



ORAU TEAM Dose Reconstruction Project for NIOSH

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ANL	Argonne National Laboratory
AQL	Acceptance Quality Level
AWE	atomic weapons employer
cGy	centigray
Ci	curie
cm	centimeter
d	day
D&D	decontamination and decommissioning
DCF	dose conversion factor
DOE	U.S. Department of Energy
dpm	disintegrations per minute
DWE	daily weighted (average) exposure
ECI	Export Controlled Information
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
F ₂	fluorine gas
ft	foot
FUSRAP	Formerly Utilized Sites Remedial Action Program
g	gram
gal	gallon
GI	gastrointestinal
GSD	geometric standard deviation
H ₂	hydrogen gas
HASL	Health and Safety Laboratory
hr	hour
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
keV	kiloelectron-volt, 1,000 electron-volts
KOH	potassium hydroxide
kVp	applied kilovoltage; peak kilovoltage
L	liter
lb	pound
LOD	limit of detection
m	meter
mCi	millicurie
MCW	Mallinckrodt Chemical Works
MED	Manhattan Engineer District
MeV	megavolt-electron, 1 million electron-volts
mi	mile

min	minute
mm	millimeter
mo	month
mR	milliroentgen
mrem	millirem
mrep	millirep
NG	Not Good, aqueous phase of the ether extraction [liquor] that contained a fair percentage of uranium
NIOSH	National Institute for Occupational Safety and Health
NYOO	New York Operations Office (AEC)
oz	ounce
PA	posterior-anterior
pCi	picocurie
PER	Program Evaluation Report
ppb	parts per billion
Q	quarter (used to indicate portion of calendar year in several tables)
qt	quart
R&D	research and development
RU	recycled uranium
s	second
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
TBP	tributyl phosphate
U.S.C.	United States Code
UNH	uranyl nitrate hexahydrate
WL	working level
WLM	working level month
yr	year
µg	microgram
µm	micrometer
§	section or sections

1.0 INTRODUCTION

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

In this document, the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer [AWE] facility" or a "Department of Energy [DOE] facility" as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 [EEOICPA; 42 U.S.C. § 7384l (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 during which 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 occupationally derived radiation exposures at the facility must be included in dose reconstructions. NIOSH does not consider the following exposures to be occupationally derived:

- Radiation from naturally occurring radon present in conventional structures
- Radiation from diagnostic X-rays received in the treatment of work-related injuries

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- or U.S. Atomic Energy Commission (AEC)-related work) must be included in dose reconstructions. Radiation dose received from DOE/AEC-related work includes (1) radiation from radon consistent with NIOSH's policies for including such radiation in the contract period and (2) medical screening X-rays (but not diagnostic X-rays for the treatment of work-related injuries). It should be noted that (1) under subparagraph A of 42 U.S.C. § 7384n(c)(4), radiation associated with the Naval Nuclear Propulsion Program is specifically excluded from the employee's radiation dose; and (2) under subparagraph B of this section, radiation from a source not covered by subparagraph A that cannot be reliably distinguished from radiation that is covered by subparagraph A is considered part of the employee's radiation dose. This site profile covers only exposures resulting from nuclear weapons-related work. Exposures resulting from non-weapons-related work, if applicable, will be covered elsewhere.

1.1 PURPOSE

This document establishes the technical basis for the reconstruction of radiation doses to workers at the Harshaw Chemical Company's Harvard-Denison Plant at 1000 Harvard Avenue in Cleveland, Ohio, which received feed materials from uranium mills throughout the United States and Canada (DOE 1984) and refined it to produce various uranium compounds under contract to the U.S. Government from 1942 to 1955.

A Special Exposure Cohort (SEC) class established for Harshaw includes all AWE employees who were monitored or should have been monitored while working at the Harshaw Harvard-Denison Plant for a number of workdays aggregating at least 250 workdays from August 14, 1942, through

November 30, 1949, or in combination with workdays within the parameters established for one or more other classes of employees in the SEC (Elliott 2007). This document also provides guidance for EEOICPA-covered employees who participated in Harshaw operations, specifically for non-SEC cancers and those presumptive cancer claims for workers who have fewer than 250 workdays under this employment or in combination with workdays within the parameters established for other classes of employees in the SEC.

1.2 SCOPE

This document covers workers at the Harshaw Harvard-Denison Plant at 1000 Harvard Avenue in Cleveland, Ohio, principally in Plant C and several additional buildings listed in Table B-2 (e.g., Foundry, K-1, and an Annex facility), which were associated with the chemical conversion and production of various uranium compounds. Some early uranium tetrachloride (UCl_4) laboratory-scale work appears to have been done at a Harshaw laboratory at 1945 East 97th Street (Gamertsfelder ca. 1944), but little else is known about this facility. (Therefore, while the known exposure rate and position information are included in Table B-5, more investigation could be needed about the location and use of this facility and it is not included in the scope of this site profile.) In addition, a site radiological survey conducted in the 1980s (DOE 1984) indicated that radiological contamination was detected inside and outside several other structures at the main plant site.

This document covers the period from the start of contract operations for the AEC and its predecessor agency [the Manhattan Engineer District (MED)] through the cessation of operations. In addition, this document discusses the periods of decontamination and decommissioning (D&D) for each operational area and the period from AEC release of the site to the point where the Formerly Utilized Sites Remedial Action Program (FUSRAP) took over decontamination responsibilities for the parts of the site at which eligible operations had been performed. Table B-1 lists a detailed site chronology. In general, the period of production operations was from January 1942 to about September 1955. However, the period of AWE operations started on August 14, 1942, with the establishment of the MED. The period of D&D under AEC supervision was from November 1951 to an unspecified point in 1960 when Harshaw received an unrestricted release from AEC (OEPA 2001). The period of free release and non-AEC use until FUSRAP took over is considered to be from 1960 until June 1999 when DOE (which was then in charge of FUSRAP) issued a letter stating that Harshaw was a candidate site for remediation and that it would support preliminary characterization by providing documentation (Fiore 1999). The latter period includes the time after the purchase of the site by the Engelhard Corporation in 1988. Dose reconstructions could be needed from August 14, 1942, to the present.

This site profile consists of six sections: (1) Introduction, (2) Site Description, (3) Occupational Medical Dose, (4) Occupational Environmental Dose, (5) Occupational Internal Dose, and (6) Occupational External Dose. Attachment A describes the assessment of internal coworker data from Harshaw. Attachment B contains the tables important to dose reconstruction.

2.0 SITE DESCRIPTION

2.1 INTRODUCTION

Table B-1 lists the chronology of site use. Figure 2-1 shows the buildings that existed at the Harshaw site during a radiological survey in 1984 (DOE 1984). Layouts of the uranium tetrafluoride (UF_4) and uranium hexafluoride (UF_6) areas can be found in AEC (1948, 1951a) and that of the ore concentrate to uranium dioxide (UO_2) area in AEC (1951b).

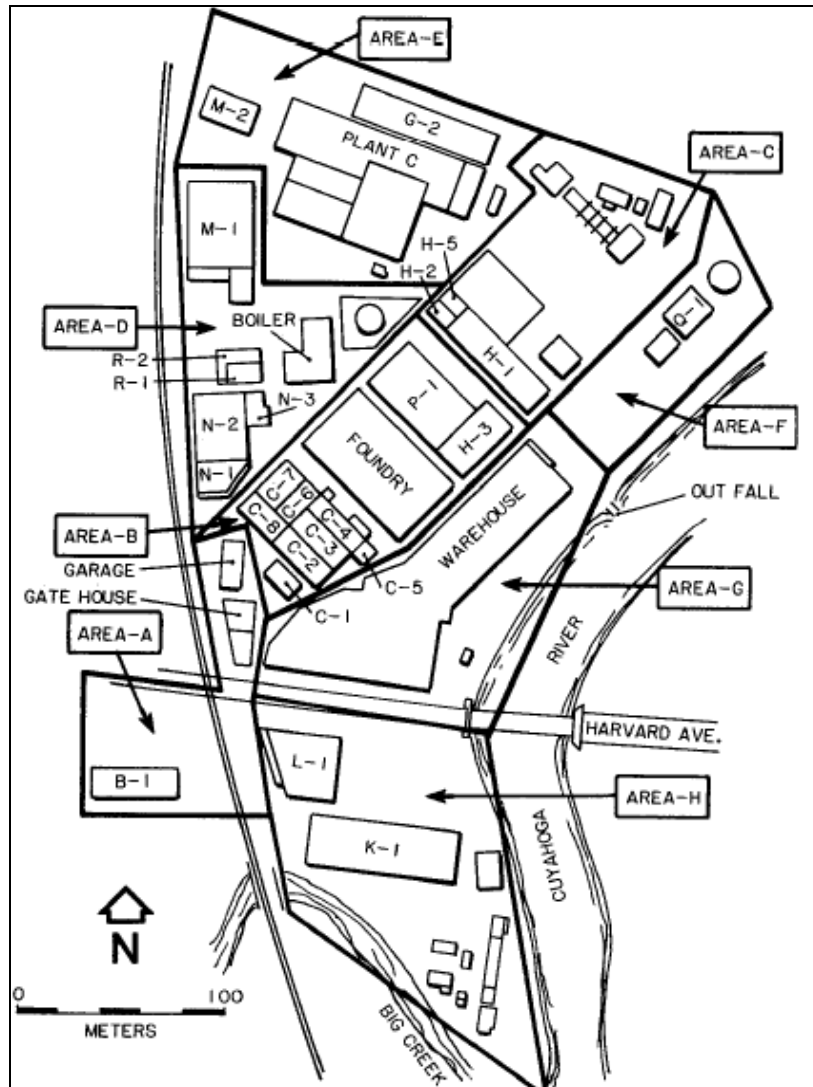


Figure 2-1. Harshaw Chemical Company identifying area designations during FUSRAP Site Survey (DOE 1984).

Little information about the early history of site use is available. The principal source of information is an account by the president of Harshaw of the early work under contracts with the MED or other entities that were working for the MED (Harshaw 1945). Harshaw was first approached in 1941 about Office of Scientific Research and Development/MED-associated work when it was asked to produce UF_6 using fluorine gas (F_2) based on the company's experience manufacturing anhydrous hydrofluoric acid and laboratory experience in manufacturing F_2 (Harshaw 1945). Harshaw had limited experience working with uranium, having manufactured uranium-containing ceramic glazes and similar products.

2.2 SITE ACTIVITIES

The entire Harshaw Chemical Company operational area for AEC work was referred to as "Plant C" or "Area C" (Velten 1949). Plant C (refinery building also referred to as Building G-1) included all of the individual production operations using radiological materials (UO_2 -to- UF_4 , UF_4 -to- UF_6 plus distillation, and ore-to- UO_2 or - UO_3). The locations where these three individual production operations were performed were also referred to as "plants." The common usage of the time might have referred to the ore-to- UO_2 operations as the "brown," "new brown," or "493" plant; the UF_4 -to- UF_6 operations as

the "hex" or "492" plant; and the UO_2 -to- UF_4 operations as the "green" or "491" plant. In 1945, UF_6 operations were expanded in an annex to Building G-1, called the "hex annex."

Uranium production operations for the MED war effort began in August 1942 when Harshaw produced laboratory quantities of UCl_4 . This work would have been done using ore material. Generally speaking, this material would have been in the form of triuranium octaoxide (U_3O_8), sometimes referred to as black oxide or yellow cake. Harshaw also produced UCl_4 for the National Bureau of Standards. After 1943, other research contracts followed, including a contract to research methods of improving uranium capture and recovery (Harshaw 1945).

From March 1942 until 1951, Harshaw produced UF_6 (DOE 1997; Stefanec 1951). In 1944, under contract to the MED, Harshaw built and operated a full-scale UF_6 pilot plant (Harshaw 1945).

In May 1949, the AEC contracted with Harshaw to convert uranium concentrate to uranium trioxide (UO_3) (also called orange oxide) and then to UO_2 (NYOO Medical Division 1949; Mayer and Proschan 1949). In 1951, AEC discontinued conversion of UO_3 to UO_2 and production of UO_2 ceased. At that point, Harshaw began to produce UO_3 only from milled ore (DOE 1997). Production of UO_3 continued until about September 1955, when the plant was shut down.

D&D activities were conducted under AEC supervision starting in November 1951. Decontamination of the UF_4 and UF_6 areas began after early December 1951 (Clarke 1963). Dismantling of equipment in the UF_6 area and probably the UF_4 area began in January 1952 (HCC 1950–1953). In about February 1952, an area of the Building G-1 annex, which had held an extension of the UF_6 production area and also housed some offices and shops, was torn out to make a warehouse, forcing the relocation of various offices and shops to other existing areas (Klevin 1952a). AEC equipment was removed by September 30, 1955 (NIOSH 2002). AEC D&D surveys of the various production areas were performed since November 1951 (Klevin 1953a). The buildings were decontaminated by Harshaw in the late 1950s and released from AEC control in 1960 (FUSRAP 2001).

Although Building G-1 and its annex were subjected to D&D from 1952 to 1959 for the older areas and 1956 to 1959 for the newer areas (the brown area), it appears that other buildings that had some residual contamination – either from work there in the MED years, such as the foundry and Building K-1, or from track-in, such as the Boiler House – were not subject to D&D in these time periods. After release of the site by the AEC in 1960, Harshaw used all the buildings until the site was purchased by the Engelhard Corporation in 1988. Engelhard then did D&D between 1990 and 1997 on all buildings except (apparently) Plant C, but did not seem to have used Plant C. Thus, the postoperations years would be from 1960 to 1989 for all buildings and the Engelhard D&D years would be 1990 to 1992 for those buildings that were demolished (Buildings K-1, M-1, and P-1); 1990 to 1992 for the Boiler House (used to store parts of the demolished buildings); and 1990 to 1997 for the remaining buildings other than Plant C. Table 2-1 lists the assumed periods of D&D and postoperations use. Note that it is assumed that those buildings that were not in use were kept locked and access to them was controlled, as is suggested or stated explicitly by all relevant references.

The D&D of the green area, the "old hex" area, and the distillation and recovery areas that took place from 1952 to 1959 was not continuous [1]. However, it appears that no operations were performed there because the AEC did not release the site until 1960, and it should thus be assumed that this period was entirely a D&D period for these areas. The D&D of the "new hex" or annex area that took place in 1952 left some residual contamination because the space was to be converted to other uses associated with the ore concentrate to UO_3 process. This work was true D&D work and the workers engaged in it likely had a notation in film badge records such as "decontamination," but the space was not completely clean afterward and was used for process-associated work such as warehousing and

Table 2-1. Assumed periods of D&D and postoperations use.

Year	UF ₄ area	Old UF ₆ area	Recovery /still area	New UF ₆ /UO ₃ area	UO ₃ -Only area	Lab	Boiler house	Foundry	Garage	Warehouse	P-1	K-1	M-1
1942-1951	3Q 1944-3Q 1951	3Q 1944-4Q 1951	3Q 1944-4Q 1951	3Q 1944-4Q 1951	2Q 1949-	3Q 1944-	Start of use?	1942-?	Start of use?	Converted from New Hex, 1952?	Start of use?	Start of use?	Start of use?
1952	1Q	1Q	1Q	1Q-3Q									
1953-1954													
1955				3Q1955	3Q 1955	3Q 1955							
1956				4Q	4Q	4Q?							
1957-1959													
1960-1989													
1990-1992							1Q	1Q	1Q	1Q	1Q	1Q	1Q
1993-1997													
1998													

Shading legend:

In use
 D&D period
 Nonuse (locked)
 Demolished

Q means calendar quarter

storage of contaminated items. Thus, while the post-D&D exposure levels in this area probably did not approach the exposure levels of the process area, there was still the potential for non-negligible exposure [2].

The period of free release and non-AEC use until FUSRAP took over is considered to be from 1960 until June 1999. FUSRAP work at the site is not included in the scope of this site profile. Table 2-2 provides a summary of the operations, postoperations, and D&D history to clarify the application of calculated dose estimates.

Table 2-2. Operations, postoperations, and D&D activity.

Area	Operations	AEC decontamination	Use after AEC operations and/or D&D	Final (Engelhard) decontamination	Use after final decontamination
UO ₂ -UF ₄ (491, Green)	1944-1951	1952-1959	1960-1989	---(a)	---
UF ₄ -UF ₆ (492, Old hex)	1944-1951	1952-1959	1960-1989	---	---
Recovery/Still (492, Distillation)	1944-1951	1952-1959	1960-1989	---	---
UF ₄ -UF ₆ annex (492, New hex)		1952-1959	1960-1989	---	---
UF ₆ production	1948-1951	1952	---	---	---
Storage for UO ₃ production	1953-1955	1956-1959	1960-1989	---	---
Laboratory	1944-1955	1956-1959	1960-1989	---	---
Boiler House	1942?-1955	---	1956-1989	1990-1997	---
Foundry	1942-1955	---	1956-1989	1990-1997	1998 on
Garage	1942?-1955	---	1956-1989	1990-1997	1998 on
Warehouse ^b	1953-1955	---	1956-1989	1990-1997	1998 on
Building K-1	1942-1955	---	1956-1989	1990-1992	---
Building M-1	1942?-1955	---	1956-1989	1990-1992	---
Building P-1	1942?-1955	---	1956-1989	1990-1992	---

a. Table entries that are blank (---) represent periods of inactivity.

b. Assumed to be different from the New hex annex, which was partly converted to extra warehouse space in 1952.

Table B-21 gives information about the number of workers based on AEC dust study reports. Job titles should be listed in employment records for individual workers, but these might not correspond to job titles or work descriptions in the badge and bioassay records because informal terms or area references were often used in the latter. Table B-4 (keywords and codes used) and Table B-22 (job titles and functions) should be used to help establish the actual category to be used for an individual worker with respect to later tables in this site profile and with respect to interpreting records.

Most of the Harshaw uranium processing operations were carried out 24 hr/d, 7 d/wk (Lippmann 1958). AEC dust study reports [e.g., Klevin (1950a) and Lippmann (1958)] indicate that the workday for all workers was 8 hours. Klevin (1950a) indicated that there were three shifts per day for every type of listed worker except the plant superintendent, general foreman, process engineer, health physicist, and laundry workers. The Harshaw operating manual (HCC 1946) stated that the Timken Roller Bearing Company plan of rotation was used: there were four crews for each major process or activity, with each crew working five 8-hour days for 3 weeks, then working six 8-hour days the fourth week, for an average of 42 hr/wk. This was necessary so the plant could operate 24 hr/d, 7 d/wk.

In addition to Harshaw employees, employees from at least two subcontractors were mentioned in site documents, M. K. Ferguson (Klevin ca. 1948) and General Welding (AEC 1950a; HCC 1950–1953).

2.3 SITE PROCESSES

2.3.1 Uranium Tetrachloride Production

Harshaw shipped its first order of UCl_4 to the National Bureau of Standards in March 1942. Harshaw began larger scale laboratory production of UCl_4 in November 1942; by April 1943, Harshaw was producing up to 100 lb of UCl_4 daily (Harshaw 1945). Harshaw set up a new production area in October 1944. This new production area was initially used to process up to 1 ton per day of UCl_4 , with the production rate continuing to increase monthly thereafter (Harshaw 1945; Ferry 1944a; MED 1945). In January 1945, MED ordered an additional amount of 65,000 lb of UCl_4 . This was the final order of the material (Simmons 1945). Harshaw stopped production of UCl_4 in February 1945 and dismantled the UCl_4 production area. Parts were shipped to Oak Ridge, Tennessee (Harshaw 1945).

2.3.2 Uranium Hexafluoride Production

In February 1942, Harshaw Chemical Company first produced UF_6 and maintained a production rate of 5 lb/d throughout 1942. This material was sometimes referred to as “hex” and the processing area as the “hex” or “492” plant.” By 1943, Harshaw was producing as much as 50 lb of UF_6 per day in a pilot plant. Harshaw operated the pilot plant until February 1944, producing a total of 9,000 lb of UF_6 (Harshaw 1945).

In 1944, Harshaw built a new UF_6 production facility containing three units. Harshaw erected electrolytic cells to obtain the fluorine that was used to produce UF_6 (Quigley 1951a). By July 1944, Harshaw was producing as much as 3,300 lb of UF_6 per day, and as much as 4,500 lb/d of UF_6 by April 1945 (AEC 1951c; Harshaw 1945). An auxiliary building known as the “hex annex” or “new hex” area was added in 1945 to provide additional UF_6 production capacity. The hex annex was mentioned in some references as an addition to Building G-1 and in others as a separate structure. Although this level was not reached on a regular monthly basis, by December 1947, Harshaw was producing up to 46,000 lb of UF_6 per month. In December 1951, Harshaw Chemical Company stopped producing UF_6 (Sargent 1951).

2.3.3 Uranium Tetrafluoride Production

In 1942, at the request of Standard Oil, Harshaw Chemical Company began to produce UF_4 from UO_2 . This material was sometimes referred to as “green salt” and the processing area as the “green” or “491” plant. The first UO_2 material processed at Harshaw was supplied by Westinghouse Electric and Manufacturing Company (HCC ca. 1945). In July 1942, the MED asked Harshaw to produce

1,200 lb of UF₄ per day from UO₂ produced and supplied by DuPont and Mallinckrodt (Harshaw 1945).

In September 1942, Harshaw implemented large-scale production using a new facility with a production capacity of 50,000 lb (25 tons) of UF₄ per month (HCC ca. 1945; AEC 1951c). By December 1943, continued improvements in the conversion process increased the production level to 60 tons per month (Harshaw 1945; AEC 1951c).

In December 1944, Harshaw moved production of UF₄ to Building G-1, where the production rate was about 3,000 lb/d (HCC ca. 1945). In February 1946, anticipating a later full production of 28,000 lb (14 tons) per week, the MED authorized an increase to 15,000 lb/wk.

Although discrepancies in production quantities exist and are likely due to the reporting of theoretical capacity versus actual production, the final full production level for UF₄ (in February 1948) appears to have been 81 tons/mo (AEC 1951c). In October 1951, Harshaw stopped producing UF₄ (DOE 1997).

2.3.4 Uranium Trioxide and Uranium Dioxide Production

In 1947, Harshaw Chemical Company constructed an ore-to-UO₃-to-UO₂ batch production facility for AEC use (Velten 1949). This facility was constructed so Harshaw could produce UO₂ on the site, alleviating the need to bring in UO₂ from other suppliers. "Ore" was typically received as milled ore.

By July 1949, the ore-to-UO₃-to-UO₂ batch production facility was operating (AEC 1951c). In 1951, through process modification that included a switch to the use of tributyl phosphate-kerosene rather than ether, UO₂ production increased (AEC 1951c). However, the UO₃-to-UO₂ portion of the operation stopped entirely in 1951 although the AEC implies this occurred as of October 1952 (DOE 1997; Stefanec 1951; Termini 1952).

The UO₃ contractual amount was 200,000 lb/mo from October to December 1952 (Neumann 1952). Harshaw continued to produce UO₃ from ore until August 1953, when UO₃ production was placed on standby, and the AEC directed Harshaw to end all processing except for a final conversion of all leftover feed materials to UO₃ (Neumann 1953).

2.3.5 Uranyl Nitrate Hexahydrate Production

Throughout the 1950-through-1951 timeframe, uranyl nitrate hexahydrate (UNH) [chemical formula, UO₂(NO₃)₂·6 H₂O], which is an intermediate liquid produced in the initial processing of ore and uranium extraction, was reportedly produced as "research material." Documentation available to NIOSH does not indicate if the UNH was produced for use at Harshaw or elsewhere. Beginning in 1952, Hanford sent UNH to Harshaw, sometimes via the Brush Beryllium Company, to be converted into UO₃ (Klevin 1952b; Termini 1952; DOE 2000). Hanford produced UNH using a tributyl phosphate chemical process and transported it in tank cars to Harshaw.

2.3.6 Operations Involving Other Radiological Materials

Between 1943 and 1944, Harshaw Chemical Company manufactured a number of special radiological materials, including uranium oxyfluoride (UO₂F₂), sodium uranate (Na₂UO₇) at 84%, and uranium nitrate [U(NO₃)₂] at 56% (presumably the percentages were of U₃O₈ equivalent) (Harshaw 1945). However, NIOSH has not located documentation describing how Harshaw processed these materials. Between February 1947 and August 1950, Harshaw prepared short-lived ²³⁴Th (known as UX1) from

a residue of the UF₄-to-UF₆ conversion process (Stefanec 1951). Thorium-234 was produced in a laboratory in bench quantities (Stefanec 1951).

On at least two occasions, Harshaw Chemical Company processed some low-enriched uranium, in the form of UF₆, from Hanford (Kelley 1946). In 1945 and 1946, Harshaw was asked to mix natural UF₆ with UF₆ that had been slightly enriched. The resulting slightly enriched UF₆, now referred to as low-enriched UF₆, appears to have been enriched to less than 1% ²³⁵U by weight and was shipped to K-25 in Oak Ridge, Tennessee (Kelley 1946). The specific activity of 1% enriched uranium is 0.783 pCi/μg.

In 1952 and 1953, various shipments of UO₃ were sent from Hanford to Harshaw for purification (Klevin 1952b; BJC and Haselwood Enterprises 2000), which began on 1 October 1952 (Klevin 1952b). This was the so-called Redox material, which was made from recycled uranium (RU) recovered from irradiated uranium fuel at Hanford. At first, Hanford had trouble producing from this RU material enough UO₃ of adequate purity for refining into UF₆ (BJC and Haselwood Enterprises 2000; DOE 2000), mainly due to the difficulties in removing transuranic elements (such as plutonium and neptunium) and fission products, principally ⁹⁹Tc (BJC and Haselwood Enterprises 2000), and hence it was shipped to Harshaw. Klevin (1952b) states that the RU UO₃ was shipped in 30-gal drums and that it had been produced based on depleted uranium. The AEC further indicated that the material contained “practically all the elements in the periodic table,” as well as, 9 ppb Pu and 0.64% of ²³⁵U (Klevin 1952b, BJC and Haselwood Enterprises 2000). The purified product was sent to K-25 by way of Oak Ridge National Laboratory.

UNH was also sent, beginning in 1952, from Hanford to Harshaw (at least sometimes via the Brush Beryllium Company) to be converted into UO₃ (Termini 1952; Klevin 1952b; DOE 2000). The UNH sent from Hanford in 1952 and 1953 appears to have been RU like the UO₃ shipped from Hanford at about the same time.

To estimate the activity fractions of RU at Harshaw during the period from July 1, 1952, to June 1954, the maximum radionuclide mass fractions were used with an assumption of specific activity for depleted uranium of 0.4 pCi/μg [3]. Table 2-3 lists the results. These fractions will overestimate the activity of RU in the source term for most exposure scenarios [4].

Table 2-3. Assumed fractions of recycled uranium contaminants at metal handling facilities.

Recycled uranium contaminant	Pu-239	Np-237	Tc-99
Contaminant, ppb of uranium	30	780	12,000
Activity fraction of contaminant in uranium	0.00464	0.00137	0.506

2.4 INCIDENTS

In early 1950, one worker in the green plant and five or six workers in the hex plant showed abnormal urine readings (high albumin, variously called albumen-urea, albuminaria, and albuminuria in AEC references) (AEC 1950b; Lippmann 1958). At least one worker was immediately removed from uranium work (AEC 1950b; Lippmann 1958) and was sent to the hospital at Brookhaven National Laboratory for 10 days of study in August and September 1950 (AEC 1950b; Quigley 1950). In October 1950, an AEC New York Operations Office (NYOO) doctor met with Harshaw to discuss these cases (Sargent 1950a, which lists the workers and their hire dates, work locations, and job titles). He emphasized to Harshaw that seven confirmed cases (of 11 reported) for 200 workers was far above normal and that tests had eliminated all causes except uranium damage to the kidneys. He pointed out that the condition had not occurred in those known to be most heavily exposed, so it could

be that the affected workers had a special susceptibility to uranium, or that short intense exposures were not being picked up by the existing exposure evaluations. A urologist at Deaconess Hospital was unable to find any reason for the condition of the two workers tested there other than their uranium exposure (Quigley 1950). A follow-up study initiated by the AEC showed that for those followed up, urinary findings were normal within a few months (Lippmann 1958).

2.5 SUMMARY OF POTENTIAL EXPOSURES

Harshaw AWE employees could have received internal and external radiation exposures from uranium and nonuranium contaminants, including radium, and thorium in the milled and composite materials. The uranium content of mined uranium ores varied based on the quality of the rock being mined. Natural conditions resulted in varying degrees of disequilibrium between decay series radionuclides within particular ore deposits. A majority of the ^{226}Ra and thorium isotopes would have been removed by the milling processes.

While the activity of the ^{226}Ra and thorium isotopes was reduced, much of the potential for external exposure to AWE workers at Harshaw was likely due to uranium progeny. Radium-226, a gamma emitter, likely produced some of the external whole-body dose received by the Harshaw workers. Thorium-234 and $^{234\text{m}}\text{Pa}$, both primarily beta emitters, likely produced whole-body skin and extremity dose for workers involved in handling the fluorination ash or decontaminating equipment used to contain or transport the bed ash (ORAUT 2006a).

Internal exposures would have included alpha radiation resulting from uranium and uranium progeny emissions [5]. Workers involved in the UX-1 operation were potentially exposed to alpha emissions from thorium. AWE workers were also likely exposed to elevated levels of radon. The concentration of radium (and radon) and other progeny present in the ore concentrates, processed uranium, and processing residue at any given time depended on various factors, including the concentration of uranium in the original ore body; how much uranium progeny remained in the U_3O_8 product received from the mill, the total amount of U_3O_8 product processed, and how long the U_3O_8 was stored prior to use as feed at Harshaw [6].

Little monitoring information is available on the radiological aspects of the D&D work. No records of how the workers were monitored appear to have been made, other than that the monitoring was typical of normal operations and would appear among the regular operations records, perhaps with the notation "decontamination."

3.0 OCCUPATIONAL MEDICAL DOSE

3.1 INTRODUCTION

Harshaw uranium processing workers were given a preemployment physical that included a chest X-ray (Lippmann 1958). They were also given an annual physical that included a chest X-ray (Rauch 1948; Ferry 1944a). There might have been a hiatus during which annual chest X-rays were not given. In March 1947, the AEC specifically approved annual X-rays (Howland 1947a). However, this might have represented the AEC agreeing to assume the expense of the annual X-rays rather than Harshaw and so might not indicate an interruption of annual chest X-rays.

Workers were also given preemployment and annual X-rays of the pelvis for a time, but in January 1944, the AEC recommended that the X-rays of the pelvis be discontinued (Ferry 1944a), and it appears that this was done. In 1947, AEC recommended that as part of Harshaw's health and safety program upgrade, an "anterior-posterior radiograph of the thoracic spine" be included in

preemployment and termination medical examinations to detect fluorosis (Kelley 1947). No evidence has been found to date that indicates Harshaw performed thoracic spine X-rays. It does not appear that this was done. However, the X-rays of the pelvis appear to have been resumed at some point because in June 1951 an AEC doctor who visited Harshaw recommended discontinuing the annual X-rays of the pelvis done on all employees (Quigley 1951a), while in November 1951 another AEC doctor recommended (Tabershaw 1951) that even with the contraction in the number of workers, the medical program be continued as previously, with X-rays of the pelvis to be continued for new hires and terminating employees but not for the annual physical – implying that annual X-rays of the pelvis were no longer being done.

Due to the corresponding dates, it appears that X-rays of the pelvis were probably resumed (if they ever were discontinued) in the second half of 1951 in response to the finding of albuminuria in some workers (see Section 2.4). Thus, it would appear that X-rays of the pelvis were done from 1942 to 1944 and again from about 1950 to 1951; this would be on an annual basis. The rationale for the X-rays of the pelvis was to detect bone effects due to fluoride exposure, so it is likely that only the workers in the UF₄-to-UF₆ process area had X-rays of the pelvis.

No information is available regarding whether all workers (e.g., clerical and other support workers) received annual X-rays. In addition, because AEC recommended in October 1948 that Harshaw keep a single file of records for each worker and a complete X-ray file (Kelley 1949a) (which implies that they had not been doing so), it seems likely that early records of X-ray examinations might be in disarray or missing.

No information appears to be available concerning what the practices were during the postoperations phase, 1956–1999.

3.2 EXAMINATION FREQUENCIES

Therefore, the assumption favorable to claimants is made that all personnel who worked in the areas in which work was done for the MED/AEC received initial, annual, and termination chest X-rays [7]. Unless there is evidence to the contrary in the employee's file, the chest X-rays should be assumed to be posterior-anterior (PA) radiographic only. Those process area workers who were working in MED/AEC work from 1942 to 1944 and from 1950 to 1951 should be assumed to have received annual X-rays of the pelvis as well [8].

3.3 EQUIPMENT AND TECHNIQUES

Photofluorography was widely used in the United States for medical screening in the 1930s and was used at some sites supporting the Manhattan Project. No evidence has been located that it was used for Harshaw workers. Therefore, the assumption is made that only conventional X-rays were used for Harshaw.

Because no actual X-ray output measurements or X-ray technique factors are available in Harshaw records, default values for entrance kerma appropriate for this period should be used in the calculation of organ dose for use in dose reconstruction. Information to be used in dose reconstruction for the early years, for which no specific information is available, is provided in ORAUT-OTIB-0006, the dose reconstruction project technical information bulletin covering occupational medical X-ray procedures (ORAUT 2005a).

Guidance regarding organ doses from X-rays of the pelvis has been included in an update to ORAUT (2005a). Note that for Harshaw, such contributions would be applicable only to those workers

handling or working around fluorides (i.e., workers involved with the UO_2 to UF_4 , UF_4 to UF_6 , and UF_6 distillation processes, including workers who handled the hydrogen fluoride supply and storage functions). Also, such contributions would be applicable only from 1942 to 1944 and 1950 to 1951, as indicated above. Organ doses from thoracic-spine projections should be available in the near future.

3.4 ORGAN DOSE CALCULATIONS

ORAUT-OTIB-0006 (ORAUT 2005a) lists the chest X-ray organ doses to be assumed in default of more specific information as approved by the Dose Reconstruction Project. ORAUT (2005a) should also be used to assign exposures for individuals who would have had X-ray examinations of the pelvis.

4.0 OCCUPATIONAL ENVIRONMENTAL DOSE

4.1 INTRODUCTION

Although atmospheric releases appear to have been significant, especially in the early years, little useable information appears to be available.

4.2 INTERNAL DOSE FROM ONSITE ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

In the production of UF_6 , most of the uncondensed gas remaining after condensation and distillation passed through a device called the "large turbosaturator" (also referred to as a "turbo-agitator" or "turbo") to remove solids and then went out a blower on the roof as effluent to the environment. The rest of the uncondensed gas passed through the "small turbosaturator." The large turbosaturator handled the fumes from four sources: (1) the small vent (exhaust) hood over each individual hex reactor; (2) the distillation unit dry ice trap; (3) the bleeds from the distillation unit floor receivers; and (4) the bleeds from the dry ice trap receivers. It used a potassium hydroxide (KOH) spray as a fume-condensing or capture method and was linked to a packed rooftop tower that created the suction and scrubbed the effluent using circulating acidified water. The small turbosaturator handled the fumes from the five ice traps serving the hex reactors; it created its own suction and pulled the fumes through a well of KOH (Rauch 1948).

The small turbosaturator was highly efficient, but the large turbosaturator was very inefficient during most of its process life (Rauch 1948): it was so inadequate that when a receiver was being bled, the suction vents sometimes discharged air into the reactor hoods instead of drawing it out (Hunter 1949a). To solve this problem, Harshaw's practice was to keep the smaller suction vents closed, which negated their effectiveness (Hunter 1949a). In addition, the exhaust system that provided suction across the face of the line of reactors was not exhausted through either turbosaturator or any other scrubbing or filtration system, but was vented straight to the outside atmosphere (Burman 1949). This was a different system from the individual hex reactor hood system and was in fact a much stronger exhaust than the individual hex reactor hood system (Burman 1949; Hunter 1949a) and so actually received more of the contaminants (Hunter 1949a). These systems contained UF_4 dust from the loading process and UF_6 fumes from the disconnecting of the reactors and receivers (Burman 1949). The UF_6 fuming was exacerbated when there was a plugging of one of the hex headers because in that case the nitrogen purge of the hex reactor prior to disconnection had to be skipped and the concentration in the effluent was thus all the higher (Burman 1949).

