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Summary of Extrusion Plant Site Information for Dose Reconstruction

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
04/26/2007	00	Approved new document to provide information to perform dose reconstructions for Extrusion Plant claims. Incorporates formal internal review comments. Added additional references and Table A-1. Modified to address formal NIOSH review comments and typographical errors. Adds a Glossary, and Attributions and Annotations section. There is no change to the assigned dose and no PER is required. Initiated by Robert Hysong.
03/07/2017	01	Revision initiated to incorporate a rewrite of Section 3.0 - Occupational Medical X-rays, based on revisions to ORAUT-OTIB-0079, Rev. 01, Guidance on Assigning Occupational X Ray Dose Under EEOICPA for X-Rays Administered Off Site, and ORAUT-OTIB-0006, Rev. 04, Dose Reconstruction from Occupational Medical X-Ray Procedures. Revised Section 4.0 - Unmonitored and Environmental Dose, based on 2016 internal ORAUT and NIOSH comments for unmonitored and environmental dose. Revised Section 5.0 – Internal Dose, for referencing the Fernald Internal Dosimetry TBD section for recycled uranium. Revised neutron discussion, and resolved NIOSH comments from 2008 and 2016 to incorporate the external dosimetry data evaluation in Section 6.0 - External Dose. Revised Attachment E – External Data Evaluation of Doses, to give the 95th and 50th percentile doses to be used for unmonitored workers. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Louise M. Buker.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AP	anterior-posterior
AWE	Atomic Weapons Employer
BZ	breathing zone
CEP	Controls for Environmental Pollution
CFR	Code of Federal Regulations
cm	centimeters
CTW	construction trade workers
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ES&H	Environment, Safety and Health
FEMP	Fernald Environmental Management Project
FMPC	Feed Material Production Center
g	gram
GM	geometric mean
HASL	Health and Safety Laboratory
HEPA	high-efficiency particulate air
hr	hour
ICRP	International Commission on Radiological Protection
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volts, 1,000 electron-volts
L	liter
lb	pound
LOD	limit of detection
m	meter
MDA	minimum detectable activity
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
min	minute
mL	milliliter
mrad	millirad
mrem	millirem
MT	metric ton
MTU	metric tons of uranium
NAD	nuclear accident dosimeter
nCi	nanocurie

NIOSH	National Institute for Occupational Safety and Health
NLO	National Lead Company of Ohio
NRC	U.S. Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
ODH	Ohio Department of Health
ORAU	Oak Ridge Associated Universities
PA	posterior-anterior
pCi	picocurie
PER	program evaluation report
PNL	Pacific Northwest Laboratory
POC	probability of causation
RCG	radioactivity concentration guide
RCRA	Resource Conservation and Recovery Act of 1976
RMI	Reactive Metals, Inc.
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
SRS	Savannah River Site
t	ton
TBD	technical basis document
TLD	thermoluminescent dosimeter
U.S.C.	United States Code
USTC	United States Testing Company
yr	year
µCi	microcurie
µg	microgram
°F	degree Fahrenheit
§	section or sections

1.0 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies). These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used to refer to an area, building, or group of buildings that served a specific purpose at a DOE or AWE facility. It does not mean nor should it be equated to an "AWE facility" or a "DOE facility." The terms AWE and DOE facility are defined in sections 7384l(5) and (12) of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA), respectively. An AWE facility means "a facility, owned by an atomic weapons employer, that is or was used to process or produce, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining or milling." 42 U.S.C. § 7384l(5). On the other hand, a DOE facility is defined as "any building, structure, or premise, including the grounds upon which such building, structure, or premise is located … in which operations are, or have been, conducted by, or on behalf of, the [DOE] (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program);" and with regard to which DOE has or had a proprietary interest, or "entered into a contract with an entity to provide management and operation, management and integration, environmental remediation services, construction, or maintenance services." 42 U.S.C. § 7384l(12). The Department of Energy (DOE) determines whether a site meets the statutory definition of an AWE facility and the Department of Labor (DOL) determines if a site is a DOE facility and, if it is, designates it as such.

Accordingly, a Part B claim for benefits must be based on an energy employee's eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility's designated time period and location (i.e., covered employee). After DOL determines that a claim meets the eligibility requirements under EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and the types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. Unlike the abovementioned statutory provisions on DOE facility definitions that contain specific descriptions or exclusions on facility designation, the statutory provision governing types of exposure to be included in dose reconstructions for DOE covered employees only requires that such exposures be incurred in the performance of duty. As such, NIOSH broadly construes radiation exposures incurred in the performance of duty to include all radiation exposures received as a condition of employment at covered DOE facilities in its dose reconstructions for covered employees. For covered employees at DOE facilities, individual dose reconstructions may also include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction.

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

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

1.1 PURPOSE

The purpose of this summary document is to provide consistency in dose reconstructions for the Extrusion Plant (also known by the primary contractor's name, Reactive Metals, Inc., or RMI) and to ensure that all components of dose are adequately addressed. This document provides information on the radiological processes and source terms and on the radiological controls and monitoring practices at the Extrusion Plant. While not meant to substitute for a complete site profile, this document represents the best understanding of the site at this time and provides assumptions for estimating doses when specific dose-related information is not available in individual records.

1.2 SCOPE

Section 2.0 of this document describes the Extrusion Plant and its history, including information about the radiological processes and source terms as well as radiological controls and monitoring practices. Section 3.0 describes occupational medical dose. Sections 4.0 describes occupational environmental dose. Sections 5.0 and 6.0 describe internal and external occupational dose.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 7.0.

2.0 SITE INFORMATION

The DOE Office of Worker Screening and Compensation Support defines the following information for the Extrusion Plant (DOE 2012):

Table 2-1. Site Information.

Site	Extrusion Plant
Alternative names	Reactive Metals, Inc. (RMI)
Location	Ashtabula, Ohio
Covered period	1962 to November 1, 2006
Facility type	Department of Energy

The Extrusion Plant received uranium from the Feed Material Production Center (FMPC) and the Weldon Spring Plant, as well as lesser quantities from other sites for extrusion, closed-die forging, or both (DOE 2000). Most of the uranium arrived in the form of billets, which were extruded into feedstocks for fabrication of fuel and target elements for use in nuclear production reactors. In addition, smaller quantities of thorium were processed at the plant.

The Extrusion Plant was the successor of the Bridgeport Brass facility in Adrian, Michigan. The work at Adrian was very similar to the work at the Extrusion Plant, and the same extrusion press was used at both facilities. The press, a 3,850-t Loewy horizontal extrusion press owned by the U.S. Atomic Energy Commission (AEC), was moved from Adrian to Ashtabula, Ohio, in November or December 1961 (Haywood et al. 1982; Jefferson 1961, 1962a) after the end of work in Adrian. Uranium extrusion at Ashtabula began in January 1962 (Koh 1997). The majority of material that was processed at the facility was for AEC and DOE, but nonradioactive metals such as copper, zirconium, titanium, and molybdenum were also extruded for commercial firms (ORAU 1985).

The Extrusion Plant conducted its radiological operations under Section 110, “Exclusions,” of the Atomic Energy Act of 1954 (Koh 1997, p. 47) and as a licensee of both AEC and the U.S. Nuclear Regulatory Commission (NRC) (DOE 2000). Section 110 of the Act as amended (42 U.S.C. § 2140) states:

> *Nothing in this chapter shall be deemed-*
>
> *a. to require a license for (1) the processing, fabricating, or refining of special nuclear material, or the separation of special nuclear material, or the separation of special nuclear material from other substances, under contract with and for the account of the Commission; or (2) the construction or operation of facilities under contract with and for the account of the Commission; or*
>
> *b. to require a license for the manufacture, production, or acquisition by the Department of Defense of any utilization facility authorized pursuant to section 2121 of this title, or for the use of such facility by the Department of Defense or a contractor thereof.*

Extrusion Plant operations for DOE and its predecessors were conducted under a prime contract from 1962 through August 1987, a subcontract under the DOE Fernald Environmental Management Program (FEMP) from September 1987 through November 1992, a prime contract under the DOE Oak Ridge Operations Office from December 1992 through March 1993, a prime contract under the DOE Chicago Operations Office from April 1993 through March 1995, and a prime contract under the DOE Ohio Field Office (DOE 2000). On December 22, 2003, the Ohio Department of Health (ODH) received notification that the DOE prime contract (DE-AC24-93-CH1055) was terminated (ODH 1999–

2004). DOE contracts with the Extrusion Plant included DE-AC05-760R 01405, administered by the Oak Ridge Operations Office (ORAU 1985) and DE-AC24-93CH10555 (DOE 2000).

The Extrusion Plant's NRC-regulated work was conducted under source material License Number SMB 602, first issued in June 1962, which initially allowed the use of 500,000 lb (230 MT) of uranium and thorium. This was changed to 400,000 lb (180 MT) of uranium and 100,000 lb (45 MT) of thorium on October 31, 1973 (NRC 1962–1999), and to 10,000 lb (4.5 MT) of natural uranium, 600,000 lb (270 MT) of depleted uranium, and 10,000 lb (4.5 MT) of thorium in 1979 (NRC 1979–1980). On June 7, 1985, the NRC license allowed possession, use, and storage of 5,000 kg (5 MT) natural uranium and 300,000 kg (300 MT) of depleted uranium; thorium was no longer listed.

On October 15, 1991, the NRC license amounts remained the same, but the condition of use was changed to possession incidental to site characterization and decommissioning plan preparation. On September 11, 1997, the condition of use was changed to "possession incident to decommissioning, remediation, restoration, and waste disposal." On May 26, 1999, the quantities were removed from the license and the materials were listed as natural uranium, depleted uranium, enriched uranium, and ^{99}Tc; the amount was described as "contaminated materials present at the site as of July 1, 1998" (NRC 1962–1999). The authority for licensing was later transferred to ODH on August 31, 1999, and ODH issued License Number 11900040004, which was similar to SMB-602 (DOE 2000; ODH 1999–2004). RMI also held NRC License 34-10618-01 (Van Loocke 1979), which was reportedly for a 1-mCi sealed ^{137}Cs source used to calibrate a gamma alarm system (RMI 1995). As of 1995, RMI possessed 171 sealed sources for use in instrument calibrations, including ^{60}Co, ^{90}Sr/Y, ^{99}Tc, ^{137}Cs, ^{210}Pb, ^{226}Ra, ^{230}Th, natural uranium, ^{239}Pu, ^{241}Am, and ^{242}Am. The quantities were not noted, but it was stated that some of the sources to be disposed would be characterized as Class C waste under 10 CFR. § 61.55 (RMI 1995). On March 6, 2004, ODH issued approval for RMI to use sealed sources of ^{55}Fe, ^{109}Cd, and ^{241}Am, in an X-ray fluorescence analyzer, ^{133}Ba in a liquid scintillation counter, and ^{63}Ni in a gas chromatography instrument.

DOE work at the Extrusion Plant included the extrusion of primarily depleted and enriched uranium for the Hanford N-Reactor and the Savannah River Site (SRS). License SMB-602 allowed extrusion of uranium and thorium (NRC 1962–1999); it was also RMI's authorization for use of source material for U.S. Department of Defense (DOD) armor-piercing penetrator work from 1974 through 1985 for DOD contractors (DOE 2000).

Westinghouse Materials Company of Ohio (WMCO) noted in its June 1989 review of the Extrusion Plant program that the most recent DOE extrusion campaign had been completed in September 1988 and that commercial extrusion work was occurring during the June 1989 review (WMCO 1989). Westinghouse also stated that no uranium was being extruded during the June 1989 review (WMCO 1989, p. 20). Other references (DOE 2000; RMI 1995) report that uranium extrusion for DOE ended in September 1988, and all other extrusion operations ceased on October 31, 1990. However, in Section 2.2 of this document, it is shown that uranium was received in 1989, 1990, and 1993, albeit in quantities much less than during the pre-1989 years. NRC License SMB-602 authorized uranium extrusion until the issuance of Amendment 4 on October 15, 1991.

This document refers to the period from January 1, 1962, through October 14, 1991, as the "radiological production period" (the time during which uranium was being extruded or during which uranium could have been extruded under the NRC license). The "postproduction period" is defined as October 15, 1991, to the present, during which no extrusion of radioactive metals occurred and predecommissioning and decommissioning activities were underway. Although there appears to be receipts for 0.05 MTU in 1993, there is currently no evidence that the material was processed.

2.1 PROCESS DESCRIPTION AND FACILITIES

The Extrusion Plant was in Ashtabula County, Ohio, slightly east of the city of Ashtabula. The facility consisted of 25 buildings occupying 7 acres of a 32-acre site. A floor plan of the buildings is shown in Figure 2-1. Table 2-2 lists buildings where uranium was processed or process equipment was stored. Attachment A contains detailed site figures by year that show changes to the site over time (RMI 1995).

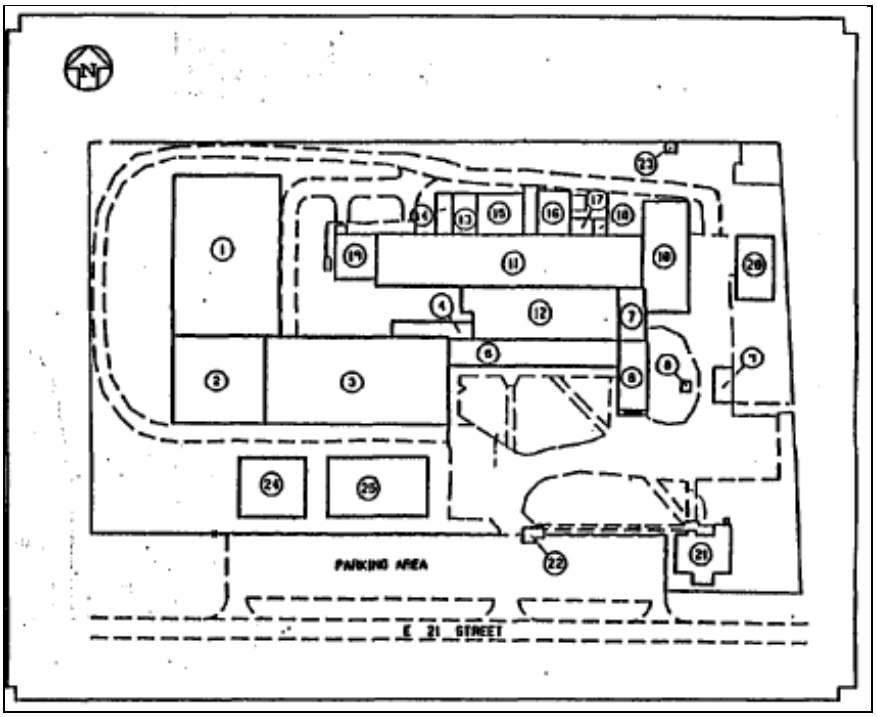

Figure 2-1. Extrusion Plant site map (DOE 1993, p. 13).

The normal operations of extrusion and forging at the Extrusion Plant can be generally described as a metals fabrication process. The plant's primary function was to change the shape or configuration of the received materials and then to ship these reconfigured metals to the receiving sites.

RMI's primary equipment for handling the uranium and thorium metals included the Loewy extrusion press, a runout table, a cooling table, and a cut-off saw. The process components also included a straightener, a degreaser, an oil bath, and a cleaning hood. Three gas-fired incinerators were in an auxiliary building (RF-3 Butler Building). Two of the incinerators were used to oxidize uranium sludges and residues while the third was used to incinerate contaminated combustible materials.

The process steps varied with different materials but generally consisted of heating the metal in a salt bath for 1.5 hours after extruding, quenching, a degreasing step, packaging, and weighing. A portion of the extruded uranium metal was pickled in a nitric acid solution for the purpose of cleaning the material.

The DOE work supported the N-Reactor at the Hanford Site and the reactors at SRS. The Extrusion Plant also supported armor-piercing penetrator programs for DOD contractors. The plant produced N-Reactor fuel and targets from 1962 to 1988. From 1962 until 1970, uranium processing for N-Reactor consisted of receiving uranium, primarily from the FMPC, extruding the metal, and returning the extrusions and scrap (again primarily to FMPC). In 1971, RMI began using a forge process for N-Reactor fuel, which involved further processing of the extrusions before sending them directly to Hanford (DOE 2000).

Table 2-2. Primary processing and storage buildings.

Building number	Building name[a]	Description/comments[b]
1	Northwest storage building	Storage of contaminated process equipment
2	RF-6 Butler Building addition	Acid neutralization tanks
3	RF-6 Butler Building	Storage of contaminated process equipment, lathes, drill press, small extrusion press, warehouse, laboratory, and offices; pickling, inspection, machining, and packaging of uranium
4	Enclosed ramp	None
5	Locker rooms/foreman's office	None
6	Enclosed truck ramp	None
7	Dock area	None
8	Emergency equipment storage building	None
9	RCRA storage building	None
10	Billet storage warehouse	Storage of incoming and outgoing uranium
11 and 12	Main plant high bay and Main plant low bay	Uranium processing, 3,850-t extrusion press, transfer table, pickling tanks, furnaces, abrasive saw
13	Runout table filter building	None
14	Saw filter building	None
15	Tool crib	Extrusion tooling storage
16	Die head filter building	None
17	Switchgear room	None
18	Compressor room	None
19	Wastewater treatment plant	None
20	RF-3 Butler Building	Uranium incinerator (oxidizer) and volume reduction equipment
21	ES&H Building	None
22	Guard house	None
23	Sewage disposal plant	None
24	Modular laboratory	None
25	Modular office	None

a. DOE (1993).
b. RMI (1995).

The detailed steps involved in this process are described in Attachment B. Production for SRS occurred from 1962 to 1988. The general steps consisted of receiving uranium, extrusion and sectioning, and then returning the extrusions and scrap. Some minor changes were made to the process over time. These changes, as well as a detailed description of production for SRS, are included in Attachment C. Work on armor-piercing penetrators for DOD contractors occurred from 1974 until 1985. The process was similar to that for SRS except for three process changes:

1. The extrusions were hydraulically sheared and air-cooled before water quenching.

2. The extrusions were not generally cut on the abrasive saw and were not run through the roll straightener.

3. The extrusions were pickled with nitric acid and then rinsed before shipment.

A detailed description of the process is included in Attachment D.

In addition to uranium, thorium metal (with no cladding) was extruded for DOE from 1962 through 1971 (RMI 1996, pp. 19–29). A small number of clad beryllium ingots for the New Production Reactor program were also extruded. This involved a process step described as hand rolling (Breslin and

Glauberman 1964, pp. 14–16). Because the ingots were identified as beryllium, they would not have been a source of radiation exposure.

2.2 PRODUCTION QUANTITIES

The quantities of uranium sent to the Extrusion Plant for the N-Reactor and SRS production are listed in Table 2-3.

Table 2-3. Uranium receipts for N-Reactor and SRS work (MTU).[a]

Uranium type	1962–1970	1971–1990
Enriched	13,442	11,829
Normal	4,904	330
Depleted	5,094	30,778
All	23,440	42,937

a. DOE (2000).

Production for the DOD penetrator program was reported as approximately 9,488 MTU of depleted uranium from 1974 to 1985 (DOE 2000). According to this inventory information, the total uranium sent to the plant was 75,757 MTU. This amount is in basic agreement with the 76,721.78 MTU of uranium compiled from annual receipts of recycled uranium at RMI as shown in Table 2-4 (DOE 2000).

According to the RMI Environmental, Safety and Health (ES&H) Director in 2006, site documentation indicates that the amount of uranium (and presumably thorium) the Extrusion Plant processed and handled was larger than the amount it shipped because some of the material was processed more than one time (ORAUT 2007a). The total amount of processed uranium, reported as 121,224 MT (Saito 1993; RMI 1995) and as 130,712 MT (Koh 1997), provides an upper amount of uranium the plant handled. The difference between the latter two numbers is an additional 9,488 MTU from DOD work (Koh 1997). Tables 2-5 and 2-6 indicate the enrichment of the uranium handled and received at RMI.

A search of Extrusion Plant records by RMI indicated that thorium was only processed at the plant between 1962 and 1971 (RMI 1995). The amounts of uranium the plant processed for DOE and the amounts of thorium for SRS, Hanford, Y-12, and Davison Chemical (also known as W. R. Grace) between 1962 and 1971 are listed by year in Table 2-7 (Britcher 1992). Consideration of the plant's total uranium process inventory would reduce the Table 2-7 ratios if the thorium inventories are complete.

2.3 SOURCE TERM

2.3.1 Processed Uranium and Thorium

Uranium was assumed to be extruded from January 1, 1962, through October 14, 1991, and thorium was extruded intermittently from May 1, 1962, through December 31, 1971. Depleted, normal, and enriched uranium were processed. Most of the uranium was probably recycled uranium. Uranium factors are listed in Table 2-8. As listed in Tables 2-4 and 2-5, the enrichment of uranium the Extrusion Plant handled was typically well below 2.1%. For dose reconstruction, the default assumption is that the uranium was 2% enriched.

Table 2-4. Annual recycled uranium receipts.[a]

Year	MTU
1962	526.85
1963	2,977.63
1964	4,518.48
1965	2,213.02
1966	2,694.82
1967	3,249.29
1968	2,875.46
1969	2,657.32
1970	1,731.69
1971	1,920.31
1972	1,898.09
1973	3,083.36
1974	2,226.30
1975	1,547.26
1976	2,076.36
1977	2,232.35
1978	2,314.72
1979	2,355.17
1980	3,175.80
1981	3,794.46
1982	5,873.91
1983	6,619.13
1984	4,832.58
1985	3,697.74
1986	4,322.51
1987	800.76
1988	496.52
1989	9.67
1990	0.17
1991	0
1992	0
1993	0.05
Total	**76,721.78**

a. The information comes from DOE (2000). It is not clear if this is only a listing of DOE-contract processed uranium or if it also includes NRC-licensed materials.

Table 2-5. Uranium-235 weight percentage of uranium handled (Koh 1997).

U-235 wt%	MTU	Percentage of total uranium processed
0.14 and 0.2	64,438	49.3%
0.71	25,178	19.3%
0.86	3,932	3%
0.95	28,115	21.5%
1.02	8,108	6.2%
2.1	941	0.72%
Total	**130,712**	Not applicable

Table 2-6. Annual receipts of enriched uranium.[a]

Year	MTU	U-235 MTU	Average % enrichment
1962	244.7	2.3	0.947
1963	1,027.3	9.7	0.947
1964	2,503.1	24.2	0.965
1965	1,931.1	18.8	0.972
1966	2,357.3	25.6	1.083
1967	3,102.2	35.3	1.139
1968	1,068.5	18.5	1.732
1969	761.4	7.6	0.995
1970	448.4	4.2	0.946
1971	343.4	3.7	1.064
1972	554.2	5.6	1.005
1973	398.5	3.8	0.962
1974	918.2	9.2	0.999
1975	663.1	6.5	0.986
1976	321.9	3.4	1.046
1977	297.7	2.9	0.980
1978	535.0	5.4	1.009
1979	426.8	4.1	0.970
1980	266.3	2.8	1.048
1981	607.6	6.0	0.982
1982	670.5	6.5	0.970
1983	1,372.3	14.1	1.026
1984	1,324.0	13.1	0.987
1985	1,242.4	12.2	0.982
1986	1,262.3	12.5	0.991
1987	421.6	4.4	1.053
1988	257.7	2.4	0.947
1993	0.002084	0.000025	1.199
Total	**25,327.5**	**264.8**	Not applicable

a. DOE (2000).

