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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 W. 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. Approval: Document Owner: 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

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ACRONYMS AND ABBREVIATIONS

µg	microgram
ACE	Army Corps of Engineers
AEC	Atomic Energy Commission
BNI	Bechtel National, Inc.
Bq	becquerel
CFR	Code of Federal Regulations
Ci	curie
d	day
DOE	Department of Energy
dpm	disintegrations per minute
FUSRAP	Formerly Utilized Sites Remedial Action Program
GM	geometric mean
GSD	geometric standard deviation
h	hour
hr	hour
hrs	hours
ICRP	International Commission on Radiological Protection
keV	kiloelectron volt
L	liter
LOD	limit of detection
LOOW	Lake Ontario Ordnance Works
MAC	maximum allowable concentration
m	meter
MED	Manhattan Engineer District
mm	millimeter
mr	radiation exposure unit used by MED in the 1940s; interpreted here as mR for photons and mrem for beta radiation
mR	milliroentgen
mrad	millirad
mrem	millirem
mrep	millirep
NIOSH	National Institute for Occupational Safety and Health
NRC	Nuclear Regulatory Commission
NYDO	New York Directed Operations
NYOO	New York Operations Office
ORNL	Oak Ridge National Laboratory

PAEC	potential alpha energy concentration (a measure of concentration of radon daughters)
PAEE	potential alpha energy exposure (a measure of exposure to radon daughters)
pCi	picocuries
PL	preferred level
R	roentgen
R&D	research and development
Sv	sievert
TBD	technical basis document
USBM	U.S. Bureau of Mines
U.S.C.	United States Code
wd	workday
wk	week
WL	working level
WLM	working level month
y	year

1.0 INTRODUCTION

Technical Basis Documents and Site Profile Documents are general working documents that provide guidance concerning 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 the National Institute for Occupational Safety and Health (NIOSH) 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" (AWE facility) or a "Department of Energy facility" as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 [EEOICPA; 42 U.S.C. Sections 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 contamination period.

Employment at an AWE facility is categorized as either (1) during the 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 outside of the period in which weapons-related production occurred). For contract period employment, all radiation exposures must be included in dose reconstructions. For residual contamination period employment, only the radiation exposures defined in 42 U.S.C. § 7384n(c)(4) (*i.e.,* radiation doses received from DOE/AEC-related work) must be included in dose reconstructions. 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.

This document provides an exposure matrix for workers at the following facilities of the Linde Air Products Company: Tonawanda Laboratory and the Ceramics Plant in Tonawanda, New York.

2.0 SITE DESCRIPTION AND OPERATIONAL HISTORY

The information in this Technical Basis Document supports the assumed operational and residual contamination periods listed below.

Facility	Operational and initial cleanup period	Residual contamination period[a]
Tonawanda Laboratory	October 1, 1942 to December 31, 1946	January 1, 1947 to present
Ceramics Plant	October 1, 1942 to December 31, 1954 Stand by: August 1, 1946 to September 14, 1947	January 1, 1955 to present

a. Also called the "post-cleanup period" in this document.

When the United States government and its contractors first became interested in uranium, Linde Air Products, 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 the 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. Linde worked with two types of radioactive material:

- refined uranium materials and preprocessed (partially refined) domestic ores;
- 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 non-uranium 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.

Linde Air Products Company 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 on uranium processing for the MED beginning in late 1942 and probably ending in 1946. In the early years, it operated pilot plants in order to develop procedures used at the Ceramics Plant. The Ceramics Plant produced uranium materials for the MED and the AEC from 1943 to 1946 and from 1947 to 1949. In the 1947 to 1949 period (and perhaps earlier), Linde received UO_2 for processing from Mallinckrodt Chemical Works (AEC 1949a). Linde 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 was described as involving 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).

More detailed histories of Linde facilities, radiation sources and processes related to MED/AEC radiation exposures follow. Key dates are noted in Table 1.

2.1 FACILITIES

Tonawanda Laboratory and the Linde Ceramics Plant were located on land 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 involved in MED/AEC work are shown in Figure 1. 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 Linde 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, Inc. (ACE Buffalo 2003). As of this writing (2005), it is owned by Praxair.

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 may be different from the official Linde or MED designations. Buildings A and B were no longer standing in 1978, when the 1976 radiological survey was issued by ORNL (1978).

This figure is based on Figures 3-1 and B-11 of BNI 1982, Figure 3 of Frame et al 1981 and Linde Database 1945.

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

2.1.2 Building 14 (Tonawanda Laboratory)

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

Table 1. Key dates.

Assumed date or period[a]	Event or activity
Ceramics Plant	
6/1/43 – 7/31/46[b]	Step I production (U_3O_8 from ore or sludge)
6/1/43 – 11/30/43	Preprocessed ores (domestic and scrap)
12/1/43 – 11/12/44	African ores
11/13/44 – 1/31/46	Preprocessed domestic ores
2/1/46 – 2/28/46	African ores (48%) and preprocessed domestic ores (52%)
3/1/46 – 6/30/46	African ores
7/1/46 – 7/31/46	African ores and preprocessed ash
4/27/43 – 3/8/44[c]	Step II production (UO_2 from U_3O_8)
7/25/43 – 6/26/46[d]	Step III production (UF_4 from UO_2)
8/1/46 – 9/14/47	Standby
9/15/47 – 10/31/47[e]	Step III rehabilitation
11/1/47 – 6/30/49[f]	Step III production
10/1/44 – 2/28/46[g]	Production of nickel material for K-25 diffusion barrier
7/1/49 – 3/31/50[h]	Cleanup of Building 30
7/1/49 – 12/31/54[i]	Cleanup of Linde Ceramics Buildings
Prior to 1978	Demolition of Buildings A and B (office buildings on Figure 1)
After 1954	Post-cleanup period
8/31/81[j]	Demolition of Building 37 began
8/31/96[j]	Demolition of Building 38 began
9/5/98[k]	Demolition of Building 30 began
9/30/00[j]	Soil remediation began (scheduled for completion in 2007)[j]
Tonawanda Laboratory	
10/1/42 – 7/31/46[l]	MED-related research and development operations
8/1/46 – 12/31/46[m]	Cleanup period
After 1947	Post-cleanup period
4/30/04[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 indicated 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 Aerospace Corporation 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 text Section 4.1.3.2).
c. Linde Air Products Company Ceramics Plant 1946b, p. 40.
d. Linde Air Products Company Ceramics Plant 1946c, p. 42.
e. Start date based on Kent 1947, p. 1.
f. Start date per Rennich 1947. End date per AEC 1951, p. 30.
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 per Eisenbud 1950.
i. Start date assumed to be day after end of Step III production; end date estimated per Harris 1954.
j. Pilon 2004.
k. ACE Buffalo 1998.
l. Start date per Jenness and Ewing 1943. End date assumed to be the end date of 1943-1946 production at the Ceramics Plant.
m. Start date based on estimated end date of research and development period; end date estimated.

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 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.

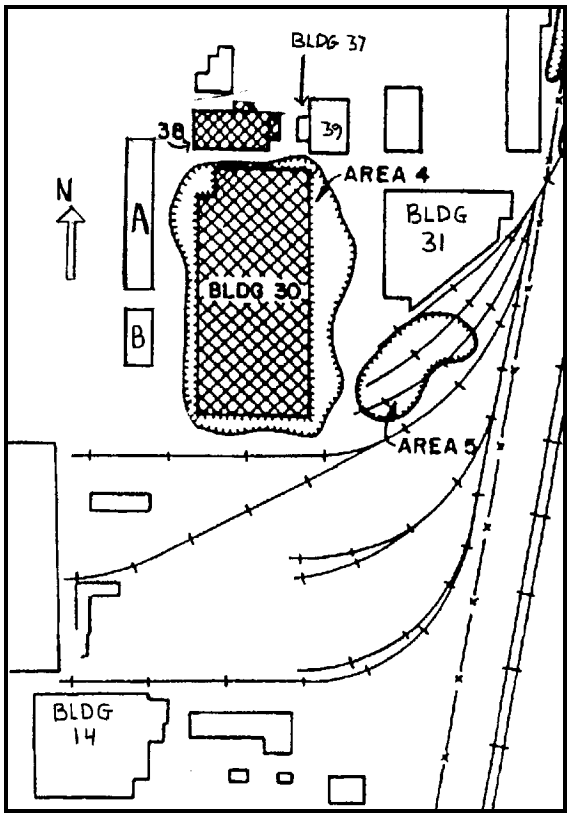

Figure 1. Linde Ceramics and Tonawanda Laboratory buildings on Linde Tonawanda site.

A 1942 Tonawanda Laboratory directory listed approximately 200 employees (Tonawanda Laboratory 1942). The locations of most were designated by room numbers. Some employees were said to be located in the "Eng. Lab" and others in the "Proving Lab." A 1944 employee list reported some to be located in the "Pilot Plant" (Tonawanda Laboratory 1944). 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 and 3 show the layout. It had two floors. Large pieces of equipment such as digestion tanks were located on the ground floor. Process workers operated equipment from the second floor (Dupree 1983a,b). The 1949 to 1950 decontamination work done by Linde 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 September 5-19, 1998 (ACE Buffalo 1998).

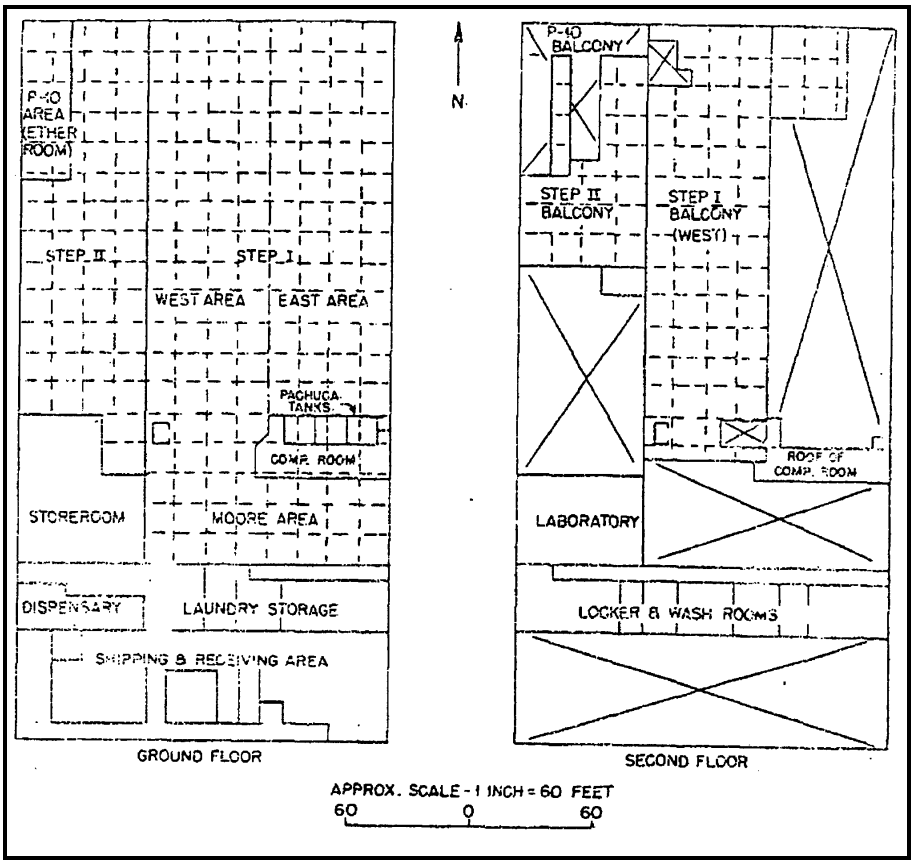

Figure 2. Building 30 layout (from Heatherton 1950).

2.1.4 Building 31 (Ceramics Plant)

Building 31 is described as "used in uranium separation process during MED operations" (BNI 1993, p. 1-65). 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 (Linde Database 1945). Building 31 might have been the final location of Linde's Tonawanda nickel processing operations.

As of 1976, the building was described as housing fabrication facilities, offices and storage areas and was occupied by about 12 to 15 employees. Radioactive contamination was found during a 1976

survey (ORNL 1978). Building 31 was decontaminated in 1997 (ACE Buffalo 2002). As of 2004, it was still standing and in use (Pilon 2004).

2.1.5 Building 37 (Ceramics Plant)

Building 37 was a small appendage to Building 39, which was located east of Building 38 (Figure 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 ft x 36 ft (BNI 1982 Figure B-11), indicates at most a minor role. The 1945 plant drawing labeled it "Test Bldg" (Linde Database 1945). Radioactive contamination was found in 1976 (ORNL 1978). Linde demolished Building 37 in 1981 (Pilon 2004).

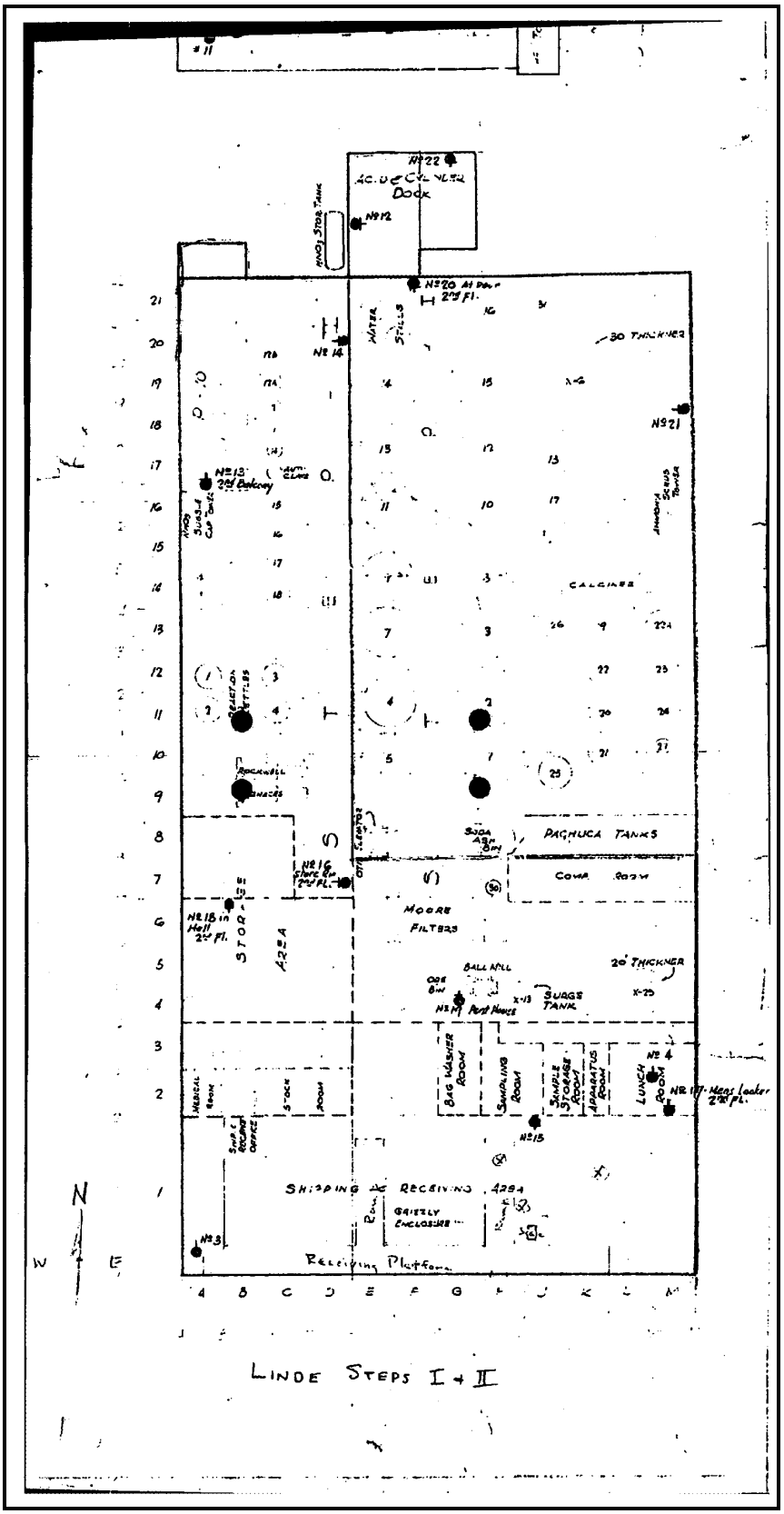

Figure 3. Building 30 layout (from Linde Database Undated A).