Releases were highest before the turbosaturator system was installed in about 1947, when Harshaw's neighbors (including homeowners) were threatening lawsuits due to, for example, the etching of glass

in car and home windows that took place as a result of the high F_2 releases (Ray 1947). The turbosaturator system helped a great deal, but levels of both fluorine and uranium dust were still high and neighbors still complained (AEC 1949a) until the problems with the large turbosaturator and the unfiltered exhaust were finally corrected in 1949 with the installation of two "Buffalo" (Buffalo Forge) scrubbers (AEC 1949a; Harris 1949a). These scrubbers were of the same type that ElectroMet had found to be effective on its effluent particulates, a "microcrystalline" dust that was very similar to what Harshaw was dealing with (Hunter 1949a). The efficiency of the Buffalo scrubbers ranged from 75% to 99% (HCC 1950–1953; Stefanec 1951) and they made a significant difference in the Harshaw effluent concentration (AEC 1950a) and in, for example, exposure to workers in the shipping and receiving area (AEC 1950c). However, even these scrubbers were thought by the AEC to be somewhat inadequate with regard to flow rate (Hunter 1949a).

Regarding effluents from the UF_4 plant, the roof exhaust vent from the blower had no filtration device at all; it was pointed downward and thus much of the particulates accumulated on the gravel of the roof (Burman 1949). Harshaw installed a capture device on this vent in about 1949 (Burman 1949). Regarding effluents from the brown plant, a micropulverizer between the denitration pots and the Rockwell (UO_3 -to- UO_2) furnaces was exhausted through two cyclones and a Hersey-type bag collector; the latter discharged directly to the atmosphere of the room it was in and the collected contents discharged down a hopper and chute into a drum on the floor below it (Wolf 1948a). In addition, exhaust ventilation was installed for the UO_3 packaging hood in January 1952 (Klevin 1952c).

Stack measurements appear to have been done to estimate losses (AEC 1949a) (i.e., usually the quantity was expressed in uranium mass lost per hour, per day, or per month). Harris (1949b) concluded that the AEC and Harshaw sampling methods were comparable and could be used interchangeably, and after that Harshaw appears to have done all its own stack measurements. Few measurements are available. The AEC stack effluent samples showed losses of 21 g U/min total from the six reactor stacks and 13 g/min from the turbosaturator stack, for an hourly total loss of about 5 lb of uranium, which AEC thought was mostly in the form of UF_6 (Eisenbud 1949a). Between 100 and 10,000 ft from the Plant C area, AEC found no sample above $10 \mu\text{g}/\text{m}^3$, with multiple such samples not being above background; the maximum concentration was at about 0.3 mi, where the average concentration was $3 \mu\text{g}/\text{m}^3$ (Eisenbud 1949a). The losses, and thus undoubtedly the concentrations, decreased significantly with the addition of the Buffalo scrubbers; by July 1949, the hourly loss rate was down by a factor of about 2 or 3 (Harris 1949c).

Those not working in process areas had a potential for exposure. Because the work was done mostly in one building, the process areas were not always well enclosed or well ventilated, and there was a considerable loss of material out the various stacks that could be carried by building drafting back into the building. Also, contamination appears to have been tracked out of the process areas in the early years at Harshaw. For example, this is suggested by the practices discussed in the 1946 Harshaw operating manual (HCC 1946), by the observation that doors were left open between areas and there was two-way traffic through the one-way turnstiles (Klevin ca. 1948), and by the fact that visitors could not be issued cover clothing until the new guardhouse was installed in 1949 (Morgan 1949). Some potential for tracking likely existed in later years too, even though revisions of the change room operation were intended to prevent this (Ray 1947; Rauch 1948; Kelley 1949a; Eisenbud 1949b; Wolf 1948a,b).

No information is available about releases, if any, in the postoperations and D&D phase, 1956-1999.

4.3 EXTERNAL DOSE

Environmental ambient radiation levels have not been recovered for this site.

5.0 OCCUPATIONAL INTERNAL DOSE

5.1 INTRODUCTION

Few radiation measurements or evaluations of dust exposure were made in plants doing MED/AEC work in the first few years of operations because it was anticipated that the processing of uranium ores, ore concentrates, and compounds would involve little risk of radiation injury. This belief was based on the low specific activity of uranium and on what was thought at the time to be the temporary nature of the work. In January 1944, the Special Materials Division of the MED Medical Section recommended a medical monitoring program to its contractors that included routine physicals, urine sampling, and X-ray examinations for worker protection (Ferry 1944a). It is not clear whether MED prioritized certain sites to implement these programs, but the same document indicated that several contractors had instituted the recommendations while others had not. In addition, both air and dust samples had been collected and analyzed at several of the contractor sites, but not initially at Harshaw. When the MED/AEC NYOO evaluated the results of these surveys at various sites in light of the government intent to continue uranium processing work in the late 1940s, it was determined that the hazards were not negligible. Kelley (1949a) listed the requirements for a minimally acceptable medical program (including maintaining X-ray records) that it wanted Harshaw to institute in addition to what Harshaw might have been doing already.

5.2 URANIUM SOLUBILITY AND PARTICLE SIZE

The uranium processing operations at Harshaw produced some insoluble uranium compounds, such as UO_2 and U_3O_8 ; some moderately insoluble compounds, such as UF_4 , and UCl_4 ; and some soluble compounds, such as UF_6 and its byproduct UO_2F_2 . The default absorption types for radioactive materials that were likely to have been present at Harshaw can be determined from International Commission on Radiological Protection (ICRP) Publication 68 (ICRP 1995). Information on likely uranium absorption type by job title is listed in Table B-22.

There is little information on particle sizes at Harshaw except for one 1950 study done by Klevin (1950b), when U_3O_8 , UO_3 , UO_2 , UF_4 , and UF_6 were all being handled at Harshaw. Table 5-1 lists a summary of the mass median particle sizes measured as well as the result of calculations of the aerodynamic median activity diameter done for this site profile. In the particle size study, the sizes of breathing zone dusts were found to be consistently larger than those in the general air. There was a variation in particle size for duplicate impactor runs of up to $0.5 \mu\text{m}$ for mass median diameters from 1.28 to $4.7 \mu\text{m}$. A "rigorous statistical analysis" showed both impactors to be exactly the same, so it was concluded that the size variations were random. A comparison of the total impactor concentrations to the filter paper samples showed that 17 out of 22 impactor samples were higher; this was found to be statistically significant, although no explanation could be offered for this. When adjusted for density, these particle size results are consistent with ICRP Publication 66 (ICRP 1994) default particle size distributions. Thus, ICRP (1994) defaults should be assumed, including an activity median aerodynamic diameter of $5 \mu\text{m}$.

Table 5-1. Air-sampling information.

Sample location/activity	Form	Type ^a	Mass median diameter, μm^b	Density, g/cm^3	Activity median aerodynamic diameter, μm^c
Nongeneric					
Gulping orange (UO_3)	UO_3	BZ	3.1	7.29	8.37
Near pots, NE corner	UO_3	GA	1.8	7.29	4.86
Rockwell 1st deck	UO_2	GA	2.3	10.96	7.61
Rockwell discharge	UO_2	GA	2.1	10.96	6.95
Dumping 8 brown (UO_2) trays	UO_2	BZ	3.9	10.96	12.9
Brown-green (UO_2 - UF_4) loading platform	UO_2	GA	4.4	10.96	14.6
20 ft south of laundry	UO_2	GA	2.6	10.96	9.27
Dumping 8 green (UF_4) trays	UF_4	BZ	4.7	6.7	12.2
Hex area near operator's desk	UF_6	GA	1.28	4.68	2.77
Removing ice trap	UF_6	BZ	1.35	4.68	4.22
Center of still area	UF_6	GA	1.35	4.68	2.92
Generic					
Generic UO_3	UO_3	BZ	3.1	7.29	8.37
	UO_3	GA	1.3	7.29	3.51
Generic UO_2	UO_2	BZ	3.9	10.96	12.9
	UO_2	GA	2.4	10.96	7.95
Generic UF_4	UF_4	BZ	4.7	6.7	12.2
	UF_4	GA	4.4	6.7	11.4
Generic UF_6	UF_6	BZ	2	4.68	4.33
	UF_6	GA	1.3	4.68	2.81

- BZ: breathing zone; GA: general area
- For the top (nongeneric) block: the filter paper concentration is from a single sample, while impactor concentration and the mass median particle size are an average of the two impactor samples.
- Mass median diameter* is the term used in the reports but activities not masses were actually measured. The mass median aerodynamic diameter is the equivalent of the aerodynamic median activity diameter if the activity is considered to be homogenous. The aerodynamic median activity diameter calculation assumed that the particles measured could be treated as spherical so that the Stokes diameter is equal to the geometric diameter (measured) and the slip correction factor is equal to one.

5.3 IN VITRO MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PROTOCOLS

5.3.1 In Vitro Urine Analysis

5.3.1.1 Early Urine Studies

Ferry (1944a) stated that Harshaw was one of the sites following the MED-recommended medical program, which included urinalysis for uranium done as a "screening experiment" with regard to both acute and chronic exposures. HCC (ca. 1945) stated that monthly urine samples for all employees exposed to UF_6 were sent to the Medical Division of the MED (to be sent on to the University of Rochester) for uranium analysis. The 1945 measurements might have been part of a volunteer study of 24-hour excretion of uranium; the volunteers, who included the 491/492 plant superintendent, attempted to limit voiding to "the hour of arising, noon hour, completion of the day's work, and on retiring" (Mears 1945). In January 1946, MED informed the plant superintendent that its Medical Section had decided to terminate all urinalyses for uranium and fluoride content because of the contamination of a high percentage of the samples.

In August 1947, AEC told Harshaw that its health program had to be improved, including a routine urine testing program for uranium and fluorine to be instituted after a spot check program was

completed (Kelley 1947). However, the samples submitted in 1947 again appear to have been contaminated on the basis of both the fluorine and the uranium analyses (Howland 1947b).

Because of the questions regarding the validity of sample results prior to December 1949, the apparent variations in sample analysis methods, and even who was doing the analyses, the Harshaw urinalysis data prior to December 1, 1949 should not be used. It appears that the errors, if any, are in the conservative (high) direction and thus would be favorable to claimants [9]. However, urine samples are also likely to be rare, necessitating an air sample-based approach.

It has been determined that it is not feasible to perform dose reconstructions from August 14, 1942, through November 30, 1949, due to the lack of internal dosimetry data for the radionuclides associated with uranium for operations at Harshaw [10]. Beginning December 1, 1949, adequate information to perform dose reconstructions is available (Elliott 2007).

5.3.1.2 Routine Urine Program

Some routine urinalyses appear to have begun in late 1948 (Harris 1949d). However, there is a description of a (new) pilot program to measure uranium in urine in March 1949 [described in Harris (1949d) and Eisenbud (1949c)], and Lippmann (1958) stated that the urine sampling program began in January 1950. Measurements for the 1948-to-early-1949 period for a few people appear in the Harshaw records, and an increasing number of workers appear to have been included in the urine sampling program as of late (December) 1949. Sargent (1950b), while it referred to a previous lot of urine samples, also requested that Harshaw institute a urine sampling program "on a running basis" to sample about 100 workers per month, including occupations that the AEC specified. Thus, the January 1950 time point could represent an acceleration of urine sampling rather than its start. In 1951, an AEC doctor recommended continuing the urinalyses as part of the Harshaw medical program even though the green and hex plants were being phased out, but it is not clear if the urinalyses were to include a uranium measurement or not (Tabershaw 1951).

Sargent (1950a) stated that 200 workers were subject to urinalysis, which appears to have included workers at all three major areas of Plant C. While there was no tally kept by Harshaw of the number of people exposed to uranium who had left their employment, in November 1951, Harshaw provided the AEC an estimate of several hundred such people who had been exposed for more than a year.

As noted above, the AEC specified at one point the worker categories to be sampled: hex loaders, hex operators, still operators, brown and green loaders, orange pot unloaders, Rockwell operators, black oxide (ore concentrate) loaders, and shipping and receiving personnel (Turner 1947a). A former AEC official stated that samples were taken from "hexafluoride" (UF₆) and "nitrate" (UF₄) workers and that one could see a 10-fold drop in the uranium content between Friday night and Monday morning samples for those exposed to soluble forms, but little drop for those exposed to insoluble forms (ORAU 1983). Lippmann (1958) describes a study of 1950 data to determine the correlation between urinalyses and exposure of Harshaw workers in the green and hex plants; he mentions workers showing elevated urinary albumin measurements, so this apparently retroactive study might have been motivated by that occurrence. Table B-20 is drawn from Lippmann (1958); the results are listed along with statistical data calculated for this site profile. Thus, Table B-20 could be useful for trend or comparison purposes or for interpreting worker bioassay records.

Before the UF₄ and the UF₆ production plants were closed in 1951, urine samples were usually taken in pairs on a before-and-after weekend basis (Lippmann 1958); these were called the "before weekend" or "Friday" sample and the "after weekend" or "Monday" sample, respectively. Samples were taken every week but not from every worker or group of workers, so the average was a pair of

samples per worker per month (Lippmann 1958). From September 1951 on, the AEC appears to have directed Harshaw to discontinue all Friday samples from the 491 area; to obtain Monday samples from each 491 operator every 2 weeks; to discontinue urine samples from health physics personnel and guards; and to include all process personnel in the urine sampling program (Harris 1951). The reason for including the last group was that many process area workers had rarely or never submitted urine samples, resulting in serious gaps in the data that the AEC had collected (Harris 1951). A coworker worker study was conducted to aid in filling these data gaps and is described in Section 5.9.

Samples do not appear to have been 24-hour samples. Quigley (1951b) suggests that at least some might have been 6- to 8-hour samples. Referring to urine samples sent to the University of Rochester from any AEC site, ORAU (1983) stated that samples were collected from workers in 4-oz glass bottles with Bakelite caps and shipped offsite for analyses.

It should be noted that many workers, over the years of their employment, worked both at the green and brown plants, where the uranium form was fairly or highly insoluble, and at the hex plant, where the uranium form could be either soluble or insoluble. Urinalysis data appeared to end in 1953.

5.3.1.3 Minimum Detectable Activities

Uranium fusion photofluorimetry urinalyses performed by the University of Rochester and the AEC NYOO were similar to those performed at other AEC facilities. Consistent with the information above, the default detection threshold for uranium urinalysis can be 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). Lippman (1958) stated that the analytical precision of the urine samples was about ±10 µg/L for samples reading less than 100 µg/L; Howland (1947b) stated that the limit of reliable determination by the fluorimetric method was about 0.01 mg/L, or again, about 10 µg/L.

5.3.2 In Vitro Methods for Uranium

Harshaw urinalysis samples were sent for analysis to the University of Rochester directly (HCC 1950–1953) or to Health and Safety Laboratory (HASL) (Eisenbud 1975) to be sent on to Rochester. In about January 1949, the University of Rochester work in support of the AEC was switched to HASL (ORAU 1983) and thus HASL itself was analyzing the Harshaw urine samples. ORAU (1983) stated that the HASL urinalysis program ended in 1955 or 1956 and that Harshaw was the last plant for which HASL did analyses. However, HCC (1950–1953) indicated that urine samples were sent to National Lead Company (Fernald) for analysis late in the plant's life; this could have been only during the decontamination phase.

The radiological analysis was apparently only for uranium content (referred to as "X in urine" or "uranium-in-urine"). ORAU (1983) stated that all urinalyses done at the University of Rochester used the fluorometric method; the urine was dried in a platinum dish, then fluxed with either sodium fluoride or a lithium-calcium fluoride mix and counted.

ORAU (1983) also stated the following about samples analyzed at the University of Rochester. The samples were run in triplicate, with the results usually being within ±2 µg/L of one another; if this turned out not to be the case, it indicated that there had been poor fusion of the flux and the samples were re-fused and rerun, which usually corrected the problem. The value recorded was the median value of the three. For insoluble uranium, it was considered that 30 µg/L in the urine corresponded to an air concentration of 50 µg/m³. The analysts were confident of readings greater than or equal to

5 µg/L, but if more confidence was desired for lower level samples (e.g., for special projects), the urine was concentrated either by ion exchange or by extraction, or more aliquots were run.

5.4 OTHER BIOASSAY METHODS

Fecal sample analyses do not appear to have been a routine part of the Harshaw bioassay program. No whole-body or lung counts appear to have been performed for Harshaw employees during the covered period. A few blood measurements were done in 1950 and 1951. Data have been found in Harshaw bioassay records for about 16 people (including one AEC person), with most volumes given and with the notation that the blood was assayed using fluorometric analysis [11]. No analytical precision has been found for the blood measurements.

5.5 RADON LEVELS DURING OPERATIONS

As indicated above, uranium bearing ore (unprocessed rock) does not appear to have occurred at Harshaw, at least in any large quantities. Because the feed material that was received by Harshaw had been processed prior to receipt, the activity concentrations of both ^{226}Ra and ^{230}Th would have been much lower than that of uranium, and radon concentrations would likewise have been reduced. No records of radon measurements made by the AEC or Harshaw have been located for the period that the Harshaw facilities are covered by this document.

Because there are no known radon measurements during the period of operations, it is difficult to determine what the Harshaw radon levels actually were. It can be inferred that the AEC concluded that the Harshaw radon levels were always or nearly always below the radon tolerance level of 1×10^{10} Ci/L and likely usually below the detectable level of 1×10^{-12} Ci/L due to the low radium content of the received material. But to make an upper bound estimate of potential radon levels, measurements from the Mallinckrodt site (ORAUT 2005b, Table 25) can be evaluated as a bounding set. Table 5-2 lists Mallinckrodt levels from 1954 to 1957, which was after high-radium ores had been processed and the feed was mostly MGX (a Belgian Congo ore tailings concentrate that still had a high radium level), soda salt, and lower grade ore. These measurements would be higher than the levels at Harshaw because of the higher radium content in the received feed and the likely comparable residence time of the material in containers. The measurements should be comparable also because the Mallinckrodt process included the ore concentrate-to- UO_3 , UO_3 -to- UO_2 , and UO_2 -to- UF_4 steps [12].

The results of calculations of working level months (WLMs) are listed in Table B-18. In these calculations, the figures in the rightmost column of the table above were used and the following assumptions were made [these are the same as those used for Mallinckrodt (ORAUT 2005b)].

1. For ore concentrate storage areas, the equilibrium factor for the radon daughters was assumed to be 1.0. For process and maintenance areas, the equilibrium factor was 0.50. For nonprocess areas, the equilibrium factor was 0.40.
2. For workers in ore concentrate storage and process areas and in shops and support facilities other than laboratories, the occupancy factor was assumed to be 0.75 for the normal work area and 0.25 for the break room, locker room, and other low-exposure areas. For maintenance workers who visited the process areas, the occupancy factor was assumed to be 0.50 for the process area, 0.25 for the maintenance shop area, and 0.25 for the break room, locker room, and other low-exposure areas. For laboratory workers, the occupancy factor was assumed to be 0.88 (i.e., 7 hr/d) in the laboratory area and 0.12 in the break room, locker

Table 5-2. Radon levels in the Mallinckrodt plants in the later period of operation, in units of 10^{-10} Ci/L of radon.

Area	Room or building	1954	1955	1956	1957	Harshaw ^a
Indoor	Scalehouse/Ore Storage/Warehouse	0.01	0.01			0.01
	Digest/Feed	0.03		0.01	0.03	0.03
	Extraction Cells	0.26		0.01	0.01	0.01
	Centrifuge Area	0.07		0.01	0.01	0.01
	Feinc/Filter/Raffinate/Cloth Storage/Niagara	0.14	0.50	0.07	0.01	0.07
	Orange Packing	0.12				0.12
	Pot Room	0.02				0.02
	Shotgun Lab (UO ₃ assay)	0.04				0.04
	Lab (Research/ Control /X-ray/Radium)	0.01			0.04	0.04
	Decontamination Room	0.01				0.01
	Nitric Acid House	0.01				0.01
	Ether House	0.02				0.02
	Receiving (non-ore shipping and receiving)	0.01				0.01
	Welding, Millwright, and Electrical Shops	0.01				0.01
Maintenance Shop	0.04				0.04	
Outdoors/yards	Smoking (Break) Room, Production Office	0.01				0.01
	Near Ether House/Acid unloading station	0.01				0.01

a. Mallinckrodt 1956 and 1957 figures correspond to lower grade ores, soda salt, etc. that Harshaw used in its ore concentrate to UO₂/UO₃ processing. These values are recommended for use in dose reconstructions.

room, and other low-exposure areas. For office workers, the occupancy factor was assumed to be 1.0 in low-exposure areas.

3. The conversion to WLM per year from the measured values in units of pCi/L was assumed to be:

$$\text{Intake (WLM/yr)} = 0.12 \sum [\text{Equilibrium factor} \times \text{Occupancy factor} \times \text{Radon level (pCi/L)}] \quad (5-1)$$

5.6 AIRBORNE RADIOACTIVE DUST MEASUREMENTS

A formal program of airborne radioactive dust measurements taken by the AEC began when the first formal report of Harshaw radioactive dust levels was issued by the AEC (1948), although dust samples were said to have been received periodically from Harshaw from perhaps 1943 on (Ferry 1944a,b). Eisenbud (1975) reported that AEC air sampling was generally done by collecting the dust on Whatman #41 filter paper and counting total alphas; a correction for self-absorption in the paper was then applied and the results were reported as alpha disintegrations per minute per cubic meter. Table 5-3 lists production information for UF₄ and UF₆ and the start of dust sampling. Additional details of the air sampling methods and practices are found in AEC (1950b,c,d), Glauberman and Harris (1958), Hayden (1948), Klevin (1950a), and Lippmann (1958).

The results are listed in Tables B-10 to B-15 in disintegrations per minute per cubic meter. Tables B-10, B-11, and B-12 list results of instantaneous (spot) measurements for various areas and particular jobs, while Table B-13 lists results for a beta count of some air samples taken in the green and brown plant. Table B-14 lists results for individual areas, expressed as the daily [time-] weighted average exposure (DWE) as calculated by the AEC. Table B-15 lists results for individual occupations (job titles), also expressed as the DWE.

Note that Breslin (1958) stated that the DWEs calculated by NYOO from measured data do not include any correction for respirator use and should be viewed as (only) potential exposure. Harshaw workers were supposed to wear respirators when loading or unloading hex reactors (Ferry 1944b; Rauch 1948; HCC 1946) and whenever it was thought that there was a significant potential for

Table 5-3. Production of UF₄ and UF₆ and the start of dust sampling.^a

Date	Average, lb/d	Notes
UF4 production		
Apr–Aug 1942	---	Laboratory production
Sep 1942–Jul 1944	1,650	Production level; Feb 1944: first MED dust sampling ^b
Aug 1944–Dec 1946	2,650	
1947	5,500	
1948	5,800	May, Sept 1948: first two AEC dust studies using DWEs
1949	5,000	
1950	5,250	
Jan–Mar 1951	5,650	
UF6 production		
To April 1944	400	Pilot Plant; Feb 1944: first MED dust sampling
May–Jun 1944	2,300	
Jul–Nov 1944	3,300	
Dec 1944–1945	4,500	
1946	4,500	
1947	5,700	
1948	6,800	May, Sept 1948: first two AEC dust studies using DWEs
1949	6,300	
1950	6,900	
Jan–Mar 1951	6,600	

a. Data are from AEC (1951c) supplemented by HCC (ca. 1945).

b. The February 1944 MED dust sampling date represents the earliest date found in records; samples might have been taken earlier.

exposure to elevated dust or fume concentrations (HCC 1946). However, several documents indicate that respirators were not always worn or worn effectively (see Hayes 1947; HCC ca. 1945; HCC 1950–1953; Kelley 1947; Klevin ca. 1948; Lippmann 1958; Long 1947; Morgan 1949; Rauch 1948; Sargent 1948; Turner 1947a,b,c,d). Therefore, no credit for respirator use should be taken when applying air sample measurements. The contribution of resuspended dusts is assumed to be included in all air sample data cited in this site profile.

5.7 POSTOPERATIONS INTERNAL DOSE

To estimate doses to workers from MED/AEC contamination, use was made of the results of the various Harshaw contamination and dose rate surveys (i.e., from Blatz 1951; Klevin 1955a; Schoen 1958; DOE 1984; and FUSRAP 2001). Data taken from these references and used in the calculation of exposure are listed in Tables B-23 and B-26; these represent only a small and select subset of data from larger sets of many data points although they were representative and conservatively chosen.

The RESRAD-BUILD computer code (ANL 2003) was used to calculate annual exposures from inhalation of airborne particulates and radon (and its progeny). Maximum averages of surface contamination were used to produce the inhalation and radon source terms that are favorable to claimants for RESRAD-BUILD; and the inhalation RESRAD-BUILD results were then used to produce the source term for the ingestion calculations. For RESRAD-BUILD parameters other than the source term, values favorable to claimants were used when they could be determined; when no specific or suitable values could be determined, conservative default values given in the RESRAD-BUILD manual (ANL 2003) or other guidance documents were used.

The results of the RESRAD radon calculations for the postoperations years were compared to measured radon concentrations given in DOE (1984) as listed in Table 5-4, with the working level (WL) values given in DOE (1984) converted to WLM per year. In DOE (1984), the Argonne National

Table 5-4. Comparison of RESRAD-calculated and DOE (1984)-measured radon concentrations.

Building	DOE (1984) ^a			RESRAD-BUILD ^a	
	Range (WL)	Range (WLM/yr)	Range-ABG ^a (WLM/yr)	Range (WLM/yr)	Mode (WLM/yr)
Bldg G-1	2.5–6.9	0.033–0.093	0.0–0.053	0.0–0.069	0.017
Boiler House	4.7	0.063	0.023	0.0–.00063	0.00031
Foundry	1.7–5.6	0.023–0.075	0.0–0.035	0.0–.027	0.0027
Garage	3.7	0.05	0.01	0.0–.00063	0.00031
Warehouse	3.3–6.5	0.044–0.088	0.004–0.048	0.0–.0021	0.0010
Bldg K-1	1.1–5.7	0.015–0.077	0.0–0.037	0.0–.021	0.0031
Bldg M-1	3.5	0.047	0.007	0.0–.00002	0.00001
Bldg P-1	1.2–3.3	0.015–0.044	0.0–0.004	0.0–0.052	0.015

a. The ANL values include natural background while the RESRAD-BUILD values do not. Thus, for illustrative purposes, the "Range - ABG" (Range minus ABG) column gives the ANL range figures minus an average annual indoor radon background (ABG), taken to be 0.040 WLM/yr based on 2,000 hr/yr.

Laboratory (ANL)-reported values were based on 100% equilibrium, while the RESRAD-calculated values corresponded to the degree of equilibrium present given the assumed room volume, air changes per hour, etc. The ANL-measured values also included natural radon background for the buildings, which were typically constructed of concrete and brick. Thus, it would be expected that in general, the ANL-measured values would be somewhat higher than the RESRAD-calculated values. Comparing columns 4 and 6 of the table, it is clear that there is agreement between the two sets of values, considering that the radon added by the residual contamination appears to be at or below the level of background.

5.8 ACTIVITY FRACTIONS

Harshaw urinalyses measured uranium only. Air samples usually were only analyzed in terms of alpha activity. For other radionuclides, Table 5-5 lists the activity fractions relative to uranium activity, assuming that the uranium daughter products are in equilibrium [13]. Nearly all of the radium and thorium was removed from the milled ore, but no assays of the incoming ore were located. Equilibrium fractions were chosen because they are favorable to claimants.

Table 5-5. Bioassay and alpha air sample activity fractions by period and radionuclide.

Period	Radionuclide	Bioassay activity fractions	Alpha air sample activity fractions
8/14/1942 – present	U-natural	1	0.402
	Th-230	0.489	0.196
	Ra-226	0.489	0.196
	Po-210	0.489	0.196
	Pa-231	0.0228	0.00916
	Ac-227	0.0228	0.00916
7/1/1952 – present	Pu-239	0.00464	0.00464 ^a
	Np-237	0.00137	0.00137 ^a

a. Assumed to be the same as bioassay activity fraction.

5.9 DETERMINATION OF INTERNAL DOSES (INSTRUCTIONS TO DOSE RECONSTRUCTORS)

Prior to December 1, 1949, bioassay results are not reliable. Therefore dose reconstructors should use air-sample based intakes as described in Sections 5.9.1, 5.9.2, and 5.9.3. From December 1, 1949 on, individual uranium urinalysis results for Harshaw workers should be used to determine internal exposure to the individual when they are available. Where individual urinalysis results are not

available, illegible, or inadequate, the coworker data included in Attachment A and summarized in Table 5-6 are to be used to estimate internal exposures that are favorable to claimants [14].

Table 5-6. Chronic intake rates for types F, M, and S ²³⁴U.

Dates	Type F			Type M			Type S		
	Percentiles		GSD ^a	Percentiles		GSD ^a	Percentiles		GSD ^a
	50th (pCi/d)	84th (pCi/d)		50th (pCi/d)	84th (pCi/d)		50th (pCi/d)	84th (pCi/d)	
12/1/49–3/31/50	650.3	2,607	4.01	3,934	16,220	4.12	19,910	79,860	4.01
4/1/50–12/31/51	157	650.5	4.14	460.1	1,830	3.98	3,651	18,070	4.95
1/1/52–12/31/53	37.43	86.85	3.00	115.9	201.7	3.00	1,071	2,655	3.00

a. Geometric standard deviations (GSDs) less than 3.0 were assigned a GSD of 3.0.

The Harshaw bioassay data analyzed and assessed to be used as coworker data were verified and statistically analyzed, and intake estimates were generated to aid dose reconstructors when sample data might either be inadequate or unavailable for estimating internal exposures to unmonitored or marginally unmonitored workers [14].

The 1948 figures (the earliest DWE figures available) in Table B-15 were used to estimate the 1942 to 1947 period in Tables B-16 and B-17. There was significant variability in exposure by job, by plant, and by year at the Harshaw site. Hence, it is not feasible to calculate a matrix of intakes for all occupational types, all locations, and all periods for inclusion in this site profile. The data in Tables B-16 and B-17 could then be used for the period including the SEC class, when appropriate, to help determine intakes based on the time-weighted daily average exposure level. Note, however, that nonuranium doses are not to be assigned for the SEC period, August 14, 1942, through November 30, 1949 [10].

Care must be taken in the case of workers whose work histories show them apparently still doing work with radioactive materials after the period of AEC-sponsored decontamination (1956-1960). This is because such work during the postdecontamination years was not AEC-contracted work; it is uncertain what work might have been done and what materials might have been used; there might be no available records covering these years, and such records as there are might reflect doses associated with private work, not with residual contamination from AEC activities. Coworker data could not be developed for the AEC-sponsored D&D years. Thus, where there are no individual data available for this period, reference should be made to the data in Tables B-25 and B-26 [15]. For the postdecontamination years, where individual data cannot be located, as is likely, the data of Tables B-25 and B-26 should be used, as discussed in Section 5.9.5 below.

5.9.1 Determining Annual Inhalation Intakes Based on Time-Weighted Daily Average Inhalation Exposure Data (Table B-16), 8/14/1942-11/30/1949

Table B-16 lists the DWEs from Table B-15 as converted to effective annual intakes of radioactivity in picocuries. The conversion was done by assuming a breathing rate of 1.2 m³/hr for 2,000 hr/yr (ICRP 1994). Some ratioing was done to cover job titles for unreported periods or for periods when some job titles were lumped together. Also, for the early period when only spot air samples were taken and DWEs were not used, measured data from 1948 or (if 1948 data were not available) 1949 data were used for 1942 to 1947. The justification for this is as follows. While control of dust and fumes undoubtedly improved somewhat from 1942 to 1948, it is clear that administrative and physical control measures were often ignored after they were implemented and that increases in production also increased dust levels as a general rule. Also, the aging of equipment undoubtedly resulted in more leakage and more need for maintenance. The start of intermittent air sampling in about 1944, regular air sampling in 1948, and urinalysis in 1949 are correlated with significant UF₄ and UF₆ production

increases (see Table 2-2) or with the peak of production. Hence, it is deemed that the later measurements bound the earlier ones.

Table B-16, the table of annual intakes based on time-weighted, daily average inhalation of uranium and its daughters, can be used to estimate inhalation intakes on an individual basis if urinalysis and related information is unavailable or spotty (e.g., for 1942 to 1948) or to estimate doses for comparison to doses calculated from individual urinalysis and other data [16]. Note, however, that nonuranium doses are not to be assigned for the SEC period, August 14, 1942, through November 30, 1949 [10]. For workers whose work history showed job rotation, but the time spent in each job is uncertain, a choice of the job title giving the maximum exposure applied through the whole year, will be favorable to claimants. The steps are as follows:

1. The job title or work area selection(s) from Table B-22 should be made on the basis of the claimant's submitted information, urinalysis records, film badge records (if helpful), employment records, and other information. Table B-4 can also be used as an aid.
2. The annual intakes from Table B-16 should be selected to correspond to the job title or work area, plants, and periods.
3. Assumptions regarding isotopic content of the radioactivity in the air should be made as listed in Table 5-5, above.

5.9.2 Determining Annual Ingestion Intakes Based on Time-Weighted Daily Average Inhalation Exposure Data (Table B-17), 8/14/1942-11/30/1949

Because health physics practices at Harshaw appear to have been substandard, ingestion intakes might have been significant. The effects of ingestion on the gastrointestinal (GI) tract are not well accounted for by the assumption of inhalation intakes. Thus, if the organ of concern is a GI tract organ, chronic ingestion intakes should be included in addition to the inhalation intakes (NIOSH 2004). Table B-17, the table of annual ingestion intakes based on time-weighted daily average inhalation of uranium and its progeny, can be used to determine ingestion intake on an individual basis if urinalysis and related information is unavailable or spotty (e.g., for 1942 to 1948) or to generate doses for comparison to doses calculated from individual urinalysis and other data [17]. Note, however, that nonuranium doses are not to be assigned for the SEC period, August 14, 1942, through November 30, 1949 [10]. The steps are as given below.

1. The job title or work area selection(s) from Table B-22 should be made on the basis of the claimant's submitted information, urinalysis records, film badge records (if helpful), employment records, etc. Table B-4 can also be used to help make the selection.
2. The annual intake from Table B-17 should be selected to correspond to the job title or work area, plants, and periods.
3. Assumptions regarding isotopic content of the radioactivity in the air should be made as listed in Table 5-5, above.

5.9.3 Estimating Inhalation and Ingestion Intakes by Using Time-Weighted Daily Average Exposure Data (Table B-15), 8/14/1942-11/30/1949

Table B-15, the time-weighted daily average exposures for specific job titles, and Table B-14, the time-weighted daily average exposures for specific work areas, can be used to estimate intakes using

assumptions different from the standard ones used for Tables B-16 and B-17 [18]. The job title selection should be made on the basis of the claimant's submitted information, urinalysis records, film badge records (if helpful), employment records, and other information. Tables B-4, B-21, and B-22 should be used to help make the selection [19]. The air concentrations from Table B-15 (or in default of information in Table B-15, use Table B-14) should then be selected to correspond to the job title(s), work area(s), and periods. Any necessary adjustments should be made for partial years, overtime, etc. Note, however, that nonuranium doses are not to be assigned for the SEC period, August 14, 1942, through November 30, 1949 [10].

The intakes, in picocuries, should be calculated by multiplying the appropriate air concentrations by the breathing rate(s) and the hours, and dividing by 2.22 dpm/pCi. Once the inhalation intake has been determined, the ingestion intake can be calculated by using the assumptions in NIOSH (2004):

$$\text{Ingestion intake (pCi/yr)} = 0.021 \times \text{Inhalation intake pCi/yr} \quad (5-2)$$

5.9.4 Estimating Annual Radon Exposure by Using Bounding Radon Exposure Data (Table B-18), 12/1/1949 on

Table B-18, the table of annual radon exposures based on Mallinckrodt radon measurements (ORAUT 2005b), should be used to determine bounding estimates of radon exposures if the organ of concern is a respiratory tract organ. Note, however, that nonuranium doses are not to be assigned for the SEC period, August 14, 1942, through November 30, 1949 [10]. Table B-18 was based on spot measurements usually taken at times of representative or maximum radon emanation, not on daily weighted average exposures, which are unavailable.

1. The job title or work area selection(s) from Table B-22 should be made on the basis of the claimant's submitted information, urinalysis records, film badge records (if helpful), employment records, and other information. Table B-4 can also be used as needed.
2. The annual radon intake(s) from Table B-18 should then be selected to correspond to the job title or work area, plants, and periods.

