Mallinckrodt

C-T Project

Decommissioning Plan

Part I

November 11, 1997

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1. GENERAL INFORMATION

1.1 INTRODUCTION

Mallinckrodt Chemical, Inc. (Mallinckrodt) has had a license, Number STB-401, docket number 40-6563 , since 1961, issued by the Atomic Energy Commission, and later the NRC, to extract columbium and tantalum from natural ores and tin slags. This license is currently a possession-only license. A license was required because the ores and the by-products of processing contained uranium and thorium isotopes. The Columbium-Tantalum (C-T) decommissioning project is designed to decontaminate and decommission the areas of Mallinckrodt's St. Louis Plant involved with the processing, storage, and handling of radioactive materials associated with the extraction of columbium and tantalum from ores and slag. The C-T production plant and associated support facilities are located within the boundaries of the Mallinckrodt site.

The ultimate goal of the C-T decommissioning project is to remediate the radiological constituents associated with C-T production to the extent required to terminate license STB-401. Remediation of radiological constituents in other areas of the St. Louis Plant is being performed under the Department of Energy's (DOE's) Formerly Utilized Sites Remedial Action Program (FUSRAP).

The C-T Project Decommissioning Plan will be submitted in two parts because the C-T decommissioning project will be conducted in two phases. Phase I will decommission the buildings and equipment to the extent that whatever remains on-site will be released for *unrestricted use* based on an industrial use scenario. Phase II will complete the decommissioning of the building slabs and foundations, paved surfaces, and all subsurface materials to the extent that they can be released for *restricted use*.

This document is the Phase I C-T Decommissioning Plan (hereafter, Phase I Plan), describing only the activities involving above-grade decommissioning of the buildings and equipment. The Phase II C-T Decommissioning Plan (hereafter, Phase II Plan) will cover the decontamination and decommissioning of the grade-level and below-grade facilities including the unreacted ore (URO) burial pits, pavement and building slabs, wastewater basins, and other subsurface materials. The Phase II Plan will be submitted later under separate cover.

The two-phase decommissioning plan provides Mallinckrodt the flexibility necessary to deal with the inherent complexities of the site and to take immediate steps to reduce the amount of residual radioactive material at the St. Louis Plant. The advantages of a two-phase decommissioning plan are as follows:

- A two-phase decommissioning plan follows a logical sequence. The removal of above-grade contaminated equipment and the decommissioning of the buildings in the C-T project site must be accomplished prior to addressing the subsurface material.

- There are limited staging areas available for use during decommissioning. The St. Louis Plant is an ongoing operational facility which manufacturers a variety of bulk pharmaceuticals and specialty chemicals. Manufacturing will continue during and after the decommissioning of the C-T project site. The decommissioning project must be carefully planned and staged to allow the ongoing operations to continue with minimal impact.

- The approval of the Phase I Plan allows physical remediation work to commence in those areas where on-site workers have the greatest chance of being exposed to residual radioactive material. Building surveys indicate that the majority of residual radioactive material is fixed and, although there are no immediate health and safety issues, the primary process buildings and equipment have not been in use for several years and are starting to physically deteriorate. Subsurface radioactive materials, by comparison, are not easily accessible to on-site workers due to the physical barriers presented by the paved surfaces and building slabs.

- The goal of the Phase I Plan is straightforward: decommission the facility for release for unrestricted industrial use. Approval of the Phase I Plan will allow Mallinckrodt to make progress toward license termination in a timely manner.

- Plan approval and implementation in phases will allow progress toward decommissioning, and allows decommissioning to proceed while NRC develops guidance for subsurface remediation.

Submittal of the Phase I Plan has been preceded by more than two years of planning and site characterization. These efforts significantly enhanced Mallinckrodt's knowledge regarding the current radiological status of the site. The recent characterization results have been combined with historical knowledge and previous characterization efforts to provide a more complete understanding of the radiological status of the C-T project area. The characterization results for the buildings and equipment are incorporated into the Phase I Plan. Characterization results for the unreacted ore (URO) burial pits, pavement and slabs, wastewater basins, and subsurface materials will be presented in the Phase II Plan.

1.2 LICENSE INFORMATION

The name of the STB-401 licensee is Mallinckrodt Chemical, Inc., a subsidiary of Mallinckrodt Inc. which has its principal offices at 7733 Forsyth Blvd., St. Louis, MO 63105-1820. Mallinckrodt Chemical, Inc. is responsible for site operations. The mailing address of the St. Louis Plant is Mallinckrodt & Second Streets, P.O. Box 5439, St. Louis, MO, 63147.

1.3 SITE DESCRIPTION

The St. Louis Plant has been in operation since 1867 and has produced a wide range of products including metallic oxides and salts, ammonia, organic chemicals, and various uranium compounds under contract to the Manhattan Engineering District and the Atomic Energy Commission (MED-AEC). The plant currently produces a variety of products for the food, drug, cosmetic, pharmaceutical, and specialty chemical industries.

1.3.1 The St. Louis Plant

The St. Louis Plant is a 43 acre site located adjacent to the west bank of the Mississippi River in an area zoned and developed for industrial use (Figure 1-1). The plant is bounded by: Angelrodt Street on the south; Salisbury Street and the McKinley Bridge on the north; Broadway Street on the west, and Wharf Street on the east, except for the small Plant 7E area just east of the railroad tracks.

The St. Louis Plant currently contains more than 50 buildings in an area of approximately twelve city blocks. The St. Louis Plant has traditionally been subdivided into areas, called Plants, based on similar operations being performed. These Plants have been named historically in one of two ways: either using a letter or a number from 1 to 10 (with or without a letter), such as Plant 5 or Plant 7E. Additionally, areas of particular operations could be described by a process-related name, e.g. C-T Plant. Currently the St. Louis site is divided into numbered plant areas. This Phase I Plan will use either descriptive names or numeric names for these Plant areas. Individual buildings within each Plant are numbered, lettered, or both. A plot of the current arrangement of Plants within the St. Louis Plant is shown in Figure 1-2.

In addition to production facilities, the St. Louis Plant includes support facilities such as maintenance shops, research and quality control laboratories, warehouses, steam boilers, inactive wastewater neutralization basins, and a permitted facility for drum storage of hazardous waste.

1.3.2 The C-T Process and Support Buildings

The former C-T Process and Support Buildings within the St. Louis Plant, shown in Figure 1-3, encompass 21 buildings on approximately 4.2 acres, primarily within Plant 5 but also within portions of Plants 1, 3, 6, 7N, 7W, and 8. A more detailed description of the processing history of the various buildings is given in section 1.4 of this plan.

1.3.3 Physiography

The St. Louis Plant site is located in the Oak-Hickory-Bluestem Parkland section of the Prairie Parkland Province (Bailey 1980) and within the Florissant Basin (Lark 1992). The area adjacent to the St. Louis Plant is completely developed, with no pre-settlement vegetation existing. The ground upon which the St. Louis Plant is situated was built up

using cinders, sand, dirt, construction debris, and gravel. Only hardy vegetation such as annual bromegrass and mustard survive under these soil conditions.

Numerous buildings and facilities cover a large portion of the St. Louis Plant and the majority of the area is paved with asphalt or concrete. Mallinckrodt restricts access to the site 24 hours a day to employees, contractor construction workers, and authorized visitors.

Land use within a 1.6-km (1-mi) radius of the St. Louis Plant reflects a mixture of commercial, industrial, and residential uses. Commercial or industrial properties in the vicinity of the St. Louis Plant include the McKinley Iron Company, Thomas and Proetz Lumber Company, and PVO Foods (closed). Three other nearby properties are owned by the Chicago, Burlington, and Quincy Railroad, the Norfolk and Western Railroad, and the St. Louis Terminal Railroad Association, all of which have rail lines that transect the St. Louis Plant from north to south.

The property located between the St. Louis Plant and the Mississippi River is owned by the city of St. Louis. The land is undeveloped and unfenced. The St. Louis Plant, its vicinity properties, and most properties east of Broadway Street and south of Merchants Bridge are zoned "K" (unrestricted district) by the City of St. Louis. This industrial zone allows all uses except residential, provided that no other city codes are violated. Some of the uses within this zone allowed by the City of St. Louis Zoning Code, Section 26.60, under conditional use permits, are acid manufacture, petroleum refining, and stockyards. It is anticipated that this area will remain in industrial use.