2.1.6 Building 38 (Ceramics Plant)

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

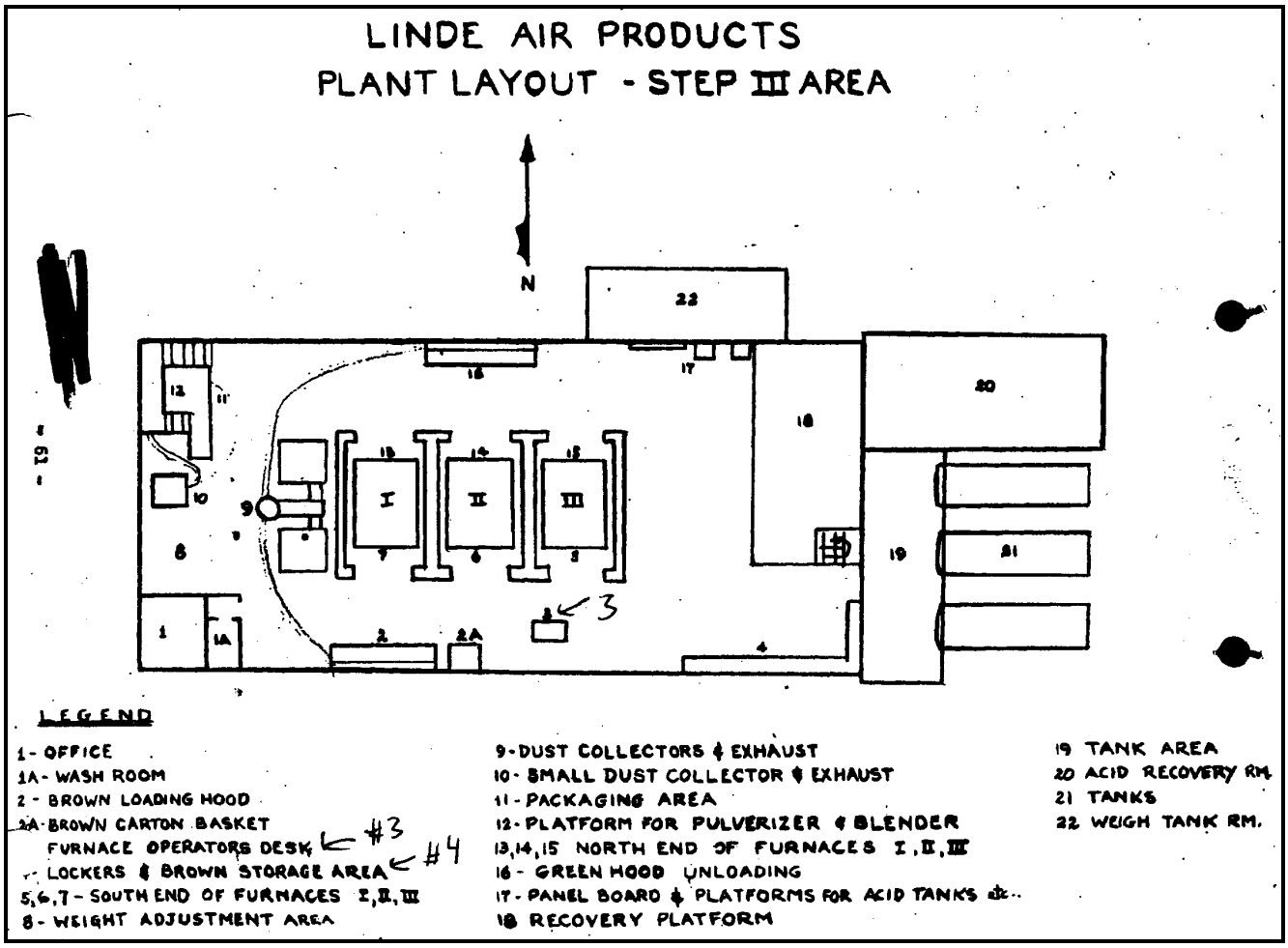

Figure 4. Building 38 layout (from Klevin 1949a).

2.2 SOURCE TERM — 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 from Africa; some was torbernite (Aerospace Corporation 1981, p. 1). Neither African ore underwent chemical preprocessing before being shipped to Linde. Thus, they contained not only uranium, but 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 (Aerospace Corporation 1981, p. 5). The preprocessing may have also separated uranium from other radionuclides in the ore. If the separation of uranium were complete, then the radioactivity in the preprocessed ore delivered to Linde would have consisted primarily of the uranium radionuclides U-234, U-235, and U-238 (in their natural abundance) 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 used at Linde is not known. A review of the chemical processing concluded only "the radium in the domestic ore was significantly less than that in the African ores" (Aerospace Corporation 1981, p. B-1).

The distinction between the two types of ore is important. The primary radiological hazards from an ore containing only uranium and its short-lived progeny would be due to alpha and beta emissions. However, radium and other progeny in the African ores would produce, in addition, 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 (35% or less by mass) might have been used on a limited basis for K-25 barrier testing and research and development 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

Linde Air Products Company 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 tons 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 Linde Air Products entered into contract W-7401 eng-14 with the MED to design and construct facilities at the Ceramics Plant for processing uranium-containing ores and to produce U_3O_8, UO_2 and UF_4 (Cornell 1942; Marshall 1943). The contract also called for research and development regarding the 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's work covered the period October 1942 to November 1943 (Jenness and Ewing 1943). Most of the work was devoted to study of the problems encountered 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 tons 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 (Tonawanda Laboratory 1946b). The final report discussed work that was done 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 since 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 was related to conversion of UF_6 process gas to UO_3. Research reports (Brimm and Schubert 1945a,b) and a final report (Tonawanda Laboratory 1946a) indicated that the associated pilot plant operations used pounds rather than tons of material.

A later Linde research report on uranium processing dated 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 containing 3-20% U_3O_8 by weight. Approximately 70% of the ore was African and 30% domestic by weight. About 26,000 metric tons of ore were processed, and about 2,300 metric tons of U_3O_8 were produced (Aerospace Corporation 1981, Table B-1).

Step I production 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 produced 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 pounds of uranium ore concentrates from Colorado and 8,900 pounds of residue, which were the "property of Contractor as a result of operations prior to his present CPFF [cost plus fixed fee] work for the Manhattan District (Contract No. W-7401 eng-14)." The transfer was made under contract W-17-028 eng-29 dated May 4, 1945 (U.S. Engineer Office 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 2-3 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 out 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, leaving 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. Also, some of the other chemical additives used at various points were different (Linde Air Products Company Ceramics Plant 1946a; Aerospace 1981, p. A-1).

Step I Operations

Ore was delivered by rail in boxcars. African L-30 pitchblende arrived packaged in burlap bags that were inside paper bags. Each ore bag weighed 50-80 lbs. For African ores, the car doors were left open for 12-24 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; Linde Air Products Company Ceramics Plant 1946a; Cranch 1944c).

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 tons 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 half pound 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 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 out 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 containing 24 rectangular leaves with each leaf approximately 35 sq. ft. 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-inch 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 A, third page; Olevitch 1944; Linde Air Products Company Ceramics Plant 1946a; 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 ether and water 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 (Linde Air Products Company Ceramics Plant 1946b; 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" (Linde Air Products Company Ceramics Plant 1946b).

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 1000°F. Gaseous anhydrous HF gas was flowed over the UO_2, which was placed on magnesium trays stacked on spacer bars inside 9" square tubes 10 feet long that were inside a furnace (Linde Air Products Company Ceramics Plant 1946c; 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 re-treatment. Good material was pulverized, blended, sampled, packaged, weighed, and shipped (Linde Air Products Company Ceramics Plant 1946c; Hickey, Crawford-Brown, and Tankersley 1988, p. 8).

2.3.3 Nickel Production

After the shutdown of Linde Ceramics Step II in 1944, the Step II equipment was converted for use in production of nickel powder, a non-radioactive 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 (U.S. Government 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 tons 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 used in connection with Contract W-7401 eng-14, the contract for the uranium refining work. A drawing 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. Thus, nickel processing may have occurred in Building 31 (see Figure 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 may be that nickel processing initially occurred in the Step II area and later was moved to Building 31.

Evidence supporting Building 31 as at least the ultimate location for Linde's Tonawanda nickel processing is found a 1945 plant drawing (Linde Database 1945). The visible portions of partially obscured label on the drawing are consistent with the interpretation that it says "Nickel Plant." Also, 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 Linde Air Products Company 1945, 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 into 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 1948h; 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 box cars to Building 30 (see the section on 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/h 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 1948h).

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 were used to unload drums of K-65 shipped by rail to the Lake Ontario Ordnance Works at Modeltown, New York (Wolf 1949; Heatherton 1949a). Linde film badges were worn during this work.

At the April 18, 2005 Worker Outreach Meeting, strontium was mentioned. Summary information from the meeting did not indicate if this was radioactive on nonradioactive strontium, or if this was process material or a sealed source. Because no mention of radioactive strontium was found in the available Linde documents, it is believed likely that this source might have been a sealed radioactive source, nonradioactive material or a small quantity in comparison to the uranium source term. At the June 27, 2005 Worker Outreach Meeting, it was clarified that the strontium was in the form of a strontium-90 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.

2.4 PROTECTIVE MEASURES

Steps were taken to protect against radiological and chemical hazards during processing (Linde Air Products Company Ceramics Plant 1946a,b,c; Linde 1945; 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 regarding 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 wearing of 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 working with hydrofluoric acid.

According to interviewees (Dupree 1983b), use of respirators was often at the worker's discretion, but they were "usually used." 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 1944f). 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 1945d). 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 WORKHOURS

For Tonawanda Laboratory, two listings of employees may aid in determining whether a claimant worked there. A September 1942 Laboratory directory (Tonawanda Laboratory 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 (Tonawanda Laboratory 1944) 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 worker jobs. A December 1945 job description specifies the duties of 51 categories of workers and lists a department code for each position (Linde Air Products Company 1945). Since 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 may help identify the type of activity in which a worker was involved. Table 2 shows Ceramics Plant department codes associated with various worker activities as determined in a 1988 study of worker hazards. 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. 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.

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

Table 3. Worker information.

Item	Reference
Tonawanda Laboratory employees and room numbers dated September 2, 1942. Job titles are not stated.	Tonawanda Laboratory 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 1948f
Ceramics Plant Step III process area personnel (including operators, millwrights, and stores attendants) included in report published 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 included in report published August 26, 1949	Klevin 1949b, p. 11

The term “day” in this document refers to a calendar day. The term “workday” is used to describe a day at work. Default assumptions are 250 workdays per year and 8 hours per workday: this results in 2000 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 integrating over calendar days.

The Ceramics Plant work schedule during the production period (1943-1946) is described as involving eight-hour shifts, six days per week (Dupree 1983a,b,c). A nine-hour workday with a lunch period included is assumed because there is evidence that many employees worked overtime (Dupree 1983b, p. 4; MED Undated A, p.1). An eight-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 six-day workweek to a five-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).

Based on the above, the periods and work schedule at the Ceramics Plant from the onset of MED operations through 1954 are assumed to have been as shown in Table 4.

Table 4 Ceramics Plant assumed work schedule (including lunch and breaks), 1942-1954.

Period	Start	End	h/wd	wd/wk	wk/y
Pre-production	10/1/42	4/26/43	9.0	6	50
Production	4/27/43	7/31/46	9.0	6	50
Standby	8/1/46	9/14/47	8.5	6	50
Rehabilitation & Production	9/15/47	6/30/49	8.5	6	50
Cleanup	7/1/49	12/31/50	8.5	6	50
	1/1/51[a]	12/31/54	8.5	5	50

a. Assumed date of transition from six-day to five-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 1948c,d,e). 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 1949d, e).

Shortly after the shutdown, a comprehensive cleanup effort was undertaken to reduce levels of radioactivity in Building 30 in order to enable its release to Linde for unrestricted use (Heatherton 1950). After removal of the bulk of the process equipment from Building 30, 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, performing surface abrasion activities, were provided with a U.S. Bureau of Mines (USBM) approved rubber hood 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-type respirators or, for short periods, USBM respirators approved for all dusts.

An AEC internal memorandum dated 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 dated April 1954 (Harris 1954) reported levels obtained after decontamination. Average readings on categories of equipment to be left at the plant ranged from 0.05 to 0.6 mrep/h 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 "mreps/hr/ft^2" beta plus gamma. This was considered excessive. The memorandum cited data indicating that this could be reduced to 0.065 (mreps/hr)/ft^2 by covering contaminated floor areas with asphalt tile. It recommended release of the building to Linde once this was done. The date at which Linde met this condition was not found, but since the requirement was modest, it is likely that the release to Linde occurred in 1954.

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

This document assumes the end date of the Ceramics Plant cleanup period to be the date of turnover of the four Ceramics Plant production buildings to Linde for its use. This date is sometimes stated as 1953 (see, for example, ACE Buffalo 2004a, Response to Question 4). However, Harris (1954) indicates that the decontamination of Building 38 was not complete as of April 1954. For dose reconstruction, it is assumed that turnover did not occur until December 31, 1954.

2.7 POST MED/AEC OPERATIONS

The Formerly Utilized Sites Remedial Action Program (FUSRAP) began in 1976. Oak Ridge National Laboratory (ORNL 1978) surveyed the Linde Tonawanda from October 18 through November 5, 1976 to determine if remediation would be required. 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 associated with Building 57, and an additional review of the BNI1993 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, 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 buildings involved in MED/AEC work except Building 31. Table 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, 1942-1954

This section develops parameters for reconstruction of doses due to internal exposures from October 1, 1942, the assumed start date of MED work at Linde, until December 31, 1954, the assumed date of initial cleanup completion and building turnover from MED/AEC to Linde.

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 estimated in this section.

This document assumes that Tonawanda Laboratory's primary internal exposure occurred from October 1, 1942 to December 31, 1946. At 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, 1954 (operations and cleanup). From October 1, 1942 and 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 remaining 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's progeny and to radon would have decreased at the Ceramics Plant after August 31, 1946, because ore was no longer processed during this later period. Continued lower level exposures to uranium progeny and to radon are assumed, because some radioactive

waste was disposed on 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 1947 to 1954 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 Manhattan Engineer District (MED), the permissible level for natural uranium air dust concentration was set at 500 $\mu g/m^3$ for insoluble uranium compounds and 150 $\mu g/m^3$ for soluble uranium compounds. After the war, the University of Rochester (Rochester, NY) recommended lowering the permissible level to 50 $\mu g/m^3$ for all uranium compounds on the basis of chemical toxicity. This level was also stated as 70 disintegrations per minute per cubic meter (70 dpm/m^3) for natural uranium and was based primarily on animal studies. The Medical Division of the AEC New York Operations Office (NYOO) thought that the "maximum permissible level" should be based on human data and was thus unknown. Therefore, the level of 50 $\mu g/m^3$ was generally referred to as the "preferred level" (AEC 1949a). Oftentimes the contractors of the AEC used the term "Maximum Allowable (air) Concentration" (MAC) interchangeably with the "preferred level" (PL) and reported air-sampling results as multiples of the MAC (NLO 1952, AEC 1953). When considering air concentrations 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, Linde Air Products, 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 weighted by exposure times and summed these time-weighted air concentrations to determine daily time-weighted average air concentrations by job categories. Up until the time of the 1949 AEC report, 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 6250 $\mu g/m^3$), 9% were exposed at 25-125 MAC (1250-6250 $\mu g/m^3$), and 82% were exposed to less than 25 MAC (less than 1250 $\mu g/m^3$). 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 re-design 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 1958).

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 prior to their 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 prior to 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 that occurred 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.