In general, only workers who spent time in an area where radon might concentrate significantly, such as in the ore concentrate storage area, were likely to be exposed to any significant level of radon. Nonprocess workers, particularly office workers, can be assumed to have insignificant radon exposures.

5.9.5 Estimating Intakes During the D&D/Postoperations Years by Using Calculated Data (Table B-25)

As stated above, if urinalysis data are available for an individual, they should be used to determine the internal exposure to the individual. Little if any urinalysis data are expected to be documented in the D&D/postoperations years.

Because there is not a formal date that can serve as a cutoff for the dose reconstructor to use, judgment will have to be applied as to how much of the dose record after 1955 should be counted as contributing to dose from AEC decontamination operations [20]. Note that after the operations and D&D years, Harshaw received an AEC license to use certain radioactive materials. Care must be taken in the case of workers whose work histories show them apparently still doing work with radioactive materials after the D&D period; such work during the postdecontamination years was not AEC-contracted work; it is uncertain what work might have been done and what materials might have

been used, and any intakes found in records from the post-D&D years might be attributable to commercial operations with radioactive material, not to residual contamination from AEC work. But it is generally acceptable to assume that operations continued until the end of 1955 in the ore-concentrate-to- UO_3 (brown) plant and the associated laboratories [21].

Annual intake estimates were calculated from measured and interpolated data listed in Table B-24 and the results are listed in Table B-25. See the notes following the table for information on how the table was developed. For the D&D and postoperations years if individual data cannot be found or where it is not clear what the source of the intake data was, the data of Table B-25 should be used [22]. In using Table B-25, the applicable years of employment for each indicated period should be determined and the number of years in the covered period and area should be multiplied by the annual value. The annual value can be prorated for partial years.

If the claimant was a process or other worker likely to have spent considerable time in areas of significant residual contamination and it is not clear in which building the worker actually spent time, the Building G-1 values can be used [23]. Intakes listed in Table B-25 can be ratioed to indicate time spent in the contaminated area.

6.0 OCCUPATIONAL EXTERNAL DOSE

6.1 INTRODUCTION

Conditions for external radiation exposure are best summarized by the description in AEC (1949b): "Severe exposures to external beta radiation ... exist in this plant." Beta dose and gamma dose to the extremities was potentially high for those workers handling hex ash and other residues. Because little individual worker monitoring data are available before about 1949, some extrapolation of existing data to cover the unmonitored periods is necessary; however, with the significantly lower quantities of material handled and produced in the laboratories and pilot plants, the external doses were not likely to be greater than they were later. Exposure rate information retrieved from various AEC reports was condensed into Tables B-5, B-6, B-7, and B-8 and is to be used to conduct external dose reconstructions for periods where personnel exposure data are not available. (A coworker assessment was not conducted for the Harshaw external exposure data set.) Individual extremity dose data appear to be lacking and must be inferred mostly from measured and/or calculated dose rates.

The nominal period of operations was 1942 to 1955. However, as Table 2-2 shows, not all of the operating areas started in 1942 or operated until 1955. Thus, for example, an entry of "491 Area" (UO_2 - UF_4 area) in 1953 would indicate that D&D was being performed rather than process operations, while an entry of "493 Area" (ore-concentrate-to- UO_3 area) at the same time would indicate process operations. If there is no information to the contrary, the periods of operation and D&D should be assumed to be as given in Table 2-2.

6.2 BASIS OF COMPARISON

Since the initiation of the MED in the early 1940s, various radiation dose concepts and quantities have been used to measure and record occupational dose. A basis of comparison for dose reconstruction is the Personal Dose Equivalent, $H_p(d)$, where d identifies the depth (in millimeters) and represents the point of reference for dose in tissue. For weakly penetrating radiation of significance to skin dose, $d = 0.07$ mm and is noted as $H_p(0.07)$. For penetrating radiation of significance to whole-body dose, $d = 10$ mm and is noted as $H_p(10)$. Both $H_p(0.07)$ and $H_p(10)$ are the radiation quantities

recommended for use as the operational quantities to be recorded for radiological protection purposes.

Film badge records contain "beta" values that were obtained by subtracting the optical density of the film behind the cadmium shield from that behind the open window (Blatz 1950a,b,c). These recorded values can be assumed to be equivalent to Hp(0.07). The performance of the MED badge in determining Hp(10) under field conditions is less certain. Therefore, an approach favorable to claimants is adopted for converting exposure to tissue dose. See Section 6.3.2.

6.3 DOSE RECONSTRUCTION PARAMETERS

6.3.1 Site Historical Administrative Practices

Available film badge data tabulated weekly begin on 25 August 1947 as shown in dose records collected in various data captures by the EEOICPA Dose Reconstruction Project; these data are more or less continuous until the end of production.

However, it seems clear that film badging started earlier than that. First, the document titled *Harshaw Radiation Summary, Aug 44 to Mar 48* (University of Rochester ca. 1948) indicates the character of early doses as recorded in film badge results. The earliest results given appear to correspond to a badge start date of 29 August 1944. Although the author of this document is not clear, this file is similar to a summary prepared for the Mallinckrodt site and is clearly from the same film badge service, the University of Rochester (e.g., there are entries termed "Rochester Control" for control badges used during processing). The Harshaw summary contains badge results for 187 individuals, by name and Social Security Number; the number of weeks of employment for each employee (range from 1 to 181); the employee's total gamma results (range from 0 to 6,890 mrem); the total beta results (range from 120 to 139,740 mrem); and the starting and ending dates of monitoring. Weekly doses from this list, averaged over all employees, are 2.5 mrem/wk from gamma radiation and 741 mrem/wk beta equivalent. This corresponds well with Figure 8 of AEC (1949b), which centers most individual exposures on an axis of "700 mrep/wk." Note that Table B-7 of this site profile includes data from Figure 8 of AEC (1949b).

Second, there are various mentions of film badging being done prior to 1947, although possibly with interruptions. AEC (Hayes 1949) discussed a Harshaw individual who was badged from late 1945 on. The MED and AEC (Tybout 1944a, b; Schoen 1958) mentioned beta or gamma measurements using film badges worn by workers apparently routinely. Becker (1946) stated that while the plant superintendent would go along with the discontinuance of film badge monitoring, he wanted some kind of substitute badge for security reasons; Mears (1946a) quoted an MED consultant as questioning the advisability of discontinuing film badge monitoring on the basis that the exposure to the hands was high and the (chest) film badge thus provided some indication of the extremity dose, even if the registered (whole-body dose) was below tolerance. Despite this, an MED manager advocated discontinuing film badge monitoring on the grounds that the indicated exposures were consistent over the 2.5 years and that it should be continuation of monitoring that needed to be justified, and not the discontinuation (Mears 1946a). But Mears (1946b) decided that film badging was to be continued and every employee was to wear a badge, including supervisors. The issuance of film badges to everyone was so that the badge could double as a security badge (Mears 1946b). In addition, Blatz (1949a,b) and Eisenbud (1949d,e), dated February or March 1949, discussed a review of Harshaw film badge readings, with the details suggesting that badging had been going on for some time, and Klevin (ca. 1948) stated in November 1948 that film badges were not always being worn as required. Thus, it appears that a significant period of weekly film badge data, from approximately late

1945 through 1948, might be missing, although the data for most individuals will likely be found in a totaled format in the Harshaw summary discussed above (University of Rochester ca. 1948).

From film badge records, some scapular (shoulder-placed) film badges appear to have been worn in May 1949. As discussed in Section 6.3.2, extremity film badges were used only on an experimental or study basis due to difficulties (e.g., heat damage).

Film badges, generally, were issued weekly. Badges were turned in at the guardhouse when workers left the site and picked up at reentry (HCC 1946). AEC (1951c) stated that over the 13-wk period of 1 November 1948 to 24 January 1949, 91 badges were worn per week; over the period 4 July 1949 to 19 June 1950, 180 badges were worn per week.

Most existing records are labeled as being for Plant C, but occasionally, a film badge results card is labeled as being for "Plant E." It is unclear what the "E" might stand for. The dose records reflect decreasing numbers of badges issued through the early 1950s, to the point that the latter records for Plant C have no doses entered for most of the subjects, indicating that most were no longer badged and the plant was in shutdown status. At the end of 1954, only six names are listed on the film badge results card. This agrees with the statement in Klevin (1954) that in May 1954, there were only five workers left in the plant.

Most of the film badge results are listed on forms from the NYOO, with beta and gamma results for all monitored workers. Some records also list the optical densities from which the doses were calculated and some records list the total dose, computed from adding the beta and gamma together without modification. Many of the film badge records are handwritten and, for even the typewritten records, personnel names are omitted from some weeks' results; also, Harshaw dose records list individuals to whom badges are issued, apparently, without regard to their week-to-week assignment. However, because individuals are associated with consistent badge numbers from week to week, EEOICPA Dose Reconstruction Project personnel compiling individual dose histories should have little difficulty in assigning doses with no listed names. It should also be noted that badge numbers were initially listed serially, with names in alphabetical order, but as employees were added or removed from the dose monitoring roster, the badge results continued to be listed in serial order and the names departed from strict alphabetical order.

Doses were recorded as "beta" and "gamma." Consistent with the demonstrated practice at the University of Rochester for Mallinckrodt (ORAUT 2005b), the result for beta is the open-window result with the dose under the shield subtracted. The result for gamma is the dose calculated from the optical density under the shield. Confirmation by inference was possible, as it was for Mallinckrodt, because there are a few beta results for Harshaw that record beta as less than the limit of detection (LOD) and gamma as some number greater than the LOD.

As the set of monitored individuals shrank, individuals were gradually removed from the roster, often by simply lining through their names. The handwritten records often record results less than detectable as "X" in the appropriate box, and these are to be interpreted as zeros. As more and more individuals returned zero results, the handwritten Xs expanded to cover the appropriate number of columns and rows to indicate zero results (i.e., a single X could be written across multiple columns and rows). A careful distinction seems to have been made between lined-out entries in the rows and crossed-out dose results column. For the purposes of dose reconstruction, X results are to be interpreted as individually entered zeros for beta and gamma, while lined-out names are to be considered unmonitored personnel for that week. Notations in the film badge records indicate when a badge was missing, when a readout was unsuccessful, etc.

Note that the Rochester Harshaw dose summary for 1944 to 1948 (University of Rochester ca. 1948) leaves blanks in the gamma column (though never in the beta column), which are interpreted to be zero results. Individual dose report results list many zeros for gamma also, with some also recorded as "less than 50" (i.e., they are denoted as 50* with a footnote indicating that the asterisk "denotes less than"); some weekly badge records give zeroes as blanks or Xs. All these should be considered zero results for the purposes of dose reconstruction [24]. Also, most gamma results in the Rochester-generated records are rounded to the nearest 10. Those results listed with 1 to 9 in the last column might be the result of averaging because a handwritten note at the top of the dose summary file states that "these figures include average values inserted where badges were lost or readings missed for some other reason."

For 1944 to 1946, it is likely that the film badge exchange frequency was weekly, but individual records or use of the Rochester summary of dose prior to August 1947 might involve a different assumption according to the individual case. The exchange frequency should be assumed to be weekly for 1947 on, except for the several years near the end of operation when records indicate the consolidation of the last 2 weeks of a year on a single film badge.

6.3.2 Site Dosimetry Technology and Calibration

No procedures and little other film badge specification data have been found to date (e.g., there is no specific calibration information). However, because early badges were processed by Rochester, it is very likely that the calibration methods were those of MED/Rochester and later, of the AEC HASL.

There is no information to suggest calibration using a phantom, so open-air calibrations were likely performed. Thus, it is recommended that the Harshaw-recorded gamma doses be converted using dose conversion factors for roentgen-to- H_T dose for photons from Appendix B of NIOSH (2006). Because exposure to organ dose conversion factors result in a higher organ dose and higher probability of causation, given the Radiation Effective Factors of the intermediate energy photons, these dose conversion factors will be used to convert recorded film badge doses to organ dose. The low-energy component does not seem to be a significant characteristic of the Harshaw spectrum, thus no modification is proposed to recorded deep doses, once converted to organ doses.

Extremity dosimeters were not worn routinely at Harshaw. However, some information regarding extremity dose measured with films is available. The MED attempted to measure beta dose to the hands of hex loaders (from the ash) by using film strips (HCC ca. 1945; Engel 1946; Tybout 1946). In February 1946, an MED sergeant was sent to Harshaw to work for a few days as a hex tray loader and hex ash handler (Engel 1946). While working, he wore a film strip around each finger and a small piece of film in each palm, on each wrist, and on the back of the hand; all of these films were fastened on with Scotch[®] tape. Also, he wore one regular (chest) film badge and an additional chest film badge. The regular badge was worn all 4 days, but the hand films and the additional chest films were changed every day, except that only new palm badges were worn on the extremities during the ash handling operation. Although gloves were worn, perspiration and the heat of the trays rendered some of the films unreadable; still, some results were obtained as listed in Table B-6. Because of the heat problem, films were not used further as extremity dosimeters.

Although extremity dose was likely to have been high at Harshaw, the proportion of claims requiring calculation of extremity dose is unlikely to be high. Thus, this subject is not treated in detail in this site profile. Extremity dose estimates, when necessary, should be formulated on a case-by-case basis [25].

6.3.3 Workplace Radiation Fields

6.3.3.1 Beta/Photon Dosimeter Response

Both ^{235}U and ^{238}U are primarily alpha-particle emitters. However, ^{235}U does emit a 185-keV photon in 54% of its decays. Most of the external dose from ^{238}U comes from its short-lived ^{234}Th , $^{234\text{m}}\text{Pa}$, and ^{234}Pa decay products. From an external dose standpoint, the most significant radiations emitted by these decay products of ^{238}U are: (1) the 2.29-MeV beta particle from $^{234\text{m}}\text{Pa}$, and (2) the photons emitted by ^{234}Pa with energies as large as 1.962 MeV. Photons should be assumed to be in the 30- to-250 keV energy range, consistent with NIOSH (2006). Beta (electron) radiation should be assumed to be in the range greater than 15 keV. Radiation in this range is the primary external dose component for Harshaw.

Table B-5 lists measured gamma and total dose rates at various locations and times as reported by the AEC and Harshaw. Table B-6 lists chest and hand beta doses from ash residue handling as measured by film badge and films taped on the hands. Table B-7 lists weekly doses tabulated for the period from August 1944 to January 1949 by the AEC (1949b).

The total dose on film badges was more than 95% beta as noted by the AEC [1949b, 1951c (referring to 1948 to 1951)] and as shown by film badge records. The percentage might have been even higher for extremities. The AEC was concerned by Harshaw's lack of close monitoring of beta exposures and cited this several times as a failure of the Harshaw health and safety program (e.g., Kelley 1947). The AEC (1951c) noted that in 1949, about 25% of the hex plant workers received beta exposures higher than tolerance, but by 1951 that figure had dropped to 10%; similarly, in 1949, "hardly a month went by" without a number of badges registering over 1,000 mrep/wk, but by early 1951 there was only an occasional badge registering that high. This was attributed to better personnel control and to the 1950 addition of the central loading/unloading station for hex reactor trays.

Doses registered on film badges worn by people not working directly with the uranium and the process and analytical equipment, such as guards and office workers, were more likely from gamma exposure than from beta exposure. This is because these workers were usually at some distance from the source (the uranium and its progeny). It is true that uranium-containing dust was found throughout the plant to varying extents, but that would likely not contribute substantially to the external dose rate much in buildings or areas distant from the process areas [26].

In addition to the beta dose rate from the uranium as natural uranium, uranium oxide, etc., two waste concentrates produced significant beta dose rates. First, when ether was used with the uranyl nitrate to extract the uranium in the ore-to- UO_3 production process, ^{234}Th and $^{234\text{m}}\text{Pa}$ (again, UX1 and UX2, respectively) were left in the aqueous phase (Eisenbud 1975). When this aqueous solution was filtered, the resulting cake(s) contained most of these beta-gamma emitters (Eisenbud 1975). Second, the hex reactor ash, as stated above, was highly concentrated in the ^{234}Th and $^{234\text{m}}\text{Pa}$ from the UF_4 (AEC 1949b). The highest extremity doses at Harshaw were probably from this source.

6.3.3.2 Uncertainty and Bias for Beta/Photon Dosimeters

There was no quantitative information recovered for the film badge used at Harshaw. Information from other sites (Y-12, Hanford) indicates that the uncertainty for a two-element badge of the era could be estimated as $\pm 30\%$. Similarly, information on bias was not located. It is likely that some factors such as the calibration techniques could have resulted in recorded doses that were too high while other factors such as the angular response and wear location could have resulted in recorded doses that were too low for the organ(s) of interest [27].

6.3.3.3 Neutron Doses

No neutron exposure measurements are available. However, Dupree-Ellis et al. (2000) deemed neutron exposures at a similar uranium production facility (the Mallinckrodt site) to be minimal. This conclusion seems to be correct for the Harshaw UO_3 , UO_2 , and UF_4 production processes too, due to the similarity of the production processes, and likely for the UF_6 production as well. The enriched uranium hexafluoride (LEUF_6) from Hanford (Kelley 1946) and the Hanford recycle UO_3 (BJC and Haselwood Enterprises 2000) that Harshaw processed from 1952 to 1954 also do not appear to have involved significant levels of neutrons because no extra precautions appear to have been thought necessary due to the low enrichment level of the former and the relatively low transuranic content of the latter. (See Table B-3 for details of the content of the LEUF_6 and the RU.)

In analyzing neutron production by the alpha-neutron reaction, the forms of uranium that would produce neutrons at the highest rates were identified as UF_4 and UF_6 , but UO_2 , UO_3 , and U_3O_8 and the soda salt form $\text{Na}_2\text{U}_2\text{O}_7$ were also considered [28]. Little information could be found about UCl_4 , so it was considered in terms of identifying another form that would bound its contribution (i.e., UF_4). As long as an adequate amount of target material (fluorine, oxygen, sodium, or chlorine) is available and it is intimately mixed with the source material (uranium, thorium, or their alpha-emitting progeny), as would typically be true in the forms handled at Harshaw, neutron production essentially depends on the amount of the source material. The bounding assumption is therefore made that the maximum neutron emission occurs (i.e., that there is an adequate amount of target material whatever the form) [28].

As listed in Table B-3, a fiber or steel drum container of UO_3 , UO_2 , or soda salt would weigh 75 lb and a steel drum container of U_3O_8 (ore feed as black oxide) would weigh 100 lb; most of this weight would be uranium, so it can reasonably be assumed that the entire weight is uranium. While a larger volume could be found in, for example, a digest tank, the liquid and the thick tank wall would provide a great deal of shielding. A larger volume could be found in a massed array of containers, but a great deal of self-shielding would be involved and a person would likely not spend a great deal of time near an array. Thus, it is likely that the dose rate from a single container (being temporarily stored, loaded, transported, or dumped) would be the typical dose source and this was the source form analyzed. The dose rate from a massed array, however, could be estimated by multiplying the single-container dose rate by the number of containers.

In the dose rate calculations, assumptions and data from ORAUT (2005c) were used; these included the assumptions that there was a point-source geometry, the isotopic composition in the source container was that of natural uranium, and the energy of the neutrons produced was 2.0 MeV. The resulting whole-body neutron dose rates are listed in Table B-8. To consider the effect of including progeny contributions, for full equilibrium of the progeny of natural uranium down to radon (which would not be in chemical union with the target and would likely have been vented when the container was opened, the form processed, etc.) or for the extreme case to polonium, the additional contribution of the alpha-emitting progeny of ^{238}U and ^{235}U was also considered where appropriate. Thus, the progeny contribution was included in Table B-8 when there was time for the progeny to build in significantly (i.e., the ore concentrate and soda that were being sent from various other sites and the UO_2 and UF_4 that were being sent from Mallinckrodt Chemical Works in St. Louis). Note that the UF_4 appears to have been sent in significant quantities only from 1945 to 1947, constituting at most 28% of the amount used (i.e., Harshaw produced 72% of the UF_4 it used during that period and 100% at other times).

In the Harshaw-specific calculation of the annual whole-body doses performed for inclusion in this site profile, exposure time and distance estimates favorable to claimants were made based on time

measurements and estimates given in memoranda, dust studies, etc [29]. Although production increased over time, it is assumed that the process worker population grew more or less proportionately so that each worker type did the same amount of work on an annual basis (e.g., loaded or moved the same number of drums). It was assumed that the process worker-receptor spent 1 hr/d at 1 ft from the container with the uranium form in it and 3 hr/d at 3 ft from it, every working day for a 2,000-hour workyear [29]. The laboratory worker was assumed to handle much smaller sizes of containers (e.g., 2-quart sample jars and the like), taken to contain 10% of the mass of the corresponding larger containers, but to spend all of the work time near the containers (e.g., sitting on benches or hoods in the laboratory), taken to be 2 hr/d at 1 ft and 6 hr/d at 3 ft from the containers [29]. The office/clerical/ management worker was assumed to spend an insignificant amount of time near these containers, with their resulting doses being negligible; this is reasonable based on the inspection of the process workers and laboratory worker doses [29].

The occupancy time assumptions used in the calculation of Table B-8 should be adjusted for workers not likely to have spent considerable time near the uranium forms; for these workers, the doses should be ratioed by an appropriate fraction to reflect the time spent near uranium forms in bulk. A fraction of 0% is suggested for office workers and shop workers, 5% for higher level managers, and 25% for maintenance and safety workers who were likely to have spent time in process areas [based on engineering judgment, given the information in the various dust studies (including time-and-place information), observations in AEC and Harshaw memos and reports, and statements in HCC (1946)]. Note that no estimate of neutron dose for postoperations and decontamination work need be made because of the small volumetric concentrations of the residual contamination (i.e., mostly surface-type deposits on walls, floors, and equipment). Neutron radiation should be assumed to be in the range of 0.1–2.0 MeV. The neutron doses in Table B-8 should be multiplied by 2 to correct the values calculated from ORAUT (2005c) to the ICRP Publication 60 radiation weighting factor for this energy range (ICRP 1991).

6.4 ADJUSTMENTS TO RECORDED DOSE

No adjustments to recorded dose are proposed for Harshaw at this time.

6.5 MISSED DOSE

The LOD for beta reported from 9 May 1948 on AEC forms (NYOO 1948) was less than 50 mR for both beta and gamma. Since the University of Rochester read the Harshaw films up to May 1948, it will be assumed that the LOD was 60 mrep up to May 1948 and 50 mrep from May 1948 on. Thus, per NIOSH (2006), it should be assumed that beta missed dose should be applied as LOD/2, or a lognormal distribution with a mean of 0.030 rem/wk prior to May 1948 and 0.025 rem/wk thereafter, with a GSD of 1.52 for Rochester records.

The LOD for gamma dose should be assumed to be 50 mR/wk, based on records generated by the University of Rochester; these record gamma results as either zero or "less than 50" if they were less than measurable. (It is not clear why they chose to use one or the other.) It should be assumed that per NIOSH (2006), gamma missed dose should be applied as LOD/2, or a lognormal distribution with a mean of 0.025 rem/wk with a GSD of 1.52.

6.6 DETERMINATION OF EXTERNAL DOSES (INSTRUCTIONS TO DOSE RECONSTRUCTORS)

Representative external dose histories can be compiled for employees with work histories beginning after about August 1944. When available, individual film badge data should be used to determine the

exposure. Some help in interpreting the film badge records can also be found in the urinalysis records because the latter could clarify what type of work the worker was doing at a particular time. Most workers employed in the early years will have some gaps in monitoring because routine film badging did not begin until 1947 (although some data are available from 1944 on) and because there were undoubtedly some missed readouts. If individual data are not available, the exposure rate information listed in Tables B-5 through B-8 are to be used to reconstruct a claimant's external exposure.

Although some improvements were instituted from 1942 to 1946, the start of routine film badging in 1947 is correlated with significant UF_4 and UF_6 production increases or with the peak of production, and the years after 1946 mostly cover the period of significant external exposure problem reporting. Hence, it is deemed that the later measurements are reasonably representative and their use would be favorable to claimants if data gaps exist for earlier conditions. To what extent exposures during early bench-level operations differed from production-level exposures is not known. However, processes were developed on the bench and pilot plant levels and then in many cases quickly scaled up to production levels; the production levels then increased repeatedly throughout the wartime and early postwar years. So, it can be concluded that the conditions for bench-level and pilot plant operations were similar to production level operations, but on a much smaller scale (including generally much smaller source quantities) [30].

Note that many workers worked at two or more of the UO_2 -to- UF_4 , UF_4 -to- UF_6 , and ore-concentrate-to- UO_2 or - UO_3 plants, with their differing potential for external exposure; especially for beta exposure. So if it is necessary to determine the maximal job for an individual having several jobs over the course of a year, care should be taken to identify which job is indeed the maximal one for the organ of interest. Care must also be taken in the case of workers whose work histories show them apparently still doing work with radioactive materials after the D&D period; such work during the postdecontamination years was not AEC-contracted work, it is uncertain what work might have been done and what materials might have been used, and any doses found in records from the post-D&D years might be attributable to private operations with radioactive material, not to residual contamination from AEC work. But it is generally acceptable to assume that operations continued until the end of 1955 in the ore-concentrate-to- UO_3 (brown) plant and the associated laboratories. If there are no individual data available for this period, Table B-26 should be used, as explained below. See the notes following the table for information on how the table was developed.

6.7 DETERMINING EXPOSURE DURING THE OPERATIONS YEARS

6.7.1 Reconstruction of Doses from August 1944 to 1955

As noted, film badging at Harshaw began in at least intermittent fashion in August 1944, although it does not appear to have become routine until 1947. The AEC directive to badge all employees who might be subject to significant external exposure suggests that it is reasonable to assume that exposed employees engaged in MED/AEC work will have at least some film badge results for their covered employment, and individuals with no badge results are unlikely to have received anything but incidental exposure. Individual dose histories are likely to contain gaps due to missing or damaged badges and, especially for earlier periods, "fogged films."

How missing doses can be most accurately estimated is based on what dose data are available. As discussed by Watson et al. (1994), the most accurate estimate for a missing annual dose is an average of the doses recorded in the years before and after. For the purposes of dose reconstruction, this will be assumed to apply to shorter periods as well if the worker clearly worked in the same work area and did the same type of work. In such cases, data for a missing year should be filled by

averaging the dose for the year before and the year after; data for shorter periods (a missing quarter, month, or week) should be filled by averaging appropriate periods before and after the missing period.

If the worker did different work in the years or periods bracketing the missing year or period, the data for the missing period should be based on the maximally dose-producing of the two types of work. In the case that this method produces clearly adequate results or the method produces inconsistent results, exposure rates listed in Table B-5 are to be used with time estimates from dust studies (e.g., AEC 1953a) to produce reasonable exposures for comparison.

6.7.2 Reconstruction of Doses Prior to August 1944

For workers involved in uranium processing between the beginning of Harshaw work for the MED in 1942 and the beginning of monitoring in August 1944 who have later dose monitoring results, the most accurate estimate of annual doses is likely to come from dose monitoring information for later years, if available. As stated in Watson et al. (1994), dose information from the closest period should be used to estimate missing dose information. For the early period, the best information is likely in the Rochester Radiation Summary (University of Rochester ca. 1948). For individuals listed in that summary, the average dose should be applied to each year preceding the period covered by the summary. Although application of a simple average will introduce considerable uncertainty, given changes in the process and job assignments, it is likely to be an assumption favorable to claimants because production levels were steadily increasing over the period (from about 5 lb/d in 1942 to about 4,200 lb/d in June 1947, shortly before the date of the first routine individual dose monitoring results) so the period of the radiation summary covers a time with likely overall higher dose hazards than the period before the summary [31].

For workers involved in uranium processing between the beginning of Harshaw work for the MED in 1942 and the beginning of monitoring in August 1944 who have no later dose monitoring results (e.g., those who might have terminated prior to the start of monitoring), the data in Tables B-5 through B-8 should be used to establish an exposure estimate. It is unlikely that individuals with no available dose monitoring records worked in the uranium operations at Harshaw for more than a short time. However, if such a case should present itself – if the evidence indicates that the worker was present in the uranium operations and the monitoring results are missing – the data in the tables should be used to provide an external exposure estimate that is favorable to claimants.

6.7.3 Estimating Incidental Dose for Individuals Employed In Uranium Processing but Not Involved in Operations

External dose monitoring was performed for workers directly involved in the uranium processing operations, but possibly not always for workers employed in a support capacity. For example, process workers, warehouse workers, maintenance workers who entered process areas or received process equipment in their shops, safety workers, and all these workers' supervisors and managers were badged; laboratory personnel and process area clerks also were badged [32]. But it is not clear that secretaries, nonprocess clerks, and the like were badged. It is reasonable to assume that these unmonitored individuals associated with the AEC uranium operations did receive some radiation dose, however, due to the possibility that their work locations were in buildings in or near the uranium processing area. Dose rates from surveys are listed in Table B-5. The 50th percentile of this data is about 7 mrem/hr (presumably 95% or greater is beta dose based on film badge results) [33]. Weekly doses for monitored workers are listed in Table B-7, which can serve as an upper bound on dose estimates for unmonitored workers. Dose reconstructors should estimate the dose based on the information in the individual case. For example, a worker described as a laborer might be assumed to have a higher occupancy in radiation areas than a secretary.

6.8 DETERMINING EXPOSURE DURING THE POSTOPERATIONS AND D&D YEARS

Some judgment will have to be applied as to how much of the dose record after 1955 should be counted as contributing to dose from AEC decontamination operations. This is particularly true because, after the period of covered operations and D&D years, Harshaw received a license from the AEC for the use of certain radioactive materials and some workers might have continued to be badged [34]. It will be generally acceptable, however, to assume that operations continued until the end of 1955 in the ore-concentrate-to-UO₃ (brown) plant and the associated laboratories [21].

Because measured dose rates were available, RESRAD-BUILD was not used to calculate external exposures. Instead, manual calculations of annual external exposures were performed to estimate gamma and beta dose rate values that are favorable to claimants.

Annual external exposure estimates were calculated from measured data and are listed in Table B-26. The "AEC Decontamination" set should be applied for the D&D of Plant C only; the "Post-Decontamination" set should be applied for 1960 to 1989 to Plant C only; and the "Post-AEC Operations, Decontamination (Continuing Source Term)" set should be applied for 1960 to 1992 or 1960 to 1997 to the remaining buildings, as appropriate from Table 2-2. The latter set is applied as if the dose rates were continuous (i.e., not being reduced during D&D) due to the lack of dose rate data for D&D, an assumption favorable to claimants.

Energy bin assignment should be made as given in Section 6.3.3. The stay time assumptions used for Table B-26 (see the text after the table) should be adjusted for workers not likely to have spent considerable time in the areas of residual contamination, especially Plant C; the doses listed in Table B-26 should then be ratioed by an appropriate fraction to indicate a reasonable amount of time spent in the contaminated area. For example, a claimant who was a secretary in the postoperations years likely did not spend much time in the areas of significant contamination and should be assigned only a small fraction of the doses listed in Table B-26. A fraction of 5% is suggested for office workers, 10% for higher managers, and 25% for maintenance and safety workers [based on engineering judgment, given the information in the various dust studies (including time-and-place information), observations in AEC and Harshaw memos and reports, and statements in HCC (1946)].

7.0 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database.

- [1] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This observation was based on several documents. Although hex operations had ended, these documents give the impression that the area was not decontaminated immediately. The area was instead converted to other uses or held in a standby condition. It is therefore logical that it was not completely decontaminated.
- [2] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. While this statement is speculative, it is based on the common sense observation that more airborne dust would be likely during operations than during D&D. While dust could be raised

during D&D, it would most likely decrease with time unlike the high steady-state levels during continuous operations.

- [3] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. The percent of ^{235}U by weight in depleted uranium varies according to the reference used but is generally in the range of 0.2% to 0.25%. Because of this uncertainty, the specific activity of depleted uranium is rounded to one significant figure here.
- [4] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. The activity fractions of plutonium, neptunium, and technetium were chosen to be conservative. For example, BJC and Haselwood Enterprises (2000) gives the maximum fraction of plutonium as 11 and 9 ppb for 1953 and 1954, respectively. Therefore, the resultant doses from the RU contaminants are very likely to be overestimates.
- [5] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. Uranium isotopes and their progeny are primarily alpha emitters. Other emissions are not as important for internal dose.
- [6] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. An increase in the various factors mentioned would increase the exposure of AWE workers to radon.
- [7] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. A review of approximately 50% of Harshaw claimant files in 2007 did not find any X-ray records. The estimate of an initial, annual, and termination X-ray is based on limited site documentation and practices at other sites. It was judged to be more likely that some of these X-rays would be missed rather than occur more frequently. Therefore, this assumption is favorable to claimants.
- [8] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. A review of approximately 50% of Harshaw claimant files in 2007 did not find any X-ray records. Assumption of X-rays of the pelvis for process workers in the given date range is favorable to claimants.
- [9] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. If used in dose reconstructions, contaminated samples would result in a higher intake estimate and would therefore be favorable to claimants.
- [10] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This statement was added to ensure dose reconstructors do not attempt to use the tables to assign nonuranium doses during the SEC period.
- [11] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. The examples can be found in several documents in the SRDB. See for example SRDB Ref IDs 10503, 10671, 10675, 11162, and 11646.
- [12] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This statement is made based on professional judgment because of the similarity of the uranium operations at Mallinckrodt.

- [13] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. Bioassay activity fractions were calculated assuming that ^{230}Th , ^{226}Ra , and ^{210}Po are in secular equilibrium with $^{238}\text{U}/^{234}\text{U}$ and that ^{231}Pa and ^{227}Ac are in secular equilibrium with ^{235}U . The specific activities of $^{238}\text{U}/^{234}\text{U}$ and ^{235}U in natural uranium were divided by the specific activity of natural uranium to determine the fractions. Alpha air activity fractions were calculated by summing the activities of these isotopes in the bioassay activity fraction column and then dividing each activity fraction in that column by the total.
- [14] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. The coworker study was conducted using methods known to be favorable for uranium sites. Details are provided in Attachment A.
- [15] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. These tables provide the only information likely to be available for the D&D period. The methods used and assumptions are listed after each table.
- [16] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. Coworker data is not available before December 1949; therefore, Table B-16 is to be used to estimate inhalation doses for this period. Values were calculated by multiplying the DWEs in Table B-15 by the breathing rate per day and the days worked per year. DWEs were averaged when more than one value is given per year in Table B-15.
- [17] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. Coworker data is not available before December 1949; therefore, Table B-16 is to be used to estimate ingestion doses for this period. The values in this table were calculated by multiplying the annual inhalation intake by 0.021. This factor is derived from NIOSH (2004).
- [18] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This statement is based on the fact that in some cases more specific information might be available. In these cases, dose reconstructors can use the specific information rather than the Table B-16 and B-17 annual intakes.
- [19] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. These tables provide information that could be useful to the dose reconstructor to determine individual-specific factors when calculating inhalation and ingestion doses from air-sampling data.
- [20] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. Individual-specific information might or might not be available to the dose reconstructor. If, for example, the record indicates that the employee was transferred to an administrative position as of a certain date, dose calculations could end as of that date. However, it is favorable to claimants to calculate doses to the employment termination date, in lieu of such information.
- [21] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This statement is based on information presented in Section 2.0.
- [22] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. Table B-25 provides the means to assign internal doses in lieu of bioassay, which is unlikely to be available for the postoperations years. The assumptions that were used to calculate the values are given in the text following the table.

- [23] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. "Building G-1" was the term used to describe the area of most of the operations. The values are representative for most workers.
- [24] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. A review of the forms indicates that this was the probable meaning of the entries.
- [25] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. Not enough information is available to formulate a site-specific method for calculating extremity dose.
- [26] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. Levels of contamination were likely to be lower in buildings or areas distant from the process areas. See examples of dose rates in areas closer to highly contaminated areas such as offices etc. in Table B-5. It is logical that the rates in more distant areas were lower.
- [27] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This statement is based on the fact that calibration with a source of higher energy than was found in the workplace was common in the early days. This would typically result in an over-response in the dosimeter. The location and orientation of the dosimeter in relation to the radiation sources (above, behind, below), would typically cause an under-response of the dosimeter relative to the dose received.
- [28] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This is the same approach as in ORAUT (2005c).
- [29] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. See Tables B-10 to B-13 for examples of dust studies. The time and distance values here are based on professional judgment, and dose reconstructors can change them as appropriate in individual cases. See also the times for close contact in Table B-6.
- [30] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This is a reasonable default assumption since no information is available.
- [31] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. Because production levels were lower during the period before the summary, use of the summary data should be favorable to claimants.
- [32] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. See Table B-7 for examples.
- [33] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. The total dose rates (beta plus gamma) in Table B-5 were analyzed to produce the estimate of the 50th-percentile value. The statement that 95% of the total dose on film badges was beta is based on the AEC references listed in Section 6.3.3.1.
- [34] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This statement is an assumption based on the continued use of radioactive material at Harshaw as indicated by the AEC license.