The area where the St. Louis Plant is located is within Census Tract 1267. The residential population within Census Tract 1267 was 2,867 in 1990, and the total residential population within 1.6 km (1 mile) of the St. Louis Plant was 10,054, approximately 2.5 percent of the population of the City of St. Louis. The number of occupied dwellings within this 1.6 km radius was 4,710 for an average occupancy of 2.1 people per dwelling. The closest residential dwelling is located on North Broadway Street, 61 m (200 ft.) southwest of the southwestern corner of the St. Louis Plant (City of St. Louis Community Development Agency 1992). Most of the population in this area is located west of I-70.

The major surface water bodies in the area are the Mississippi, Missouri, and Meramec Rivers, which supply 97 percent of the 4.5 billion liters (1.2 billion gallons) per year of drinking and industrial water for the St. Louis area (BNI 1989). The area where the St. Louis Plant is located is at River Mile 182.5, on the western bank of the Mississippi River (Figure 1-1), 20 km (12.7 miles) downstream from the confluence of the Mississippi (River Mile 195.2) and Missouri (River Mile 0) Rivers. Storm runoff from the St. Louis Plant flows via a combined sewer system to the Metropolitan Sewer District (MSD). All City of St. Louis municipal water intakes are located upstream from the underground drainage system that drains the St. Louis Plant.

The Mississippi River at the St. Louis gauging station has a drainage area of approximately 1.8×10^6 km^2 (700,000 sq. miles). The average flow for a 114-year period is 5×10^6 m^3/s

[177,000 cubic feet per second (cfs)]. The minimum flow recorded in this period is 5×10^5 m^3/s (18,000 cfs), and the maximum measure flow is 3×10^7 m^3/s (1,019,000 cfs). Although flooding has occurred every month of the year, higher flows are frequently associated with snow melt and heavy rains in spring. Lowest flows typically occur during December or January. A levee constructed in 1964 on city property east of the plant protects the plant from Mississippi River floodwaters. Flooding information for the Mississippi River is shown in Table 1-1.

Table 1-1
Flood Frequency Distribution for the Mississippi Ri[illegible] St. Louis, MO (1934-1979)

Recurrence Interval (years)	Discharge Level (cubic feet per second)	Exceedence Probability (%)
1.01	146,000	0.99
1.05	223,000	0.96
1.11	273,000	0.90
1.25	341,000	0.80
2.00	490,000	0.50
5.00	651,000	0.20
10.0	735,000	0.10
25.0	820,000	0.04
50.0	872,000	0.02
100	915,000	0.01
200	952,000	0.01

1.3.4 Climatology and Meteorology

The climate of the St. Louis area is characterized as warm and moist in the summer and cold and dry in the winter. The region is dominated by warm, moist maritime tropical air masses, which flow northward from the Gulf of Mexico region, and by cooler, drier polar air masses, which drift down from the Canadian Provinces. The annual average of monthly high and low temperatures are 31°C and -5°C (88°F and 23°F), respectively. The area averages 91 cm (36 inches) per year in total precipitation (i.e., rainfall plus water content of melted snowfall). Average annual snowfall is approximately 66 cm (26 inches).

Tornadoes are the most common form of severe weather occurring in the region. From 1916 through 1985, 52 recorded tornadoes occurred in the St. Louis metropolitan area. In 1990, Missouri had 31 storms in 14 days, most of them in May and June. Based on the record between 1953 and 1990, Missouri is ranked seventh nationally in the occurrence of tornadoes and averages 11 tornado and 27 storm days per year (NOAA 1990).

1.3.5 Geology and Hydrology

The St. Louis Plant is underlain by two unconsolidated soil units and one bedrock unit. Figure 1-4 is a typical cross-section through the site. The upper unconsolidated unit is mostly fill material and varies between 12 and 30 feet in thickness. The lower unconsolidated unit varies from 0 to 60 feet in thickness and consists of relatively impermeable alluvial silt and clay above relatively permeable sandy alluvial sediments. The deepest unit is limestone bedrock.

The subsurface hydrologic conditions in the Plant 5 area of the St. Louis Plant are known from the installation of over sixty exploratory borings in Plant 5 during previous investigations. Selected borings, monitoring wells, and groundwater level contours in Plant 5 are shown on Figure 1-5. A typical subsurface profile of Plant 5, developed from borings is shown in Figure 1-6 where four stratigraphic units exist.

The topmost unit is 7-18 feet thick and consists of brick, clay, coal slag, cinder, concrete, construction rubble, glass, and sand fill materials. A perched groundwater unit occurs within the fill in Plant 5 at depths ranging from approximately three to nine feet below ground surface. Groundwater elevation measurements performed between 1982 and 1989 indicate that the perched groundwater flows generally to the northeast with a horizontal hydraulic gradient of approximately 0.006 ft/ft.

A relatively impermeable fine-grained alluvial unit, 20-30 feet thick, is found below the fill unit. The fine-grained alluvial unit consists of interbedded silty clay, clayey silt, and sandy silt. Representative hydraulic and geotechnical properties of this unit were evaluated by Department of Energy (DOE) under FUSRAP. An *in-situ*, changing head test performed in this unit in Plant 7 yielded a horizontal hydraulic conductivity of 9.9×10^{-6} cm/s. DOE also collected fourteen undisturbed samples of this unit from various boreholes across the St. Louis Plant and tested them for laboratory permeability, cation exchange, and geotechnical parameters. The vertical hydraulic conductivity ranged from 4×10^{-4} to 1×10^{-6} cm/s with a geometric mean of 1×10^{-5} cm/s. Cation exchange capacities ranged from 70 to 200 meq/100 g, with an average value of 39 meq/100 g.

The sandy alluvial unit beneath Plant 5 ranged from zero to ten feet thick, and consists of fine-to-coarse sand. Groundwater in this unit may be confined or semi-confined by the overlying relatively impermeable fine-grained unit. The groundwater potentiometric surface occurs at depths of about 10 to 35 feet below ground surface. Groundwater in this unit flows eastward towards the Mississippi River with a horizontal hydraulic gradient ranging from 0.01 to 0.02 ft/ft. See Figure 1-7 for a typical groundwater contour map.

The limestone bedrock surface beneath Plant 5 occurs at depths ranging from approximately 30 to 55 feet and slopes gently towards the Mississippi River (Figure 1-5).

Groundwater in the sandy alluvial unit is locally saline and generally very hard, with high iron and manganese content. Groundwater found in the underlying bedrock is generally saline and non-potable. The groundwater found beneath the St. Louis Plant and surrounding industrial areas is not used as a source of drinking water. There are no known drinking water wells in the vicinity of the plant. The St. Louis Plant is within the City of St. Louis, which has an established public water supply system. This system treats and distributes water obtained from the Mississippi River north of the St. Louis Plant.

1.4 PLANT DESCRIPTION AND OPERATION

Various past production operations at the St. Louis Plant have involved radioactive materials:

- From 1942 to 1957, Mallinckrodt extracted uranium from ores and produced purified uranium and uranium compounds for national defense purposes, first for the MED, and later for the AEC.

- From 1956 to 1960, Mallinckrodt extracted columbium, tantalum, uranium, thorium, and rare earth elements from euxenite mineral ore and converted the ore into useful products for delivery to the AEC and the General Services Administration (GSA) as part of the Defense Materials Procurement Program. The euxenite operation was performed under AEC source material license No. R-226 which expired in 1960. It is estimated that a total of 95 Curies (Ci) of natural uranium (U-238, U-234, and U-235) and 10 Ci of natural thorium (Th-232, Th-228) were extracted from the ore during this time period.

- From 1956 to 1977, Mallinckrodt subdivided small quantities of uranyl and thorium salts for resale under AEC/NRC license SUC-872. A report of Mallinckrodt's final radioactivity survey for this license was submitted to NRC on December 13, 1979.

- From 1961 to 1985, Mallinckrodt extracted columbium and tantalum from ores and tin slags containing low levels of natural uranium and thorium, under AEC/NRC rare earth extraction license No. STB-401. A two-month trial production run was performed in early 1987, but C-T extraction has not been performed since that time. A total of approximately 6 Ci of natural uranium and 19 Ci of natural thorium isotopes were contained in the ores and tin slags processed during the period of columbium and tantalum extraction.

