equal to the remainder-of-the-body dose *for the high category* or 74 rem.

4.1.2.2 Gamma

Gamma film badge data are available for Step I process workers for a portion of the 1943 to 1946 production period. The data covers the period from January 31, 1944, through February 26, 1945,

except for a 3-week gap from April 18 to May 5, 1944, for which data were not obtained due to a film handling error (Ferry 1944d). Badges were usually worn for about a week, but during the first 3 months some of the badging periods were longer (2 to 4 weeks). In the data analysis, each multiweek measurement was treated as a set of individual weekly results with each equal to the weekly average that was determined from the multiweek measurement. For example, a 4-week result of 300 mR was treated as four individual 1-week results of 75 mR.

About 20 workers were typically badged in each period – about 10 to 12 Loaders and 7 to 9 processors. It is not known whether all Step I personnel were badged. Film badges were provided by the Medical Section of the MED (presumably the University of Rochester).

From January 31, 1944, through March 27, 1944, results were in units of roentgen per 8-hour day (the designations "Roentgen," "R," and "r" in different reports of the data were interpreted as synonymous). After March 27, 1944, results were reported in tolerance units (designated "fractions of tolerance" or "times tolerance"). The MED tolerance value for gamma radiation in this period was stated as 0.1 R/8-hr day and also as 0.5 R/wk and 0.6 R/wk (Ferry 1944e; Tybout 1945c). Linde employees are reported to have worked a 6-day 48-hour workweek (Dupree1983a,b). For analysis of the gamma data, a 6-day workweek was assumed. To obtain roentgen per week, results in roentgen per 8-hour day were multiplied by 6 and results in tolerance units were multiplied by 0.6.

Some of the gamma results were reported in the form "<*X*," where the value of *X* varied. The average of the *X* values was 44 mR/wk. In the analysis of the data, 44 mR/wk was taken as the LOD, and results reported as "<*X*" or zero were considered to be <LOD results. However, all nonzero results, even if below 44 mR/wk, were treated as ≥LOD results.

During the first part of the measuring period, the plant processed radium-containing L-30 African pitchblende ore. In an analysis of plant operations (Wallo, Vierzba, and Roberts 1981, Table B-1), it was estimated that processing of the African ore began in December 1943 and continued through November 1944 and that processing of domestic ore with minimal radium content began in December 1944. The film badge data showed a sharp drop in the number of ≥LOD results during the first 2 weeks of November with the low point during the week ending November 12, so this was taken as the date when processing of the African ore ended. Exposures to workers were averaged separately for the periods from January 31 to November 12, 1944 (41 weeks) and November 13, 1944 to February 26, 1945 (15 weeks).

The data identified each worker's job activity. The job activities were grouped into categories. Each category consisted of jobs that were judged to have similar exposure potential based on job descriptions and the data. Table 4-7 lists the job activities and the category to which each was assigned.

Table 4-7. Job categories for Step I gamma dose analysis, 1943 to 1946.

Job activity[a]	Category
Ball mill	Ball mill operator
Ball mill operator	Ball mill operator
Cleaning up plant	Cleanup
Loader	Loader
Loader (handling ore)	Loader
Loader foreman	Loader
Ore sampler	Ore sampler
Sampler	Ore sampler
Barreling of tails from Moore filter	Process operator
Chief operator	Process operator
Digest	Process operator
Digest & V.P.	Process operator

Job activity[a]	Category
Eimco & prod. precipitation	Process operator
Foreman	Process operator
Lead removal	Process operator
Making ore digest in Pachuca tanks	Process operator
Moore filter operator	Process operator
Moore operator	Process operator
Moore tailings	Process operator
Moore tailings/Moore operator	Process operator
Moores	Process operator
Pachuca digest	Process operator
Pachuca tanks	Process operator
Product precipitation	Process operator
V.P. press	Process operator
V.P. removal	Process operator
Weighmaster	Weighmaster

a. Job activity descriptions are from the film badge records.

African Ore

Table 4-8 displays statistical characteristics of the film badge data for the period from January 31, 1944 to November 12, 1944, when African ore was processed. For each category, a lognormal distribution was assumed and the following parameters were determined:

- The number of film badge measurements,
- f_D = dosimeter fraction = the fraction of results in a year at or above the LOD,
- f_M = missed fraction = the fraction of results in a year below the LOD,
- m_D = GM of all results above the LOD, and
- The GSD of all results above the LOD based on the assumption that their distribution was lognormal.

Table 4-8. Step I gamma exposure data for African ore processing, January 31 to November 12, 1944.

Category	# weekly results	Dosimeter fraction f_D	Missed fraction f_M	For results in dosimeter fraction	
				GM m_D (mR/wk)	GSD
Ball mill operator	17	0.9412	0.0588	93	1.81
Loader	507	0.8383	0.1617	123	2.00
Ore sampler	36	0.5000	0.5000	109	2.61
Process operator	202	0.6980	0.3020	71	1.93
Weighmaster	31	0.8387	0.1613	100	2.15

The subscripts *D* and *M* denote dosimeter dose and missed dose, respectively, as defined by NIOSH (2007a, pp. 30 and 32). For the current set of data, the value of f_D was estimated as the number of ≥LOD results divided by the number of measurements, and f_M was estimated as the number of <LOD results divided by the number of measurements. (Case-by-case consideration is required when the number of measurements is very low. Such situations are dealt with in Sections 4.1.4.1 and 4.1.4.2.)

The estimated median annual dosimeter dose D_D for a 50-week year is shown in Table 4-9. This was calculated as follows:

$$D_D = 50 f_D m_D \quad (4\text{-}1)$$

Table 4-9. Step I gamma exposure rate parameters for African ore processing, January 31 to November 12, 1944.

Category	D_D[a] (R/yr)	GSD[b]	D_M[a] (R/yr)	GSD	$D_D + D_M$
Ball mill operator	4.37	2.61	0.06	1.52	4.44
Loader	5.17	2.61	0.18	1.52	5.35
Ore sampler	2.72	2.61	0.55	1.52	3.27
Process operator	2.47	2.61	0.33	1.52	2.80
Weighmaster	4.21	2.61	0.18	1.52	4.39

a. Each dose value represents the median of a lognormal distribution.
b. Highest GSD for D_D values in Table 4-8 is assumed to apply to all categories.

For simplicity and because a GSD value based on small number of measurements might not be valid, the highest GSD in Table 4-8 was assigned to every job category.

In accordance with NIOSH (2007a), the annual missed dose was assumed to have a lognormal distribution with median D_M and the 95% confidence limit D_{M95}. For a 50-week year, these were calculated as follows:

$$D_M = 50 f_M (LOD/2) \quad (4\text{-}2)$$

$$D_{M95} = 50 f_M (LOD) \quad (4\text{-}3)$$

A GSD of 1.52 was calculated from D_M and D_{M95}. The results for D_M are in Table 4-9.

Domestic Ore

The analysis of exposure data for the period November 13, 1944, to February 26, 1945, when domestic ore with minimal radium content was being processed, was similar to that made for the preceding period. The results are displayed in Table 4-10 and Table 4-11. For routine processing of domestic ore, the weighmaster and ball mill operator were assigned the highest annual dose. Most of this dose is missed dose.

Table 4-10. Step I gamma exposure data for cleanup and domestic ore processing, November 13, 1944, to February 26, 1945.

Category	# Weekly results	Dosimeter fraction f_D	Missed fraction f_M	For results in dosimeter fraction	
				GM m_D (mR/wk)	GSD
Ball mill operator	0	(a)	(a)	(a)	(a)
Cleanup[b]	15	0.3333	0.6667	30.0	1.00
Loader	164	0.0061	0.9939	34.2	(c)
Ore sampler	11	0.0000	1.0000	0.0	(d)
Process operator	74	0.1486	0.8514	32.9	1.17
Weighmaster	11	0.1818	0.8182	35.5	1.27

a. Not available because no weekly data was reported.
b. Cleanup was an occasional short-term activity that was not part of routine domestic ore processing.
c. Not available because only one result was ≥LOD.
d. Not available because no results were ≥LOD.

Table 4-11. Step I gamma exposure rate parameters for cleanup and domestic ore processing, November 13, 1944, to February 26, 1945.

Category	D_D[a] (R/yr)	GSD	D_M[a] (R/yr)	GSD	$D_D + D_M$	Relative $D_D + D_M$[b]
Ball mill operator[c]	0.32	1.27	0.90	1.52	1.22	28%
Cleanup[d]	0.50	1.27	0.73	1.52	1.23	(e)
Loader	0.01	1.27	1.09	1.52	1.10	21%
Ore sampler	0.00	1.27	1.10	1.52	1.10	34%
Process operator	0.24	1.27	0.94	1.52	1.18	42%
Weighmaster	0.32	1.27	0.90	1.52	1.22	28%

a. Each dose value represents the median of a lognormal distribution.
b. $D_D + D_M$ for domestic ore processing as percent of its value for same category for African ore processing.
c. There were no measured data available for the ball mill operator. This category was assigned the same dose as weighmaster because of the similarity of their exposure potential and because of the possibility that processing of some types of domestic ore might require handling by the Ball Mill Operator.
d. Cleanup was an occasional short-term activity that was not part of routine domestic ore processing.
e. Not available.

The results obtained for this period also apply to the processing of scrap and residues from uranium processing. The uranium processing would have removed most of the content of radium and radium progeny.

4.1.2.2.1 Estimation of Worker Doses

African Ore

Step I Process Workers

The annual gamma exposures of the monitored Step I workers (Table 4-9) were considered to be sufficiently similar that all could be assigned the same value. For dose reconstruction purposes, all were assigned an exposure rate of 5.35 R/yr, the highest value of $D_D + D_M$ in Table 4-9, and a GSD of 2.61, the highest GSD in Table 4-9.

Other Workers

For other workers, most of the task-related sources of gamma exposure were probably comparable to or weaker in effect than they were for Step I process workers. Step II and Step III process workers were handling refined uranium materials with much lower gamma emission rates because much of the radium and radium progeny had been removed. Nickel process workers were handling nonradioactive materials. In general, support personnel – such as laboratory, maintenance, and janitorial personnel – had less frequent exposure to radium-containing materials or dealt with smaller quantities or worked at greater distances from the materials. They also worked in less dusty environments and so had less exposure to airborne radioactivity and to floor and wall contamination.

One source of gamma exposure, however, was more dependent on worker location than on worker task. This was radiation from large quantities of ore that were stored in ore piles. A report of radiation surveys on March 2 and 3, 1944 (Ferry 1944f) identified two 200,000-lb piles of 10% ore and one 300,000-lb pile of 10% ore "in the receiving room" as well as one 30- by 30- by 12-ft pile of 8% ore at an unspecified location (percentages specify weight % of U_3O_8). Based on an analysis of typical pitchblende ore at Linde, the ore density was estimated as 3.25 g/cm^3 (Wallo, Vierzba, and Roberts 1981, Table C-1). Therefore, the 30- by 30- by 12-ft pile contained approximately 2 million lb of ore and was the largest of the four piles.

The probable location of the ore was the shipping and receiving platform at the south end of Building 30 (Figure 2-3). This would have placed the ore near the "grizzly enclosure" through which

ore was dumped onto the conveyor belt at the start of Step I processing. A letter to Linde reporting on a March 2, 1944, inspection recommended cleaning up and rearranging the platform so that ore would be stored in its southeast corner (Cranch 1944a), and a letter from the Ceramics Plant on May 8, 1944, indicated compliance with this recommendation (Rehm 1944b).

In Ferry (1944f), an exposure rate of 0.23 R/8-hr day was reported at 1.5 ft from the 30- by 30- by 12-ft pile. Potential exposures at more distant locations were estimated by mathematical analysis. For the region very close to an ore pile, the dependence of exposure rate on distance was determined by fitting data from Linde pilot plant studies for a 4- by 5- by 20-ft pile of 8% ore (Skinner 1944; Wiesendanger 1944). The fit was obtained using the MicroShield computer program (MicroShield Team 2003), which is based on the point kernel methodology. The values from the program were multiplied by a single scaling factor that was chosen to provide a best fit to the measured data. The adjustment compensated for simplifications in the modeling. Figure 4-1 shows the data and the fit. Gamma exposure rates measured during pilot plant studies at Linde Ceramics (Skinner 1944) are shown by the diamond symbols. The rectangles show the fit to the data with distance assumed to be measured along the perpendicular bisector of a 4- by 20-ft. face. The ore grade (8%) was assumed to be that determined in the analyses reported by Wiesendanger (1944, p. 18).The data covers the range of 0 to 10 ft from the 4- by 20-ft face. The adjusted MicroShield output fit the data to within

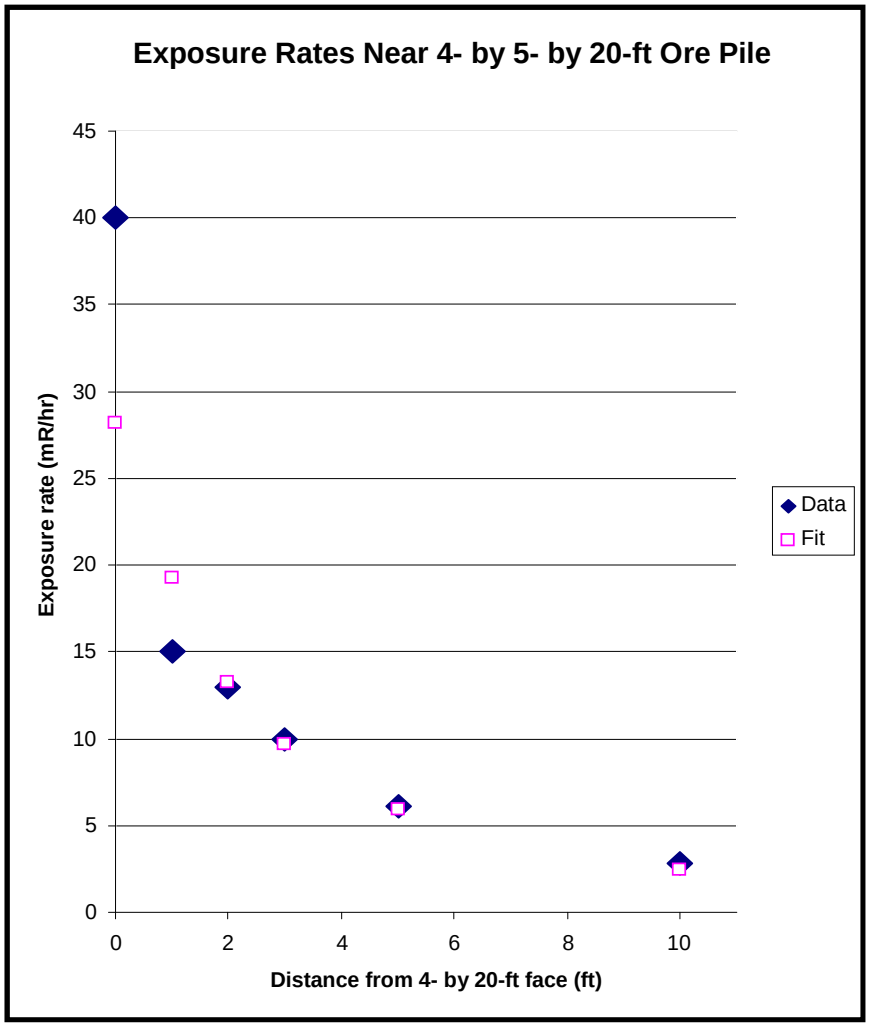

Figure 4-1. Fit to gamma exposure data for a 4- by 5- by 20-ft pile of 8% African pitchblende ore.

±10% between 3 ft and 10 ft from the face. The results were extended to larger distances and to a 30- by 30- by 12-ft pile by more MicroShield calculations that were scaled with the same factor.

Figure 4-2 shows the MicroShield results and an analytic function fit to them. The diamond-shaped symbols show calculated values of exposure rate as a function of distance from the pile center. Distance is measured along the perpendicular bisector of a 12- by 30-ft face. The values were calculated using MicroShield (MicroShield Team 2003). The line is a plot of the analytic function obtained by a least-squares fit to the MicroShield results. The function is:

$$(1/r^2) \times \exp[a \times \exp(-b \times r) + (c - d \times r)] \qquad (4\text{-}4)$$

where $a = 1.906$, $b = 3.126\times10^{-02}$, $c = 7.660$, and $d = 1.521\times10^{-03}$. The fit agreed with the calculated values to within ±3.0% over the range of 16 to 2,000 ft from the center of the pile.