- [35] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. This does not affect the fitting of the data for intake determination because all uranium isotopes behave the same biokinetically and the isotopes considered in this analysis have long half-lives relative to the assumed intake period. The ICRP Publication 68 dose coefficients (also referred to as dose conversion factors) for ^{234}U are larger than those for ^{235}U and ^{238}U . Because of the isotopic compositions of the source terms, the ^{234}U dose conversion factor will overestimate doses.
- [36] Potter, Eugene. M. H. Chew & Associates. Consultant Health Physicist. February 2007. The coworker study for Harshaw was developed using the methods described in ORAUT (2005d). The 50th-percentile intakes are assigned as the intake, and the 84th-percentile is used to determine the GSD for each intake ($\text{GSD} = 84\text{th}/50\text{th}$ percentile intake for each period), with the exception that no GSD less than 3 may be assigned.

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GLOSSARY

air kerma

Air kerma means kerma in a given mass of air. Kerma means the sum of the initial energies of all the charged particles liberated by uncharged ionizing particles in a material of given mass. Kerma is closely related to the energy absorbed per unit mass (absorbed dose). See *rad*.

background radiation (also background or natural background)

Radiation from cosmic sources, naturally occurring radioactive materials including naturally occurring radon, and global fallout from the testing of nuclear explosives. Background radiation does not include radiation from source, byproduct, or Special Nuclear Materials regulated by the U.S. Nuclear Regulatory Commission. The average individual exposure from background radiation is about 360 millirem per year.

beam quality

A measure of an X-ray beam's ability to provide useful diagnostic information without unnecessary exposure to the patient. Usually expressed in *half-value layers* of aluminum.

beta radiation

Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron. Most of the direct fission products are (negative) beta emitters. Exposure to large amounts of beta radiation from external sources can cause skin burns (erythema), and beta emitters can be harmful inside the body. Thin sheets of metal or plastic can stop beta particles.

contamination, radioactive (also residual contamination)

Radioactive material in an undesired location including air, soil, buildings, animals, and persons.

curie (Ci)

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

daily weighted (average) exposure

The average concentration calculated by summing the products of the concentration measured by an air sampler and exposure time (in hours) for each period or task in a day and dividing by the total time per day (typically 8 hours).

decontamination

Reduction or removal of radioactive material from a structure, area, object, or person. Decontamination can occur through (1) treating the surface to remove or decrease the contamination or (2) allowing natural radioactive decay to occur over a period of time.

depleted uranium (DU)

Uranium with a percentage of ^{235}U lower than the 0.7% found in natural uranium. As examples, spent (used) fuel elements, byproduct tails, residues from uranium isotope separation, and some weapons materials contain DU. DU can be blended with highly enriched uranium to make reactor fuel or used as a raw material to produce plutonium.

dose

In general, the effects of ionizing radiation in terms of the specific amount of energy absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rads, rems, or grays.

dosimeter

Device that measures the quantity of received radiation, usually a holder with radiation-absorbing filters and radiation-sensitive inserts packaged to provide a record of absorbed dose received by an individual.

dosimetry

Measurement and calculation of internal and external radiation doses.

enriched uranium

Uranium in which processing has increased the proportion of ^{235}U to ^{238}U to above the natural level of 0.7%. Reactor-grade uranium is usually about 3.5% ^{235}U ; weapons-grade uranium contains greater than 90% ^{235}U .

equilibrium factor

A measure of the degree of radioactive equilibrium between radon and its short-lived radioactive decay products. The equilibrium factor may be expressed as a fraction or a percent. For example, an equilibrium factor of 0.4 (40%) means that the concentration of short-lived radioactive decay products is 0.4 of the radon concentration.

exposure

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

extremity

That portion of the arm extending from and including the elbow through the fingertips, and that portion of the leg extending from and including the knee and patella through the tips of the toes.

film

Radiation-sensitive photographic film in a light-tight wrapping.

fission

Splitting of the nucleus of an atom (usually of a heavy element) into at least two other nuclei and the release of a relatively large amount of energy. This transformation usually releases two or three neutrons.

fission product

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

gamma radiation

Electromagnetic radiation (photons) of short wavelength and high energy (10 kiloelectron-volts to 9 megaelectron-volts) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma rays are very penetrating, but dense materials such as lead or uranium or thick structures can stop them.

Gamma photons are identical to X-ray photons of high energy; the difference is that X-rays do not originate in the nucleus.

half-life

Time in which half of a given quantity of a particular radionuclide disintegrates (decays) into another nuclear form. During one half-life, the number of atoms of a particular radionuclide decreases by one half. Each radionuclide has a unique half-life ranging from trillionths of a second to billions of years.

half-value layer

Thickness of a specified material (usually aluminum for X-rays) which reduces the exposure rate to one-half of its initial value.

irradiate

To expose to ionizing radiation.

isotope

One of two or more atoms of a particular element that have the same number of protons (atomic number) but different numbers of neutrons in their nuclei (e.g., ^{234}U , ^{235}U , and ^{238}U). Isotopes have very nearly the same chemical properties but often have different physical properties.

limit of detection (LOD)

The lowest quantity of radiation exposure or dose that can be distinguished from background. Measurements below the LOD are usually recorded as zero, but may be noted in some other way (for example, "< 50" or blank entries).

natural uranium

Uranium as found in nature, approximately 99.27% ^{238}U , 0.72% ^{235}U , and 0.0054% ^{234}U by weight. The specific activity of this mixture is 2.6×10^7 becquerel per kilogram (0.7 microcuries per gram).

neutron

Basic nucleic particle that is electrically neutral with mass slightly greater than that of a proton. There are neutrons in the nuclei of every atom heavier than normal hydrogen.

nuclide

Stable or unstable isotope of any element. Nuclide relates to the atomic mass, which is the sum of the number of protons and neutrons in the nucleus of an atom. A radionuclide is an unstable nuclide.

occupancy factor

The fraction (or percentage) of time that a given area is occupied by workers.

open window

Area on film dosimeter that implies the use of little (i.e., only security credential) shielding over the film. Commonly used to label the film response corresponding to the open-window area on dose reports.

personal dose equivalent [Hp(d)]

Represents the dose equivalent in soft tissue below a specified point on the body at an

appropriate depth *d*. The depths selected for personnel dosimetry are 0.07 mm and 10 mm for the skin and body, respectively. These are noted as Hp(0.07) and Hp(10), respectively.

photon

Basic unit of electromagnetic radiation. Photons are massless “packages” of light energy that range from low-energy microwave photons to high-energy gamma rays. Photons have energies between 10 and 100 kiloelectron-volts.

rad

Traditional unit for expressing absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium. A dose of 1 rad is equivalent to the absorption of 100 ergs per gram (0.01 joules per kilogram) of absorbing tissue. The rad has been replaced by the gray in the International System of Units (100 rads = 1 gray). The word derives from radiation absorbed dose.

radiation

Subatomic particles and electromagnetic rays (photons) that travel from one point to another, some of which can pass through or partly through solid materials including the human body.

radioactive

Giving off ionizing radiation such as alpha particles or X-rays.

radioactivity

Disintegration of certain elements (e.g., radium, actinium, uranium, and thorium) accompanied by the emission of alpha, beta, gamma, and/or neutron radiation from unstable nuclei. See *radionuclide*.

radioactive waste

Radioactive solid, liquid, and gaseous materials for which there is no further use. Wastes are generally classified as high-level (with radioactivity as high as hundreds of thousands of curies per gallon or cubic foot), low-level (in the range of 1 microcurie per gallon or cubic foot), intermediate level (between these extremes), mixed (also contains hazardous waste), and transuranic.

radionuclide

Radioactive nuclide. See *radioactive* and *nuclide*.

reactor

A container or vessel in which a chemical reaction takes place. Not a nuclear reactor as it is used in this document.

recycled uranium

Uranium from spent nuclear fuel from Government reprocessing plants at the Hanford, Savannah River, and Idaho sites and also at the commercial West Valley site. These plants recovered plutonium and uranium from spent nuclear fuel and target material irradiated in nuclear reactors. Recycled uranium contains trace amounts of fission products, activation products, and transuranic elements.

Redox material

Uranium recovered at Hanford using a solvent chemical extraction separation technique, the Reduction-Oxidation process.

rem

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The average American receives 360 millirem a year from background radiation. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

rep

An early unit of absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium. A dose of 1 rep is equivalent to the absorption of 93 ergs per gram of absorbing tissue. It is approximately equal to 1 roentgen of 250 kVp X-radiation in soft tissue, or 0.93 rads, or 9.3 milligray. The rep was replaced by the *rad*.

reprocessing

Normally mechanical and chemical processing of spent nuclear fuel to separate useable fissionable products (i.e., uranium and plutonium) from waste material. At Harshaw this term applies to recycled uranium material (not spent fuel) received from Hanford for further purification.

roentgen (R)

Unit of photon (gamma or X-ray) exposure for which the resultant ionization liberates a positive and negative charge equal to 2.58×10^{-4} coulombs per kilogram (or 1 electrostatic unit of electricity per cubic centimeter) of dry air at 0° Celsius and standard atmospheric pressure. An exposure of 1 roentgen is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher energy photons (generally greater than 100 kiloelectron-volts).

shielding

Material or obstruction that absorbs ionizing radiation and tends to protect personnel or materials from its effects.

skin dose

Dose equivalent at a depth of 0.007 cm in tissue.

specific activity

A measure of radioactivity per unit mass, such as pCi/g.

spent fuel

Fuel that has been in a reactor long enough to become ineffective because the proportion of fissile material has dropped below a certain level.

U.S. Atomic Energy Commission (AEC)

Federal agency created in 1946 to assume the responsibilities of the Manhattan Engineer District (nuclear weapons) and to manage the development, use, and control of nuclear energy for military and civilian applications. The Energy Research and Development Administration and the U.S. Nuclear Regulatory Commission assumed separate duties from the AEC in 1974. The U.S. Department of Energy succeeded the Energy Research and Development Administration in 1979.

waste

See *radioactive waste*.

working level (WL)

Any combination of short-lived radon decay products in one liter of air that will result in the ultimate emission of 130,000 MeV of potential alpha energy. Approximately the total alpha energy released from the short-lived decay products in equilibrium with 100 pCi of Rn-222 per liter of air.

working level month (WLM)

A unit of exposure used to express the accumulated human exposure to radon decay products. 1 WLM = 1 WL exposure for 170 hours.

whole body dose

Commonly defined as the absorbed dose at a tissue depth of 1.0 cm (1,000 mg/cm²); however, also used to refer to the recorded dose.

X-ray radiation

Penetrating electromagnetic radiation (photons) of short wavelength (0.001 to 10 nanometers) and energy less than 250 kiloelectron-volts. X-rays usually come from excitation of the electron field around certain nuclei. Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.

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Due to the limited availability of bioassay data for use at the Harshaw site, it was necessary to conduct a coworker study of all the bioassay data for use in determining intake estimates. In short, the data used in this study were transcribed directly from hardcopy (or electronic copy) into worksheets. The data in the worksheets were verified (as indicated below), a statistical analysis conducted (and verified), and intake assessment conducted (and verified). Each of these processes is further described below. The resulting intake tables are provided in the intake assessment section.

A.1 DATA VERIFICATION

The Harshaw bioassay data were verified as follows.

1. Data were transcribed directly from source documents to spreadsheets by Data Entry personnel.
2. The transcribed data entered into eight spreadsheets were evaluated for acceptability using the statistical sample procedure of "Sampling by Attributes," which is based on DOE (1989). Spreadsheets were deemed acceptable when they passed a completeness and accuracy quality control review. These reviews were conducted by comparing the data on the individual source documents to the transcribed data.
3. The completeness quality control review was a review to ensure that the transcribed data reflected the total amount of data that was available for entry (i.e., the number of individual data items on a page of the source document was actually entered onto the spreadsheet for that page). If this was acceptable, the spreadsheet accurately reflected the amount of data present in the source document.
4. After the completeness quality control review indicated that the spreadsheet was acceptable, a quality control review for accuracy was performed. The accuracy quality control review ensured that the data entered into the spreadsheet accurately reflected the information from the source document. If this was acceptable and the completeness review was acceptable, there was reasonable assurance that the data contained in the spreadsheet accurately reflected the data from the source document.
5. After the data contained in individual spreadsheets were determined to accurately reflect the applicable individual source document information, a review to determine whether duplicate entries existed was conducted. The multiple spreadsheets were consolidated into a single spreadsheet that included all the data that were reviewed. Data contained in this consolidated spreadsheet were then sorted by sample date, last name, and first name. Once this was done, the data were reviewed to determine if duplicate data had been entered from source documents containing redundant information. As potential duplicate source document pages were identified, they were verified by viewing each potential duplicate page simultaneously (i.e., they were tiled to allow viewing of both pages). If it was determined that the pages were duplicates, they were evaluated for legibility, with the most legible data becoming the "original" and the poorer copy becoming the "duplicate." The consolidated spreadsheet was then sorted by file name (or Ref ID) and page number; and the information from the previously noted "duplicate" page was deleted.

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6. This process was repeated multiple times until each source document had been incorporated into the consolidated spreadsheet and no additional duplicate pages were noted.

Table A-1 identifies the parameters used to verify the Harshaw bioassay data in accordance with MIL-STD-105E (DOD 1989); Table A-2 lists the details of the values for those parameters for the 4% sample.

Table A-1. Parameters for Harshaw data verification.

Values Used for Sampling by Attributes (MIL-STD-105E^a)			
Parameter	Definition	Value	
		Completeness review	Accuracy review
Batch Size	The batch size is the number of items in a lot or a batch.	Number of pages in source document	Number of lines of data entered into spreadsheets
AQL	The maximal percent of nonconforming items (or the maximal number of nonconformities per 100 items), which is considered, for inspection purposes, as a satisfying process mean.	4% (Set by Task 3)	4% (Set by Task 3)
Inspection Level ^b	The inspection level determines the relation between the batch size and sample size.	III	III

- a. DOD (1989).
- b. Inspection Levels I, II, and III are *general inspection levels*:
 - Level I requires about half the amount of inspection as level II, and is used when reduced sampling cost are required and a lower level of discrimination (or power) can be tolerated.
 - Level II is designated as Normal.
 - Level III requires about twice the amount of inspection as level II, and is used when more discrimination (or power) is needed.

A.2 STATISTICAL ANALYSIS

The verified data were analyzed according to the requirements within ORAUT-OTIB-0019, *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005d) and ORAUT-PROC-0095, *Generating Summary Statistics for Coworker Bioassay Data* (ORAUT 2006b). The data analysis report follows.

A.3 INTRODUCTION

This is a report on the validation check of the 1947 through 1953 summary statistics for Uranium Urine data collected from Harshaw. The summary statistics were compiled annually for 1947 through 1949; the remaining years were compiled quarterly.

For each period, the geometric mean, GSD, and number of samples were calculated independently by two individuals and separate spreadsheets created and results compared. The spreadsheets were created in accordance to the methodology in ORAUT (2005d) and ORAUT (2006b).

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Table A-2. MIL-STD-105E (DOD 1989) 4% AQL sampling of Harshaw bioassay data files.

Name of spreadsheet	Harshaw Chemical Co-Bio-1949-1953-SAGJ-EmlBx15 HarshawUrineEarly1950s	Harshaw-Bio-Per-1950-51-CSW-SA-TMB-KH-UrineUrAnalysesPeople Data	Harshaw Chemical-Bioassay-1950-SA-Eml2Bx29 Harshaw Dust1948-50	Harshaw Chemical Co-Bio 1950-GAJ-EmlBox4MedHealth&Safety3_9-Gasior Corresp	Harshaw Chemical Co-Bioassay-Per-47 to 48-SA-250000064	Harshaw Chemical Co-Bio-Med-Pers-48&50-GAJ-Eml2Box24HarshawChemCo PlantC_People Data	HarshawChemical & Mallinckrodt-Bio-Per-1952 to 1953-csw-EML2Box14HarshawChemUrine1952to55	HARSHAW CHEMICAL CO-Middlesex-Sylvania-NYOO-Vitro-MCW-Simonds-Bio-Per-Ext-1947-1950-MDE-Eml2BX29 Harshaw Urine & Feces 1947-50	
Total number of pages in source document	1,052	155	102	11	9	46	245	245	122
Minimum sample size	125	50	32	5	5	20	50	50	32
Total number of pages reviewed in source document during the completeness review	1,052	50	102	11	9	46	50	245	104
Maximum error rate to achieve 4% AQL	8	3	2	0	0	1	3	3	2
Number of pages having incorrect number of data entry items listed for the page	0	0	0	0	0	0	0	0	3
Completeness review completed successfully?	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Total number of lines of data entry in the spreadsheet	5,935	1,689	8	16	64	8	1,623	1,623	2,831
Minimum sample size	315	200	5	16	20	5	200	200	200
Total number of lines of data entry reviewed during the accuracy review	315	200	8	16	64	6	200	200	629

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Name of spreadsheet	Harshaw Chemical Co-Bio-1949-1953-SAGJ-EmlBx15 HarshawUrineEarly1950s	Harshaw-Bio-Per-1950-51-CSW-SA-TMB-KH-UrineUrAnalysesPeople Data	Harshaw Chemical-Bioassay-1950-SA-Eml2Bx29 Harshaw Dust1948-50	Harshaw Chemical Co-Bio 1950-GAJ-EmlBox4MedHealth&Safety3_9-Gasior Corresp	Harshaw Chemical Co-Bioassay-Per-47 to 48-SA-250000064	Harshaw Chemical Co-Bio-Med-Pers-48&50-GAJ-Eml2Box24HarshawChemCo PlantC_People Data	HarshawChemical & Mallinckrodt-Bio-Per-1952 to 1953-csw-EML2Box14Hars hawChemUrine1952to55	HARSHAW CHEMICAL CO-Middlesex-Sylvania-NYOO-Vitro-MCW-Simonds-Bio-Per-Ext-1947-1950-MDE-Eml2BX29 Harshaw Urine & Feces 1947-50	
Maximum error rate to achieve 4% AQL	18	12	0	1	1	0	12	12	12
Total number of lines of data entry that were found to have at least one error	11	2	0	0	0	0	14	2	4
Accuracy review completed successfully?	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes
Reviewer	RM	EP	EP	EP	EP	EP	HWJ	BF	SW

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Methods

Data were supplied in a spreadsheet (ORAUT 2007). All samples without a value for either "sample end date" or "sample conc" were excluded. In addition, one value in 1952 that included a "?" was excluded.

Independent spreadsheets were created by two individuals to compute the relevant statistics and compared.

Results

Periods, Effective Bioassay dates, and units appear to be correct. The comparison of the two spreadsheets resulted in a difference of within one-tenth of 1%.

A.4 INTAKE ASSESSMENT

A lognormal distribution was assumed for the urinary excretion data and the 50th- and 84th-percentile uranium (mass) excretion rates were calculated using the method prescribed in ORAUT (2005d) and ORAUT (2006b). These excretion rates are listed in Table A-3. The uranium mass excretion rates were converted to the uranium activity excretion rates in Table A-4 by applying the specific activity of natural uranium (0.68296 pCi/μg) and a urination rate of 1.4 L/d. Bioassay data collected over a specified period are analyzed to determine the 50th- and 84th-percentile excretion rates for that period. The effective bioassay dates are the midpoints of the periods and they are used in the Integrated Modules for Bioassay Analysis (IMBA) software to calculate the intake rates. Data collected for 1947 through November 1949 are not considered to be reliable (see Section 5.3.1.1) and were not used to determine the intake rates.

Table A-3. Uranium mass urinary excretion data (mg U/liter).

Period	Effective bioassay date	50th percentile (mg/L)	84th percentile (mg/L)
1949 Dec	12/15/49	0.1990	0.7163
1950 Q1	2/15/50	0.1373	0.6173
1950 Q2	5/15/50	0.0948	0.4674
1950 Q3	8/15/50	0.0367	0.1221
1950 Q4	11/15/50	0.0406	0.1610
1951 Q1	2/15/51	0.0383	0.1830
1951 Q2	5/15/51	0.0367	0.1327
1951 Q3	8/15/51	0.0428	0.1630
1951 Q4	11/15/51	0.0348	0.1205
1952 Q1	2/15/52	0.0207	0.0403
1952 Q2	5/15/52	0.0124	0.0345
1952 Q3	8/15/52	0.0099	0.0244
1952 Q4	11/15/52	0.0142	0.0296
1953 Q1	2/15/53	0.0139	0.0358
1953 Q2	5/15/53	0.0079	0.0238
1953 Q3	8/15/53	0.0072	0.0185
1953 Q4	11/15/53	0.0058	0.0189

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A.5 INTAKES

All urinary excretion rates were modeled as normally distributed 24-hr urine samples with a uniform absolute error of 1 (which results in all results being weighed equally). The excretion data were modeled with IMBA Expert ORAU-Edition for multiple chronic intakes of type F, type M, or type S uranium. Plots of expected and observed urinary excretion from these fits are shown in Figures A-1 through A-10. While it may be unlikely for all workers at Harshaw to be chronically exposed to uranium, it will approximate a series of acute intakes with unknown intake dates. Intakes were assumed to be via inhalation using a default breathing rate of 1.2 m³/hr and a 5- μ m activity median aerodynamic diameter particle-size distribution.

Because uranium has a very long half-life and because the type S material is retained in the body for long periods, excretion results are not independent. To avoid potential underestimation of intakes for people who worked for relatively short periods, each type S intake period was fit independently, using only the bioassay results from that intake period. This will result in a best estimate of dose if the person works in only one period and can result in an overestimate if an individual works in multiple periods.

Table A-4. Uranium activity urinary excretion data (pCi/d).

Period	Effective bioassay date	50th percentile (pCi/d)	84th percentile (pCi/d)
1949 Dec	12/15/49	190.29	684.86
1950 Q1	2/15/50	131.23	590.22
1950 Q2	5/15/50	90.67	446.92
1950 Q3	8/15/50	35.05	116.73
1950 Q4	11/15/50	38.78	153.92
1951 Q1	2/15/51	36.66	174.96
1951 Q2	5/15/51	35.13	126.90
1951 Q3	8/15/51	40.96	155.87
1951 Q4	11/15/51	33.25	115.19
1952 Q1	2/15/52	19.81	38.49
1952 Q2	5/15/52	11.81	33.00
1952 Q3	8/15/52	9.46	23.32
1952 Q4	11/15/52	13.57	28.28
1953 Q1	2/15/53	13.26	34.19
1953 Q2	5/15/53	7.59	22.73
1953 Q3	8/15/53	6.88	17.64
1953 Q4	11/15/53	5.58	18.07

The intake rates, GSDs, and periods in which they are applicable are listed in Table A-5 for type F uranium, Table A-6 for type M uranium, and Table A-7 for type S uranium. The fits to the data are shown in Figures A-1 to A-10. The natural uranium is assumed to be 100% ²³⁴U for the purpose of calculating dose [35].

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Table A-5. Chronic intake rates for type F ^{234}U .

Start date	End date	50th percentile (pCi/d)	84th percentile (pCi/d)	GSD
12/1/1949	3/31/1950	650.3	2,607	4.01
4/1/1950	12/31/1951	157	650.5	4.14
1/1/1952	12/31/1953	37.43	86.85	2.32

Table A-6. Chronic intake rates for type M ^{234}U .

Start date	End date	50th percentile (pCi/d)	84th percentile (pCi/d)	GSD
12/1/1949	3/31/1950	3,934	16,220	4.12
4/1/1950	12/31/1951	460.1	1,830	3.98
1/1/1952	12/31/1953	115.9	201.7	1.74

Table A-7. Chronic intake rates for type S ^{234}U .

Start date	End date	50th percentile (pCi/d)	84th percentile (pCi/d)	GSD
12/1/1949	3/31/1950	19,910	79,860	4.01
4/1/1950	12/31/1951	3,651	18,070	4.95
1/1/1952	12/31/1953	1,071	2,655	2.48

In most cases, doses to be assigned to individuals potentially exposed on a routine basis are calculated from the 50th-percentile intake rates assuming the solubility type that results in the largest probability of causation [36].

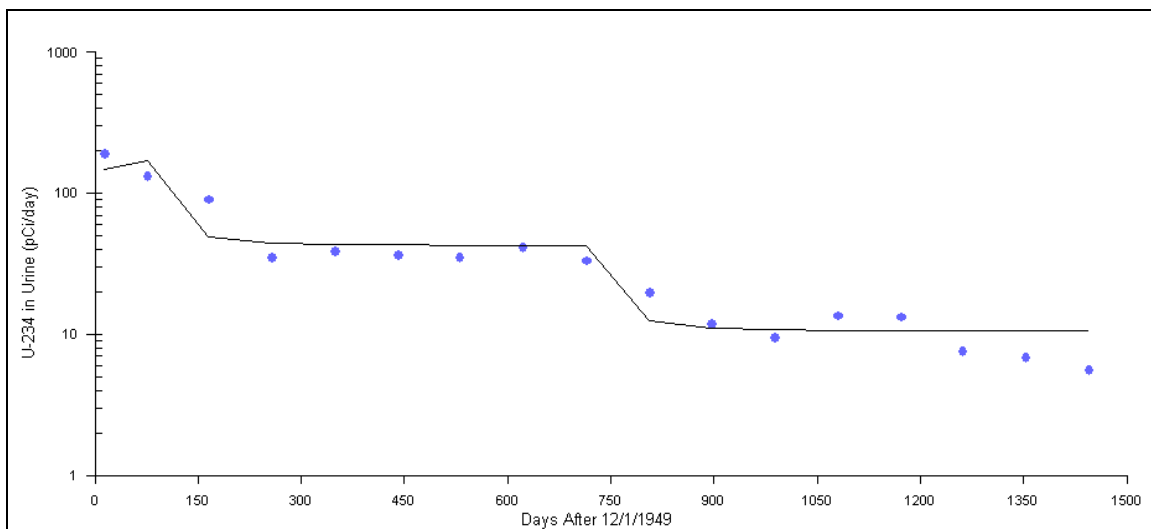


Figure A-1. Type F 50th percentile for 12/1/1949 to 12/31/1953.

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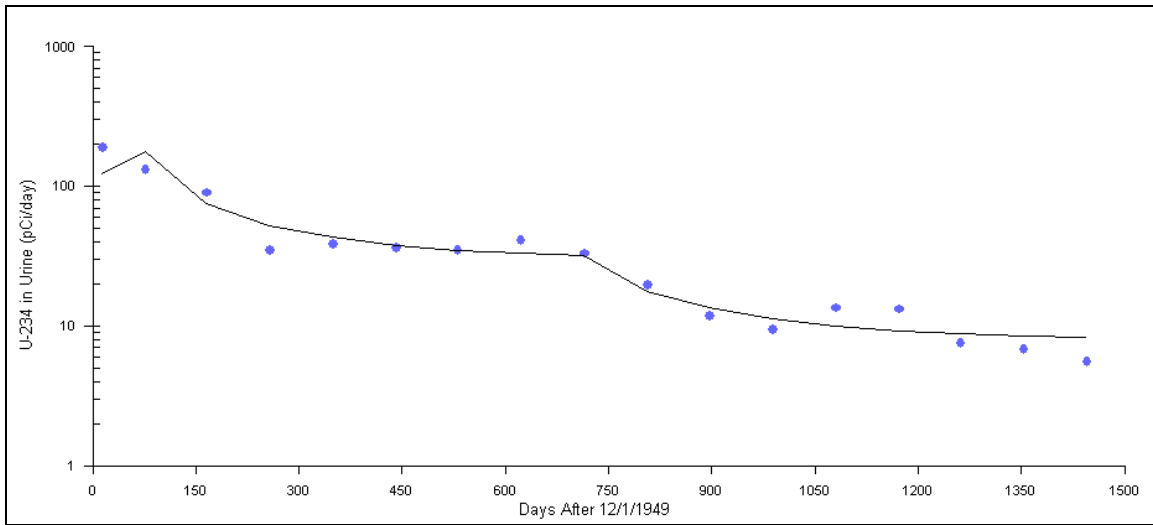


Figure A-2. Type M 50th percentile for 12/1/1949 to 12/31/1953.

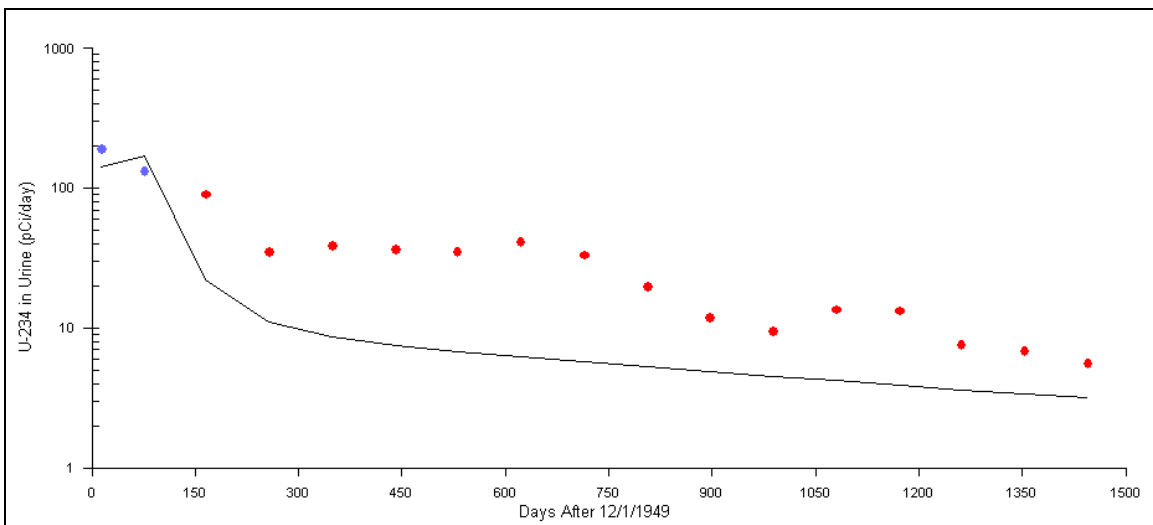


Figure A-3. Type S 50th percentile for 12/1/1949 to 3/31/1950.

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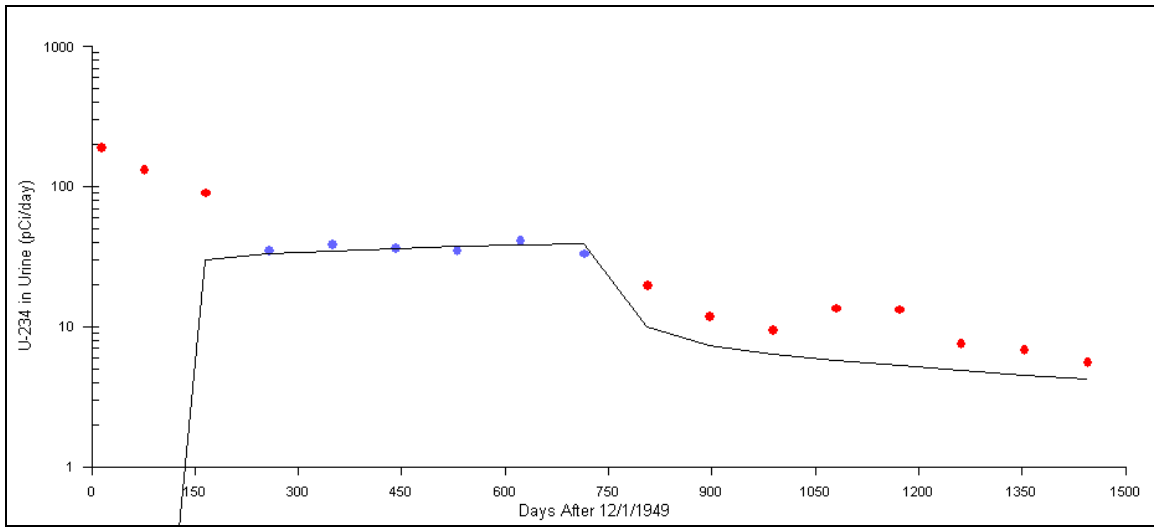


Figure A-4. Type S 50th percentile for 4/1/1950 to 12/31/1951.

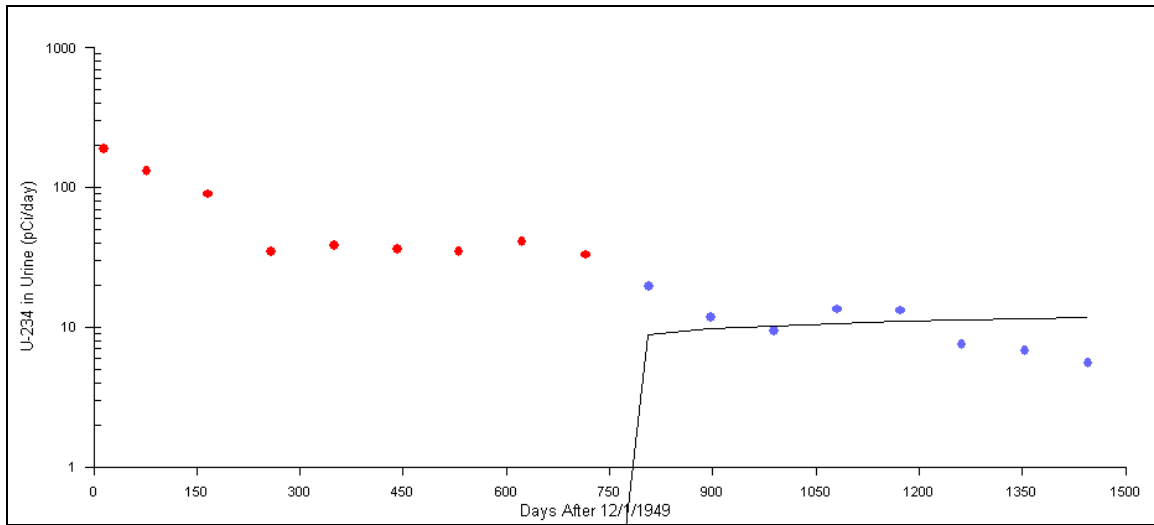


Figure A-5. Type S 50th percentile for 1/1/1952 to 12/31/1953.

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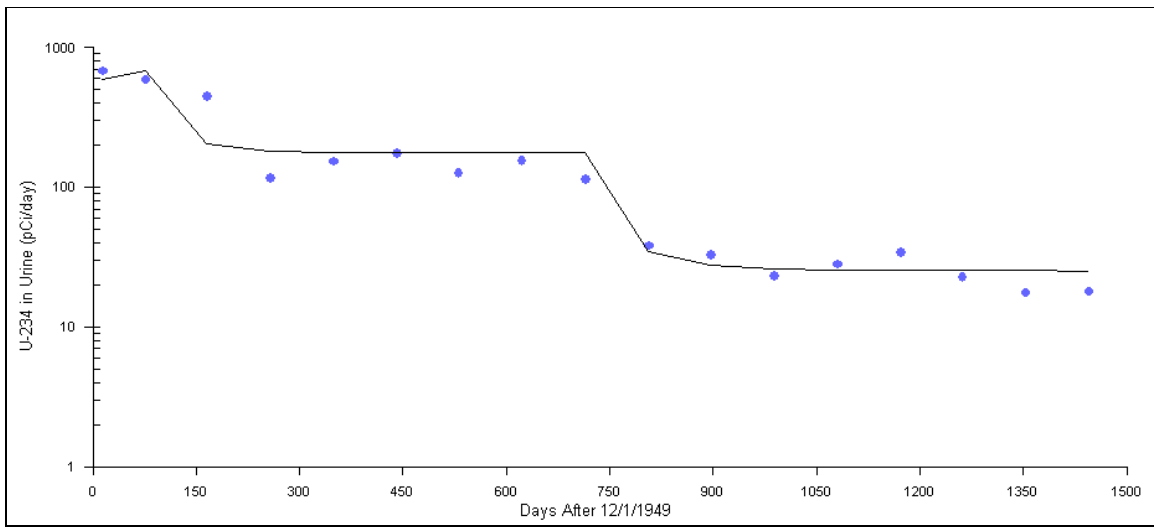


Figure A-6. Type F 84th percentile for 12/1/1949 to 12/31/1953.