Table 2-7. Masses of DOE uranium and thorium processed and mass ratio of thorium to uranium.

Year	U (MT)[a]	Th (lb)[b]	Th (MT)	Th:U
1962	526.9	7,674	3.5	6.64E–03
1963	2,977.6	47,320	21.5	7.22E–03
1964[c]	4,518.5	4,200	1.9	4.20E–04
1965	2,213.0	0	0	0.00E+00
1966	2,694.8	1,170	0.5	1.86E–04
1967	3,249.3	883	0.4	1.23E–04
1968	2,875.5	0	0	0.00E+00
1969	2,657.3	3,200	1.5	5.64E–04
1970	1,731.7	1,300	0.6	3.46E–04
1971	1,920.3	1,925	0.9	4.69E–04

a. DOE (2000).
b. Britcher (1992).
c. An additional amount of thorium was processed for Hanford's Project A-801-38 (Britcher 1992), but the amount is currently unavailable. It consisted of only one extrusion.

Table 2-8. Uranium mixtures, specific activity, and isotopic fractions.

Uranium mixture and radionuclide[a]	Activity fraction	Mass fraction	Activity ratio to U-235	Mass ratio to U-235
Slightly enriched (2%), U-234[b]	0.7694	0.0002	28.76	0.01
Slightly enriched (2%), U-235[b]	0.0268	0.02	1	1
Slightly enriched (2%), U-238[b]	0.2038	0.9798	7.618	48.99
Natural, U-234[c]	0.4886	5.37E-05	21.4	0.00745
Natural, U-235[c]	0.0228	7.20E-03	1	1
Natural, U-238[c]	0.4886	9.93E-01	21.4	138
Depleted, U-234[d]	0.1546	1.00E-05	14.45	0.00502
Depleted, U-235[d]	0.0107	1.99E-03	1	1
Depleted, U-236[d]	0.0005	3.11E-06	0.0467	0.00156
Depleted, U-238[d]	0.8342	9.98E-01	78	501

a. Although listed as a recycled uranium component, U-236 represents <1% of the dose from exposure to uranium (ORAUT 2016a).
b. Useful factors: Overall activity of 2% enriched uranium is 1.616 pCi/µg and 80.8 pCi/µg U-235.
c. Useful factors: Overall activity of natural uranium is 0.683 pCi/µg and 94.9 pCi/µg U-235.
d. Useful factors: Overall activity of depleted uranium is 0.4021 pCi/µg and 202 pCi/µg U-235.

The Extrusion Plant was licensed by NRC to use thorium from mid-1962 to mid-1985 (NRC 1962–1999) and might also have processed thorium under an Atomic Energy Act exception (Koh 1997). A review of plant records (Britcher 1992) indicated that thorium was not processed by RMI after 1971. Table 2-7 shows thorium to uranium mass ratios of less than 1%, and the NRC (1962–1999) licensing documents indicate that, for most periods, authorized mass ratio of thorium to uranium (natural and depleted) was no more than 25% before 1979, and was less than 2% by 1979. These ratios would be lower if consideration was given to the mass of enriched uranium. The low ratios of thorium to uranium are also supported by later environmental sampling and characterization surveys that identified only uranium and technetium contamination at the site (RMI 1995, 1996, pp. 19–29). Thorium-232 (which is the most significant isotope by mass in natural thorium) contamination was not detected on site. Thorium-230, a long-lived uranium progeny, was found on site but was characterized as consistent with background levels (NRC 1962–1999).

2.3.2 Industrial X-Ray Sources

No Extrusion Plant site documentation indicates that industrial radiography sources were used at the Ashtabula facility. Furthermore, the 1965 RMI application for renewal of NRC License SMB-602 specifically stated that no provision for metallographic laboratory handling activities would be included in the application like those previously carried out at the Bridgeport Brass facilities in Seymour, Connecticut (Bean 1965a).

2.4 RADIOLOGICAL SAFETY PROGRAM

Information about the Extrusion Plant's early radiological safety program was described in licensing documentation. Other information sources included the *Health Protection Rules and Regulations* (RMI 1973); AEC Health and Safety Laboratory (HASL) reports; and health protection, nuclear safety, and environmental inspections and appraisals that were conducted periodically throughout the history of the Extrusion Plant. In 1985, an independent health, safety, and environmental review at RMI resulted in a recommendation that the entire area of industrial hygiene and health physics be upgraded starting with a thorough health protection program evaluation and fundamental assistance on establishing written procedures, sampling, documentation, and recordkeeping, followed by a quality assurance evaluation (Row 1985). Consequently, between July 1985 and 1988, Battelle, the Pacific Northwest Laboratory (PNL) contractor, conducted an in-depth review of the RMI radiation protection program that included air sample particle size, solubility studies, and recommendations for internal and external dose control, contamination control, training, respiratory protection, and

associated procedures (Munson 1985). RMI established an As Low as Reasonably Achievable Committee, which had its first organizational meeting on April 2, 1985. A Health Physics Appraisal in May 1987 indicated that a draft health physics procedure manual was completed on September 25, 1986, and was formally adopted in July 1989 (RMI 1989a) after DOE uranium extrusion had ceased.

After passing through the guardhouse, hourly production and maintenance workers entered and exited the main plant building through the locker room door and performed a complete clothing and shoe change (Jefferson 1962a). By 1973, caps were also required by for hourly and production personnel (AEC 1973). A storage bin containing clean coveralls and gloves was located in the locker room (Jefferson 1962a). Lockers were used to store uncontaminated street shoes and personal clothing; contaminated shoes were stored below benches adjacent to the clothing lockers and a drum for disposal of contaminated work coveralls and gloves was located just outside the door leading from the locker room into the plant production area (Jefferson 1962b). At the end of a work shift, workers sat down at a bench in front of their lockers and removed contaminated coveralls and shoes. They then put on their shower clogs and deposited their contaminated clothing in the disposal drum as they proceeded to the shower area. After showering, a worker would return to his locker, put on personal clothing, and exit the plant through the locker room door (Jefferson 1962b).

Salaried workers and visitors were allowed through the main plant office area entrance door, which originally required an immediate change into smocks (lab coats) and shoes, or the use of rubber shoe covers, for use in contaminated areas (Jefferson 1962a). The location of the rack containing reusable smocks was moved a short time later from the shoe change area just inside the main plant office area entrance door to just outside the door leading from the main plant office into the main production area next to a drum for contaminated smocks (Jefferson 1962b). Entrance to the plant production area from the office area required the use of reusable smocks and rubber shoe covers. The entire floor of the plant office area was treated as a radiologically controlled area. Spot check surveys of desktops and other spots above the floor were used as a means of controlling the spread of contamination to desks (Jefferson 1962b).

Contamination monitoring of personnel exiting the operating area was not performed in the early years of operation because it was believed that good housekeeping, use of protective clothing, and washing of hands before leaving would afford adequate protection against inhalation or ingestion of contamination (Jefferson 1962a). Daily surveys of the locker room, office area, and production area access points was practiced “rigidly,” and hourly production employees were encouraged to take showers at the end of their shifts (Jefferson 1962a). Site documentation implies that contamination monitoring was performed routinely in later years. The findings of an independent review of the RMI radiation protection program in July 1985 indicated that contamination control boundaries and frisking were lax but the nature of materials available to be spread along with existing controls were such that the accidental transport of significant quantities of material off site was “unlikely” (Munson 1985). Correspondence dated November 7, 1985, indicates that RMI had procured some “needed personnel friskers” based on a recommendation from Battelle at PNL and that the radiation worker training planned for January 1986 would correspond with the implementation of an improved contamination control program (Munson 1985). WMCO (1989) observed that contamination control and monitoring practices were being implemented in a June 1989 review of the RMI program. By 1991, personnel contamination monitoring when workers exited a contamination control zone was being performed with alarming beta-gamma frisker instrumentation. Protective clothing and personal clothing contamination limits were 15,000 and 3,700 dpm/100 cm^2 beta-gamma (or alpha), respectively (RMI 1990a), while the skin contamination limit was 3,700 dpm/100 cm^2 beta-gamma (RMI 1994).

Before the construction of a new Butler Building sometime in the first quarter of 1965, the plant changing facility (locker room) was also used as a lunchroom (Ruch 1965). The dining table in the locker room was used at lunch breaks and coffee breaks (three per shift), where plant personnel were required to wear blue smocks over contaminated coveralls to prevent the spread of contamination to

the table. In addition, the washing of hands was supposed to be "strictly enforced" at the time the blue smocks were used during breaks (Jefferson 1962a). The lunchroom and offices were relocated to the new Butler Building by March 1965 (Ruch 1965), but alpha contamination survey records from 1977 and 1978 indicate removable contamination levels up to 1,500 dpm/100 cm^2 in the hourly employee lunch room as well as elevated fixed contamination (Featsent 1977a). Similar results were recorded on 1975 and 1977 alpha contamination survey records from the hourly employee locker room (Featsent 1975, 1977b). Contamination surveys of the lunchroom and change room in September 1989 indicated total beta-gamma levels of 29,526 and 76,798 dpm/100 cm^2, respectively (RRA 1989).

Early site documentation indicates that there were designated eating and smoking areas, but in May 1985 an independent health, safety, and environmental review at RMI resulted in RMI instituting a no-smoking policy in the production and incineration areas of the plant because "employees were smoking throughout the plant" (Row 1985).

Written standard operating procedures that incorporated criticality controls were used for extrusion and handling of slightly enriched uranium (Puterbaugh and Van Loocke 1964), and nuclear accident dosimeters (NADs) had been placed inside access doors to plant areas shortly before the 1.95%-enriched uranium (clad with beryllium) New Production Reactor campaign in September 1964 (Bean 1964). A fourth NAD was installed in the warehouse area of the RF-6 Building in June 1965. By April 1965, a criticality alarm system had been installed (Bean 1965b).

Ventilation was the primary means used to limit radioactivity in the air at the Extrusion Plant, but respirators were provided to personnel for certain operations that caused excessive airborne contamination (Jefferson 1962c). HASL conducted dust and ventilation surveys at RMI in June 1962 and March 1964 that resulted in ventilation system modifications and procedural changes to reduce dust levels. Four ventilation systems with hoods were used for the early extrusion process. The Loewy extrusion press, die head and runout table, extrusion cooling table, and cut-off saw were each serviced by a separate ventilation system. There was also a small hood adjacent to the extrusion press used to clean die parts, which had its own fan and exhaust system. Another ventilation system serviced three gas-fired incinerators in an auxiliary building (Scrap Building). Two of the incinerators were used to oxidize uranium sludges and residues while the third was used for contaminated combustible materials (Breslin and Glauberman 1964, pp. 14–16). The gas-fired incinerators were equipped with a Type N Rotoclone dust collector (Breslin and Glauberman 1964, pp. 14–16; ORAU 1985). Additional processes that were served at one time or another by a ventilation or exhaust system included the resistance heater-roll straightener, vapor degreaser, acid pickle tanks, lathes, forge booths, hand filing and forge area, and the ingot acid etch booth and grinding booths in the warehouse portion of the RF-6 Building (Bean 1973). Site documentation indicates that during most of the production period of the Extrusion Plant, the primary source of stack emissions was the abrasive saw, followed by the scrap incinerator (Ruch 1964; Smith 1973; Hibbitts and Wing 1980; Wing 1982). The abrasive saw exhaust stack was not equipped with an emission control system until 1984 when a precipitator was installed (ORAU 1985).

Isokinetic sampling probes equipped with filter paper discs were periodically used to obtain representative airborne particulate samples from each plant stack for alpha counting. One stack sample a week was collected by the Safety Officer on a rotating basis so that each exhaust system was sampled at least every 7 weeks (AEC 1973). Continuous monitoring of all seven plant stacks, along with the emission control improvements for the stacks, serving the abrasive saw, forge area, and uranium scrap incinerators was recommended in 1985 (Row 1985). Installation and testing of a new abrasive saw ventilation system, which utilized high-efficiency particulate air (HEPA) filters and a baghouse filter system, were completed in August 1985, and RMI established a new policy of performing continuous stack sampling when the saw was in use. In addition, a ventilation and filter (non-HEPA) system was installed on the scrap incinerator sometime between June 1985 and June

1987, but isokinetic exhaust sampling was only performed when the ventilation equipment for the incinerator facility was operating (Davis 1987). WMCO (1989) noted in a June 1989 review of the RMI program that applications had been submitted for permission to install the extrusion press, runout table, cooling table, abrasive saw, scrap incinerator, forge area, pickling tanks, tool coating dip tank, and lathe emission sources, and that ventilation systems with HEPA filtering and stack emission monitoring were in place for each of the sources.

2.4.1 Area Surveys

According to the Extrusion Plant 1962 license application, floor contamination surveys in the locker room and plant office areas were to be performed daily and weekly floor surveys were to be performed in the production areas associated with the highest contamination levels (including the Scrap Building and warehouse portion of the RF-6 Building). Surveys of the surfaces of equipment above floor level were to be made based on the judgment of the Safety Officer or whenever equipment was shipped or removed from a contaminated area. The routine in-plant clean-up cycles were expected to keep the surface contamination limits below the levels listed in Table 2-9. Available portable instruments at the start of operations included alpha detectors and Geiger-Müller counters. Decontamination procedures generally involved the use of simple detergents, solvents, or steam cleaning. Broom sweeping was prohibited; dry vacuuming could be performed using an electrically operated industrial vacuum (Jefferson 1962b, 1962c). According to the independent health, safety, and environmental review in May 1985, there was no delineation of contaminated zones in areas of the plant where uranium was received, handled, processed, machined, and prepared for delivery (Row 1985). By June 1985, establishment of contamination control levels were consistent with DOE policy, but the total area of the facility was considered a contamination zone. By May 1987, progress had been made in reducing the area of contaminated zones to approximately two-thirds of the facility (Davis 1987).

Table 2-9. Recommended surface contamination limits in 1962.[a]

Location	Removable alpha (dpm/100 cm^2)	Fixed alpha (dpm/100 cm^2)	Removable beta-gamma (mrad/hr)	Fixed beta-gamma (mrad/hr)
Hot working areas	3,000	6,000	5	10
Other production plant areas	1,500	3,000	2	5
Locker room floor	250	500	0.2	2.5
Eating table	100	200	Not detectable	Not detectable
Plant office	500	1,000	0.3	2.5
Shipping-receiving floor	500	1,000	0.4	2.5
Truck beds (after use)	Not detectable	500	Not detectable	0.4
Equipment-in-place (accessible areas; above floor level)	10,000	20,000	5	10
Before removal (to other plant property sites; above floor level)	2,000	4,000	0.4	2.5

a. Jefferson (1962a, 1962b, 1962c).

2.4.2 Personnel Monitoring

Based on a review of the claim files and available site documentation, it appears that greater than 50% of the RMI employees were monitored for external and internal radiation. External monitoring for most years of RMI operations consisted of the use of a film or thermoluminescent dosimeter (TLD) badge worn on the chest to monitor radiation from photons and electrons. Neutron monitoring and extremity monitoring studies occurred in later years, and extremity monitoring was routine for some employees in later years. Determination of uranium levels in urine began in 1962 for workers believed to have the larger potential for internal exposures. Beginning in 1968, in vivo chest counts were also

being used to monitor internal exposure from uranium. Beginning in 1979, in vivo chest count results sometimes included monitoring for other radionuclides such as thorium and technetium.

2.4.3 Air Monitoring

Early site documentation (Jefferson 1962c) indicates that the Safety Officer periodically performed breathing zone (BZ) air sampling at each production worker's operating position such that each position was sampled routinely every 2 months for comparison to the maximum allowable concentration of 6×10^{-11} µCi/mL, but it is not clear that specific actions were taken if this level was exceeded. BZ sampling appears to have continued throughout the history of the plant. Later documentation (Brewer 1990) indicates that workers were required to wear respiratory protection and BZ monitoring samplers when air concentrations reached 2×10^{-11} µCi/mL. In addition, at least one general area high-volume air sample was collected weekly at different locations around the plant, as chosen by the Safety Officer, based on the activities being conducted (Jefferson 1962c). General area air sampling appears to have continued throughout the history of the plant. Between 1962 and 1965, HASL conducted stack, environmental, general area, cyclone, and BZ air sampling at the Extrusion Plant. The most comprehensive and complete data report available (Breslin and Glauberman 1964, pp. 14–16) has been used as a basis for comparison to airborne exposure at the Bridgeport Brass plant. The conclusion that can be reached from this comparison is that, in general, exposures at the Extrusion Plant were less than those at the Bridgeport Brass plant. In addition, two-stage air-sampling studies that were conducted between 1963 and 1964 at the Extrusion Plant indicated that 90% of the air contamination was in a nonrespirable particle size range (Ruch 1964). Later studies conducted from 1985 to 1986 indicated that over 90% of the aerosol in the Scrap Building was nonrespirable (Munson 1986a) while 66% of the aerosol in the extrusion press area was nonrespirable (Munson 1986b).

2.5 RADIOLOGICAL INCIDENTS

There are two notes written beside urine bioassay results that refer to fires in the Scrap Building in late June 1965 and in November 1965. No additional information on the 1965 fires has been found. A 1995 uranium fire incident involving a drum containing slightly enriched uranium oxide waste (the folder that contained the report was labeled "1% Oxide Issue") occurred inside a glovebox (probably in October) (DOE 1995). The initial report indicated a small pie-shaped area (one-sixth the area of the drum and a few inches deep) in the bottom of the drum was smoldering like burning embers (DOE 1995). A later review indicated that the smoldering had changed to flames inside the enclosure, that the ventilation remained on during the fire, and that at least one worker in the area was not wearing the required gloves (DOE 1995).

2.6 PHYSICAL EXAMINATIONS INCLUDING OCCUPATIONAL MEDICAL X-RAYS

Section 3.0 provides information about what is known about the occupational health program at RMI, including the X-rays taken for screening for occupational disease. The Extrusion Plant shared medical facilities with a nearby RMI plant (Hibbitts and Pryor 1970), and a dispensary was co-located with the main guardhouse. The medical program, described in 1970 and 1976 health and nuclear safety appraisals (Hibbitts and Pryor 1970; Johnson 1976), consisted of preemployment, annual, and termination physicals including blood tests, audiograms, immunizations, urinalyses, and chest X-ray examinations for everyone except female clerical employees. In addition, laborers had a preemployment lower back ("normal spine") X-ray examination (Johnson 1976) because they were more "prone" to back injuries (Hibbitts and Pryor 1970).

2.7 WORK PERIODS

For much of the operational period at RMI, there were at least two shifts: day and night. The 1964 HASL air concentration study (Breslin and Glauberman 1964, pp. 14–16) used 480 minutes for the total exposure time per day on the site, but assigned 40 minutes for lunch with a morning break and 10 minutes in the locker room, which would reduce the time engaged in work to 430 min/d. In general, bioassay and external dosimeters should be used to determine doses and, in those cases, the period of exposure is irrelevant. To account for exposures that were not monitored by bioassay or dosimeters, this summary assumes a default exposure of 2,000 hr/yr.

2.8 PERIOD AFTER RADIOLOGICAL PRODUCTION

This summary refers to January 1, 1962, through October 14, 1991, as the "radiological production" period, during which uranium was being extruded or might have been extruded based on uranium receipts. The "postproduction period" is that during which extrusion of radioactive metals ceased and predecommissioning and decommissioning activities were underway (from October 15, 1991, to December 2006). On October 15, 1991, License Amendment 4 converted NRC License SMB-602 from an operating license to a license only for possession incident to site characterization and decommissioning plan preparation. On May 14, 1993, License Amendment 5 was approved to designate a new Radiation Safety Officer and contact person for RMI. License Amendment 6 was approved on November 9, 1993, to enable predecommissioning activities that included surveys, equipment removal, waste handling, shipment, and disposal. On September 11, 1997, the decommissioning plan was approved and incorporated into the license as Amendment 8. Internal and external monitoring continued throughout the postproduction period until 2004 when internal monitoring practices changed. According to the current RMI ES&H Director, starting in January 2004, the only urine bioassay samples submitted by workers were preemployment, termination, and "for cause" whenever an intake was suspected (ORAUT 2007b). By January 2004, building decontamination was completed and work at the site consisted primarily of soil and ground-water remediation (ORAUT 2007b). In March 2004, decontamination and decommissioning activities at the site temporarily ceased until November 2005 when a remediation services contractor (LATA-SHARP Remediation Services) was hired to complete decontamination and decommissioning (ORAUT 2007b). By March 2004, less than 20 RMI management and compliance personnel remained at the Extrusion Plant to provide oversight of the remaining decontamination and decommissioning field activities, which were likely completed by December 2006 (ORAUT 2007b).

Uranium extrusion for DOE at RMI ceased in September 1988, and all other extrusion operations reportedly ceased on October 31, 1990 (DOE 2000), although the inventory receipt information compiled in Table 2-3 shows receipt of 0.1 MTU in 1993. This document refers to the radiological production period as January 1, 1962, through October 14, 1991, and the postproduction period as the period during which extrusion of radioactive metals is believed to have ceased (despite the later receipt of uranium in 1993, which appears to not have been extruded based on the license change in 1991).

3.0 OCCUPATIONAL MEDICAL X-RAYS

The medical program that is described in 1970 and 1976 health and nuclear safety appraisals consisted of preemployment, annual, and termination physicals including a chest X-ray for everyone except female clerical employees (Hibbitts and Pryor 1970; Johnson 1976). The female clerical employees did not appear to receive the annual physicals, only the preemployment and termination physicals (Hibbitts and Pryor 1970, p. 8). In addition, laborers received one preemployment lumbar spine X-ray (Hibbitts and Pryor 1970; Johnson 1976).

The 1970 appraisal specifically states that X-ray examinations were made off the site at Ashtabula General Hospital; claim data support this statement. The RMI ES&H Director stated in an interview in 2006 that RMI obtained its own X-ray equipment for chest X-ray examinations in later years, although the specific date of equipment acquisition is unknown (ORAUT 2007b). Based on information in the claim file records, it appears that X-rays were taken off site at Ashtabula General Hospital through 1980 [1]. According to the guidance in ORAUT-OTIB-0079, *Guidance on Assigning Occupational X Ray Dose Under EEOICPA for X-Rays Administered Off Site* (ORAUT 2016b), the dose from X-rays performed off site at a noncovered facility (such as the Ashtabula General Hospital) should not be included in dose reconstruction.

X-ray examinations were performed on site at the Extrusion Plant starting in 1981. The practice of not providing annual physicals to female clerical employees was an early one (Hibbitts and Pryor 1970, p. 8), and might not have been retained into the 1980s. The practice of taking preemployment lumbar spine X-rays was phased out sometime before 1986 and, beginning in 1997, chest X-rays were only taken at the physician's discretion (ORAUT 2007b). There is no evidence collected to date that establishes when the X-ray equipment might have been removed from the site, so it is assumed that X-rays were taken on site from 1981 to 2006, when the site was decommissioned. What is known about RMI's occupational X-ray screening frequency is summarized in Table 3-1.

Table 3-1. Frequency of occupational X-ray screening at the Extrusion Plant.