As of this writing, the pre-1947 operational period intakes are reserved. Therefore, the pre-1947 information is provided only as a description of what the likely upper bound exposures might have been, and is not currently planned for use in Linde dose reconstruction.

For the pre-1947 period, the MAC would have been assumed to be based on inclusion of uranium's alpha emitting progeny. Although short-term exposures might have exceeded 300 MAC, it is very unlikely that long-term exposures would have. A review of the predicted urinalyses, kidney burdens and lung burdens, indicate that it is highly unlikely that an individual would have sustained exposures like these for any length of time. Evidence of sustained exposure to the more soluble uranium compounds might have shown up in the medical urinalyses, as increases in proteins and glucose in the urine (note that other conditions can also account for these increases). The assumption of air concentrations at 300 MAC seems adequate to provide a quick estimate of exposure, and although the type F uranium bioassay results are high, they do not seem inconceivable for some workers during this early period. However, it is also likely that Linde workers were exposed to a mixture of uranium absorption types. The analysis of radium exposures in Section 3.8 is partially based on the assumption of alpha activity air concentrations of 300 MAC during Linde's ore processing period.

After the ore processing, Linde began a standby period. It was initially and arbitrarily assumed that exposures decreased to 1 MAC during the standby period at the Ceramics Plant, and that exposures decreased to 0.1 MAC at the Tonawanda Laboratory after cleanup in 1946 until the end of cleanup at the Ceramics Plant in 1954. Based on reviews of later air concentrations at Linde, and reviews of air concentration data from other sites, it is believed that 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. Intakes from the standby and rehabilitation periods are reserved. Beginning November 1, 1947 at Linde Ceramics, workers were assumed to be exposed to 33 MAC and it was assumed this exposure continued through cleanup in 1954. Uranium progeny are not included in this later period, because only refined uranium was used and because the dose from intakes of contamination left from earlier work would have been insignificant compared to the dose to uranium during operations.

To simplify calculations, it assumed that the workweek was 40 hours long during all years, although it is likely that the workweek for many was in excess of 40 hours especially during the earlier years. The assumed air concentrations are sufficiently large to account for any differences in actual hours exposed.

Dose reconstructions should assume International Commission on Radiological Protection (ICRP) Publication 66 default parameters for particle deposition (ICRP 1994).

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 prior to standby. In addition, workers may have worked in multiple uranium process areas. This analysis assumed that types F, M and S were available prior to 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 (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 1944a), but no records are available indicating the analyses were done. A page (no author, circa 1947) titled, "Schedule of Examinations, Contract AT-30-1-GEN-165," which commenced sometime in 1947 (Rennich 1947), indicates that 60 cc of urine from each employee was to be sent monthly to 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 (Linde Ceramics Plant Urinalysis Data 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 prior to 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, with many of the higher results reported in the November to December 1947 "pre-employment urines." Because cleanup/preparation activities were occurring prior to collection of these samples, it is possible the elevated urinalysis results represented exposures received either during rehabilitation of the Step III equipment or perhaps from the less soluble component of uranium 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 individuals' results are consistent with an intake 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 two 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 1948g). The uranium fusion photofluorimetry urinalyses performed by the University of Rochester and the AEC NYOO were similar to those performed at other AEC facilities. The default detection threshold for uranium urinalysis is 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 transmitted by the AEC to Linde (Wolf 1948) included a note that indicates the MDA might have been much higher.

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 November1948 uranium urinalysis results include a similar note, but 0.01 mg U/L was reported as the limit of accurate quantitative determination or reliable determination for photofluorometric analysis. Based on this information, the assumed detection threshold through January 31, 1948 is 0.1 mg/L for Linde uranium photofluorimetry urinalyses. Beginning February 1, 1948 the assumed uranium urinalysis detection threshold is 0.01 mg/L. Note that it is possible that the January 1948 determination level of 0.1 mg/L is 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 numbers reported.

The AEC (Wolf 1948) 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 three months. Linde's reply (Heatherton 1948g) requested clarification, because it was their previous understanding that resampling was to occur monthly.

3.2.2 Uranium Air Concentrations

Information regarding particulate air concentration measurements at Linde is provided in this section. This information might be useful in future analyses, however, this site profile currently uses an estimate of Linde's largest time weighted air concentration reported by AEC (1949a) to estimate intakes during uranium operations. Other Linde particulate air concentration data were reviewed and considered, but are not discussed here.

1943-1944
Air concentrations were measured at Linde Ceramics at selected locations during 1943-1946. These early results were reported in units of $\mu g/m^3$ (Linde Ceramics Plant Airborne Dust Uranium Data 1943-1946). Laboratory data sheets are available for measurements made in 1943. These indicate that each sample was collected in a tube. The mass of "X material" (uranium) in the tube was determined by multiplying the mass of uranium in a standard by the ratio of instrument response measured for the sample to that measured for the standard, and dividing by the volume of air sampled. The instrument response was in units of volts/sec, but no description of the instrument was found. In a discussion of 1945 results that also involved determination of the radioactivity of a dust sample in a tube, the methodology was characterized as involving "radioactive methods" and "radioactivity measurements" (Bryan 1945; Tybout 1945c), although another document referring to sampling of the nickel operations seemed to imply that perhaps not all samples included radioactivity analyses (Rehm 1944b).

Although there is the suggestion that radiation from air samples was measured, it is possible that some results reflect the mass of all materials suspended in the sampled air rather than the radioactivity in air. If this were the case, the reported air concentrations would be an overestimate of the uranium mass-based air concentration. In addition, if the uranium standard contained only refined uranium, then the reported results might overestimate uranium concentrations from African ore, because the progeny would also be counted as uranium. It is likely that the early measurements did account for radionuclides (or particulate) other than uranium, so although the early measurements are reported in terms of uranium mass, it is assumed that they are actually radiation-based values, which can be apportioned to different uranium progeny.

1947-1949
Airborne dust radioactivity concentrations were measured in general areas and breathing zones associated with various locations and tasks in 1947-1949. Air particulate samples were collected by pumping air through filter paper disks, and an alpha counter was used to measure the radioactivity on a filter (Klevin 1949a). Results were reported in units of α dpm/m^3 or of "preferred level" (noted then as 70 α dpm/m^3). The AEC combined the results with data on the amounts of time workers spent at

various locations and tasks in order to determine daily time-weighted average radioactivity concentrations in the air breathed by workers in various jobs (Klevin 1949a, b).

3.3 RADIUM

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. Assignment of radium exposures when radon breath analyses are not available or cannot be interpreted is addressed below in Section 3.4.

3.3.1 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 collected from Linde workers between June 12, 1944 and January 17, 1945 (Linde Ceramics Plant Breath Radon Data 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 (Tybout 1944b), room background concentrations were reported respectively as 0.6 and 0.3 pCi/L (a picocurie equals a micro-microcurie). 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 Ra-226 level was not stated, but in many cases, the reported burden of radium in µg was numerically equal to 10% of the breath radon concentration in pCi. Radon breath results are the starting point for dose reconstruction, so it might be necessary to back-calculate breath results from either Ra-226 burdens or tolerance levels. The original records might have sufficient information for this determination. This information has not yet been tabulated for inclusion in the site profile.

3.4 URANIUM PROGENY

Ceramics Plant 1943-1946 Production, and Tonawanda Laboratories
In the absence of data on exposures to uranium progeny, their intake rates are determined by assuming secular equilibrium. Table 5 lists equilibrium-based ratios for uranium progeny of particular interest in dose reconstruction. Absorption types for their likely chemical forms are also shown. The intake ratios provide reasonably realistic estimates of intakes of progeny due to dust from African ore. The uranium activity fractions overestimate relative intakes of most progeny when the dust is from preprocessed domestic ore. They may underestimate intakes of progeny when the dust is from filter cakes or waste products that contain uranium progeny but very little uranium. The ratios in Table 5 are for use for the entire 1943-1946 production period for all workers even though only about 70% of the ore processed was African ore (see Section 2.3.2) and many workers handled only refined uranium materials. This along with the claimant-favorable assumptions made in the estimation of worker dust exposures is judged to provide sufficient overestimation to balance any underestimation associated with the handling of waste products.

Note that the uranium fractions are applied when the activity of uranium is known. The activity fractions for gross alpha are applied to data measured as alpha activity.

Table 5. Intake ratios and absorption types for uranium progeny.

Nuclide	Uranium activity fractions	Gross alpha activity fractions	Absorption type
U-natural	1	4.02E-01	F, M, S
Th-230	4.89E-01	1.96E-01	M, S
Ra-226	4.89E-01	1.96E-01	M

Po-210	4.89E-01	1.96E-01	F,M
Pa-231	2.28E-02	9.16E-03	M, S
Ac-227	2.28E-02	9.16E-03	F, M, S

Ceramics Plant 1947-1949 Step III Production, and Subsequent Initial Cleanup
During this period, refined uranium materials were handled. None of the progeny listed in Table 5 would have been present in significant quantities compared to the uranium at the Ceramics Plant.

3.5 RADON

More than 200 measurements of radon concentrations were made during 1942-1946 pilot plant and production processing of African ore at Linde (Linde Ceramics Plant Radon Data 1944-1946). The early measurements were made using liter glass bulbs that were evacuated immediately before sampling to below 50 microns (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 Rn-222), the concentration *C* in pCi/L was converted to potential alpha energy concentration (*PAEC*) in units of working level (WL) using:

$$PAEC = C * F / (100 \text{ pCi/L per WL})$$

where the equilibrium factor *F* was taken as 0.4, a value recommended by the 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 month (WLM) was obtained from *PAEC* and the number of months of exposure *M* using:

$$PAEE = PAEC * M$$

where a month is assumed to be 170 work-hours.

3.5.1 Ceramics Plant Radon Exposures

During Ceramics Plant pre-production 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 resulting from Tonawanda Laboratory work. No direct measurement of this was available. An estimate was made based on the lowest indoor concentrations measured at the Ceramics Plant during African ore processing. These were viewed as indicating the upper limit to the outdoor concentration since outdoor air is 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. Therefore, 10 pCi/L was taken as the estimated outdoor concentration.

Radon concentrations measured during processing of African ore at the Ceramics Plant are summarized in Table 6. Based on geometric means, locations were grouped into three categories — high, medium, and low, and the geometric mean and 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).

To simplify, this analysis assumes that workers, who were likely to spend the majority of their time in process areas, or in boxcars (where some of the highest radon levels were measured, about 200 times tolerance), or whose jobs were unknown were exposed to 99.3 pCi/L of radon for 2040 hours

Table 6. 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 geometric mean, a <LOD value was assumed to equal the LOD.
b. GSD could not be determined.

(12 work-months) per year prior to standby. Workers who did not work or have their offices in the process buildings are assumed to have been exposed to 22.4 pCi/L of radon prior to standby. Because a job in current times might not be in or near a process area, does not mean the same held true 60 years ago. Nurses, some stenographers, launderers and seamstresses and some clerical workers had jobs or locations that put them in contact with the uranium and progeny (Homes 1944b).

The initial period of African ore processing was followed by a second period of domestic ore processing. Thirteen measurements of radon concentration during the domestic ore processing were available. The geometric mean of the measurements, assuming the <LOD values were equal to the LOD, was 9.1 pCi/L. To estimate exposure during this domestic ore processing period, both indoor and outdoor radon concentration were assumed to be 10 pCi/L.

After the end of African ore processing, concentration in the main ore processing building, Building 30, was assumed to remain at the level measured during the second period of domestic ore processing, 10 pCi/L, until the end of its cleanup. Concentrations in other Ceramics Plant buildings were also assumed to be 10 pCi/L until the end of their cleanups. Since the locations of many workers are likely to be unknown, it was assumed that all workers were exposed to 10 pCi/L of radon until the assumed end of the Ceramics Plant building cleanup in 1954.

Table 7. Ceramics Plant worker radon exposures rates, 1942-1954.

Period/work location	Time-weighted concentration (pCi/L)	Exposure rate (WLM/y)
10/1/1942 to 10/31/1947		
In process and research areas	Reserved	Reserved

Not in process and research areas	Reserved	Reserved
11/1/1947 to 12/13/1954		
All workers	10.0	0.480

3.5.2 Tonawanda Laboratory Radon Exposures

Few measurements were available, 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 been ongoing 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 preceding the beginnings of the two African ore processing periods at the Ceramics Plant. For dose reconstruction, it is assumed that African ore processing at Tonawanda Laboratory occurred during its whole MED period, but that the peak concentrations of radon were equal to the medium values at the Ceramics Plant. It is 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 its cleanup, radon exposure in the Laboratory was based on the highest geometric mean PAEC determined for a Tonawanda site building from measurements made in 1981, 1.68E-02 WL for Building 31 (based on analysis of data in BNI 1982, Table B-3).

Table 8 summarizes the assumed radon concentrations and resulting exposures.

Table 8. Tonawanda Laboratory radon exposure rates, 1942-1954

Period		Time-weighted concentration pCi/L	Exposure rate (WLM/y)
Start	End		
R&D and cleanup			
10/01/42	10/31/47	Reserved	Reserved
Post-cleanup			
11/01/47	12/31/54	—	0.202

Radon and Radon Daughters
Radon exposure rates provided here are annual PAEE rates in WLM/y. 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 6).

3.6 INHALATION INTAKE ESTIMATES OF RADIOACTIVE PARTICULATES

Airborne concentrations were assumed in Section 3.1, for the purpose of estimating intakes for Linde workers. This analysis assumes that a worker was chronically exposed to alpha airborne concentrations as stated in Section 3.1 and summarized in Table 9.

Table 9. Assumed airborne concentrations used to estimate intakes.

Start	End	Activity description	# MAC	alpha dpm/m^3	Source
Ceramics plant					
10/1/1942	7/31/1946	Uranium ore processing	Reserved	Reserved	Uranium and progeny
8/1/1946	9/14/1947	Standby	Reserved	Reserved	Uranium and progeny
9/15/1947	10/31/1947	Rehabilitation	Reserved	Reserved	Reserved
11/1/1947	12/31/1954	Step III processing	33	2,310	Uranium
Tonawanda plant					
10/1/1942	10/31/1947	Uranium ore research	Reserved	Reserved	Uranium and progeny
11/1/1947	12/31/1954	Post	0.1	7	Uranium and progeny

This analysis assumes Linde workers were exposed to the given air concentrations for eight hours per day, five days per week, 50 weeks per year. Time-weighted exposure studies of even the larger plants like Mallinckrodt showed that the majority of workers were exposed at lower levels than those 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 9 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 is estimated by multiplying the air concentration of 7 dpm/m^3 by the alpha fraction of uranium, 0.402; the ICRP 66 (ICRP 1994) recommended breathing rate of 1.2 m^3/h; and the assumed 2000 work-hours per calendar year. This results in an annual chronic inhalation intake of 6.75E+03 dpm, which is equal to a daily intake rate of 18.5 dpm/day. For the assumed exposure at 33 MAC, no alpha activity is apportioned to progeny so the daily uranium intake would be 1.52E+04 dpm/day.

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

3.7 INGESTION INTAKE ESTIMATES

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/workday. To adjust this to ingestion intake per calendar day, 0.563 dpm/workday is multiplied by 250 workdays per year and divided by 365 days per year, which equals 0.385 dpm/day. For the assumed exposure at 33 MAC, no alpha activity is apportioned to progeny so the daily uranium intake would be 316 dpm/day. In accordance with NIOSH 2004, the f_1-value used for inhalation dose calculations is to be used for ingestion dose calculations.

3.8 CONSIDERATION OF BIOASSAY DATA

Predicted uranium urinalysis results, provided in Table 10, were calculated for the last day of assumed chronic intake periods of 30 and 60 days, 0.5 years, 1 year and extended annually thereafter through the end of operations, assuming the estimated inhalation and ingestion intakes of natural uranium were based on a uranium air concentration of 33 MAC. A cursory review of the highest uranium urinalysis data from facilities that handled uranium in large quantities (Mallinckrodt, Harshaw, Hanford, ORNL, K-25, Paducah, and Portsmouth) indicates that results exceeding 10 mg/L are rare and that most results are less than 1 mg/L. At the Ceramics Plant, where the first Linde uranium bioassays were performed after standby, two of the available urinalysis results exceeded 1 mg/L. Subsequent results from these individuals were much lower. From November 1947 through January 1950, most Linde uranium urinalyses (about 95%) were less than 0.1 mg/L, but it is notable

that exposures would likely have been lower during this period than in the earlier days of operations. The predicted results in Table 10 do not seem inconsistent with the limited Linde urinalyses.