The function facilitates calculation of exposures at arbitrary distances. As a check on the methodology, the analytic function was used to calculate the exposure rate at a point where it had been measured, which was 1.5 ft from the face. The result was 0.19 R/8 hr, which is 17% below the measured value. This degree of difference is consistent with the uncertainties typical of this type measurement and the approximations inherent in the MicroShield code.

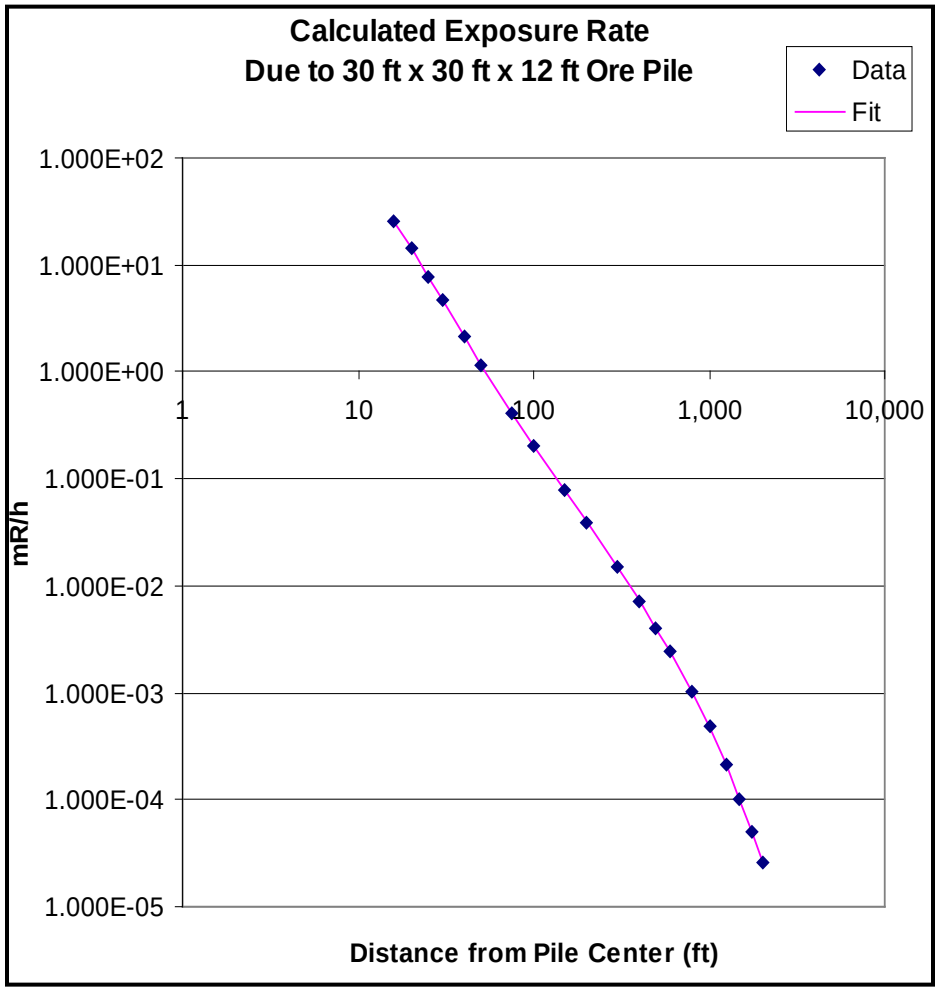

Figure 4-2. Calculated exposure rates and analytic fit for a 30- by 30- by 12-ft pile of 8% African pitchblende ore.

The calculated results are expected to overestimate radiation levels well away from the pile. The calculations account for air attenuation, but not for attenuation due to objects such as walls, floors, and equipment. In addition, the calculations are based on distance along the perpendicular bisector of a pile face. Exposure rate at any distance is lower for a point not on the perpendicular bisector.

To estimate the impact on the other workers in Building 30, who did not wear dosimeters, exposure rates due to a 30- by 30- by 12-ft pile in the southeast corner of Building 30 were calculated for various facilities in the building. In each case, the exposure rate was calculated for the point closest to the center of the pile. The calculated rates are in Table 4-12. They range from 0.28 R/yr (for the laboratory) to 4.9 R/yr (for the lunchroom). The results indicate a potential for exposures to the other workers in Building 30 comparable to the exposures to Step I process workers.

Although few of the other workers spent the major portion of their work time near the southeast corner of Building 30, some might have worked near other ore piles, near tailings piles, or near large quantities of ore in process. Because the locations and quantities of these other sources are not known in sufficient detail, any worker who spent a substantial amount of time in Building 30 had a potential for significant gamma exposure. Such workers included not only the production personnel in the building, but also nurses (whose office was there), maintenance personnel (who came there to maintain, repair, or refurbish), janitors, personnel who used locker rooms or the lunchroom, and office personnel who might have had to regularly visit the building (e.g., product accountants, industrial relations representatives, the fire marshal, and engineers). Last, it appears that there were office

Table 4-12. Estimated maximum exposure rates in selected Building 30 facilities due to 30-by 30- by 12-ft pile of 8% ore in southeast corner.

Facility	Distance (ft)[a]	Exposure rate (R/yr)[b]
Bag washer room	73	1.09E+00
Laboratory	128	2.81E–01
Locker & wash rooms	43	4.46E+00
Lunchroom	42	4.89E+00
Sampling room	57	2.10E+00

a. Distance is the distance from the pile center to the closest point that is inside the facility and 3 ft above the floor.
b. Based on 2,550 hr/yr exposure.

personnel who were permanently stationed in the building. A February 1944 letter from the plant addressing the frequency of required medical examinations states that it would not be appropriate to reduce the examination frequency for all office personnel because “we have some stenographers and clerks, stores clerks, etc., who work in the main building and are from time to time exposed to the same hazards as the operating personnel” (Holmes 1944a). In view of the likely difficulty of establishing that a particular worker did not spend a substantial portion of time in Building 30, for dose reconstruction it is assumed that all plant personnel had a significant potential for gamma exposure. All Ceramics Plant employees were assigned a gamma dose of 5.35 R/yr for the whole body, based on the results for the most exposed group of monitored process workers (Table 4-9).

Domestic Ore

During domestic ore processing, the gamma exposure rates were lower because the ore contained lower proportions of radium and other uranium series progeny than African ore. However, it is difficult to estimate how much lower because of uncertainty as to how much African ore radioactivity remained in the preprocessed ores at Linde (Section 2.2). The value of $D_D + D_M$ for Step I process operators during domestic ore processing (Table 4-11) is 42% of the value during African ore processing (Table 4-9). This difference is judged not to be large enough to justify the complication of using lower doses for periods of domestic ore processing. Therefore, for dose reconstruction during the 1943 to 1946 production period, the gamma exposure rate of 5.35 R/yr was assumed to apply for the whole period.

4.1.2.2.2 Outdoor Exposure Rate

The average outdoor gamma exposure rate to which workers could have been exposed during production was assumed to be at most equal to the indoor level in Building 30 based on the reasoning

in Section 4.1.1. Therefore, the outdoor gamma rate was estimated as 0.131 mR/hr based on the level before vacuuming and flushing in Table 4-1. For 0.5 hr/wd exposure, 6 wd/wk, and 50 wk/yr, the average worker exposure would have been 0.020 R/yr. This is negligible in comparison with the indoor exposures given the approximate nature of the estimates and was ignored.

4.1.3 Standby, 1946 to 1947

Little information is available about the status of activities during the standby period. It is likely that the onsite staff consisted primarily of a small number of management and janitorial personnel – both of whom worked primarily in an office environment – and guards. For dose reconstruction, each worker during standby was classified as either a guard or a general worker, and worker time was assumed to have been spent in an office building, in production buildings, and outdoors. Averaged over the entire standby period, each worker's allocation of time was assumed to have been as indicated by the occupancy factors in Table 4-13.

Table 4-13. Ceramics Plant beta and gamma radiation rates during standby.

Parameter	Category			Time-weighted radiation rate[a]	
	Office	Production	Outdoors	Beta (rem/yr)	Gamma (R/yr)
Beta (mrem/hr)	0.000	0.676	0.676	(b)	(b)
Gamma (mR/hr)	0.000	0.131	0.131	(b)	(b)
Occupancy factor					
General worker	0.833	0.111	0.056	3.04E-01	5.91E-02
Guard	0.756	0.111	0.133	4.46E-01	8.67E-02

a. Based on 9.0 hr/d exposure, 6 d/wk, 50 wk/yr. Based on the underlying data and judgment, a GSD of 3 is assigned. The beta and gamma rates are for the whole body.
b. Not applicable.

Gamma radiation levels were measured at six locations in Building 30 on October 22, 1946. Measurements were made at 1 in. from the surface of interest. The results were reported as 0 R/8 hr for four of the locations and 0.005 R/8 hr (0.625 mR/hr) for the other two locations (each near an ore dumping grill) (Howland 1946). Because the dumping grill was one of the most contaminated spots in the plant, the exposure rate there was not considered typical of the conditions that would have been encountered upon occasional entry during standby. Instead, the indoor gamma and beta levels for a production building were taken as the values in Table 4-1 before vacuum cleaning and flushing. Outdoor gamma and beta levels were taken as equal to the indoor rates based on the reasoning used above in the discussion of the preproduction period. The gamma and beta radiation rates in an office building were assumed to be zero.

Table 4-13 summarizes the calculation of annual radiation rates based on the above parameters.

Because there would have been little need for direct handling of radioactive materials by Ceramics Plant workers in this period, beta dose rate to the hands and forearms was taken as equal to the beta dose rate to the remainder of the body.

4.1.4 Rehabilitation and Production, 1947 to 1949

4.1.4.1 Beta

For part of the period from 1947 to 1949, weekly film badge measurements of beta and gamma exposure were available (LAPC undated b*)*. The data had been “de-identified” by removal of personnel names (Wallace 2003), but many of the records were labeled with a job title. The data file contained about 6,000 records from January 7, 1948 through December 12, 1949. By comparison with a copy of a laboratory film badge report (AEC 1949b), it was determined that the date identified the beginning of the week when the badge was worn. For records before March 15, 1948, badges were processed by the University of Rochester (Heatherton 1948g, Osinski 1948a); thereafter, they

were processed by the AEC New York Directed Operations Office Radiological Laboratory (Osinski 1948b; AEC 1949b). When use of the badges was initiated, the intention was for them to "be available for use by all entering a contaminated area" (Heatherton 1947).

The computer file had three columns of data but no labels to indicate units or type of radiation. The third column was obviously the total radiation dose. The first column was determined to be beta radiation and the second gamma radiation by comparison with a laboratory film badge report (AEC 1949b). The report stated both the beta and gamma units to be "exposure, mr." Progress reports by the Linde health physicist referred to the film badge results sometimes as "mr" and sometimes as "mrep" (Heatherton 1949d,e). The beta units are designated here as mrem and are assumed for purposes of dose reconstruction to be equivalent to shallow dose at 0.07 mm, $H_p(0.07)$. Before March 15, 1948, a beta value below the limit of detection was reported as "<80." After March 15, 1948, the limit of beta detection was not explicitly stated, but it was deduced to be 35 mrem from an examination of the data. Some of the results were reported as "0," "*," or "**". These were interpreted as meaning less than the LOD.

The beta film badge data was analyzed to determine dose rates for various job categories. Approximately one-third of the badge records were labeled with the job title NA. These were ignored because it could not be determined which had been worn by a worker and which had been used for other purposes (e.g., as controls or to measure process dose). Job titles that were specified in the records were binned into categories that combined jobs that were judged to have had similar exposure potential. Table 4-14 displays the job titles in each category.

Table 4-14. Job categories for 1947 to 1949 Step III film badge analysis.

Job title	Category
AEC	Office
Asst engineer	Engineer
Asst supt	Superintendent
Carpenter	Maintenance
Chem oper	Process operator
Chem oper A	Process operator
Chem oper B	Process operator
Chem oper C	Process operator
Chem oper G L	Process operator
Chemist	Chemist
Control engr	Engineer
Electrician	Maintenance
Electrician hlpr	Maintenance
Fire chief	Engineer
Foreman (unspecified type)	(a)
Group leader (unspecified type)	(a)
Health physicist	Engineer
Janitor	Janitor
Labor foreman[b]	Loader
Laundry helper	Laundry worker
Laundry worker	Laundry worker
Loader	Loader
Loader foreman	Loader
Mail girl	Office
Maint	Maintenance
Maint foreman	Maintenance
Maint g l	Maintenance
Maint group leader	Maintenance
Maintenance	Maintenance
Millwright A	Maintenance
Millwright C	Maintenance
Moveman	Loader
Painter	Maintenance
Painter B	Maintenance
Personnel	Office
Pipefitter	Maintenance
Process foreman	Process operator
Prod accounting	Office
Property dept	Office
Safety engr	Engineer
Seamster	Seamster
Security agent	Office
Shift storekpr	Storekeeper
Shift stores att	Storekeeper
Shipping & rec	Storekeeper
Storekeeper	Storekeeper
Superintendent	Superintendent
Tool crib keeper	Storekeeper
Trades helper	Maintenance
Truck driver	Truck driver
Weighmaster	Weighmaster
Welder	Maintenance

a. There were two records with the job title Foreman and one with the job title Group Leader. They were not included in the gamma dose analysis because the title was not descriptive enough to identify the work activity.
b. From available data (Heatherton 1949b; AEC 1949b), it was deduced that the worker with the job title Labor Foreman was supervising loaders.

Table 4-15 displays statistical characteristics of the beta film badge data. The data were assumed to have a lognormal distribution. The parameters f_D, f_M, and m_D were defined as in the analysis of the 1943 to 1946 gamma dosimetry data (Section 4.1.2.2). For categories in which there were 50 or more badge results, f_D was taken as equal to the number of results at or above the LOD divided by the total number of badge measurements and f_M was taken as $(1 - f_D)$. Categories with fewer than 50 results were examined on a case-by-case basis. If the workers in the category were rarely badged, it was assumed that they only occasionally entered the Linde radiation area. The value assigned to f_D was the number of results at or above the LOD divided by 50, which is the number of workweeks per year. The missed fraction f_M was taken as (1/50) times the highest number of results below the LOD for any member of the category. In the present case, this alternative method was applied to the categories Office and Superintendent.

Table 4-15. Step III beta dose data, 1947 to 1949.

Job	# Badges	Dosimeter fraction f_D	Missed fraction f_M	For results ≥LOD	
				GM m_D (mrem/wk)	GSD
Chemist	183	0.0055	0.9945	95	(a)
Engineer	243	0.0535	0.9465	113	2.12
Janitor	261	0.0421	0.9579	132	2.17
Laundry	115	0.1565	0.8435	106	1.65
Loader	304	0.0559	0.9441	78	1.47
Maintenance	913	0.0526	0.9474	146	1.62
Office	9	0.0000	0.0600	(b)	(b)
Process operator	1,814	0.1527	0.8473	145	1.81
Seamster	44	0.0909	0.9091	141	1.31
Storekeeper	207	0.0048	0.9952	120	(a)
Superintendent	3	0.0000	0.0400	(b)	(b)
Truck driver	18	0.0000	1.0000	(b)	(b)
Weighmaster	61	0.0492	0.9508	136	1.42

a. Only one result was ≥LOD; GSD could not be determined.
b. No results ≥LOD; GM and GSD could not be assigned.

The estimated median annual dosimeter dose D_D and the median annual missed dose D_M for a 50-week year and their GSDs were determined in the same way as for the 1943 to 1946 gamma data (Equations 4-1, 4-2, and 4-3 in Section 4.1.2.2). Because badges would not have been worn until workers arrived indoors, an outdoor beta dose rate of 0.10 rem/yr (from Section 4.1.2.1.2) was added to the D_M values. For the office worker and superintendent, the outdoor contribution was larger than the indoor D_M value, so the outdoor GSD of 3 was assigned. For the other categories, the outdoor contribution was not the major portion of the sum, so the GSD was taken as not changed by the addition. Table 4-16 displays the results.

Table 4-16. Beta dose rates for Step III, 1947 to 1949.