Period	Workers	X-ray projections	Frequency
Before 1981 (off site)	None	X-rays not included in dose reconstruction[a]	Not applicable.
1981–1986	All workers	PA chest	Preemployment, annual, and termination.
1981–1986	Laborers and trade workers (use job title)	AP and AP spot, lateral, and lateral spot lumbar spine[b]	Preemployment only.
1987–2006	All workers	PA chest	Per physician direction and as contained in the claim file records. Default to preemployment, annual, and termination if no records in the claim file.

a. ORAUT (2016b), Hibbitts and Pryor (1970), Johnson (1976).
b. ORAUT-OTIB-0006 (ORAUT 2011).

No information is currently available about the type of X-ray equipment used at the Extrusion Plant. Specific X-ray techniques for the examinations have not been found. Some X-ray technique factors have been collected from RMI records, but they are for examinations that would not have been performed in an industrial setting (such as pediatric and upper and lower gastrointestinal examinations) (Heinlein 1983), so it is believed that these might have actually come from another facility such as the Ashtabula General Hospital. Therefore, only doses for X-rays on site at RMI from 1981 through 2006 should be assigned in accordance with ORAUT-OTIB-0006, *Dose Reconstruction from Occupational Medical X-Ray Procedures* (ORAUT 2011).

4.0 UNMONITORED AND ENVIRONMENTAL DOSE

An extensive data evaluation of externally monitored workers was conducted and the results are given in Attachment E. In accordance with ORAUT-OTIB-0020, *Use of Coworker Dosimetry Data for External Dose Assignment*, unmonitored employees with low potential for exposure (those not working in hot areas) can be assigned the values in Table E-4 at the 50th percentile for gamma as a constant distribution from 1962 to 2004.

Also, a site characterization report (Koh 1997) concluded that, based on actual site conditions in 1992, the doses that were received by the workers and the public were below NRC and DOE limits and did not pose a significant health risk. The mean external worker dose was 0.0039 rem/yr, which is below the reasonable maximum dose of 0.006 rem/yr for a worker at the RMI site. The *Environmental Assessment of the Safety Evaluation* for the license amendment of 1995 indicated that most outside areas in the main plant area were at ambient background levels.

The study by Breslin and Glauberman (1964, pp. 14–16) specifically indicates that the internal environmental exposure for operating personnel such as draftsmen and secretaries (see Table 4-1) was less than 0.1 of the radioactivity concentration guide (RCG), which was <28 dpm/m^3. Although the report shows that the average air concentrations from uranium dust were less than 10% of the RCG, an intake rate based on 10% of the RCG is considered bounding and should be assigned to nonradiation workers. For the International Commission on Radiological Protection (ICRP) Publication 66 breathing rate of 1.2 m^3/hr for an 8-hour workday (ICRP 1994), this is 184 dpm per calendar day.

Table 4-1. Occupational exposure intakes for uranium dust.

Job description or location	Air concentration (dpm/m^3)	Intake rate (dpm/workday)	Intake rate (dpm/calendar day)
Secretary	8.5	81.6	55.9
Draftsman	11	105.6	72.3
Office	8.5	81.6	55.9
Locker room for lunch period	20	192	131.5
Locker room for clothes changes	18	172.8	118.4
10% of the RCG	28	268.8	184.1

5.0 INTERNAL DOSE

The source term at RMI consisted of depleted uranium, normal uranium, slightly enriched uranium, recycled uranium contaminants, and thorium. Uranium urinalyses were performed, but it appears that urinalysis for thorium was never performed at the Extrusion Plant. Some in vivo chest counts are available beginning in 1968. Uranium intakes should be assessed based on bioassay data. Thorium and recycled uranium contaminant intakes are derived from the uranium intakes. Because of the tendency of technetium to become airborne more readily than uranium (DOE 2004), chronic ^{99}Tc intakes might have higher than the intake that would be derived using Table 5-5 later in this section. Therefore, if in vivo monitoring records indicate ^{99}Tc lung count results greater than the in vivo counter minimum detectable activity (MDA) of 0.5 µCi, then in vivo monitoring records should be used to assess ^{99}Tc intakes.

5.1 URANIUM BIOASSAY DATA

Uranium urinalyses for workers who RMI judged to be exposed were performed quarterly at the Extrusion Plant beginning in 1962. By 1963, urine sampling was performed quarterly for production and maintenance workers and semiannually for salaried personnel. Special studies involving urine sampling were performed when new processes were started. The RMI site investigation level for uranium in urine was 50 µg/L until 1985 when it was changed to 15 µg/L. Work restrictions were implemented if urinalysis results exceeded 30 µg/L and continued until repeated analysis indicated levels less than 15 µg/L. By 1997, the investigation level was decreased to 1 µg/L (Henderson 1997).

From 1962 through 1964, bioassay consisted of uranium photofluorimetry urinalyses by HASL. From 1965 to the early 1970s, uranium urinalyses were by Tracerlab of Waltham, Massachusetts, also using photofluorimetry methods. For most of the 1960s, three spot urine samples (the first submitted on a Monday morning before work, the second on the following Friday morning before work, and the third on the following Monday morning before work) were obtained every 3 months from personnel judged to have significant exposure potential. Starting sometime in the mid-1970s until 1988, total uranium urinalysis was conducted by the United States Testing Company (USTC) of Richland, Washington, using a mass-based uranium measurement technique. Site documentation suggests that single urine voidings were obtained at an onsite restroom on two consecutive Mondays every 3 months until 1986 (Manninen 1986). Beginning in 1986, workers were provided sample containers to take home during the weekend instead of providing urine samples in the potentially contaminated restroom on return to work on Monday morning. Also beginning in 1986, workers provided a "simulated 12 hour [sic] sample" that consisted of all urine passed by the employee between 2 hours before bedtime and 0.5 hours after waking the next morning (Manninen 1986). From 1988 to 2006, uranium urinalysis was performed by several vendors including Controls for Environmental Pollution (CEP) of Santa Fe, New Mexico; Quanterra and Severn Trent Laboratories in Richland, Washington; and General Engineering Laboratories in Charleston, South Carolina. Based on available information in claim files, all urinalysis vendors used a mass-based uranium measurement technique, and results are presented in either milligrams or micrograms of uranium per liter. Table 5-1 lists the Extrusion Plant uranium urinalysis vendors and detection levels based on a review of claimant records and site documentation.

According to the 1990 RMI license renewal application, RMI employees were required to submit weekly, monthly, or quarterly urine samples depending on work assignments (Marsh 1990). According to the 1992 RMI urine bioassay program description, bargaining unit personnel were sampled monthly and salaried personnel were sampled quarterly (Gammon 1992). According to the RMI ES&H Director, starting in January 2004, the only urine bioassay samples submitted by workers were preemployment, termination, and "for cause" whenever an intake was suspected. By January 2004, work at the site consisted primarily of soil and ground-water remediation. Intakes for personnel who might not have been monitored after 2003 are presented in Section 5.5.

Table 5-1. Uranium urinalysis vendors and detection levels.

Analyst[a]	Year(s)	Detection threshold concentration (µg/L)	Reference
HASL	1962–1964	2.0	AEC 1964
Tracerlab and NLO	1965–1972	1.0	RMI 1967
USTC	Early 1973–1983	3.0	RMI 1974
USTC	1985	5.0	Manninen 1988
USTC	1986	0.1	Manninen 1988
USTC	1987	0.5	Manninen 1988
USTC, CEP, Quanterra, Severn Trent, and General Engineering Laboratories	1988–2003	0.5 or less	CEP 1991 and claim file records

a. CEP was the RMI vendor from 1989 through 1993. In April 1994, Sandia National Laboratory stopped using CEP's bioassay services because quality control testing had raised questions about the reliability of CEP urinalysis results (NRC 1994). CEP results are considered invalid for the purpose of this Project, and CEP analysis should not be used for any year.

5.2 IN VIVO MONITORING

Beginning in 1968, RMI began using the results of either in vivo lung counting or uranium urinalysis to determine internal exposures, although site documentation indicates a much greater reliance on lung counts for demonstrating regulatory compliance. Annual in vivo chest counts for total uranium and enriched uranium were performed by an unidentified vendor (probably the Y-12 Plant in vivo mobile counter) in 1968, 1969, and 1971 through 1985. From 1968 through 1978, lung count results are reported in micrograms of ^{235}U, milligrams of total uranium, grams of potassium, and nanocuries of ^{137}Cs on an "Invivio [sic] Radiation Monitoring Report" for individual workers at the plant. There is no evidence of occupational intakes of ^{137}Cs at the Extrusion Plant, so no dose of record should be associated with these measurement results in any year. Table 5-2 lists general information about the detection capabilities of in vivo lung counting at the Extrusion Plant for various periods. Table 5-3 lists codes with their interpretations. A review of available records indicates that from 1979 through 1985 some in vivo lung-counting records also contain results for thorium activity inferred from the ^{228}Ac and/or ^{212}Pb lung activity. However, the Y-12 mobile counter did not provide adequate detection sensitivity for either transuranic nuclides or thorium until 1986 (ORAUT 2016a), by which time it was no longer used at the Extrusion Plant. According to correspondence about the RMI radiation protection program review by PNL, evaluation of the previous whole-body counting data revealed "mostly problems and questions" (Munson 1985). The action level for recounts and work restrictions could not be determined.

Table 5-2. In vivo lung measurement types and detection levels for various periods.[a]

Period	Equipment	Radionuclide	MDA[b,c]
1968–1985	Y-12 mobile counter	Total uranium	4 mg
1968–1985	Y-12 mobile counter	Enriched uranium (2% U-235)	0.1 mg
1968–1985	Y-12 mobile counter	Depleted uranium	4 mg
1968–1985	Y-12 mobile counter	Np-237	200 pCi
1968–1985	Y-12 mobile counter	Tc-99	0.5 µCi
1968–1985	Y-12 mobile counter	Th-232	6 mg
1986–1995	Helgeson counter	Total uranium	2–4 mg
1986–1995	Helgeson counter	Enriched uranium	0.04–0.07 mg

a. Adapted from ORAUT (2012).
b. The Tc-99 MDA is based on McDougal (1980).
c. The Th-232 MDA is based on Scott et al. (1969) and ORAUT (2012).

Beginning in 1986, Helgeson Scientific was contracted to perform annual in vivo lung counts for natural uranium and ^{235}U because of their greatly improved sensitivity (2.5 nCi) in comparison with the Y-12 mobile counter (Manninen 1988). March 1990 site correspondence indicates that the lung count

results from Helgeson indicated system bias and that a background level would need to be established to test the data for significance at a 95% confidence level (Aldridge 1990). Correspondence containing corrected lung count data (dated October 14, 1997) from Helgeson Scientific to RMI indicated that Helgeson reports from 1986 to September 1996 have headings that were incorrectly labeled milligrams of uranium rather than nanocuries of uranium, and that the percent of the annual limit on intake calculation was low by a factor of 1.4 (Helgeson 1997). A review of several claim files indicates that these corrections were made to records in the available claim files. From a review of claim files, it appears that annual lung counting was discontinued around 2002 when Helgeson ceased operation.

Table 5-3. In vivo record codes.

Form identification	Measurement type	Code	Interpretation
In Vivo Radiation Monitoring Report	Lung	F/B Ratio	This is a measure of how close to the front or back the internal contamination is. A ratio of greater than 1 could indicate external contamination.
In Vivo Radiation Monitoring Report	Lung	A. Enriched Uranium	The maximum U-235 enrichment at the Extrusion Plant was 2.1%.
In Vivo Radiation Monitoring Report	Lung	J. NLO Uranium	Refers to the special spectrum region of interest for NLO, early operator of the FMPC.
Helgeson	Lung	n-u handwritten on forms with a spectral print out	Natural uranium result usually with corresponding handwritten result in nanocuries with 2-sigma uncertainty.
Helgeson	Lung	U-235 handwritten on forms with a spectral print out	U-235 (enriched uranium) result usually with corresponding handwritten result in micrograms with the 2-sigma uncertainty.

5.3 INTAKE ASSUMPTIONS

5.3.1 Uranium

Uranium intakes are assumed to be type M or S because the processes at the Extrusion Plant were similar to those at the Adrian Plant and receipt of material in metal or billet form. The predominant form of uranium that was processed at the site is uranium metal, and no uranium hexafluoride or uranium nitrate materials (type F) were received. Further, the in vitro dissolution studies of compounds at uranium facilities have shown that oxides of uranium can exhibit moderate solubility, which indicates absorption type M based on Eidson (1994).

For calculating annual organ doses, the uranium intake (in disintegrations per minute) can be assumed to be entirely ^{234}U. Because most of the uranium came from Fernald, the ratio of the recycled uranium contaminants to uranium is based on Section 5.5.1.4 of ORAUT-TKBS-0017-5, *Feed Materials Production Center – Occupational Internal Dose* (ORAUT 2016a). Recycled uranium intakes should be assigned by applying the contaminants and ratios from the table containing contaminant intakes per unit activity of uranium. The selected material types for the contaminants should be assigned using the direction in ORAUT-OTIB-0060, *Internal Dose Reconstruction* (ORAUT 2014b).

5.3.2 Thorium

In vitro analyses to estimate thorium exposures at the Extrusion Plant are unavailable, and thorium activity determined from chest counts was not reported until 1979, after the reported thorium processing period. Table 2-7 indicates that the thorium received by RMI in any year was less than 1% of the uranium received by mass. Because there are no specific process and monitoring records for

thorium use, this document assigns a U:Th mass intake ratio as 0.01 during the thorium production period, which is assumed to be from January 1, 1962, through December 31, 1971. See Table 2-7.

The specific activity of normal uranium is a factor of 1.7 larger than that of depleted uranium, but on the order of a factor of 2.4 smaller than that of 2%-enriched uranium. As a consequence, it appears to be favorable to claimants to assume natural uranium when determining the relative activity of thorium based on the relative mass comparisons. To determine the relative activities of uranium to thorium, the specific activity of ^{232}Th is divided by the specific activity of natural uranium and multiplied by a ^{232}Th:U mass intake ratio of 1%. This results in a relative ^{232}Th:U activity intake ratio of 0.00161.

It is assumed that the thorium is natural because this the dose from this form is more favorable to claimants. Therefore, it is assumed that the 232Th, 228Ra, 228Ac, 228Th, 224Ra are in secular equilibrium and an equivalent intake should be assigned for each. This 1% factor overestimates 232Th:U mass ratio for the production years and could be an order of magnitude higher for some years (ORAUT 2016c). Therefore, exposures from thoron are accounted for with the application of this 1% factor for thorium.

Thorium was not processed in 1965, 1968, or after 1971 (Britcher 1992). Site documentation indicates that RMI cleaned up after thorium extrusions to prevent commingling of thorium and recyclable uranium (Jefferson 1962a), but it is uncertain if the clean-up included all areas of the plant. Even though the thorium was cleaned up before uranium extrusions, it is favorable to the claimant to assume that residual contamination resulted in thorium intakes in 1965 and 1968 when no thorium processing was recorded. No thorium exposure should be assumed for periods after 1971.

5.4 INTERNAL DOSE ASSIGNMENT SUMMARY

Depleted, natural, and slightly enriched uranium were all source terms at the Extrusion Plant and are assumed to be types M or S. For results in mass units, it is favorable to claimants to assume exposures to slightly enriched uranium (2.0% ^{235}U by mass) if no better information is available. If uranium urinalysis measurements were made in both mass and activity units, the activity units should be used to calculate intakes and ^{234}U can be assumed for uranium dose calculations.

Uranium urinalysis, when available, should be used to estimate uranium intakes; the default detection thresholds are 3 µg/L for samples analyzed between 1962 and 1984, 5 µg/L in 1985, and 0.5 µg/L from 1986 to 2003. It should be noted that the forms used to record uranium urinalysis results at the Extrusion Plant had a column labeled “Mg/L” (which meant milligrams per liter rather than megagrams per liter); nevertheless, the results were sometimes recorded in micrograms per liter.

It should be noted that bioassay data from 1989 to 1993 analyzed by CEP are not to be used in any internal dose assessment. For individuals with potential intakes during that period, the following priorities should be used for assessing uranium intake rates:

1. Urinalysis data after 1993. These can be used to bound the earlier years.

2. Body counts during 1989 to 1993.

3. Urinalysis data before 1989 (extension of the dose calculation beyond the last bioassay result).

4. Data from an Extrusion Plant coworker with the same job (following the same priorities as above).

5. Unmonitored uranium intakes from a similar site (Bridgeport Brass, Adrian facility).

A ^{232}Th:U mass intake ratio of 1% is assumed. This results in a relative ^{232}Th-to-uranium intake activity ratio of 0.00161. Based on the assumption that the thorium is natural, the ^{232}Th, ^{228}Ra, ^{228}Ac, ^{228}Th, ^{224}Ra are in secular equilibrium and an equivalent intake should be assigned for each.

Thorium was not processed in 1965, 1968, and after 1971 (Britcher 1992); site documentation indicates that RMI cleaned up after thorium extrusions to prevent commingling of thorium and recyclable uranium (Jefferson 1962a), but it is uncertain whether the clean-up included all areas of the plant. Even though the thorium was cleaned up before uranium extrusions, it appears favorable to the claimant to assume that residual contamination resulted in thorium intakes in 1965 and 1968 when no thorium processing was recorded. No thorium exposure should be assumed after periods after 1971.

5.5 UNMONITORED INTERNAL DOSE

Breslin and Glauberman (1964, pp. 14–16) reported the daily average exposures of operating personnel. Air samples were obtained at locations selected to characterize workers' exposures, and both breathing zone and general air samples were collected. Table 5-4 lists the various job categories that were assessed. The intake rates are based on the daily average exposures (dpm/m^3) for operating personnel (assuming a breathing rate of 9.6 m^3/d for an 8-hour day).

Table 5-4. Occupational exposure intakes from uranium dust.[a]

Job description	Air concentration (dpm/m^3)	Intake rate (dpm/workday)	Intake rate (dpm/calendar day)
Heater	40	384	263.0
Heater Helper	40	384	263.0
Press Operator	63	604.8	414.2
Die Head Man	83	796.8	545.8
Extrusion Puller	150	1,440	986.3
Saw Man	280	2,688	1,841.1
Stamper	410	3,936	2,695.9
Inspector (Rod)	1,200	11,520	7,890.4
Oil Bath and Roll Straightener	96	921.6	631.2
Billet Inspector	46	441.6	302.5
Packer	54	518.4	355.1
Jeep Driver	60	576	394.5
Tool Crib	83	796.8	545.8
Scale Clerk	29	278.4	190.7
Scrap Handler	81	777.6	532.6
Maintenance	110	1,056	723.3
Engineering	33	316.8	217.0
Foreman	100	960	657.5
Head Foreman	60	576	394.5
Health and Safety	48	460.8	315.6

a. (ORAUT 2016d).

5.6 INTERNAL DOSE DURING THE POSTPRODUCTION PERIOD

During the postproduction period at the Extrusion Plant (after October 14, 1991), uranium extrusion activities, along with the handling of bulk uranium metal for extrusion, had ceased. Worker exposures and intakes occurred from residual uranium contamination on structures and components and in soil during characterization surveys and sampling, equipment dismantlement and removal, decontamination, demolition, and waste packaging and shipping activities. Section 2.8 contains a summary of worker monitoring changes that occurred during the postproduction period at the Extrusion Plant (after October 14, 1991), but based on a review of available site documentation such as personnel dosimetry files and claim files, it is not evident that site personnel from the remediation

contractor (LATA-SHARP Remediation Services), or even RMI, participated in an internal or external monitoring program after 2003. As stated in Section 2.8, by January 2004 building decontamination was complete and work at the site consisted primarily of soil and groundwater remediation.

To calculate the intakes from soil to unmonitored site personnel during soil remediation activities that began in 2004, this analysis assumed that uranium exposure was to the average soil concentration from locations within the main plant area and the property to the east of the main plant at a level of 350 pCi/g total uranium and to the maximum ^{99}Tc concentration of 49 pCi/g within the main plant area (RMI 1996, pp. 19–29). Using a mass loading factor of 2×10^{-4} g/m^3 and a breathing rate of 2,400 m^3/yr, the calculated annual uranium inhalation intake was 168 pCi and the annual ^{99}Tc intake was 24 pCi. The daily inhalation intakes for total uranium and ^{99}Tc are 0.460 pCi and 0.065 pCi. It is assumed that other recycled uranium contaminants would be a small fraction (see Table 5-4). The mass loading factor of 2×10^{-4} g/m^3 takes into account short periods of high mass loading and sustained periods of normal activity on a typical farm (Yu et al. 2001).

According to NIOSH (2004), the daily ingestion rate in picocuries can be estimated by multiplying the daily air concentration in picocuries per cubic meter by a factor of 0.2, which results in the daily ingestion of 0.014 pCi of uranium and 1.96×10^{-3} pCi ^{99}Tc.

An estimate of the uncertainty that is associated with soil resuspension intakes has been made by assuming (1) that the soil concentrations are lognormally distributed, (2) that the average uranium soil concentration (350 pCi/g) can be used to underestimate the 50th-percentile concentration, and (3) that the maximum uranium soil concentration ever measured at the plant (2,600 pCi/g) represents the upper 95th percentile (RMI 1996). The resultant geometric standard deviation is 3.4.

6.0 EXTERNAL DOSE

RMI employees were exposed to radiation from uranium, thorium, and their short-lived progeny. This document assumes that photon energies were in the 30- to 250-keV range, which is favorable to claimants when considering both organ dose conversion factors and radiation effectiveness factors. Shallow or open-window dose is assumed to be from electrons with energies greater than 15 keV.

From 1962 to the mid-1980s, claim files contain a handwritten Film Badge Assignment Sheet containing a name, badge number (typically between 1 and 80 assigned by RMI), and notes or comments such as lost film, damaged film, inverted film, quit, terminated, retired, vacation, wore two badges in monitoring period, and issue date (if not on the first day of the month or quarter). The next page(s) contain the dosimeter vendor report with the badge number corresponding to the Film Badge Assignment Sheet handwritten next to the appropriate results. Beginning in 1986, RMI personnel names are specifically provided in the vendor reports, and there either is no Film Badge Assignment Sheet or there is a database printout (1987 through 1996) that contains personnel names and external dosimetry data. Some claim files contain a Terminated Employee Radiation Exposure Report containing skin and whole-body dose for the employment period. In addition, some claim files contain Form AEC-5 (beginning in the early 1960s), later Form NRC 5. All available external dosimetry data, was evaluated for and stripped of duplicates, and is summarized in several tables in Attachment E. An evaluation of the almost 10,000 lines of data at the 95th and 50th percentiles was also given for gamma and beta doses.

6.1 PENETRATING DOSES

6.1.1 Gamma Dose

Table 6-1 lists available information on badges used from 1962 to 2004; whole-body external dosimetry results appear to be present in claimant files. There are also some extremity badge results beginning in 1974.

Routine film badge results included whole-body beta and gamma monitoring results. Table 6-1 provides some of the available information on the monitoring methods, detection limits, and reporting.

6.1.2 Neutron Dose

Neutron measurements began with the use of the five-element TLD in January of 1986 (badges were received in late December of 1985). Neutron doses appear to have been monitored for most RMI employees in 1986, 1987, and through the third quarter of 1988. No basis was given for starting or ending the neutron measurements. There were no requests for area neutron surveys and none were conducted. A Pacific Northwest National Laboratory audit of the RMI health physics program (Munson 1985) did not suggest or indicate there was a concern about neutron exposure. It appears the monitoring of neutron dose was initiated because of the ability of the new five-element TLD.