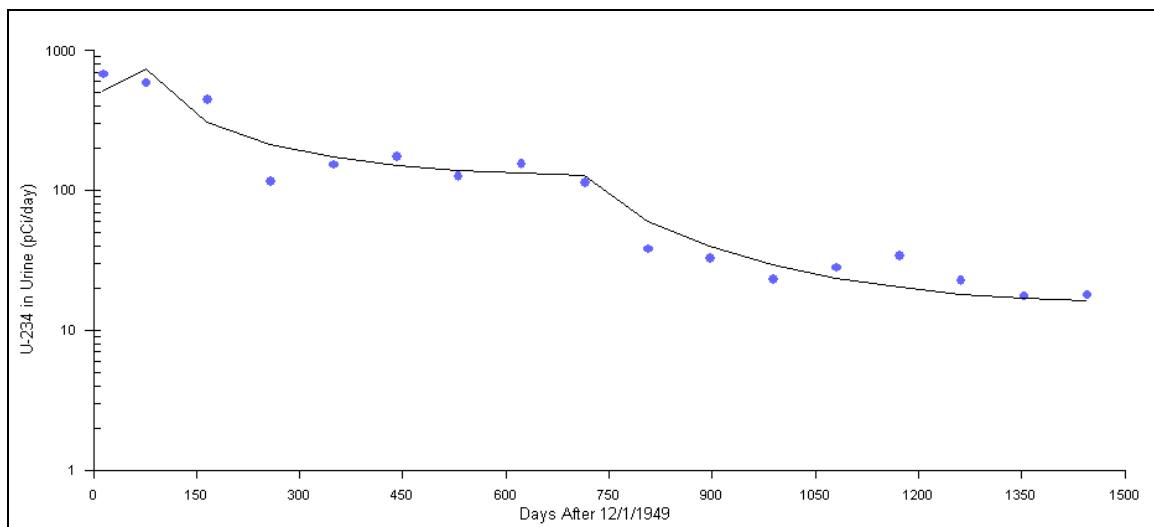


Figure A-7. Type M 84th percentile for 12/1/1949 to 12/31/1953.

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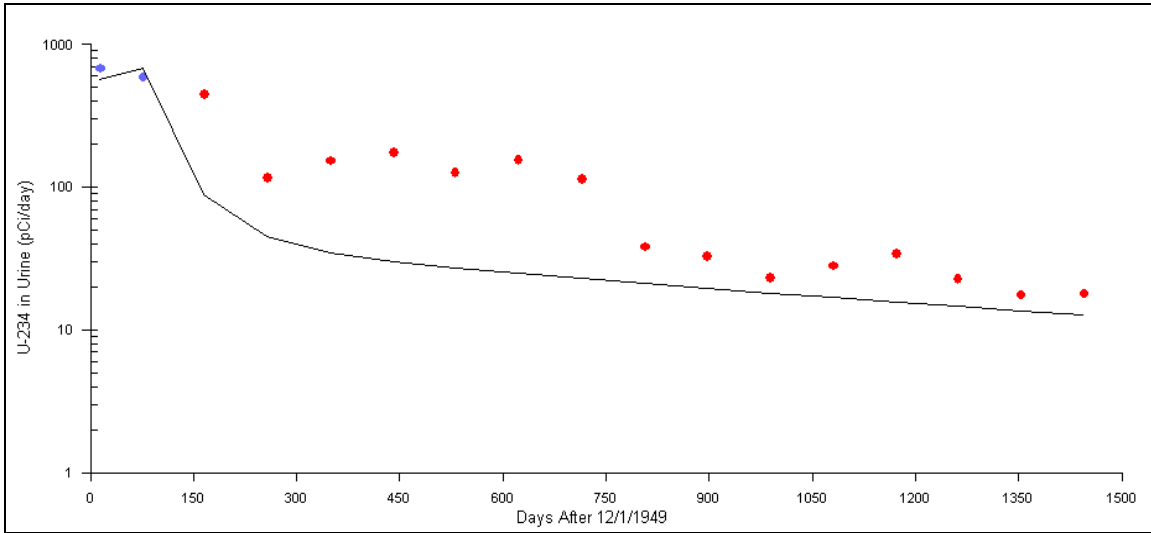


Figure A-8. Type S 84th percentile for 12/1/1949 to 3/31/1950.

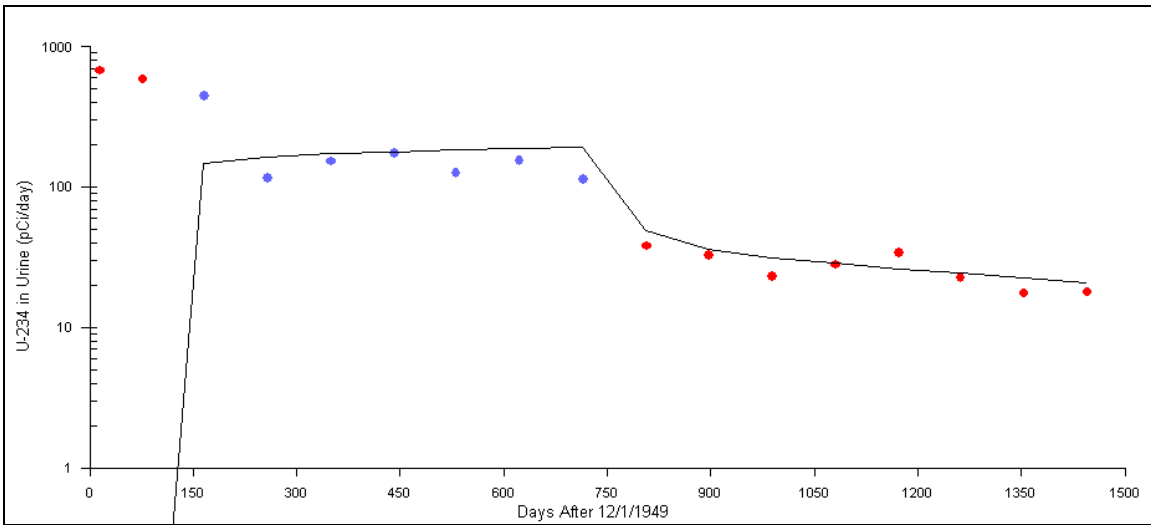


Figure A-9. Type S 84th percentile for 4/1/1950 to 12/31/1951.

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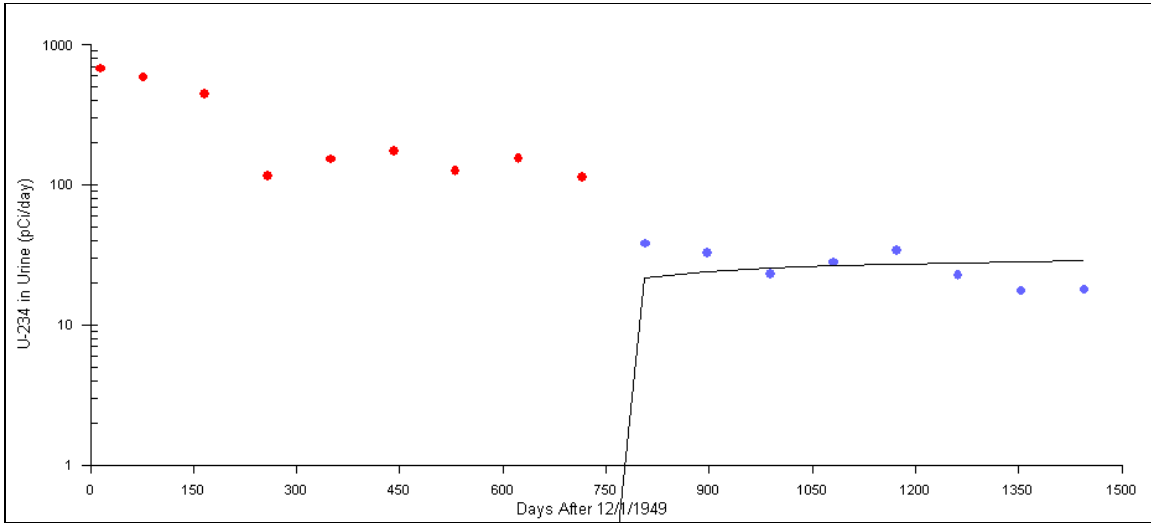


Figure A-10. Type S 84th percentile for 1/1/1952 to 12/31/1953.

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Table B-1. Chronology of Harshaw site operations based on available reference material.

Event	Form	Date	Brief description	Level	Amount
Columbia University asks Harshaw to make UF ₆ using F ₂	UF ₆	Oct 41	UF ₆ production request	Laboratory	
National Bureau of Standards orders experimental quantities of UF ₆	UF ₆	Dec 41	UF ₆ production request	Laboratory	
Production of UCl ₄ begins	UCl ₄	Jan 42	UCl ₄ production begins	Laboratory	
Production of UF ₆ begins	UF ₆	Feb 42	UF ₆ production begins	Laboratory	5 lb/d (1942 only)
First shipment of UCl ₄	UCl ₄	Mar 42	First shipment of UCl ₄	Laboratory	
UF ₄ production (dry process, for Standard Oil)	UF ₄	11 Mar 1942	UF ₄ production begins	Laboratory	Batches: tens of lb
First shipment of UF ₆ to Columbia Univ (later to Naval Research Lab., Nat'l Bureau of Stds)	UF ₆	18 Mar 1942	First shipment of UF ₆	Laboratory	
First shipment of UF ₄ , to Westinghouse Elec.	UF ₄	23 Mar 1942	First shipment of UF ₄	Laboratory	20 lb
Westinghouse asks Harshaw to make UF ₄ ; sends special oxide as feed	UF ₄	14 Apr 1942; May 1942	Increased UF ₄ production	Laboratory	About 10 lb
Second shipment of UF ₄ , to Westinghouse,	UF ₄	11 May 1942	Second shipment of UF ₄	Laboratory	About 10 lb
Third shipment of UF ₄ , to Princeton University	UF ₄	8 Jun 1942	Third shipment of UF ₄	Laboratory	
MED directs Harshaw to produce UF ₄ . Small-scale production in laboratory, then plant built	UF ₄	11 Jul 1942	Significant UF ₄ production increase	Laboratory	Tens of lb/d, then to 1,200 lb/d
Making of UF ₄ in production-level plant begins	UF ₄	22 Sep 1942	Sustained UF ₄ production	Production	1,200–1,800 lb/d
Laboratory production of UCl ₄ for University of California, with production rate increasing	UCl ₄	Nov 1942	Sustained UCl ₄ production begins	Laboratory	Up to 100 lb/d
Harshaw constructs pilot plant to produce UF ₆	UF ₆	Mar 1943	Sustained UF ₆ production	Pilot plant	50 lb/d
Peak UCl ₄ production for U. of California	UCl ₄	Apr 1943	Peak UCl ₄ production	Laboratory	100 lb/d
Harshaw manufactures special materials, including uranium oxyfluoride	UO ₂ F ₂ ; Misc	1943-1944	Special material production	Laboratory	
MED and Harshaw agree to build UF ₆ plant to produce 5,000 lb/d	UF ₆	Jan 1944	UF ₆ plant planned	Production	
Harshaw closes the UF ₆ pilot plant	UF ₆	Feb 1944			Total of 9,000 lb
Two units of new UF ₆ plant go into operation	UF ₆	Apr 1944	Volume production of UF ₆	Production	400 lb/d
Harshaw begins work on UF ₆ analysis, recovery (improvements applied in new plant)	UF ₆	Spring 1944	UF ₆ yield and recovery investigations begin	Laboratory	
Building G-1 completed		Jul 1944	Building G-1 operation	Production	
Full-scale production achieved in new UF ₆ plant (six sets of reactors in Building G-1)	UF ₆	Jul 1944	Significant UF ₆ production increase	Production	Believed to be 2,000 lb/d
Harshaw begins research on recovery of UF ₆ from spent carbon trap residues	UF ₆	Aug 1944	Another UF ₆ recovery investigation begins		
Film badging starts		Aug 1944	Film badging		
New UF ₆ reactor design plan	UF ₆	Sep 1944	UF ₆ production changes	Production	
Harshaw agrees to set up new plant for UCl ₄	UCl ₄	15 Oct 1944	UCl ₄ production plant		1 ton/d or more
Shipment of first ton of UCl ₄ from UCl ₄ plant	UCl ₄	30 Oct 1944	UCl ₄ production		
Expansion in new UF ₆ plant (12 sets reactors)	UF ₆	Nov 1944	UF ₆ production increase	Production	Up to 4,600 lb/d
UF ₄ production moves to Bldg G-1, increases	UF ₄	Dec 1944	UF ₄ production increase	Production	3,000 lb/d
MED orders additional UCl ₄ by mid-Feb 1945	UCl ₄	1 Jan 1945	UCl ₄ production extension	Production	65,000 lb (add'l)
UCl ₄ production work completed in February 1945, equipment dismantled	UCl ₄	Feb 1945	UCl ₄ production ends	Production	
UF ₆ production increase	UF ₆	1 Apr 1945	UF ₆ production increase	Production	4,500 lb/d
MED orders shipment to Harshaw of slightly enriched UF ₆ , to be mixed with normal UF ₆	LEUF ₆	28 May 1945	Blending of normal and LEUF ₆	Limited basis	126 lb (after blending)
Expansion of new UF ₆ plant (14 sets reactors)	UF ₆	Jun 1945	UF ₆ production increase	Production	4,600 lb/d
MED orders shipment to Harshaw of slightly enriched UF ₆ , to be mixed with normal UF ₆	LEUF ₆	17 Jan 1946	Blending of normal and LEUF ₆	Production	
MED authorizes UF ₄ production increase	UF ₄	4 Feb 1946	UF ₄ production increase		15,000 lb/wk
MED authorizes UF ₆ production increase	UF ₆	1 Jun 1946	UF ₆ production increase		4,200 lb/d
Fume recovery system planned for UF ₆ plant	UF ₆	Jan 1947	Pollution concerns		
Central waste disposal system completed		Oct 1947			
UF ₄ production expansion	UF ₄	4th Q 1947?	UF ₄ production increase	Production	5,600 lb/d
UF ₆ production expansion	UF ₆	4th Q 1947?	UF ₆ production increase	Production	6,300 lb/d
Construction of brown plant	Ore-UO ₂	1948–1949	Construction		

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Event	Form	Date	Brief description	Level	Amount
New laundry begins operation	Misc	Jan 1949?	Improved laundering		
The new brown plant begins operation	Ore-UO ₂	May 1949	Brown plant operation	Production	
AEC air pollution survey in, around Harshaw site (measuring radioactivity, fluoride levels)		Jun 1949	Air pollution survey		
Continuous green salt reactor use begins	UF ₄	Jan 1950?	Continuous UF ₄ reactor		
Central loading station use begins in UF ₆ area	UF ₆	Jun 1950	Mechanized loading: UF ₄	Production	
Central vacuum system use in the UF ₆ area	UF ₆	2nd half 1950	Central vacuum system	Production	
Dust control improvements, ore-to-UO ₂ area	Ore-UO ₂	Early 1951?	Dust control measures		
UF ₄ production shut down	UF ₄	17 Oct 1951	UF ₄ production ends	Shutdown	
UO ₂ production shut down; only UO ₃ produced in former brown plant	UO ₂ , UO ₃	4th Q 1951	UO ₂ production ends	Shutdown	
AEC contamination/radiation survey of UF ₄ and UF ₆ areas (prior to decontamination)	UF ₄ , UF ₆	Nov 1951	AEC survey	D&D	
UF ₆ production shut down, placed on standby	UF ₆	Dec 1951	UF ₆ production ends	Shutdown	
Decontamination of UF ₄ and UF ₆ areas begins	UF ₄ , UF ₆	After 6 Dec 1951	D&D: UF ₄ , UF ₆ areas	D&D	
Equipment dismantling in UF ₆ area begins	UF ₆	Jan 1952	Equipment dismantling	D&D	
Mezzanine area of annex (1945 hex bldg) torn out, forcing relocation of offices and shops		Jan/Feb 1952?	Modification		
Final shipment of hex ash to Vitro		Apr 1952	Hex ash shipment ends		
Dust control improvements in UO ₃ production	Ore-UO ₃	Jun-Jul 1952	Dust control measures		
Hanford ships UNH to Brush Beryllium for storage, transfer to Harshaw to make UO ₃	UNH	Mid-1952?	UNH processed to UO ₃	Production	
Change in feed materials (black oxide; also soda salts, Hanford UNH)	Ore-UO ₃	1 Oct 1952	Change in feed materials	Production	200,000 lb/mo
Capacity expansion in UO ₃ production plant	UO ₃	Oct 1952	Capacity expansion	Production	600,000 lb/mo
Start of processing of Hanford recycled UO ₃	Recycle U	4th Q 1952	Recycle U processing	Production	1,700 tons total
UO ₃ production shut down, placed on standby	UO ₃	Aug 1953	UO ₃ production ends	Shutdown	
AEC contamination/radiation survey, UF ₆ area	UF ₆	22-23 Oct 1953	AEC survey	D&D	
Harshaw finishes processing recycled U	Recycle U	4th Q 1953	Recycle U processing end	Special	
AEC check/survey of D&D progress		6 May 1954		D&D	
AEC check/survey of D&D progress		22 Jul 1954		D&D	
Virtual completion of UF ₆ plant dismantlement		Jul 1954		D&D	
AEC contam/radiation survey: UF ₆ , UF ₄ areas		22-23 Dec 1954	AEC survey	D&D	
AEC contam/radiation survey: UF ₆ , UF ₄ areas		31 May – 1 Jun 1955	AEC survey	D&D	
AEC directs Harshaw to end processing, convert leftover materials to UO ₃		17 Aug 1955	Completion of processing	Production	
Contract for removal of AEC equipment ends		30 Sep 1955	Equipment removal		
AEC-Harshaw contamination/radiation survey to track decontamination progress		14 Nov 1956	AEC-Harshaw D&D survey	D&D	
AEC-Washington orders AEC area office to dismantle Harshaw refinery		17 Dec 1956	Dismantlement of refinery – equipment only	D&D	
Harshaw license termination (D&D prep)		31 Dec 1956	License termination	D&D	
AEC contam/radiation survey, ore-UO ₂ area		21 Nov 1957	AEC survey	D&D	
Decontamination by Harshaw, AEC oversight		1955(?) - 1960	Decontamination	D&D	
Final AEC contamination/radiation survey		15 Apr 1959	AEC survey	D&D	
Site released from AEC control		1960	Site release	Release	
Argonne Nat'l Lab survey of site for DOE		12-20 May 1976	DOE-sponsored survey	D&D	
Engelhard Corp decontaminates some buildings, demolishes others		1990–1997	Further decontamination	D&D	
NRC releases 3 bldgs for unrestricted use; Plant C stays shuttered; Boiler House used to store rad materials from demolition		20 Mar 1998	Final disposition of site	Release	

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Table B-2. Buildings known to have been used at the Harshaw site for uranium processing work.

Building	Room or area	Use
G1 (Plant C)	All levels, partial floors	493: Brown oxide (ore-to- UO_2) production
	All levels, partial floors	493: Orange oxide (ore-to- UO_3) production
	1st floor	492: Hex (UF_6) production, recovery
	2nd floor	491: Green salt (UF_4) production
Annex	All areas	492: Hex (UF_6) production, storage of contaminated equipment and black oxide
K1		Some analytical work
Foundry		Early processing work

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Table B-3. Types and quantities of materials used and produced in Harshaw uranium processing.^a

Material	Process or operation	Content and form notes	Amount
ORES AND OTHER FEEDS			
All ores and feeds	It does not appear that uranium bearing ores were processed at Harshaw, but rather U ₃ O ₈ was received after milling (DOE 2000).		Mar 1953: 1 ton of "contained uranium" of each to be shipped to Harshaw: Colorado soda salt, Colorado black oxide, Canadian black oxide, South African black oxide, "Congo precipitate" (Belmore 1953).
Domestic ore and tailings	It does not appear that uranium bearing ores were processed at Harshaw, but rather U ₃ O ₈ was received after milling (DOE 2000).	Colorado ores (e.g., Uravan, Durango, Grand Junction, and Naturita) were carnotite type (Eisenbud 1975); North American ore contained less than 1% U ₃ O ₈ .	
U ₃ O ₈ (milled ore or black oxide)	Ore arrived at Harshaw in milled or concentrated form, as black oxide.	1952: Canadian black oxide at 96% U ₃ O ₈ , Colorado black oxide at 70-85% U ₃ O ₈ (Termini 1952).	4Q 1952: feed included 159 tons Colorado black oxide (Neumann 1952).
Sodium diuranate (soda salt)	Packed in fiber containers (MED 1945).	1952: Vitro soda salt at 75% U ₃ O ₈ (Termini 1952). Some possibly from Anaconda and Durango. 1952: Colorado soda salt at 60-72% U ₃ O ₈ (Termini 1952). 1952: "recycle soda salt" also used (Termini 1952). Fiber containers weighed about 75 lb each when full.	4Q 1952: feed included 43 tons Vitro soda salt, 53 tons Colorado soda salt; and 45 tons Eldorado soda salt (Neumann 1952).
UNH	1952: Hanford transferred UNH to Harshaw via the Brush Beryllium Company in Luckey, Oregon, for conversion to UO ₃ (DOE 2000).	Shipped by rail in tanker cars (DOE 2000), at least once via the Brush Beryllium Company (DOE 2000). 1952: "recycle NG liquor" used (Termini 1952), presumably from Hanford.	1,700 tons total.
REFINING PRODUCTS			
UO ₃ (orange oxide)	Feed digested in nitric acid to convert to nitrate form; precipitation of Ra-Pb with sulfuric acid (pitchblende ores only); filtration to remove the acid-insolubles; sulfate removal with Ba salt addition; centrifuging of solution to remove solids; boiling of "liquor"; double extraction of U with diethyl ether; purification; water wash to remove uranyl nitrate from ether; dewatering in Sperry press; boiling of the molten salt to "hex liquor" (uranyl nitrate hexahydrate); decomposition in gas-fired pots to form UO ₃ ; UO ₃ scooped or "gulped" out of pot using vacuum system, packed in fiber containers for shipment.	Digestion took 4-8 hr. Solid and liquid wastes, including most residues below. Full 2.5-gal fiber containers weighed 75 lb. Typical, 1952 (Termini 1952): 4,000-5,000 lb of Canadian and Colorado black oxides yielded 900 gal of digested slurry at 400-500 g U/L; pumped to holding tank and mixed with 600 gal dilute NG liquor, more acid, recycle soda salt, and Vitro soda salt, to produce new slurry at 300 g U/L; after extraction, before water wash, the organic liquid saturated to 80-90% uranium; after water wash, preboildown OK liquor at 50-75 g U/L. UO ₃ shipped to K-25 in 16-drum lots (800 lb/drum gross), 2 lots per shipment; sample taken during drum filling was supplied for each lot; 2 lot samples composited, analyzed for U content and isotopic ratio at K-25 (BJC and Haselwood Enterprises 2000).	1951: 100 tons/mo capacity (AEC 1951c). 3Q 1952: 55 tons/mo to be sent to K-25 and about 23 tons/wk to Mallinckrodt (Fernelius 1952). 4Q 1952: 200,000 lb/mo of UO ₃ to be produced (Neumann 1952); Sep 1952: 200,000 lb/mo based on operation 24 hr/d, 7 d/wk (Termini 1952).
UO ₂ (brown oxide) (received)	Small quantities of UO ₃ not meeting K-25 acceptance criteria for purity were sent to Harshaw for purification (DOE 2000).	2.5-gal fiber containers weighed about 75 lb each when full (Burman 1949). Shipped by rail from Mallinckrodt (AEC 1949b). Had to be 97% or more free UO ₂ (the rest was impurities) (HCC 1946).	Mallinckrodt produced 2/3 of US total; 20% went to Harshaw (MED 1949). In 1944-1945, 10,000 lb/wk to Harshaw; in Sept-Oct 1944; 28,000 lb in Nov; and 13,000 lb/wk after that (Simmons 1945).

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Material	Process or operation	Content and form notes	Amount
REFINING PRODUCTS (Cont'd.)			
UO ₂ (brown oxide) (produced)	UO ₃ was transferred from fiber containers onto monel trays; reduced with cracked ammonia in batch electric (muffle) furnace to form UO ₂ (AEC 1949b); scooped from trays into fiber containers for transfer elsewhere.	In June 1949, 300 lb/drum (Mayer and Proschén 1949).	In Oct 1949, 56 tons produced (Kelley 1949b), design capacity 54 tons/mo (AEC 1949b, Kelley 1949b). Jul -Dec 1949: avg 42.5 tons/mo; 1950: avg 53 tons/mo; Jan-Mar 1951: avg 71 tons/mo (AEC 1951c). 1951, after process mods (e.g., using TBP), capacity up to 100 tons/mo (AEC 1951c).
UF ₄ (green salt) (received)		Fiber containers weighed about 75 lb each when full.	To 1949, some UF ₄ from ElectroMet (DOE 1997) (probably to Harshaw). In 1945, 8,000 lb/wk were being sent to Harshaw from Mallinckrodt (Simmons 1945).
UF ₄ (green salt) (produced)	UO ₂ placed on nickel trays (Rauch 1948; HCC 1946), loaded into reactor tubes, and placed in hydrofluorination reactor (furnace); HF gas passed over it to form UF ₄ ; UF ₄ removed from furnace and put through pulverizer; UF ₄ weighed (Mayer and Proschén 1949) and packed into fiber containers (Simmons 1945) or 5-gal containers for transfer to another site (AEC 1949b). Excess anhydrous HF gas was bled off the reactors and sent back to the recovery system (Mayer and Proschén 1949).	Four trays per reactor; 9 lb in the top and bottom and 13 in each middle tray, for a total charge of 44 lb/reactor (Mayer and Proschén 1949; Klevin 1955b; HCC 1946). Oct 1946: one bank of 37 reactors, for a total (fresh) charge of 1,893 lb per 14-hr cycle; however, reruns of incompletely reacted material amounted to 7-10% of the material made (HCC 1946). UF ₄ had to be minimum of 97.5% UF ₄ , with less than 0.3% UO ₂ and 1.2% UO ₂ F ₂ (HCC 1946). Unreacted or partly reacted material was recharged into trays at 75 lb/tube and rerun through the normal 14-hr cycle (HCC 1946). Oct 1948: two banks of 37 reactors each operated on 12-hr cycles (Rauch 1948). 51 lb UF ₄ produced per 44 lb UO ₂ charged (HCC 1946). Filled drums contained 200 lb each (Burman 1949). Gas sent to the recovery system was not analyzed (Mayer and Proschén 1949). On a metal basis, from 100% charged as UO ₂ , about 0.4% was recovered as scrap and 0.57% was unaccounted for (HCC 1946). Sent to K-25? (DOE 1997). Fiber containers: 75 lb each, full (Simmons 1945).	Sep 1942: 1,800 lb/d (HCC ca. 1945), avg 25 tons/mo (AEC 1951c). Dec 1944: 3,000 lb/d (HCC ca. 1945). Feb 1946: 18,000 lb/wk for 1Q 1946, then 15,000 lb/wk (Eisenbud 1949b). Oct 1946: 2 d' inventory normally in process (Gates 1946), theor. capacity 3,235 lb/d but actual production 3,000 lb/d due to reruns (HCC 1946). Jul 1944 - Dec 1946: avg 40 tons/mo (AEC 1951c). 1947: avg 84 tons/mo; 1948: 88.5; 1949: 76; 1950: 80; and Jan-Mar 1951 (projected): 86 (AEC 1951c). Oct 1948: theor. capacity 7,600 lb/d (Rauch 1948). Dec 1947, Jan 1948, Feb 1948: 20.2, 20.0, and (projected) 20.5 tons produced. respectively (Chrestia 1948). 4Q 1949, 5,500 lb/d UF ₄ to be produced (Hunter 1949b).

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Material	Process or operation	Content and form notes	Amount
REFINING PRODUCTS (Cont'd.)			
UF ₆ (hex)	UF ₄ placed on nickel trays in sealed boxes; fluorine passed over at an elevated temperature; UF ₆ gas condensed in a series of three receivers, with any uncondensed gas caught on traps (Gates 1946; HCC 1946). Completeness of reaction was judged by receiver weight, at which time the third reactor per unit was reloaded (Rauch 1948). The third receiver normally was not filled in each cycle and was not removed until full, so was not removed every cycle (HCC 1946).	Jul 1944 (HCC ca. 1945): 18 reactors. Nov 1944 (HCC ca. 1945): 36 reactors. June 1945: 36 reactors (Tybout 1945a) or 42 reactors (HCC ca. 1945). Oct 1946: each set of reactors produced 340 lb/d of crude UF ₆ ; there were 14 units (sets of 3 reactors) in 7 hoods, 14 pairs of fluorine cells, and a 24-hr cycle (HCC 1946). Oct 1948: 96 reactors in 16 hoods and 52 fluorine cells; three reactors in series formed a unit, with two fluorine cells for each unit; 18 reactors without dedicated fluorine cells drew on excess fluorine capacity (Rauch 1948). Two reactors per regular unit operated on 24-hr cycles, the third on 2- to 3-d cycles; the 18 extra reactors operated on 3- to 4-d cycles (Rauch 1948). Each reactor had two trays, each charged with 90 (HCC 1946) or 100 (Rauch 1948) lb UF ₄ ; receivers held up to 435 lb of condensed UF ₆ . Theoretical output was based on a 100-lb/d charge, 100% efficiency, 78 reactors (2/3 on a daily cycle, 1/3 on 3-d cycle), plus 18 (4-d cycle) (Rauch 1948). 1945: UF ₄ feed rate was 4,200 lb/d (HCC ca. 1945).	Apr 1944: 400 lb/d UF ₆ (HCC ca. 1945). Sept 1944: new semicontinuous process reactor to produce 800 lb/d UF ₆ (per set) (Tybout 1944a), 3,300 lb/d total (AEC 1951c). Nov 1944: 4,500 lb/d (AEC 1951c). Nov 1944 (HCC ca. 1945), Jun 1945 (Tybout 1945a): capacity 4,000 lb/d. Feb 1946: 4,200 lb/d to Jun 1946, then 4,500 lb/d (Eisenbud 1949b). Oct 1946: theor. capacity 4,600-4,700 lb/d or about 14.5 lb/hr of crude UF ₆ , actual production 3,000 lb UF ₄ and 4,600 lb UF ₆ per day (HCC 1946). Dec 1946: 6,300 lb/d; Nov 1947: 9,000 lb/d (AEC 1951c). By Mar 1948: expansion to produce 5,600 lb/d or 121,000 lb/mo (Chrestia 1948). Oct 1948: theor. capacity 14,600 lb/d (assumes total UF ₄ input of 13,000 lb/d, more than Harshaw itself produced) (Rauch 1948). 1950: production cut back to 6,300 lb/d (AEC 1951c).
Distilled UF ₆ (hex)	Crude UF ₆ purified by distillation: most noncondensable gases were drawn off via floor bleeders (HCC 1946); the HF, the rest of the condensables, and some UF ₆ were drawn off via scale bleeders (HCC 1946); the rest of the UF ₆ was distilled into nickel or steel cylinders for shipment (Gates 1946).	1946: 5% by weight of crude UF ₆ was bled off to a scale bleeder (HCC 1946). The rest was distilled into nickel or steel shipping containers, with nickel container capacity of 462 lb UF ₆ (HCC 1946). Sample cylinders held 5 lb UF ₆ , with ~4.5 lb left after 7-oz sample draw (HCC 1946). On a metal basis, from 100% of U charged as UF ₄ , 0.69% was recovered as scrap metal and 1.82% unaccounted for. Cylinders sent to K-25 (Gates 1946; AEC 1949b) by government truck (AEC 1949b). Oct 1943: 5 lb UF ₆ to be shipped to Oak Ridge (Clinton Engineer Works) (Russell 1943). Destination of this UF ₆ was the S-50 project (1,050 lb/d, May - Nov 1945, total of 500,000 lb by mid-Oct 1945) (Hearon 1945).	May 1945: capacity 4,600 lb/d (Pinkston 1945). Normal in-process inventory, Oct 1946: 1 week's production (Gates 1946). Dec 1947, Jan 1948, Feb 1948 (Chrestia 1948): 45,800, 46,400, and 42,500 lb produced, respectively. By Mar 1948: expansion to 6,300 lb/d, 136,000 lb/mo (Chrestia 1948). Jun 1948: 7,600 lb/d produced (Kelley 1948). Oct 1948: theor. capacity 14,600 lb/d (required UF ₄ input beyond what Harshaw produced) (Rauch 1948). Apr 1949: 90 tons/mo to K-25 (AEC 1949b). 4Q 1949, 6,000 lb/d UF ₆ planned (Hunter 1949b).
UCl ₄		Assumed production 7 d/wk (Belmore 1953; AEC 1949b). For Jan 1945 contract extension, feed materials to be "Product 306, Product B, Product G, and other similar materials" (MED 1945).	Oct 1944: new plant capacity of 2,000 lb/d by Nov 1944, 80,000 lb to be supplied by 1 Jan 1945 (Belmore 1953; AEC 1949b). Jan 1945: 65,000 lb by 15 Feb 1945 (MED 1945).

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Material	Process or operation	Content and form notes	Amount
SPECIAL PROCESS MATERIALS AND SAMPLING			
Slightly enriched UF ₆	Shipped from Hanford (Kelley 1946).	After blending in May 1945: .3% enriched (Kelley 1946).	In May 1945, 126 lb shipped from Harshaw after blending (Kelley 1946).
RU	Shipped from Hanford (BJC and Haselwood Enterprises 2000).	Oct 1952-Jun 1953: RU from Harshaw to K-25, ~1,403,000 kg UO ₃ at 0.666%; Jul 1953-Dec 1953, ~300,000 kg at 0.671%; in this total of ~1,702,000 kg, there were 5.96 g Pu, 1328 g Np, and 11,900 g Tc (BJC and Haselwood Enterprises 2000).	
Sampling Harshaw-produced UO ₃	20-qt jar of UO ₃ unloaded into grinder; ground UO ₃ loaded into 2-qt jar and put into blender, which was then collected in 2-qt jars.	1953: tests on Harshaw UO ₃ were performed daily (AEC 1953b).	
Sampling of Hanford-produced UO ₃	In lab, UO ₂ scooped from cans into blenders; blended; scoop-unloaded into grinder; ground; and collected in 2-quart cans. Shotgun tests: samples of ground UO ₃ weighed out; loaded into a die; pressed in hydraulic press; and used in the assay process (AEC 1953b).	1953: tests on Hanford UO ₃ were performed about 3 times every 2 wk (AEC 1953b). Shotgun samples consisted of ten 45-g aliquots, apparently per 2-quart jars (AEC 1953b).	
RESIDUES, OTHER WASTES, AND RECOVERED MATERIALS			
UF ₄ -to-UF ₆ fluorination ash	Vacuum-conveyed from fluorination trays to a bag dust collector (AEC 1949b; Rauch 1948) on the roof (Piccot 1949), where it was removed by turning a crank that allowed the ash to fall through a star valve into a drum. The drum was filled about every 2 wk, at which time it was removed, lidded, and conveyed by a truck to the first floor (Piccot 1949). It was then stored temporarily for decay (Mayer and Proschon 1949; HCC 1946); from 1946 or before, until at least 1949, the storage time was 6 mo (HCC 1946; Eisenbud 1949f).	Ash: 0.1% of the original mass of the uranium, but practically all of the UX1-UX2; beta activity of ~1 mCi/g (AEC 1949b). May 1948: 100 g of ash had volume of 100 cm ³ ; spread out in flat dish 10 cm in diameter, gave a reading corresponding to 15 mCi. Sep 1949: ash being collected (and probably shipped in) 15-gal drums (Sargent 1949; Piccot 1949) lined with 1/16 in. or 3/32 in. of lead on top, bottom, and sides (Piccot 1949). April 1949 (AEC 1949b): shipped to Oak Ridge or Lake Ontario Ordnance Works for storage; (Mayer and Proschon 1949, Jun 1949): being weighed and sent to Vitro. (Hunter 1949c, Nov 1949): 2,250 lb of Harshaw ash in storage at Oak Ridge, with newest well over 1 yr old (i.e., no longer being sent to Oak Ridge); this ash supposedly contained 10-13% uranium. Jun 1949 (Piccot 1949): ash was about 1 g/cc and contained about 18% metal.	In Jul 1944, about 5 lb/d of ash produced (Parke 1944); in Apr 1949, 2,500 g/d (AEC 1949b). In Apr 1949, 2,000 gal/d apparently shipped for storage (AEC 1949b).
UO ₂ loading dust	Vacuum-conveyed from fluorination trays to a bag dust collector (AEC 1949b; Rauch 1948).		
Ash from miscellaneous combustible waste	Produced from burning UO ₂ containers, floor sweepings ("black ash"), and scrap from the recovery room (Rauch 1948).		Shipped to Vitro (Mayer and Proschon 1949).
Floor drainage	Floor drainage sent through filter press, producing a filtrate and a press cake (AEC 1949b).	Press cake shipped to Vitro for processing (AEC 1949b); filtrate sent to the Cuyahoga River (AEC 1949b).	Filtrate: about 150 gal/d in April 1949 (AEC 1949b).
UF ₆ condenser water		Radionuclide concentration normally none (AEC 1949b).	
Condenser (receiver) tail gas	Passed through a dry ice trap to remove residual UF ₆ (AEC 1949b).		