1.4.1 MED-AEC Operations

In April 1942, Mallinckrodt, then called Mallinckrodt Chemical Works (MCW), was contracted to extract uranium from ore concentrates for eventual use in the first self-sustaining nuclear chain reaction in the graphite reactor being built at the University of

Chicago. The initial contract was signed on July 20, 1942. Within 50 days of accepting the assignment from the War Department, MCW began producing highly refined uranium dioxide (UO_2) for the CP-1 pile reactor at the rate of 1 ton per day in Plant 2, supported by research activities in Plant 1. The UO_2 was also shipped to another MED site for reduction to metallic fuel for the reactor. The intermediary products, uranyl nitrate and uranium trioxide, were produced both as intermediaries to the production of uranium dioxide and as final products. A process to convert UO_2 to uranium tetrafluoride was begun as a batch process in 1942. A process to convert uranium tetrafluoride to uranium metal started in 1943. This activity was performed in process buildings located on the east side of Broadway Street, immediately west of Plant 5. At that time, this area was designated as Plant 4. This designation is no longer used for this area. The company was the sole supplier of uranium compounds for the Manhattan project well into 1943, and provided high purity uranium products for the duration of the war.

In 1945, the Destrehan Plant (Plants 6, 6E, and 7 north and south of Destrehan Street) was built to process pitchblende ore and to increase the capacity of the refinery. Plant 6 production began in 1946. In 1950 and 1951, the MED-AEC facilities in Plants 1 and 2 were partially decommissioned and finally decommissioned in the early 1960s. In 1958, the Destrehan plant was put on standby, and uranium processing was transferred to the MCW Weldon Spring Plant. In 1960 and 1961, the MED-AEC facilities in the former Plant 4 area (part of the area now known as Plant 10) and the Destrehan Plant were decommissioned.

Figure 1-8 illustrates the areas at the St. Louis Plant site that were used for MED-AEC production. The St. Louis Plant processed approximately 50,000 tons of uranium products during the 1942-1958 MED-AEC operations. Based upon the estimated 50,000 tons of uranium products produced from the ore concentrates and pitchblende ore, it is estimated that the minimum radioactivity throughput was approximately 30,000 Ci of uranium isotopes and 10 Ci of thorium isotopes.

DOE, under FUSRAP, had responsibility for remediating radioactive and chemical contamination in the areas of the St. Louis Plant that formerly housed MED-AEC operations. However, the U.S. Army Corps of Engineers (COE) now has responsibility for these activities because in October, 1997, Congress transferred responsibility for FUSRAP implementation from DOE to COE. FUSRAP was created to identify and control or remediate sites where residual radioactivity remains from activities conducted under contract to MED and AEC during the early years of the nation's atomic energy program. Some facilities that produced radioactive materials for commercial sale are also included under FUSRAP at the direction of Congress. The COE is responsible for the cleanup of both radioactive and hazardous chemical contamination at the St. Louis Plant with oversight by the U. S. Environmental Protection Agency (EPA). These responsibilities are outlined in a Federal Facilities Agreement (FFA) negotiated by EPA Region VII and DOE in June of 1990 (Docket No. VII-90-F-0005). It is our understanding that the FFA will be amended to transfer these responsibilities to COE. The FFA further defines the conditions dictated by EPA to manage remediation at St.

Louis. The document creates broad obligations for clean up of all residual waste from uranium processing, including such waste that might have mixed or commingled with other hazardous material substance at the site.

FUSRAP is responsible for the remediation of Building 25, 50, 51, 51A, 52, 52A, 100, 116, 117, 219, 700, 704, 705, 706, 707,and 708, as illustrated in Figure 1-9. FUSRAP has already demolished all of these structures with the exception of Building 25. Some Plant 6 and 7 buildings and adjacent open areas were used to support C-T manufacturing following their decontamination and release to Mallinckrodt by the AEC in the early 1960s. The buildings in Plants 6 and 7 that supported C-T are summarized in Table 1-2. Buildings and soil in these areas contain substantial volumes of residues from uranium refining and are therefore subject to remediation by DOE under FUSRAP. All Plant 6 and 7 buildings previously used for uranium processing, including those that subsequently supported C-T manufacturing, are being demolished by DOE. Demolition will be completed in October 1997. DOE will remediate Plant 6 and 7 soils over the next several years.

1.4.2 Euxenite Operations

From approximately 1956 to 1960, MCW extracted columbium, tantalum, uranium, thorium, and rare earth elements from euxenite mineral ore for national defense purposes. The operation was performed in connection with the Defense Materials Procurement Agency and AEC. MCW was a subcontractor to Porter Brothers Corporation and performed the euxenite processing under an AEC source material license: R-226. The license expired March 1, 1960. Former euxenite process and support areas are shown in Figure 1-10. (Please note that some of these buildings no longer exist.)

Columbium (now known as niobium) and tantalum are naturally occurring metallic elements with various uses. Columbium is used as an alloying agent in carbon and alloy steels and in nonferrous metals to improve material strength, and is also used in an alloy form for high temperature structural applications. Tantalum and its alloys are used in electronic components, chemical process equipment, and high temperature aircraft and missile parts.

1.4.3 C-T Production

In 1961, after the end of the euxenite operation, MCW began production of columbium and tantalum compounds from natural and synthetic ores and tin slags at the St. Louis Plant under an AEC rare earth extraction license, number STB-401 issued August 23, 1961. From 1961 to approximately 1974, Mallinckrodt purchased feed materials from Malaysia, Thailand, and other countries. From approximately 1975 to 1985, feed materials consisted of a blend of upgraded synthetic tin slag concentrates and natural columbite mineral ores and were processed under a long-term contract with National Research Corporation, Inc. In February 1986, Mallinckrodt informed the NRC that National Research Corporation, Inc. did not renew its contract and that the C-T Process

and Support Areas would be placed on standby. Following a one-year shutdown, a two-month pilot production run was performed in early 1987 to explore new business opportunities. The pilot run processed approximately 20,000 pounds of tin slag from Thailand. The C-T Process and Support Areas returned to standby mode following the trial run and no C-T operations have occurred since that time. On July 12, 1993, NRC amended the license to a possession-only license for D&D and license termination.

1.4.4 Buildings Supporting C-T Production

C-T production and support buildings are listed in Table 1-2 and displayed in Figure 1-3. Although C-T production occurred in Plant 5, support activities were conducted in portions of Plants 1, 3, 6, 7, and 8. Process building 238 in Plant 5 was constructed for use by the euxenite operations, while other process buildings in Plant 5 were constructed specifically for the C-T operation. Selected buildings and areas in Plants 6 and 7 were used to receive and store feed materials and drummed URO waste. Some URO was buried in the western portion of Plant 6 in 1972 and 1973 in conformance with 10 CFR 20.304 (Figure 1-11).

Mallinckrodt began construction of Plant 5 in 1947 with the construction of Building 200 and 201 along Angelrodt Street at the southern end of the block. These buildings have been used for several types of processing and are still in operation today processing both organic and inorganic materials. New sewers were installed as Plant 5 was being developed. Wastewater was conveyed in underground sewers to the northwest corner of Plant 7. From Plant 7, an underground sewer carried the Plant 5 effluent east and connected to the sewer and outfall system constructed to support the MED-AEC Destrehan Street Facility. In the early 1970s, a pair of wastewater neutralization basins were constructed in the northwest corner of Plant 7. These basins were used until 1993.

TABLE 1-2
C-T PROCESS AND SUPPORT BUILDINGS

Building No. and Location	C-T Process and Support Areas
Plant 1 Area	
Building 25 (FUSRAP)*	Laboratory
Plant 3 Area	
Building 62	Change Rooms (Lockers)
Plant 5 Area	
Building 213	Change and Break Rooms
Building 214	Transformer/Switchgear Room
Building 235	Feed Material/Storage (East Half)
Building 236	Feed Material Storage
Building 238	C-T Ore Grinding/Dissolving/T Processing
Building 246A	Offices
Building 246B	Solvent Extraction Process
Building 247A	C-T Solvent Extraction/Product Storage
Building 247B	Columbium Filtration and Drying
Building 248	Columbium Filtration/Drying/Calcining
Building 250	Offices and Quality Control Labs
Plant 6 Area	
C-T Incinerator	C-T Incinerator
Building 116 (FUSRAP)	Receipt/Unloading of C-T Ore
Building 117 (FUSRAP)	URO Drum Preparation and Staging
Plant 7 Area	
Building 700 (FUSRAP)	Storage of Tin Slag Feed Material
Building 704 (FUSRAP)	URO Drum Storage
Building 705 (FUSRAP)	C-T Ore Storage
Building 706 (FUSRAP)	C-T Ore Storage
Building 708 (FUSRAP)	Storage of Tin Slag Feed Material
Plant 8 Area	
Building 90/91	Maintenance Areas

* (FUSRAP) These buildings are being addressed under FUSRAP.

Specific buildings that supported C-T production are described below.

- Buildings 213 and 236 were constructed in 1953. Building 213 was originally used as a locker and break facility for Plant 5 operations, including C-T. It now houses plant utility operations, as well as the break room. Building 236 is currently used as a maintenance shop. At one time, C-T product was dried in tray dryers in Building 236.