The neutron measurements appear to be invalid, or the results are spurious due to cross-contamination of the TLD chips in the dosimetry badge by other contaminated badges. See *Summary for Extrusion Plant (RMI) Research, High 1986 Results* (ORAUT 2015), which describes the research that was conducted and which concluded that the neutron results were not valid and likely the result of improper handling of the badges, which caused them to be contaminated. In 1987, contamination control practices were used to place the badges in separate thin plastic bags. The photon dose results then dropped back to the 1985 levels. Another reason for not considering the neutron measurements valid is the lack of the sensitivity algorithm that was not added until 1988. The lack of this added sensitivity feature might have allowed for erroneous neutron and gamma results to go unchecked, possibly contributing to the neutron dose signal that was measured by the algorithm.

Table 6-1. External dosimetry

Period	External dosimetry vendor	Exchange frequency/ monitoring method	Detection threshold reported in records
1962–1966	Controls for Radiation, Cambridge, Massachusetts	Monthly/film	A zero indicates less than minimum detectable dose of 5 mrem for X-ray and gamma <175 keV or 10 mrem for >175 keV X-ray and gamma and beta.[a]
1967–1987	USTC, Richland, Washington	Quarterly/film until TLDs in 1986	1972 and 1976 film badge reports indicate film detection limits of 10 mrem for X-ray, 20 mrem for gamma ray, and 20 mrem for beta.[b] According to a 1977 film badge report calibrations for gamma performed with Cs-137, beta performed with Sr-90, and X-ray performed with 16- to 17-keV photons. 1983, 1984,[c] and 1985[d] film badge reports indicates film detection limits of 10 mrem for photons and 20 mrem for beta. 1986 and 1987[e] TLD badge reports indicate detection limits of 10 mrem for X-ray, gamma, beta, and neutron radiation.
1988–1991	USTC and WMCO at FEMP[f]	Monthly/TLD	1988 and 1989 TLD badge reports from USTC indicate detection limits of 10 mrem for X-ray, gamma ray, and beta radiation. The minimum detection limit for FEMP TLDs was 5 mrem gamma and 10 mrem beta from 1989 to 1992.[g,h]
1992–2004	WMCO at FEMP, TMA Eberline, ThermoNutech, Landauer, ICN, and Global Dosimetry Solutions	Probably quarterly/TLD	TLD badge reports typically indicate detection limits of 10 mrem for X-ray, gamma ray, and beta, but information was not found for every vendor's badge type. Site documentation implies that TLD vendors were DOE Laboratory Accreditation Program or National Voluntary Laboratory Accreditation Program certified from 1989 to 1992.

a. CR (1964)
b. USTC (1974–1982).
c. USTC (1984).
d. USTC (1985).
e. USTC (1986), Mason (1988).
f. Mason (1988).
g. ORAUT (2014a)
h. ORAUT (2013)

Further, a review of the sources and source term at the Extrusion Plant did not indicate there were any sealed sources for neutrons or neutron generating equipment (ORAUT 2015). No records have been found that indicate the source of neutron exposure. In addition, Bridgeport Brass the predecessor to the Extrusion Plant conducted a coworker study and considered the neutron exposures minimal or negligible. Neutron exposures at the Extrusion Plant are also assumed to be minimal or spurious for those years when neutrons were monitored. The neutron doses were added into the photon or deep penetrating doses, these practices are considered favorable to claimants (ORAUT 2013; 2016e). Therefore, no additional neutron doses should be assigned based on the neutrons being included into the dosimetry reports for penetrating dose.

6.1.3 Historical Annual Exposures for Penetrating Radiation

Table 6-2 summarizes RMI-reported external penetrating doses for 1962 through 1999 and provides a general sense of the program size, although the number of persons monitored appears to include visitors and temporary workers beginning in the 1980s. The doses in Table 6-2 do not include nonpenetrating and extremity doses. The term "no data" means there were no reported doses for that period. For 1962 through 1973, the bounding dose for unmonitored workers is assumed to be 1 R/yr based on Table 6-2.

Table 6-2. Penetrating whole-body external doses, 1962 to 1999.

Year	Number not monitored	Number monitored	Dose (rem) 0–1	Dose (rem) 1–2	Reference
1962	7	52	51	1	Sapirie 1963
1963	9	71	70	1	Sapirie 1964
1964	14	81	81	No data	Sapirie 1965
1965	8	66	66	No data	Sapirie 1966
1966	27	57	57	No data	Sapirie 1967
1967	28	65	65	No data	Lenhard 1968
1968	31	62	62	No data	Hibbitts 1969
1969	33	41	41	No data	Smith 1970
1970	25	47	47	No data	Lenhard 1971
1971	24	46	46	No data	Lenhard 1972
1972	20	46	46	No data	Travis 1973
1973	27	53	53	No data	Travis 1974
1974	Unknown	69	69	No data	Van Loocke 1975
1975	Unknown	62	62	No data	Heiser 1976
1976	Unknown	58	58	No data	Schaeffer 1977
1977	Unknown	62	62	No data	Schaeffer 1978
1978	Unknown	68	68	No data	Schaeffer 1979
1979	Unknown	80	80	No data	Van Loocke 1980
1980	Unknown	80	80	No data	Schaeffer 1981
1981	Unknown	89	89	No data	Van Loocke 1982
1982	Unknown	116	116	No data	Schaeffer 1983
1983	Unknown	118	118	No data	Schaeffer 1984
1984	Unknown	117	117	No data	Schaeffer 1985
1985	Unknown	124	124	No data	Manninen 1988
1986	Unknown	134	130	4	Manninen 1988
1987	Unknown	822	822	No data	Brewer 1988
1988	Unknown	920	920	No data	RMI 1989b
1989	Unknown	No data	No data	No data	Not applicable
1990	Unknown	282	282	No data	Rizzi 1991
1991	Unknown	No data	No data	No data	Not applicable
1992	Unknown	No data	No data	No data	Not applicable
1993	Unknown	No data	No data	No data	Not applicable
1994	Unknown	332	332	No data	TMA 1995
1995	Unknown	255	255	No data	TMA 1996
1996	Unknown	285	285	No data	TN 1997
1997	Unknown	290	290	No data	TN 1998
1998	Unknown	No data	No data	No data	Not applicable
1999	Unknown	1,202	1,202	No data	Eberline 2000

Dosimeters consisted of film badges for beta, X-ray, and gamma radiation from 1962 through 1985. TLDs replaced film in 1986. In the early period of Extrusion Plant operations, a strip of indium metal foil was placed in badges for criticality monitoring. In 1986, 1987, and 1988, neutron monitoring was

performed in addition to beta, X-ray, and gamma monitoring. Dosimetry badges were exchanged monthly from 1962 through 1966 and quarterly or monthly thereafter.

Film badge detection thresholds for beta and gamma are based on information provided by film badge vendors in Table 6-1 and ORAUT-OTIB-0010, *A Standard Complex-Wide Correction Factor for Overestimating External Doses Measured with Film Badge Dosimeters* (ORAUT 2006a). The 0.010 rem given by the film badge vendor from 1962 to 1966 is not in line with guidance from OTIB-0010 and therefore, to be consistent with the values given later, the limit of detection for the earliest period of operation is 0.030 rem and from 1967 to 1985 was estimated to be 0.020 rem per exchange period. The beta-gamma TLD from Fernald has a detection threshold of 10 mrad beta and 5 mrad gamma (ORAUT 2014a). Therefore, the value was set at 0.010 rem for both beta and gamma through 2004.

The overall uncertainty in recorded dose is dependent on (1) administrative practices, (2) dosimetry technology, (3) calibration, and (4) workplace radiation fields. The Extrusion Plant used a variety of dosimetry vendors from 1962 to 2004, and the precise details of dosimeter type and calibrations are not available; consequently, uncertainty estimates are based on ORAUT (2005b). The uncertainty associated with beta-gamma film badge results from 1962 through 1985 are estimated to be ±30%, and the uncertainty associated with beta-gamma TLD badge results from 1986 through 2004 are also estimated to be ±30%.

Tables 6-2 through 6-4 are for information purposes and are based on the upper bound of reported doses received at the Extrusion Plant. The doses are listed in Table 6-3 for 1974 to 1984 and Table 6-4 for 1994 to 1999.

Table 6-3. Upper bound of reported whole-body deep doses, 1974 to 1984.

Year	Number monitored	No measurable data	<0.100 (rem)	0.100–0.249 (rem)	0.250–0.499 (rem)	0.500–0.749 (rem)	Reference
1974	69	22	27	17	3	No data	Heiser 1975
1975	62	19	31	11	1	No data	Heiser 1976
1976	58	30	20	8	No data	No data	Schaeffer 1977
1977	62	26	36	No data	No data	No data	Fletcher 1978, p. 167
1978	68	16	21	26	5	No data	Schaeffer 1979
1979	80	14	45	19	2	No data	Van Loocke 1980
1980	80	16	32	28	4	No data	Schaeffer 1981
1981	89	34	40	15	No data	No data	Fletcher 1978, p. 167
1982	116	20	51	34	10	1	Schaeffer 1983
1983	118	32	39	32	15	No data	Schaeffer 1984
1984	117	23	48	33	13	No data	Schaeffer 1985, p. 134
1985	124	38	58	22	4	2	Theisen 1986
1987	129	30	74	21	3	1	Theisen 1988
1988	117	45	68	4	No data	No data	Theisen 1989

Table 6-4. Upper bound of reported whole-body deep doses, 1994 to 1999.

Year	Number monitored	<0.010 (rem)	0.01–0.099 (rem)	0.100–0.249 (rem)	95th percentile	Reference
1994	330	291	38	1	0.099	TMA 1996
1995	255	220	35	No data	0.099	TN 1996
1996	285	241	44	1	0.099	TN 1997
1997	290	271	19	No data	0.099	TN 1998
1998	No data	No data	No data	No data	No data	Not applicable
1999	1,202	1,170	32	No data	<0.010	Eberline 2000

Table E-4 or E-5 should be used to assign external doses for either workers or construction trade workers (CTW) for the rare occasion when they are unmonitored CTW.

6.1.4 Missed Doses

Missed dose occurs when the dose of record is zero because the interpreted dose had a negative bias, dosimeter response was less than the minimum detectable level, or there was no dose of record for an assigned badge for a badge cycle. Estimates for missed dose can be assigned from Table 6-5 considering that the maximum potential annual missed dose is the 95th-percentile and missed dose is reported at the median (NIOSH 2007).

Table 6-5. Missed external doses (rem).

Monitoring period	Penetrating and nonpenetrating LOD	Exchange frequency	Maximum potential annual missed penetrating and nonpenetrating dose
1962–1966	0.030	Monthly	0.360
1967–1985	0.020	Quarterly	0.080
1986–1988	0.020	Quarterly	0.080
1989–1991	0.010	Monthly	0.120
1992–2003	0.010	Quarterly	0.040

6.2 NONPENETRATING DOSE

Because film badge and or TLD data are available for most of the period from 1962 through 1991, dose should usually be assigned based on the dose of record. Nonpenetrating dose associated with natural uranium consists primarily of electrons with energies above 15 keV.

For those workers who were not monitored or with unexplained gaps in their monitoring, the dose reconstructor should assign doses based on Table E-4 or E-5 for construction trade workers.

6.3 EXTREMITY MONITORING

The earliest extremity badge results were for wrist badge assignments to 12 personnel from July to September 1, 1974. Quarterly wrist badge results were also noted in December 1983 in two claim files. Finger rings appear to have replaced wrist badges in 1986. Monthly finger ring results (left and right) were present for 1986. Extremity doses were also noted in database output forms in the claim files beginning in 1987 and on some AEC-5 Forms in claim files.

According to the 1974 annual health protection appraisal, RMI was advised to begin immediate monitoring of extremity radiation exposures to workers who closely handled or inspected uranium billets to determine if continuous extremity monitoring would be required. This was in anticipation that the AEC would soon revise the radiation standard for forearm exposures to conform to a National Council on Radiation Protection and Measurements recommendation to decrease allowable exposures from 75 to 30 rem/yr. Several employees were expected to exceed 10% of the anticipated 30-rem limit that would require monitoring (Jelinek 1974). The use of wrist film badges was initiated in July 1974. The annualized results of a 6-month study of skin doses to the forearm and chest from July 1 to December 31, 1974, are listed in Table 6-6. According to correspondence from RMI to the U.S. Energy Research and Development Administration (a DOE predecessor agency) in January 1977, ring badge dosimeters had not yet been used at the Extrusion Plant to assess hand doses, but RMI estimated that hand exposures would run as high as 10 to 15 rem/yr for four to six workers at the plant (Van Loocke 1977).

A study was performed in November 1985 to determine the ratio between the fingertip dose and the TLD ring dose on workers whose inspection tasks required them to run their bare fingers across the

Table 6-6. Forearm and chest skin dose monitoring results for personnel who closely handled or inspected uranium billets in 1974.[a]

Job/task	Wrist badge (mrem/yr)	Chest badge (mrem/yr)
Lathe Operator	100	1,120
Billet Inspector	2,460	3,860
Lathe Operator	680	1,120
Forge Helper	5,120	5,420
Forge Helper	2,080	2,600
Roll Operator	360	2,500
Runout Table	820	1,900
Salt-Bath Operator	1,320	460
Forge Inspector	8,900	4,660
Saw Operator	820	1,960
Extrusion Inspector	3,020	4,580
Forge Inspector	3,640	5,580

a. Johnson (1976).

surface of uranium metal. Calibrated tissue-equivalent dosimeters were exposed on a hand phantom on a uranium billet at the plant. The results indicated that a properly worn finger ring would respond a maximum of 60% lower in comparison with a fingertip in contact with a billet. The study showed that a finger ring dosimeter worn on the inside of the little finger was as accurate as one worn on the inside of the finger ring. Finger rings worn on the outside of the hand significantly under-responded (Munson and Stacy 1985).

For most work categories, if extremity monitoring is not available, it is reasonable to estimate extremity entrance dose as being equal to the shallow dose measured by the chest badge result. For workers whose job category was inspector, the factor of 3 (derived from the ratio of the 1974 Salt Bath Operator's annual chest badge exposure to his annual wrist badge exposure), which is more favorable to the claimant, should be applied. Based on the Munson and Stacy (1985) study, finger doses could be a factor of 1.67 larger or more, which would result in a whole-to-finger ratio of 5.0.

6.4 MISCELLANEOUS INFORMATION CONCERNING EXTERNAL EXPOSURE

Early documentation from the Extrusion Plant indicates that whole-body film badges were to be displayed in the open and not obscured (RMI 1973), the results of a May 1985 independent health, safety, and environmental review at the Extrusion Plant indicated that film badges were inappropriately being placed into oversized plastic wrap to prevent them from becoming contaminated. By June 4, 1985, this was corrected by placing the badges into a properly sized heat-sealed plastic holder (Row 1985).

Personal contamination surveys were nonexistent or lax until 1986.

No Extrusion Plant site documentation was found that indicates industrial radiography sources were used at the facility.

6.5 EXTERNAL DOSE ASSIGNMENT SUMMARY

Many Extrusion Plant employees were monitored for beta, X-ray, and gamma external exposures. The dose reconstructor should assign the gamma doses as 30-250 keV photons, acute, 100% anterior posterior geometry, unless to lung, esophagus, or red bone marrow in which case isotropic and rotational geometries must be evaluated. All electron or beta doses are assigned as greater than

15 keV. All monitored and unmonitored doses are assigned in IREP as a constant distribution. Missed doses for monitored workers are assigned as a lognormal distribution in IREP.

For those employees not monitored sufficiently or unmonitored, the data in Attachment E summarizes the dose assignment values for unmonitored workers and construction trade workers. Dose reconstructors should assign dose to unmonitored workers with minimal potential for radiation exposure from RMI operations using the 50th-percentile dose. Electron (beta) doses are all >15 keV. Dose reconstructors should exercise judgment about the energy employee's work location and duties to determine the probability of extremity dose. For example, administrative personnel would be unlikely to receive extremity dose.

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] Elyse Thomas. Oak Ridge Associated Universities (ORAU) Team. Principal Medical Dosimetrist. May 2014.
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GLOSSARY

accreditation
For external dosimetry, the assessment of whether or not a personnel dosimetry system meets specific criteria. The assessment includes dosimeter performance and the associated quality assurance and calibration programs.

background radiation
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.

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.

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.

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.

deep dose equivalent
Dose equivalent in units of rem or sievert for a 1-centimeter depth in tissue (1,000 milligrams per square centimeter). See *dose*.

depleted uranium
Uranium with a percentage of ^{235}U lower than the 0.7% found in natural uranium.

DOE Laboratory Accreditation Program (DOELAP)
Program for accreditation by DOE of DOE site personnel dosimetry and radiobioassay programs based on performance testing and the evaluation of associated quality assurance, records, and calibration programs.

dose
In general, the specific amount of energy from ionizing radiation that is 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, rad, rep, 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. See *thermoluminescent dosimeter*.

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% by mass. Reactor-grade uranium is usually about 3.5% ^{235}U; weapons-grade uranium contains greater than 90% ^{235}U.

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.

film
(1) In the context of external dosimetry, radiation-sensitive photographic film in a light-tight wrapping. (2) X-ray film.

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.

gamma radiation
Electromagnetic radiation (photons) of short wavelength and high energy (10 kiloelectron-volts to 9 megaelectron-volts) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma photons are identical to X-ray photons of high energy; the difference is that X rays do not originate in the nucleus.

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.

in vitro bioassay
Measurements to determine the presence of or to estimate the amount of radioactive material in the excreta or in other biological materials removed from the body.

in vivo bioassay
Measurements of radioactive material in the human body utilizing instrumentation that detects radiation emitted from the radioactive material in the body.

ionizing radiation
Radiation of high enough energy to remove an electron from a struck atom and leave behind a positively charged ion. High enough doses of ionizing radiation can cause cellular damage. Ionizing particles include alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, photoelectrons, Compton electrons, positron/negatron pairs from photon radiation, and scattered nuclei from fast neutrons.

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.

natural uranium

Uranium as found in nature, approximately 99.27% ^{238}U, 0.72% ^{235}U, and 0.0054% ^{234}U by mass. 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.

nucleus

Central core of an atom, which consists of positively charged protons and, with the exception of ordinary hydrogen, electrically neutral neutrons. The number of protons (atomic number) uniquely defines a chemical element, and the number of protons and neutrons is the mass number of a nuclide. The plural is nuclei.

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.

photon

Quantum of electromagnetic energy generally regarded as a discrete particle having zero rest mass, no electric charge, and an indefinitely long lifetime. The entire range of electromagnetic radiation that extends in frequency from 10^{23} cycles per second (hertz) to 0 hertz.

proton

Basic nucleic particle with a positive electrical charge and mass slightly less than that of a neutron. There are protons in the nuclei of every atom, and the number of protons is the atomic number, which determines the chemical element.

quality factor

Principal modifying factor (which depends on the collision stopping power for charged particles) that is employed to derive dose equivalent from absorbed dose. The quality factor multiplied by the absorbed dose yields the dose equivalent. See *dose*.

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) with kinetic energy that interact with matter through various mechanisms that involve energy transfer. See *ionizing radiation*.

radioactive

Of, caused by, or exhibiting radioactivity.

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.

radioactivity
Property possessed by some elements (e.g., uranium) or isotopes (e.g., ^{14}C) of spontaneously emitting energetic particles (electrons or alpha particles) by the disintegration of their atomic nuclei. See *radionuclide*.

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

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

roentgen
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 degrees Celsius and standard atmospheric pressure. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher energy photons (generally greater than 100 kiloelectron-volts).

thermoluminescent dosimeter (TLD)
Device for measuring radiation dose that consists of a holder containing solid chips of material that, when heated, release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

transuranic elements
Elements with atomic numbers above 92 (uranium). Examples include plutonium and americium.

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 U.S. 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.

whole-body dose
Dose to the entire body excluding the contents of the gastrointestinal tract, urinary bladder, and gall bladder and commonly defined as the absorbed dose at a tissue depth of 10 millimeters (1,000 milligrams per square centimeter). Also called penetrating dose. See *dose*.

X-ray radiation
Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.

ATTACHMENT A
PLANT LAYOUTS

LIST OF TABLES

LIST OF FIGURES

ATTACHMENT A
PLANT LAYOUTS (continued)

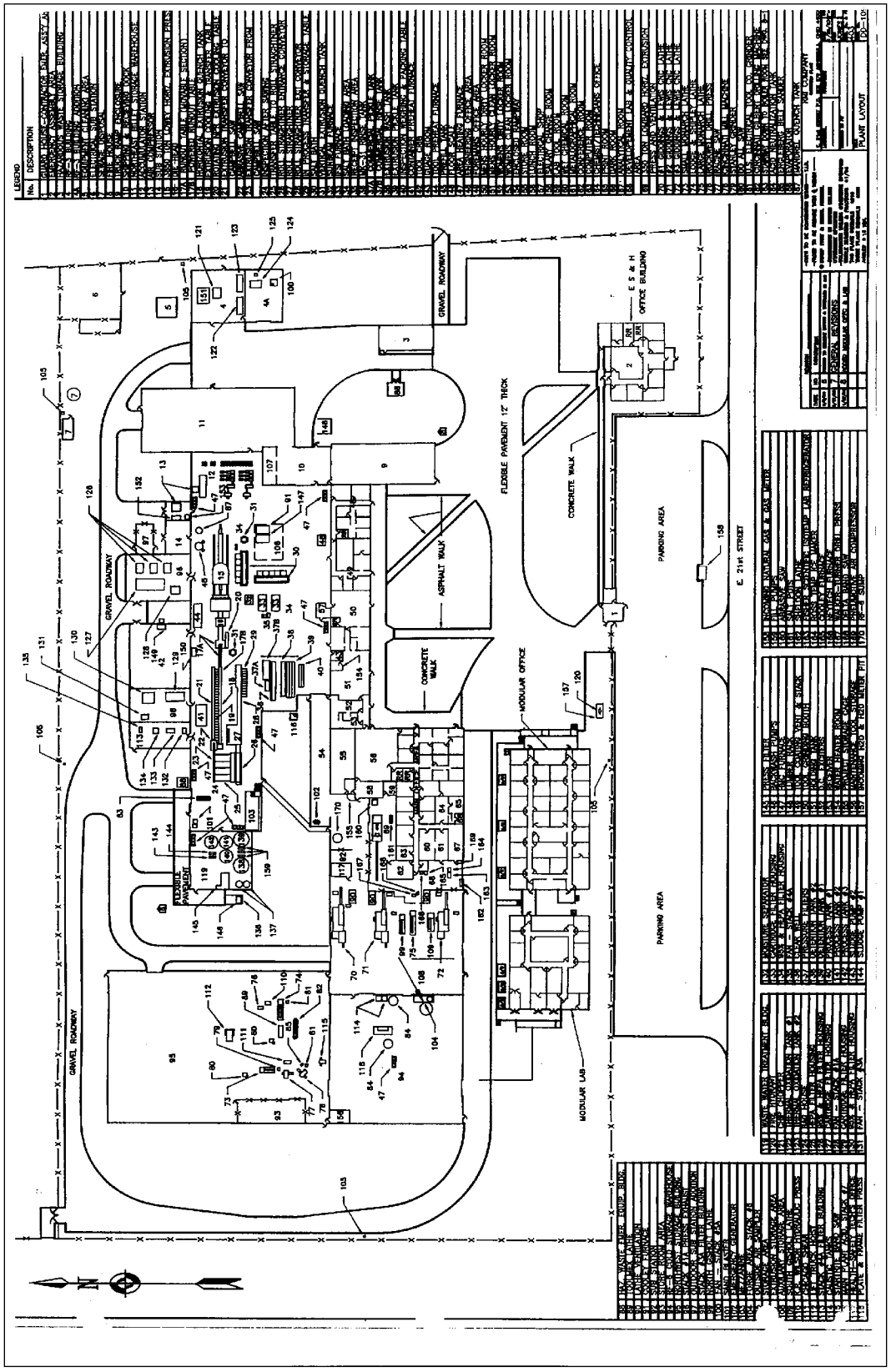

Figure A-1. Plant layout (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

Table A-1. Key for Figure A-1.