Job	D_D[a] (rem/yr)	D_D[a] GSD[b]	$D_M + OD$[a] (rem/yr)	$D_M + OD$[a] GSD	$D_D + D_M + OD$ (rem/yr)
Process operator	1.11	2.17	0.84	1.52	1.95
Laundry	0.83	2.17	0.84	1.52	1.67
Seamster	0.64	2.17	0.90	1.52	1.54
Maintenance	0.38	2.17	0.93	1.52	1.31
Weighmaster	0.33	2.17	0.93	1.52	1.26
Engineer	0.30	2.17	0.93	1.52	1.23
Janitor	0.28	2.17	0.94	1.52	1.22
Loader	0.22	2.17	0.93	1.52	1.15
Chemist	0.03	2.17	0.97	1.52	1.00
Storekeeper	0.03	2.17	0.97	1.52	1.00
Truck driver	(c)	(c)	0.98	1.52	0.98
Office	(c)	(c)	0.15	3.00	0.15
Superintendent	(c)	(c)	0.14	3.00	0.14

a. Each D_D and D_M value represents the estimated median of a lognormal distribution. OD = estimated outdoor dose rate.
b. The highest GSD from Table 4-15 was assumed for all categories for which a GSD can be assigned.
c. No result ≥LOD.

For most jobs, the largest contribution to $D_D + D_M + OD$ was missed dose D_M. The production workers in 1947 to 1949 were probably scattered among three buildings – Building 38, the location of the Step III processing facilities; Building 30, which contained a loading dock, locker rooms, lunchroom, nurses' office, and chemistry laboratory; and Building 31, which contained shop facilities. None of the buildings had been decontaminated. The average beta radiation level in Building 30 for this period was estimated (Section 4.1.1) as 0.676 mrem/hr (value before vacuuming and flushing in Table 4-1), which is equivalent to 34.5 mrem/wk and 1.7 rem/yr for a person spending 8.5 hr/d, 6 d/wk, and 50 wk/yr in the building. This is comparable to the film badge LOD for beta radiation (35 mrem/wk) and so might have been a substantial portion of the missed dose. Based on this estimate of 1.7 rem/yr of possible dose and the lowest film badge $D_D + D_M$ value in Table 4-16 of approximately 1 rem/yr for workers, who were regularly badged, it was assumed that all workers, even office workers, received a beta dose of at least 1 rem/yr. Office workers were included because it is unknown which office workers spent substantial portions of their time in production buildings. Some workers who might have had offices in the office building might also have spent much of their workday in the production area (e.g., engineers, health physicist, and fire marshal), and some clerical workers might have actually had work locations in the production area [discussion near the end of Section 4.1.3.2.1 and Holmes (1944a)].

Categories

For dose reconstruction, the jobs were grouped into two categories – medium and low. The medium category was assigned a dose rate of 1.95 rem/yr. Workers whose jobs might have required frequent handling of radioactive materials were placed in the medium category. The tool crib attendant was assigned to the medium category because of the possible handling of contaminated tools. The truck driver was placed in the medium category because it was the practice at the plant in 1943 to 1946 for truck drivers to work as loaders when not driving a truck (LAPC 1945b, p. 10); it was assumed that the same practice applied from 1947 to 1949. Engineers and the plant health physicist (considered a type of engineer) were placed in the medium category because of their roles in closely monitoring and troubleshooting production activities.

The low category was assigned a dose rate of 1.00 rem/yr. Assigned to it were most office workers, the superintendent, and storekeepers. It also included guards and the fire marshal.

A list of jobs with the category assignment of each is provided in Section 4.4.

Rehabilitation
Beta dose rates during rehabilitation were assumed to be the same as during production to account for exposures while cleaning contaminated facilities, setting up, and testing equipment.

Hands and Forearms Dose
Film badges were worn on the chest (Heatherton 1948h). Measurements during various Step III process operations showed that dose rates to other areas of the body differed from dose rates at the film badge location with the difference dependent on the operation and the area of the body (Heatherton 1948h). The results are shown in Table A-6. Based on these data, it was estimated for this site profile that the average dose to the hands and forearms of a worker handling radioactive material was 3 times the chest dose and that average dose to the remainder of the body was equal to the chest dose. For each body area, the estimate was based on the average ratio for the studied operations with the result rounded up to the nearest integer. No credit was taken for wearing gloves. For dose reconstruction, a hands and forearms dose of 5.85 rem/yr was assumed for the medium category. Workers in the low category were assumed to have a hands and forearms dose of 1 rem/yr, which is the same as the dose assigned to the remainder of the body.

4.1.4.2 Gamma

As discussed in Section 4.1.4.1, weekly film badge measurements of beta and gamma exposure during 1947 to 1949 production were available (LAPC undated b*)*. As noted, the measurement units were specified as “mr” or “mrep.” It is assumed here that for gamma radiation the units were equivalent to milliroentgen. Results reported as “0,” “*,” or “**” were interpreted as meaning less than the LOD. For records before March 15, 1948, a gamma value below the LOD was reported as “0” or “<50.” For records after March 15, 1948, the limit of gamma detection was not explicitly stated but was deduced to be 45 mR from an examination of the data. A few scattered reports of nonzero exposures below 45 mR were judged not to reflect routine measuring capabilities.

The procedure for analyzing the gamma data and determining annual gamma exposures was the same as that for the analysis of the beta data (Section 4.1.4.1). Table 4-17 summarizes the statistical characteristics of the gamma data.

Table 4-17. Step III gamma exposure data, 1947 to 1949.

Category	# Badges	Dosimeter fraction f_D [a]	Missed fraction f_M [b]	For results ≥LOD	
				GM m_D (mR/wk)	GSD
Chemist	183	0.0164	0.9836	82	1.41
Engineer	243	0.0247	0.9753	145	2.13
Janitor	261	0.0115	0.9885	62	1.27
Laundry	115	0.0783	0.9217	81	1.46
Loader	304	0.0789	0.9211	141	2.14
Maintenance	913	0.0164	0.9836	87	1.38
Office	9	0.0200	0.0400	80	(c)
Process operator	1,814	0.0309	0.9691	86	1.52
Seamster	44	0.0000	1.0000	(d)	(d)
Storekeeper	207	0.0048	0.9952	65	(c)
Superintendent	3	0.0200	0.0400	50	(c)
Truck driver	18	0.0000	1.0000	(d)	(d)
Weighmaster	61	0.0000	1.0000	(d)	(d)

a. f_D = fraction of measurements ≥LOD.
b. f_M = fraction of measurements <LOD.
c. Only one result was ≥LOD; GSD could not be determined.
d. No results ≥LOD: GM and GSD could not be assigned.

The estimated median annual dosimeter dose D_D and median annual missed dose D_M for a 50-week year and their GSDs were determined in the same way as for the 1947 to 1949 beta data

(Section 4.1.4.1). Because badges would not have been worn until workers arrived indoors, an outdoor gamma exposure rate of 0.02 R/yr (Section 4.1.2.2.2) was added to the D_M values. Because the outdoor contribution was not the major portion of any of the sums, the GSD was taken as not changed by the addition. Table 4-18 displays the gamma exposure rates that were determined.

As in the case of 1947 to 1949 beta dose (end of Section 4.1.4.1), it is to be expected that there was a contribution to gamma dose due to floor and wall contamination in the buildings. The average gamma radiation level in Building 30 for this period is estimated in Section 4.1.1 as 0.131 mR/hr (value before vacuuming and flushing in Table 4-1), which is equivalent to 6.7 mR/wk and 0.34 R/yr for a person who spent 8.5 hr/d, 6 d/wk, and 50 wk/yr in the building. This is below the film badge LOD for gamma radiation (45 mR/wk). Any clerical personnel who worked in production buildings (see Holmes 1944a) were assumed to have received this estimated floor and wall radiation plus the 0.14 R/yr that was received by the occasionally badged office worker for a total of 0.48 R/yr. Because it might not be easy to identify which clerical personnel worked in a production building, all unbadged or infrequently badged personnel were assumed to have received 0.48 R/yr.

Categories
For dose reconstruction, the jobs were grouped into two categories – medium and low – with the gamma category assignment for each job the same as its 1947 to 1949 beta category assignment (Categories discussion in Section 4.1.4.1). Jobs in the medium gamma category were assigned an

Table 4-18. Gamma exposure rates for Step III, 1947 to 1949.

Job	D_D[a]		$D_M + OD$[a]		$D_D + D_M + OD$
	(R/yr)	GSD[b]	(R/yr)	GSD	(R/yr)
Loader	0.56	2.14	1.06	1.52	1.61
Laundry	0.32	2.14	1.06	1.52	1.37
Engineer	0.18	2.14	1.12	1.52	1.30
Process operator	0.13	2.14	1.11	1.52	1.24
Maintenance	0.07	2.14	1.13	1.52	1.20
Chemist	0.07	2.14	1.13	1.52	1.19
Janitor	0.04	2.14	1.13	1.52	1.17
Storekeeper	0.02	2.14	1.14	1.52	1.15
Seamster	(c)	(c)	1.14	1.52	1.14
Truck driver	(c)	(c)	1.14	1.52	1.14
Weighmaster	(c)	(c)	1.14	1.52	1.14
Office	0.08	2.14	0.06	1.52	0.14
Superintendent	0.05	2.14	0.06	1.52	0.11

a. Each D_D and D_M value represents the estimated median of a lognormal distribution. *OD* = estimated outdoor dose rate.
b. The highest GSD from Table 4-17 was assumed for all categories for which a GSD can be assigned.
c. No results ≥LOD.

exposure rate of 1.61 R/yr for the whole body. Jobs in the low category were assigned an exposure rate of 0.48 R/yr for the whole body.

Rehabilitation
Gamma exposure rates during rehabilitation were assumed to be the same as during production in order to account for exposures while cleaning contaminated facilities, setting up, and testing equipment.

4.1.5 Cleanup, 1949 to December 31, 1953

Cleanup of the Ceramics Plant is discussed in Section 2.6. Floor and wall radiation levels, which were measured at the start and end of the 1949 to 1950 decontamination of Building 30, were available and are summarized in Table 4-1. In addition, results were available from film badge

measurements of beta and gamma radiation during the 1948 removal of equipment from Building 30. These results are summarized in Table 4-19. Because the results in Table 4-19 are film badge measurements, they include floor and wall radiation in addition to radiation from contaminated equipment.

Table 4-19. Beta and gamma radiation measured during equipment removal, 1948.

Beta dose rate			
D_D (rem/yr)[a]	GSD	D_M (rem/yr)[a]	GSD
1.99	1.53	0.52	1.52
Gamma exposure rate			
D_D (R/yr)[a]	GSD	D_M (R/yr)[a]	GSD
0.89	1.70	0.94	1.52

a. D_D and D_M are dosimeter dose and missed dose, respectively, as defined in NIOSH (2007a), pp. 30 and 32.

For external dose reconstruction, each worker is classified in one of the following categories:

- Cleanup worker,
- Cleanup support worker, and
- Non-cleanup worker.

A cleanup worker is defined as a worker who was directly engaged in the removal of radioactive equipment, components, or contamination (e.g., by sandblasting, vacuuming, or washing) for a substantial portion of the workday. This category includes equipment operators (such as sandblaster operators) and laborers who were involved in moving and cleaning activities. A cleanup support worker is one who spent a substantial portion of the workday in the building or area being decontaminated to support the cleanup activities but was not actively engaged in the removal activities. Examples are the plant health physicist and a stores worker in a building being cleaned. All remaining plant personnel are considered non-cleanup workers. For a cleanup worker, the radiation estimate for both beta and gamma was taken as:

$$D_D + D_M + OD \quad (4\text{-}5)$$

where D_D and D_M were taken from Table 4-19. The parameter OD is the outdoor dose rate. As for the preproduction and standby periods, it was estimated as equal to the estimated indoor floor and wall level at 3 ft before vacuum cleaning and flushing in Table 4-1. A cleanup support worker was assumed to have only half as much exposure to indoor radiation. The radiation level estimate was taken as:

$$0.5(D_D + D_M) + OD \quad (4\text{-}6)$$

for both beta and gamma. Finally, a non-cleanup worker was assumed to have only 5% of the exposure to gamma radiation of a cleanup worker and to have a beta exposure equal to that calculated for Tonawanda Plant workers after 1946 (numbers were not adjusted for exposure time). The radiation level was estimated as:

$$0.05(D_D + D_M) + OD \quad (4\text{-}7)$$

for gamma radiation. The beta radiation values specified are for all parts of the body. Table 4-20 presents the results.

The beta dose rates in Table 4-20 are based primarily on film badge measurements. The film badges were worn on the chest (Heatherton 1948h). The hands and forearms would have been closer than the chest to radioactive materials during some of the steps involved in cleanup (e.g., dismantling and

removing equipment, and scrubbing contaminated surfaces). For dose reconstruction, the beta dose rate to the hands and forearms of a cleanup worker was taken as 3 times that to the remainder of the body. The factor three was based on measurements discussed in Hands and Forearms Dose in Section 4.1.4.1.

4.2 TONAWANDA LABORATORY BETA AND GAMMA EXPOSURE

For external dose reconstruction, workers at Tonawanda Laboratory are classified into two categories – research and office. The research category includes all personnel who performed hands-on work in research facilities (laboratories, fabrication facilities, or pilot plants) or who provided some type of support for these facilities that involved working in or very close to them. It includes scientists; technicians; and shop, maintenance, stores, and janitorial personnel. The office category includes all personnel who had primarily desk jobs although they might occasionally have visited the research facilities. It includes secretaries, clerks, draftspersons, and high-level managers. Three periods (R&D, cleanup, and postcleanup) are considered for Tonawanda Laboratory external dose reconstruction. Data on worker doses were not available for Tonawanda Laboratory, but survey results indicated that radiation levels during pilot plant operations were similar to later measurements at the Ceramics Plant. Therefore, radiation level estimates were based to a large extent on the

Table 4-20. External dose reconstruction parameters for cleanup.

Period/category	Beta[a]		Gamma[b]	
	Median (rem/yr)[c]	GSD[d]	Median (R/yr)[c]	GSD[d]
07/01/1949–12/31/1950, 6-day week				
Cleanup worker	2.61E+00	4.04	1.85E+00	4.04
Cleanup support worker	1.36E+00	4.04	9.34E-01	4.04
Non-cleanup worker	3.26E-01	3.00	1.11E-01	4.04
01/01/1951–12/31/1953, 5-day week				
Cleanup worker	2.18E+00	4.04	1.54E+00	4.04
Cleanup support worker	1.13E+00	4.04	7.78E-01	4.04
Non-cleanup worker	3.26E-02	3.00	9.26E-02	4.04

a. For the cleanup support and non-cleanup workers, the indicated beta rate is to the whole body. For the cleanup worker, the rate is to all parts of the body except the hands and forearms. For these, the rate is 3 times higher. See text.
b. Gamma exposure rates are for the whole body.
c. Annual rates based on 8.5 hr/wd indoors, 0.5 hr/wd outdoors, 50 wk/yr, except as noted for the non-cleanup worker.
d. The GSD for the beta value for the non-cleanup worker is the estimated GSD for outdoor beta dose. The other GSD values are based on the assumption that for indoor exposure the 95th-percentile value is 10 times the median.

estimates for workers who did similar work at the Ceramics Plant. Table 4-21 shows the assumed annual radiation rates and their bases.

Table 4-21. Tonawanda Laboratory beta and gamma radiation, 1942 to 1953.