- Building 238 was constructed in 1954 to house the euxenite process, a predecessor to the C-T operation, then was used for C-T operations.

- Building 235 was constructed in 1959. Building 235 was used as a returned-goods warehouse and at one time was used to store C-T feed materials and URO. All areas of Building 235 have been renovated for manufacturing and associated support activities.

- Buildings 246A and 246B were built in 1961 as Building 238 was being converted for C-T processing. C-T operations offices were located in Building 246A. The original C-T extraction operations were performed in Building 246B.

- Building 250 was constructed in 1967 to support C-T and other manufacturing operations. The C-T quality control and research laboratories were located in Building 250, as were manufacturing and laboratory facilities for other Mallinckrodt products. Prior to Building 250 construction, C-T laboratories were located in Building 25 (in Plant 1). Building 25 was also used as a laboratory to support AEC-MED operations and will be remediated under FUSRAP.

- Buildings 247A, 247B, and 248 were constructed in 1967 to house expanded C-T extraction and finishing operations.

All Plant 5 streets are paved with asphalt or concrete. Paved streets were installed to serve manufacturing and warehouse buildings as they were constructed.

1.4.5 C-T Process Description

A generalized C-T process flow diagram is shown in Figure 1-12. The feed materials included ore, slag, sodium hydroxide, hydrofluoric acid, sulfuric acid, aqueous ammonia, methyl isobutylketone, hydrochloric acid, and potassium chloride. Products from this process included tantalum oxide, potassium fluotantalate, and columbium oxide. Columbium and tantalum oxides and salts were produced in a batch process that included five major steps:

Step 1: Feed materials were received by truck in burlap bags and drums. Usually, the bags were placed in drums or boxes for storage. The drums or boxes were stored in Plant 6 and 7 prior to forklift transport to the ore staging area in Plant 5, where ore batches were selected.

The ore (feed material) was arranged into feed batches in the ore staging area east of Building 245. Ore was also staged on the other paved streets in Plant 5. The feed material was ground into fine grained slurry in the ball mill room (Building 238 annex) using a wet milling process. The slurry was then pumped into boil-down tanks where excess water was evaporated.

Due to the value of columbium and tantalum, the burlap ore bags were incinerated, and the ash was recycled back to the process to recover columbium-tantalum. The incinerator was originally located west of Building 248. In 1980, the incinerator was installed in its present location west of Building 101 in Plant 6.

Step 2: The ore slurry was pumped into large rubber-lined acid-dissolving tanks in building 238. Hydrochloric, sulfuric, and hydrofluoric acids were used during the tin slag processing. Hydrofluoric and sulfuric acids alone were used in dissolving/leaching columbite and tantalite ores and synthetically upgraded tin slags.

Step 3: The acid C-T mother liquor was decanted from the unreacted ore by mixing and settling. A flocculating agent was utilized to enhance separation. The decanted liquor was filtered and pumped to Building 247 for Step 4 processing. Initially, the URO acid slurry was filtered on a plate and frame press, washed with water and the cake discharged to the plant sewer system. Between 1975 and 1980, the URO press cake was drummed for future use or disposal. Beginning in 1980, the stored URO was reprocessed by slurrying in Step 4 raffinate in order to form a homogeneous mixture. This mother liquid was then decanted. The URO slurry was diluted with water, neutralized with caustic or ammonia, dewatered in a filter press, dried in a pancake dryer and drummed for disposal. All of the URO processing was performed in Building 238.

Step 4: The acid mother liquor was subjected to a two-series extraction/purification process. In the first series, the C-T mother liquor was extracted using methyl isobutylketone (MIBK) and sulfuric acid. This generated a C-T-MIBK stream (organic end) and a raffinate (aqueous end) consisting of hydrofluoric and sulfuric acids, salts, and residual URO material. In the second series, the C-T-MIBK stream was contacted with water in a second extractor to separate the columbium from the MIBK phase. This yielded a tantalum-MIBK stream (organic end) and a fluocolumbic acid stream (aqueous phase). MIBK was removed from the tantalum-MIBK stream by steam stripping, yielding a fluotantalic acid stream. The first series raffinate stream was used to wash columbium and tantalum acid liquors from the URO, reused as feed liquors for the solvent extraction step, or neutralized with ammonia and discharged to the sewer. These process steps were performed in Buildings 246B and 247B. Solvent extraction was not utilized until approximately 1964. Prior to this time, the columbium and tantalum were separated from the mother liquor by precipitation.

Step 5: The primary C-T process products were columbium oxide and potassium fluotantalate salt. Approximately five percent of the tantalum product was produced as tantalum oxide. Columbium and tantalum oxides were precipitated from their respective product streams (fluocolumbic acid, fluotantalic acid) by addition of ammonia. Finishing steps included filtration, drying, and calcining. Columbium oxide precipitation and finishing were performed in Building 248. The potassium fluotantalate salt was precipitated from the fluotantalic acid stream by addition of potassium chloride, separated in a centrifuge, and dried in tray dryers. These steps were conducted in Building 238.

1.4.6 C-T Operations Environmental Controls and Waste Management

Air emissions from C-T processing in Building 238 were controlled using a dust collector in the ball mill area, two acid fume scrubbers and an ammonia scrubber. Certain decanted liquors, raffinate from the solvent extraction step, and C-T filtrate from the final process steps were discharged to the plant sewer in accordance with NRC license conditions and St. Louis Metropolitan Sewer District (MSD) permit.

Prior to 1975, most of the URO was discharged to the sewers in accordance with License No. STB-401 and other regulatory requirements, except for some drums of URO that contained higher than normal levels of tantalum. These drums were held on-site for reprocessing but were not reused due to the high cost of recovery. As a result, approximately 300 cubic yards of drummed URO were buried on-site (Figure 1-11) in 1972 and 1973 in conformance with 10 CFR 20.304. Starting in 1975, all URO solids were drummed and stored on site. Processing of the stored URO started in 1980 using the new URO neutralization, filtration and drying system. All URO processed through this system was drummed for transfer to a licensed uranium mill or a licensed low level radioactive waste disposal facility.

1.4.7 Historical Occurrences

Documentation does not describe any accidental releases of radioactive material from the time period in which the C-T process site was operational. Therefore, interviews were conducted to obtain historical information from past and present employees involved in C-T operations. The following events were described in these interviews.

- Raffinate Tanks - During the operational period of the C-T process site, raffinate tanks located north of Buildings 246 and 247 overflowed on more than one occasion. In the event of an overflow of the main tanks, raffinate was diverted to a backup tank. However, in some instances, the backup tank did not contain all materials.

- Steam Jet Emissions - The entrained liquid from a high pressure vacuum steam jet on the southwest roof of Building 238 occasionally sprayed into the air potentially contaminating roofs of surrounding buildings.

1.5 APPROACH TO C-T SITE DECOMMISSIONING

1.5.1 Site-Specific Considerations and Objectives

Based on the information presented in the prior sections, a site-specific characterization approach was proposed. The proposal was based on facts related to the St. Louis site that are important considerations for the C-T site decontamination and decommissioning pursuant to license No. STB-401:

- The areal extent and scale of MED-AEC operations as shown by historical records, maps, and photographs, as well as the radioactivity throughput, indicate that virtually all of the radiological contamination outside the well defined Plant 5 C-T process and support areas is derived from MED-AEC operations and is unrelated to the NRC licensed operations.

- The areal extent of the former MED-AEC process and support areas surrounding Plant 5 (see Figure 1-8), and routes of contamination dispersal (e.g. loss during materials transport, atmospheric deposition, and flooding) suggest that MED-AEC contamination may have migrated into Plant 5.

- The DOE has been investigating and remediating residual contamination from the MED-AEC operations under FUSRAP and in accordance with the terms of a FFA. The COE will propose a remedial action plan for the site and in coordination with the EPA, Missouri Department of Natural Resources (MDNR) and with public notice and comment will select and implement the appropriate remediation activities. The COE will prepare a remediation plan which will be implemented over the next several years.

The presence of MED-AEC contamination of the St. Louis plant, and the overlap of the Plant 6 and 7 MED-AEC process areas and C-T support areas indicate that the plans for the C-T site characterization and decommissioning must seek to limit the potential conflict, duplication, and inconsistency between Mallinckrodt's responsibilities under NRC license No. STB-401 and those of the COE under the FFA and FUSRAP. Therefore the objectives considered in developing the Phase I Plan are as follows:

- Develop decontamination and decommissioning goals and approaches that are protective of human health and the environment and consistent with applicable NRC, EPA, and state regulations.