No.	Description	No.	Description
1	Guard House - Contractor Emergency Assembly Area	53	Women's Clean Locker Room
2	Emergency Assembly Area	54	Enclosed Rampway
3	Hazardous Waste Storage Building	55	Storeroom
4	RF-3 Building	56	Lunchroom
4A	RF-3 Building Addition	57	Electrical Shop
5	Equipment Cleaning Area	58	Salary Change Room
6	Electrical Substation	59	Lab Tool Room
7	Sewage Disposal	60	Wet Chemical Room
8	Firehouse	61	Metalographical Room
9	Truck Ramp Enclosure	62	Conference Room
10	Shipping and Receiving Back	63	Computer Room
11	Northeast Billet Storage Warehouse	64	Chemist/Technicians Office
12	Accumulator Station	65	Print Room
13	Air Compressor	66	Dark Room
14	Substation	67	Analytical Room
15	3850 Ton Loewy Horizontal Extrusion Press	68	Development Lab and Quality Control Area
16	Die Head	69	350 Ton Lombard Horizontal Extrusion Press and Ventilation
17A	Runout Table (Movable Section)	70	#1 Giddings and Lewis CNC Lathe
17B	Powered Runout Table	71	#2 Giddings and Lewis CNC Lathe
18	Horizontal Extrusion Quench Tank	72	#3 Giddings and Lewis CNC Lathe
19	Extrusion Cooling and Transfer Table	73	Small Monarch Lathe
20	Rotating NPR Extrusion Cooling Table	74	Lodge and Shipley Lathe
21	Extrusion Transfer Conveyor To Campbell Saw	75	Large Monarch Lathe
22	Abrasive Campbell Saw	76	Rockwell Drill Press
23	Extrusion Transfer Conveyor From Campbell Saw	77	Racine Saw
24	Inspection After Sawing	78	Cincinnati Mill Machine
25	Transfer Table To Roll Straightener	79	Stanley Grinder
26	Roll Straightener Entrance Conveyor	80	Do All Saw
27	Roll Straightener	81	U.S. Electrical Tool Co. Grinder
28	Roll Straightener Exit Conveyor	82	Cincinnati Gilbert Drilling Machine
29	In Process Transfer and Storage Cable	83	Stairway Down To Boiler Room
30	Salt Bath	84	Acid Neutralization Tank
31	Vertical Extrusion Quench Tank	85	Engleberg Belt Sander
32	Sunbeam Furnace	86	Evaporator
33	FSI Furnace	87	Mandrel Quench Tank
34	Salt Bath Loading Area	88	Hazardous Waste Emergency Equipment Building
35	Receipt Inspection Area	89	Toolmex Lathe
36	MK-31 Rinse Tank	90	Lathe Ventilation
37A	Commercial Pickle Tank	91	Cooley Furnace
37B	Commercial Rinse Tank	92	Substation
38	Extrusion Wash Tank	93	Storeroom Area
39	Extrusion Pickle Tank	94	RF -- 6 Cold Storage Warehouse
40	Inspection Weighing and Packing Table	95	Northwest Storage Building
41	Container Preheat Furnace	96	Stack #1A Press Exhaust
42	Toll Crib	97	Outdoor Substation Addition
43	Gauge Room	98	Stack #3A Filter Building
44	Tool Preheat Furnace	99	North Gisholt Lathe
45	Prefill Tank	100	Fan -- Stack #5A
46	Floor Scale	101	Sandblaster
47	Area Heating Furnace	102	Emergency Generator

ATTACHMENT A
PLANT LAYOUTS (continued)

No.	Description	No.	Description
48	Engineering Office Area	103	Mezzanine
49	Foremen's Office Area	104	Forge Area Stack #8
50	Men's Hourly Dirty Locker Room	105	Outside Air Sampler
51	Men's Hourly Clean Locker Room	106	Storage Area
52	Women's Dirty Locker Room	107	Extrusion Storage Area
108	Auxiliary Storage Area	134	95% and HEPA Filter Housing
109	South Gisholt Lathe	135	Fan -- Stack #4A
110	K. R. Wilson Hydraulic Press	136	Clear Well
111	Chicago Shear	137	Pressure Filters
112	Lift Truck Hoist	138	Detention Tank #2
113	Stack #4Afilter Building	139	Detention Tank #1
114	Caustic Tanks	140	Processed Tank #1
115	Startrite Band Saw	141	Processed Tank #2
116	Main Plant Acid, Stack #7	142	Processed Tank #3
117	Health-Safety Technician's Office	143	Sludge Pump #2
118	Plate and Frame Filter Press	144	Sludge Pump #1
119	Wastewater Treatment Building	145	Pressure Filter
120	Fire Hydrant	146	Backwash Pumps
109	South Gisholt Lathe	147	H. M. C. Furnace
110	K. R. Wilson Hydraulic Press	148	Lumber Rack
111	Chicago Shear	149	Tool Coating Vent and Stack
112	Lift Truck Hoist	150	Tool Grinding Booth
113	Stack #4Afilter Building	151	Holding Tank
114	Caustic Tanks	152	D. C. Exciters
115	Startrite Band Saw	153	Receiver
116	Main Plant Acid, Stack #7	154	Water Heater Room
117	Health-Safety Technician's Office	155	Project Storage Cage
118	Plate and Frame Filter Press	156	Maintenance Tool Storage
119	Wastewater Treatment Building	157	Incoming Water and Water Meter Pit
120	Fire Hydrant	158	Incoming Natural Gas and Gas Meter
121	Chip Chopper	159	Still Pumps
122	Thermal Oxidation Tank #1	160	Abrasive Saw
123	Thermal Oxidation Tank #2	161	Salt Pots
124	Baghouse	162	Sheldon Lathe
125	HEPA Filter Housing	163	Fisher Scientific Iso-Temp Lab Refrigerator
126	95% and HEPA Filter Housing	164	Ross Temp Icemaker
127	Cartridge Filter Housing	165	Cooley Furnace
128	Fan -- Stack #1A	166	Neytech Furnace
129	Cartridge Filter Housing	167	Walker-Turner Drill Press
130	95% and HEPA Filter Housing	168	Delta Band Saw
131	Fan -- Stack #3A	169	Pneumotive Air Compressor
132	Moisture Separator	170	RF -- 6 Sump
133	Cartridge Filter Housing	–	–

ATTACHMENT A
PLANT LAYOUTS (continued)

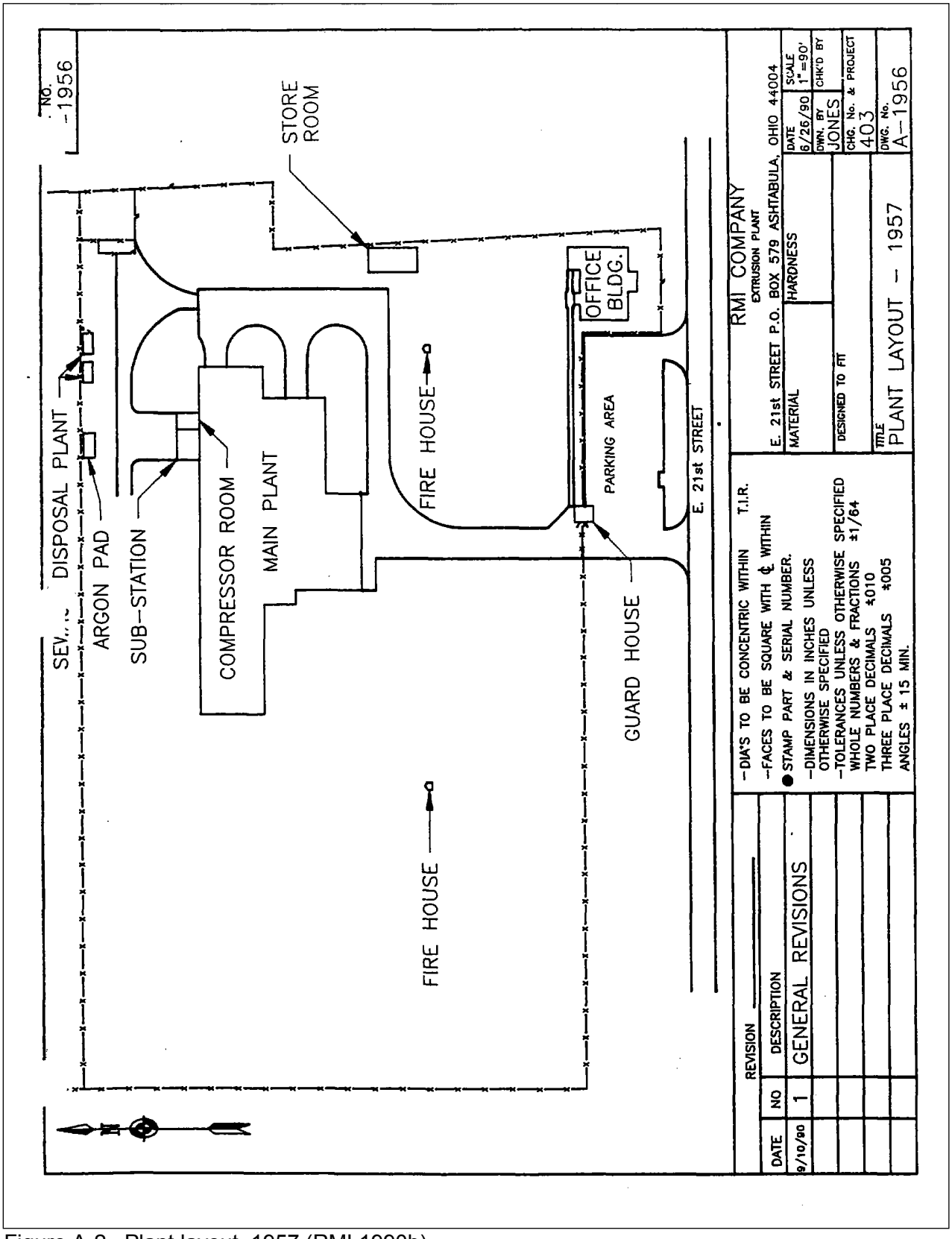

Figure A-2. Plant layout, 1957 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

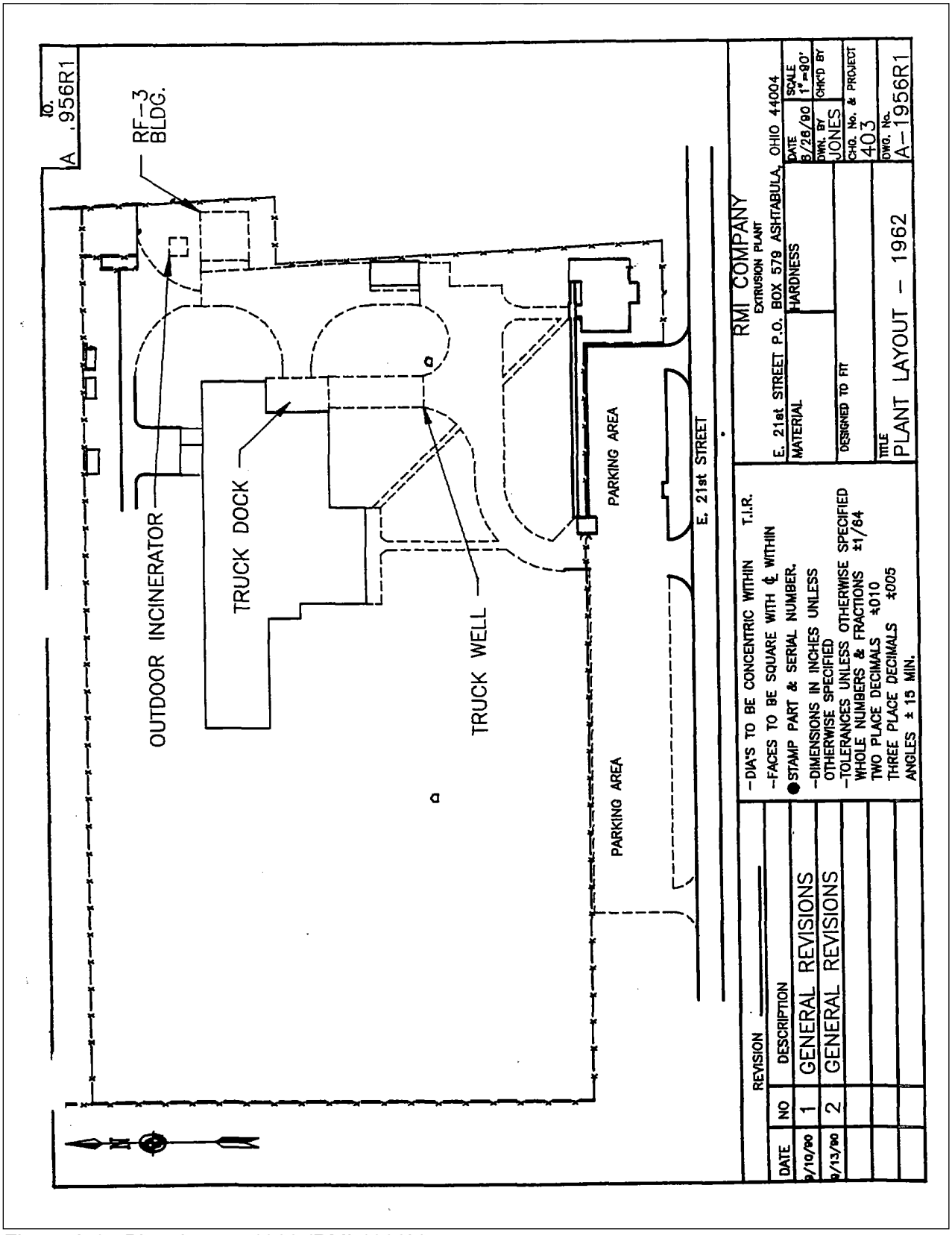

Figure A-3. Plant layout, 1962 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

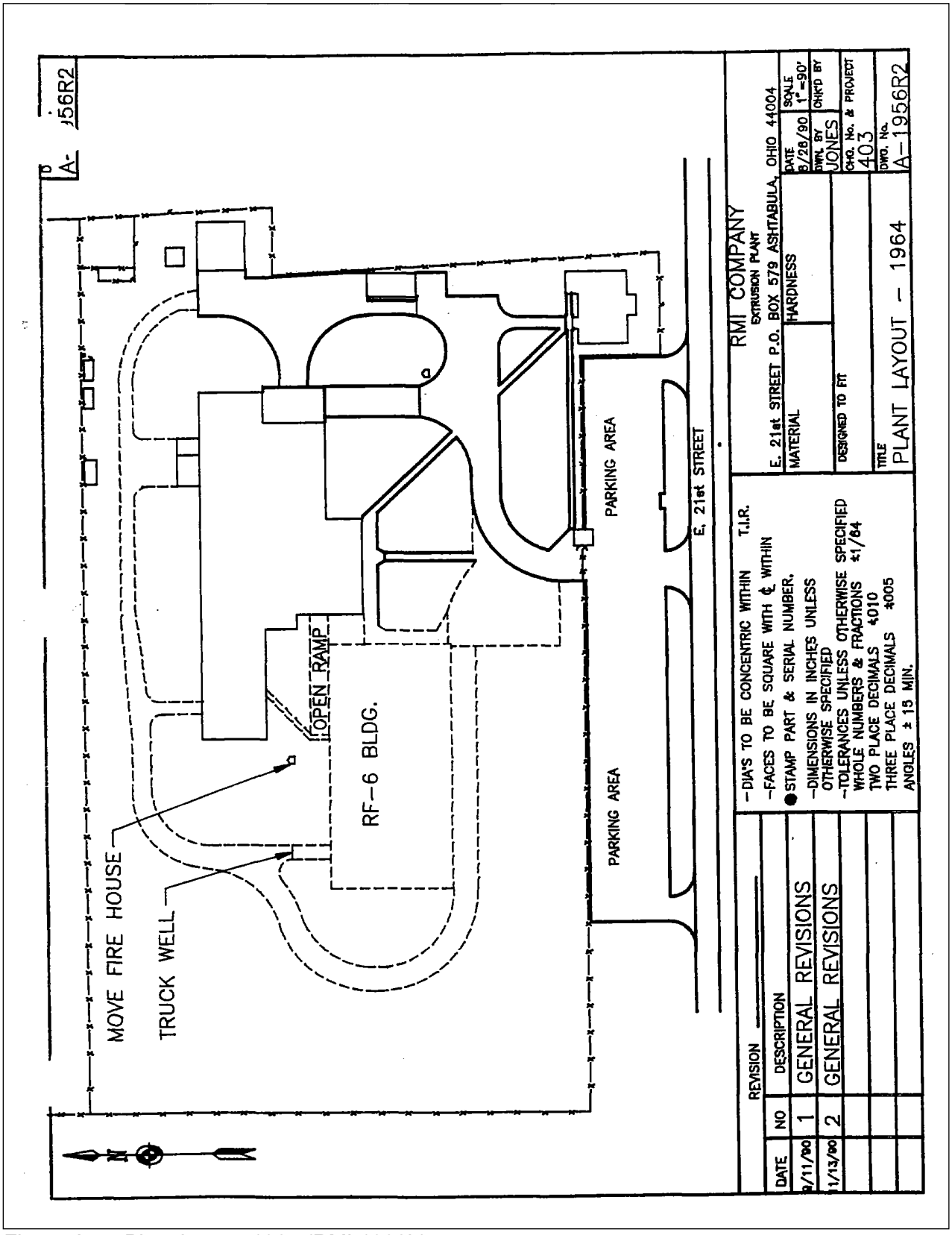

Figure A-4. Plant layout, 1964 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

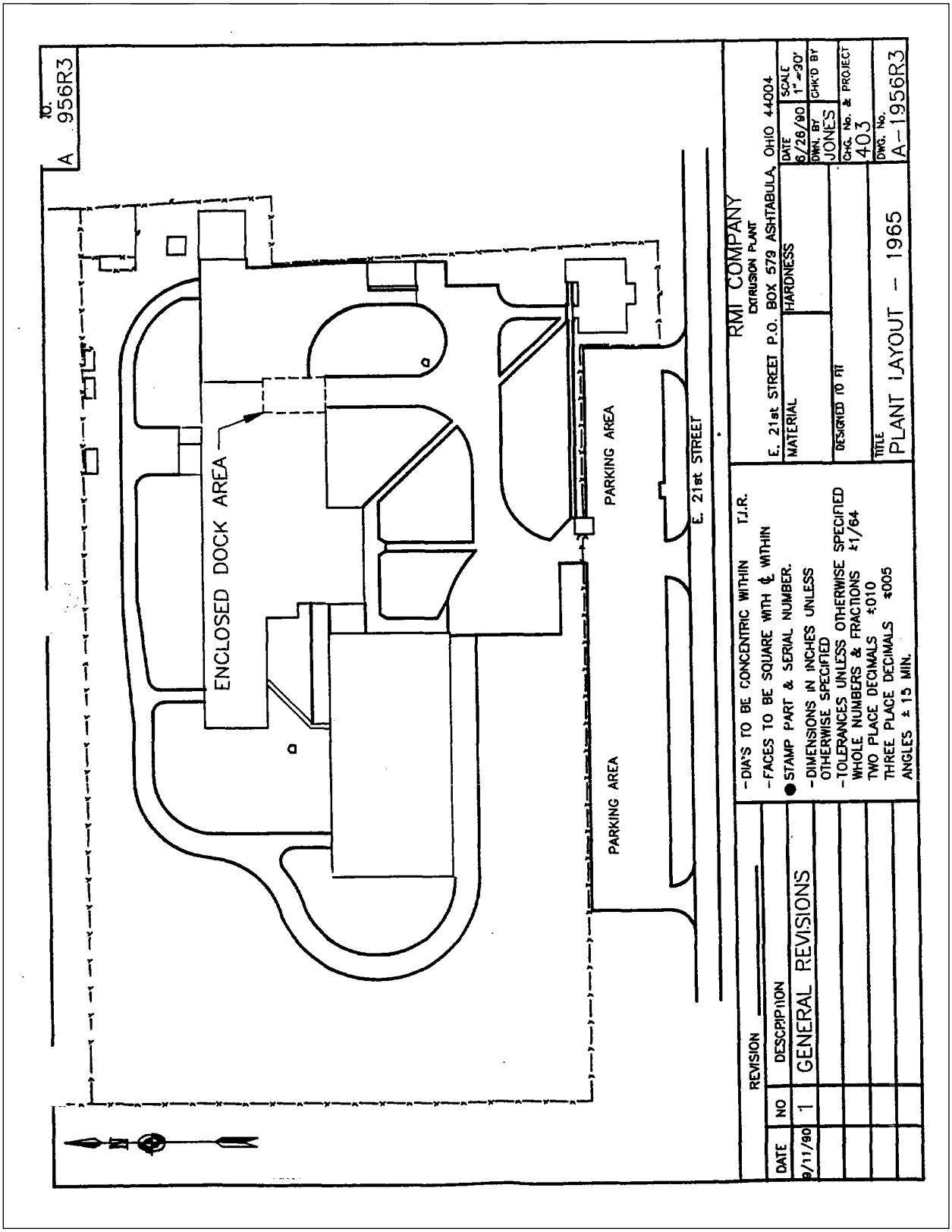

Figure A-5. Plant layout, 1965 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