Period			Beta				Gamma		
			rem/yr						
Start	End	Job	Hands & forearms	Remainder of body	GSD	Basis[a]	R/yr[b]	GSD	Basis[a]
R&D									
10/01/1942	07/31/1946	Research	1.11E+02	3.70E+01	1.52	A	5.35E+00	2.61	F
		Office	3.00E+00	3.00E+00	2.65	B	5.35E+00	2.61	B
Cleanup									
08/01/1946	12/31/1946	Research	7.83E+00	2.61E+00	4.04	C	1.85E+00	4.04	C
		Office	1.01E-01	1.01E-01	3.00	D	1.11E-01	4.04	D
Postcleanup									
01/01/1947	12/31/1953	All	3.26E-01[c]	3.26E-01[c]	(d)	E	6.80E-02	(d)	E

a. Basis:
   1. One-half of value for most exposed Ceramics Plant process worker during 1943 to 1946 production (Step I process operator). The factor of 2 reduction allows for the lower intensity of research work in comparison with production. For example, the amount of material processed and the day-to-day hands on radiation work would have been significantly lower than for the Step I to III operators.
   2. Based on Ceramics Plant office worker during 1943 to 1946 production.
   3. Cleanup worker during Ceramics Plant decontamination.
   4. Non-cleanup worker during Ceramics Plant decontamination.
   5. Building 30 postcontamination level (Table 4-1) for 8.5 hr/wd plus outdoor level on Tonawanda site for 0.5 hr/wd. Outdoor level assumed equal to Building 30 level before vacuum cleaning and flushing (Table 4-1). A 6-day week was assumed.
   6. Based on most exposed Ceramics Plant process worker during 1943 to 1946 production (Step I process operator).

b. Where the gamma exposure rate estimate was based on dosimetry data for the Ceramics Plant (available as values of D_D and D_M) plus possibly also outdoor dose OD, the gamma exposure rate for Tonawanda Laboratory was approximated as the sum of the parameters (e.g., $D_D + D_M + OD$). Gamma exposure rates are for the whole body.

c. The increased beta radiation level for office workers after the end of cleanup is due in part to allowing for the possibility that any worker could have been stationed anywhere in the facility once it was all considered clean.

d. Not estimated.

4.3 NEUTRON EXPOSURE

No neutron exposure measurements are available for the Linde site. Neutron production by means of the alpha-neutron reaction would have resulted in a relatively small neutron dose component during Step I conversion of African ore to purified U_3O_8; Step II conversion of U_3O_8 to UO_2; and Step III conversion of UO_2 to UF_4. Neutron dose rates per gram of natural uranium in the compounds that were processed at Linde are presented in Table 4-22.

Table 4-22. Natural uranium per gram dose rates at 1 ft.

Chemical form	Dose rate[a] (rem/hr-gram)
African ore (U_3O_8) with alpha emitting progeny in secular equilibrium through Ra-226	2.05E-11
Purified UO_3 and UO_2 with no alpha emitting progeny	7.91E-12
UF4/UF6 without alpha emitting progeny	6.62E-10

a. ORAUT-OTIB-0024 (ORAUT 2005b) data used to calculate the alpha-neutron dose rates is from Shleien, Slaback, and Birky (1998), Salmon and Hermann (1992), Reilly et al. (1991), and DOE (2000).

The following assumptions were made about the conversion from neutron production rate from the alpha-neutron reaction to an annual dose at the receptor point:

1. Point source geometry was used to produce a nominal ambient neutron dose rate.

2. Point source strength was based on the daily rate of U_3O_8 handling or UF_4 production. Self-shielding within the source-target compound matrix was assumed to be negligible.

3. A ^{238}U:^{235}U composition of 99.3% to 0.711% by weight for natural uranium was assumed.

4. The entire mass of uranium compounds was conservatively assumed to be attributable to the uranium (source) content.

5. Average neutron energy from alpha-neutron reactions was 2 MeV.

6. Dose equivalent rate-to-fluence rate conversion factor for 2 MeV was 1.3×10^{-4} rem/hr per neutron/cm^2-s.

7. The work year consisted of 8 hr/d, 6 d/wk, and 50 wk/yr.

During the Step I and Step II conversion processes in Building 30 (Ceramics Plant) the bounding neutron dose rate would be attributable to operations that involved handling of African Ore with alpha-emitting progeny in secular equilibrium through ^{226}Ra. This arises from the fact that the African ore feed material contained alpha-emitting progeny that contributed to the alpha-neutron production rate. As indicated in Table 4-22, the African ore form produced neutron dose rates that are 2.6 times greater than the purified UO_3 and UO_2 with no alpha-emitting progeny. In addition, during Step I processing, water, a neutron moderator, was added to the digestion process after ore milling. Similarly, during Step II processing, uranyl nitrate hexahydrate was added to ether, an organic hydrocarbon, and the solution was washed with water. The presence of these neutron moderators in the Step I and Step II processes would decrease the average neutron energy and consequently reduce the dose equivalent rate-to-fluence rate conversion factor. As a consequence, the dose that was received by loaders (or "Movemen") would be bounding for all personnel who were engaged in Step I and Step II processing activities. As indicated in the Table 4-22, during the Step III conversion process in Building 38, the neutron dose rate from UF_4 is almost 85 times greater than from the UO_2 feed material. Therefore, the dose contribution from handling UO_2 feed material can be neglected.

As discussed in Section 2.3.2, the Linde Ceramics Plant (Building 30) processed approximately 26,000 MT (26 million kilograms) of ore over a period of 37 months beginning in June 1943 and ending in July 1946. Assuming a 6-day workweek for 50 wk/yr, and a uniform rate of ore processing, 28,111 kg/wd (5,622 kg of uranium) were handled at Linde. The assumption favorable to claimants that all of the ore was African ore has been made although 30% of the ore was known to be preprocessed and would contain significantly less alpha-emitting progeny. The assumption favorable to claimants that all of the ore was 20% U_3O_8 by weight has also been made although the Linde site literature indicates that the ore contained 3% to 20% U_3O_8 by weight. According to the available Linde literature, the highest weekly production rate (Step III) of UF_4 at the Linde Ceramics Plant (Building 38) was 41,624 pounds (Kent 1949a) or 3,147 kg/d.

The only potentially significant source of neutron exposure at Linde would have been neutrons produced by the alpha-neutron reaction in materials where uranium was mixed with elements of low atomic number such as fluorine and oxygen. Ceramics Plant (Buildings 30 and 38) personnel are assumed to be exposed to one-tenth of the daily production amount of U_3O_8 or UF_4 at a distance of 1 ft. This quantity of material is an estimate that is favorable to claimants of the time-averaged amount of material likely to have been close to the maximally exposed worker during a work shift. The factor of 10 reduction takes into account several factors: that the plant operated around the clock so that each shift dealt with only one-third of a day's throughput; that many workers were involved in each type of operation so that each worked closely with only a portion of a shift's throughput; that an individual who worked in the vicinity of a large quantity of material (e.g., barrels of finished product) would on the average have been much more than 1 ft away because of the large volume it would have occupied; and that even a worker in a job that involved being close to large quantities of material also had other activities at larger distances from the source term.

Major uranium-related projects at Tonawanda Laboratory were as follows:

- R&D for Steps I, II and III. This work occurred primarily from October 1942 to December 1943 (Jenness and Ewing 1943). The largest-scale efforts were pilot plant projects. The Step I pilot plant produced 2 to 3 t of U_3O_8 per week in November 1942 (Bonsib 1942). The Step III equipment could convert up to 90 lb of UO_2 to UF_4 in a single run (LAPC 1946c, p. 84).

- Conversion of UF_6 to UO_3. This work appears to have been done after November 1943 (because it is not mentioned in the progress report for work from October 1942 to November 1943; Jenness and Ewing 1943). Interim reports were issued from July 1944 through July 1945, and a final report was issued in May 1946 (LAPC 1946d, p. 111). Typical runs involved use of a 150-lb cylinder of UF_6 and produced a few pounds of product.

- Grinding UF_4. The laboratory conducted research on grinding UF_4 at an unknown time between October 1942 and May 1946 (LAPC 1946c, p. 89). The work involved processing 2,750 lb of UF_4. The description of the program suggests that its duration was short (a few weeks).

Based on the above and allowing for date uncertainties and the storage of materials before and after processing or research, the following model was used to estimate neutron doses at Tonawanda Laboratory:

- Ore containing 3 t of U_3O_8 was processed per week, which corresponds to a daily processing rate of 454 kg from October 1, 1942, through February 29, 1944.

- Exposures were to 100 lb of UF_4 or UF_6 per week daily from October 1, 1942, to July 31, 1946. This accounts for the Step III and UF_6 to UO_3 work.

- Exposures were to 3,000 lb of UF_4 for 2 months at some point between October 1, 1942, and July 31, 1946. This accounts for the UF_4 grinding research. The neutron dose in a year is equivalent to that which would be produced by continuous exposure to 500 lb of UF_4 for a year. Because the dates of the work are uncertain, it was assumed that there was continuous exposure to 500 lb of UF_4 from October 1, 1942, to July 31, 1946.

To reflect all types of UF_4 and UF_6 work, a daily source term of 600 lb (272 kg) was assumed. All source terms (both oxide and fluoride) were divided by 3 to account for the fact that no worker would be within close range of the source material at all times. This yielded a daily processing rate of 151 kg of U_3O_8 and a continuous exposure to 90.7 kg of UF_4 or UF_6. Tonawanda Laboratory (Building 14) personnel were assumed to be exposed to these source amounts at a distance of 1 ft. Table 4-23 summarizes the model parameters and presents calculated dose rates.

Table 4-23. Neutron dose rates and annual doses from the alpha-neutron sources.

Location/time period	Process	Alpha source	Target atom	Daily source term (g)	Dose rate at 1 foot (rem/hr)	Dose rate (rem/yr)
Building 30 06/01/1943–07/31/1946	Steps I and II	African ore 20% U by weight	Oxygen	5.62E+6	1.15E-5	2.77E-2
Building 38 07/25/1943–08/31/1946[a] 09/15/1947–09/30/1949[a]	Step III	UF_4	Fluorine	2.76E+6	2.08E-4	5.0E-1
Tonawanda Laboratory (Building 14) 10/01/1942–02/29/1944	Steps I and II	U_3O_8	Oxygen	1.51E+5	3.10E-6	7.44E-3
Tonawanda Laboratory (Building 14) 10/01/1942–07/31/1946	Step III; UF_4 grinding; UF_6 process gas to UO_3	UF_4 or UF_6	Fluorine	9.07E+4	6.01E-5	1.44E-1

a. The period of exposure was extended beyond the end of the production period to account for inventory left on hand. UF_4 that was produced at Linde was shipped to Electromet. The dates when shipping of all Linde inventory was completed were estimated from data in a Linde progress report (Kent 1949b) and data on production at Electromet (DOE undated, pp. 23-24).

4.4 EXTERNAL DOSE RECONSTRUCTION SUMMARY, OCTOBER 1, 1942, TO DECEMBER 31, 1953

This section summarizes guidelines and parameters for reconstruction of doses due to external radiation during October 1, 1942, to December 31, 1953 except for doses due to occupational medical exposure, which are addressed in Section 5.0. The parameters that are provided in this section are for use when individual worker data are unavailable or inadequate. Because of the SEC determination (Leavitt 2005) that it is infeasible to adequately reconstruct internal dose for the period from October 1, 1942, through October 31, 1947, dose estimates for this period are considered partial dose estimates.

For dose reconstruction, exposures for different jobs have been grouped into categories. Table 4-24 specifies the categories and the annual radiation doses or exposures assigned to each. For the production periods at the Ceramics Plant, categories were designated high, medium, and low. In other cases, designations more descriptive of the work activity are used.

Where the dose or exposure estimates were based on dosimetry parameters (dosimeter dose D_D and missed dose D_M) plus possible outdoor dose OD, the dose was estimated as the sum of the parameters ($D_D + D_M + OD$). Each estimate is considered to be the median of a lognormal distribution. A GSD of 3 was assigned to all beta and gamma dose estimates based on the typical GSD levels estimated for the underlying data; that is, beta and gamma doses were assigned as lognormal

Table 4-24. Summary of annual external exposure from AWE operations, 1942 to 1953.

Year[a]	Work category	Beta (rem)[a]		Gamma (R)[a]	Neutron (rem)[a,b]
		Hands & forearms	Rest of body		
Ceramics Plant (Buildings 30, 31, 37, and 38)					
1942[c]	All workers	2.55E–02	2.55E–02	4.97E–03	(d)
1943[e]	High	1.51E+02[f]	5.05E+01	3.65E+00	3.41E–01[g]
		5.05E+01[h]			
	Medium	1.20E+01	3.97E+00		
	Low	2.08E+00	2.08E+00		
1944 1945	High	2.21E+02[f]	7.40E+01	5.35E+00	5.00E–01[g]
		7.40E+01[h]			
	Medium	1.76E+01	5.85E+00		
	Low	3.00E+00	3.00E+00		
1946[i]	High	1.28E+02[f]	4.32E+01	3.11E+00	3.33E–01[g]
		4.32E+01[h]			
	Medium	1.04E+01	3.59E+00		
	Low	1.93E+00	1.93E+00		
1947[i]	Medium	2.04E+00	8.91E–01	5.37E–01	1.48E–01[g]
	Low	6.10E–01	6.10E–01	2.03E–01	
1948	Medium	5.85E+00	1.95E+00	1.61E+00	5.00E–01[g]
	Low	1.00E+00	1.00E+00	4.80E–01	
1949	Medium/low[k]	6.85E+00	2.28E+00	1.73E+00	3.74E–01[g]
	Cleanup				(d)
	Non-cleanup[l]	6.60E–01	6.60E–01	2.94E–01	
1950 1951 1952 1953	Cleanup[m]	7.83E+00	2.61E+00	1.85E+00	(d)
	Non-cleanup[n]	3.26E–01	3.26E–01	1.11E–01	
Tonawanda Laboratory (Building 14)					
1942[c]	Research	2.80E+01	9.33E+00	1.35E+00	3.63E–02
	Office	7.56E–01	7.56E–01		
1943	Research	1.11E+02	3.70E+01	5.35E+00	1.44E–01
	Office	3.00E+00	3.00E+00		
1944 1945	Research	1.11E+02	3.70E+01	5.35E+00	1.44E–01
	Office	3.00E+00	3.00E+00		
1946	Research[o]	6.78E+01	2.26E+01	3.88E+00	8.36E–02[p]
	Office[q]	1.78E+00	1.78E+00	3.15E+00	
1947 1948 1949 1950 1951 1952 1953	All workers	3.26E–01	3.26E–01	6.80E–02	(d)

a. Total annual exposure (dose) for the designated year. Prorated based on calendar year and applicable notations below.
b. Because of the possible difficulty in determining whether a worker was working with oxide or fluoride materials, each worker was assigned the larger neutron dose due to fluorides.
c. Exposure for the period from October 1 through December 31, 1942 only.
d. Neutron dose rate was negligible.
e. Values prorated: For January 1 through April 26, 1943 (preproduction period), applicable values from Table 4-1 applied; for April 27 to December 31, 1943, applicable 1944 to 1945 values applied.
(Example calculation: 1943 high-ball mill operator = 0.315 × 1.01E-01 + 0.685 × 2.21E+02 = 1.51E+02).
f. Based on 221 rem/yr for ball mill operator, Step I and Step II process operators, and weighmaster.
g. The Building 38 neutron dose rate for Step III processing was assumed to apply from April 27, 1943, to August 31, 1946, and from September 15, 1946, to September 30, 1949. The neutron dose rate was negligible from September 1, 1946, to September 14, 1949 (standby), and after September 30, 1949 (cleanup and postcleanup). The period of neutron exposure extended beyond the end of production in 1946 and 1949 due to remaining inventory of UF_4.
h. Based on 74 rem/yr for loader per Section 4.1.2.1.3.

i. Values prorated: For January 1 to July 31, 1946, applicable 1944 to 1945 values applied; for August 1 to December 31, 1946 (standby period), applicable values from Table 4-4 (guard) applied.
j. Values prorated: For January 1 to September 14, 1947 (standby period), applicable values from Table 4-4 (guard) applied; for September 15 to December 31, 1947, applicable 1948 values applied.
k. Values prorated: For January 1 to June 30, 1949 (Step III production), applicable 1948 medium values applied; for July 1 to December 31, 1949, 1950 to 1953 cleanup values applied.
l. Values prorated: For January 1 to June 30, 1949 (Step III production), applicable 1948 low values applied; for July 1 to December 31, 1949, 1950 to 1953 non-cleanup values applied.
m. All cleanup workers and cleanup support workers as defined in Section 4.1.5 are assigned to the cleanup exposure category. Parameters are those of the cleanup worker for a 6-day week in Table 4-20.
n. All non-cleanup workers as defined in Section 4.1.5 are assigned to the non-cleanup exposure category. Parameters are those of the non-cleanup worker for a 6-day week in Table 4-20.
o. Values prorated: For January 1 to July 31, 1946, applicable 1944 to 1945 values applied; for August 1 to December 31, 1946, applicable values from Table 4-21 (cleanup-R&D scenario) applied.
p. Includes neutron exposures through July 31, 1946.
q. Values prorated: For January 1 to July 31, 1946, applicable 1944 to 1945 values applied; for August 1 to December 31, 1946, applicable values from Table 4-21 (cleanup-office scenario) applied.

distributions with GSDs of 3 and an acute exposure rate. Neutron doses were assigned as a constant distribution and a chronic exposure rate.