Table 10. Predicted uranium urinalyses from Ceramics Plant assumed inhalation and ingestion chronic uranium intake from November 1, 1947 to December 31, 1954 based on 33 MAC in air.

Bioassay date	Type M		Type S	
	dpm/d	mg/L	dpm/d	mg/L
12/1/1948	566	0.3	18	0.01
12/31/1948	661	0.3	20	0.01
5/2/1949	853	0.4	28	0.01
11/1/1948	961	0.5	36	0.02
11/1/1949	1,013	0.5	48	0.02
11/1/1950	1,022	0.5	57	0.03
11/1/1951	1,026	0.5	64	0.03
11/1/1952	1,028	0.5	70	0.03
11/1/1953	1,031	0.5	74	0.03
11/1/1954	1,033	0.5	77	0.04
12/31/1954	1,033	0.5	78	0.04

*Mass results assume natural uranium exposure

Given a chronic exposure to uranium and its alpha emitting progeny at 300 MAC, the activity fraction of Ra-226 would be 0.196, which means that the chronic inhalation rate would be 2.7E+04 dpm/d. This gives a whole body activity of 2.6E+05 dpm at one year, and about 4.0E+05 dpm at 4 years (calculated using IMBA Expert (OCAS), Version 3.2.20). The Ra-226 body activity was estimated using the largest breath radon result found for Linde, 2.2 pCi/L, by multiplying the radon result by a conversion factor of 2.52E+05 pCi/(pCi/L) (ORAUT 2005). This gives a body activity of 5.5 E+05 pCi, which is equal to 1.2 E+06 dpm, and is within a factor of 3 of the estimated intake from a 4-year chronic exposure to 300 MAC. Because other Linde radon breath analyses are lower and because a chronic exposure scenario may not best represent a worker's exposure pattern, the assumption of 300 MAC chronic exposure was believed to be adequate for reconstructing doses in the pre-1947 research and production period, but at this time this period is reserved.

3.9 OCCUPATIONAL INTERNAL DOSE RECONSTRUCTION ASSUMPTIONS AND SUMMARY 1942-1954

Intake estimates and radon exposures estimated for Linde workers are shown in the Tables 11 and 12. Uranium bioassays are available for some workers during the period November 1947 through January 1950. If uranium bioassays are used to reconstruct an individual's dose, additional intakes from uranium progeny might need to be added for Tonawanda Laboratories and for intakes assumed to occur before November 1947 at the Ceramics Plant. Table 5 lists the activity ratio of the progeny to uranium.

For workers, whose work location is considered indeterminate, intakes for the Ceramics Plant are assumed.

For dose calculations, uranium intakes can be assumed to be U-234. Absorption types are listed in the table.

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.

Table 11. Internal exposure summary for the Ceramics Plant, October 1, 1942, through December 31, 1954

Radionuclide	Start	End	Intake route	Absorption type	Intake or exposure	Units
U-234	10/1/1942	7/31/1946	Inhalation	F, M, S	Reserved	dpm/day
	10/1/1942	7/31/1946	Ingestion	(a)	Reserved	dpm/day
	8/1/1946	9/14/1947	Inhalation	F, M, S	Reserved	dpm/day
	8/1/1946	9/14/1947	Ingestion	(a)	Reserved	dpm/day
	11/1/1947	12/31/1954	Inhalation	M, S	1.52E+04	dpm/day
	11/1/1947	12/31/1954	Ingestion	(a)	3.16E+02	dpm/day
Th-230	10/1/1942	7/31/1946	Inhalation	M, S	Reserved	dpm/day
	10/1/1942	7/31/1946	Ingestion	(a)	Reserved	dpm/day
	8/1/1946	9/14/1947	Inhalation	M, S	Reserved	dpm/day
	8/1/1946	9/14/1947	Ingestion	(a)	Reserved	dpm/day
Ra-226	10/1/1942	7/31/1946	Inhalation	M	Reserved	dpm/day
	10/1/1942	7/31/1946	Ingestion	(a)	Reserved	dpm/day
	8/1/1946	9/14/1947	Inhalation	M	Reserved	dpm/day
	8/1/1946	9/14/1947	Ingestion	(a)	Reserved	dpm/day
Po-210	10/1/1942	7/31/1946	Inhalation	F, M	Reserved	dpm/day
	10/1/1942	7/31/1946	Ingestion	(a)	Reserved	dpm/day
	8/1/1946	9/14/1947	Inhalation	F, M	Reserved	dpm/day
	8/1/1946	9/14/1947	Ingestion	(a)	Reserved	dpm/day
Pa-231	10/1/1942	7/31/1946	Inhalation	M, S	Reserved	dpm/day
	10/1/1942	7/31/1946	Ingestion	(a)	Reserved	dpm/day
	8/1/1946	9/14/1947	Inhalation	M, S	Reserved	dpm/day
	8/1/1946	9/14/1947	Ingestion	(a)	Reserved	dpm/day
Ac-227	10/1/1942	7/31/1946	Inhalation	F, M, S	Reserved	dpm/day
	10/1/1942	7/31/1946	Ingestion	(a)	Reserved	dpm/day
	8/1/1946	9/14/1947	Inhalation	F, M, S	Reserved	dpm/day
	8/1/1946	9/14/1947	Ingestion	(a)	Reserved	dpm/day
Rn-222	10/1/1942	7/31/1946	Inhalation	-	Reserved[b]	WLM/y
	10/1/1942	10/31/1947	Inhalation	-	Reserved[c]	WLM/y
	11/1/1947	12/31/1954	Inhalation	(d)	0.48E+00	WLM/y

a. Choose same f_1-value as used for inhalation per NIOSH (2004).
b. Assign to workers who were who were likely to spend the majority of their time in process areas, or in boxcars or whose jobs were unknown.
c. Assign to workers who did not work or have their offices in the process buildings.
d. For the year 1947, assume that 048 WLM/y applies to the partial exposure period.

Table 12. Internal exposure summary for the Tonawanda Laboratory, October 1, 1942, through December 31, 1954

Radionuclide	Start	End	Intake route	Absorption type	Intake or exposure	Units
U-234	10/1/1942	10/31/1947	Inhalation	F, M, S	Reserved	dpm/day
	10/1/1942	10/31/1947	Ingestion	(a)	Reserved	dpm/day
	11/1/1947	12/31/1954	Inhalation	M, S	1.85E+01	dpm/day
	11/1/1947	12/31/1954	Ingestion	(a)	3.85E-01	dpm/day
Th-230	10/1/1942	10/31/1947	Inhalation	M, S	Reserved	dpm/day
	10/1/1942	10/31/1947	Ingestion	(a)	Reserved	dpm/day
	11/1/1947	12/31/1954	Inhalation	M, S	9.04E+00	dpm/day
	11/1/1947	12/31/1954	Ingestion	(a)	1.88E-01	dpm/day
Ra-226	10/1/1942	10/31/1947	Inhalation	M	Reserved	dpm/day
	10/1/1942	10/31/1947	Ingestion	(a)	Reserved	dpm/day
	11/1/1947	12/31/1954	Inhalation	M	9.04E+00	dpm/day
	11/1/1947	12/31/1954	Ingestion	(a)	1.88E-01	dpm/day
Po-210	10/1/1942	10/31/1947	Inhalation	F, M	Reserved	dpm/day
	10/1/1942	10/31/1947	Ingestion	(a)	Reserved	dpm/day
	11/1/1947	12/31/1954	Inhalation	F, M	9.04E+00	dpm/day
	11/1/1947	12/31/1954	Ingestion	(a)	1.88E-01	dpm/day
Pa-231	10/1/1942	10/31/1947	Inhalation	M, S	Reserved	dpm/day
	10/1/1942	10/31/1947	Ingestion	(a)	Reserved	dpm/day
	11/1/1947	12/31/1954	Inhalation	M, S	4.22E-01	dpm/day
	11/1/1947	12/31/1954	Ingestion	(a)	8.79E-03	dpm/day
Ac-227	10/1/1942	10/31/1947	Inhalation	F, M, S	Reserved	dpm/day
	10/1/1942	10/31/1947	Ingestion	(a)	Reserved	dpm/day
	11/1/1947	12/31/1954	Inhalation	F, M, S	4.22E-01	dpm/day
	11/1/1947	12/31/1954	Ingestion	(a)	8.79E-03	dpm/day
Rn-222	10/1/1942	10/31/1947	Inhalation	-	Reserved	WLM/y
	11/1/1947	12/31/1954	Inhalation	-	2.021E-01	WLM/y

a. Choose same f_1-value as used for inhalation per NIOSH (2004).

4.0 ESTIMATION OF EXTERNAL EXPOSURE, 1942-1954

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, 1954, the assumed date of completion of initial cleanup at the Ceramics Plant and its turnover to Linde. In some parts of this section, measurements and parameters for the period after 1954 are analyzed because they provide information used to estimate external exposures before 1955. Occupational medical exposures are treated in Section 5.0. Dose reconstruction for the period after 1954 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, and 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 (e.g., gloves) 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 collected at the Ceramics Plant (see Table A-6).

4.1 CERAMICS PLANT BETA AND GAMMA EXPOSURE

4.1.1 Post-production Radiation in Building 30

Little information was available on radiation levels in Ceramics Plant buildings during periods of non-production. Estimates for these periods were based on measurements made after the end of production in Building 30, the main processing building.

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'x15' 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 (mrep/h) was measured at contact at the square's center and corners and at 3' 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 13 display the results of a statistical analysis of the data.

Table 13. Floor and wall radiation in Building 30 before and after the 1949-1950 decontamination.

Time	Location	Measured beta plus gamma[a]		Estimated	
		Median (mrep/h)	GSD	Beta (mrem/h)	Gamma (mR/h)
Before vacuum cleaning and flushing	Contact	—[b]	—[b]	2.03[c]	0.131[c]
	3'	—[b]	—[b]	0.676[c]	0.131[c]
Pre-decontamination (just before decontamination and after vacuum cleaning and flushing)					
	Contact	0.719	1.67	0.675[d]	0.0438[d]
	3'	0.240	1.67	0.225[d]	0.0438[e]
After decontamination					
	Contact	0.311	2.04	0.292[d]	0.0189[d]
	3'	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/h was stated). These were used with the computer program LOGNORM4 (Strom and Stansbury 2000) to estimate the median and GSD. For measurements at 3 feet, 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 pre-decontamination values. The factor three is based on April 1949 data; see discussion in text.
d. Estimates based on beta and gamma percentages in Table 14.
e. Assumed to equal contact gamma exposure rate.

Another comprehensive survey of Building 30 was made in 1976 (ORNL 1978). Beta plus gamma levels (mrad/h) were measured at one centimeter from the surface, and gamma levels (µR/h) were

measured at one meter. Table 14 displays the results of a statistical analysis of the data. The levels found were similar to the post-decontamination results of 1950, indicating little change in conditions. Since 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 shown in Table 13 for the pre- and post-decontamination results.

The measurements just before decontamination in 1949-1950 were made after the building had undergone vacuum cleaning and flushing. Brief, semi-quantitative reports were available for two one-

Table 14. 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/h	3.47	—[b]
Gamma at 1 m[c]	0.0154	mR/h	1.69	—[d]
Estimated gamma at 1 cm[e]	0.0154	mR/h	—[d]	6.09%
Estimated beta at 1 cm[f]	0.238	mrad/h	—[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 out from the external gamma values. Outdoor external background in the Tonawanda area was reported to be 8-15 µR/h (ORNL 1978, p. 15).
d. Not applicable.
e. Estimated as equal to gamma at 1 meter.
f. Estimated as 0.253 (beta plus gamma) – 0.0154 (gamma).

day surveys taken in April 1949, before the vacuum cleaning and flushing (Blatz 1949; Wolf 1949). Typical levels of gamma radiation measured at 3' on April 19 and at contact on April 22 were similar and averaged about 0.18 mR/h, about four times higher than the estimated median contact gamma level before decontamination. Typical levels of beta measured at contact on April 22 averaged about 0.87 mrep/h, about 1.3 times higher than the estimated median beta level before decontamination. The April results were the basis of the estimate in Table 13 that beta and gamma radiation levels before vacuum cleaning and flushing were three times higher than the values measured afterward.

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

4.1.2 External Exposure During Non-production Periods at the Ceramics Plant

4.1.2.1 Pre-production, 1942-1943

For the pre-production period, it was assumed that there was no significant external beta or gamma exposure indoors for Ceramics Plant workers since their buildings were under construction or new. Some exposure while outdoors was possible since the transport of ore to Tonawanda Laboratory (Building 14) and the work there could have produced some site contamination. Resulting outdoor radiation levels are very likely to have been less than floor and wall radiation levels within Building 14 since 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 13 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 h/d. Table 15 shows estimated annual exposure rates.

Table 15. Pre-production beta and gamma radiation levels.

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

a. Based on Table 13 beta and gamma levels at 3' before vacuum cleaning and flushing. Assumed exposure period: 0.5 h/wd, 6 wd/wk, 50 wk/y. The beta and gamma rates are for the whole body.
b. Estimated on the basis of the GSD values in Tables 13 and 14.

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.

4.1.2.2 Standby, 1946-1947

Little information is available about the status of activities during the standby period. It is likely that the staff on site 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 whole standby period, each worker's allocation of time is assumed to have been as indicated by the occupancy factors in Table 16.

Gamma radiation levels were measured at six locations inside Building 30 on October 22, 1946. Measurements were made at one inch from the surface of interest. The results were reported as 0 R/8 hours for four of the locations and 0.005 R/8 hours (0.625 mR/h) for the other two locations (each near an ore dumping grill) (Howland 1946). Since 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 13 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 pre-production period. The gamma and beta radiation rates in an office building were assumed to be zero.

Table 16 summarizes the calculation of annual radiation rates based on the above parameters.

Table 16. Ceramics Plant beta and gamma radiation rates during standby.

	Category			Time-weighted radiation rate[a]	
Parameter	Office	Production	Outdoors	beta (rem/y)	gamma (R/y)
Beta (mrem/h)	0.000	0.676	0.676	—[b]	—[b]
Gamma (mR/h)	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 h/d exposure, 6 d/wk, 50 wk/y. 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.

Since 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.

4.1.2.3 Cleanup, 1949-1954

Cleanup of the Ceramics Plant is discussed in Section 2.6. Floor and wall radiation levels, measured at the start and end of the 1949-1950 decontamination of Building 30, were available and are summarized in Table 13. 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 17. Since the results in Table 17 are film badge measurements, they include floor and wall radiation in addition to radiation from contaminated equipment.

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

- Cleanup Worker
- Cleanup Support Worker
- Non-cleanup Worker

Table 17. Beta and gamma radiation measured during equipment removal, 1948.

Beta dose rate			
D_D (rem/y)[a]	GSD	D_M (rem/y)[a]	GSD
1.99	1.53	0.52	1.52
Gamma exposure rate			
D_D (R/y)[a]	GSD	D_M (R/y)[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 (2002), pp. 30 and 32.

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 a sandblaster operator) and laborers involved in moving and cleaning activities. A Cleanup Support Worker is one who spent a substantial portion of the work day 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 stationed 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:

$$D_D + D_M + OD$$

where D_D and D_M were taken from Table 17. The parameter OD is the outdoor dose rate. As was done for the pre-production and standby periods, it was estimated as equal to the estimated indoor floor and wall level at 3' before vacuum cleaning and flushing as given in Table 13. A cleanup support worker was assumed to have only half as much exposure to indoor radiation. The radiation level estimate was taken as

$$0.5*(D_D + D_M) + OD$$

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 worker's after 1946 (numbers were not adjusted for exposure time). The radiation level was estimated as

$$0.05*(D_D + D_M) + OD$$

for gamma radiation. The beta radiation values specified are for all parts of the body. Table 18 presents the results.