- Develop practical, cost-effective decommissioning approaches that limit potential interference and duplication of effort with remediation activities pursuant to the FFA and FUSRAP.

- Limit major inconsistencies with the remedial goals and approaches established by COE, EPA, and MDNR for remedial work under the FFA and FUSRAP.

1.5.2 Proposed Approach and Rationale

To achieve the objectives listed above, Mallinckrodt proposes an approach to C-T decommissioning pursuant to NRC License No. STB-401 that addresses the unique mix of regulatory and technical issues at this facility. The approach is designed to protect human health and the environment consistent with NRC goals and accommodate FUSRAP activities.

The proposed scope of the decommissioning program involves all C-T process and support areas in Plant 5 (except as described in the following paragraph) and certain C-T support areas in Plants 3, 6, 7, and 8. Although the presence of MED-AEC contamination cannot be precluded, the contamination in these areas appears to be primarily related to C-T operations.

DOE has identified radiological contamination in Plant 10 (the former MED-AEC Plant 4 area) and in areas along Second Street between Angelrodt and Salisbury Streets. Surficial and subsurface contamination at the northwest corner of building 250 has been documented by DOE. Mallinckrodt has also identified radiological contaminants along the west side of building 250. The area was used by Mallinckrodt as an employee parking lot prior to building 250 construction. Contamination of these areas including the soils beneath Buildings 250 and 240 and other non-C-T Plant 5 buildings is believed to be the result of MED-AEC activity (MED activities in Plant 4 and materials transport along streets and tracks west of Plant 5), and therefore subject to remediation under FUSRAP and the FFA. These areas are therefore not addressed in this Phase I Plan.

Certain of the C-T support areas in Plants 1, 6, and 7 contain widespread MED-AEC contamination that is subject to remediation under the FFA and FUSRAP. Since these C-T support areas will be addressed pursuant to the FUSRAP program and the FFA, they will not be included in the Phase I and II Plans, including the C-T incinerator and pad and subsurface burials near Building 101.

1.6 CHARACTERIZATION STATUS

The characterization status of the C-T project is based on historical characterization programs and the data resulting from the Columbium-Tantalum Characterization Plan submitted to the NRC in January 1994. The following discussions provide information about each of the characterization efforts performed to date, as well as the resulting evaluation and current radiological status, and the planned disposition of each of the C-T process and support area buildings, and associated equipment.

1.6.1 Radiological Characterization Programs

In addition to the DOE site-wide investigations referenced in paragraph 1.4.1 above, several site assessments specific to the C-T process have been conducted in Plant 5 and other support areas of the C-T project. The site assessments included hydrogeologic, radiological, and chemical investigations.

The site assessments were designed to quantify the physical and chemical characteristics of the C-T process and process support areas, and perform an initial assessment of the suspected areal and vertical extent of radioactive contamination. Affected and potentially affected building structures and equipment, as well as subsurface materials were characterized. The primary isotopes of concern in these characterization studies were U-

238, U-234, Th-230, Ra-226, Th-232, Th-228, Ra-228, and the unregulated radionuclide K-40.

Measurement techniques for buildings and equipment included gamma scanning while walking over the areas and gamma and beta-gamma activity measurements on a gridded surface using hand-held, direct reading instruments. Discrete samples were also gathered and sent for gamma spectroscopy and alpha spectroscopy at the TMA/Eberline laboratory in Oak Ridge, Tennessee.

Summaries of the radiological investigation programs that involved equipment and building surfaces, the focus of each program, and the techniques used are discussed below. A summary evaluation of the C-T characterization program results and the current radiological status of each building is provided in Appendix A and discussed in paragraph 1.6.2 below. A typical graphical representation of the characterization data is included as Appendix B.

FUSRAP

As discussed in paragraph 1.4.1, the DOE has performed extensive radiological investigations of the St. Louis Plant and has remediated some residual contamination from the MED-AEC operations under FUSRAP and in accordance with the terms of the FFA.

Preliminary Radiological Investigation (PRI)

In 1992, TMA/Eberline removed loose contamination from equipment, interior walls, and floors of selected C-T process buildings in Plant 5. Following this removal activity, TMA/Eberline performed a Preliminary Radiological Investigation of loose and fixed contamination of selected C-T and Plant 5 buildings and roofs. This activity provides a significant portion of the surface data required for the characterization.

Scoping Surveys

In 1993, Rust Remedial Services, Inc.(Rust) performed a radiological scoping survey in selected C-T support areas; Buildings 90, 91, and 236 maintenance shops; Buildings 62 and 213 locker rooms; and Building 250 laboratories. Fixed and removable beta-gamma measurements were performed.

1.6.2 Columbium-Tantalum Plant Phase I Characterization

Based on the knowledge of site operations and results of previous radiological assessments, the significant radionuclides of concern were determined to be the radionuclides found in natural uranium and natural thorium. A characterization plan was submitted to NRC in January, 1994. Comments on the plan were received from NRC in June, 1994 and considered prior to implementation of the program. Characterization

activities were completed in 1996. Characterization activities focused on the identified radionuclides of concern and the former C-T process and support areas in Plant 5. C-T support areas outside of Plant 5 were also characterized: employee change rooms in Building 62, Plant 3; maintenance areas in Buildings 90 and 91, Plant 8; C-T incinerator and adjacent Building 101 roof in Plant 6, and the wastewater basins in Plant 7.

The preliminary radiological investigation and scoping survey direct measurements were performed using instruments sensitive to gamma or beta/gamma activity. The characterization plan stated that direct measurements would measure alpha, beta, and/or beta/gamma activity. Beta radiation was selected as the basis for characterization activity and any subsequent direct measurements of fixed contamination. Removable contamination was measured using the typical alpha measurement of surface wipe samples. Beta activity was selected for the following reasons:

- Direct measurement of alpha particles is unreliable due to self adsorption; contamination of painted or other coated areas would not be detected.

- Direct measurement of beta/gamma activity would not provide results representative of actual contamination due to contribution of "shine" from subsurface and other areas containing natural gamma emitters (e.g., brick and concrete).

- Beta activity measurements can achieve lower MDAs as beta background is approximately 1/3 that of background beta/gamma activity.

- Awareness of NRC recommendation to a licensee and draft survey guidance relying on beta in lieu of alpha measurements.

A series of guideline values were established for each radionuclide of concern and were based on the NRC's Branch Technical Position, Option 1, and Policy and Guidance Directive FC 83-23.

Surface contamination guideline values were established for each support and process building. Six surface samples were obtained from buildings 62, 90/91, 213, 238, 246, 247, and 248; and analyzed to define the presence and distribution of radionuclides. Samples of surface contamination on concrete and brick were collected and analyzed for relevant radionuclides to determine average release guidelines for these surface materials.

In order to determine criteria for the various areas, a scan was taken over an area to find locations with high readings. These locations were outlined and scabbled samples were taken from these locations. The samples were then analyzed for the seven isotopes of interest. A beta surface activity guideline was derived based upon the relative distribution of radionuclide activity in the samples and NRC's Acceptable Surface

Contamination Guidelines for nuclide groups using equation A-2 from NUREG/CR 5849:

Derived Guideline = $1/(f_1/G_1 + f_2/G_2 + \ldots f_n/G_n)$,

where: f_n = fraction of total sample activity contributed by each radionuclide "n", and
G_n = NRC free release surface guideline value for radionuclide "n".

For a more complete discussion of this derived beta surface activity derivation, see Appendices C and D.

The C-T process and support areas were initially separated into affected and unaffected areas based on historical knowledge and facility use. Areas with known contamination and areas for which the previous surveys identified activities in excess of 25% of the guideline values were also identified as affected areas. C-T areas for which there was a low likelihood of contamination based on historical knowledge or for which previous surveys identified activities less than 25% of the guideline values, were identified as unaffected areas. Areas surveyed as unaffected areas were resurveyed as affected areas if any activity exceeding 25% of the guideline value. During the Phase I characterization program a duplicate reading was obtained for every twentieth survey point during the fixed gamma/beta-gamma surveys.

A smear for the measurement of removable contamination was taken at every fifth survey point. A duplicate smear was taken for every tenth smear collected. Approximately 2,000 random smears (every fifth survey data point) were taken during the characterization program. Of these, about 200 had detectable measurements. Only five of these smears had readings greater than 20% of the characterization surface release criteria. Upon examination of the location of these five samples, it was determined that all were in relatively inaccessible areas such as inside of duct work , behind access panels and p-traps of drain lines. In addition, a smear sample was taken from the area of the highest measurable surface contamination detected during the survey of asphalt surfaces. Evaluation of this smear sample indicated no removable contamination was present.