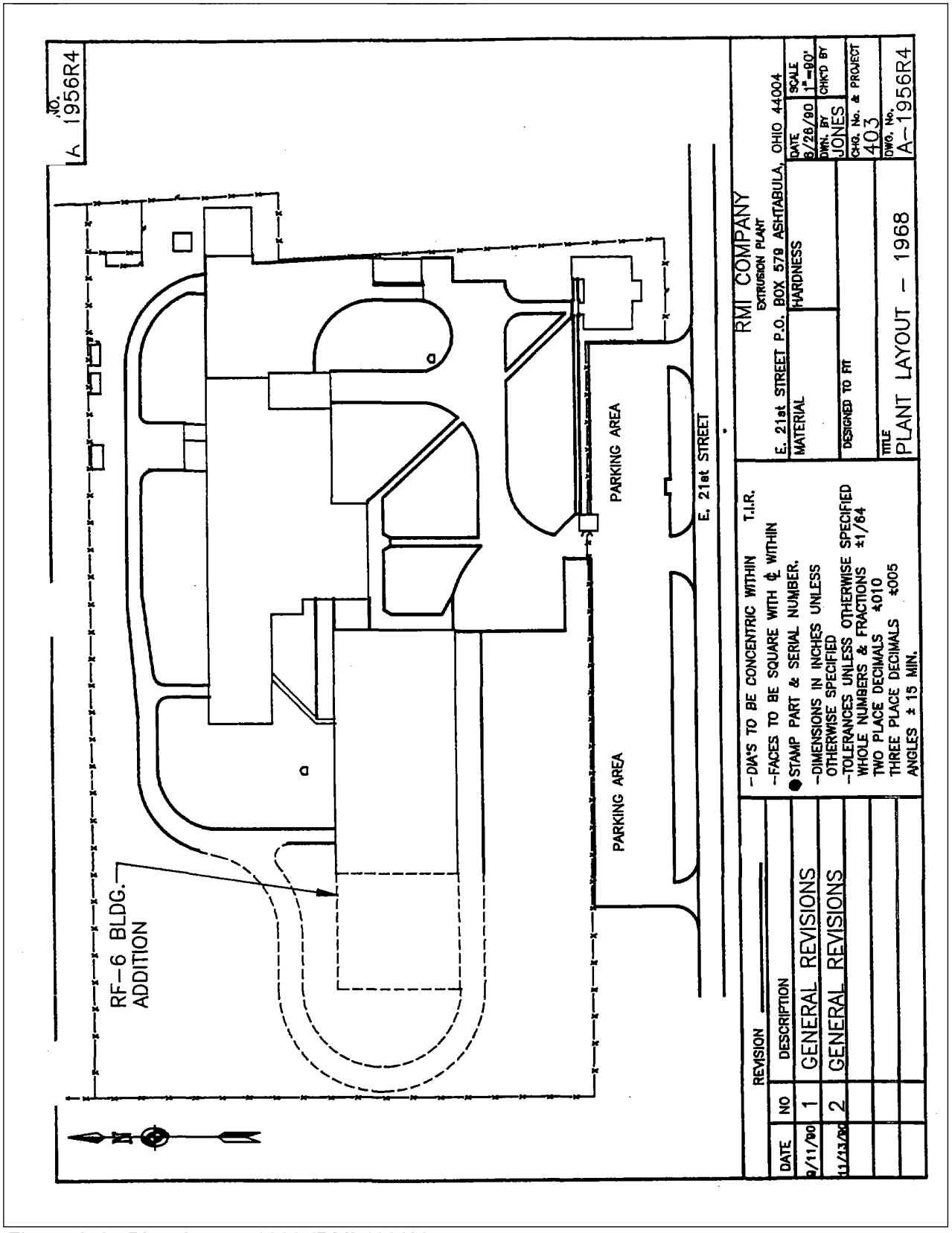

Figure A-6. Plant layout, 1968 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

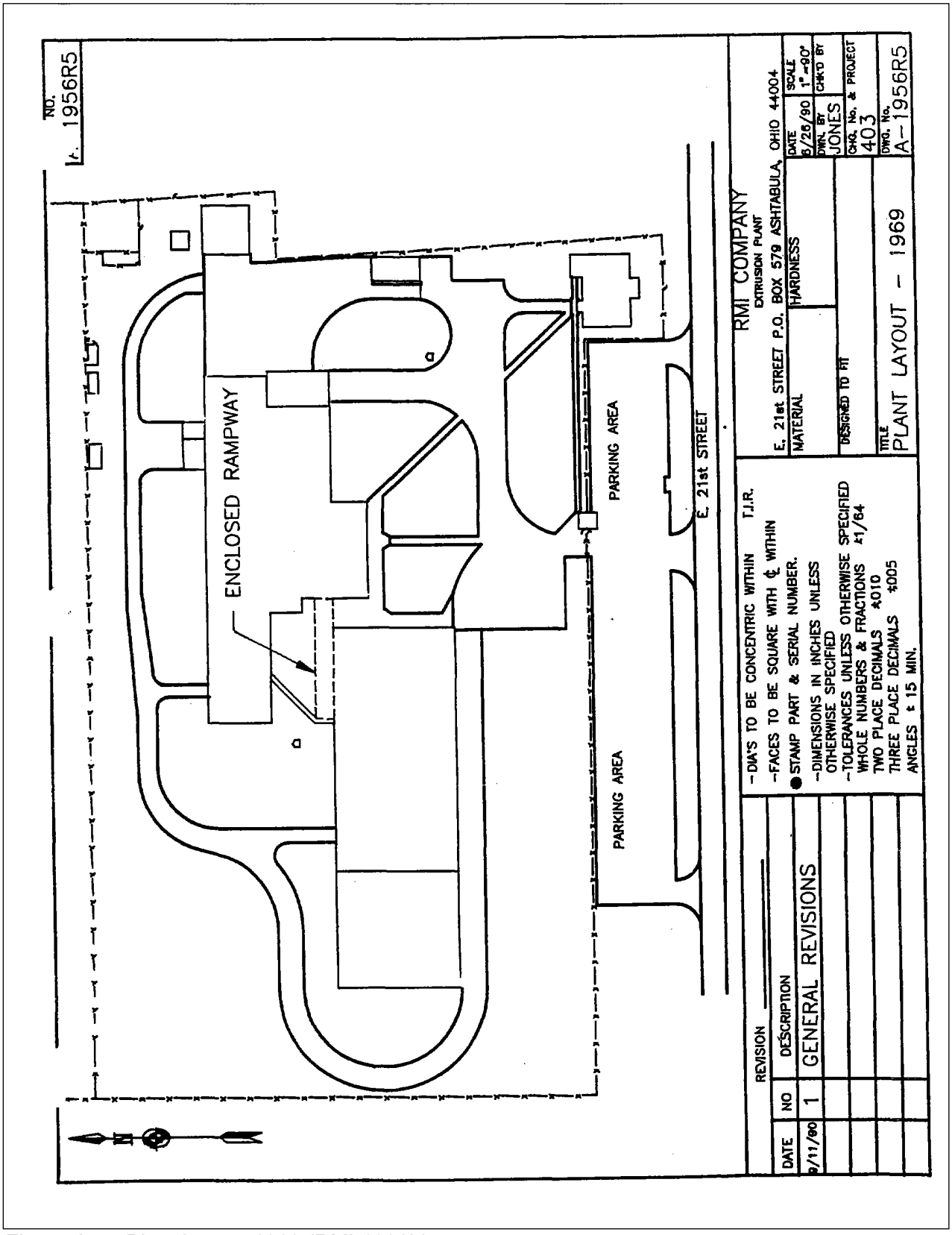

Figure A-7. Plant layout, 1969 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

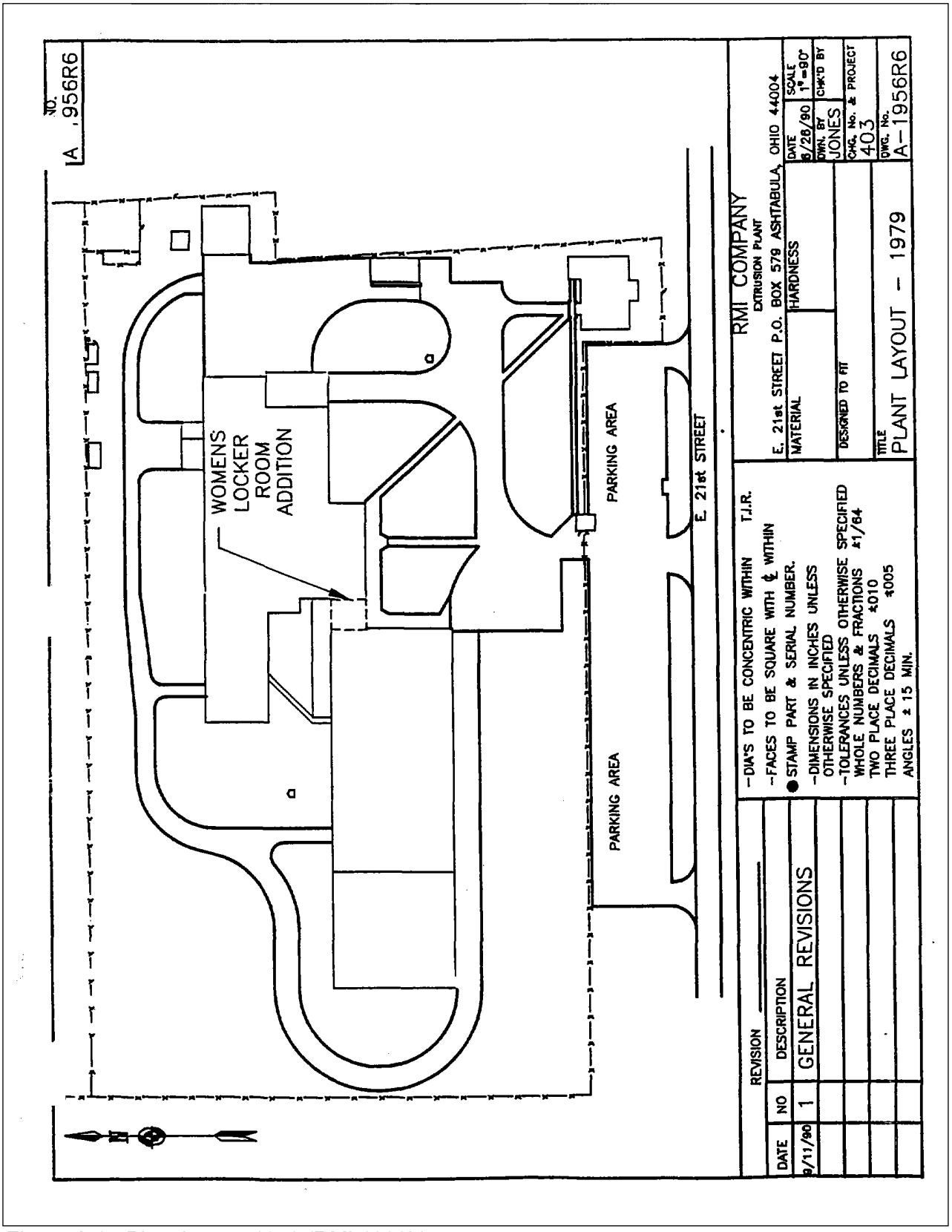

Figure A-8. Plant layout, 1979 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

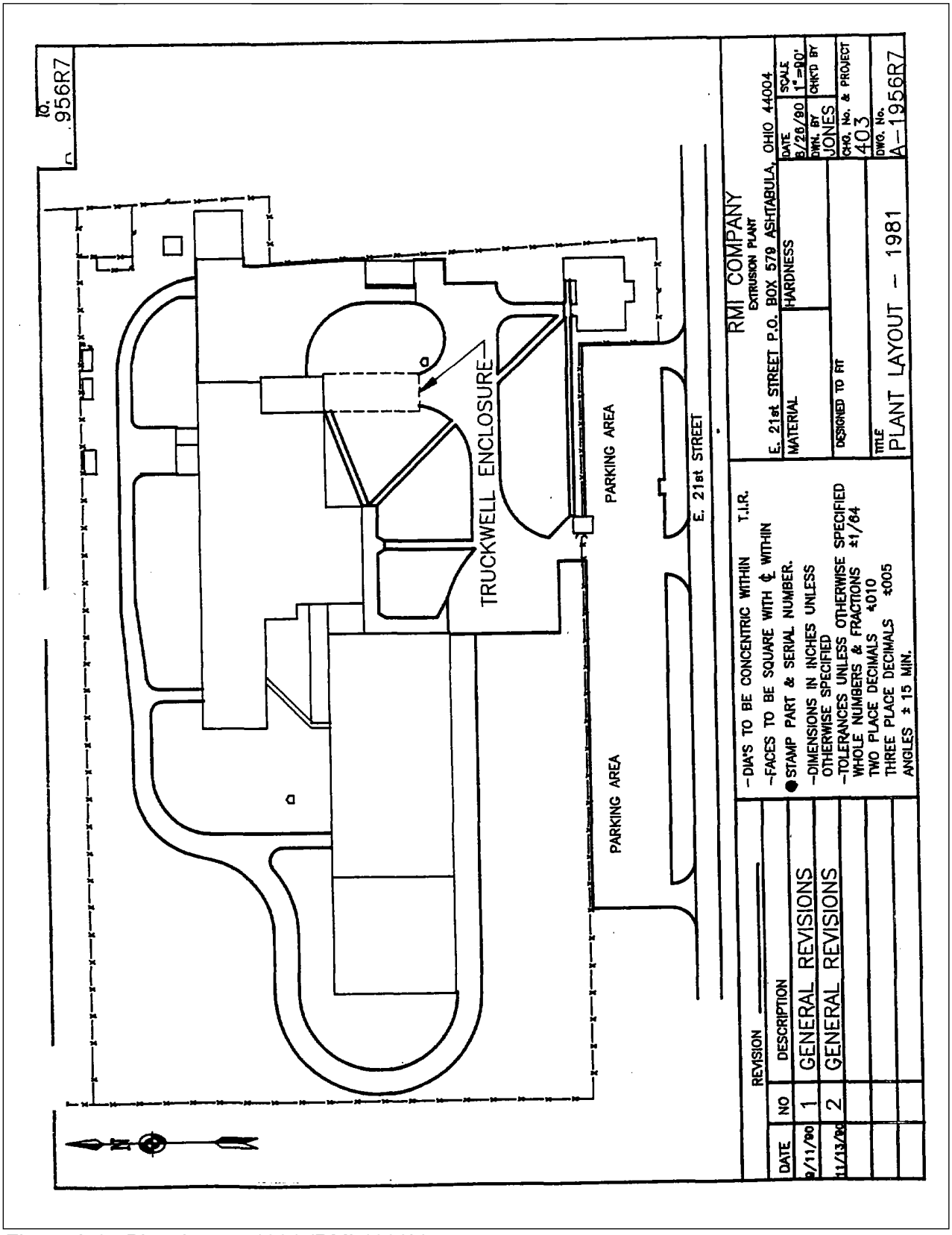

Figure A-9. Plant layout, 1981 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

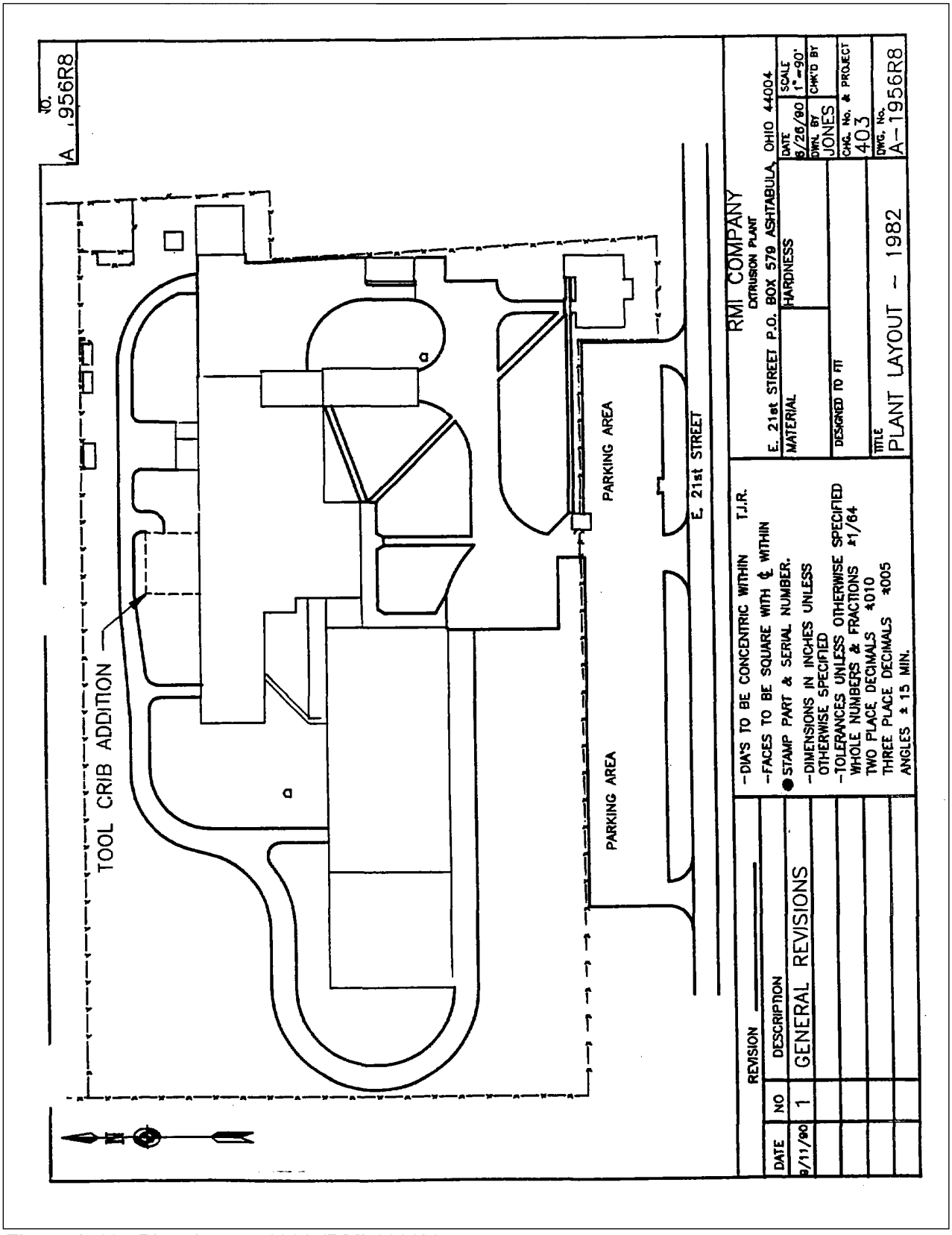

Figure A-10. Plant layout, 1982 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

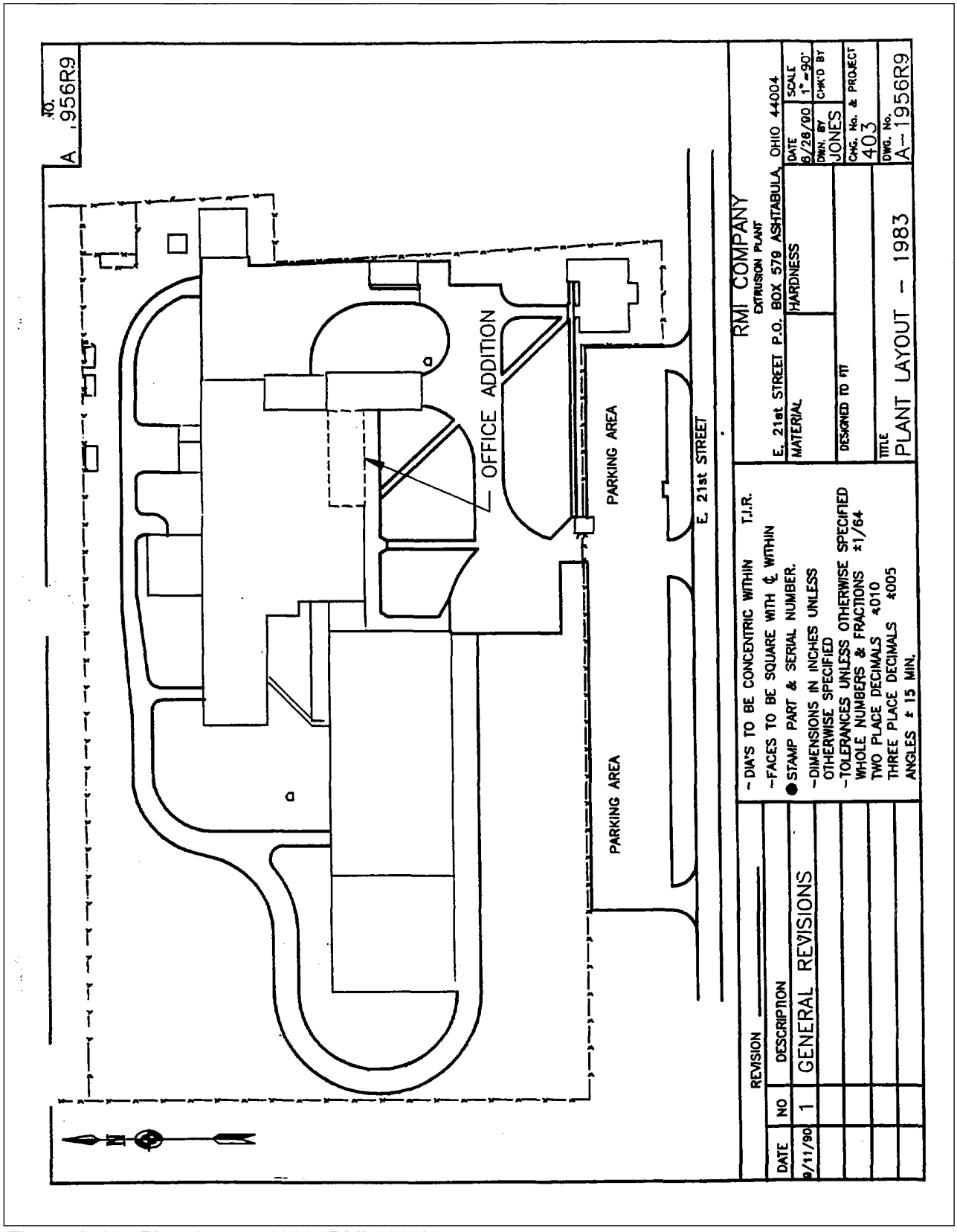

Figure A-11. Plant layout, 1983 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