The beta dose rates in Table 18 are based primarily on film badge measurements. The film badges were worn on the chest (Heatherton 1948a). 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, scrubbing contaminated surfaces). For dose reconstruction, the beta dose rate to the hands and forearms of a cleanup worker was taken as three times that to the remainder of the body. The factor three was based on measurements discussed under the heading "Hand and Forearm Dose" at the end of Section 4.1.4.1.

Table 18. External dose reconstruction parameters for cleanup.

Period/Category	Beta[a]		Gamma[b]	
	Median (rem/y)[c]	GSD[d]	Median (R/y)[c]	GSD[d]
7/1/49-12/31/50, 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
1/1/51-12/31/54, 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 three times higher. See text.
b. Gamma exposure rates are for the whole body.
c. Annual rates based on 8.5 h/wd indoors, 0.5 h/wd outdoors, 50 wk/y, 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.

4.1.3 Production, 1943-1946

4.1.3.1 Beta

No beta film badge data were found for 1943-1946. There were beta film badge data for 1947-1949; however, none of the data pertained to Steps I or II, as these processes were no longer in operation. For 1943-1946, one report of workplace beta exposure rates was found. Solid samples collected at 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 1944b) and are reproduced here in the first five columns of Table 19. Technically, the unit roentgen (R) is defined only for photon radiation; beta dose should be expressed in absorbed dose units such as rad or rem. However, since 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 units of rem.

From the fact that no distance values are 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 (Table A-1).

The fourth and fifth columns of Table 19 contain the allowable times of exposure based on a 3.0 R/week 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 hours per week, much greater than the allowable exposure time of 18 hours. An 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.…*

Table 19. Beta radiation exposure rates measured at Linde Ceramics January 23, 1944.

From MED report of data during processing of L-30 African ore[a]					
Sample #	Sample location	Beta radiation (roentgen /8-hr day)[b]	Allowable time of exposure per man (hrs/week)[c]	Actual time of exposure per man (hrs/week)	Estimated worker dose rate (rem/y)[d]
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 1944b).
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/day and 3.0 R per six-day week for beta radiation, where R denotes one "Roentgen."
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 current text section. 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/8h)	hr/wk	hr/y	rem/y
Hands in tailings	1.4	10	500	88
Within range of radiation	0.7	40	2,000	175
Total				263

> *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 19 shows worker doses derived from the data. These range from 2 to 263 rem/y.

It is striking that the 1943-1946 dose rates indicated in Table 19 are much higher than the 1947-1949 rates discussed in Section 4.1.4.1. For 1943-1946 Step I Process Operators, the beta dose rate is estimated as 263 rem/y. For 1947-1949, 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/y (approximate value of annual D_D+D_M; see Table 30). Despite the high estimates, the 1943-1946 data does not appear unreasonable. The dose rates in Table 19 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 six-day week (Ferry 1944b) or 150 R/y. Therefore, the estimates in Table 19 are considered a valid basis for dose reconstruction.

4.1.3.1.1 Estimation of Worker Doses

Beta doses for specific jobs are estimated below and summarized in Table 36.

1943-1946 Loaders, Step I Process Operators, Step II Process Operators, Ball Mill Operator, and Weighmaster

The calculated dose shown in Table 19 for sample 13 was assumed to apply to a Loader. The doses shown in Table 19 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 Operator 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 Linde Air Products Company 1945). Reductions for shielding by apparel and containers were not taken into account in estimating doses because of insufficient information. The dose rates in Table 19 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 discussion under the heading "Hand and Forearm Dose" at the end of 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 Linde Air Products Company 1945) 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 hours a day at a location 4 ft from a pile of ore bags. The data in Table 19 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 19 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 pre-refined ore (which the domestic ore used at Linde may 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.

In order to use the data in Table 19 to estimate time-averaged beta dose rates applicable to the entire 1943-1946 production period at Linde, the ratio of the average dose rates to those measured in Table 19 was estimated. Table 20 documents the determination of the ratio. The grade range of the L-30 ore used at Linde was 8-12% (Aerospace Corporation 1981, Table B-1). To obtain the highest ratio, it was assumed that the measurements in Table 19 were made on the lowest L-30 ore grade, 8%. It was also assumed that beta dose was proportional to electron energy released per decay and 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 predicted by using the data in Table 19. Therefore, the estimated dose rates in Table 19 were multiplied by 0.84 to obtain time-averaged

dose rates for 1943-1946 production. The results are in Table 21, which summarizes the results of all beta dose rate estimates in this section and groups the job categories into three groups (high, medium, and low).

1943-1946 Workers with Jobs Analogous to 1947-1949 Jobs
Workers in 1943-1946 who held jobs analogous to ones in 1947-1949 were assigned three times the 1947-1949 beta dose rates. The factor three increase accounts for potential exposure to radiation from waste products from unrefined uranium ore and for the possibility that procedures in 1943-1946 did not involve as much radiological protection. (Per Tables A-2 and A-3, unrefined uranium materials release approximately 2.6 times as much electron energy per uranium decay as refined uranium materials. The plant had a health physicist in 1947-1949 but not in 1943-1946.) Workers with jobs analogous to those assigned to the medium beta exposure category for 1947-1949 were assigned dose rates of 17.6 rem/y to the hands and forearms and 5.85 rem/y to the remainder of the body.

Table 20. 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]
6/43 to 11/43	Not stated	Domestic & scrap	0.447	15.0%	0.72	1,000	0.028
12/43 to 10/44	L-30	African	1.157	10.8%	1.35	8,504	0.447
10/44 to 11/44	L-50	African	1.157	6.7%	0.84	1,486	0.048
12/44 to 2/46	L-19	Domestic	0.447	14.0%	0.68	6,102	0.161
2/46 to 7/46	R-10	African	1.157	3.5%	0.44	8,492	0.145
6/46 to 7/46	Q-20	African	1.157	17.7%	2.21	82	0.007
					Sum:	*25,666*	*0.84*

a. Per Aerospace Corporation 1981, Table B-1. A metric ton is 1000 kilograms.
b. Per Table A-2. Nuclear transformation is abbreviated nt.
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 1943-1946 ore beta dose to the ore dose due to 8% L-30 ore.

Workers with jobs analogous to those assigned to the low beta exposure category for 1949-1949 were assigned a beta dose rate of 3.00 rem/y to the whole body. Table 21 shows the assignments made.

1943-1946 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/y to the hands and forearms and 5.85 rem/y to the remainder of the body.

4.1.3.1.2 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 4.1.2.1. Therefore, the outdoor beta rate was estimated as 0.676 mrem/h (based on beta dose rate at 3' before vacuum cleaning and flushing given in Table 13). For 0.5 h/wd exposure, 6 wd/wk, and 50 wk/y, the average worker exposure would have been 0.10 rem/y. This is negligible compared to the indoor doses given the approximate nature of the estimates and was ignored.

4.1.3.1.3 Categories

Table 21 summarizes beta dose rates assigned for the 1943-1946 production period. All are considered to be median values of a lognormal distribution. The dose rates derived from the 1943-1946 data in Table 19 were judged to be based on near maximum exposure parameters. They were

assigned a GSD of 1.52, which corresponds to a ratio of 95th percentile value to median of two. The other dose rates were considered to be more uncertain. They were assigned a GSD of 2.65, which corresponds to a ratio of 95th percentile value to median of five.

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 job, it should be obtained as follows:

- For the low and medium categories, the values in Table 21 should be used.

Table 21. Assigned 1943-1946 beta dose rates.

Category/job	Median dose rate (rem/y)		GSD
	Hands & 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 & 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 text of Section 4.1.3.1.3.

- For the Ball Mill Operator, Step I Process Operator, and Weighmaster, the value in Table 21 should be used.

- For the Step II Process Operator, the dose to the hands and forearms should be taken as three times the remainder-of-the-body dose *for the high category*, i.e., as 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*, i.e., as 74 rem.

4.1.3.2 Gamma

Gamma film badge data are available for Step I process workers for a portion of the 1943-1946 production period. The data covers the period January 31, 1944 through February 26, 1945 except for a 3-week gap (4/18/44-5/8/44) for which data were not obtained due to a film handling error (Ferry 1944g). Usually, the badges were worn for about a week, but during the first three months, some of the badging periods were longer (two to four weeks). In the data analysis, each multi-week measurement was treated as a set of individual weekly results with each equal to the weekly average determined from the multi-week measurement. For example, a four-week result of 300 mR was treated as four individual one-week results of 75 mR.

Typically, about 20 workers were 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 or only some.

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). Afterwards, 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 per 8-hour day and also as 0.5 R/week and 0.6 R/week (Ferry 1944e; Tybout 1945a). Linde employees are reported to have worked a six-day, 48-hour workweek (Dupree1983a, b). For analysis of the gamma data, a six-day workweek was assumed. To obtain R/week, results in R/8h 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/week. In the analysis of the data, 44 mR/week was taken as the limit of detection (LOD), and results reported as "<X" or zero were considered to be "<LOD results." However, all non-zero results, even if below 44 mR/week, 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 (Aerospace Corporation 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 two weeks of November with the low point reached 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 1/31/44 – 11/12/44 (41 weeks) and 11/13/44 – 2/26/45 (15 weeks).

The data identified each worker's job activity. The job activities were grouped into categories. Each category consisted of jobs judged to have similar exposure potential based on job descriptions and the data. Table 22 lists the job activities and the category to which each was assigned.

African Ore
Table 23 displays statistical characteristics of the film badge data for the period 1/31/44 – 11/12/44, 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 limit of detection (LOD);

- f_M = missed fraction = the fraction of results in a year below the LOD;
- m_D = geometric mean of all results above the LOD;
- the geometric standard deviation (GSD) of all results above the LOD based on the assumption that their distribution was lognormal.

The subscripts *D* and *M* denote "dosimeter dose" and "missed dose," respectively, as defined in the NIOSH 2002 (pp. 30 and 32). For the current set of data, the value of f_D was estimated as the number of results ≥LOD divided by the number of measurements, and f_M was estimated as the number of results <LOD 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.)

Table 22. Job categories for 1943-1946 Step I gamma dose analysis

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
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.

Table 23. Step I gamma exposure data for African ore processing, 1/31/44-11/12/44.

Category	# Weekly results	Dosimeter fraction f_D	Missed fraction f_M	For results in dosimeter fraction	
				Geometric mean m_D (mR/week)	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 estimated median annual dosimeter dose D_D for a 50-week year is shown in Table 24. This was calculated as follows:

$$D_D = 50\ f_D\ m_D \quad (2)$$

For simplicity and because a GSD value based on small number of measurements might not be valid, the highest GSD in Table 23 was assigned to every job category.

In accordance with NIOSH 2002, 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:

$$D_M = 50\ f_M\ (LOD/2) \quad (3)$$

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

A GSD of 1.52 was calculated from D_M and D_{M95}. The results for D_M are in Table 24.

Table 24. Step I gamma exposure rate parameters for African ore processing, 1/31/44-11/12/44.

Category	D_D[a] (R/y)	GSD[b]	D_M[a] (R/y)	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 23 is assumed to apply to all categories.

Domestic Ore

The analysis of exposure data for the period 11/13/44 – 2/26/45, 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 25 and Table 26. 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 25. Step I gamma exposure data for cleanup and domestic ore processing, 11/13/44-2/26/45.

Category	# Weekly results	Dosimeter fraction f_D	Missed fraction f_M	For results in dosimeter fraction	
				Geometric mean m_D (mR/week)	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 as 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 as only one result was ≥LOD.
d. Not available as no results were ≥LOD.

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.

4.1.3.2.1 Estimation of Worker Doses

African Ore

Step I Process Workers

The annual gamma exposures of the monitored Step I workers (Table 24) 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/y, the highest value of D_D+D_M in Table 24, and a GSD of 2.61, the highest GSD in Table 24.

Table 26. Step I gamma exposure rate parameters for cleanup and domestic ore processing, 11/13/44-2/26/45.

Category	D_D[a] (R/y)	GSD	D_M[a] (R/y)	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.

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 non-radioactive 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 stored in ore piles. A report of radiation surveys conducted March 2 and 3, 1944 (Ferry 1944c) 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 ft x 30 ft x 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 used at Linde (Aerospace Corporation 1981, Table C-1), the ore density was estimated as 3.25 g/cc. Therefore, the 30x30x12 pile contained approximately 2,000,000 lbs 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 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 1944c), and a letter from the Ceramics Plant dated May 8, 1944 indicated compliance with this recommendation (Rehm 1944a).

In Ferry 1944c, an exposure rate of 0.23 R/8-hr day was reported at 1.5 ft from the 30x30x12 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 obtained in Linde pilot plant studies for a 4 ft x 5 ft x 20 ft pile of 8% ore (Skinner 1944; Wiesendanger 1944). The fit was obtained using the MicroShield computer code (MicroShield Team 2003), which is based on the point kernel methodology. The values calculated by the code were multiplied by a single scaling factor chosen to provide a best fit to the measured data. The adjustment compensated for simplifications in the modeling. Figure 5 shows the data and the fit obtained. The data covers the range 0 ft to 10 ft from the 4x20 face. The adjusted MicroShield output fit the data to within ±10% between 3 ft and 10 ft from the face. The results were extended to larger distances and to a 30x30x12 pile by more MicroShield calculations scaled with same factor. Figure 6 shows the MicroShield results and an analytic function fit to them. The function facilitates calculating 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, 1.5 ft from the face. The result was 0.19 R/8 hours, 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.

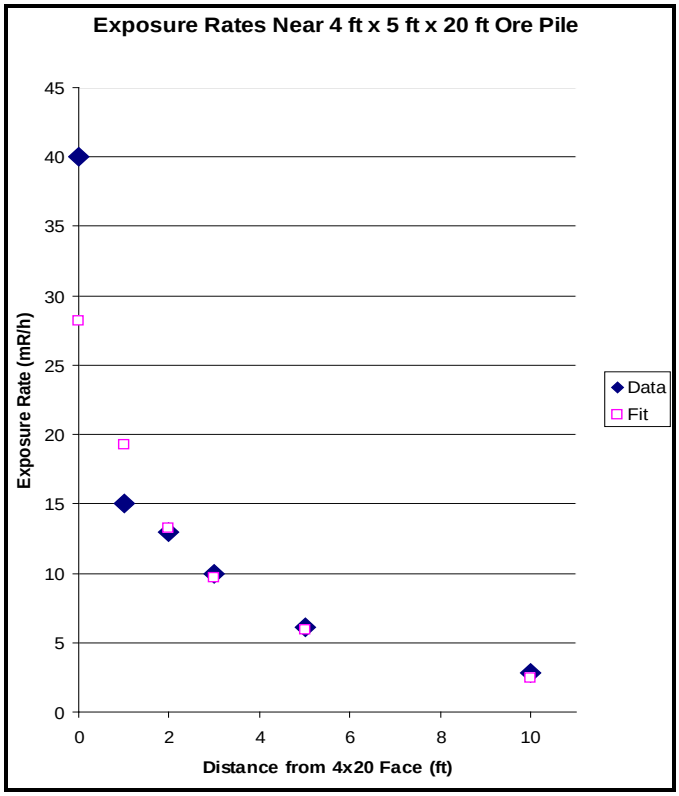

Figure 5. Fit to gamma exposure data for a 4 ft x 5 ft x 20 ft pile of 8% African pitchblende ore.

Gamma exposure rates measured during pilot

plant studies at Linde Ceramics (Skinner 1944) are shown by the diamond-shaped symbols. The rectangles show the fit obtained to the data by the method described in the text (Section 4.2.1.2) with distance assumed measured along the perpendicular bisector of a 4x20 face. The ore grade (8%) was assumed to be that determined in the analyses reported on p. 18 of Wiesendanger 1944.

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. Also, the calculations are based on distance along the perpendicular bisector of a pile face. Exposure rate at any distance will be lower for a point not on the perpendicular bisector.