Due to the large number of smear samples taken and the small number of positive readings, and considering the location of those samples, the possibility of finding large amounts of removable contamination is very small. Further, although some additional characterization will be necessary for waste segregation, data of sufficient quantity and accuracy exists to safely initiate remediation and in-process sampling activities.

Building Interiors - A 6-foot grid system was installed on the floors and mezzanines of each affected building surveyed. The origin of the grid system was the southwest corner of the respective building or room. Direct radiological measurements for beta and/or beta-gamma activity were taken at each 6-foot grid intersection. Removable measurements were taken at every fifth survey point and at locations where direct

measurement values exceeded the removable criteria. Table 1-3 identifies the building surfaces surveyed as part of this characterization program.

Piping, Vessels and other Equipment - Certain vessels and other equipment in the C-T process buildings were surveyed as part of the PRI site assessment (see Appendix A). No additional equipment was surveyed as part of the Phase I characterization. Table 1-3 identifies buildings in which equipment surveys were performed.

Building Exterior Walls - The exterior walls of Buildings 213, 238, 246A&B, 247A&B, and 248A&B were surveyed on a 6-foot grid for gamma/beta-gamma activity by taking direct measurements. Removable measurements were taken at every fifth survey point and at locations where direct measurement values exceeded the removable criteria. Buildings 235, 236, and 245 were surveyed on a 12-foot grid. See Table 1-3 for a list of building exterior walls surveyed as part of this characterization program.

TABLE 1-3
BUILDING SURFACE SURVEYS

BUILDING/ AREA	101	213	235	236	238	246A	246B	247A	247B	248A	248B
First Floor, Floors		X	X	X	X	X	X	X	X	X	X
First Floor, Walls		X	X	X	X	X	X	X	X	X	X
First Floor, Ceiling		X	X	X	X	X	X	X	X		X
Second. Flr., Floors					X	X		X	X	X	X
Second. Flr., Walls					X	X		X	X		X
Second. Flr., Ceiling					X	X		X			X
Equipment					X	X	X	X	X	X	X
Exterior Walls		X	X	X	X	X	X	X	X	X	X
Outside Fence Area	X										

Building Roofs - Based upon operating history, proximity to C-T operations, and data obtained during the PRI and scoping surveys, the roofs of Buildings 213, 222, 223, 235, 236, and 245 were surveyed using a gamma walkover scan. Core samples were taken on each roof at locations with elevated gamma activity, or at locations identified in the PRI or scoping surveys, for sample analysis and direct beta-gamma measurements.

For building roofs with no previous survey data (Buildings 250, 240, and 101), a six-foot grid system was installed. Direct measurements were taken for gamma/beta-gamma activity. Removable contamination measurements were taken at every fifth survey location and at locations where direct measurement values exceeded the removable criteria.

Roofs of Buildings 200E, 200W, and 204, which did not house any C-T processing operations, were surveyed as unaffected areas. A total of 20% of the surface area was surveyed with a gamma walkover scan, and direct measurement for gamma/beta-gamma activity.

Roof drains on C-T process buildings were surveyed at their determined end points. All roof penetrations and ventilation systems were surveyed at access/discharge points on the roof. Table 1-4 provides a complete list of building roof surveys conducted during the Phase I characterization.

In addition to the activities described in the characterization plan, Mallinckrodt performed additional measurements in selected internal and external areas of Buildings 101, 235, 250, and 245. In most instances, these measurements were performed to support building remodeling and renovation projects and represent final surveys for these areas. These measurements are described in additional detail in Section 1.6.4.

TABLE 1-4
BUILDING ROOF SURVEYS

BUILDING	RADIOLOGICAL SURVEY	CORE SAMPLES
101	gamma/beta-gamma	4
200E	gamma/beta-gamma	0
200W	gamma/beta-gamma	0
204	gamma/beta-gamma	0
213	gamma	4
222	gamma	4
223	gamma	4
235	gamma	4
236	gamma	4
238	none	4
240	gamma/beta-gamma	4
245	gamma	3
246A	none	2
246B	none	2
247A	none	2
247B	none	2
248	none	4
250	gamma/beta-gamma	4

1.6.3 Radiological Data Analysis

The radiological information that follows is a summary of all data collection efforts on buildings and structures conducted to-date at the C-T project site. Mallinckrodt has compiled the data from the characterization programs into a single database and the C-T Project buildings and equipment have been divided into 321 survey areas with each wall, roof, floor, and equipment item representing an individual survey area, as shown in Appendix A. Appendix A also provides the detailed radiological data collected to-date for each building and equipment survey area, and in addition to the survey area number,

identifies the building, surface, survey results (mean and upper 95% confidence level of the mean of all the samples available for each survey area, and using the Students' t-test with appropriate degrees of freedom), surface release criteria value applied, survey evaluation summary and planned classification for final release survey. A summary evaluation of these results and the planned disposition of buildings and equipment follows.

Buildings

C-T process and support buildings have been extensively surveyed using direct reading instruments to determine the gamma and beta-gamma activity, and using smear samples for evaluation of removable contamination. These surveys have included interior and exterior walls, floors, ceilings, roofs, and equipment (i.e., associated pipes, ducts, fixtures, and other equipment). These surface areas have been identified and data collected as individual survey areas. Each wall, floor, ceiling, and roof is an individual survey area and the equipment is combined into logical survey groupings or areas. Results for each of the 321 survey areas are shown in Appendix A.

Survey Data/Reports: Survey results for individual survey areas have been plotted (mapped) to record the location of each survey point and identify areas of contamination. In addition to the grid indicating the location of each survey point, each surface map also includes:

- Number of Data Points for that Survey area;
- Surface Area involved;
- Average, Maximum, and 95% Confidence value for all Data Points;
- Number of points greater than criteria, and number greater than three times criteria.

Selected examples of these building surface maps are provided as Appendix B. Appendix B includes maps of eight different building surfaces. These surfaces were selected to depict typical results for various surface materials (concrete, brick, steel, etc.), surface locations (interior or exterior wall, floor, roof, etc.), and degree of contamination (highly contaminated, clean, etc.). The eight selected surfaces are:

SURFACE	MATERIAL	STATUS
• Building 248 First Floor	Concrete	Locally Contaminated
• Building 213 North Exterior Wall	Cinder Block	Locally Contaminated
• Building 238 Mezzanine	Steel	Clean
• Building 235 East Wall	Brick	Partially Contaminated
• Building 248 West Exterior Wall	Brick	Locally Contaminated
• Building 213 Break Room Ceiling	Ceiling Tiles	Clean
• Building 246B South Wall	Fiberglass	Locally Contaminated
• Building 204 Roof	Composite	Locally Contaminated

This mapped data will be used during the remediation phase of the project to identify the location of contamination and support safe removal, decontamination, and demolition.

1.6.4 Current Radiological Status

Buildings

Buildings within the C-T process and support areas have been grouped according to their radiological status and intended disposition. Buildings 213, 214, 238, 246A&B, 247A&B, and 248 will be demolished. Contaminated equipment will be removed from the roof of Building 204. Contaminated roofing material will be removed from Buildings 222 and 223. Localized decontamination will be performed in Building 250 laboratory and office areas which supported C-T.

The interiors of Buildings 62, 90, 91 and the exteriors of Building 101 and other non C-T Plant 5 buildings are clean and will be surveyed for release. Specific areas of Building 250 were surveyed during a recent renovation while others will be locally decontaminated and surveyed for release. Interior and exterior surfaces of Building 245 were surveyed during a recent renovation. Building 250 and 245 surveys will be included in the Final Status Survey Reports.

The buildings which have been excluded from this plan because of demolition and/or remediation under the FUSRAP Program are Buildings: 25, 116, 117, 219, 700, 704, 705, 706, 707, 708, and 100.

Each building associated with the C-T process or located within Plant 5, including material storage and other support areas in other Plants, is identified in Table 1-5 below. In addition to the list of structures, the table identifies the C-T operational usage, and the planned disposition of each structure.