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Material	Process or operation	Content and form notes	Amount
RESIDUES, OTHER WASTES, AND RECOVERED MATERIALS (Cont'd.)			
Floor spillage	UF ₄ mostly washed to the sewer in 1945 (Anderson 1945).	Some washed out to Cuyahoga River, some lodged in sewer (Anderson 1945).	Up to 40 lb/d of UF ₄ (Anderson 1945).
Scale and floor bleeds	Crude UF ₆ in the receivers, UF ₆ on the associated dry ice traps: "bled" to remove volatile gases. The former bleed sent via floor and scale bleeders (HCC 1946) to large turbosaturator on the roof and the latter to the small one (Rauch 1948). Floor receivers in distillation unit and dry ice trap receivers bled directly to large turbosaturator, bypassing scale and floor bleeders and the dry ice trap (Hunter 1949a). When full, floor and scale bleeders were processed like receivers, minimizing loss (HCC 1946).	The large turbosaturator and tower handled the fumes from the small vent hoods over each individual reactor, the dry ice trap for the distillation unit, and the bleed from the floor receivers in the distillation unit and the dry ice trap receivers; the last two of these were bled directly to the large turbosaturator, thereby bypassing the scale and floor bleeders and the dry ice trap (Hunter 1949a). Condensate from the dry ice traps contained up to 40% HF (HCC 1946).	Up to 10% of the original quantity was bled off (Rauch 1948).
UF ₄ -to-UF ₆ reactor hood exhaust	Exhausted to the large turbosaturator on roof, then to the atmosphere (Rauch 1948).		
Direct vents	April 1949: area in front of UF ₄ reactor hoods vented directly to atmosphere; no provision for collection of dust; mostly UF ₄ being loaded but also UF ₆ fuming from reactors (Burman 1949; Hunter 1949a). UF ₆ process area ceiling exhaust also vented directly to atmosphere (Hunter 1949a).		
Sump liquid	Mainly washings from process areas, especially the hex reactor area (Burman 1949). This was filtered; the filtrate was sent to the sewer (Lynch 1949) and the press cake was weighed and sent to Vitro (Burman 1949; Mayer and Proschén 1949; Lynch 1949).	Floor drainage and the water from the turbosaturator were collected and treated in a recovery process, producing press cake and ~150 gal/d of filtrate (Lynch 1949). April 1948: Cuyahoga River samples showed that U concentration to be 0.00-0.006 µg/cm ³ in water and 0.00-0.58 µg/g in mud; said to be about normal background (Lynch 1949).	
Digestion process residues	Slurry from digester was sent through Niagara filter; insoluble cake was removed, treated again with nitric acid, filtered; U adhering to cake after 2nd acid digestion and filtration was precipitated with strong caustic. Slurry was filtered; filtrate was sent to sewer and cake to Vitro in 1949 (Mayer and Proschén 1949). Filtrate sent to sewer was not checked for uranium content, but the final cake was analyzed (Mayer and Proschén 1949).	1952 (Termini 1952): raffinate from the final extraction tank was collected in holding tanks for analysis; if > 0.1 g U/L, raffinate was recycled or treated with caustic to precipitate soda salt (latter removed as a cake by filtration); if < 0.1 g U/L, raffinate was discarded. Average weight of cake from good (70%) black oxide feed, was 504 lb per digest batch; 1,975 lb when operating on a 70%-30% batch (Fernelius 1950).	In Oct 1949, accumulations of 70,000 lb of press cake in 55-gal drums and 35,000 lb of Bird (centrifuge) residue were being stored temporarily at Harshaw (Blatz 1949c).
Metal scrap	Harshaw received Hanford metal scrap (DOE 2000); also generated its own metal scrap (AEC 1949b).	In April 1949, scrap stored at Harshaw; might have been disposed of in unknown ways previously (AEC 1949b).	Probably to Vitro.
30-, 55-gal drums	Empty drums, stored as contaminated waste.		
Turbosaturator slurry	The slurry from both turboagitators was filter-pressed to remove the uranium precipitates. The filtrate was sent to the sewer and the cake was weighed and sent to Vitro (Mayer and Proschén 1949).	Floor drainage and water from fume scrubbing (turbo-sat) was collected and treated in a recovery process, producing a press cake and 150 gal/d of filtrate (Lynch 1949). Apr 1948: samples from Cuyahoga River showed that U concentration was 0.00-0.006 µg/cm ³ in water and 0.00-0.58 µg/g in mud, said to be about normal background (Lynch 1949).	

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Material	Process or operation	Content and form notes	Amount
RESIDUES, OTHER WASTES, AND RECOVERED MATERIALS (Cont'd.)			
Stack discharges			Sep 1948: discharge, 40 lb/d U (Harris 1949c). Jun 1949: 21 g/min total from 6 reactor stacks, 13 g/min from 2 turbosaturator stacks, 5 lb/hr from main sources (likely UF ₆) (Eisenbud 1949a); in outdoor air, no samples >10 µg/m ³ at 100-10,000 ft from plant; average conc at 0.3 mi was 3 µg/m ³ (Eisenbud 1949a). Samples up to Jul 1949: 22 lb/d (Harris 1949c). 15 Nov -15 Dec 1951: losses of 530 lb for UF ₆ area, 14 lb for UO ₃ area (Stefanec 1951).
Condenser cooling water	Normally not contaminated (AEC 1949b); sent to the Cuyahoga River (Lynch 1949; AEC 1949b).	April 1948: samples taken from the Cuyahoga River showed that the uranium concentration in the water was 0.00-0.006 µg/cm ³ and the mud concentration 0.00-0.58 µg/g, which were said to be of the order of normal background (Lynch 1949).	
Aqueous tails from the ether extraction	The water fraction from the ether extraction was heated to boil off ether, which was recycled. The de-etherized solution was partly neutralized to precipitate impurities. The cake was given a second acid leach and filtered. The filtrate was returned to the process flow; the cake was treated with caustic and filtered. The final cake was sent to Vitro. Caustic was added to the filtrate; after U precipitated, the slurry was filtered; the filtrate was alkalinized, refiltered, and discarded (Mayer and Proschen 1949).	The de-etherized solution had a low percentage uranium content (Mayer and Proschen 1949).	

a. The text describes process details; Table B-4 lists other code numbers and terms.

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Table B-4. Functional and process keywords and codes.

Process	Keyword	Notes
UF ₄ -UF ₆ , UF ₆ Distill	1st Level	First floor of Bldg G1: all UF ₄ -UF ₆ , distillation processes located there
	192	MED code number for the Harshaw laboratory
	2	Generic MED code for soluble uranium forms
	2048	Generic MED code for radiation
	216	Fluorine or hydrofluoric gas
UF ₄ -UF ₆	2nd Level	Second floor of Bldg G1: all UO ₂ -UF ₄ processes located there
	32	Generic MED code for handling of (uranium) metal
	36	MED code number for the Harshaw production plant
	4	Generic MED code for insoluble uranium forms
UF ₄ -UF ₆	490	Usage suggests this is an early synonym for 492
UO ₂ -UF ₄	491	The UO ₂ -to-UF ₄ (green salt) operation or area
UF ₄ -UF ₆	492	The UF ₄ -to-UF ₆ operation or area (hex area), including still ops
Ore-UO ₂ , Ore-UO ₃	493	The ore-to-UO ₂ or -to-UO ₃ operation or area (brown plant)
	516	UCl ₄
UF ₄ -UF ₆ , Distill	616	UF ₆ (and sometimes its ash residue)
	64	Generic code for UF ₆
All	Acid	Usage suggests this is an early synonym for 492
UF ₄ -UF ₆	Ash	Left after fluorination to UF ₆ and removal of UF ₆
Ore-UO ₃	Batch	In batches (i.e., as opposed to a continuous or semicontinuous process)
Ore-UO ₃	BDT & DEET	Boildown tank and de-etherizing extraction tanks
Ore-UO ₃	Bird	Centrifuge used in the ore-to-UO ₃ production process
Ore-UO ₃	Black oxide	U ₃ O ₈
Ore-UO ₃	Boildown	Reduction used in the ore-to-UO ₃ production process
Ore-UO ₂ , UO ₂ -UF ₄	Brown	UO ₂
UO ₂ -UF ₄	Brown loader	Job title for loader of UO ₂ ; also mechanical loader
All	Buffalo	Gas scrubber
	C-216	Fluorine or hydrofluoric gas (HF)
	C-516	UCl ₄
	C-616	UF ₆
Ore-UO ₃	Centrifuge	Centrifuge used in the ore-to-UO ₃ production process
All	Cleanup	Usually refers to general area cleanup (e.g., the vacuuming or washdown of floors)
UO ₂ -UF ₄	Continuous furnace	Used in the UO ₂ -to-UF ₄ production process
	CPM	Chief process man (lead operator)
Ore-UO ₃	CR-15	UO ₃
UF ₆ Distillation	Cylinder	Container for UF ₆ storage and shipment
UO ₃ -UO ₂	Deck	In general, a localized or intermediate partial deck or floor for process access
All	Decontamination	Cleaning of equipment or areas beyond normal housekeeping by operators
All	Develop, Dev, Dev Engr	Development of engineering-type modifications, research
Ore-UO ₃	Digester	Digest tank
UF ₆ Distillation	Distillation	Distillation of crude UF ₆ into specification-grade UF ₆
Ore-UO ₃	Ether	Used in extraction process
Ore-UO ₃	Extraction	Ether extraction of uranium in the ore-to-UO ₃ process
Ore-UO ₃	Feed	Input to the digestion process (e.g., U ₃ O ₈ or soda salt)
Ore-UO ₃	Frame	Frame-and-press-type filter, producing a cake
UO ₃ -UO ₂	Furnace	(1) Rockwell furnace, (2) brown-to-green furnace
All	G1	Building G1 (i.e., Plant C, where uranium processing took place)
UO ₂ -UF ₄ , UF ₄ -UF ₆	Green	UF ₄
Ore-UO ₃	Grinding	Grinding orange oxide (UO ₃) lumps
Ore-UO ₃ , Ore-UO ₂	H-32	Milled ore (black oxide)
Ore-UO ₃	HE	Code name for Hanford
Ore-UO ₃	HE-33	Hanford-produced UO ₃ (sent to Harshaw for reprocessing)
UF ₄ -UF ₆	Hex Area Loader	Worker who loaded UF ₄ and (usually) who unloaded ash from the hex reactors
	HGE	Code name for Hanford, used as a prefix to identify Hanford-produced UO ₃ being cold-pressed prior to analysis at Harshaw
UO ₂ -UF ₄	HL-7, HL7	UF ₄ , the UO ₂ -to-UF ₄ production process
UF ₄ -UF ₆	Hopper	For loading green salt
	JH-6	Harshaw-produced UO ₃ made from recycle UO ₃ from Hanford
	K1	Building K1, where analytical work was done
	KoZ	Unclear

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Process	Keyword	Notes
UF ₄ -UF ₆	Lift truck	Used for loading green salt
UO ₂ -UF ₄ , UF ₄ -UF ₆	Loader	Usually, worker who loaded UO ₂ or UF ₄
Ore-UO ₃	NG Liquor	NG = UNH (including the nitric acid?); also used as name of tanks in which extraction was done
Ore-UO ₃	Niagara	Filter used in the ore-to-UO ₃ process
	O.G.	Fluorine (F ₂) gas
Ore-UO ₃ , UO ₃ -UO ₂	Orange	UO ₃
Ore-UO ₃ , UO ₃ -UO ₂	PB-13, PB13	UO ₂ ; the ore-to-UO ₂ or -to-UO ₃ operation or area (brown plant)
	P.G.	UF ₆
	PH	Process helper
	PH-30	Unclear
	Pilot Plant	Any of several small-scale facilities set up to produce a U form for a limited period
All	Plant C	The Harshaw uranium processing plant; usually, Bldg. G1
	PM	Process man (operator)
	POB	Unclear; it might be associated with the term "Process Man".
Ore-UO ₃	Pot	Denitration pot
Ore-UO ₃ , UF ₄ -UF ₆	Press cake	Residue containing trace uranium as potassium uranate (or typical product of a filter press, e.g., for sump washings or extraction fluids)
UF ₆ distillation	Purification	Synonym for distillation
	Q-2-X	A type of ore feed (U ₃ O ₈)
Ore-UO ₃	Raffinate	Postdigestion residue
UF ₄ -UF ₆	Reactor	Hex reactor (i.e., the reactor in which UF ₆ forms)
UF ₄ -UF ₆	Receiver	Container into which UF ₆ is drawn as it forms
All	Recovery	Recovery of wastes, for the ore-UO ₂ processes, or recovery of UF ₆ and associated vapors, for the UF ₄ -UF ₆ and UF ₆ distillation processes
UF ₄ -UF ₆ , UF ₆ Distill	Recovery Room	Fume recovery room
UO ₃ -UO ₂	Rockwell	Brown process production furnace
UF ₄ -UF ₆	RT-12, RT12	UF ₆ , the UF ₄ -UF ₆ production process
	RX-10	UF ₆
	S-15	Soda salt
All	Scrubber	Gas scrubber
Ore-UO ₃	Soda salt	Sodium diuranate (Na ₂ U ₂ O ₇), a feed form
	SS-20	It is unclear what this is. It may be a nonradioactive process material.
UF ₆ Distillation	Still	Distillation still used in the processing of crude UF ₆ to specification UF ₆
	T	Code synonym for U (e.g., TCl ₄ , TF ₄ for UCl ₄ , UF ₄)
Ore-UO ₃	TBP	Ether form used for extraction
UF ₄ -UF ₆ , UF ₆ Distill	Trap	Usually, dry ice trap for UF ₆ ; could also be carbon or other trap for UF ₆
UO ₂ -UF ₄	Tube	Reaction container for UO ₂
UF ₄ -UF ₆ , UF ₆ Distill	Turbosaturator	Scrubber used to collect acid vapors and U gas forms
All	Vacuum	Portable vacuum cleaners or material transfer devices; the central vacuum system
	WE-22	Probably ash
	WE-22S	Probably scrap material
UO ₂ -UF ₄	WE-61	UF ₄ or its processing
	X-32	Same as H-32?

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Table B-5. Measured dose rates.^a

Area or source item	Position/distance	Gamma dose rate, mR/hr	Beta dose rate, mrep/hr	Total dose rate, mrep/hr	Date	Reference
UCl₄ production laboratory ("1945 East 97th Street")						
Near shelves with bottles of UCl ₄	Highest rate			1.3	Circa 1944	Gamertsfelder ca. 1944
Storeroom	Highest rate			1.8	Circa 1944	Gamertsfelder ca. 1944
In fluorinat'g hood, not being used				0.8	Circa 1944	Gamertsfelder ca. 1944
UF₄ plant ("1000 Harvard Avenue")						
Dust-covered UF ₄ canning table	2 in. away			3.3	Circa 1944	Gamertsfelder ca. 1944
	At worker position			1.2		
	2 ft from table, 4.5 ft from floor			0.4		
Man's shirt, worn in lab 5 d	Contact with electroscope			0.6	Circa 1944	Gamertsfelder ca. 1944
Storeroom, Fe can with 140 lb UF ₄	Top of can			4	Circa 1944	Gamertsfelder ca. 1944
UO₂-to-UF₄ (Green Area, 2nd level)						
Corridor of UO ₂ storage area, avg	43 in. above floor	1.5	1.1	2.6	May 1948	Hayden 1948
UO ₂ carton by Furnace #41	Top, 1 in.			10	Aug 1947	Turner 1947e
Put-up bench, layer of fine UO ₂	Top, 1 in.			20	Aug 1947	Turner 1947e
Under put-up bench, r sweepings	Floor, 1 in.			2	Aug 1947	Turner 1947e
Under furnace loading stand	Floor, 1 in.			<2	Aug 1947	Turner 1947e
Furnace purging rack	Floor, 1 in.			<2	Aug 1947	Turner 1947e
Under furnaces	Floor, 1 in.			ND-<2	Aug 1947	Turner 1947e
Rear of furnaces	Floor, 1 in.			ND	Aug 1947	Turner 1947e
Burned trays	Top, 1 in.			<2	Aug 1947	Turner 1947e
In front of panel board	Floor, 1 in.			5	Aug 1947	Turner 1947e
Metal cans near furnace	Side, 1 in.			<2	Aug 1947	Turner 1947e
Repair bench, fine UF ₄ layer	Top, 1 in.			2	Aug 1947	Turner 1947e
Worker's right shoe top, UF ₄ on it	Top, 1 in.			17	Aug 1947	Turner 1947e
Fence (inside) in SW area	Floor, 1 in.			ND	Aug 1947	Turner 1947e
At south wall, over storage boxes	Wall, 1 in.			5	Aug 1947	Turner 1947e
By window upstairs (green area?)	Floor, 1 in.			ND	Aug 1947	Turner 1947e
Stairs between UF ₄ , Hex areas:						
Top landing	Floor, 1 in.			<2	Aug 1947	Turner 1947e
Middle of stairs	Floor, 1 in.			10		
Bottom of stairs	Floor, 1 in.			10.5		
Hoods	43 in. above floor	0.6	4.4	4.9	May 1948	Hayden 1948
General hood area	43 in. above floor	1	1.3	2.3	May 1948	Hayden 1948
At a reactor tube rack	43 in. above floor	0.5	1.2	1.7	May 1948	Hayden 1948
At reactor tube cooling rack	43 in. above floor	0.4	1.9	2.3	May 1948	Hayden 1948
Product container cleanout area	43 in. above floor	1	7.8	8.8	May 1948	Hayden 1948
UF₄-to-UF₆ (Hex Area, 1st level)						
Catch pans on scale, UF ₄ in pan	1 in.			100-240	Aug 1947	Turner 1947e
Above a UF ₄ tray	1 in.			170	Aug 1947	Turner 1947e
Above a UF ₄ tray	3 in.			240	Aug 1947	Turner 1947e
Near tray, some UF ₄ salt and ash	Floor, 1 in.		15	160	Aug 1947	Turner 1947e
Loading rack w/ UF ₄ and hex ash	Surface, 1 in.		<2	52	Aug 1947	Turner 1947e
Loading rack hood end: UF ₄ , ash	Surface, 1 in.		60	1,200	Aug 1947	Turner 1947e
Loading rack	Top, 1 in.		18	240	Aug 1947	Turner 1947e
Loading rack	Top, 1 in.			240	Aug 1947	Turner 1947e
2 ft in rear of cell rooms	Floor, 1 in.			ND	Aug 1947	Turner 1947e
Inside cell room	Floor, 1 in.			ND	Aug 1947	Turner 1947e
Instrument panel for Cells #18, 18A	Floor, 30 in.			10	Aug 1947	Turner 1947e
In front of Panel Board #5	Floor, 1 in.			5	Aug 1947	Turner 1947e
2 ft S of Hood #3, UF ₄ on wet floor	Floor, 1 in.		10	135	Aug 1947	Turner 1947e

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Area or source item	Position/distance	Gamma dose rate, mR/hr	Beta dose rate, mrep/hr	Total dose rate, mrep/hr	Date	Reference
Various points around hoods	Floor, 1 in.			15-40	Aug 1947	Turner 1947e
Top of reactor rod rack, Crew #3	Floor, 1 in.			20	Aug 1947	Turner 1947e
Various pts betw. cells, reactors	Floor, 1 in.			22-25	Aug 1947	Turner 1947e
At hoods, 1 ft from receivers	Floor, 1 in.		5-7	25-55	Aug 1947	Turner 1947e
Near receivers opposite cells	Floor, 1 in.		7.5	20-60	Aug 1947	Turner 1947e
6 in. from receiver opposite a cell	Floor, 30 in.		<1	25	Aug 1947	Turner 1947e
4 ft in front of receivers opp. a cell	Floor, 30 in.			20	Aug 1947	Turner 1947e
Main entrance to building (Annex)	Floor, 1 in.			ND	Aug 1947	Turner 1947e
Doorway opp. Furnace Bank #20	Floor, 1 in.			10	Aug 1947	Turner 1947e
Various pts around furnace banks	Floor, 1 in.			5-20	Aug 1947	Turner 1947e
Above a tray containing hex ash	1 in.		600 (3 in.)	6,400 (1 in.)	Aug 1947	Turner 1947e
Floor drain betw. Cells #9A, #10	Floor, 1 in.			22	Aug 1947	Turner 1947e
Floor drain in front of Hood #2	Floor, 1 in.			12.5	Aug 1947	Turner 1947e
Drain in front of hex work table	Floor, 1 in.			5	Aug 1947	Turner 1947e
Dolly of Hood #6	Top, 1 in.		2	25	Aug 1947	Turner 1947e
Dolly of Hood #1	Floor, 1 in.		5	50	Aug 1947	Turner 1947e
"Cap dollie" (receiver cap dolly?)	Floor, 1 in.			50	Aug 1947	Turner 1947e
Around various ice traps	Floor, 1 in.			10-20	Aug 1947	Turner 1947e
Around ice traps, hydrolized UF ₆ on floor	Floor, 1 in.		2-5	20-25	Aug 1947	Turner 1947e
Top of Ice Trap #69	Top, 1 in.			3	Aug 1947	Turner 1947e
Hoods	43 in. above floor	2.6	24.5	27.1	May 1948	Hayden 1948
Near loading hood	43 in. above floor	0.5	4.4	4.9	May 1948	Hayden 1948
Near UF ₄ loading rack	43 in. above floor, 5 ft away	0.3	2.6	2.9	May 1948	Hayden 1948
UF ₄ weighing scale	43 in. above floor	1	6	7	May 1948	Hayden 1948
Control aisle (panel boards)	43 in. above floor	0.93	6.57	7.5	May 1948	Hayden 1948
Reactor area				3	Apr 1952	HCC 1950-1953
Dry ice trap	43 in. above floor	2.2	14.9	17.1	May 1948	Hayden 1948
Walls with hydrolized UF ₆	Wall, 1 in.			110-150	Aug 1947	Turner 1947e
Near steam table	Floor, 1 in.			110	Aug 1947	Turner 1947e
Wall E of steam table, 4 ft over floor	Wall, 1 in.			80	Aug 1947	Turner 1947e
Wall W of clean'g table, 4 ft over flr	Wall, 1 in.		10	120	Aug 1947	Turner 1947e
Near hot water container	Floor, 1 in.			240	Aug 1947	Turner 1947e
Opposite ice box	Floor, 1 in.			<2	Aug 1947	Turner 1947e
Empty receiver storage area	Floor, 1 in.			<2	Aug 1947	Turner 1947e
Under off-gas vent line 3 ft S of east wall	Floor, 1 in.			30	Aug 1947	Turner 1947e
Door, N ₂ trailer/cold H ₂ O machine	Floor, 1 in.			<2	Aug 1947	Turner 1947e
Opposite CuF ₂ drums	Floor, 1 in.			70	Aug 1947	Turner 1947e
Line unplugging area (UF ₆ lines)	43 in. above floor	8.7	104.3	113	May 1948	Hayden 1948
Line cleaning vat (UF ₆ lines)	43 in. above floor	6.4	35.4	41.8	May 1948	Hayden 1948
Line repair area	43 in. above floor	1.2	6.1	7.3	May 1948	Hayden 1948
Empty used cylinder to be painted	Floor, 1 in.			5	Aug 1947	Turner 1947e
Full cylinder on storage rack	Floor, 1 in.			<1 -2	Aug 1947	Turner 1947e
Cylinder work table	Top, 1 in.			2	Aug 1947	Turner 1947e
Between, behind stills, trap scale	Floor, 1 in.			ND - 2	Aug 1947	Turner 1947e
2" in front of still room	Floor, 1 in.			<2	Aug 1947	Turner 1947e
Outside still area	Floor, 1 in.			ND	Aug 1947	Turner 1947e
Still area (UF ₆ distillation)	43 in. above floor	BG	BG	BG	May 1948	Hayden 1948
1 ft south of Recovery Hood #16	Floor, 1 in.		<2	20	Aug 1947	Turner 1947e
1 in, S of Recovery Hood #20, UF ₄ on floor	Floor, 1 in.		2	25	Aug 1947	Turner 1947e
6 ft S of recovery hoods, ash on flr	Floor, 1 in.		2-130	50-2,400	Aug 1947	Turner 1947e
Asbestos gloves, used, hex crew	Surface, 1 in.			20	Aug 1947	Turner 1947e
Asbestos gloves on flr, hex panel	Surface, 1 in.			140	Aug 1947	Turner 1947e
Rubber gloves worn by operator	Surface, 1 in.			5	Aug 1947	Turner 1947e

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Area or source item	Position/distance	Gamma dose rate, mR/hr	Beta dose rate, mrep/hr	Total dose rate, mrep/hr	Date	Reference
Tray welding, cutting (subcontr.)			40		Oct 1950	HCC 1950-1953
Residue storage room	Center of room	21	Nil	21	Sep 1944	Tybout 1944a
Outside door to ash collector rm	43 in. above floor	7.7	6.2	13.9	May 1948	Hayden 1948
Ash collector room, average	At bin, 43 in. above floor	42	44	86	May 1948	Hayden 1948
Tray of fresh ash (before putting into can)	4 in. above tray		4,000, 2,000		Feb 1946	Tybout 1946
	3 in. above tray		3,000		Feb 1946	Tybout 1946
Smaller container (can) of ash (1/32-in, steel wall)	Side, contact		11,000		Feb 1946	Tybout 1946
	Side[?], 1.5 ft		2,000		Feb 1946	Tybout 1946
Ash can from center separ. (filter)	Touching the top			3,300	Aug 1947	Turner 1947e
Ash collector drum on roof	Contact	>20			Apr 1949	Wolf 1949
Drum of roof ash ^a	No lid, 2 in. above ash			300	Jun 1949	Piccot 1949
	Side, contact			164[?]		
	Side, 8 in.[?]			Illegible		
	Side, 12 in.			Illegible		
	Side, 24 in.[?]			Illegible		
Smaller container (can) of ash (1/8-in. steel wall)	Side, contact			110 (2.5")	Jun 1949	Piccot 1949
	Side, 8 in.[?]			35		
	Side, 12 in.			11		
	Side, 18 in.[?]			3		
	Side, 24 in.[?]			3		
Fresh ash in large beaker	6 in.	160	2,300	>2,500 α	May 1950	Klevin 1950c
Fresh ash in large beaker covered with a watch glass	6 in.	12.5	350	410 α	May 1950	Klevin 1950c
Fresh ash, fines	6 in.	650	>2,500	>2,500 α	May 1950	Klevin 1950c
Fresh ash, coarse residue	6 in.	54	950	1,100 α	May 1950	Klevin 1950c
Service corridor, west end	43 in. above floor	0.04	0.9	0.9	May 1948	Hayden 1948
Service corridor, center	43 in. above floor	0.12	0.8	0.9	May 1948	Hayden 1948
Macadam floor (hex area?)	Contact			17	May 1952	Blatz 1952
Ore-to-UO₂ or -UO₃ (Brown/Orange Area)^a						
Extraction tanks NG-1, NG-2	Contact	3.5			Apr 1949	Wolf 1949
Extract. boildown tanks (4a, 4b, 5)	Contact	1.2			Apr 1949	Wolf 1949
Extraction tanks NG-1, NG-2 (higher of the two readings is shown)	Contact	4		4	Oct 1949	Blatz 1952
	6 in.	2		2		
	12 in.	1		1		
	18 in.	0.9		0.9		
	24 in.	0.6		0.6		
36 in.	0.5		0.5			
Bird centrifuge	2 ft?	1			Apr 1949	Wolf 1949
Niagara and miscellaneous filters	Half-full, contact	1			Apr 1949	Wolf 1949
Niagara filters 1, 2	Contact	0.4, 0.3	20, 18		May 1950	Sargent 1950c
	6 in.	0.25, 0.2	9, 7.5			
	12 in.	0.2, 0.1	4, 3.5			
Plate-and-frame filter	Contact	1.6			May 1950	Sargent 1950c
	6 in.	1.4				
	8 in.		20			
	12 in.	1	14			
	24 in.		8		May 1950	Sargent 1950c
Postdigestion, Hanford HG-33:					Mar 1953	Klevin 1953b
Digest tank (empty)	1 in.			2.5		
NG liquor (extraction)	1 in.			2.5		
Raffinate liquor	1 in.			10.5		
Strip water	1 in.			0.05		
Stripped solvent	1 in.			0.25 - 2.0		
Recovered acid	1 in.			0.25		
Tank #9 recovered acid	1 in.			0.5		
Bi-Carbonate scrub	1 in.			2		

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Area or source item	Position/distance	Gamma dose rate, mR/hr	Beta dose rate, mreps/hr	Total dose rate, mreps/hr	Date	Reference
Raffinate cake	1 in.			2.5 - 5.0		
Discharge vapor: stack exhaust (mixer-settler extractor fumes)	1 in.			7-9		
Vapors off strip water exhaust	1 in.			0.02		
Nitric acid recovery exhst stack	1 in.			0.02		
Waste liquor to sewer	1 in.			10.5		
Digester tank	1-2 ft above liquid	1	5-9		May - Jul 1953	HCC 1950-1953
Raffinate tank	2 in.-3 ft above liquid	0-0.2	4-12		May - Jul 1953	HCC 1950-1953
Press cake filtrate, in process	3 in. above liquid	0.05	4		May - Jul 1953	HCC 1950-1953
Bi-Carb scrub tank	1 in.-6 ft above liquid	0	0-0.7		May - Jul 1953	HCC 1950-1953
Strip water tank	2 in.-3 ft above liquid	0	0-0.8		May - Jul 1953	HCC 1950-1953
Strip solvent tank	2 ft above liquid		0.2		May - Jul 1953	HCC 1950-1953
Recovery acid tank	1 in.-4 ft from liquid	0	0-2		May - Jul 1953	HCC 1950-1953
Press cake filtrate, in process	3 in. above liquid	0.05	4		May - Jul 1953	HCC 1950-1953
Sewer, 0.01-0.02 g/L raff. expelled	2-3 in. from liquid	0	6-11		May - Jul 1953	HCC 1950-1953
Digest tank exhaust	Stack, discharge	0	0.8		May - Jul 1953	HCC 1950-1953
Strip water exhaust	Stack, discharge	0	0-0.1		May - Jul 1953	HCC 1950-1953
Extractor fume exhaust	Stack, discharge	0	0-3		May - Jul 1953	HCC 1950-1953
Spilled cake from frame/press filt.	2-3 ft from spill	2		>20	Sep 1949	Blatz 1949a,b
Group 55-gal drums, press cake	Contact?	1		8	Sep 1949	Blatz 1949a, 1949b
Group 55-gal drums, Bird residue	Contact?	3			Sep 1949	Blatz 1949a,b
Group 55-gal drums, Bird residue	2 ft	2			Apr 1949	Wolf 1949
Pot, during denitration		1	5		May - Jul 1953	HCC 1950-1953
Pot, gulping	~3 in. above	0.5-1	5-6		May - Jul 1953	HCC 1950-1953
Pot, opened for repair	2 in. above bottom	0	2		May - Jul 1953	HCC 1950-1953
Other areas within Plant C						
Wooden pallets in storage area	Surface, 1 in.			10	Aug 1947	Turner 1947e
Floor betw. stock room, elevator	Floor, 1 in.			ND	Aug 1947	Turner 1947e
Stacks of UO ₃ cardboard containers, being stored	Contact	1.5			Apr 1949	Wolf 1949
	General area	1				
UO ₃ containers to be shipped	Contact			1.5	May 1952	HCC 1950-1953
Laboratory	Floor, 1 in.			ND	Aug 1947	Turner 1947e
Laboratory, sample & other tables	Top, 1 in.			ND	Aug 1947	Turner 1947e
Work bench, Maintenance Shop	Top, 1 in.			ND	Aug 1947	Turner 1947e
Maintenance room	Floor, 1 in.			ND-<2	Aug 1947	Turner 1947e
Leather glove, Maint Shop, left hand, still being used	Surface, 1 in.			10	Aug 1947	Turner 1947e
Leather glove, Maint Shop, right hand, still being used	Surface, 1 in.			60	Aug 1947	Turner 1947e
Locker Rm #1: shower, lav., area	Floor, 1 in.			<2	Aug 1947	Turner 1947e
Lunchrm #1 (cafeter.), kitchen, rec room	Floor, 1 in.			ND	Aug 1947	Turner 1947e
Lunchroom #2, floor and benches	Floor, 1 in.			<2	Aug 1947	Turner 1947e
Lunchroom #2, table	Top, 1 in.			ND	Aug 1947	Turner 1947e
Lunch, locker, shower rooms	43 in. above floor	BG	BG	BG	May 1948	Hayden 1948
Superintendent's desk in office	Top, 1 in.		<1	<1	Aug 1947	Turner 1947e
1 ft in front of lavat. in Supt's office	Floor, 1 in.		<1	<1	Aug 1947	Turner 1947e
1 ft in front of clerk's desk in office	Floor, 1 in.		<1	5	Aug 1947	Turner 1947e
1 ft in front of filing cabinet in office	Floor, 1 in.		<1	<1	Aug 1947	Turner 1947e
Aisle near hex area, 15 ft W of office	Floor, 1 in.			2	Aug 1947	Turner 1947e
Office, superintendent's	43 in. above floor	BG (.2)	BG	BG	May 1948	Hayden 1948
Drinking fountain	43 in. above floor	3.4	1.8	5.2	May 1948	Hayden 1948
Roof and yard areas^b						
Annex roof, 20 ft fr WE-61 [vent?]						
Roof, main building	Roof, 1 in.			ND	Aug 1947	Turner 1947e
Roof S of UO ₂ -to-UF ₄ tube rack	43 in. above floor	0.64	5.1	5.2	May 1948	Hayden 1948

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Area or source item	Position/distance	Gamma dose rate, mR/hr	Beta dose rate, mreps/hr	Total dose rate, mreps/hr	Date	Reference
Surface, can of UF ₆ cylinder heel	Inside asbestos gloves	2.5			Jun 1945	Tybout 1945a
	Inside leaded glove	136.3				
Roof, main building	Roof, 1 in.			ND	Aug 1947	Turner 1947e
Drum 2/3 full: heel + water washed from cylinders	Contact	13.8			Jun 1945	Tybout 1945a
80-90 horiz. cylinders, at 1 lb UF ₆	6 in. above cylinder pile	2.0-2.5			Jun 1945	Tybout 1945a
Two stacks cylinders: 4 ft high, 20 ft long, 1.5 ft apart, 6 in. from ground	Between the two stacks	0.5			Jun 1945	Tybout 1945a
Pile of cylinders	10 ft from bottom	0			Jun 1945	Tybout 1945a
Pile of cylinders	3 ft from broadside end	0.5-.75			Jun 1945	Tybout 1945a
Line of 25 full cylinders of UF ₆ , laid singly on ground	6 in. above	0.3			Jun 1945	Tybout 1945a
Cylinder storage area (yard?)	General area	3			Apr 1949	Wolf 1949
Outside wall of guard shack near cylinders	Contact, 2 ft off ground	2			Jun 1945	Tybout 1945a
Badge board at guard shack	1 ft in front	0.5			Jun 1945	Tybout 1945a
Railcar floors	Contact?	1	0.0-0.5	1.5	Apr 1951	HCC 1950-1953
Fence opposite trailer, outside	Surface, 1 in.			ND	Aug 1947	Turner 1947e
2 ft down inside sewer	Inside, 1 in.		<2	<2	Aug 1947	Turner 1947e

- a. Question marks indicate that the reference was illegible or that the information was incomplete or ambiguous. For example, "Contact?" indicates that the distance was not specified but the likely measurement point was at contact.
- b. Ash from a faulty collector blew over the roof at times. The above roof ash readings were not considered reliable because the roof and surroundings were contaminated.
- c. Cylinders are UF₆ cylinders, empty except as noted.