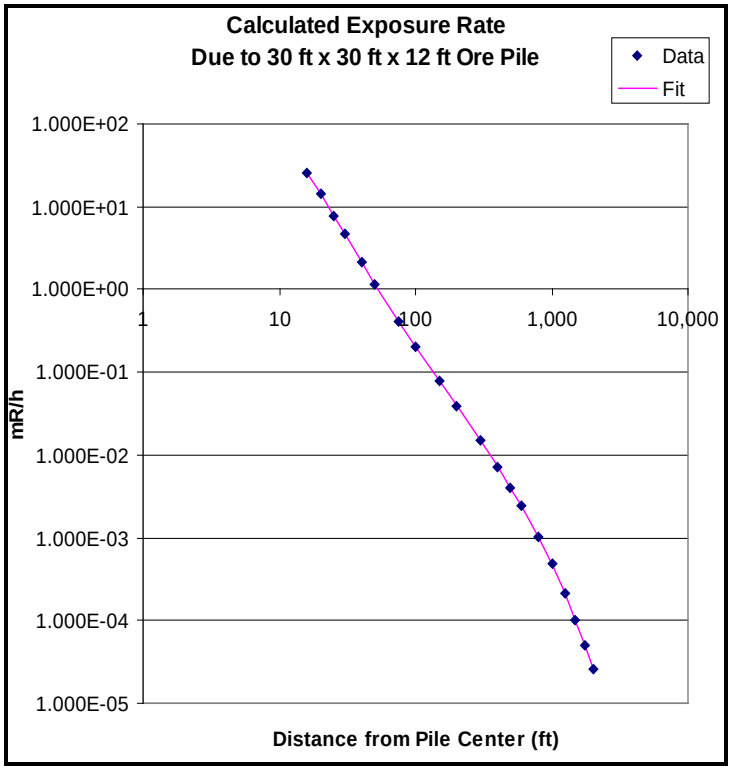

Figure 6. Calculated exposure rates and analytic fit for a 30 ft x 30 ft x 12 ft pile of 8% African pitchblende ore. The diamond-shaped symbols show calculated values of exposure rate as a function of distance from the pile center. Distance is measured along the perpendicular bisector of a 12x30 face. 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 $(1/r^2)*\exp[a*\exp(-b*r) + (c-d*r)]$, where a = 1.906, b = 3.126E-02, c = 7.660, and d = 1.521E-03. The fit agreed with the calculated values to within ±3.0% over the range 16 ft to 2,000 ft from the center of the pile.

To estimate impact on the other workers in Building 30, who did not wear dosimeters, exposure rates due to a 30x30x12 pile located 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 27. They range from 0.28 R/y (for the Laboratory) to 4.9 R/y (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 may have worked near other ore piles, near tailings piles, or near large quantities of ore in process. Since the locations and quantities of these other sources are not known in sufficient detail, any worker who spent a substantial of work time in Building 30 had a potential for significant gamma exposure. Such workers included not only the production personnel who worked 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 lunch room, and office personnel who may have had to regularly visit the building (e.g., product accountants, industrial relations representatives, the fire marshal, engineers). Finally, it appears that there were office personnel who were permanently stationed in the building. A February 1944 letter from the plant addressing the frequency of required medical exams, states that it would not be appropriate to reduce the exam 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 1944b). In view of the likely difficulty of establishing that a particular worker did not spend a substantial portion of work 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 are assigned a gamma dose of 5.35 R/y for the whole body, based on the results for the most exposed group of monitored process workers (Table 24).

Table 27. Estimated maximum exposure rates in selected Building 30 facilities due to 30 ft x 30 ft x 12 ft pile of 8% ore in southeast corner.

Facility	Distance (ft)[a]	Exposure rate (R/y)[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 2550 h/y exposure.

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 used at Linde (see Section 2.2). The value of D_D+D_M for Step I Process Operators during domestic ore processing (Table 26) is 42% of the value during African ore processing (Table 24). 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 1943-1946 production period, the gamma exposure rate of 5.35 R/y is assumed to apply for the whole period.

4.1.3.2.2 Outdoor Exposure Rate

The average outdoor gamma exposure 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 4.1.2.1. Therefore, the outdoor gamma rate was estimated as 0.131 mR/h based on the level before vacuuming and flushing in Table 13. For 0.5 h/wd exposure, 6 wd/wk, and 50 wk/y, the average worker exposure would have been 0.020 R/y. This is negligible compared to the indoor exposures given the approximate nature of the estimates and was ignored.

4.1.4 Rehabilitation and Production, 1947-1949

4.1.4.1 Beta

For part of 1947-1949, weekly film badge measurements of beta and gamma exposure were available (Linde Database 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 6000 records dated from 1/7/48 through 12/12/49. 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 dated before 3/15/48, badges were processed by the University of Rochester (Heatherton 1948b, Osinski 1948a); thereafter, they were processed by the U.S. AEC NYDO 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 label indicating 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 1949c,d). 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 3/15/48, a beta value below the limit of detection was reported as "<80." After 3/15/48, 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 in order 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 specified in the records were binned into categories that combined jobs judged to have had similar exposure potential. Table 28 displays the job titles in each category.

Table 29 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 1943-1946 gamma dosimetry data (Section 4.1.3.2). 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, 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.

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 1943-1946 gamma data (see Equations 2, 3, and 4 in Section 4.1.3.2). Since badges would not have been worn until workers arrived indoors, an outdoor beta dose rate of 0.10 rem/y (from Section 4.1.3.1.2) 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 30 displays the results.

Table 28. Job categories for 1947-1949 Step III film badge analysis.

Job title	Category	Job title	Category
AEC	Office	Maint G L	Maintenance
Asst Engineer	Engineer	Maint Group Leader	Maintenance
Asst Supt	Superintendent	Maintenance	Maintenance
Carpenter	Maintenance	Millwright A	Maintenance
Chem Oper	Process Operator	Millwright C	Maintenance
Chem Oper A	Process Operator	Moveman	Loader
Chem Oper B	Process Operator	Painter	Maintenance
Chem Oper C	Process Operator	Painter B	Maintenance
Chem Oper G L	Process Operator	Personnel	Office
Chemist	Chemist	Pipefitter	Maintenance
Control Engr	Engineer	Process Foreman	Process Operator
Electrician	Maintenance	Prod Accounting	Office
Electrician Hlpr	Maintenance	Property Dept	Office
Fire Chief	Engineer	Safety Engr	Engineer
Foreman (Unspecified Type)	— [a]	Seamster	Seamster
Group Leader (Unspecified Type)	— [a]	Security Agent	Office
Health Physicist	Engineer	Shift Storekpr	Storekeeper
Janitor	Janitor	Shift Stores Att	Storekeeper
Labor Foreman[b]	Loader	Shipping & Rec	Storekeeper
Laundry Helper	Laundry Worker	Storekeeper	Storekeeper
Laundry Worker	Laundry Worker	Superintendent	Superintendent
Loader	Loader	Tool Crib Keeper	Storekeeper
Loader Foreman	Loader	Trades Helper	Maintenance
Mail Girl	Office	Truck Driver	Truck Driver
Maint	Maintenance	Weighmaster	Weighmaster
Maint Foreman	Maintenance	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 29. 1947-1949 Step III beta dose data.

Job	# Badges	Dosimeter fraction f_D	Missed fraction f_M	For results ≥ LOD	
				Geometric mean m_D (mrem/week)	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.

For most jobs, the largest contribution to D_D+D_M+OD was missed dose D_M. The production workers in 1947-1949 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, lunch room, nurses' office, and chemistry lab; 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 is estimated in

Table 30. Beta dose rates for Step III, 1947-1949.

Job	D_D [a]		D_M+OD [a]		D_D+D_M+OD
	(rem/y)	GSD [b]	(rem/y)	GSD	(rem/y)
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 29 is assumed for all categories for which a GSD can be assigned.
c. No result ≥ LOD.

Section 4.1.1 as 0.676 mrem/h (value before vacuuming and flushing in Table 13), which is equivalent to 34.5 mrem/wk and 1.7 rem/y for a person spending 8.5 hours per day, 6 days per week, and 50 weeks per year in the building. This is comparable to the film badge LOD for beta radiation (35 mrem/wk) and so may have been a substantial portion of the missed dose. Based on this estimate of 1.7 rem/y of possible dose and the lowest film badge D_D+D_M value in Table 32 of approximately 1 rem/y for workers, who were regularly badged, it was assumed that all workers, even office workers, received a beta dose of at least 1 rem/y. Office workers were included because it is unknown which

office workers spent substantial portions of their time in production buildings. Some workers who may have had offices in the office building may have spent much of their work day in the production area (e.g., engineers, health physicist, fire marshal) and some clerical workers may have actually had work locations in the production area (see discussion near the end of 4.1.3.2.1 and Holmes 1944b).

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/y. 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 1943-1946 for Truck Drivers to work as Loaders when not driving a truck (Linde Air Products Company 1945, p. 10); it was assumed that the same practice applied in 1947-1949. 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/y. 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 in order to account for exposures incurred while cleaning contaminated facilities, setting up, and testing equipment.

Hand and Forearm Dose
Film badges were worn on the chest (Heatherton 1948a). 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 depending on the operation and the area of the body (Heatherton 1948a). The results are shown in Table A-6. Based on these data, it was estimated for this TBD that the average dose to the hands and forearms of a worker handling radioactive material was three 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 operations studied with the result rounded up to the nearest integer. No credit was taken for wearing gloves. For dose reconstruction, a hands and forearm dose of 5.85 rem/y was assumed for the medium category. Workers in the low category were assumed to have a hands and forearm dose of 1 rem/y, 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 1947-1949 production were available (Linde Database 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 (mR). Results reported as "0", "*" or "**" were interpreted as meaning "less than the LOD." For records date before 3/15/48, a gamma value below the LOD was reported as "0" or "<50." For records dated 3/15/48 or later, the limit of gamma detection was not explicitly stated, but it was deduced to be 45 mR from an examination of the data. A few scattered reports of non-zero 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 employed in the analysis of the beta data (see Section 4.1.4.1). Table 31 summarizes the statistical characteristics of the gamma data.

Table 31. 1947-1949 Step III gamma exposure data.

Category	# Badges	Dosimeter fraction f_D [a]	Missed fraction f_M [b]	For results ≥ LOD	
				Geometric mean m_D (mR/week)	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 1947-1949 beta data (see Section 4.1.4.1). Since badges would not have been worn until workers arrived indoors, an outdoor gamma exposure rate of 0.020 R/y (from Section 4.1.3.2.2) 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 32 displays the gamma exposure rates that were determined.

Table 32. Gamma exposure rates for Step III, 1947-1949.

Job	D_D [a]		D_M+OD [a]		D_D+D_M+OD (R/y)
	(R/y)	GSD [b]	(R/y)	GSD	
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 31 is assumed for all categories for which a GSD can be assigned.
c. No results ≥ LOD.

As in the case of 1947-1949 beta dose (see 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/h (value before vacuuming and flushing in Table 13), which is equivalent to 6.7 mR/wk and 0.34 R/y for a person spending 8.5 hours per day, 6 days per week, and 50 weeks per year 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 1944b) are assumed to have received this estimated floor and wall radiation plus the 0.14 R/y received by the occasionally badged office worker for a total of 0.48 R/y. Because it may not be easy to identify which clerical personnel worked in a production building, all unbadged or infrequently badged personnel are assumed to have received 0.48 R/y.

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 1947-1949 beta category assignment (see subsection "Categories" in Section 4.1.4.1). Jobs in the medium gamma category were assigned an exposure rate of 1.61 R/y for the whole body. Jobs in the low category were assigned an exposure rate of 0.48 R/y 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 incurred while cleaning contaminated facilities, setting up, and testing equipment.

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 may occasionally have visited the research facilities. It includes secretaries, clerks, draftspersons, and high-level managers. Three periods (research and development, cleanup, and post-cleanup) 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 those found later at the Ceramics Plant. Therefore, radiation level estimates were based to a large extent on the estimates for workers who did similar work at the Ceramics Plant. Table 33 shows the assumed annual radiation rates and their bases.

Table 33. Tonawanda Laboratory beta and gamma radiation, 1942-1954.

Period			Beta				Gamma		
			rem/y						
Start	End	Job	hands & forearm	remainder of body	GSD	Basis[a]	R/y[b]	GSD	Basis[a]
R&D									
10/01/42	07/31/46	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									
08/01/46	12/31/46	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
Post-cleanup									
01/01/47	12/31/54	All	3.26E-01[c]	3.26E-01[c]	—[d]	E	6.80E-02	—[d]	E

a. Bases:
   A. One half of value for most exposed Ceramics Plant process worker during 1943-6 production (Step I Process Operator). The factor 2 reduction allows for the lower intensity of research work in comparison with production, e.g., 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.
   B. Based on Ceramics Plant office worker during 1943-6 production.
   C. Cleanup Worker during Ceramics Plant decontamination.
   D. Non-cleanup Worker during Ceramics Plant decontamination.
   E. Building 30 post-contamination level (Table 13) for 8.5 h/wd plus outdoor level on Tonawanda site for 0.5 h/wd. Outdoor level assumed equal to Building 30 level before vacuum cleaning and flushing (Table 13). A 6-day week is assumed.
   F. Based on most exposed Ceramics Plant process worker during 1943-6 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 present in the compounds processed at Linde are presented in Table 34 below.

Table 34. Natural uranium per gram dose rates at 1 foot.

Chemical form	Dose rate (rem/h-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

ORAUT-OTIB-0024 Data used to calculate the alpha-neutron dose rates is from Shleien et al (1998), Salmon and Hermann (1992) NRC (1991) and DOE 2000.

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

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

2. Point source strength is based on the daily rate of U_3O_8 handling or UF_4 production. Self-shielding within the source-target compound matrix is assumed to be negligible.
3. A U-238/U-235 composition of 99.3% to 0.711% by weight for natural uranium.
4. The entire mass of uranium compounds is conservatively assumed to be attributable to the uranium (source) content.
5. Average neutron energy from alpha-neutron reactions of 2.0 MeV.
6. Dose equivalent rate-to-fluence rate conversion factor for 2.0 MeV of 1.3×10^{-4} rem/hour per neutron/cm^2-s.
7. The work year consisted of 8 hours per day, 6 days per week, and 50 weeks per year.

During the Step I and Step II conversion processes in Building 30 (Ceramics Plant) the bounding neutron dose rate would be attributable to operations involving the handling of African Ore with alpha emitting progeny in secular equilibrium through Ra-226. This arises from the fact that the African ore feed material contains alpha emitting progeny that contribute to the alpha-neutron production rate. As indicated in Table 34, the African ore form produces 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 following 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. Consequently, the dose received by "Loaders" or "Movemen" would be bounding for all personnel engaged in Step I and Step II processing activities. As indicated in the Table 34, during the Step III conversion process in Building 38, the neutron dose rate from UF_4 is almost eighty-five times greater than from the UO_2 feed material. Consequently, 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 metric tons (26 million kilograms) of ore over a period of 37 months beginning June 1943 and ending July 1946. Assuming a six-day workweek for 50 weeks per year, and a uniform rate of ore processing, 28,111 kg per workday (5,622 kg of uranium) were handled at Linde. The claimant favorable assumption 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 claimant favorable assumption 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 3-20% 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 1949d) or 3,147 kg per day.

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 1/10 of the daily production amount of U_3O_8 or UF_4 at a distance of 1 foot. This quantity of material is a claimant-favorable estimate 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 1/3 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 working 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 foot distant 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:

- Research and development for Steps I, II and III.

  This work occurred primarily in the period October 1942 to December 1943 (Jenness and Ewing 1943). The largest scale efforts were pilot plant projects. The Step I pilot plant was producing 2 to 3 tons of U_3O_8 per week in November 1942 (Bonsib 1942). The Step III equipment could convert up to 90 lbs of UO_2 to UF_4 in a single run (Tonawanda Laboratory 1946b, p. 84).

- Conversion of UF_6 to UO_3

  This work appears to have been done after November 1943 (since it is not mentioned in the progress report describing work from October 1942 to November 1943; see Jenness and Ewing 1943). Interim reports were issued from July 1944 through July 1945, and a final report was issued in May 1946 (Tonawanda Laboratory 1946a, 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 (Tonawanda Laboratory 1946b, p. 89). The work involved processing 2750 lbs 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 tons of U_3O_8 processed per week corresponding to a daily processing rate of 454 kg from 10/1/42 through 2/29/44.