Prorating of exposures over less than a year is not straightforward for all periods. If doses are to be prorated, the footnotes to Table 4-24 should be checked to determine how the dose should be distributed throughout the year. Beta doses were assigned as electrons with energy >15 keV. Gamma doses were assigned as photons with energy from 30 to 250 keV, which is favorable to claimants. Neutrons are assigned as neutrons of 0.1- to 2-MeV energy.

Table 4-25 shows how Ceramics Plant production jobs were assigned to the various categories. For 1943 to 1946, the assignments were based on Table 4-25 and the discussion in Section 4.1.2.1.3. For 1947 to 1949, the assignments were based on Tables 4-16 and 4-18 and the discussion in Section 4.1.4.2. In each case, some job titles that reflect subcategories of titles in the earlier tables were added.

For the Ceramics Plant cleanup period, the cleanup category includes both cleanup workers and cleanup support workers as defined in Section 4.1.5. For Tonawanda Laboratory, the research and office categories are defined in Section 4.2.

If the exact job of a worker is not listed, dose reconstruction should be based on the most similar job.

Table 4-25. Ceramics Plant worker beta and gamma external exposure categories for production periods

1943–1946 production		1947–1949 production	
Job	**Beta category**	**Job**	**Beta and gamma category[a]**
Ball mill operator	High	A operator	Medium
Chemist	Medium	B operator	Medium
Draftsman	Low	C operator	Medium
Engineer	Medium	Chemist	Medium
Fire inspector	Low	Fire marshal	Low
Guard	Low	First aid nurse	Low
Janitor	Medium	Foreman, labor	Medium
Lab technician	Medium	Foreman, loader	Medium
Laundry worker	Medium	Foreman, maintenance[b]	Medium
Loader/moveman	High	Foreman, step III	Medium
Maintenance worker[b]	Medium	Group leader, maintenance[b]	Medium
Nickel operator	Low	Group leader, step III	Medium
Nurse	Low	Guard	Low
Office worker[c]	Low	Janitor step III	Medium
Ore sampler	Medium	Janitor, outside step III	Low
Plant superintendent, asst supt	Low	Lab technician	Medium
Seamster, seamstress	Medium	Laundry man	Medium
Shipping & receiving clerk	Low	Loader/moveman	Medium
Step I process operator	High	Maintenance[b]	Medium
Step II process operator	High	Office employee, AEC	Low
Step III process operator	Medium	Office employee, Linde[c]	Low
Storekeeper/stock clerk	Low	Shipping & receiving clerk	Low
Tank farm operator	Low	Storekeeper/stock clerk	Low
Tool crib attendant	Medium	Timekeeper	Low
Truck driver/operator	High	Tool crib attendant	Medium
Weighmaster	High	Truck driver/operator	Medium
		Weighmaster	Medium

a. The category assignments are the same for beta and gamma radiation.
b. Maintenance personnel include all craftspersons who constructed, fabricated, repaired, or refurbished (e.g., carpenter, electrician, instrument repairman, lathe operator, mason, millwright, oiler, painter, pipe fitter, sheet metal worker, trades helper, and welder).
c. Office workers include accountant, bookkeeper, clerk, comptometer operator, draftsman, industrial relations representative, mail person, plant superintendent and assistant superintendent, product accountant, property department worker, secretary, security agent, stenographer, telephone operator, and typist.

5.0 OCCUPATIONAL MEDICAL EXPOSURE

NIOSH is required to account for dose from medical X-rays performed on an EEOICPA-covered site (either the covered site where the AWE work was being performed or a covered site where medical X-rays were performed as a service). Linde X-rays were performed off-site at a non-covered commercial facility (Holmes 1944b; Van Horn 1943c; Author unknown 1948, 1950). Therefore, the dose from medical X-rays is not accounted for in the overall estimated dose calculation (ORAUT 2011).

6.0 ESTIMATION OF EXPOSURES FROM RESIDUAL CONTAMINATION AFTER 1953

This section develops parameters for reconstruction of doses due to internal and external exposures at the Ceramics Plant starting January 1, 1954.

NIOSH has determined, with concurrence from the Secretary of DHHS, that internal doses at the Linde Ceramic Plant cannot be reconstructed with sufficient accuracy from the beginning of 1954 through the end of 1969. If monitoring data are available for workers who are included in the SEC

class, dose is to be assigned as appropriate based on such data. However, such dose reconstructions are still considered partial dose reconstructions because NIOSH has determined that internal exposures during the SEC class period cannot be bounded.

6.1 INTERNAL EXPOSURE

There are no radiological survey data available for the period during which Building 30 underwent remodeling. However, air samples were collected during decontamination of Building 30 (Heatherton 1950) and Building 31 (Author unknown 1954). Based on the nature of the likely activities during remodeling, the maximum air concentration from removing concrete with a pneumatic hammer (surfaces previously cleaned by sand blasting) was determined to be most representative. The maximum air concentration that was reported in the 1950 NYOO report (Heatherton 1950) for removing concrete with a pneumatic hammer (surfaces previously cleaned by sand blasting) was 2.3 MAC (where 1 MAC = 70 dpm/m^3) or 161 dpm/m^3.

Comprehensive surveys of the Tonawanda site buildings were made in 1976 (ORNL 1978) and 1981 (BNI 1982). Building 30 was found to be the most contaminated building on the site. The indoor airborne uranium concentration was measured as 1.9×10^{-2} pCi U/m^3 (ORNL 1978). This air concentration was used to evaluate personnel internal exposures to uranium for the residual period.

Therefore, an exponential interpolation was made between the uranium air concentrations that were assumed for the remediation period (161 dpm/m^3) and the levels that were measured in the 1976 survey (4.22×10^{-2} dpm/m^3). After 1976, the 1981 survey indicates a more consistent level of contamination. Therefore, the air concentrations after 1976 are assumed to be constant at the 1976 levels.

Intakes of the uranium progeny thorium and radium were based on a ratio that was calculated from the highest monthly outdoor airborne radionuclide concentrations at the Linde Tonawanda site from 2000 to 2004. The ingestion intake rates in Tables 5-1 and 6-2 were estimated using the steps in Section 3.7.

Table 6-1. Uranium and progeny inhalation intake rates (dpm/d) for January 1, 1970, through 2009.

Years	U-234[a]	Th-230[a]	Ra-226[a]	Po-210[a]	Pa-231[a]	Ac-227[a]
1970	1059	275	222	222	10.6	307
1971	268	69.7	56.3	56.3	2.68	77.7
1972	67.8	17.6	14.2	14.2	0.678	19.7
1973	17.1	4.45	3.59	3.59	0.171	4.96
1974	4.33	1.13	0.909	0.909	0.043	1.26
1975	1.10	0.285	0.230	0.230	0.011	0.318
1976–2009	0.277	0.072	0.058	0.058	0.003	0.080

a. Values listed are assigned as a constant distribution.

Table 6-2. Uranium and progeny ingestion intake rates (dpm/d) for January 1, 1970, through 2009.

Years	U-234[a]	Th-230[a]	Ra-226[a]	Po-210[a]	Pa-231[a]	Ac-227[a]
1970	22.1	5.73	4.63	4.63	0.221	6.40
1971	5.58	1.45	1.17	1.17	0.056	1.62
1972	1.41	0.367	0.297	0.297	0.014	0.410
1973	0.356	0.093	0.075	0.075	0.004	0.103
1974	0.090	0.023	0.019	0.019	0.001	0.026
1975	0.023	0.006	0.005	0.005	0.0002	0.007
1976–2009	0.006	0.002	0.001	0.001	0.0001	0.002

a. Listed values are assigned as a constant distribution.

Radon Progeny

There are no radiological radon survey data available for the remodeling period (1954 to 1969). Therefore the radon air concentrations for the end of the remodeling period are assumed to be the same as the previous period (0.480 WLM/yr) as described in Section 3.5.1.

Fifty-five measurements of radon progeny concentration were made between 1976 (ORNL 1978) and 1981 (BNI 1982) in the Tonawanda site buildings. The highest radon concentration (from Building 31 in 1981) was used to evaluate radon exposures. No correction was made for natural background radon, although these levels were comparable to the levels in Building 31 in 1981.

A fit was made between the levels that were used for the remediation period (0.48 WLM/yr) to the 1981 results (0.201 WLM/yr). During the remediation of Building 31 between 1995 and 1998 (McKenzie and Uziel 1998), radon concentration levels were measured. These levels were consistent with the levels that were measured in the 1976 and 1981 surveys. Therefore, the exposure rates after 1981 for radon are assumed to be constant at the 1981 levels. Table 6-3 lists these rates.

Table 6-3. Radon exposure summary from January 1, 1970, through 2009 (WLM/yr).

Years	Exposure rate[a]
1970	0.480
1971	0.444
1972	0.410
1973	0.379
1974	0.350
1975	0.323
1976	0.299
1977	0.276
1978	0.255
1979	0.235
1980	0.218
1981–2009	0.201

a. Values listed are assigned as a constant distribution.

6.2 EXTERNAL BETA AND GAMMA EXPOSURE

As noted, Building 30 was found in 1976 to be the most contaminated building on the site (ORNL 1978). Measured floor and wall radiation levels in Building 30 in 1976 were compared with similar measurements from 1950 immediately after its decontamination (see Section 4.1.1 and Heatherton 1950). The results were similar, but the 1950 values were slightly higher.

Outdoor gamma radiation levels at 1 m above the surface on and near the Tonawanda site were measured in 1976 and 1981. Natural background gamma radiation levels in the Tonawanda area were said to be 8 to 15 µR/hr (ORNL 1978, p. 15; BNI 1982, p. B-12). ORNL 1978 (p. 15) reported only isolated areas with levels significantly higher than background; readings in these areas were described as very nonuniform and varying from background up to 250 µR/hr. Regions with readings above about 20 µR/hr were indicated in a figure in the report (Fig. 26). Figure B-5 of BNI (1982) summarized the combined results of the two studies. Locations and values were shown for readings ≥25 µR/hr. BNI (1982, p. B-12) stated that the points of maximum radiation that were determined in the ground surveys were slightly displaced from but in general agreement with those that were determined in a 1979 aerial radiological survey. The total number of reported readings ≥25 µR/hr was 16. The net readings (after subtraction of 8 µR/hr to correct for background) had a GM of 94 µR/hr and a GSD of 3.95. This was taken as an estimate of worker exposure rate when outdoors. This estimate was assumed to apply starting January 1, 1954. No credit was taken for the soil remediation at the site that began in 2000 because the remediation has not been completed (Pilon 2004). The estimate is probably an overestimate for the following reasons: measurements were reported only for

areas that were identified as having above-background radiation levels, these areas occupied much less than half of the open area of the site (based on Fig. B-5 of BNI 1982), and the reported measurements were not averages over these areas but only readings at a few hot spots.

The beta dose rate at 3 ft above the ground that corresponded to the adopted value of gamma exposure rate was estimated as 4.38×10^{-1} mrem/hr, which is 4.66 times the gamma rate where 4.66 is the ratio of beta mrem/hr at 3 ft to gamma mR/hr at 3 ft for the floors and walls in Building 30 after decontamination (Table 4-1). It is assumed that the GSD for the beta dose rate would be the same as for the photon dose rate.

Because the radiation levels seemed to remain fairly constant, and because the levels were fairly low, the Tonawanda Laboratory exposures after cleanup, based on Building 30 contamination levels, were used to estimate the external exposure rate. No adjustments for changes in work hours were made. The radiation energy distributions were assumed to be same as those during the operational period (lognormal distribution with a GSD of 3). This results in an exposure rate of 0.068 rem/yr penetrating dose and 0.326 rem/yr nonpenetrating dose.

6.3 NEUTRON EXPOSURE

Because only small quantities of radioactive material were on the site in the period after cleanup, there were no significant neutron exposures.

6.4 UTILITY TUNNEL EXPOSURE

Tunnel Contamination
There are two pathways for the radiological contamination of the tunnels: (1) transport of the effluent through the ground formations into the tunnels directly, and (2) stormwater runoff contaminated by the waste effluent overflow (possibly remaining in the soil following well overflow) into the tunnels during flooding conditions.

The scenario of effluent transport through the ground formations and groundwater transport into the tunnels has multiple challenges. The effluent was pumped into the bedrock level at depths between 90 and 150 ft. Documentation of the greatest depth of the tunnel has not been found as yet, but it is likely that the tunnel would not have been nearly that deep given that access could be accomplished through covered manways. The information NIOSH has about accessing the tunnel for maintenance activities implies the tunnel would not have exceeded 20 to 30 ft in depth. Consideration of the cost associated with digging in bedrock to build the utility tunnel, and the associated cost of running the utilities back up to the buildings they would supply, supports this conclusion. If the tunnel did have a depth of 20 to 30 ft from the surface, upward transport of the radiological contamination would be required for the effluent to directly contaminate the tunnel level. The remedial investigation report (BNI 1993) describes, in some detail, the soil interval between the ground surface and the water table, known as the vadose zone, and how flow occurs through this layer. The report made a conservative estimate of vertical groundwater velocity and applied a retardation factor to account for the slower movement of contaminants through the vadose zone due to interactions between the contaminants and the soil particles. The report states the uranium contamination in groundwater would migrate 47 times slower than the estimated vertical groundwater velocity of 0.09 m/yr (0.3 ft/yr). The retardation factors for thorium and radium are typically greater than for uranium, so thorium and radium would migrate at a slower rate. Radiological contaminants from the effluents injected into the contact-zone aquifer underneath the Linde site appear to be immobilized in the particulate phase (BNI 1993, p. 5-17).

Another challenge to overcome in terms of tunnel contamination directly from the effluent is the construction of the tunnel. The Linde utility tunnel is constructed of reinforced concrete. There is evidence of degradation in several older sections where the concrete is brittle and cracked. Such

locations, along with cracks and seams, would be the best pathways for contaminated soil and groundwater to infiltrate the tunnel.

The assumption that the infiltration of the tunnel was due to contaminated stormwater runoff is also due consideration. It is possible that stormwater passing through contaminated soils and flooding the tunnel could leave contamination behind. It is unlikely that any soluble uranium contamination in the soil around the injection wells would remain available for contamination of the tunnel for very long beyond the end of use and overflow of the wells. Natural weather action would be likely to have scoured the soluble components from the soil long before the timeframe under evaluation (1954 to 2006; the final injections occurred in mid-July 1946), leaving insoluble uranium to contaminate the tunnel.

The limited distribution of the contamination in the tunnel, as demonstrated by the 2001 survey (USACE 2002b), is difficult to reconcile with the idea of contamination washing into the tunnel during flooding. The dose assessment performed by the USACE concludes the contamination in the utility tunnel is a result of operations associated with Building 14 and contaminated solid and in-leakage near Building 31. This conclusion is supported by the limited distribution of contamination in the tunnels. Had the tunnels been contaminated by stormwater runoff flooding as has been suggested, the distribution of the tunnel contamination would be expected to be more uniform because there is no evidence of any remediation of the tunnels. The action of pumping the stormwater from the tunnel would have further distributed and diluted any contamination that came in via floodwaters. This is not consistent with the distribution seen on the survey map.

From review of the locations of the former injection wells near Building 8 and the section of the utility tunnel closest to those wells, the tunnel is approximately 25 ft to the north of the injection wells at the closest point. The nearest contaminated (greater than 5,000 beta/m/100cm^2) tunnel areas (identified by USACE 2002b) are approximately 50 to 100 ft northwest of the closest injection well. The second set of injection wells, in the vicinity of Building 30, are approximately 80 ft, 180 ft, 280 ft, and 400 ft from the closest contaminated section of utility tunnel (which runs north to south just to the west of Building 31).

Source Term

The detailed radiological characterization performed in 2001 was based on beta surface contamination levels. Six to eight circumferential measurements were taken in 1-m increments along the tunnel. The maximum result was then recorded. In addition, hot spot readings were collected.

The beta contamination levels were then converted to individual isotopic activities based on gamma spectroscopy analysis of soil samples that were taken during the site remediation. The analysis of this data is presented in Table 6-4.