TABLE 1-5
DISPOSITION OF C-T PROCESS AND SUPPORT AREA BUILDINGS

Bldg. and Location	C-T Operational Usage	Planned Disposition
Plant 1 Area		
Building 25	Laboratory	FUSRAP Remediation
Plant 3 Area		
Building 62	Change (Locker) Rooms	Survey and Release
Plant 5 Area		
Building 200	Organic/Inorganic Manufacturing Process	Roof Sampling/Decontamination
Building 201	Organic/Inorganic Manufacturing Process	Roof Sampling/Decontamination
Building 204	Inorganic Chemical Manufacturing	Roof Sampling/Decontamination
Building 213	Change and Break Rooms	Demolish
Building 214	Transformer/Switchgear Room	Demolish
Building 219	Warehouse/Shed	FUSRAP
Building 222	Warehouse	Roof Sampling/Decontamination
Building 223	Warehouse	Roof Sampling/Decontamination
Building 235	Feed Material/URO Storage (East Half)	Surface Decon
Building 236	Maintenance Area, Product Drying	Surface Decon
Building 238	C-T Ore Grinding/Dissolving/T Processing	Demolish
Building 240	Offices	Survey and Release
Building 245	Not Used for C-T (Renovated)	Survey and Release
Building 246A	Offices	Demolish
Building 246B	Solvent Extraction Process	Demolish
Building 247A	C-T Solvent Extraction/Product Storage	Demolish
Building 247B	Columbium Filtration and Drying	Demolish
Building 248	Columbium Filtration/Drying/Calcining	Demolish
Building 250	Offices and Quality Control Labs	Local Decon - Survey and Release
Plant 6 Area		
Building 101 Roof	N/A	Roof Sampling/Decontamination
Bldg 101 Incin Pad	C-T Incinerator	Eqpt. Removal - Local Decon
Building 116	Receipt/Unloading of C-T Ore	FUSRAP
Building 117	URO Drum Preparation and Staging	FUSRAP
Plant 7 Area		
Building 700	Storage of Tin Slag Feed Material	FUSRAP
Building 704	URO Drum Storage	FUSRAP
Building 705	C-T Ore Storage	FUSRAP
Building 706	C-T Ore Storage	FUSRAP
Building 708	Storage of Tin Slag Feed Material	FUSRAP
Plant 8 Area		
Building 90/91	Maintenance Areas	Survey and Release

Equipment

Equipment within the C-T process and support area buildings has been surveyed using direct reading instruments to determine the gamma/beta-gamma activity, and using smear samples for evaluation of removable contamination. The disposition of equipment by building and disposition category is covered in the following paragraphs. Specific survey results for equipment within each building by floor/room location is provided in Appendix A.

Considered Potentially Contaminated: The C-T process equipment within Buildings 238, 246A, 246B, 247A, 247B, and 248, is considered potentially contaminated, even though more data are needed on some of the equipment to verify the contamination status and to support segregation and disposal. This equipment will be segregated by known characterization results and if practical, decontaminated and/or volume reduced.

Equipment on the roofs of Buildings 200E, 200W, 201, 204, 213, 235, and 236 is also considered potentially contaminated because of their proximity to the processing buildings, however, further testing is required to support segregation and disposal, decontamination and/or volume reduction.

In addition, the equipment located on the Building 101 Incinerator Pad is considered contaminated and will be segregated by known characterization results and volume reduced if practical.

Considered Clean: Equipment within Buildings 90 and 91 is considered clean This equipment will be surveyed and released.

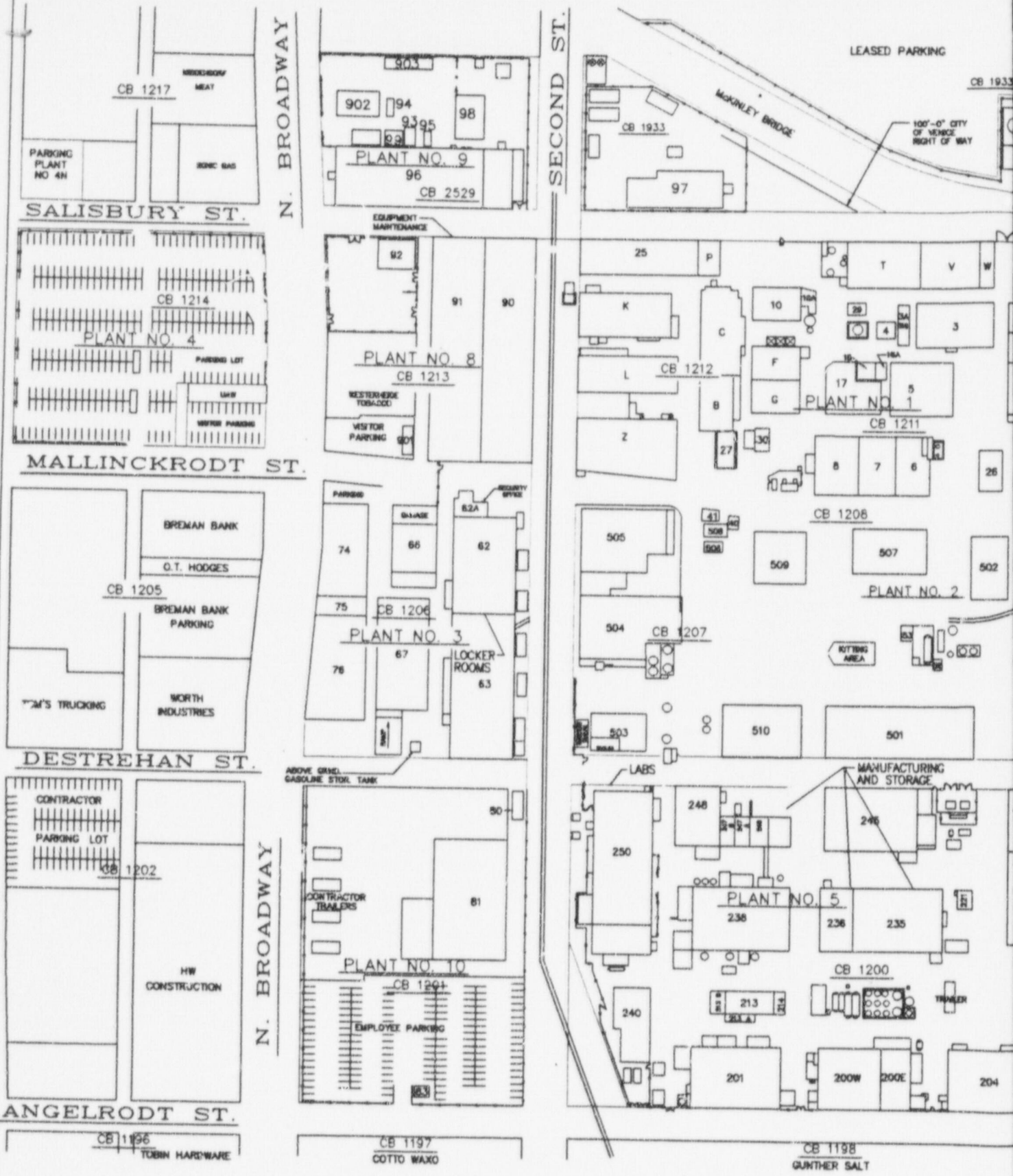
1.6.5 Conclusion

The C-T project buildings and equipment have been subjected to comprehensive radiological characterization investigations. These programs have delineated the nature and extent of radiological contamination present and have confirmed the suspected radiological conditions, for both affected and unaffected areas, based on process history and site knowledge, using surface beta measurements. Concentrations of radionuclides that exceed acceptance criteria have been identified on building surfaces and equipment, and these locations accurately recorded for subsequent remediation.