Table B-6. Chest and hand beta doses from ash residue handling, as measured by films.^a

Operation	Dose, R	Dose rate, R/hr
Hand (film) dose during ash container transfer	0.4	2.4
Tray handling, day 1 (0.35 hr)		
Chest (regular) film badge, worn 1 d	0.21	0.60
Right wrist	0.225	0.64
Right back of hand	0.2	0.57
Left wrist	0.2	0.57
Left back of hand	0.15	0.43
Tray handling, day 2 (1.05 hr)		
Chest (regular) film badge, worn 1 d	0.28	0.27
Tray handling, day 3 (1.75 hr)		
Chest (regular) film badge, worn 1 d	0.35	0.20
Right wrist	0.45	0.26
Right back of hand	0.3	0.17
Left wrist	0.4	0.23
Tray handling, day 4 (1.75 hr)		
Chest (regular) film badge, worn 1 d	0.35	0.20
Right wrist	0.45	0.26
Right back of hand	0.4	0.23
Left wrist	0.4	0.23
Left back of hand	0.3	0.17
Chest (regular) film badge, worn all 4 d (5.07 hr total)	1.4	0.28

- a. These films were exposed at the same time as the ash residue dose rate measurements in Table B-5 for February 1946 were taken (Tybout 1946; Engel 1946). Dose rate measurements were taken with an ion meter and did not take into account the shielding provided by the cotton and asbestos gloves worn by the tray handlers. The hand films listed above were taped on the skin under the gloves. Note that units of

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roentgen were used by the reference, although the measurement was specifically beta. Missing measurements above are due to the films' being damaged by perspiration and heat.

Table B-7. Weekly dose rates for various workers and areas (from AEC 1949b, Figures 8 and 10).

Position	No.	Average dose rate (mrep/wk)	Median dose rate (mrep/wk)	Minimum dose rate (mrep/wk)	Maximum dose rate (mrep/wk)
August 1944 – January 1949 ^a					
Hex area operator	24	600	630	345	1,145
Hex area loader	25	970	905	270	2,000
Still operator	8	350	250	165	840
Brown loader, reactor operator	16	265	240	170	650
Maintenance	4	280	315	120	595
Recovery	1	360	---	---	---
Supervisor	5	200	220	195	335
Foreman	3	220	210	190	275
Miscellaneous and unclassified	5	190	195	110	335
Clerks	1	100	---	---	---
Plant manager	1	80	---	---	---
		Beta-gamma (mrep/wk at 4 in.)	Beta-gamma (mrep/wk at 43 in.)		
May 1948 ^b					
UO ₂ tray loaders	6?	340	70		
UO ₂ tube handlers	6?	145	80		
UF ₄ Loaders, etc. (day shift)	9	1,640	463		
UF ₄ Loaders, etc. (day shift)	9	885	250		
Still area	6?	137	38		
Recovery room	1	830	243		
Office	4	93	33		

- The figures given for August 1944 to January 1949 are individual averages based on the total dose accumulated over the period, from film badge data. The maximum dose rate is the average dose rate to the most exposed person (in terms of total dose) in the position; the minimum is the average dose rate to the least exposed person.
- The exposures given for May 1948 appeared to arise mostly from UX1 and UX2. Daily dose rates were multiplied by 5 to get weekly dose rates (the probable original units).

Table B-8. Annual neutron whole-body doses from the alpha-neutron reaction^a, various uranium forms.

Worker type	Form	Source ^b	Target	Weight of form in container ^c	Applicable years	Dose rate, 1 ft, rem/hr	Dose rate, 3 ft, rem/hr	Annual dose, rem ^d
Ore/digestion	U ₃ O ₈ (covers Na ₂ U ₂ O ₇)	NU+D	O (Na,O)	100 (75) lb	1949–1954	4.74E-06	5.27E-07	1.58E-03
UO ₃ , UO ₂ load/unloader	UO ₃ , UO ₂	NU	O	75 lb	1949–1954	2.70E-07	3.00E-08	9.00E-05
UF ₄ load/unloader	UF ₄	NU	F	75 lb	1942–1944; 1948–1951	2.26E-05	2.51E-06	7.53E-03
		NU+D			1945–1947	4.04E-04	4.49E-05	1.35E-01
UF ₄ -UF ₆ loading	UF ₄	NU	F	75 lb	1942–1944; 1948–1951	2.26E-05	2.51E-06	7.53E-03
		NU+D			1945–1947	4.04E-04	4.49E-05	1.35E-01
		UF ₆ (receiver)			NU	435 lb	1942–1951	1.31E-04
UF ₆ distillation	UF ₆ (cylinder)	NU	F	462 lb	1942–1951	1.39E-04	1.54E-05	4.63E-02
Shipping & Receiving	UF ₆ (cylinder)	NU	F	462 lb	1942–1948	1.39E-04	1.54E-05	4.63E-02
	U ₃ O ₈ (covers Na ₂ U ₂ O ₇)	NU+D	O (Na,O)	100 (75) lb	1949–1951	4.74E-06	5.27E-07	1.58E-03
	UO ₃	NU	O	75 lb	1952–1954	2.70E-07	3.00E-08	9.00E-05
Laboratory	UO ₂	NU+D	O	75 lb	1942–1951	4.74E-06	5.27E-07	1.58E-03
	Ore/HGE-33/other feed	NU+D	O (Na,O)	---	1949–1954	4.74E-07	5.27E-08	3.16E-04
	UF ₄ (from Harshaw)	NU	F	---	1949–1954	2.26E-06	2.51E-07	1.51E-03
	UF ₆	NU	F	---	1942–1951	2.26E-06	2.51E-07	1.51E-03
	UO ₃ , UO ₂ (from others)	NU+D	O	---	1942–1953	4.74E-07	5.27E-08	3.16E-04
	UO ₃ , UO ₂ (fr Harshaw)	NU	O	---	May 1949-1953	2.70E-08	3.00E-09	1.80E-05

- Dose rates were calculated using data and assumptions from ORAUT (2005c). The neutron doses in Table B-8 should be multiplied by 2 to correct the values calculated from ORAUT (2005c) to the ICRP Publication 60 radiation weighting factor for the energy range (ICRP 1991).
- NU: natural uranium mix, uranium only; NU+D: natural uranium mix with daughters through polonium (i.e., full secular equilibrium).
- Production data are from Gates (1946), Rauch (1948), HCC (1946), Tybout (1944a), Tybout (1945a), Simmons (1945), and AEC (1951c).
- In calculating annual dose, it was assumed for all but the laboratory worker and the office/clerical/manager worker that 4 hr/d was spent handling the container: 1 hour at 1 foot and 3 hours at 3 feet. For the laboratory worker, source quantities were much smaller but close access time was greater; thus

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the dose rates were multiplied by 0.1 to allow for smaller quantities, while for the doses the times spent per day were doubled. For the office/clerical/manager worker, the dose was assumed to be negligible (by inspection of the more exposed worker doses and by consideration of the likelihood of exposure). In all cases, the work year was taken to be 2,000 hours.

Table B-9. Reserved.^a

Table B-10. Air concentrations in various areas in the green and hex plants (Ferry 1944b; Tybout 1945b).

Location	X dust ($\mu\text{g}/\text{m}^3$) ^a	Uranium dust exposure (dpm/m^3) ^b
Green (UF_4) Plant		UO_2/UF_4 exposure (dpm/m^3)
Center of room 69 in. high	230	322
In front of hood, unloading	240	336
Charging in front of hood by scale	200	280
Front center of hood	350	490
In front of tube rack 69 in. high	320	448
Next to record desk	240	336
Hex (UF_6) Plant		UF_6 exposure (dpm/m^3)
East end of corridor	30	42
Center of corridor	89	125
West end of corridor	40	56
East end of Hood 9	370	518
Center of Hood 9	30	42
West end of Hood 9	9,130	12,782

- a. All $\mu\text{g}/\text{m}^3$ values are $\pm 10\%$, except for the last entry, which corresponds to $\pm 20\%$.
- b. The tolerance level is given as $150 \mu\text{g}/\text{m}^3$ of "X dust" (uranium dust) and $40 \mu\text{g}/\text{m}^3$ for UF_6 . For this table, it was assumed that $50 \mu\text{g}$ of uranium was equal to 70 dpm.

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Table B-11. General area and breathing zone dust measurements in the green and hex plants (491 and 492), in alpha dpm/m³.

	Ferry 1944b	Tybout 1945b	Hayden 1948	Harris 1949d, AEC 1951d	AEC 1950d	AEC 1950e, AEC 1950c	AEC 1950c	HCC 1950- 1953	AEC 1951b	AEC 1951d	AEC 1953a	AEC 1953a	AEC 1953a	AEC 1953a
	MED	MED	AEC	AEC	Harshaw	AEC	AEC	Harshaw	AEC	AEC	Harshaw	AEC	Harshaw	AEC
	Feb 1944	Sep 1944	May 1948	Sep 1948	Feb 1949	Sep 1949	Jul 1950	Oct 1950	Jan 1951	Sep 1951	Sep 1952	Jan 1953	Feb 1953	Aug 1953
General Area Samples														
491 general area, postoperations											3	26	7	1
492 general area, postoperations											0	0	11	2
Reactor furnace area			260	532	63	336	126		238	63				
HF tank scale area				469	42	399	49		133	203				
Brown loading room area	336		18,000	441	1,610	525	140		350	77				
Still area			1,300	406	280	602	210		1,274	63				
SW corner of the still area				294	77	658	280							
Hex area		125	175	1,260	511	7,490	252		217	189				
Line cleaning area			1,000											
Central loading room			21,000				98		434	140				
Lunchroom No. 1			150	126	77	56	42		91	56				
Cell-Steam Room area				273	91	7	14		42	21				
Locker Room No. 1			150	413	70	49	56		329	98				
Lunchroom No. 2			150	42	42	91	70		28	70				
Locker Room No. 2			150	84	42	63	63		21	35				
Lunchroom No. 3			150			21	70		14	7				
Locker Room No. 3			150			63	119		84	35				
Maintenance shops				21		98	322		196	28				
Maintenance office				77		28	42		49	14				
Cylinder storage area				84		210	161		196	224				
Cylinder wash area				70		553	35		63	714				
Respirator cleaning area				1.7										
Shipping & Receiving area				2.0	3.5	83.0	0.4		0.9	0.3				
Laboratory area				1.0		3.8	1.0		1.0	0.9				
Guard office				0.8		0.1	0.3		0.2	0.3				
Health Physics						2.1	2.3		0.5	0.9				
Recovery area				18.5			0.8		2.3	1.0				
Recovery Room			35,400	180										
Production Office			160	77		119	56		28	21				
Breathing Zone Samples														
Reactor furnace area														
Disconnect HF lines				910	0.0	371	371		651	595				
Remove hot tube to rack				3,290	0.0	1,687	497		98	70				
Connect HF lines				910		637	2,660		294	259				
Clean up area				8,680										
Still operations														
Break receiver connections						672	35		203	77				
Installing receivers				7,000	210									
Removing receivers				1,610	56									
Take used receivers to storage, put in new ones						448	84		168	126				
Reconnect receiver lines						931	1,960		294	406				
Brown oxide loading														

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	Ferry 1944b	Tybout 1945b	Hayden 1948	Harris 1949d, AEC 1951d	AEC 1950d	AEC 1950e, AEC 1950c	AEC 1950c	HCC 1950– 1953	AEC 1951b	AEC 1951d	AEC 1953a	AEC 1953a	AEC 1953a	AEC 1953a
	MED	MED	AEC	AEC	Harshaw	AEC	AEC	Harshaw	AEC	AEC	Harshaw	AEC	Harshaw	AEC
	Feb 1944	Sep 1944	May 1948	Sep 1948	Feb 1949	Sep 1949	Jul 1950	Oct 1950	Jan 1951	Sep 1951	Sep 1952	Jan 1953	Feb 1953	Aug 1953
Open UO ₂ carton				95,200	57,050	37,100								
Carry tubes from storage to loading hood	490			1,680	1,330	469	126		98	70				
Open tubes				13,300	11,900	651	7,770		6,790	6,090				
Unload UF ₄	336			24,500	30,100	53,200	10,430		9,170	2,310				
Load trays with UO ₂ , put into tubes	280			14,700	5,600	3,080	5,950		7,840	1,890				
Prepare and seal tubes				6,720	9,450	5,180	9,100		4,060	3,710				
Weigh, seal, tag UF ₄ drum				35,000	6,790	115,500	35,910		2,240	10,500				
Clean up area				122,500	1,820	5,180	3,080		1,274	4,760				
Take tubes to UF ₄ storage area	448			1,680	1,330	469	126		98	70				
Hex loading, before installation of the central loading system														
Break lines				28,000	3430	14,000								
Remove hot tube to rack				25,200	2660	7,700								
Load UF ₄ into trays		4,448		210,000	25,900	12,250								
Place trays in furnace				72,100	2,100	12,250								
Secure furnace, connect lines				13,650	980	3,500								
Pull ice traps				56,000	37,800	12,250								
Clean up area				46,900	490	2,100								
Remove ash from tube						15,750								
Special operations (by subcontractor)														
Cutting supports of trays								158						
Welding								64						
Press brake operator, welding								36						

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Table B-12. General area and breathing zone dust measurements in the brown plant (493), in alpha dpm/m³^a.

	Klevin 1950a, AEC 1950e	AEC 1950e	AEC 1951a	AEC 1952a	AEC 1953b	AEC 1953b	AEC 1953b	AEC 1953b	AEC 1953a	AEC 1953a
	AEC	AEC	AEC	AEC	Harshaw	AEC	Harshaw	AEC	Harshaw	AEC
	Mar 1950	Aug 1950	Mar 1951	Jan 1952	Jun 1952	Jul 1952	Sep 1952	Jan 1953	Feb 1953	Aug 1953
General Area Samples										
X-Tank area				29	62	16	59	22	78	8
Y-Tank area				51	50	10	14	29	122	19
Bird centrifuge	126	133	35							
Digestion area	112	42	28	22	82	<1	12	33	49	45
TBP panel board area				70	31	<1	13	8	49	5
Denitration pot area	112	42	63	72	69	77	7.3	41	123	21
Niagara, frame filters (3rd deck)	259	49	21							
Digestion and recovery tank area	336	28	56							
Ether extraction area	42	91	21							
Nitric acid makeup area	140	28	28							
Rockwell area	567	441	315							
Discharge of Rockwells				48	68	8	42	58	93	49
2nd deck over Rockwells				256	114	39	36	100	43	22
3rd deck over Rockwells				1,030	61	110	363	161	75	19
Orange packing area				56	49	157	15	19	125	23
Room 1-A				33	79	86	30	20	41	1.0
Room 1-B				18	66	<1	14	5.6	28	1.0
Room 1-C				1	34	13	12	2.5	17	2.0
Room 2-A				11	78	55	8.6	17	69	4.0
Room 2-B				15	67	360	9.6	4.8	54	25
Room 3-A				17	39	43	9.7	82	34	7.0
Room 3-B (filter press)				17	18	12	14	116	28	15
Room 3-C				9						
Brown production area	546	161	140							
Recovery area				11		3	8.8	9.1	6	1
Production office	189	14	49	8	4	<1	1.6	6.3	12	1
Locker Room No. 3-C, D	49	21	84	27	0	1.5	6.3	11	6	6
Lunchroom No. 3	56	7	42	7	3	0	3.0	11	23	18
Locker Room No. 2-C, D				41	15	17	11	22	23	51
Lunchroom No. 2				32	19	18	12	2.8	33	8
Locker Room No. 1-C, D				30	4	40	2.6	48	19	2
Lunchroom No. 1				63	7	17	3	35	28	2
Laundry - dirty	126	42	84			0	11	27	4	2
Laundry - clean	14									
Laboratory grinding room						5	8.6	650	92	6
Center of main laboratory						8	5	95	23	18
Laboratory office						0	11	0	7	3
Balance Room						0	7.6	2.8	0	4

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	Klevin 1950a, AEC 1950e	AEC 1950e	AEC 1951a	AEC 1952a	AEC 1953b	AEC 1953b	AEC 1953b	AEC 1953b	AEC 1953a	AEC 1953a
	AEC	AEC	AEC	AEC	Harshaw	AEC	Harshaw	AEC	Harshaw	AEC
	Mar 1950	Aug 1950	Mar 1951	Jan 1952	Jun 1952	Jul 1952	Sep 1952	Jan 1953	Feb 1953	Aug 1953
Old laboratory						0	3.6	8.3	8	3
Guardhouse						0	16	11	12	1
Maintenance Office						0	2.8	16	4	4
Maintenance area						0	5.6	7.7	11	2
Stockroom						6	191	29	3	5
Health Physics Office	147									
Shipping & Receiving Office						4.2	4.3	27	21	1
Research & Development Office							0	30	22	8
Research & Development Laboratory						0		7	12	6
Breathing Zone Samples										
Dump drum of U ₃ O ₈ into digest hopper ^a	9,800	161	210	0.0	0.0	49	133	126	1,750	56
Clean empty drums after dumping										196
Provide makeup of nitric acid	140									
Pour liquor into pots	63	42	63							
Gulp UO ₃ from pots to collector	2,730	392	161	196	406	161	77	910	980	203
Clean pots, chip UO ₃ from paddle	43,400	(b)								
Unload UO ₃ from pot into drum, break lumps	29,400	(b)								
Shovel lumps into UO ₃ crusher	7,700	(b)								
Feed orange from pots to Rockwell (3rd deck)	2,730									
Feed orange from pots to Rockwell (vacuum system)										
Discharge UO ₂ from Rockwell into can, weigh, store	31,500	16,520	17,220							
Discharge drum UO ₃ into 30-gal drum, store				420	161	1,820	686	2,870	105	581
Shake the bag of the UO ₃ collector				14,980						
Change drum under UO ₃ collector				3,570	616	1,540	350	2,030	4,270	2,380
Shake bag of floor sweepings				3,570						
Clean and dump filter press	350	525	581	539	287	175	399	203	483	546
Grind UO ₃ ^a								140	119	21
Transfer UO ₃ from blender to sample jar ^a								280		
Transfer UO ₃ from sampling can to blender ^a								3,500		
Transfer UO ₃ from blender to grinder ^a								1,050		
Fill UO ₃ into weighing bottles from grinder ^a								3,500	<70	21
Determine density of UO ₃ ^a								133	<70	14
Clean blender with vacuum cleaner ^a									378	56
Build Bird filter cake	182	119	49							
Plow off filter cake	182	392	91							
Load press cake into drums								42		

- a. The grinding and related operations were performed infrequently, per AEC (1953b). In AEC (1953a) and AEC (1953b), the UO₃ was referred to as JH-6 or HGE, which appear to denote Hanford forms. It appears that most of the UO₃ was "gulped" (removed by vacuum system) from the pots, but that lumps too large to suction had to be removed with scoops ("Unload UO₃ from pot. ...").
- b. These operations were new in January or February 1953, per AEC (1953a) and AEC (1953b).
- c. These operations were supposed to have been discontinued prior to this survey, but per AEC (1951a), Harshaw still found it necessary to do some lump crushing.

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Table B-13. Dust measurements in the green and brown plants, in beta dpm/m³.^a

Occupation	Operation	Beta concentration, dpm/m ³	Alpha concentration, dpm/m ³	Ratio, beta to alpha
Brown oxide loader No. 1	Opening cartons of UO ₂ , dumping into barrel	46,000	95,250	0.48
Brown oxide loader No. 2	Cleaning green dumping counter	59,000	122,500	0.48
Hex loader (new area)	Dumping 300-lb barrel of UF ₄ into hopper of mechanical loader	232,000	397,000	0.58
Hex area loader	Loading tray with UF ₄	134,000	207,440	0.65
Hex area loader	Cleaning tray (ash and unreacted UF ₄)	57,000	23,300 ^b	2.45
Hex area hood	Pulling ice trap	39,000	56,000	0.70

- a. Beta data are from AEC (1948). The corresponding alpha data is from Table B-10.
b. It appears that the alpha value was not averaged in the typed record, as can be seen from the original sample values given in the handwritten sample record. The correct average has therefore been put into the table above.

Table B-14. Daily DWEs to airborne dust, in alpha dpm/m³.^a

	Hayden 1948	Klevin 1950a	AEC 1950a	AEC 1950a
	491/492	493	493	493
	May 1948	Mar 1950	Mar 1950	Aug 1950
Boildown tank and de-etherizer			<70	<70
Ether extraction		42	<70	1.4
Bird centrifuge/filter		126	1.4	2.8
Niagara and frame filter		259	4.2	1.4
No. 1 Digester (batch) and nitric acid recovery		336	4.2	1.4
Nitric acid makeup area		140		
Rockwell furnace		567	67	17
Denitration pots		112	84	210 ^b
General 493 production area		546		
Hoods, UF ₄ -to-UF ₆	21,000			
Control aisle, UF ₄ -to-UF ₆	175			
Still area	1,300			
Recovery Room	36,400			
Line cleaning area	1,000			
Hoods, UO ₂ -to-UF ₄	18,000			
General, UO ₂ -to-UF ₄	260			
Clean laundry room (East)		14		
Dirty laundry room (West)		126		
Lunch/locker-shower rooms	150	56/49		
Office	160			
Production Office (493)		189		
Health Physics Office		147		
Average of daily weighted exposures			22	2.8

- a. From the references, the data in this table represent daily weighted averages for partial time spent by a typical worker in an area, without identification of any job title. Thus, these data can apparently be used for workers with time split on a daily or weekly basis between areas in nonroutine ways, such as when substituting for a sick coworker.
b. Some major dusty parts of this operation are not included in this average.

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Table B-15. Daily DWEs to airborne dust for various job titles, in alpha dpm/m³ ^a

Operation or Job	Hayden	AEC	AEC	AEC	Klevin	AEC	AEC	AEC	AEC	AEC	AEC	Klevin	AEC	AEC	AEC	AEC	AEC
	1948	1950b	1950b	1950b	1950a,	1950b	1950c,	1951b	1951a	1951d	1952a	1952b	1952b	1953a	1953b	1953a	1953a
	491/492	491/492	491/492	491/492	493	491/492	493	491/492	493	491/492	493	493	493	493	493	493	493
	AEC	AEC	AEC	AEC	AEC	AEC	AEC	AEC	AEC	AEC	AEC	Harshaw	AEC	Harshaw	AEC	Harshaw	AEC
May	Sep	Feb	Sep	Mar	Jul	Aug	Feb	Mar	Sep	Jan	Jun	Jul	Sep	Jan	Jun	Aug	
1948	1948	1949	1949	1950	1950	1950	1951	1951	1951	1952	1952	1952	1952	1953	1953	1953	
Brown oxide loaders	11,480	11,480	5,880	9,800		2,660		2,800		1,540							
Cleaners		987	294	1,729		70											
Foremen and supervisors (491/492)		840		2,506		147		168		126							
General foreman (491/492)		420		2,520		133		182		84							
Hex area loaders	5,411	23,800	3,780	7,280		2,660		1,610		672							
Hex area operators		1,050	308	5,600		196		175		196							
Laboratory personnel		105		210		69		62		53.9							
Process engineer (491/492)		New		2,520													
Reactor furnace (incl tube handlers)	245	1,190	98	539		189		203		112							
Recovery operator	6,965	15,120		Not in operation		67		168		70							
Shift foreman						140		175		147							
Still operators	1,015	1,274	231	497		238		910		64.4							
Analyst														4.9	46.2	23.1	16.1
Assistant superintendent (493)					399				63		21.7	11.2	7.7	9.8	11.9		
Chief chemist														11.2	14	13.3	7.7
Chief/lead process man (493)					399		140		112		70	46.2	56	18.9	23.1	98	30.8
Foremen and supervisors (493)					2,520		147		105		55.3	25.2	7.7	2.8	13.3	35	6.3
General foreman (493)					399		67										
Process engineer (493)					399				119		63	31.5	23.1	25.2	23.1	30.1	7.7
Process helper/man: ether extraction					49		84		32		60.2	33.6	7.7	18.2	13.3	35	7
Process helper/man: Niagara, frame filters					203		91		77								
Process helper: Bird filter/centrifuge					140		98		44								
Process helper: boildown tank, de-etherizer					49		49									98	73.5
Process helper: general											119	91	67.2	77	49.7	105	77
Process man: Bird centrifuge					203		140										
Process man: boildown tank											91	56	77	53.9	49.7		
Process man: denitration pots					4,200		175		105		105	98	91	30.8	189	189	58.8
Process man: digester, nitric acid recovery					210		70		52		20.3	44.1	43.4	16.8	35.7	301	39.9
Process man: Rockwell furnace					3,220		728		840								
Research and development														2.8	14	13.3	7.7
Sampler (KoZ) ^b															651		9.8
Clerk		168		98		73		73		51.8							
Guards		35		7		35		42		30.8							
Health physicist					1,820	154		161		126							
Laundry man				1,680	66	154	154		88					9.8	25.2	9.1	4.9
Maintenance and repair		210		910		217		196		84				20.3	23.1	21	7
Maintenance foreman		New title		910		147		126		37.8							
Office					182												
Shipping and Receiving		133		2,450		62		93		168				35	35	30.8	9.1

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Operation or Job	Hayden 1948	AEC 1950b	AEC 1950b	AEC 1950b	Klevin 1950a, AEC 1950a	AEC 1950b	AEC 1950c, AEC 1950e	AEC 1951b	AEC 1951a	AEC 1951d	AEC 1952a	Klevin 1952b	AEC 1952b	AEC 1953a	AEC 1953b	AEC 1953a	AEC 1953a
	491/492	491/492	491/492	491/492	493	491/492	493	491/492	493	491/492	493	493	493	493	493	493	493
	AEC	AEC	AEC	AEC	AEC	AEC	AEC	AEC	AEC	AEC	AEC	Harshaw	AEC	Harshaw	AEC	Harshaw	AEC
	May 1948	Sep 1948	Feb 1949	Sep 1949	Mar 1950	Jul 1950	Aug 1950	Feb 1951	Mar 1951	Sep 1951	Jan 1952	Jun 1952	Jul 1952	Sep 1952	Jan 1953	Jun 1953	Aug 1953
Stock Room														84	18.9	9.1	4.9
Superintendent (491/492/493)		420		609	469	133		59		42						16.1	4.2
Average weighted exposures		4,690	2,730	2,380	1,155	441	182	560	154	266	63.7	51.1	42.7	23.1	45.5	60.2	22.4

- a. Improvements were made between September 1948 and February 1949, hence the change in levels for certain jobs (AEC 1948, 1949b).
- b. The job of sampler was new as of January 1952.

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Table B-16. Annual inhalation intakes based on daily weighted average exposures to airborne dust for various job titles, in pCi.

491/492 (UF₄/UF₆) workers	1942-1947	1948	1949
Brown oxide loaders	1.25E+07	1.25E+07	8.55E+06
Cleaners	1.08E+06	1.08E+06	1.10E+06
Foremen and supervisors (491/492)	9.16E+05	9.16E+05	2.73E+06
General foreman (491/492)	4.58E+05	4.58E+05	2.75E+06
Hex area loaders	5.90E+06	2.60E+07	6.03E+06
Hex area operators	1.15E+06	1.15E+06	3.22E+06
Hex central loaders ^a	---	---	---
Laboratory personnel	1.15E+05	1.15E+05	2.29E+05
Process engineer (491/492)	---	2.75E+06	2.75E+06
Reactor furnace (incl tube handlers) (hex)	2.67E+05	1.30E+06	3.48E+05
Recovery operator	7.60E+06	1.65E+07	8.25E+06
Shift foreman	8.18E+05	8.18E+05	2.30E+06
Still operators	1.11E+06	1.39E+06	3.97E+05
Generic 491 (UF ₄) process worker	1.25E+07	1.25E+07	8.55E+06
Generic 492 (UF ₆) process worker	3.20E+06	9.26E+06	3.65E+06
493 (UO₂/UO₃) workers			1949^b
Analyst			---
Assistant superintendent (493)			4.35E+05
Chief chemist			2.52E+05
Chief/lead process man (493)			4.35E+05
Foremen and supervisors (493)			2.75E+06
General foreman (493)			4.35E+05
Process engineer (493)			4.35E+05
Process helper/man: ether extraction			5.35E+04
Process helper/man: Niagara/frame filters			2.21E+05
Process helper: Bird filter/centrifuge			1.53E+05
Process helper/man: boildown tank/de-etheriz.			5.35E+04
Process helper: general			5.03E+06
Process man: Bird centrifuge			2.21E+05
Process man: denitration pots			4.58E+06
Process man: digester, nitric acid recovery			2.29E+05
Process man: Rockwell furnace			3.51E+06
Research and development			---
Sampler (KoZ)			---
Generic 493 (UO ₂ /UO ₃) process worker			1.45E+06
Support and management workers^c	1942-1947	1948	1949
Clerk (production)	1.83E+05	1.83E+05	1.07E+05
Guard	3.82E+04	3.82E+04	7.64E+03
Health physicist	4.80E+05	4.80E+05	2.87E+06
Laundry man	---	---	1.83E+06
Maintenance and repair	2.29E+05	2.29E+05	9.93E+05
Maintenance foreman	2.29E+05	2.29E+05	9.93E+05
Office	1.78E+05	1.78E+05	2.57E+05
Shipping and Receiving	1.45E+05	1.45E+05	2.67E+06
Stock Room	6.44E+04	6.44E+04	1.18E+06
Superintendent (491/492/493)	4.58E+05	4.58E+05	6.64E+05
Overall Average Weighted Exposures	1942-1947	1948	1949
491/492 basis	2.53E+06	5.12E+06	2.79E+06
493 basis	---	---	---

- a. This job began in mid-1950, so this figure should be applied during only the second half of 1950.
- b. UO₂/UO₃ plant production started in May 1949, so this column is applicable only for May–December 1949.

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Table B-17. Annual ingestion intakes based on daily weighted average exposures to airborne dust for various job titles, in pCi.

491/492 (UF₄/UF₆) workers	1942-1947	1948	1949
Brown oxide loaders	2.63E+05	2.63E+05	1.80E+05
Cleaners	2.27E+04	2.27E+04	2.31E+04
Foremen and supervisors (491/492)	1.92E+04	1.92E+04	5.73E+04
General foreman (491/492)	9.62E+03	9.62E+03	5.78E+04
Hex area loaders	1.24E+05	5.46E+05	1.27E+05
Hex area operators	2.42E+04	2.42E+04	6.76E+04
Hex central loaders ^a	---	---	---
Laboratory personnel	2.42E+03	2.42E+03	4.81E+03
Process engineer (491/492)	---	5.78E+04	5.78E+04
Reactor furnace (incl tube handlers)(hex)	5.61E+03	2.73E+04	7.31E+03
Recovery operator	1.60E+05	3.47E+05	1.73E+05
Shift foreman	1.72E+04	1.72E+04	4.83E+04
Still operators	2.33E+04	2.92E+04	8.34E+03
Generic 491 (UF ₄) process worker	2.63E+05	2.63E+05	1.80E+05
Generic 492 (UF ₆) process worker	6.72E+04	1.94E+05	7.67E+04
493 (UO₂/UO₃) workers			1949^b
Analyst			---
Assistant superintendent (493)			9.14E+03
Chief chemist			5.29E+03
Chief/lead process man (493)			9.14E+03
Foremen and supervisors (493)			5.78E+04
General foreman (493)			9.14E+03
Process engineer (493)			9.14E+03
Process helper/man: ether extraction			1.12E+03
Process helper/man: Niagara/frame filters			4.64E+03
Process helper: Bird filter/centrifuge			3.21E+03
Process helper/man: boildown tank/de-etheriz.			1.12E+03
Process helper: general			1.06E+05
Process man: Bird centrifuge			4.64E+03
Process man: denitration pots			9.62E+04
Process man: digester, nitric acid recovery			4.81E+03
Process man: Rockwell furnace			7.37E+04
Research and development			---
Sampler (KoZ)			---
Generic 493 (UO ₂ /UO ₃) process worker			3.05E+04
Support and Management Workers^c	1942-1947	1948	1949
Clerk (production)	3.84E+03	3.84E+03	2.25E+03
Guard	8.02E+02	8.02E+02	1.60E+02
Health physicist	1.01E+04	1.01E+04	6.03E+04
Laundry man	---	---	3.84E+04
Maintenance and repair	4.81E+03	4.81E+03	2.09E+04
Maintenance foreman	4.81E+03	4.81E+03	2.09E+04
Office	3.74E+03	3.74E+03	5.40E+03
Shipping and Receiving	3.05E+03	3.05E+03	5.61E+04
Stock Room	1.35E+03	1.35E+03	2.48E+04
Superintendent (491/492/493)	9.62E+03	9.62E+03	1.39E+04
Overall Average Weighted Exposures	1942-1947	1948	1949
491/492 basis	5.31E+04	1.08E+05	5.86E+04
493 basis	---	---	---

a. This job began in mid-1950, so this figure should be applied during only the second half of 1950.

b. UO₂/UO₃ plant production started in May 1949, so this column is applicable only for May–December 1949.

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Table B-18. Annual radon exposure, in WLM/yr.^a

Area or worker type	Main work area				Break and other low-exposure areas				Total, WLM/yr
	pCi/L Rn	Eq factor	Occ factor	WLM/yr	pCi/L Rn	Eq factor	Occ factor	WLM/yr	
Ore Storage/Warehouse	1.0	1.0	0.75	0.09	1.0	0.4	0.25	0.01	0.10
Digest/Feed	3.0	0.5	0.75	0.14	1.0	0.4	0.25	0.01	0.15
Extraction Cells	1.0	0.5	0.75	0.05	1.0	0.4	0.25	0.01	0.06
Centrifuge	1.0	0.5	0.75	0.05	1.0	0.4	0.25	0.01	0.06
Feinc/Filter/Raffinate/Niagara	7.0	0.5	0.75	0.32	1.0	0.4	0.25	0.01	0.33
Pot Room	2.0	0.5	0.75	0.09	1.0	0.4	0.25	0.01	0.10
UO ₃ packing area	12	0.5	0.75	0.54	1.0	0.4	0.25	0.01	0.55
UO ₃ -UO ₂ area ^b	2.0	0.5	0.75	0.09	1.0	0.4	0.25	0.01	0.10
Green (UO ₂ -UF ₄) area ^b	2.0	0.5	0.75	0.09	1.0	0.4	0.25	0.01	0.10
Hex (UF ₄ -UF ₆) area ^b	2.0	0.5	0.75	0.09	1.0	0.4	0.25	0.01	0.10
Distillation (UF ₆) and recovery ^b	2.0	0.5	0.75	0.09	1.0	0.4	0.25	0.01	0.10
Decontamination (normal oper. cleanup)	1.0	0.5	0.75	0.05	1.0	0.4	0.25	0.01	0.06
Nitric acid storage area	1.0	0.5	0.75	0.05	1.0	0.4	0.25	0.01	0.06
Ether storage area	2.0	0.5	0.75	0.09	1.0	0.4	0.25	0.01	0.10
Shipping & Receiving (non-ore)	1.0	0.5	0.75	0.05	1.0	0.4	0.25	0.01	0.06
Welding Shop	1.0	0.5	0.75	0.05	1.0	0.4	0.25	0.01	0.06
Millwright Shop	1.0	0.5	0.75	0.05	1.0	0.4	0.25	0.01	0.06
Electrical Shop	1.0	0.5	0.75	0.05	1.0	0.4	0.25	0.01	0.06
Maintenance Shop	4.0	0.5	0.75	0.18	1.0	0.4	0.25	0.01	0.19
Shotgun Lab (UO ₃ assay)	4.0	0.5	0.88	0.21	1.0	0.4	0.12	0.01	0.22
Other Lab (Research, Analytical, etc.)	1.0	0.5	0.88	0.05	1.0	0.4	0.12	0.01	0.06
Office workers	1.0	0.4	0.88	0.04	1.0	0.4	0.12	0.01	0.05
Yards and other outdoor areas	1.0	---	---	---	1.0	---	---	---	---

a. See Section 5.5 for assumptions for this table.

b. Exposures in these areas were assumed to be bounded by those in the Pot Room.