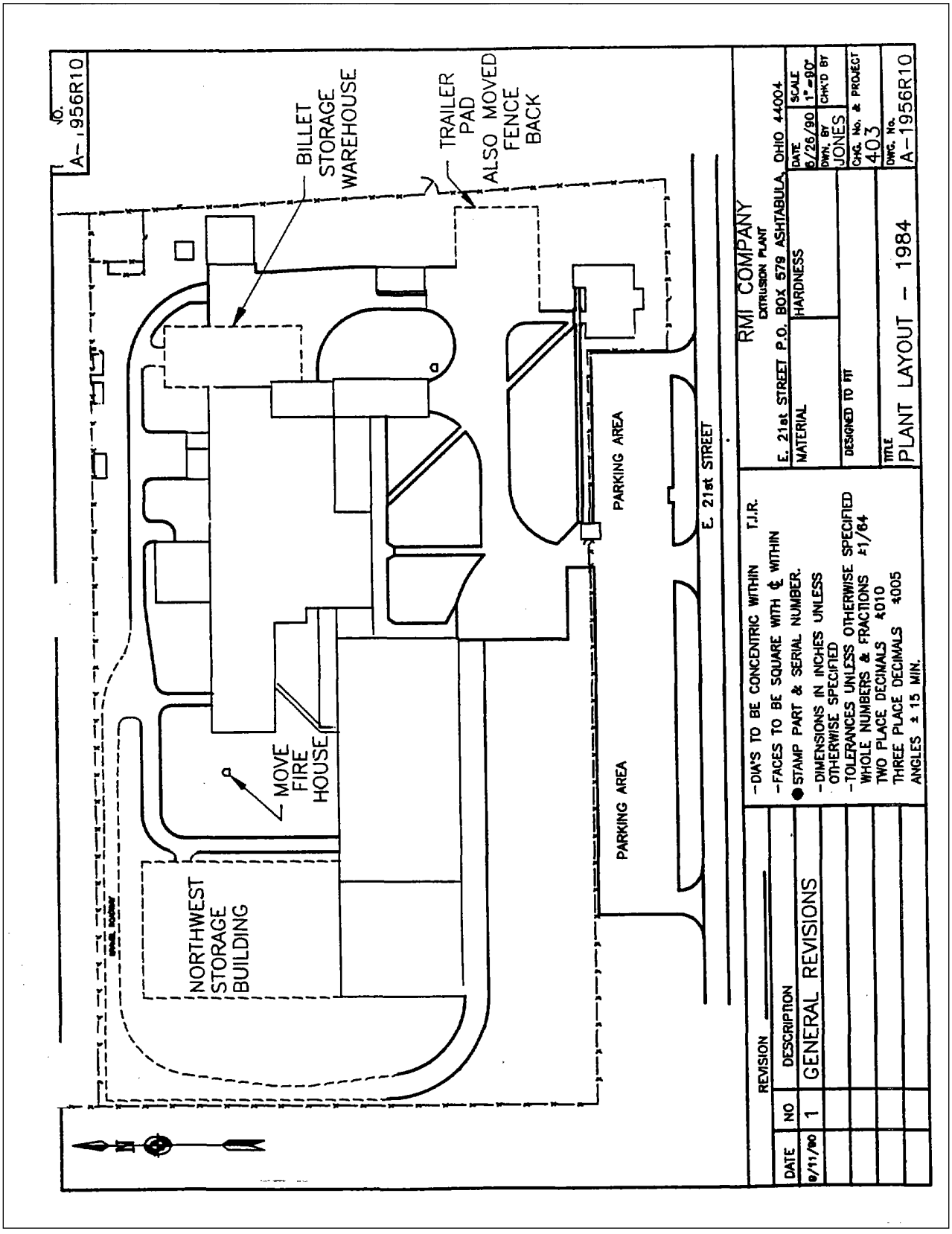

Figure A-12. Plant layout, 1984 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

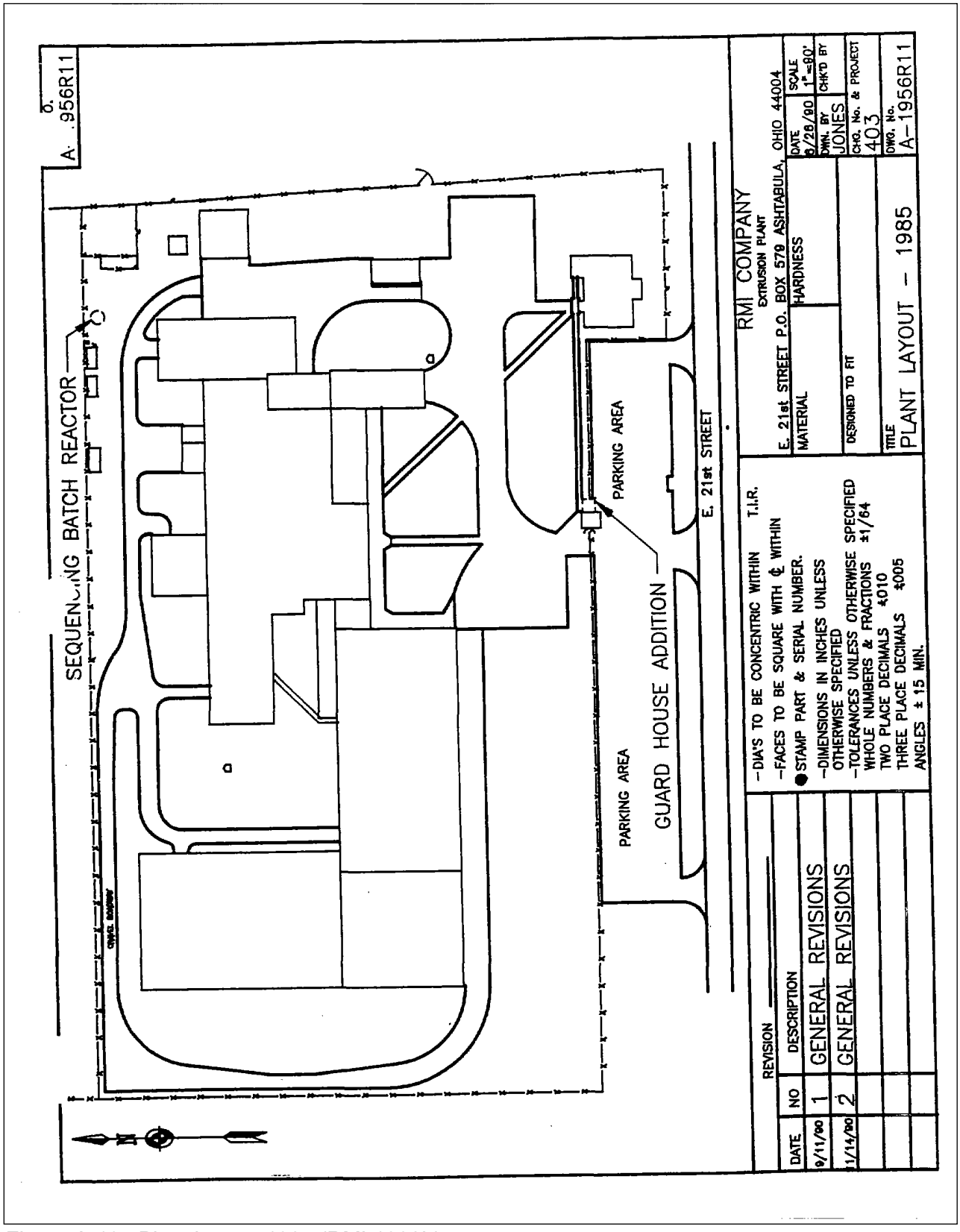

Figure A-13. Plant layout, 1985 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

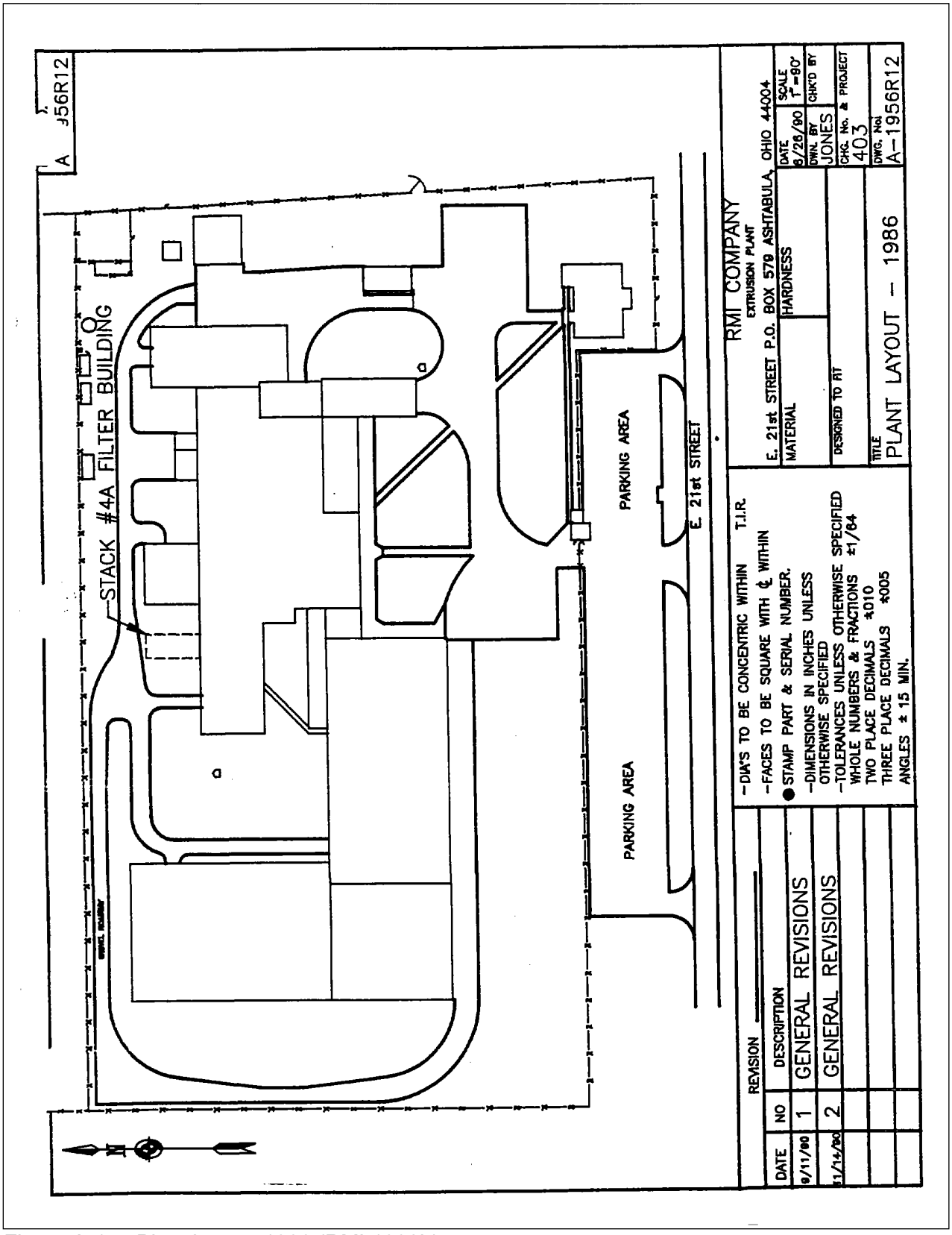

Figure A-14. Plant layout, 1986 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

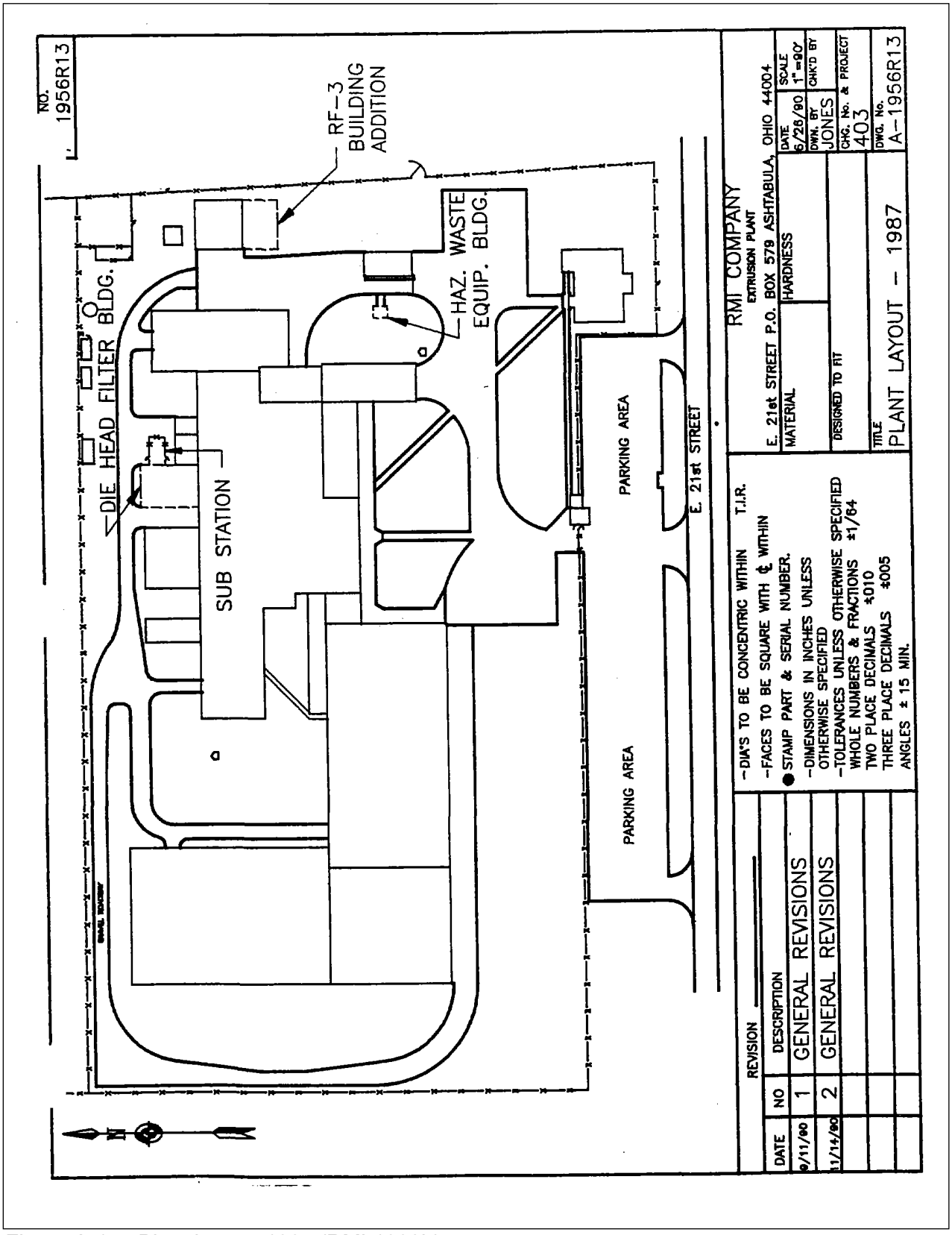

Figure A-15. Plant layout, 1987 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

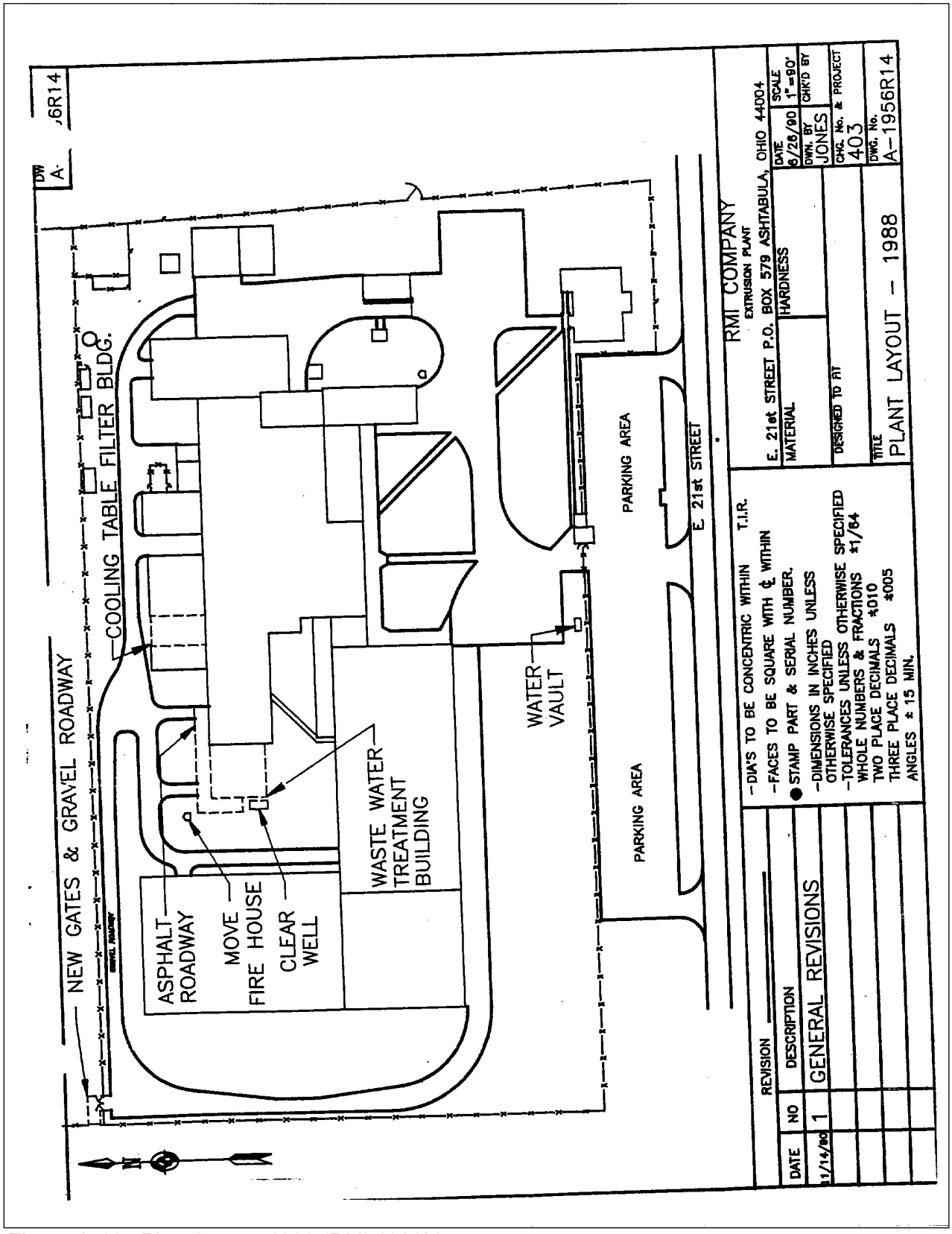

Figure A-16. Plant layout, 1988 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

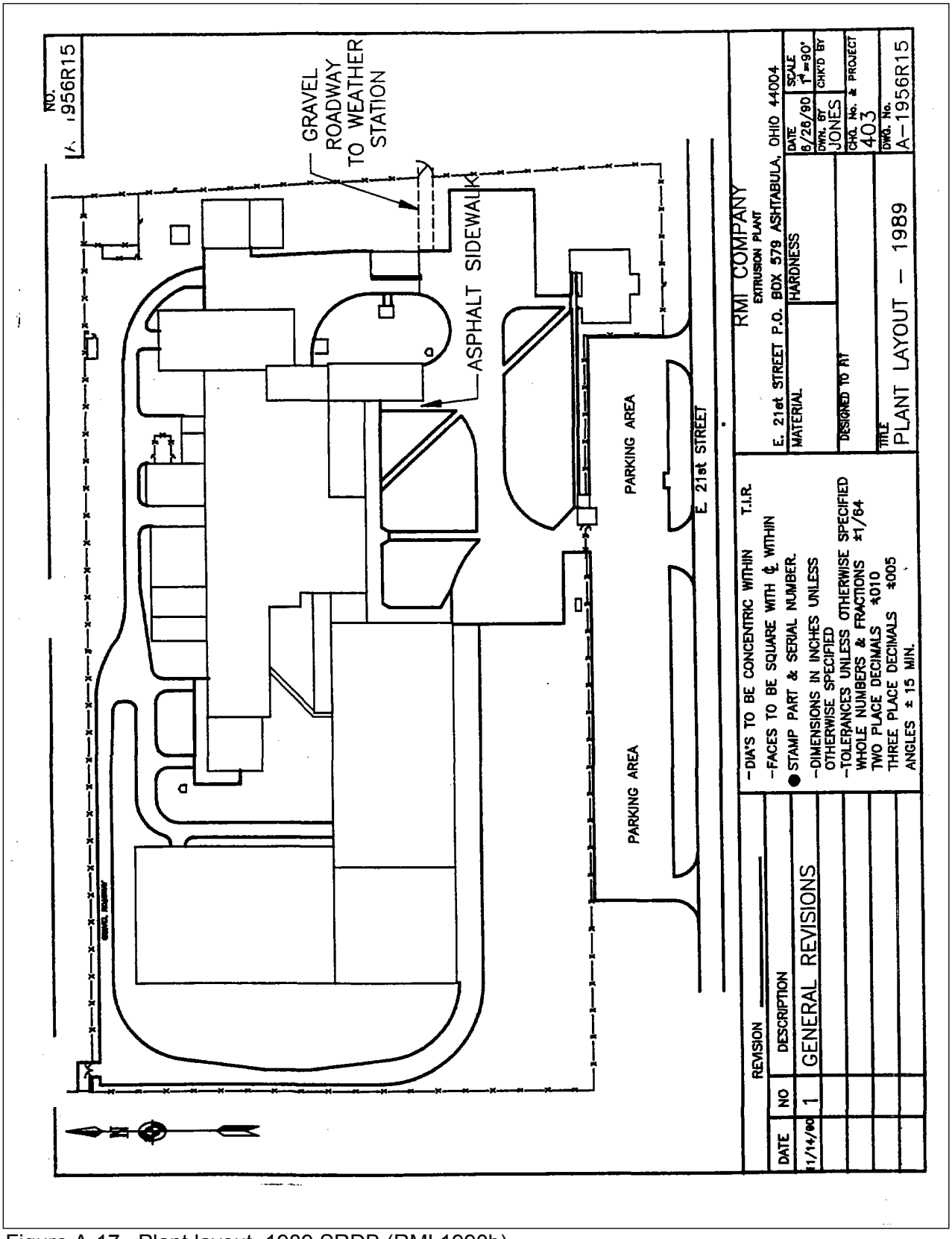

Figure A-17. Plant layout, 1989 SRDB (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

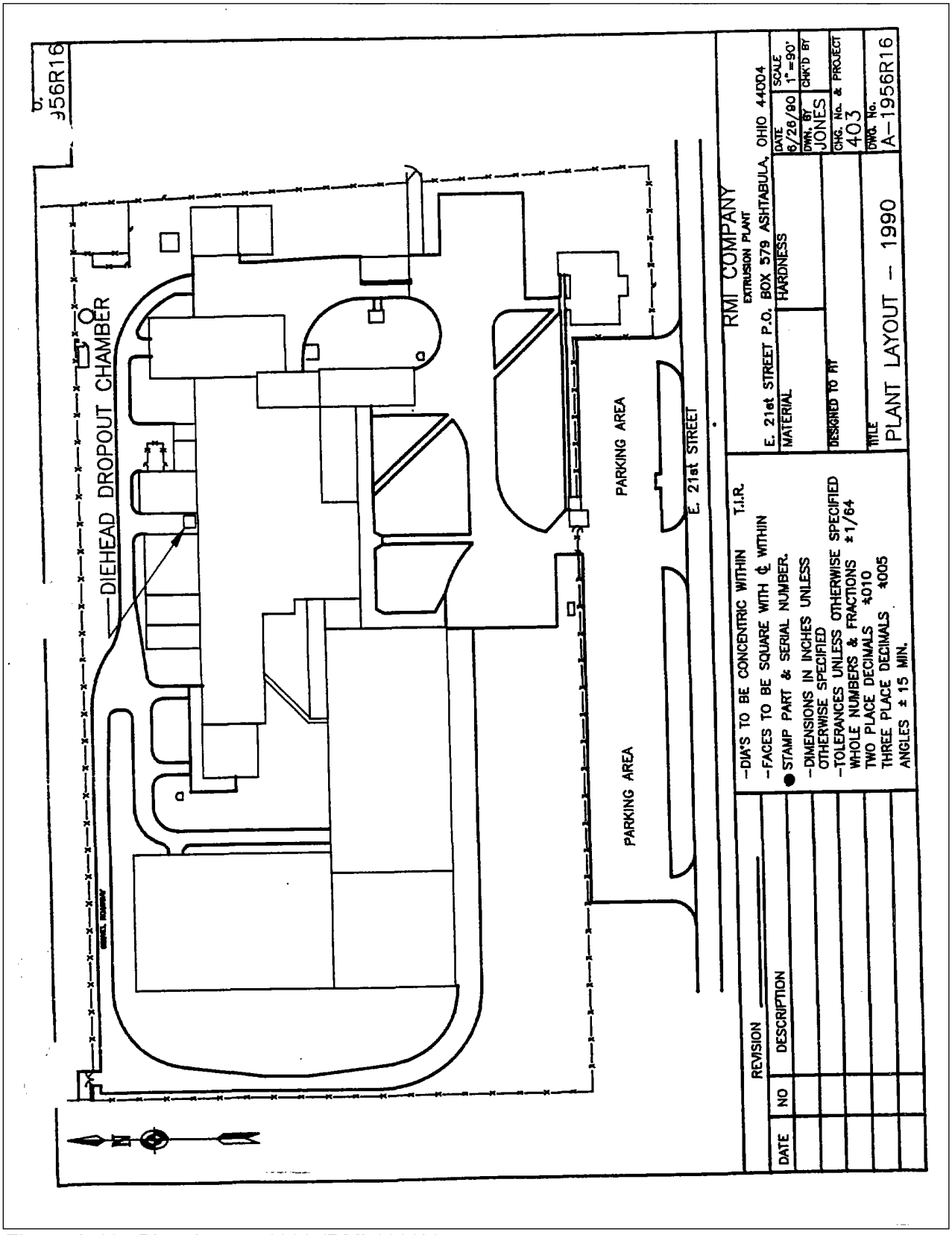

Figure A-18. Plant layout, 1990 (RMI 1990b).