- Exposure to 100 lbs of UF_4 or UF_6 per week daily from 10/1/42 to 7/31/46. This accounts for the Step III and UF_6 to UO_3 work.

- Exposure to 3000 lbs of UF_4 for 2 months at some point between 10/1/42 and 7/31/46. This accounts for the UF_4 grinding research. The neutron dose in a year is equivalent to that which would be produced by exposure to 500 lbs of UF_4 continuously for a year. Since the dates of the work are uncertain, it is assumed that there was continuous exposure to 500 lbs of UF_4 from 10/1/42 to 7/31/46.

To reflect all types of UF_4/UF_6 work, a daily source term of 600 lbs (272 kg) is assumed. All source terms (both oxide and fluoride) are divided by 3 to account for the fact that no worker would be within close range of the source material at all times. This yields 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 are assumed to be exposed to these source amounts at a distance of 1 foot.

Table 35 summarizes the model parameters and presents calculated dose rates.

Table 35. 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/h)	Dose rate (rem/y)
Building 30 6/1/43 – 7/31/46	Steps I and II	African ore 20% U by weight	Oxygen	5.62E+6	1.15E-5	2.77E-2
Building 38 7/25/43 – 8/31/46[a] 9/15/47 – 9/30/49[a]	Step III	UF_4	Fluorine	2.76E+6	2.08E-4	5.0E-1
Tonawanda Laboratory (Building 14) 10/1/42 – 2/29/44	Steps I & II	U_3O_8	Oxygen	1.51E+5	3.10E-6	7.44E-3
Tonawanda Laboratory (Building 14) 10/1/42 – 7/31/46	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 is extended beyond the end of the production period to account for inventory left on hand. UF_4 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 1949e) and data on production at Electromet (DOE undated, pp. 23-4).

4.4 EXTERNAL DOSE RECONSTRUCTION SUMMARY, 1942-1954

This section summarizes guidelines and parameters for reconstruction of doses due to external radiation during 1942-1954 except for doses due to occupational medical exposure, which are addressed in Section 5.0. The parameters provided in this section are for use when individual worker data are unavailable or inadequate.

For dose reconstruction, exposures for different jobs have been grouped into categories. Table 36 specifies the categories and the annual radiation doses or exposures assigned to each. For the production periods at the Ceramics Plant, categories are 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 possibly also outdoor dose OD, the dose was estimated as the sum of the parameters (e.g., $D_D + D_M + OD$). Each estimate is considered to be the median of a lognormal distribution. A GSD of 3.0 is assigned to all beta and gamma dose estimates based on the typical GSD levels estimated for the underlying data, i.e., beta and gamma doses are assigned as lognormal distributions with GSDs of 3, and an acute exposure rate. Neutron doses are assigned as a constant distribution and a chronic exposure rate.

Table 36. Summary – annual external exposure from AWE operations, 1942-1954.

Year[e]	Work category	Beta (rem)[e]		Gamma (R)[e]	Neutron (rem)[e,l]
		Hands & forearms	Rest of body		
Ceramics Plant (Bldgs 30, 31, 37, 38)					
1942[f]	All workers	2.55E-02	2.55E-02	4.97E-03	—[a]
1943[g]	High	1.51E+02[b1] 5.05E+01[b2]	5.05E+01	3.65E+00	3.41E-01[m]
	Medium	1.20E+01	3.97E+00		
	Low	2.08E+00	2.08E+00		
1944 1945	High	2.21E+02[b1] 7.40E+01[b2]	7.40E+01	5.35E+00	5.00E-01[m]
	Medium	1.76E+01	5.85E+00		
	Low	3.00E+00	3.00E+00		
1946[h]	High	1.28E+02[b1] 4.32E+01[b2]	4.32E+01	3.11E+00	3.33E-01[m]
	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[m]
	Low	6.10E-01	6.10E-01	2.03E-01	
1948	Medium	5.85E+00	1.95E+00	1.61E+00	5.00E-01[m]
	Low	1.00E+00	1.00E+00	4.80E-01	
1949[j]	Medium/Low	6.85E+00	2.28E+00	1.73E+00	3.74E-01[m]
	Cleanup				—[a]
	Non-cleanup	5.47E-01	5.47E-01	2.94E-01	
1950 1951 1952 1953 1954	Cleanup[c]	7.83E+00	2.61E+00	1.85E+00	—[a]
	Non-cleanup[d]	3.26E-01	3.26E-01	1.11E-01	
Tonawanda Laboratory (Building 14)					
1942[f]	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[k]	Research	6.78E+01	2.26E+01	3.88E+00	8.36E-02n
	Office	1.78E+00	1.78E+00	3.15E+00	
1947 1948 1949 1950 1951 1952 1953 1954	All workers	3.26E-01	3.26E-01	6.80E-02	—a

a. Neutron dose rate was negligible.
b. 1. Based on 221 rem/y for Ball Mill Operator, Step I and Step II Process Operators, and Weighmaster
2. Based on 74 rem/y for Loader, per Section 4.1.3.1.3.
c. All Cleanup Workers and Cleanup Support Workers as defined in Section 4.1.2.3 are assigned to the Cleanup exposure category. Parameters are those of the Cleanup Worker for a six-day week in Table 18.
d. All Non-cleanup workers as defined in Section 4.1.2.3 are assigned to the Non-cleanup exposure category. Parameters are those of the Non-cleanup Worker for a six-day week in Table 18.

Table 36 (Continued). Summary – annual external exposure from AWE operations, 1942-1954.

e. Total annual exposure (dose) for the designated year. Prorated based on calendar year and applicable notations below.
f. Exposure for the period of 10/1/42 through 12/31/42 only.
g. For pre-production period of 1/1/43 through 4/26/43, assigned exposures of 3.21E-02 rem, 3.21E-02 rem and 6.26E-03 R, respectively for the dose categories in the table.
h. For the standby period of 8/1/46 through 12/31/46, assigned beta and gamma exposures of 1.87E-01 rem, 1.87E-01 rem and 3.63E-02 R, respectively for the dose categories in the table.
i. For the standby period of 1/1/47 through 9/14/47; assigned all workers beta and gamma exposures of 3.14E-01 rem, 3.14E-01 rem and 6.10E-02 R, respectively for the dose categories in the table.
j. Includes exposures from the Step III period of 1/1/49 through 6/30/49. High/ Medium category worker exposures of 2.90 rem, 9.67E-01 rem and 7.98E-01 R and Low category workers exposures of 4.96E-01 rem, 4.96E-01 rem and 2.38E-01 R, respectively for the dose categories in the table.
k. Includes exposures from the Cleanup period of 8/01/46 through 12/31/46; assigned beta and gamma exposures for Research workers – 3.28 rem, 1.09 rem and 7.75E-01 R, respectively and Office workers – 4.23E-02 rem, 4.23E-02 rem and 4.65E-02 R, respectively for the dose categories in the table.
l. 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.
m. The Building 38 neutron dose rate estimated for Step III processing was assumed to apply from 4/27/43 to 8/31/46 and from 9/15/46 to 9/30/49. The neutron dose rate was negligible from 9/1/46 to 9/14/49 (standby) and after 9/30/49 (cleanup and post-cleanup). The period of neutron exposure extended beyond the end of production in 1946 and 1949 due to remaining inventory of UF_4
n. Includes neutron exposures through 7/31/46.

Prorating of exposures that were less than a year is not straightforward for all periods. If doses are prorated, the footnotes to Table 36 should be checked to determine how the dose is distributed throughout the year. Beta doses are assigned as electrons, >15 keV. Gamma doses are assigned as photons, 30 to 250 keV, which is claimant favorable. Neutrons are assigned as neutrons, 0.1 to 2 MeV.

Table 37 shows how Ceramics Plant production jobs are assigned to the various categories. For 1943-1946, the assignments are based on Table 21 and the discussion in Section 4.1.3.1.3. For 1947-1949, the assignments are based on Tables 30 and 32 and the discussion in Section 4.1.4.2. In each case, some job titles reflecting subcategories of titles in the earlier tables have been added.

For the Ceramics Plant cleanup period, the category “cleanup” includes both cleanup workers and cleanup support workers as defined in Section 4.1.2.3. For Tonawanda Laboratory, the categories “research” and “office” 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.

5.0 OCCUPATIONAL MEDICAL EXPOSURE

5.1 CERAMICS PLANT EMPLOYEES

Pre-employment, repeat and termination chest x-rays were required for at least some of the workers at Linde Ceramics. On certain occasions, pelvis x-rays were also required.

5.1.1 Chest X-rays

The earliest record found of a requirement for chest x-rays at Linde is in a report of a conference on worker safety at Linde held March 31, 1943 (Brimm and Neumann 1943). The conference report called for a pre-employment medical examination that included "fluoroscopic examination.” A September 4, 1943 letter from the Medical Section of the MED recommended initial and repeat examinations that included chest x-rays (Van Horn 1943a). Available medical records indicate that

Table 37 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 Marshall	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.

chest x-rays were regularly administered to workers in 1943 and later years. Based on the correspondence and the indications of plant practice found in the medical records, the assumptions below are provided for use in reconstruction of doses due to chest x-rays.

Linde Ceramics X-ray Assumptions

1. Applicability: These parameters apply to all job categories at the Ceramics Plant.

2. X-ray Category and Frequency:

   a. Pre-employment: Taken prior to starting employment at Linde Ceramics. These x-rays were also required for Linde employees transferring to Linde Ceramics from other parts of the company.

   b. Repeat

      (1) Before August 13, 1943, taken at three-month intervals.
      (2) From August 13, 1943 through December 31, 1944, taken at six-month intervals.

(3) From January 1, 1945 on, taken at 12-month intervals.

c. Termination: Taken upon termination of employee participation in MED/AEC work at Linde (either due to employee leaving or the work ending).

3. Methodology:

a. Before July 16, 1943, by fluoroscopy of the lungs.
b. On or after July 16, 1943, by taking an 11"x17" radiographic film x-ray.

4. Period: The chest x-ray program is assumed to have started on November 16, 1942 and ended on December 31, 1954.

Bases of Assumptions

1. Applicability: A MED letter (Van Horn 1943a) recommended chest x-rays for “those persons working in direct contact with special materials," where the term "special materials" appears to have denoted materials containing uranium or uranium and radium (Ferry 1944a). When the medical examination program at the Ceramics Plant was established, it was decided that all employees should receive the same physical examinations and rechecks (Cranch 1944a). By January 1944, after reports from the early exams did not indicate any noticeable medical hazard, discussions were under way on reducing the frequency of examinations for those clerical workers, who had no occasion to go into operating areas (Cranch 1944a). The suggestion was made that the examination frequency be reduced for office personnel working in the office annex building but not for certain clerical help (described as "some stenographers and clerks, stores clerks, etc."), who worked in the main building, and were "from time to time exposed to the same hazards as operating personnel" (Holmes 1944b). A reduced frequency for guards was also suggested. By November of 1944, all employees were to have a pre-employment examination, which included a chest x-ray. Employees "working in the plant or who have occasion to come in contact with any portion of the process or the materials" were given periodic repeat exams and a termination exam. However, employees "not having occasion to contact the process in any way" were exempt from the periodic and termination exams (Holmes 1944c). From the foregoing, it is clear that some employees were not subject to repeat and termination x-rays. However, it is not clear how those employees can be identified. Therefore, the general assumption for dose reconstruction is that all employees were subject to the same chest x-ray imaging requirements.

2. X-ray Category and Frequency: Pre-employment, repeat and termination x-rays are discussed in various 1943 correspondence. A three-month repeat interval was stipulated in the report of March 31, 1943 conference on Linde Step II safety (Brimm and Neuman 1943). A six-month interval was specified in a letter distributed August 13, 1943 (Cranch 1943). No medical records exemplifying a three-month interval were found, but the available medical records for 1943 appear to be incomplete. Available worker files in which the first indicated chest x-ray date is July 1943 or later are consistent with a repeat interval of approximately six months for x-rays in 1944 and a repeat interval of 12 months after 1944.

3. X-ray Type: The March 31, 1943 safety conference stipulated that the pre-employment exam should include "fluoroscopic examination." A medical record indicates chest examination by "fluoroscope, lungs" in a pre-employment exam on March 31, 1943 (Claim File *number redacted*, DOE Response *number redacted*, p. 23). A comment in the notes of a safety conference held May 7, 1943 indicates that the fluoroscopic examinations did not involve the use of film. The conference participants recommended the use of x-ray film in order to have a

permanent record of the chest condition (Brimm 1943b). The earliest chest x-ray found in available worker files was a radiographic film dated July 16, 1943 (Claim File *number redacted*, DOE Response *number redacted*, p. 28). The type of film x-rays taken at Linde (radiographic rather than photofluorographic) and their size (14"x17") was determined from several pieces of evidence:

(a) Some of the claimant files contain data sheets from the Black Rock Clinic, where Linde x-rays were taken, that state the size to be 14"x17" (e.g., Claim File *number redacted*, DOE Response *number redacted*, p. 61).

(b) Approximately 60 arbitrarily selected Linde Air Products x-ray films in the ORO (Oak Ridge Organization) vaults were pulled and viewed. All were 14"x17" (Beck 2004).

(c) In 1949, Linde requested proposals for microfilming various records on cards and x-ray films. A response to this request indicated that 3000 14"x17" x-ray films were involved and mentions no other size (Medhurst 1949).

One piece of evidence suggesting the use of photofluorographic x-ray techniques was found, a purchase requisition dated November 4, 1944 for three gross of Type F X-ray film, 4.5"x10" (Linde Database 1944). The size of this film was appropriate for photofluorographic stereoscopic chest x-rays (Cardarelli et al 2002, p. 492). However, the following factors suggest it was not used for such x-rays:

(1) No photofluorographic film images were found in the Linde claim files.

(2) The purchase requisition was issued by Linde and called for delivery of the film to the "Laboratory." If the film were for chest x-rays, it probably would have been issued by the Black Rock Clinic, the only location for taking x-rays indicated in the Linde medical files that were examined, and delivered there.

(3) The purchase requisition stated the film was for "research on army contract #W-7405 Eng. 26." This contract supported research at Linde's Tonawanda Laboratory (Brimm and Schubert 1945b) suggesting that the film was for use in such research. It is unlikely that the laboratory had its own worker x-ray program as other correspondence indicates a common program for the Ceramics Plant and the Proving Laboratory, the portion of Tonawanda Laboratory involved in developing uranium processing techniques (Cranch 1943).

4. Period: November 16, 1942 is the date on which Linde entered into contract W-7401 eng-14 for its work for the MED (Section 2.1.2). It is likely that the medical examination requirements were not implemented until some time after this date, but exactly how long after is not known. A letter to the Linde Safety Steering Committee dated August 13, 1943 indicated that pre-employment exams were already a regular practice (Cranch 1943). The earliest document found that referred to the x-ray requirements was the report of the March 31, 1943 safety conference, but the program may have been established before that date. Production work at the Ceramics Plant is assumed to have ended on June 30, 1949; cleanup work is assumed to have ended on December 31, 1954. It is likely that employees who participated in the cleanup work were required to have chest x-rays. In the absence of specific information, it is assumed that other employees were also required to have chest x-rays.

5.1.2 Pelvis/Lumbar Spine X-rays

MED's Medical Section's September 4, 1943 letter to Linde recommended initial and annual pelvis x-rays for workers "who will be exposed to fluorine or fluorine compounds" (Van Horn 1943a). However, on January 11, 1944, the MED recommended that pelvis x-rays be omitted from both pre-employment and follow-up examinations (Warren 1944). To assess company practice, Linde medical records were checked. The review was limited in its scope because only a small portion of the medical records for the period were available — records of EEOICPA claimants and records in general Linde files because of other compensation claims that had been filed against the company. Eight records were found which showed x-rays during the period 9/4/43 – 1/11/44. The records indicated that all of the subjects had had chest x-rays, but only one had also had a pelvis x-ray (Linde Workmen's Compensation File 1950, p. 67). The name of the worker for whom the pelvis x-ray was indicated was found in a Ceramics Plant employee list dated April 1944 (Neumann 1944). The list indicated that the employee worked in Step II, an activity that did not involve handling of fluorine or fluorine compounds.