Table 6-4. Radionuclide fraction calculation using average Linde soil concentrations (USACE 2002b).

Isotope	Soil concentration gamma spec (pCi/g)	Uranium fraction f(U)	Soil concentration isotopic (pCi/g)	Source fraction (f_s)	Beta emission (β/dis)	Beta fraction (f_β)	Isotopic abundance per beta (S_A)
Radium-226	7.46	–	7.46	0.080	1.82	0.170	0.094
Thorium-230	8.49	–	8.49	0.092	–	–	0.107
Uranium-238	36.33	0.473	36.33	0.392	1.98	0.830	0.419
Uranium-235	–	0.022	1.69	0.018	–		0.020
Uranium-234	–	0.505	38.79	0.418	–	–	0.447
Total	–	–	92.76	1.000	3.80	–	–

Using the data in Table 6-4, the following equations can be used to convert the gross beta surface contamination levels to isotopic activities:

^{226}Ra (dpm/m^2) = 9.4 × [(β/min)/100 cm^2]
^{230}Th (dpm/m^2) = 10.7 × [(β/min)/100 cm^2]
^{238}U (dpm/m^2) = 41.9 × [(β/min)/100 cm^2]
^{235}U (dpm/m^2) = 2.0 × [(β/min)/100 cm^2]
^{234}U (dpm/m^2) = 44.7 × [(β/min)/100 cm^2]

The 1937 tunnels were not found to be contaminated with radium because the pre-World War II uranium-refining operations in Building 14 used preprocessed ores from the Colorado Plateau vanadium deposits, which did not contain radium. Therefore, for these tunnels, 100% of the beta is assumed to be from ^{238}U. This results in the following equations to convert the gross beta surface contamination levels to isotopic activities:

^{226}Ra (dpm/m^2) = 50.5 × [(β/min)/100 cm^2]
^{230}Th (dpm/m^2) = 2.3 × [(β/min)/100 cm^2]
^{238}U (dpm/m^2) = 53.8 × [(β/min)/100 cm^2]

An analysis of all of the gross beta survey data was performed and is summarized in Table 6-5.

Table 6-5. Gross beta survey data for utility tunnels.

50th percentile	6,430 (β/min)/100 cm^2
GSD	2.10
95th percentile[a]	30,870 (β/min)/100 cm^2

a. The 95th percentile result is the measured 95th not the fitted since the tail of the curve bent above the projected fit.

Table 6-6 provides isotopic contamination levels that were calculated using the equations above and the 95th-percentile gross beta surface contamination level.

Table 6-6. Utility tunnels surface activity (dpm/m^2).

Isotope	1954–1956 surface activity	1957 on surface activity
Radium-226	-	2.90E+05
Thorium-230	-	3.30E+05
Uranium-238	1.56E+06	1.29E+06
Uranium-235	7.10E+04	6.17E+04
Uranium-234	1.66E+06	1.38E+06

These contamination levels were assumed to be uniformly present on all surfaces of the tunnels for all years. This source term is considered to be favorable to the claimant for the following reasons:

1. The beta measurements are based on the highest observed measurements.

2. The beta measurements include both fixed and removable contamination (it is normally estimated that contamination is only 10% removable and therefore a contributor to airborne contamination. Including both fixed and removable contamination overestimates the contamination available for resuspension.).

3. Hot spot contamination levels were treated as uniformly distributed contaminated surfaces and included in the distribution.

4. The tunnels were reportedly a damp environment, which would have the effect of preventing resuspension of material.

Internal
Based on the 95th-percentile beta surface contamination level (Table 6-5), an air concentration activity was calculated based on guidance provided in ORAUT-OTIB-0070, *Dose Reconstruction during Residual Radioactivity Periods at Atomic Weapons Employer Facilities* (ORAUT 2008). The following assumptions were made:

- Breathing rate: 1.2 m^3/hr

Resuspension factor: 1×10^{-6} m^{-1}

It is assumed that trade workers and laborers worked in these tunnels doing maintenance for 8 hr/wd (2 months of the year) and for the other 10 months, a transit time of 10 min/wd using the tunnels to get between buildings. For all other workers, only the transit time of 10 min/wd should be applied year-round. This results in the intakes in Tables 6-7 and 6-8.

Table 6-7. Utility tunnels trade worker intake rates (dpm/yr).

Trade workers and laborers				
	1954–1956		1957 on	
Radionuclide	Inhalation rate	Ingestion rate	Inhalation rate	Ingestion rate
Radium-226	-	-	123.6	2.58
Thorium-230	-	-	140.7	2.93
Uranium-238	664.1	13.84	551.0	11.48
Uranium-235	30.2	0.63	26.3	0.55
Uranium-234	707.5	14.74	587.8	12.25

Table 6-8. Utility tunnels other worker intake rates (dpm/yr).

All other workers				
	1954–1956		1957 on	
Radionuclide	Inhalation rate	Ingestion rate	Inhalation rate	Ingestion rate
Radium-226	-	-	14.5	0.30
Thorium-230	-	-	16.5	0.34
Uranium-238	77.9	1.62	64.7	1.35
Uranium-235	3.6	0.07	3.1	0.06
Uranium-234	83.0	1.73	69.0	1.44

Radon
Based on Tunnel Contamination
Radon is the decay product of ^{226}Ra. The level of ^{226}Ra contamination in becquerels per square centimeter multiplied by the contaminated area provides the total ^{226}Ra contamination in becquerels (disintegrations per second). Because each disintegration of a ^{226}Ra atom produces a ^{222}Rn atom, this is then the production rate of ^{222}Rn in atoms per second. Multiplying by the ^{222}Rn decay constant converts the production rate into becquerels per second. Dividing this by the volume of air into which the radon is emitted produces the rate that the air concentration is increased.

The air concentration is decreased by two factors. The first is due to the radioactive decay of the radon itself, and the second is the rate that ventilation removes the radon from the tunnel. This can be expressed mathematically by the following equation:

$$\frac{dC_t}{dt} = \frac{S \times A \times \lambda}{V} - \lambda C_t - \lambda_V C_t \qquad (6\text{-}1)$$

where:

C_t = ^{222}Rn concentration in the tunnel (Bq/cm^3)
S = ^{226}Ra contamination value (Bq/cm^2)
A = surface area of the contaminated tunnel walls/ceiling and floor (cm^2)

V = air volume in the tunnel (cm^3)
λ = decay constant of ^{222}Rn gas (1/s)
λ_V = air exchange rate in the tunnel due to ventilation (1/s)

At steady state, the derivative equals zero and the equation can be rearranged to find the steady-state radon concentration in the tunnel, which can be expressed as:

$$C_t = \frac{S \times A \times \lambda}{V \times (\lambda + \lambda_V)} \quad (6\text{-}2)$$

The area of the tunnel surfaces (walls, ceiling, and floor) can be determined on a per-unit length basis. The volume of air associated with the unit length can be similarly determined. Because the area is divided by the volume, the per-unit basis cancels out. Using the parameters in Table 6-9 and converting units, the equilibrium radon concentration in the tunnel due to surface contamination can be calculated to be 18.35 pCi/L.

Table 6-9. Radon parameters for utility tunnels.

Parameter	Value	Units
S - Radium-226 contamination value (based on 290,000 dpm/m^2)	0.483	Bq/cm^2
A – area per linear cm (based on 200 cm square cross-section)	800	cm^2
V – volume per linear cm	40000	cm^3
λ - Radon-222 decay constant	2.1E-6	1/s
λ_V - Air exchange rate	2.78E-5	1/s

For Trade Workers & Laborer it is assumed that they worked in these tunnels doing maintenance for 8 hr/wd (2 months of the year) and for the other 10 months, a transit time of 10 min/wd using the tunnels to get between buildings. For all other workers only the transit time of 10 min/wd should be applied year-round. This results in the exposure rates provided in Table 6-10. Because radium was not present in the tunnels built before 1957, these exposure rates only apply after 1957.

Table 6-10. Exposure rates from utility tunnel radon contamination (WLM/yr).

Radionuclide	Trade workers	All other workers
Radon-222	0.192	0.022

Based on Surrounding Soil

The ^{222}Rn concentration in the tunnels at the Linde Ceramics Plant ($Rn_{tunnels}$) was estimated using the following equation:

$$Rn_{tunnels} = Rn_{Erie\ Co\ basements} \times Ra_{Linde}/Ra_{Erie\ Co} \quad (6\text{-}3)$$

where:

$Rn_{Erie\ Co\ basements}$ = ^{222}Rn concentration reported for basements in Erie County.
Ra_{Linde} = ^{226}Ra concentration in soils at Linde Ceramics.
$Ra_{Erie\ Co}$ = ^{226}Ra concentration in soils in Erie County.

The Linde site is just south of the county line in Erie County. The Erie County statistics for radon in basements have a GM of 1.68 pCi/L and a GSD of 3.95 pCi/L (NYDOH 2010). The Erie County ^{226}Ra soil concentration is taken to be 0.636 pCi/g and is applied as a constant value. This number was

derived from the number for Niagara County that Sanford Cohen & Associates (SC&A) calculated, using 1.81 ppm of uranium, and then proportionally increasing the result using the Erie County figure of 1.89 ppm (LBNL 2011).

The ^{226}Ra concentration in soils at the Ceramics Plant was taken from 51 soil sample results from an aggregate 59 ft of soil samples. Discrete probabilities were assigned for each foot of soil by assuming the concentration reported for each foot of soil had a 0.0169 (1/59) probability of occurrence.

The computer program @Risk was used to solve the equation for radon concentration in the tunnels. Input parameters were applied as the distribution described above. To solve the equation, 10 simulations were run with 10,000 iterations each, with different random seed numbers. The 95th-percentile value of 99.31 pCi/L provides a plausible upper-bound estimate for radon exposure in the Linde utility tunnels for all years at the Linde site beginning from the first MED production through the present.

It is assumed that trade workers and laborers worked in these tunnels doing maintenance for 8 hr/wd (2 months of the year) and for the other 10 months, a transit time of 10 min/wd using the tunnels to get between buildings. For all other workers only the transit time of 10 min/wd should be applied year-round. This results in the exposure rates in Table 6-11. Because radium was not present in the tunnels built before 1957, these exposure rates only apply after 1957.

Table 6-11. Radon exposure rates for utility tunnels from surrounding soils (WLM/yr).

Radionuclide	Trade workers	All other workers
Radon-222	1.037	0.122

External

External contamination and submersion dose coefficients factors (DCFs) were calculated for effective whole-body and skin dose based on information in Federal Guidance Report No. 12, *External Exposure to Radionuclides in Air, Water, and Soil* (Eckerman and Ryman 1993). These DCFs are based on a single-plane exposure model. However, the fact that the 95th-percentile surface contamination level (including hotspot activities in the distribution) is being applied to the entire length of the tunnel is considered to bound any dose that would be associated with contamination on the ceiling and walls of the tunnels.

It is assumed that trade workers and laborers worked in these tunnels doing maintenance for 8 hr/wd (2 months of the year) and for the other 10 months, a transit time of 10 min/wd using the tunnels to get between buildings. For all other workers only the transit time of 10 min/wd should be applied year-round.

Using these DCFs, occupancy factors, and the calculated surface activity and air concentrations from the internal dose section, an external dose potential was estimated. These potential external annual dose rates are listed in Table 6-12. An assumed energy distribution of 100% 30- to 250-keV photons for the whole-body dose and 100% >15-keV electrons for the skin dose is recommended as an assumption favorable to claimants. Since these estimates are based on source term data, external ambient DCFs (H*(10)) should be used to assign organ dose.

Table 6-12. Utility tunnels external exposure rates (rem/yr).

	1954–1956		1957 on	
Radionuclide	Gamma	Beta	Gamma	Beta
Trade workers	<0.001	0.033	0.001	0.035
All others workers	<0.001	0.004	<0.001	0.004

7.0 ATTRIBUTIONS AND ANNOTATIONS

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

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GLOSSARY

carnotite

Mineral of the form $K_2(UO_2)_2(VO_2)_2 \cdot {}^3(H_2O)$ found in Colorado and Utah and mined for its uranium and vanadium content. Much of the domestic ore processed at Linde Ceramics Plant derived from tailings from carnotite ore. Preprocessing of this material before shipment is believed to have removed much of its radium content (Wallo, Vierzba, and Roberts 1981).

day (d)

As used in this Linde document, a day refers to a calendar day unless otherwise described. A calendar day is any 24-hour day in a year. There are at least 365 calendar days in a year. A workday (wd) refers to a day that is assigned for work. The length of a workday depends on the amount of time spent on the job. Default assumptions are that there are 8 hours in a workday and 250 workdays in a year. At Linde, the number of hours per workday and the number of workdays per year for some employees were higher than the defaults for the early years.

Linde database

Set of records of Linde Ceramics Plant operations available to the National Institute for Occupational Safety and Health dose reconstruction project.

N.G. or NG Cake

Filter cake produced in Step II processing at Linde. There were two types of cakes produced called OK and N.G.

pitchblende

Mineral containing uranium oxide (UO_X) of variable composition ranging between UO_2 and U_3O_8 as well as radium. Raw ore from Africa was processed for its uranium content at Linde Ceramics Plant the raw ore was not prerefined and so contained all of the natural uranium radioactive progeny including radium.

Step I, Step II, and Step III

Sequential uranium production processes at Linde:

- Step I, conversion of ore to U_3O_8 (black oxide)
- Step II, conversion of U_3O_8 to UO_3 (orange oxide) to UO_2 (brown oxide)
- Step III, conversion of UO_2 to UF_4 (green salts)

torbernite

Mineral in the form of $Cu(UO_2)_2(PO_4)_2 \cdot 8\text{-}12(H_2O)$ processed for its uranium content. Like the African pitchblende processed at Linde, torbernite was not prerefined before its arrival at Linde and so contained all of the natural uranium radioactive progeny including radium.

ATTACHMENT A
BETA RADIATION
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LIST OF TABLES

This attachment contains data that was used in analyzing exposures of workers to beta radiation.

Table A-1. Maximum beta surface dose rates from various uranium-containing materials.

Source	Beta surface dose rate	
	mrad/hr[a,b]	rad/8 hr
Slab of U metal	233	1.86
UO_2	207	1.66
UO_3	204	1.63
U_3O_8	203	1.62
UF_4	179	1.43

a. From Table 2-7 on p. 2-18 of DOE (2000).
b. Beta surface dose rate in air through a polystyrene filter 7 mg/cm^2 thick.

Table A-2. Electron energy released by refined uranium.

	Principal nuclides	Electron energy per decay of nuclide (MeV/nt)[a]	Electron energy per decay of parent (MeV/nt)[a]	% of total
U-238 series	U-238	0.010	0.005	1.1
	Th-234	0.060	0.029	6.6
	Pa-234m	0.822	0.402	89.8
	U-234	0.013	0.006	1.4
U-235 series	U-235	0.049	0.001	0.2
	Th-231	0.165	0.004	0.8
		Total:	0.447	

a. nt = nuclear transformation.

ATTACHMENT A
BETA RADIATION
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Table A-3. Electron energy released by unrefined natural uranium ore.

	Principal nuclides	Electron energy per decay of nuclide (MeV/nt)[a]	Electron energy per decay of parent (MeV/nt)[a]	% of total
U-238 series	U-238	0.010	0.005	0.4
	Th-234	0.060	0.029	2.5
	Pa-234m	0.822	0.402	34.7
	U-234	0.013	0.006	0.5
	Th-230	0.015	0.007	0.6
	Ra-226	0.004	0.002	0.2
	Rn-222	0.000	0.000	0.0
	Po-218	0.000	0.000	0.0
	Pb-214	0.293	0.143	12.4
	Bi-214	0.659	0.322	27.8
	Po-214	0.000	0.000	0.0
	Pb-210	0.038	0.019	1.6
	Bi-210	0.389	0.190	16.4
	Po-210	0.000	0.000	0.0
U-235 series	U-235	0.049	0.001	0.1
	Th-231	0.165	0.004	0.3
	Pa-231	0.065	0.001	0.1
	Ac-227	0.016	0.000	0.0
	Th-227	0.053	0.001	0.1
	Ra-223	0.076	0.002	0.1
	Rn-219	0.006	0.000	0.0
	Po-215	0.000	0.000	0.0
	Pb-211	0.456	0.010	0.9
	Bi-211	0.010	0.000	0.0
	Tl-207	0.493	0.011	1.0
		Total:	1.157	

a. nt = nuclear transformation.