Table B-19. General area dust concentrations in the locker rooms and lunchrooms, in alpha dpm/m³.

Survey	Locker rooms			Lunchrooms		
	1	2	3	1	2	3
AEC, 9/1948 (AEC 1948)	411	84	15	126	40	30
Harshaw(?), 2-5/1949 (AEC 1950d)	70	42		77	42	
AEC, 9/1949 (AEC 1950d)	49	63	63	56	91	21
Harshaw, 11/1949: Position 1/19C (Klevin 1950a)	78	51		194	35	
Harshaw, 11/1949: Position 2/19D (Klevin 1950a)	109	72				
Harshaw, 11/1949: Position 3 (Klevin 1950a)		37				
AEC, 3/1950 (Klevin 1950a)	50	50	34	55	55	21
AEC, 7/1950 (AEC 1950c)	56	63	119	42	70	70
AEC, 8/1950(AEC 1950e)			21			7
AEC, 1/1951 (Blatz 1950a,b,c)	329	21	84	91	28	14
AEC, 3/1951 (AEC 1951a)			84			42
AEC, 9/1951 (AEC 1951d)	98	35	35	56	70	7
AEC, 1/1952 (AEC 1952a)	30	41	27	63	32	7
Harshaw, 6/1952 (AEC 1952b)	4	15	0	7	19	3
AEC, 7/1952 (AEC 1952b)	40	17	1.5	17	18	0
Harshaw, 9/1952 (AEC 1953b)	2.6	11	6.3	3	12	3
AEC, 1/1953 (AEC 1953b): Position 1/C	31	21	22	35	2.8	11
AEC, 1/1953 (AEC 1953b): Position 2/D	64	22	0	35	2.8	11
Harshaw, 2/1953 (AEC 1953a)	19	23	6	28	33	23
AEC, 8/1953 (AEC 1953a)	2	51	6	2	8	18

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Table B-20. Average measured dust and urine concentrations by month (Lippmann 1958).^a

Month	Soluble uranium: UF ₄ -to-UF ₆		Insoluble uranium: UO ₂ -to-UF ₄							
	Hex operators Air sample Conc (µg/m ³)	Still operators Air sample Conc (µg/m ³)	UO ₂ loaders/UF ₄ unloaders			Reactor furnace operators ^b				
			Air sample Conc (µg/m ³)	Urine sample		Air sample Conc (µg/m ³)	Urine sample			
			B, µg/L	A, µg/L	B/A ratio		B, µg/L	A, µg/L	B/A ratio	
1950										
January	440	620	9,700	160	68	2.4	170			
February	370	640		110	40	2.8	150	49	18	2.7
March			9,700	120	48	2.5		110	35	3.1
April	170	420	9,500	140	33	4.2		130	7	18.6
May				200	40	5.0		71	13	5.5
June	200	320	8,100	150	70	2.1	230	67	21	3.2
July	140	170	4,400	210	45	4.7	110	65	22	3.0
August		260	3,700	50	28	1.8		33	18	1.8
September								45	8	5.6
October	170		2,600	81	49	1.7	140	30	17	1.8
November	470	160	2,500	61	35	1.7	140	30	17	1.8
December	81	310	2,800	51	15	3.4	180	21	11	1.9
1951										
January	88	500	2,200	40	20	2.0	160	40	8	5.0
February	180	940	2,600	42	21	2.0	110	30	13	2.3
March	300	160	3,600	49	29	1.7	120	12	6	2.0
April	130	120	3,600	49	29	1.7	120	12	6	2.0
May		240	1,800	54	25	2.2		53	15	3.5
June	140	90	1,800	54	25	2.2	200	75	22	3.4
July			1,300					50	12	4.2
August	100	250			18			67	11	6.1
September	130	150	1,300		21		88		18	
October	61	91								
November										
December	160									
Summary data										
Median	160	250	2,800	61	29	2.2	140	49	14	3.1
Std deviation	0.58	0.70	0.67	0.58	0.43	0.36	0.27	0.65	0.48	0.59
Sample periods	17	17	17	17	19	17	13	19	20	19

- a. Each data point is the average of worker samples. B = before-weekend sample, usually Friday; A = after-weekend sample (i.e., Monday morning). Statistical analysis of this data was done for this site profile. A lognormal distribution was assumed.
- b. The reactor furnace operators were the process men in the green (UO₂-to-UF₄) area.

Table B-21. Number of workers.

Date	Plant	No. of workers ^a	Reference
September 1948	Green/Hex	100	AEC 1948
September 1949	Green/Hex	121	AEC 1950d
July 1950	Green/Hex	131	AEC 1950c
January 1951	Green/Hex	127	AEC 1951b
September 1951	Green/Hex	125	AEC 1951d
March 1950	Brown	42	Klevin 1950a
August 1950	Brown	33	AEC 1950e
March 1951	Brown	39	AEC 1951a
January 1952	Brown (Orange)	34	AEC 1952a
July 1952	Brown (Orange)	34	AEC 1952b
September 1952	Brown (Orange)	68	AEC 1953b
January 1953	Brown (Orange)	69	AEC 1953b
February–June 1953	Brown (Orange)	69	AEC 1953a
August 1953	Brown (Orange)	69	AEC 1953a
May 1954	Brown (Orange)	5	Klevin 1954

- a. Figures shown represent all workers dedicated to the plant listed, including management and clerks, but do not include support workers from other (non-AEC/MED) areas of the Harshaw site.

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Table B-22. Job titles, functions, and appropriate absorption types.

Job title or classification	Notes	Absorption types appropriate for work area
491 Area (UO₂-to-UF₄, or green plant)		
2nd-level tray loaders	Load UO ₂ onto trays and the trays into tubes; empty trays of UF ₄	M, S
2nd-level tube handlers	Move tubes of UO ₂ to the UO ₂ -to-UF ₄ furnace and to rack	M, S
Brown (oxide) loader	Load UO ₂ onto trays and the trays into tubes	M, S
Chief process man	Control acid flow, check process and leaks, remove hot tubes from furnace; may include the term Brown	M, S
Cleaner	General area cleanup	M, S
Foreman	May include the term Green, Maintenance, or Assistant	M, S
General foreman	For both 491 and 492	F, M, S,
Green area (worker)	Generic for process man, operator, or the like; use geometry values for the appropriate such title or the generic values at right	M, S
HL-7 (worker)	See Green area (worker)	M, S
Loader	Loaded UO ₂ onto trays and the trays into tubes	M, S
Process man, PM	Assist chief process man	M, S
Reactor (furnace) operator	UO ₂ -UF ₄ reactor tender	M, S
Reaction operator	See Reactor (furnace) operator	M, S
Recovery operator	Fume recovery room/turbosaturator recovery room	F, M, S
Shift foreman		M, S
Supervisor	Might include the term <i>green</i>	M, S
Tray loader	Load UO ₂ onto trays and the trays into tubes; empty trays of UF ₄	M, S
Tube handler	Moved tubes of UO ₂ to the UO ₂ -to-UF ₄ furnace and to rack	M, S
WE-61	Function unclear	M, S
492 Area (UF₄-to-UF₆ and UF₆ distilling, or hex plant)		
1st-Level loader	Loader of UF ₄ in the UF ₄ -to-UF ₆ process	F, M
Chief process man, CPM	Check cells and all hex operations, add acid	F, M
Cleaner	Cleaned areas and respirators	F, M
Cyl cleaning	Cylinder cleaning: normally done by Shipping and Receiving, but possibly done by others; see Shipping and Receiving or appropriate other title	F, M
Engineer	Plant, process, production, dev(elopment); apparently not same as 493 eng	F, M, S
Foreman	Might include the term <i>hex, maintenance, or assistant</i>	F, M
Fume recovery room operator	See Recovery operator	F, M, S
General foreman	For both 491 and 492	F, M, S
Hex area (central) loader	Loader of UF ₄ in the UF ₄ -to-UF ₆ process, newer loading area	F, M
Hex area operator	Loader of UF ₄ in the UF ₄ -to-UF ₆ process	F, M
Laborer	Might include the term <i>maintenance or recovery</i>	F, M, S
Loader	Load trays of UF ₄ , put into furnaces, replace full receivers, clean ice traps	F, M
Meter reader		F, M
Office, office clerk		F, M, S
Operator	Same as process man?	F, M
Process helper, PH	Connect and disconnect lines, unplug (clear) lines	F, M
Process man, PM	Assist chief process man, connect lines, unplug (clear) lines	F, M
Recovery operator	Fume recovery room/turbosaturator recovery room	F, M
Shift foreman		F, M
Still operator	UF ₆ distillation process	F, M
Supervisor	Might include the term <i>hex</i>	F, M
Tray loader	Load UF ₄ into trays, load trays into hex reactors, unload trays, receivers	F, M
Welder	Contract welding	F, M
493 Area (ore-to-UO₂ and ore-to-UO₃, or brown plant)		
Analyst	Lab work	M, S
Batch makeup operator	Add ore and nitric acid to digester	M, S
Brown area (worker)	Generic for process man, process helper, or the like	M, S
Chemist	Lab work	M, S
Chief chemist	Lab work	M, S
Chief process man, CPM	Work at the TBP panel board and tanks, change drums and bags	M, S
Engineer, process	Apparently separate from the 492 engineer	M, S

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Job title or classification	Notes	Absorption types appropriate for work area
493 Area (continued)		
Foreman	Might include the term <i>brown, maintenance, or assistant</i>	M, S
Laboratory material handling	Help dump presses, make up caustic solution, handle feed material for batch makeup	M, S
Lead man	See chief process man	M, S
PH	Process helper	M, S
Process helper: Bird filter/centrifuge	Run centrifuge, draw sample and weigh, pump liquor to tanks	M, S
Process helper: boildown tank & de-etherizers	Send filtrate from Niagaras to extraction tank, de-etherize hex liquor, pump to concentrators for second boildown; send concentrate to pots	M, S
Process helper: filter/filter press	Same as process helper: Niagara filters?	M, S
Process helper: Niagara filters	Precoat, filter batches through Niagaras, pump filtrate to boildown tanks, dump cake into process tanks, clean filter; might include the term <i>frame filter</i>	M, S
Process man: batch makeup, digester, acid recovery	Make up batches of feed and acid, check digestion instruments, check recovery towers, pump acid from towers	M, S
Process man: denitrating pots	Fill pots with hex liquor, check furnace instruments, remove, delump UO ₃	M, S
Process man: extractor (ether)	Measure hex liquid, add ether; add water layer; transfer washings; reextract liquor, check instruments and ether storage	M, S
Process man: filter/filter press	Similar to process helper: Niagara filters?	M, S
Process man, PM	Similar to process helper	M, S
Process man: Rockwell	Keep feed hoppers filled, check feed table, package product in drums, check and record temperatures	M, S
Process man: tank	See process man: boildown tank	M, S
Recovery operator	Recovery of nitric acid; also cleaned cake off filters	M, S
Research and development	Work in R&D (chemistry and process) lab, office areas	M, S
Sampler	Lab prep: UO ₃ , HGE, and JH-6 grinding and blending, weigh JH-6 and UO ₃ , cold-press UO ₃ in die; might contain the term <i>KOZ</i>	M, S
Spec main	Special maintenance?	M, S
Statistician	Assay work	M, S
Stock room		M, S
Supervisor	May include the term <i>brown</i>	M, S
Generic or All Areas or "490"		
Accounting clerk	In Plant C office, but not in the plant itself	F, M, S
AEC	Employee of AEC	F, M, S
Analyst	Chemical analyses?	F, M, S
Captain of guards	For all of Harshaw site; less than 1/3 of time spent in Plant C	F, M, S
Chemist		F, M, S
Clerk	For production clerks, some production area access	F, M, S
Dev maint	Unclear: might be maintenance in pilot plant and lab areas	F, M, S
Electrician		F, M, S
Engineer	Less than 1/3 of time in plant	F, M, S
File clerk	In the Plant C office, but not in the plant itself	F, M, S
Fork truck operator		F, M, S
Guard	Less than 1/3 of time in plant	F, M, S
Health physicist	Health physics office was in the 493 area, but served all areas	F, M, S
Instrument man	Presumably serviceman for instruments	F, M, S
Janitor	If office janitor, did not enter plant	F, M, S
Laboratory assistant		F, M, S
Laboratory personnel	Generic	F, M, S
Laborer	Labor, maintenance, recovery	F, M, S
Laundry man	The two laundries were in the 493 area, but served all areas	F, M, S
Maintenance and repair	Worked out of maint. office, served all areas; might include term <i>Devel(op)</i>	F, M, S
Maintenance foreman	Worked out of the maintenance office, served all areas	F, M, S
Mechanic		F, M, S
Medical technician	Not in plant	F, M, S
Miscellaneous and unclassified	Unclear what was meant by this AEC-used term	F, M, S
Office	Generic for worker in office; might or might not have access to process areas	F, M, S

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Job title or classification	Notes	Absorption types appropriate for work area
Generic (continued)		
Office manager	Less than 1/3 of time in plant	F, M, S
Operator	Generic usage when otherwise unclear -- see "process man" and like titles	F, M, S
Plant clerk	Same as production clerk? Less than 1/3 of time in plant	F, M, S
Plant manager	Generally in charge of all Plant C areas, although at one time there appeared to be different 491/492 and 493 managers	F, M, S
Porter	Unclear if there was production area access	F, M, S
Production clerk	Some plant access	F, M, S
Production office	The Production Office was in the 493 area	F, M, S
Property clerk	Less than 1/3 of time in plant	F, M, S
Research and development	Working in the R&D lab, apparently in the 493 area	F, M, S
Shipping and receiving	Besides handling materials, washed the UF ₆ cylinders	F, M, S
Security	See Guard	F, M, S
Spec maint	Special maintenance, assumed to be the same as maintenance	F, M, S
Stenographer	In office, but not in the plant	F, M, S
Stockroom (worker)	Apparently handled uranium forms or spent time in or near process areas	F, M, S
Sweeper	Assumed to be the same as Cleaner (see 492)	F, M, S
Superintendent	See plant manager	F, M, S

Table B-23. Results of a survey of the Harshaw site by Argonne National Laboratory, 1976 to 1979 (DOE 1984).

Building	Area	Max (avg high) alpha (dpm/100 cm ²) ^a	Typ max alpha (dpm/100 cm ²)	Max b-g (dpm/100 cm ²)	Typ max b-g (dpm/100 cm ²)	Max contact b-g (mR/hr)	Avg max contact b-g (mR/hr) ^a	Avg contact b-g (mR/hr) ^a	Max 3-ft b-g (mR/hr) ^c	Radon (pCi/L)	Radon, 10 ⁻³ WL ^b
Plant C	1st floor, new	---	---	20,000	---	---	---	---	---	---	---
	1st floor, main	21,000 (13,000)	1,300	1,100,000	455,000	3	1.8	0.16	0.1	---	---
	2nd floor	30,000	7,360	1,100,000	870,000	1.4	1.2	0.6	0.07	---	---
	Unspecified	---	---	---	---	---	---	---	---	.25 - .69	2.5 - 6.9
Boiler house		300	0	150,000	0	3	0.0	0	---	0.47	4.7
Foundry		13,333	1,300	333,333	0.3	3	2.3	0.3	---	.17 - .56	1.7 - 5.6
Garage		0	0	0	0	0	0.0	0	---	0.37	3.7
Warehouse	Lab and office	1,000	1000	6,000	0	0.05	0.05	0	---	.33 - .60	3.3 - 6.0
	Second floor	0	0	2,000	0	0.02	0.02	0	---	.33 - .60	3.3 - 6.0
H-1		0	0	0	0	0.03	0	0	---	0.5	5
H-2		0	0	0	0	0.03	0	0	---	0.59	5.9
K-1		10,000	1,500	50,000	17,000	1.5	0.80	0.4	---	.11 - .57	1.1 - 5.7
M-1		10	0	50,000	12,000	0.5	0.07	0.0	---	0.35	3.5
P-1	1st floor	25,000	7,150	200,000	67,000	0.7	0.6	0.2	---	0.33	3.3
	2nd floor	0	0	20,000	6,400	0.2	0.2	0.07	---	0.12	1.2

- The "average" designation above indicates the average of the reported measurements for spots where at least one type of reading was greater than background (as interpreted from DOE 1984). The vast majority of measurements were at background and thus the true average was very much lower than is indicated here.
- This assumes radon daughter equilibrium (DOE 1984). The U.S. Environmental Protection Agency limit at the time was given as 0.02 WL (DOE 1984).
- The ambient penetrating radiation level (by Eberline PRM-7 µR meter) was found to be 40 times less than the contact Geiger-Mueller radiation level (DOE 1984).

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Table B-24. Source terms used to calculate inhalation and radon doses, D&D/postoperations period.^a

Building	Dose potential	Alpha, dpm/100 cm ²	Principal reference
AEC decontamination			
Old UF ₆ area (492)	Low	2.00E+03	Schoen 1958
(excl. Recov, Still)	Moderate	5.90E+03	Klevin 1955a
	High	6.60E+04	As per Recovery
Old UF ₆ area (492)	Low	2.00E+03	Schoen 1958
(Recov, Still only)	Moderate	1.35E+04	Schoen 1958
	High	1.50E+05	Schoen 1958
New UF ₆ area (annex)	Low	2.00E+03	Schoen 1958
(492/493)	Moderate	9.00E+03	Schoen 1958
	High	2.60E+04	Schoen 1958
UO ₂ /UO ₃ area (493)	Low	2.00E+03	Schoen 1958
(incl. Pilot Plant)	Moderate	7.25E+03	Schoen 1958
	High	2.50E+04	Schoen 1958
UF ₄ area (491)	Low	2.00E+03	Schoen 1958
	Moderate	2.00E+04	Schoen 1958
	High	1.25E+05	Schoen 1958
	---	5.00E+03	Schoen 1958
Laboratory area	---	2.00E+03	Schoen 1958
Locker area	---	2.00E+03	Schoen 1958
Postdecontamination			
Building G-1: 1st floor	Low	0.00E+00	DOE 1984
	Moderate	1.30E+03	DOE 1984
	High	2.10E+04	DOE 1984
Building G-1: 2nd floor	Low	0.00E+00	DOE 1984
	Moderate	7.40E+03	DOE 1984
	High	3.00E+04	DOE 1984

Building	Dose potential	Alpha, dpm/100 cm ²	Principal reference ^a
Postops, predecontamination; Post-AEC decontamination			
Boiler house	Low	0.00E+00	DOE 1984
	Moderate	1.50E+02	DOE 1984
	High	3.00E+02	DOE 1984
Foundry	Low	0.00E+00	DOE 1984
	Moderate	1.30E+03	DOE 1984
	High	1.30E+04	DOE 1984
Garage	Low	0.00E+00	DOE 1984
	Moderate	1.50E+02	DOE 1984
	High	3.00E+02	DOE 1984
Warehouse, incl. lab, offices	Low	0.00E+00	DOE 1984
	Moderate	5.00E+02	DOE 1984
	High	1.00E+03	DOE 1984
P-1	Low	0.00E+00	DOE 1984
	Moderate	7.20E+03	DOE 1984
	High	2.50E+04	DOE 1984
K-1	Low	0.00E+00	DOE 1984
	Moderate	1.50E+03	DOE 1984
	High	1.00E+04	DOE 1984
M-1	Low	0.00E+00	DOE 1984
	Moderate	5.00E+00	DOE 1984
	High	1.00E+01	DOE 1984

a. FUSRAP (2001) was another source of information in some cases.

Table B-25. Annual inhalation, radon, and ingestion doses, D&D/postoperations period.^a

AEC decontamination (Building G-1)		1952-1959				1955-1959		
		UF ₄ area	Old UF ₆ area	Recovery/still area	New UF ₆ area	UO ₂ /UO ₃ areas	Lab area	Locker area
Inhalation (pCi/yr)	Min	1.33E+02	1.53E+02	1.53E+02	1.53E+02	2.43E+02	---	---
	Mode	1.33E+03	4.51E+02	1.03E+03	6.87E+02	8.83E+02	6.09E+02	1.53E+02
	Max	8.31E+03	5.04E+03	1.15E+04	1.99E+03	3.04E+03	---	---
Radon (WLM/yr)	Min	4.12E-03	4.10E-03	4.10E-03	4.10E-03	4.04E-03	---	---
	Mode	4.12E-02	1.21E-02	2.77E-02	1.85E-02	1.46E-02	1.01E-02	4.10E-03
	Max	2.57E-01	1.35E-01	3.08E-01	5.33E-02	5.05E-02	---	---
Ingestion (pCi/yr)	Min	2.79E+00	3.21E+00	3.21E+00	3.21E+00	5.10E+00	---	---
	Mode	2.79E+01	9.47E+00	2.16E+01	1.44E+01	1.85E+01	1.28E+01	3.21E+00
	Max	1.75E+02	1.06E+02	2.42E+02	4.18E+01	6.38E+01	---	---
Postdecontamination		1960-1997						
		Building G-1, 1st floor	Building G-1, 2nd floor					
Inhalation (pCi/yr)	Min	0.00E+00	0.00E+00					
	Mode	6.55E+00	3.73E+01					
	Max	1.06E+02	1.51E+02					
Radon (WLM/yr)	Min	0.00E+00	0.00E+00					
	Mode	2.97E-03	1.69E-02					
	Max	4.80E-02	6.86E-02					
Ingestion (pCi/yr)	Min	0.00E+00	0.00E+00					
	Mode	1.38E-01	7.83E-01					
	Max	2.23E+00	3.17E+00					

1st floor: UF₆/ UO₃ area (in general)
 2nd floor: UF₄/ UO₂ area (in general)

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Post-AEC operations, Predecontamination		1955-1989				1959-1989		
		Boiler house	Garage	Warehouse	Bldg M-1	Foundry	Bldg P-1	Bldg K-1
Inhalation (pCi/yr)	Min	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	Mode	2.46E+00	2.46E+00	8.21E+00	8.21E-02	2.13E+01	1.18E+02	2.46E+01
	Max	4.92E+00	4.92E+00	1.64E+01	1.64E-01	2.13E+02	4.10E+02	1.64E+02
Radon (WLM/yr)	Min	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	Mode	3.13E-04	3.13E-04	1.04E-03	1.04E-05	2.71E-03	1.50E-02	3.13E-03
	Max	6.27E-04	6.27E-04	2.09E-03	2.09E-05	2.71E-02	5.22E-02	2.09E-02
Ingestion (pCi/yr)	Min	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	Mode	5.17E-02	5.17E-02	1.72E-01	1.72E-03	4.47E-01	2.48E+00	5.17E-01
	Max	1.03E-01	1.03E-01	3.44E-01	3.44E-03	4.47E+00	8.61E+00	3.44E+00
Post-AEC decontamination		1990-1992				1990-1992		
		Boiler house	Garage	Warehouse	Bldg M-1	Foundry	Bldg P-1	Bldg K-1
Inhalation (pCi/yr)	Min	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	Mode	3.03E+01	3.03E+01	1.01E+02	1.01E+00	2.63E+02	1.46E+03	3.03E+02
	Max	6.05E+01	6.05E+01	2.02E+02	2.02E+00	2.63E+03	5.05E+03	2.02E+03
Radon (WLM/yr)	Min	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	Mode	3.03E-04	3.03E-04	1.01E-03	1.01E-05	2.62E-03	1.45E-02	3.03E-03
	Max	6.06E-04	6.06E-04	2.02E-03	2.02E-05	2.62E-02	5.05E-02	2.02E-02
Ingestion (pCi/yr)	Min	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	Mode	6.36E-01	6.36E-01	2.12E+00	2.12E-02	5.52E+00	3.07E+01	6.36E+00
	Max	1.27E+00	1.27E+00	4.24E+00	4.24E-02	5.52E+01	1.06E+02	4.24E+01

a. Data in the table above were calculated using RESRAD-BUILD from information in Schoen (1958), Klevin (1955a), DOE (1984), and FUSRAP (2001). The periods of application above are based on the AEC-overseen decontamination of Plant C (UF₄, UF₆, and related areas, 1952–1959; UO₂/UO₃ and related areas, 1955–1959); the subsequent years of use of Plant C (1960–1997) before FUSRAP agreed to recharacterize the radiological status of the plant; the years of post-AEC use of the other buildings, without significant decontamination (1955–1989); and the company-sponsored decontamination of some buildings, not under AEC/U.S. Energy Research and Development Administration oversight but done to NRC standards (1990-1992). See text below regarding assumptions made, etc.

The principal assumptions made for the RESRAD-BUILD calculations (Table B-25) are given below.

1. The inhalation and radon source terms were derived on the basis of the average maximum, high average, and typical surface contamination levels, respectively, in each plant area or building regardless of surface location. However, ceiling and overhead levels were not considered because the overheads were generally found to be far less contaminated than the walls and floor and were likely not to be frequently accessed; thus, they would contribute negligibly to the total dose. The three surface contamination levels used for each plant area or building are given as "maximum,0" "mode" (average of the high readings), and "minimum" (typical; i.e., the median for all the readings), with one case run for each. The room model was assumed to have the respective surface concentration over all wall and floor surfaces, which were taken to be concrete. These sources are listed in Table B-24.
2. Source contamination was measured as gross alpha and as either total beta and total gamma separately or total beta-gamma together, depending on the survey. Because the degree of equilibrium of the uranium with its daughters was not known, it was assumed for the inhalation case that there was 100% equilibrium; this appears to be reasonable because of the length of time between the cessation of operations and the beginning of intrusive decontamination operations. The source terms were then determined by assuming that the alphas were being emitted by ²³⁸U, ²³⁴U, ²³⁰Th, and ²²⁶Ra (together yielding 97.8% of the total alpha emissions and by ²³⁵U, ²³¹Pa, and ²²⁷Ac (together yielding 2.2% of the total alpha emissions).
3. The breathing rate was taken to be 1.2 m³/hr.

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4. The worker was assumed to spend his entire work time (8 hr/workday) in the modeled room (i.e., in the contaminated work area of the given plant area or building). The worker was assumed to spend 2,000 hr/yr in one location. The takedown of a building might have been on the order of weeks and decontamination on the order of months; however, continuous decontamination and demolition work over the course of a 50-week year (2,000 hours) was assumed for such workers.
5. The room size was taken to be about 10 by 20 by 10 ft high (3 by 6 by 3 m). There were many process areas that were larger, but they were often partitioned and they were undoubtedly decontaminated in sections. Thus, assuming a smaller room would be conservative in terms of concentrating or confining the contamination in the ventilated space and, therefore, exposure estimates would be expected to be favorable to claimants.
6. One air change per hour was assumed. While only limited information is available regarding the ventilation systems at Harshaw, it was clear that process areas had forced ventilation. These were perhaps not always used during significantly dusty work (e.g., checking ducting), but in those cases respirators were likely to have been worn by the workers. Thus, it is reasonable to assume that either the normal forced ventilation was used in the general area, in which case one air change per hour is a rate favorable to claimants, or vented enclosures were used, in which case the air change rate would have been far higher. The worker would likely have been wearing a respirator, and the calculated intakes would represent a marked overestimate of the likely actual intakes.
7. For the period that decontamination took place, the resuspension factor for the transferable contamination was assumed to be 1×10^{-4} ; for the postdecontamination years case, it was conservatively assumed to be 1×10^{-6} . The latter value is based on NRC (2002) and the former is taken to be a value favorable to claimants for nonrespirator work, as is consistent with the discussion and tables in the RESRAD-BUILD manual (ANL 2003).
8. The deposition (settling) velocity was taken to be 0.00075 m/s, a reasonable value for 5- μ m sized particles, as shown in Figure J.3 of ANL (2003).
9. The removable fraction for the decontamination years case was assumed to be 30%, based on the fact that some early decontamination was done at the end of operations (rinsing out process vessels, cleaning floors, etc.). The removable fraction for the postdecontamination period was assumed to be 10%. This should be reasonable because the postdecontamination period followed an extensive decontamination. Default erosion, radon emanation, and associated values were used because they are favorable to claimants.
10. The removable/erosion portion of the source was assumed to be removed completely over the modeled time, linearly. However, the fraction of radon assumed to be released to the air (as it was formed) was 1.0.
11. Because the inhalation and radon calculations did not depend on the position of the receptor in the room model and because isotope proportions were taken to be the same at the beginning of the calculation, one wall of the maximally contaminated plant was modeled with the D&D

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sources in RESRAD-BUILD; the same was done with the post-D&D sources. The results were then ratioed for the entire wall and floor area and for other area and building cases.

For use in dose reconstruction, the annual inhalation and radon doses that were the results of computations in RESRAD-BUILD had to be converted back to activity units, in this case to picocuries and WLM of intake, respectively. The RESRAD family of codes uses the dose conversion factors for inhalation given in Eckerman, Wolbarst, and Richardson (1988), as also listed in the RESRAD-BUILD manual (ANL 2003). The radon conversion is also from the RESRAD-BUILD manual (ANL 2003). Because the conversion factors are applied at the end of the RESRAD-BUILD calculation, it is appropriate to reverse the conversion using the factors. The converted results are listed in Table B-25.

RESRAD-BUILD has a limitation on how many yearly printouts can be made. So, inhalation and radon exposures were calculated for each year for the D&D case, but only for the first few years and every 5 years thereafter for the post-D&D case. This is appropriate because as the output data show, the values change little from year to year. So, although multiple years might be indicated in the column headings in Table B-25, the figures below them are for each year, not the sum for the indicated years.

Annual inhalation intakes derived from the RESRAD-BUILD inhalation dose results were used as a basis for calculating ingestion doses according to the methodology of NIOSH (2004), as follows.

$$\text{Ingestion intake (pCi/yr)} = 0.2 \times \text{concentration (pCi/m}^3\text{)} \times 250 \text{ d/yr}$$

$$\text{Ingestion intake (pCi/yr)} = 0.021 \times \text{Inhalation intake pCi/yr}$$

The RESRAD-BUILD main output gives output only in dose units, while supplementary output does not give total picocuries per cubic meter or disintegrations per minute per cubic meter. But the supplementary output (the .diag files) shows that because equilibrium is assumed between the uranium isotopes and their progeny down to radon and because linear resuspension is assumed, there is a constant conversion factor from inhalation dose to air concentration. Thus, doses reported in the main output can be converted back to concentrations and, thus, to ingestion doses. The results of applying this conversion to obtain the ingestion are listed in Table B-25.

The principal assumptions made for the Annual external dose rates and doses, D&D/postoperations period (Table B-26) are given below.

Direct radiation levels for gamma and beta were based on the maximum dose rates in each plant area or building, regardless of location. The maximum was the absolute maximum reading for an area or building, with one case for each level and type of radiation. The source terms are listed in Table B-26. Ceiling and overhead levels were not considered because of the lower activity levels in these areas and the infrequency of access. Without regard to the actual correspondence of gamma dose rate, beta dose rate, and surface contamination levels in particular rooms, the room model was assumed to contain the highest spot and average gamma and beta dose rates found anywhere in the respective area or building. Separate manual calculations were performed for gamma and beta radiation. Other assumptions and data manipulation details for the external dose calculations are given below.

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Table B-26. Annual external dose rates and doses, D&D/postoperations period.^a

Building or area	Gamma (mR/hr)			Beta (mrep/hr)			Gamma (mR/yr)			Beta (mrep/yr)		
	Min	Mode	Max	Min	Mode	Max	Min	Mode	Max	Min	Mode	Max
Old Hex (492) (excl Recov, Still)	0.020	0.034	0.09	0.9	1.6	7.6	40	69	180	1760	3150	15200
Old Hex (492) (Recov, Still only)	0.010	0.015	0.04	0.04	0.07	0.32	20	30	80	80	133	640
New Hex (annex) (492/493)	0.01	0.015	0.04	0.11	0.19	0.93	20	30	80	220	387	1860
Brown (493) (incl Pilot Plant)	0.012	0.023	0.06	2.3	4.2	20	23	46	120	4620	8310	40000
Green (491)	0.012	0.023	0.06	0.29	0.52	2.5	23	46	120	577	1040	5000
Plant C: 1st floor (UF ₆ /UO ₃ area)	0.075	0.15	0.39	0.35	0.62	3.0	150	294	771	693	1250	6000
Plant C: 2nd floor (UF ₄ /UF ₆ area)	0.035	0.069	0.18	0.16	0.29	1.4	70	137	360	323	582	2800
Boiler House	0.075	0.15	0.39	0.35	0.62	3.0	150	294	771	693	1250	6000
Foundry	0.08	0.15	0.39	0.35	0.62	3.0	150	294	771	693	1250	6000
Garage	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Warehouse: lab, offices	0.0013	0.0024	0.006	0.006	0.010	0.05	2.5	4.9	13	12	21	100
Building P-1	0.018	0.034	0.09	0.08	0.15	0.7	35	69	180	162	291	1400
Building K-1	0.04	0.08	0.21	0.17	0.31	1.5	80	157	411	346	624	3000
Building M-1	0.013	0.024	0.06	0.06	0.10	0.5	25	49	129	115	208	1000

a. Data in the table are based on Blatz (1951) (predecontamination 491-492 dose rates), Schoen (1958) (491-492-493 predecontamination beta dose rates for all but main hex), Klevin (1955a) (hex area predecontamination dose rates), and DOE (1984) (postdecontamination dose rates). The maximum measured dose rate(s) in each area was used to calculate the annual whole-body doses. Assumptions and details are shown below.

1. Because exposure (dose) rates were used, the source terms did not have to be translated into activity units. However, while some surveys reported gamma and beta dose rates separately, others gave reported combined beta-gamma dose rates. These were ratioed using information from the surveys to produce separate gamma and beta dose rates. The resulting assumed source terms are listed in Table B-26.
2. In the surveys, the measurement point for betas or mixed beta-gamma radiation was usually a contact or near-contact dose rate for both walls and floors, while for gammas, it was most often a contact or near-distance dose rate for walls and a 3-ft measurement for floors. It was thus assumed that the measured beta dose rate and the measured mixed beta-gamma dose rate represented all-beta radiation emanating from a wall surface to a receptor point at 1 cm from a wall surface; similarly, the measured gamma dose rate was assumed to represent gamma radiation emanating from a wall surface to a receptor point at 1 m from a wall surface.
3. A preliminary set of calculations was done to see what size of source (e.g., point, small radius, large radius) was most appropriate for the measured data for each type of radiation. For both beta and gamma, a large-radius source was found to be most appropriate and favorable to claimants. For the gamma case, it was assumed that the source was of infinite radius because that was not a very large increase from a 4-m radius source and the room could thus be assumed to be on the order of the room used for the RESRAD-BUILD calculation (i.e., about 3.3 x 6.7 m for the wall lengths). For the beta case, the source was assumed to be of essentially infinite radius (i.e., 8.5 m, the range of the most energetic beta emitted from the uranium-daughter source mix).
4. Dose rates per Item 2 above were used to determine the area source strength for beta and gamma separately and then these source strengths were used to calculate the dose rates at 1 ft and 1 m for beta and at 1 cm and 1 ft for gamma as needed. Per NIOSH direction, the respective dose rates at 1 cm were then considered to be the maximum dose rates, the dose rates at 30 cm (1 ft) the most likely dose rates, and the dose rates at 1 m (3 ft) the minimum dose rates.

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5. The receptor was assumed to stay at 1 ft from the source for 2 hr/workday and at 3 ft from the source for 6 hr/workday, for a total time of 2,000 hr/yr. This ignored break time, which was usually spent in areas of very low or no contamination.

The stay time assumptions in Item 5 should be adjusted for workers not likely to have spent considerable time in the areas of residual contamination, especially Plant C; the doses listed in Table B-26 should then be ratioed by an appropriate fraction to indicate a reasonable amount of time spent in the contaminated area. For example, a claimant who was a secretary in the postoperations years likely did not spend much time in the areas of significant contamination and should be assigned only a small fraction of the doses listed in Table B-26. A fraction of 5% is suggested for office workers, 10% for higher managers, and 25% for maintenance and safety workers [based on engineering judgment, given the information in the various dust studies (including time-and-place information), observations in AEC and Harshaw memos and reports, and statements in HCC (1946)].