Additional information was found in Linde correspondence files. When the recommendation to end pelvis x-rays was relayed to the Ceramics Plant, the plant administration advised the corporate office "that we have never made any pelvic x-rays of any of the people working in the Ceramics Plant" (Holmes 1944a). The corporate office checked with the medical doctor in its Industrial Toxicology Department who had oversight over medical procedures at the Ceramics Plant. The doctor wrote

> *It is my understanding that no arrangement had as yet been made for the taking of pelvic x-rays of employees at the Ceramics plant. When this was discussed some time ago it was decided that in view of the possible fluorine exposure it might be advisable later, perhaps at the WCX plant, to include pelvic x-rays when our facilities would make this more feasible. For this reason, a space was provided in the form for "laboratory examinations." … The opinion at present is that there would be no need at the Ceramics Plant to include pelvic x-rays in the physical examinations. (Cranch 1944b)*

These records indicate that Linde did not institute routine pelvis x-ray examination of its employees in response to the 9/4/43 recommendation from the MED. The indication of a single pelvis x-ray found in the medical records may have been related to a special situation concerning the employee. While dose reconstruction for the particular employee should reflect what is in the employee's medical record, this particular record is not interpreted as evidence that Linde instituted a general policy of taking pelvis x-rays of a particular category of employees in response to the September 4, 1943 MED letter.

In the review of employee medical records, it was noticed that the record of a female employee had the notations "LMP 2/10/44" and "LMP 3/4/44" in the row on the report form reserved for recording pelvis x-ray results. Other pages of the employee's record had similar monthly entries, but they were in a section that began in the succeeding row of the report that was designated for recording results of "Other Tests." This was interpreted as indicating that the two entries on the pelvis x-ray row had been entered on that row in error. The entries were interpreted as recording the date of the last monthly period in compliance with suggestions made by the MED (Van Horn 1943b).

Pelvis x-rays of Linde employees were taken at a later time. The Ceramics Plant progress report for the week ending December 21, 1947 stated, "Complying with instructions received from the health authorities of the Atomic Energy Commission, pelvis X-rays are being taken of all Ceramics Plant employees. It is expected that this program will be completed before the end of the year" (Linde Ceramics Plant 1948). Employee medical records confirm that x-rays of the pelvic region were taken.

Images of x-ray films with the records indicate that 14"x17" films were taken and that the long direction of the film was parallel to the length of the body, an orientation like that used for standard lumbar spine x-rays and different from the 17"x14" orientation (14" parallel to the length of the body) use for standard pelvis x-rays. In each of the available files having copies of pelvis x-ray films, there were two images of the pelvic region. It was difficult to determine whether these were images of two portions of one x-ray exposure or whether two x-ray exposures had been taken at one session. To avoid underestimation of worker doses, for dose reconstruction it is assumed that two exposures were taken on each pelvis x-ray occasion.

In June 1949 as the plant was nearing shutdown, it wrote the AEC to ask whether a pelvis x-ray examination should be included in the termination medical exam of all workers or only of workers whether or not they had access to contaminated areas (Kent 1949). A statement of the decision in this matter was not found. Employee medical records indicated that x-rays of the pelvic region were taken upon termination (around the end of June 1949), but the number of records is not sufficient to indicate whether they were taken of all employees. The x-ray film images available had characteristics similar to those taken at the end of 1947.

Based on the foregoing, the assumptions below are provided for use in reconstruction of doses due to x-rays of the pelvic region.

Assumptions

1. Applicability: These parameters apply to all job categories at the Ceramics Plant.

2. Occasions: Pelvis X-rays were taken at the following times:

    a. Near the end of 1947: A pelvis x-ray was taken of every Ceramics Plant employee in November or December 1947.

    b. Upon employee termination: A pelvis x-ray was taken of every Ceramics Plant employee terminating after November or December 1947.

3. Number: Two exposures were taken at each occasion.

4. Methodology: A 14"x17" radiographic film was taken (17" parallel to the length of the body), a format similar to that for lumbar spine x-rays.

5.1.3 X-ray Dose Reconstruction Guidelines

Dose reconstruction should be based on information specific the subject to the extent that it is available and adequate. The guidelines in this section are for use when the records for an individual worker are not available or are incomplete. The guidelines are for use only to the extent that they are not inconsistent with the worker's records. For example, if the medical records are complete and indicate a lower or higher examination frequency than stated in the assumptions provided above, the data in the medical records should be used.

X-ray doses shall be determined in accordance with the latest revision of the project technical information bulletin "Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures" (current version is ORAU Team 2003) when applicable. Before using the guidelines therein, dose reconstructors shall confirm that the assumptions on which they are based are applicable.

In relation to the above, attention is called to the following:

- For many employees, medical records are available that show the dates and types of the x-rays taken.

- Often, images of the actual x-rays are also available in claimant files.

- Some chest x-rays in Linde claimant files show portions of the chin. This indicates that collimation was not as good as in current practice.

- X-rays of the pelvis thus far examined in Linde claimant files had the format of lumbar spine x-rays. The lumbar spine x-ray doses in the x-ray technical information bulletin might be more suitable for these cases than the pelvis x-ray doses.

5.2 TONAWANDA LABORATORY EMPLOYEES

The medical x-ray requirements for Ceramics Plant employees appear to have also applied to Tonawanda Laboratory employees who worked in pilot plant projects (Cranch 1943; Brimm 1943c). The extent to which the requirements were extended to other Tonawanda Laboratory employees is not known. In the absence of this information and in view of the likely difficulty of establishing whether or not a particular worker was involved in handling radioactive materials, it is assumed for dose reconstruction that chest x-rays were required of every Tonawanda Laboratory employee in the period of MED work unless it can be shown that the employee was not involved in MED projects.

The following assumptions are made regarding the frequency of x-rays of Tonawanda Laboratory employees:

1. Applicability: These parameters apply to all job categories at Tonawanda Laboratory.

2. X-ray Category and Frequency:

   a. Initial x-ray when the worker started working on an MED contract.

   b. Repeat x-rays taken at the same frequency as specified for Linde Ceramics.

   c. Termination x-ray taken upon termination of employee participation in MED/AEC work at Linde.

3. Methodology: Same as for Linde Ceramics.

4. Period: MED-related work at Tonawanda Laboratory is assumed to have started on October 1, 1942 and ended on December 31, 1946.

Dose reconstruction guidelines are the same as for Linde Ceramics employees.

6.0 ESTIMATION OF EXPOSURES FROM RESIDUAL CONTAMINATION AFTER 1954

This section develops parameters for reconstruction of doses due to internal and external exposures of Ceramics Plant and Tonawanda Laboratory workers after December 31, 1954, the assumed completion date of cleanup at the Ceramics Plant. Both facilities were on Linde's Tonawanda, New York site

Initial cleanup of Tonawanda Laboratory is assumed to have been completed on December 31, 1946. Tonawanda Laboratory workers' radiation exposures from January 1, 1947 to December 31, 1954 are discussed in Sections 3.0 and 4.0. The assumed Ceramics Plant initial cleanup date is December 31, 1954.

Beginning on January 1, 1955, It is assumed that Linde employees could have been exposed to residual contamination for 2000 hours per year.

6.1.1 Internal Exposure

Airborne Radioactivity

In recent years, outdoor air radioactivity has been measured in conjunction with the site soil remediation program (ACE Buffalo 2004b). Table 38 displays the highest monthly average air concentrations of Ra-226, Th-230 and U-238 for July 2000 through June 2004. Assuming that the measured U-238 is part of a natural uranium source term, the total uranium concentration would be the U-238 concentration divided by the fraction of activity due to U-238, 0.4886, or 4.3E-04 pCi/m^3. The activity ratios of Th-230 and Ra-226 to uranium are 0.84 and 1.7, respectively.

Table 38. Highest monthly outdoor airborne radionuclide concentrations at Linde Tonawanda site, 2000-2004.

Nuclide	Highest monthly concentration (pCi/m^3)[a]
Ra-226	7.5E-04
Th-230	3.6E-04
U-238	2.1E-04

a. Per analysis of data from ACE Buffalo 2004b.

Building 30 was found in 1976 to be the most contaminated building on the site (ORNL 1978). The indoor airborne uranium concentration in 1976 was measured as 1.90E-02 pCi U/m^3 (ORNL 1978, Table 14). This larger uranium air concentration is used to estimate a chronic intake. To maintain the consistent intake units and estimate the annual intake, the uranium air concentration of 1.90E-02 pCi/m^3 is multiplied by 2.22 dpm/pCi, a breathing rate of 1.2 m^3/h and 2000 work-hours/y. The annual intake rate is estimated at 100 dpm/y for all Tonawanda employees beginning in 1955. Dividing by 365 days/y give a daily uranium inhalation intake rate of 0.277 dpm/day. The Th-230 and Ra-226 daily inhalation intake rates would be 0.233 and 0.471 dpm/d, respectively. Ingestion intake rates are estimated using the steps described in Section 3.7. The summarized intake results are shown in Table 39. It is unlikely that uranium would be in a chemical form consistent with type F absorption during the residual contamination period, so it is assumed that only types M and S would be inhaled after 1954. Th-230 could be type M or S, and Ra-226 is type M.

Workers noted that Building 30 renovations occurred in the 1960s that could have influenced air concentrations. Specific details of the renovation, including the actual period of renovation, dust control measures, location of work and occupancy of areas are not available. It is reasonable to assume that renovations could have resulted in elevated airborne radioactivity. Because a maximum value of measured air concentration is used to estimate the intakes during the residual exposure period, a factor of 14 is assumed to be reasonable to describe the uncertainty at the 95th percentile associated with the possibility of elevated intakes during Building 30 renovations. So for organ doses, the distribution is assumed to be lognormal with a GSD of 5.

Other progeny are partially accounted for by the assigned radon exposure.

Radon Daughters
Fifty-five measurements of radon daughter concentration were made in 1976 and 1981 in the Tonawanda site buildings used in MED/AEC work (ORNL 1978, pp. 17 and 84; BNI 1982, p. B-24). The 1981 survey was more comprehensive and yielded significantly higher concentrations, so the 1976 results were ignored. Building 31 had the highest radon daughter concentration. The geometric mean PAEC was 1.68E-02 WL with a GSD of 1.89 (based on analysis of data in BNI 1982, Table B-3). No correction was made for natural background radon, as its value was unknown. Based on 12 work-months per year, the radon exposure is estimated as 2.01E-01 WLM/y, and is listed in summary Table 39.

6.1.2 External Beta and Gamma Exposure

As noted, Building 30 was found in 1976 to be the most contaminated building on the site (ORNL 1978). Floor and wall radiation levels measured in Building 30 in 1976 were compared with similar measurements made in 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 one meter 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/h (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 non-uniform and varying from background up to 250 µR/h. Regions with readings above about 20 µR/h 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/h. BNI 1982 (p. B-12) stated that the points of maximum radiation determined in the ground surveys were slightly displaced from but in general agreement with those determined in a 1979 aerial radiological survey. The total number of readings ≥25 µR/h reported by BNI was 16. The net readings (after subtraction of 8 µR/h to correct for background) had a geometric mean of 94.0 µR/h and a GSD of 3.95. This was taken as an estimate of worker exposure rate when outdoors. This estimate was assumed to apply from January 1, 1955 to the present (2005). No credit was taken for the soil remediation at the site that began in 2000, as the remediation has not been completed (Pilon 2004). The estimate is probably an overestimate for the following reasons: measurements were reported only for areas 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’ above the ground corresponding to the adopted value of gamma exposure rate was estimated as 4.38E-01 mrem/h, 4.66 times the gamma rate, where 4.66 is the ratio of beta mrem/h at 3’ to gamma mR/h at 3’ for the floors and walls in Building 30 after decontamination (per Table 13). 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 post-cleanup exposures, based on Building 30 contamination levels, were used to estimate the external exposure rate. No adjustments for changes in work-hours were made. Table 39 summarizes the results. The radiation energy distributions are assumed to be same as those during the operational period.

6.2 NEUTRON EXPOSURE

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

6.3 DOSE RECONSTRUCTION SUMMARY, 1955 TO PRESENT

This section summarizes guidelines and parameters for reconstruction of doses due to internal and external exposure beginning in 1955.

Table 39. Annual internal and external exposure to residual radioactivity.

Internal	Start[b]	Exposure	Absorption type	Intake (dpm/d)	IREP distribution
U-234	1/1/1955	Inhalation	M, S	2.77E-01	Lognormal GSD 5
	1/1/1955	Ingestion	(a)	5.78E-03	Lognormal GSD 5
Th-230	1/1/1955	Inhalation	M, S	2.32E-01	Lognormal GSD 5
	1/1/1955	Ingestion	(a)	4.84E-03	Lognormal GSD 5
Ra-226	1/1/1955	Inhalation	M	4.84E-01	Lognormal GSD 5
	1/1/1955	Ingestion	(a)	1.01E-02	Lognormal GSD 5
				WLM/y	
Rn-222	1/1/1955	Inhalation	-	2.01E-01	Lognormal GSD 1.89
External	**Start**	**Exposure**	**Basis**		
	1/1/1955	Penetrating	Survey instrument	0.068 R/y	Lognormal GSD 3
	1/1/1955	Non-penetrating	Survey instrument	0.326 rem/y	Lognormal GSD 3

a. Choose same f_1-value as used for inhalation per NIOSH (2004).
b. No end date is set, because there is no evidence that the site is restricted or has been remediated.

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GLOSSARY

carnotite

$K_2(UO_2)_2(VO_4)_2 \cdot 3(H_2O)$; a mineral found in Colorado and Utah that was mined for its uranium and vanadium content. [Much of the domestic ore processed at Linde Ceramics derived from tailings from carnotite ore. Preprocessing of this material prior to its shipment to Linde is believed to have removed much of its radium content (Aerospace Corporation 1981).]

day

As used in this report 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 refers to a day that is assigned for work. The length of a workday depends on the amount of time spent on the job. A default assumption is 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 larger than the default for the early years.

Linde database

Records of Linde operations that were included in NIOSH computer-stored data and were available as this document was being prepared.

N. G. (or NG) Cake

Filter cake produced in Step II processing at Linde. Two types of cakes were produced; they were called "OK" and "NG."

pitchblende

A mineral containing uranium oxide of variable composition ranging between UO_2 and U_3O_8. (Pitchblende ore from Africa was processed at Linde Ceramics for its uranium content. It was not pre-refined before its arrival at Linde and so contained all of the natural uranium radioactive progeny including radium.)

Step I, Step II and Step III

The three uranium production processes at Linde.

Step I, conversion from ore to U3O8 (black oxide)

Step II, conversion of U3O8 to UO3 (orange oxide) to UO2 (brown oxide)

Step III, conversion of UO2 to UF4 (green salts)

torbernite

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

ATTACHMENT A
BETA RADIATION

LIST OF TABLES

This attachment contains data 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/h[a,b]	rad/8 h
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)	Electron energy per decay of parent (MeV/nt)	% 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	

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

	Principal nuclides	Electron energy per decay of nuclide (MeV/nt)	Electron energy per decay of parent (MeV/nt)	% 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	

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 by-products[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	—[f]	0.012
12.6[e]	—[f]	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 cm. 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^{-1} was estimated from the rule of thumb that the half thickness of a beta absorber is 1/8 of the range of the beta rays (Cember 1983, p. 97). For natural uranium ore and by-products containing beta emitters other than Bi-214, the actual falloff with distance would be faster 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. Range of 3.27 MeV beta radiation per Cember 1983, p. 100.
f. No radiation at this distance.

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.

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

	(mR/h)[a]						
Operation	**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]	
	RL relative to chest[c]						**RL rel 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 1948a.
b. No data.
c. RL = radiation level. 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 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 may have had multiple meanings. The table below lists apparent definitions deduced during preparation of this document.

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 – 16.5% U_3O_8 (Aerospace 1981, Table B-1)
L-30	African pitchblende ore; estimated to contain 8 – 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