ATTACHMENT A
BETA RADIATION

Table A-4. Estimated attenuation of beta radiation with distance.

Distance from surface (m)	Relative dose rate[a]	
	Refined natural uranium[b]	Unrefined natural uranium ore or ore byproducts[c]
0.0	1.00	1.00
0.3	0.80	0.88
0.94	0.50	0.66
1.0	0.48	0.64
1.6	0.31	0.50
2.0	0.23	0.42
3.0	0.11	0.27
5.0	0.025	0.11
8.5[d]	0.000	0.024
10.0	(e)	0.012
12.6[f]	(e)	0.000

a. Only exponential attenuation due to air is considered. Additional reduction would come from geometric dispersion, which is ignored here.
b. Dose rate assumed to vary with distance as exp(-0.0074*x*), where *x* is distance from the surface in centimeters. This was derived from a fit to data in Coleman, Hudson, and Plato (1983).
c. Calculated falloff rate based on the assumption that all of the beta radiation is due to the 3.27-MeV beta ray from Bi-214. The dose rate is assumed to vary with distance as exp(-0.0044*x*), where *x* is distance from the surface in cm. The attenuation factor 0.0044/cm was estimated from the rule of thumb that the half thickness of a beta absorber is one-eighth of the range of the beta rays (Cember 1983, p. 97). For natural uranium ore and byproducts with beta emitters other than Bi-214, the actual falloff with distance would be greater because all other significant beta emissions have lower energies than 3.28 MeV.
d. Range of 2.28-MeV beta radiation per Cember (1983, p. 99).
e. No radiation at this distance.
f. Range of 3.27-MeV beta radiation per Cember (1983, p. 100).

Table A-5. Uranium beta dose reduction factors for apparel.

Item[a]	Fraction of beta dose remaining
Leather, medium weight	0.62
White cotton gloves	0.89
Tyvek coveralls	0.98
65% Dacron/35% cotton lab coat	0.91

a. Selected from Table 6-11 on p. 6-23 of DOE (2000).

ATTACHMENT A
BETA RADIATION
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Table A-6. Variation of dose rate with body location for Step III process operations.

Operation	(mR/hr)[a]					
	Hands, bare	**Hands, gloved**	**Face**	**Chest**	**Legs**	**Feet**
Handling empty trays	4	3	1.5	4	(b)	(b)
Loading trays (brown)	25	13	2	4.5	4.5	2
Handling loaded trays(brown)	25	13	5	16	(b)	(b)
Handling loaded trays (green)	25	16	7.5	25	(b)	(b)
Unloading operation (green)	25	(b)	7.5	7.5	4	2
Blending	15	10	7	8.5	12	(b)

	Radiation level relative to chest[c]						Radiation level relative to hands, bare[c]
	Hands, bare		**Face**	**Chest**	**Legs**	**Feet**	**Hands, gloved**
Handling empty trays	1.00		0.38	1.00	(b)	(b)	0.75
Loading trays (brown)	5.56		0.44	1.00	1.00	0.44	0.52
Handling loaded trays(brown)	1.56		0.31	1.00	(b)	(b)	0.52
Handling loaded trays (green)	1.00		0.30	1.00	(b)	(b)	0.64
Unloading operation (green)	3.33		1.00	1.00	0.53	0.27	(b)
Blending	1.76		0.82	1.00	1.41	(b)	0.67
Maximum	5.56		1.00	1.00	1.41	0.44	

a. Data from Heatherton (1948h).
b. No data.
c. Results in lower half of the table are based on the data in the upper half.

ATTACHMENT B
CODES AND SPECIAL TERMINOLOGY

Special codes and terminology were used in correspondence and documents that related to Linde's work for the MED and AEC in place of common names or to specify special forms of materials with particular characteristics. Over the course of time, some terms might have had multiple meanings. Table B-1 lists apparent definitions deduced during preparation of this document.

Table B-1. Codes and special terminology.

Code or term	Apparent meaning
C-103	UO_2 from DuPont; a feed material used for Linde Step III
C-306	UO_2 from Mallinckrodt; a feed material used for Linde Step III
C-316	UO_2 from Mallinckrodt; a feed material used for Linde Step III
F-29	U_3O_8 made at Linde
K-25	UF_6 gas containing various concentrations of U-235
K-65	Residues containing Ra and Pb that were byproducts of the ore to UO_3 conversion process at Mallinckrodt Chemical Works
L-19	A domestic ore processed at Linde; estimated to contain mostly 10% to 16.5% U_3O_8 (Aerospace 1981, Table B-1)
L-30	African pitchblende ore; estimated to contain 8% to 12% U_3O_8 (Aerospace 1981, Table B-1)
L-50	African pitchblende ore; estimated to contain 6.7% U_3O_8 (Aerospace 1981, Table B-1)
Mx, MX	Uranium or U_3O_8
MX-308	U_3O_8 (in ore)
My	Radium
Mz, MZ	Radon
O-71	UF_4
P-65	UO_2 produced by Linde Step II; a feed material used for Linde Step III
Product 65	P-65
Q-20	Torbernite ore processed at Linde; estimated to contain 17.7% U_3O_8 (Aerospace 1981, Table B-1)
R-10	African pitchblende ore processed at Linde; estimated to contain 3.5% U_3O_8 (Aerospace 1981, Table B-1)
T	Uranium
Tubealloy dioxide	UO_2
Tubealloy tetrafluoride	UF_4
X	Uranium or uranium ore

ATTACHMENT C
DATA SOURCES ON URANIUM PROGENY CONCENTRATIONS IN LINDE MATERIALS

calculated value
<MDA, mda shown

SRDB Ref ID	Date	Location	Description	U238	U234	U235	totalU	Th230	Ra226	Th232	Th228	Ac227	Pa231	units	Th230/U	Ra226/U	Th232/U	Ac/U	Pa/U
16294	1995	bldg 14	dry valve pit (dust)	1038	1068	53.8	2159.8	354.7	15.3	2.3	1.7			pCi/g	0.16	0.01	0.00		
			sump (dust)	1.4	1.7	0.07	3.17	0.72	0.52	0.2	0.22			pCi/g	0.23	0.16	0.06		
			corridor overhead (dust)	369	378.8	11.6	759.4	60.2	0.14	5	0.67			pCi/g	0.08	0.00	0.01		
			corridor wall (terra cotta block)	3.2	3.2	0.14	6.54	1.7	1.4	1.4	1.4			pCi/g	0.26	0.21	0.21		
14620	1978	bldg 30	air samples during D&D				1.90E-08	2.10E-09	1.10E-09					pCi/ml	0.11	0.06	0.00		
9009	1981	sediment, onsite and offsite	Ellicott Creek	0.82		0.05	1.69	0.6	0.55	0.7				pCi/g	0.36	0.33	0.41		
			Creek1	0.95		0.05	1.95	0.7	0.7	0.8				pCi/g	0.36	0.36	0.41		
			Twomile Creek - upstream	4.3		0.1	8.7	0.92	0.69	0.01				pCi/g	0.11	0.08	0.00		
			Twomile Creek - linde discharge	0.71		0.06	1.48	0.02	0.52	0.02				pCi/g	0.01	0.35	0.01		
			Twomile Creek - downstream	1.5		0.05	3.05	0.96	0.59	0.48				pCi/g	0.31	0.19	0.16		
			Storm Sewer	6.47		0.19	13.13	1.4	1.35	0.62				pCi/g	0.11	0.10	0.05		
			Storm Sewer	99		4.57	202.57	18	6.93	0.51				pCi/g	0.09	0.03	0.00		
			Storm Sewer	13		0.52	26.52	2	1.59	0.65				pCi/g	0.08	0.06	0.02		
			Storm Sewer	116		4.1	236.1	9.9	0.89	0.34				pCi/g	0.04	0.00	0.00		
			Storm Sewer	4.5		0.17	9.17	0.2	0.64	0.39				pCi/g	0.02	0.07	0.04		
			Sanitary Sewer	362		13	737	1.33	1.94	0.11				pCi/g	0.00	0.00	0.00		
			Sanitary Sewer	0.51		0.05	1.07	0.34	0.38	0.21				pCi/g	0.32	0.36	0.20		
9026	1990	Linde Soils	area 1 - mean	11.2			22.4	7.8	4.3	1.6				pCi/g	0.35	0.19	0.07		
			area 2 - mean	12.7			25.4	5.7	3.4	1.4				pCi/g	0.22	0.13	0.06		
			area 3 - mean	17.1			34.2	24.4	9.4	1.4				pCi/g	0.71	0.27	0.04		
			area 4 - mean	46.8			93.6	30.7	9.8	1.4				pCi/g	0.33	0.10	0.01		
8828	1981	Linde Soil and Sediment	near disposal well - subsurface (loc 11)	24.05		0.84	48.94	5.9	5.53	0.92		14.25	0.73	pCi/g	0.12	0.11	0.02	0.29	0.01
			test well debris (loc 13)	26.4		1.09	53.89	3.53	0.82	0.51		2.1	0.29	pCi/g	0.07	0.02	0.01	0.04	0.01
			Sanitary Sewer (loc 15)	362		12.93	736.93	1.33	1.14	0.11		5.54	0.95	pCi/g	0.00	0.00	0.00	0.01	0.00
			Storm Sewer (loc 19)	99.2		4.57	202.97	17.7	6.93	0.51		14.29	1.14	pCi/g	0.09	0.03	0.00	0.07	0.01
			Storm Sewer (loc 21)	116		4.1	236.1	3.89	0.89	0.34		3.07	0.39	pCi/g	0.02	0.00	0.00	0.01	0.00

ATTACHMENT D
LINDE URANIUM COWORKER ASSESSMENT FOR NOVEMBER 1947 TO JANUARY 1950

LIST OF TABLES

LIST OF FIGURES

Due to the limited availability of bioassay data from the Linde site, it was necessary to conduct a coworker study of all the bioassay data for use to determine intake estimates. The data that were used in this study were transcribed directly from hardcopy into worksheets. The data in the worksheets were verified (as indicated below), a statistical analysis was conducted and verified, and an intake assessment was conducted and verified. Each of these processes is further described below. The resultant intake tables are provided in the intake assessment section.

Data Verification

The Linde bioassay data were verified as follows:

1. Data were transcribed directly from source documents to spreadsheets by data entry personnel. A review of the datasets was conducted to determine the datasets in each source document (see Table D-1). The data were present in sampling lots (e.g., a group of samples related by sample date).

2. A single, master data spreadsheet was created by merging spreadsheets from individual source documents.

3. A completeness and accuracy review of 100% of the data was conducted by an individual independent of spreadsheet creation. Any discrepancies in data entry were identified and corrected. Two samples in the December 1950 dataset were excluded from further analysis because they were identified as belonging to employees at the electrometallurgical plant (i.e., not Ceramics Plant employees). One sample in the November 1948 dataset was excluded because it was a duplicate (same name and result).

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LINDE URANIUM COWORKER ASSESSMENT FOR NOVEMBER 1947 TO JANUARY 1950

Table D-1. Linde data sources in the Site Research Database (SRDB).

Period	Samples (count)	SRDB Ref ID		
		8955	11148	16979
Nov-1947 (11/11–11/12)	73	X		
Jan-1948 (01/04–01/06)	88	X	X	X[a]
Feb-1948 (02/20–02/23)	84	X	X	
June-1948[b] (06/17–06/19)	69	X	X	
Aug-1948 (08/18)	67	X	X	
Nov-1948 (11/12–11/17)	94	X	X	
Feb-1949 (02/07–02/10)	91	X	X[c]	
Apr-1949 (04/11)	4	X		
Jul-1949 (07/01–07/14)	88	X[d]		
Sep-1949 (09/11–09/12)	2	X		
Oct-1949 (10/20)	1	X		
Dec-1949 (12/30)	10	X		
Jan-1950 (01/23–01/27)	9	X		
	680			

a. Pages 97–98 and 696–697 only.
b. Datasheet in SRDB Ref ID 11148 version has handwritten correction of sampling month (from June to May).
c. Dataset is incomplete.
d. Samples are annotated as “termination”.

4. Dataset completeness was evaluated through a review of available data on the Linde urinalysis program.

   a. The urine data are from transmittals from the laboratory to Linde. Based on a review of each individual dataset, and after referencing was conducted between datasets (using the sample identifiers that did not appear to vary from dataset to dataset), it appeared that each individual dataset was complete.

   b. A handwritten listing of Linde urine datasets (ca. 1980) was found in the SRDB (Author unknown undated). Based on this listing, one dataset (October 1, 1948) is currently not available.

   c. The transmittal letter for the February 1948 urine samples (March 4, 1948), clarifies that Linde was to submit quarterly urine samples (instead of monthly, as was previously understood). The frequency of the available Linde urine data are consistent with this pattern (i.e., monthly sampling up to February 1948 and quarterly thereafter).

With the exception of the October 1948 dataset, all known urine data have been included in this study.

Statistical Analysis

The verified data were analyzed in accordance with the requirements in ORAUT-PROC-0095, *Generating Summary Statistics for Coworker Bioassay Data* (ORAUT 2006).

1. The 73 samples collected in November 1947 that were annotated as preemployment were not used in the coworker analysis because preemployment samples would not relate to exposures on the job.

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2. Seven followup samples (four in April 1949, two in September 1949, and one in October 1949) were excluded from statistical analysis. These followup samples indicated a reduction in urinary excretion in every case. Exclusion of these samples is favorable to claimants.

3. Two samples from December 1949 that were labeled "Area Plant - Electrometallurgical" were excluded from analysis because they were not Ceramics Plant employees.

4. One sample from November 1948 was a duplicate sample entry (typographical error) and therefore was excluded from further data analysis.

5. To perform statistical analysis, samples were grouped based on sampling date. Because most of the data were clustered around sampling campaigns, the midpoint of the sampling interval for each sampling event was generally assigned as the effective bioassay date. The effective bioassay dates are used in the Integrated Modules for Bioassay Analysis (IMBA) software to calculate the intake rates.

6. The uranium mass excretion rates were converted to the uranium activity excretion rates by applying the specific activity of natural uranium (683 pCi/mg) and urination rate of 1.4 L/d.

7. The analytical limit of detection was assumed to be 0.01 mg/L. This value is based on annotations on the urine data transmittal memoranda that accompanied most datasets. Because none of the data were explicitly labeled as <LOD, all nonzero data were included in the data ranking and the curve fitting.

8. A lognormal distribution was assumed for the urinary excretion data, and the 50th- and 84th-percentile uranium excretion rates were calculated using the method in ORAUT-PROC-0095 (ORAUT 2006). These excretion rates are listed in Table D-2.

Table D-2. Uranium urinary excretion data (pCi/d).

Period	Number of samples	Effective bioassay date	50th percentile	84th percentile
01/04/48–01/06/48	88	01/05/48	20.4	62.0
02/20/48–02/23/48	84	02/21/48	15.8	47.2
06/17/48–06/19/48	69	06/18/48	9.5	15.9
08/18/48	67	8/18/48	4.9	14.1
11/12/48–11/17/48	93	11/14/48	6.5	26.1
02/07/49–02/10/49	91	02/08/49	2.9	14.9
07/01/49–07/14/49	88	07/07/49	10.9	46.4
12/30/49–01/27/50	17	01/13/50	4.9	15.3
	597			

Intake Assessment
Urine results in Table D-2 were used in the intake assessment. Samples from between December 1949 and January 1950 were not used because the July 1949 dataset was labeled as termination samples for the bulk of the Linde employees. All urinary excretion rates were modeled as normally distributed 24-hour urine samples with a uniform relative error of 30%. The excretion data were modeled with IMBA for multiple chronic intakes of Type M or Type S uranium.

The intake rates, GSDs, and periods to which they apply are listed in Table D-3 for Type M uranium, and in Table D-4 for Type S uranium.

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Table D-3. Chronic intake rates (pCi/d) for Type M ^{234}U.

Start date	End date	50th-percentile value	84th-percentile value	GSD
11/01/1947	01/27/1950	74	297	4.0

Table D-4. Chronic intake rates (pCi/d) for Type S ^{234}U.

Start date	End date	50th percentile	84th percentile	GSD
11/01/1947	01/27/1950	1,884	8,085	4.3

Plots of expected and observed urinary excretion from these fits are shown in Figures D-1 through D-4. In each plot, the solid line represents a fit of the coworker bioassay data.