ATTACHMENT A
PLANT LAYOUTS (continued)

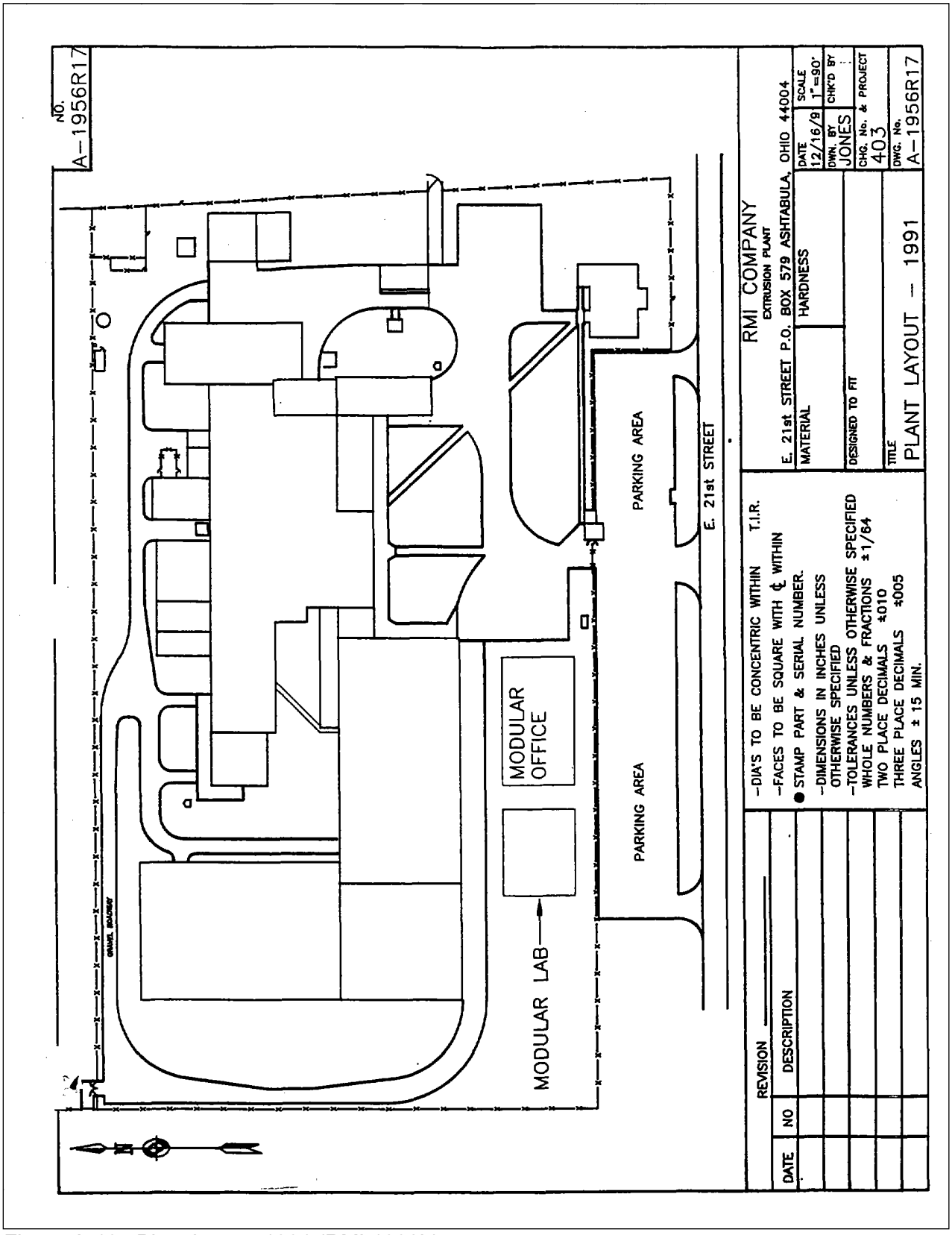

Figure A-19. Plant layout, 1991 (RMI 1990b).

ATTACHMENT B
N-REACTOR PRODUCTION

According to RMI (1995), "the N-Reactor Production consisted of a primary extrusion process followed by a forging process." The typical steps follow RMI (1995):

1. Ingots were received from FMPC.

2. Ingots were transferred to storage.

   Note: All onsite transfer of ingots, billets, forgings, and extrusions was accomplished by use of forklifts, fixtures, and overhead cranes.

3. Ingots were transferred to inspection.

4. Inspection was performed using an overhead monorail, scale, and inspection stand. Inspection included weighing and a dimensional and visual inspection.

5. Ingots were transferred back to storage.

6. Ingots were transferred to the salt baths and heated in molten salt to approximately 1,180°F for 1.5 hours minimum, 6 hours maximum.

7. Ingots were transferred to the press.

8. Ingots were extruded through the press into a heavy walled tube. This process included lubrication of press tooling to reduce friction during high-pressure extrusion.

9. The extrusion exited through the die head onto a runout table, and was placed on a rotating table and left for approximately 2.5 minutes.

10. The extrusion was transferred and lowered vertically into a water-filled quench tank for cooling for a minimum of 3 minutes.

11. The extrusion was transferred to a transfer table.

12. The extrusion was lifted horizontally and:

    a. From 1962 to the mid-1960s, was placed into a trichloroethylene vapor degreaser tank to clean the extrusion.

    b. From the mid-1960s, was placed in a nitric acid pickle tank to clean the extrusion.

13. The extrusion was transferred to a water rinse tank, rinsed, and transferred to the packing station for inspection. Then:

    a. Until the 1960s, extrusions were packed, stored, and shipped back to FMPC for further processing.

    b. In the late 1960s, further processing was performed at as described below.

14. The extrusion was cut into sections (billets).

ATTACHMENT B
N-REACTOR PRODUCTION (continued)

15. Billets were nitric acid pickled, inspected, machined, and reinspected before a closed die forging process at the press.

16. Billets were transferred to the storage area to await forging.

17. Billets were transferred to the salt baths and heated in molten salt. Billets were heated to approximately 1,170°F for 1-hour minimum, 6-hour maximum.

18. Billets were transferred to the press.

19. Billets were forged (closed die process) in the press. This process shaped the billet to the approximate dimensions of the final size and included lubrication of press tooling.

20. After forging, billets were lifted directly from the die head and lowered into a water-filled quench tank for cooling for a minimum of 3 minutes.

21. Billets were transferred to a transfer table for post forging inspection.

22. Billets were nitric acid pickled, inspected, machined (if required, repickled and reinspected), and packed for shipment.

23. Packaged billets were stored for shipment to Richland, Washington.

24. Packaged billets were shipped.

Note: Residues and metal turnings generated throughout the production process were processed (dried, sampled, oxidized, etc.) and returned to FEMP.

Note: According to Breslin and Glauberman (1964, pp. 14–16), after extrusion, hand rolling was performed for 5 minutes.

ATTACHMENT C
SAVANNAH RIVER SITE PRODUCTION

The typical steps for uranium processing for the SRS reactors follow (RMI 1995).

1. Ingots were received from FMPC.

2. Ingots were transferred into storage.

   Note: All onsite transferring of ingots, billets, forgings, and extrusions was accomplished by the use of forklifts, fixtures, and overhead cranes.

3. Ingots were transferred to inspection.

4. Inspection was performed using an overhead monorail, scale, and inspection stand. Inspection included weight and a dimensional and visual inspection.

5. Ingots were transferred back to storage.

6. Ingots were transferred to the salt baths and heated in molten salt to approximately 1,160°F for 75 minutes minimum.

7. Ingots were transferred to the press.

8. Ingots were extruded through the press into tubing. This process included lubrication of press tooling to reduce friction during high-pressure extrusion.

9. The extrusion exited through the press into tubing. This process included lubrication of press tooling to reduce friction during high-pressure extrusion.

10. The extrusions were lowered horizontally into a water-filled quench tank.

    a. Water quenching was begun in approximately 1966. Before this, extrusions were air-cooled.

11. The extrusions were cut into sections on an abrasive saw.

12. The extrusions were transferred to, and run through, the roll straightener.

    a. From 1962 until approximately 1964, the extrusions were lowered horizontally.

    b. In approximately 1964, the hot oil bath was removed and an induction heater was installed in its place. Extrusions were run through the induction heater before the roll straightener. This process continued for approximately 1 year. After this, extrusions were not heated before straightening.

13. Extrusions were stored on the process table.

14. Extrusions were lifted horizontally, then:

    a. From 1962 to approximately 1964, were placed in a vapor degreaser tank to clean the extrusion.

ATTACHMENT D
DOD PENETRATOR PRODUCTION (continued)

b. After approximately 1964, the vapor degreaser tank was no longer used and extrusions were transferred to a water rinse tank, rinsed, and transferred to the packing station.

15. Extrusions were inspected, weighed, and packed for shipment to FMPC.

16. Packaged extrusions were stored before shipment.

17. Packaged extrusions were shipped.

Note: Residues and metal turnings generated throughout the production process were processed (dried, sampled, oxidized, etc.) and returned to FMPC.

ATTACHMENT D
DOD PENETRATOR PRODUCTION

The typical steps for source material processing for DOD penetrators follow (RMI 1995).

1. Ingots were received.

2. Ingots were transferred to storage:

    a. Stored in main plant before 1984.
    b. Stored in main plant and northeast warehouse after 1984.

3. Ingot inspection:

    a. Inspected at inspection station.
    b. Inspected at floor scale.

4. Storage:

    a. Stored in main plant before 1984.
    b. Stored in main plant and northeast warehouse after 1984.

5. Heating:

    a. Salt baths.
    b. Sunbeam furnace (preheat only).
    c. IFSI furnace (preheat only).
    d. Other electric furnaces in, or transferred to the main plant salt bath area.

6. Extrusion on the extrusion press.

7. Postextrusion, which included numerous different processes:

    a. Water quench.
    b. Air cool.
    c. Saw.
    d. Transfer.
    e. Pickle.
    f. Water rinse.
    g. Inspection and weighing at packing station or floor scale.
    h. Storage:

        i. Stored in main plant before 1984.
        ii. Stored in main plant and northeast warehouse after 1984.

    i. Shipment.

ATTACHMENT E
EXTERNAL DATA EVALUATION OF DOSES

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ATTACHMENT E
EXTERNAL DATA EVALUATION OF DOSES (continued)

E.1 PURPOSE

The purpose of this attachment is to provide information to allow dose reconstructors to assign doses that based on worker dosimetry data for Extrusion Plant workers who have no or limited monitoring data. In addition, the data in this attachment should be used to assign doses for gaps in dosimetry records. The data are to be used either in addition to dosimetry data from the site or when monitoring data is lacking for covered employment at the site.

E.2 BACKGROUND

The ORAU Team has evaluated the external dosimetry records for the Extrusion Plant to permit dose reconstructors to complete certain cases for which external or internal monitoring data are unavailable or incomplete. Cases that do not have complete monitoring data could fall into one of several categories:

- The worker was unmonitored and, even by today's standards, did not need to be monitored (e.g., a nonradiological worker).
- The worker was unmonitored but, by today's standards, would have been monitored.
- The worker might have been monitored, but the data are not available to the dose reconstructor.
- Partial information is available, but it is insufficient to facilitate a dose reconstruction.

Some cases without complete monitoring data can be processed based on assumptions and methodologies of evaluating the data that existed for the Extrusion Plant and given at the 50th percentile. For example, many cases in the first category can be processed by the assignment of external and internal doses based on information in the relevant site technical basis documents.

E.3 GENERAL APPROACH

The general approach to the development of this data evaluation for cases without external monitoring data is to assign either 50th- or 95th-percentile doses with the intent that the assigned doses represent, but do not underestimate, the doses that would have been assigned had the worker been monitored.

E.4 APPLICATIONS AND LIMITATIONS

Some Extrusion Plant workers could have worked at one or more other sites in the DOE complex during their employment histories. Therefore, the data in this attachment must be used with caution to ensure that, for likely noncompensable cases, unmonitored external doses from multiple site employments have been overestimated. This typically requires the availability of the recorded doses or information for external coworker dosimetry data for all relevant sites.

The data in this attachment address penetrating gamma radiation and beta or nonpenetrating electron and/or low-energy photon radiation. These doses are to be assigned with a constant distribution in IREP.

ATTACHMENT E
EXTERNAL DATA EVALUATION OF DOSES (continued)

E.5 EXTERNAL DATA DEVELOPMENT

Dosimetry Data
The Extrusion Plant database contains four kinds of dosimetry data: gamma, beta, shallow, and deep. There is also a field for neutron dose that contains a minimal amount of data, which was insufficient for analysis. Gamma and deep are assumed to be equivalent measurements, and shallow dose was assumed to be combination of penetrating gamma and electron dose. The database contains 9,663 records.

All four dosimetry components could apply to one individual in a given year, and the periods could overlap. To perform the analysis, dose for a given person and year was prorated by summing all gamma or beta dose divided by the fraction-of-year the worker was monitored. Values for individuals that were found to be duplicates were removed.

Table E-1 contains an example calculation. This example has a gamma reading and beta reading that covers part of the year and the method for prorating the dose.

Table E-1. Example of analysis of prorating gamma and beta data and normalizing to annual, 330 days of monitoring.

Value	Gamma Dose	Beta dose
Sum of dose within a year	50	100
Calculation	= 50/(330/365)	= 100/(330/365)
Prorated dose	53.94	107.88

Personal Identifiers
The personal identifiers in the database include Social Security Number, last name, and first name. A combination of the first and last name was chosen to distinguish individuals. This is believed to represent the best possible method of separating individuals given the available data. This analysis found that 68 records did not have a last name or a first name; they were excluded from the analysis. Several of the 68 records that were not used did have a Social Security Number but no positive data was recorded for these records.

Limited Data
Adjustments were made to years with gaps or limited data. Table E-2 shows the years that were affected and the number of data points available for the original analysis. To compensate for the shortfall, data from surrounding years were added to the original data.

ATTACHMENT E
EXTERNAL DATA EVALUATION OF DOSES (continued)

Table E-2. Years with limited data for external data analysis.

Year	Data points	Years combined
1965	0[a]	1964–1966
1966	1[b]	1965–1967
1969	2[c]	1968–1971
1970	0[c]	1968–1971
1984	9[d]	1983–1985
1995	0[e]	1994–1996
2000	2[f]	1999–2001

a. Gamma data from 1964 and 1966 to give gamma data for 1965.
b. Beta results were not useable because film badge data was inverted for 1966. Gamma results were useable for 1966. Used beta results from 1964 and 1967.
c. Beta results from 1968 and 1971 to give 1969 and 1970 beta dose.
d. Gamma and beta values from 1983 and 1985 were used for 1984.
e. Gamma and beta values from 1994 and 1996 were used for 1995 because there was inadequate data for 1995.
f. Gamma and beta values from 1999 and 2001 were used for 2000.

E.6 ADJUSTMENT FOR MISSED DOSE

According to OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* (NIOSH 2007b), missed doses are to be assigned for reported zero readings for each monitoring cycle to account for the possibility that doses were received but either not recorded by the dosimeter or not reported by the site. In addition, reported dose values less than one-half the applicable MDLs are to be assigned as missed dose. Annual maximum potential missed doses are calculated by multiplying the number of zero or unrecorded badge readings by the reported dosimeter limit of detection (LOD) and summing the results. These values are used as the 95th-percentile values of a lognormal distribution to calculate the probability of causation, which is determined by DOL. Therefore, in the Interactive RadioEpidemiological Program (IREP), Parameter 1 is equal to the calculated maximum annual missed doses multiplied by 0.5, and Parameter 2 is equal to 1.52. These values represent the GM and GSD, respectively, for each year of analysis. Table E-3 lists the maximum annual missed dose by monitoring period.

Table E-3. Missed external doses (rem)

Monitoring period	Gamma and Beta LOD	Exchange frequency	Maximum potential annual missed gamma and beta dose
1962–1966	0.03	Monthly	0.360
1967–1985	0.02	Quarterly	0.080
1986–1988	0.02	Quarterly	0.080
1989–1992	0.01	Monthly	0.120
1993–2003	0.01	Quarterly	0.040

E.7 Annual Data Evaluation of Dose Summaries

Based on the described information and approaches, Extrusion Plant annual external dosimetry summaries were developed from available data for use in the evaluation of external gamma and beta dose for certain workers who were potentially exposed to workplace radiation but for whom there is no or limited monitoring data.

ATTACHMENT E
EXTERNAL DATA EVALUATION OF DOSES (continued)

These summaries were developed using the following steps:

Step 1. The reported gamma dose was modified for each worker to account for partial years of employment. This permits the dose reconstructor to assign an appropriate prorated dose to account for partial years of employment or potential exposure.

Step 2. One-half of the maximum potential annual missed doses in Table E-3 were added to the reported annual doses from Step 1 (with the exception of reported positive doses, in which case the maximum missed dose was reduced by the dose that corresponded to one badge exchange because it is not possible that all individual badge results were zero if a positive annual dose was reported).

Step 3. The 50th- and 95th-percentile annual data evaluation of gamma doses were derived from the doses from Step 2 by ranking the data into cumulative probability curves and extracting the 50th- and 95th-percentile doses for each year.

Step 4. Table E-4 lists the results of the external dosimetry data analysis. These percentile doses should be used for Extrusion Plant workers with no or limited monitoring data. In general, the 50th-percentile dose can be used as a best estimate of a worker's dose when professional judgment indicates that the worker was probably exposed to intermittent low levels of external radiation. The 50th-percentile dose should generally not be used for workers who were routinely exposed. For routinely exposed workers (i.e., workers who were expected to have been monitored and routinely exposed), the 95th-percentile dose should be applied. For instance, for cases in which routine monitoring data exist but there is a gap in the data, then Table E-4 or E-5 dose can be used to supplement missing quarters or years. The percentile dose should be the one that is consistent with the recorded doses unless there is reason to believe that the worker's job or location in that year differed significantly from the job or location during the years dose was recorded. For workers who are unlikely to have been exposed, dose reconstructors should assign external dose equivalent to the 50th percentile of Table E-4.

Step 5. Table E-5 lists gamma and beta dose values (as described in the steps above) that have been adjusted using the guidance in Section 8.0 of ORAUT-OTIB-0052, *Parameters to Consider When Processing Claims for Construction Trade Workers* (ORAUT 2014c). This guidance is applicable for workers who meet the criteria for CTW.

ATTACHMENT E
EXTERNAL DATA EVALUATION OF DOSES (continued)

Table E-4. Annual external data evaluation of doses modified to account for missed dose (mrem).

Year	Gamma 95th percentile	Gamma 50th percentile	Beta 95th percentile	Beta 50th percentile
1962	811	285	3,314	486
1963	656	376	4,316	1,438
1964	405	240	4,880	1,175
1965	402	225	5,639	1,148
1966	381	205	6,002	1,156
1967	517	167	6,056	1,029
1968	345	110	4,474	750
1969	723	271	3,330	661
1970	539	90	3,330	661
1971	504	185	2,247	576
1972	190	110	1,057	471
1973	297	105	4,422	587
1974	334	110	4,787	1,033
1975	263	90	3,590	541
1976	190	70	2,977	540
1977	321	80	2,451	541
1978	296	150	1,760	882
1979	208	95	1,552	536
1980	231	60	2,553	491
1981	391	150	2,792	692
1982	473	190	4,131	1,324
1983	403	150	2,447	732
1984	297	110	2,222	511
1985	231	110	2,276	391
1986	905	151	3,229	419
1987	238	50	1,023	91
1988	517	90	1,017	40
1989	105	75	60	60
1990	75	60	60	60
1991	75	60	60	60
1992	60	60	60	60
1993	40	40	40[a]	40
1994	53	40	74	40
1995	54	40	74	40
1996	94	40	40	40
1997	40	40	115	40
1998	94	51	85	55
1999	76	40	46	40
2000	70	40	40	40
2001	65	40	40[a]	40
2002	40	40	40	40
2003	56	40	40[a]	40
2004	55	35	40[a]	40

a. 95th percentile calculated lower, set equal to 50th percentile.

ATTACHMENT E
EXTERNAL DATA EVALUATION OF DOSES (continued)

Table E-5. Annual CTW dose based on data evaluation of worker dosimetry (mrem).

Year	Gamma 95th percentile	Gamma 50th percentile	Beta 95th percentile	Beta 50th percentile
1962	1,069	333	4,573	614
1963	853	460	5,977	1,948
1964	501	270	6,766	1,579
1965	497	249	7,829	1,541
1966	467	221	8,337	1,552
1967	712	221	8,466	1,429
1968	470	142	6,251	1,038
1969	1,000	367	4,649	913
1970	743	114	4,649	913
1971	693	248	3,134	795
1972	255	142	1,468	648
1973	404	135	6,178	809
1974	455	142	6,690	1,434
1975	356	114	5,014	746
1976	254	86	4,156	744
1977	437	100	3,419	746
1978	402	198	2,452	1,223
1979	279	121	2,160	739
1980	311	72	3,562	676
1981	535	198	3,896	957
1982	651	255	5,772	1,841
1983	552	198	3,413	1,013
1984	403	142	3,099	704
1985	311	142	3,175	535
1986	1,255	199	4,509	575
1987	321	58	1,420	115
1988	711	115	1,412	40
1989	125	83	60	60
1990	82	60	60	60
1991	83	60	60	60
1992	60	60	60	60
1993	40	40	40[a]	40
1994	62	40	92	40
1995	63	40	92	40
1996	119	40	40	40
1997	40	40	148	40
1998	119	60	107	64
1999	95	40	53	40
2000	86	40	44	36
2001	79	40	40[a]	40
2002	44	40	40	40
2003	67	44	40[a]	40
2004	65	37	40[a]	40

a. 95th percentile calculated lower, set equal to 50th percentile.