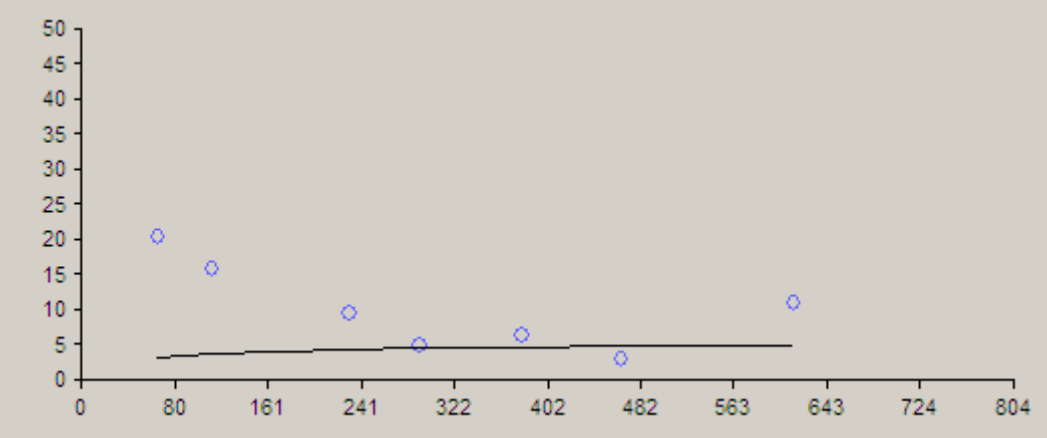

Figure D-1. Type M 50th-percentile value for November 1, 1947, to January 27, 1950.

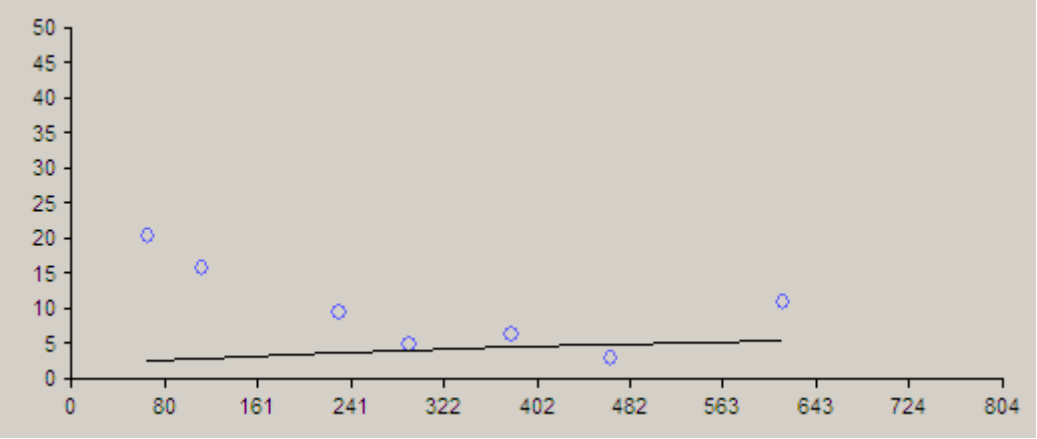

Figure D-2. Type S 50th-percentile value for November 1, 1947, to January 27, 1950.

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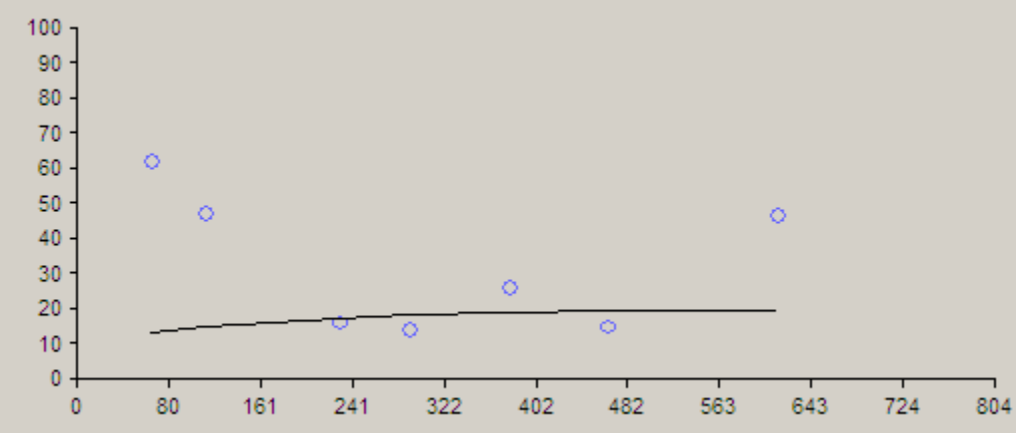

Figure D-3. Type M 84th-percentile value for November 1, 1947, to January 27, 1950.

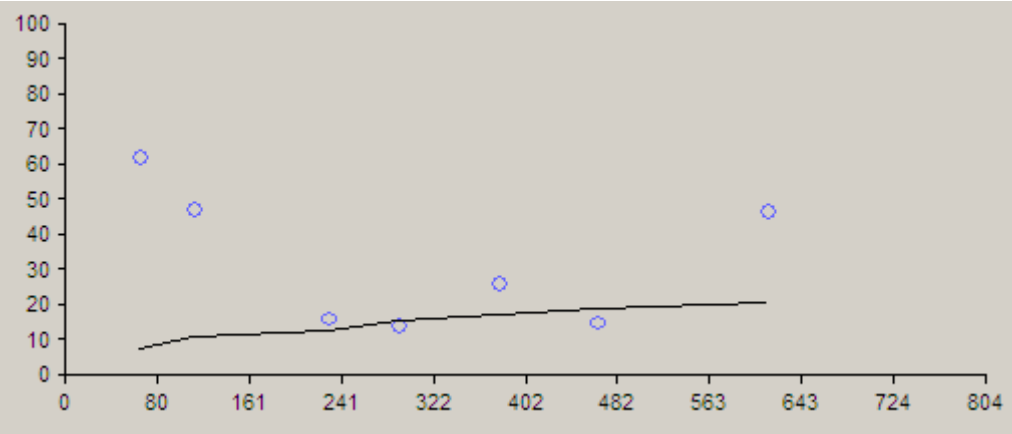

Figure D-4. Type S 84th-percentile value for November 1, 1947, to January 27, 1950.

ATTACHMENT E
FOCUSED ASSESSMENT OF DOSE CONSEQUENCES FROM URANIUM ORE BAGS ON THE SITE DURING THE POSTOPERATIONS PERIOD

Original Issue

> *During the SC&A-conducted interview in Buffalo, Linde site experts and past workers indicated that there were many thousands of used burlap bags stacked up in the open bay area behind Building 30 (see Attachment 3 of this review report). These bags were used for transporting uranium ore to Linde for processing. After the end of the operation period, the contaminated burlap bags were stored behind Building 30 awaiting disposal. Many Linde workers, operation staff, and administrative personnel sat on these contaminated bags during breaks and lunch periods. This practice continued for many years, exposing many people at close distances to beta and gamma radiation sources left over in the uranium-contaminated burlap bags. The site profile does not estimate the missed beta and gamma doses to workers resulting from sitting on or standing next to those burlap bags* (SC&A 2006).

Subsequent Analysis

During the March 26, 2007, Linde Working Group meeting the burlap bag issue was discussed by members of the working group, NIOSH, and Sanford Cohen & Associates (SC&A). The following task was defined by the working group (based on SC&A 2006):

> *NIOSH to investigate details of used burlap bags. Which bags (formerly containing African or domestic ore) were stored at which locations and during which periods of time. This may affect both internal and external exposures. Even though African ores were processed only during the SEC* [Special Exposure Cohort] *period (pre 10/31/1947), empty bags, that had contained African ore, may have been around longer (i.e., after 1950).*
>
> *NIOSH to determine whether there was an on-site incinerator to burn used burlap bags and, if so, the possible effects on internal and external exposures.*

In November 2007, an analysis of the burlap bag issue was conducted by NIOSH and submitted to the working group along with responses to the other issues that had been identified in the site profile review. Table 5-1 of the response document summarizes the issue and is reproduced below (NIOSH 2007b).

Date	Data Source	Data
Undated	Medical Section, Manhattan District, *Data Sheet for Industrial Hazard Rating* (MED undated)	• Step I material (ore) arrives in burlap bags, covered in paper sacks • Step III material (UO2) arrives in 75 lb drums
1943–1946	Linde Construction, *Operations Report* (LAPC 1946c)	• Empty bags and drums are held until they can be burned or scrapped • *Bags from dry ore are shaken in the bag shaker … the bags are next taken to the laundry room for washing. All bags are then burned in the incinerator. Ashes from the incinerator are returned to the grinding circuit for processing*.

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Date	Data Source	Data
08-25-1943	"Safety Precautions, Ceramics Plant" (Murphy 1945)	• *Bags in which low-grade ore is received are to be sent back to the original supplier* • *Bags in which African ore is delivered are to be burned to recover the elements*
07-04-1944	"Interview with [Name redacted; Privacy Act Information], a Former Linde Employee," April 10, 1981 (Dupree 1981)	• *Washing of burlap bags in which the ore was received began*
07-15-1944	*Step I - L-30 Ore* (Olevitch 1944)	• Empty ore bags contain about ½ lb of ore • Washing burlap bags recovers 70% of material • *After washing, the bags are removed, centrifuged, and stored out in the backyard of the plant. To date about 19,000 bags have been washed and stored.*
10-10-1944	"Interview with [Name redacted; Privacy Act Information], a Former Linde Employee," April 10, 1981 (Dupree 1981)	• *Burning of burlap bags in which the ore was received began*
10-20-1944	"Visit to the Tonawanda Area, 18 and 19 October 1944" (Tybout 1944a)	• *The contractor has decided not to use the bag washer for the bags from L-50 (the low grade ore)* • *The incinerator which will be used for burning the bags was inspected. At the time, it had been in operation only one or two days.*
10-??-1944	Interview with [Name redacted; Privacy Act Information], a former Linde employee, April 10, 1981 (Dupree 1981)	• *The burlap bags had to be disposed of. To control the ore dust that remained in the bags after they were dumped and prevent dust from escaping during their disposal, a series of mechanical devices were installed. First a bag shaker was added to the disposal process to shake out ore dust left behind after dumping. Than a bag washer was installed to wash any material remaining in the bag after it had been shaken. Finally, a bag incinerator was installed to burn the bags after they had been washed. This entire process was completely installed by October, 1944.*
02-22-1945	E. L. Van Horn (USACE) to A. R. Holmes (LAPC) (Van Horn 1945)	• *… It is desired, however, to continue film monitoring on employees engaged in burning L-30 ore bags …*
06-20-1946	"Interview with [Name redacted; Privacy Act Information], a Former Linde Employee," April 10, 1981 (Dupree 1981)	• *Burning of burlap bags in which the ore was received ends*

Just before the January 8 working group meeting, a review of NIOSH (2007b) from SC&A was received (SC&A 2008):

> *Section 5.0 of the NIOSH response discusses the "burlap bag issue." NIOSH concludes that "Based on the reviewed historical records, and considering the fact that the period during which the burlap bags were staged and burned is within the current SEC period, a revision to the current dose reconstruction methodology is not warranted" (NIOSH 2007[c], Section 5.0). This, however, does not adequately respond*

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FOCUSED ASSESSMENT OF DOSE CONSEQUENCES FROM URANIUM ORE BAGS ON THE SITE DURING THE POSTOPERATIONS PERIOD

> *to the site expert interview assertion that thousands of burlap bags were still stacked behind Building 30 after 1950; as stated in the SC&A site profile review: "During the MED period, they stacked all the contaminated burlap bags in storage area of Building 30. These contaminated burlap bags were kept in there until they were removed to be burned in the incinerator in the late 1950s. Many of the people working in Building 30, including operation personnel, secretaries, and maintenance workers, would sit on those bags resting or eating their lunch. This went on for many years."* (SC&A 2006, Attach. 3, p. 112)

During the January 8 working group meeting an attempt was made to resolve the burlap bag issue, considering both NIOSH (2007b) and SC&A (2006). A consensus on the subject could not be reached. The conclusion of the working group was to have a technical conference that focused on the burlap bag issue. SC&A agreed to review the interview notes that were associated with the issue. As a result, a conference call was held on February 13, 2008. The former employee that reported the burlap bag issue was able to participate on this call.

Based on clarification of the burlap bag issue from the former employee who made the original assertion, it was determined that the burlap bags in question were inside Building 30:

> *… there were two pallets of bags containing some material in Building 30 in the time period around August 1951. He and his friends would get drinks, stand by the pile of bags, and place their cups on the bags. He was told by one co-worker that the bags were left over from the Manhattan project. He was asked about the appearance of the bags and he replied that they looked like large canvas sandbags.*

An affidavit that describes the burlap bags was submitted by this former employee along with the Linde SEC Petition Documents that were submitted on March 19, 2008.

Based on this newly clarified description of the burlap bags, NIOSH was tasked with an assessment to compare calculated doses from an assumed pile of bags (two pallets of bags 2 to 3 ft high, for both ore and concentrate) to currently assigned doses.

Comparison of currently assigned dose to potential exposure from ore bags

Currently assigned dose (1950 to 1954)

1.85 R/yr (GM) with GSD of 4.04 from Table 4-20. (note: 95th-percentile value would be 18.5 R/yr).

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Dose assessment for the maximizing case:

Measurements take in 1944 on African ore (Skinner 1944)

Location	Gamma dose rate
Ore sampling room	
15 ore bags in center on ore	0.019 R/hr
1 ft from ore bags	0.0062 R/hr
Ore box car	
Center of car on ore	0.017 R/hr
Ore storage area (pile 4 by 5 by 20 ft)	
Center of 4- by 20-ft face on ore	0.040 R/hr
Center of 4- by 20-ft face on ore	0.026 R/hr
1 ft from 4- by 20-ft face	0.013 R/hr
2 ft from 4- by 20-ft face	0.013 R/hr
3 ft from 4- by 20-ft face	0.010 R/hr
5 ft from 4- by 20-ft face	0.0061 R/hr
10ft from 4- by 20-ft face	0.0028 R/hr

Based on the description of the exposure conditions from the former employee, the following exposure conditions and assumptions can be inferred:

- A dose rate of 0.0062 R/hr based on 1 ft distance from 15 ore bags, and
- An exposure duration of 1 hr/wd, 250 wd/yr.

These result in a total annual exposure of 1.5 R/yr

Summary

Based on the weight of the available evidence (tabulated below), it is unlikely that two pallets of uranium ore (which was last processed at Linde in 1946) would have been in Building 30 in 1951 (5 years after the cessation of processing of uranium ore). The current external exposure model for the period in question incorporates uncertainty in the external dose assignment by application of a lognormal distribution with a GM of 1.85 and a GSD of 4.04. This assumed distribution (with a 95th-percentile value of 18.5 R/yr) accounts for possible deviation of the actual worker exposure of the magnitude that would result from the assumption that two pallets of uranium ore were in Building 30 in 1951.

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Information that contradicts the presence of 2 pallets of uranium ore in Building 30 in 1951	Information that supports the presence of 2 pallets of uranium ore in Building 30 in 1951
1. LAPC documentation outlines the processing of uranium ore including the receipt of uranium ore in bags, handling of the ore bags, and methods that were used to recover material held up in bags after they were dumped (shaking, washing, incineration, and recycling of incinerator ash). 2. LAPC documentation establishes that uranium ore was received in bags and that ore concentrates were received in drums. 3. LAPC documentation establishes end of uranium ore processing as 1946. 4. Affidavit (1981) from a former employee substantiates the description of the handling of the uranium ore bags in the LAPC documentation. 5. Surveys of Building 30 in 1950 and the balance of the site in 1952 did not identify elevated radiation levels consistent with two pallets of uranium ore.	1. Affidavit (2008) from a former Linde employee identifies the presence of 2 pallets of bagged material in Building 30 in 1951. 2. Affidavit (2008) from a former Linde employee identifies the material as uranium ore based on statement from a coworker.

It should be noted that the uranium intake value in this revision of the site profile (15,200 dpm/d) would bound any internal exposure scenario from two pallets of uranium ore bags in the stated configuration (bagged with plastic lining) and conditions (storage).