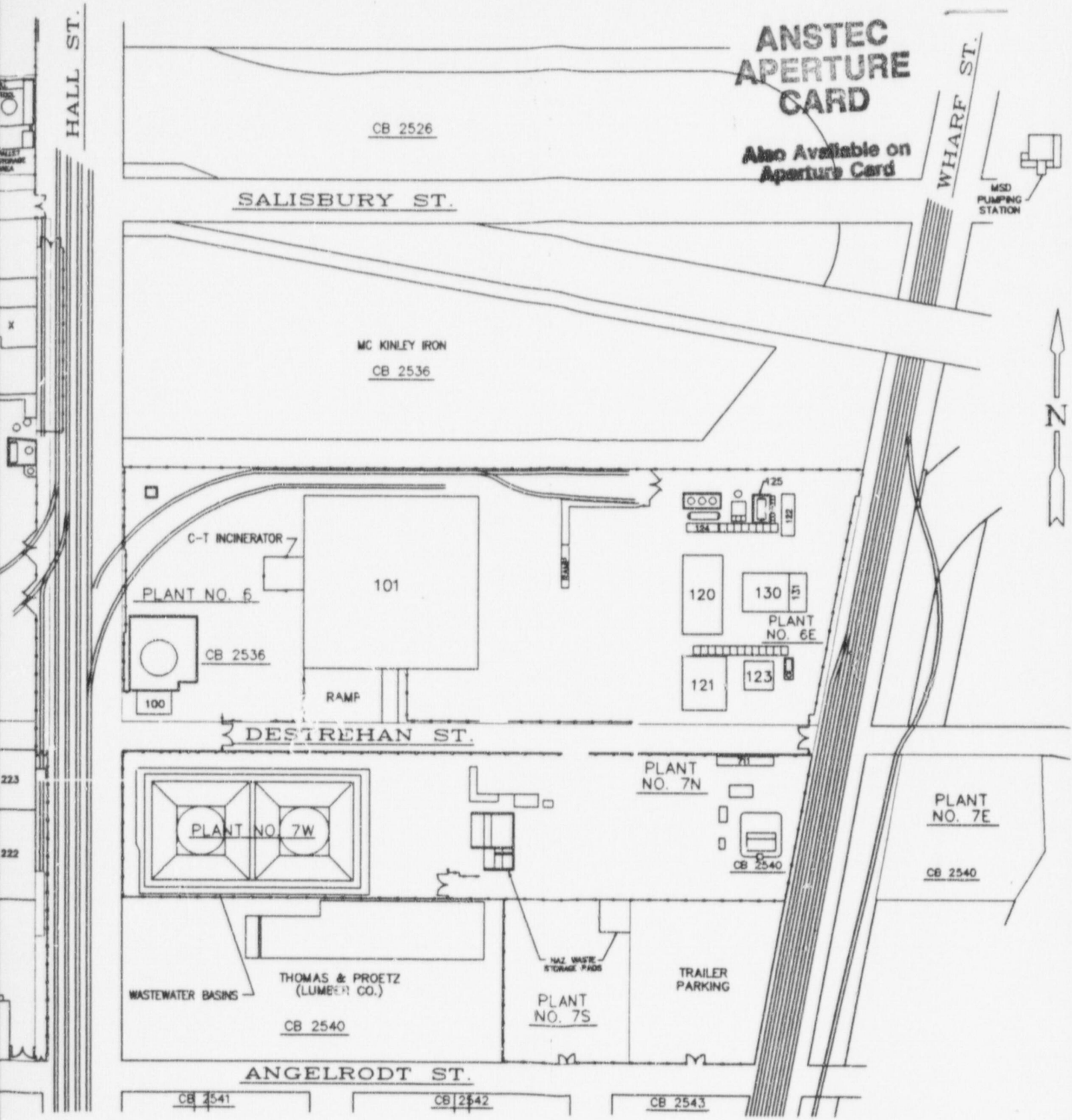


Note: Drawing taken from USGS Granite City, IL-MO quadrangle, dated 1993

Figure 1-1
Site Location Map
Rev 0


SALISBURY ST.
N. BROADWAY
SECOND ST.
MALLINCKRODT ST.
DESTREHAN ST.
N. BROADWAY
ANGELRODT ST.
LEASED PARKING
McKINLEY BRIDGE
100'-0" CITY OF VENICE RIGHT OF WAY
CB 1217
PARKING PLANT NO 4N
CB 1214
PLANT NO. 4
PARKING LOT
UNIT
VISITOR PARKING
CB 1205
BREMAN BANK
O.T. HODGES
BREMAN BANK PARKING
M'S TRUCKING
WORTH INDUSTRIES
CONTRACTOR PARKING LOT
CB 1202
HW CONSTRUCTION
CB 1196
TOBIN HARDWARE
902
903
94
95
98
PLANT NO. 9
96
CB 2529
EQUIPMENT MAINTENANCE
92
91
90
PLANT NO. 8
CB 1213
WESTERHEIDE TOBACCO
VISITOR PARKING
74
66
62
62A
75
CB 1206
PLANT NO. 3
67
LOCKER ROOMS
63
76
ABOVE GRND. GASOLINE STOR. TANK
50
CONTRACTOR TRAILERS
61
PLANT NO. 10
CB 1201
EMPLOYEE PARKING
CB 1197
COTTO WAXO
CB 1933
97
25
P
T
V
W
K
C
10
10A
29
3A
4
3
L
CB 1212
F
G
B
Z
17
5
PLANT NO. 1
CB 1211
27
30
8
7
6
26
505
41
506
508
509
507
502
CB 1208
PLANT NO. 2
504
CB 1207
KITTING AREA
503
510
501
LABS
MANUFACTURING AND STORAGE
250
248
246
PLANT NO. 5
238
236
235
CB 1200
240
213
214
TRAILER
201
200W
200E
204
CB 1198
GUNTHER SALT

ANSTEC APERTURE CARD
Also Available on Aperture Card



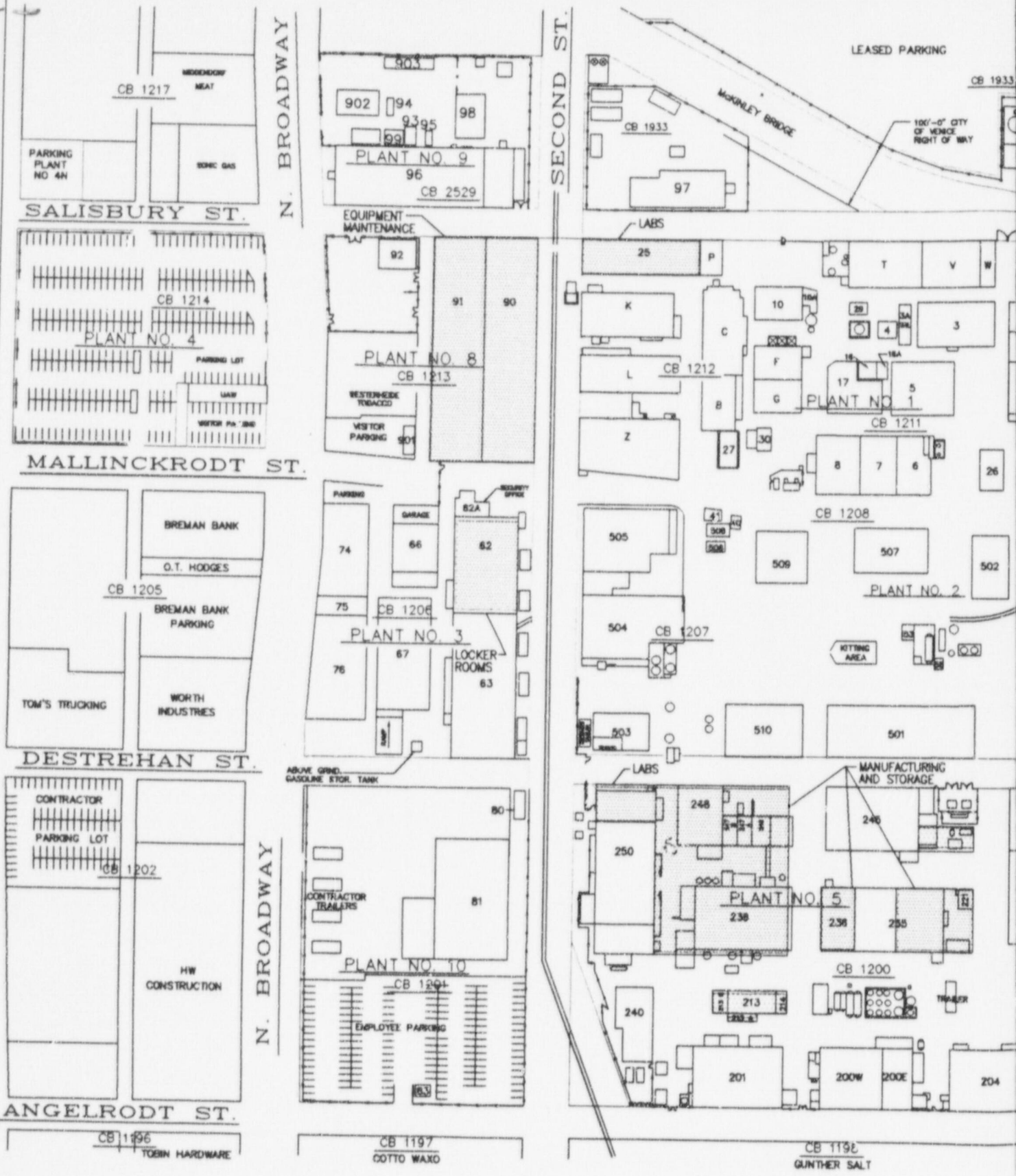
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FIGURE 1-2
MALLINCKRODT ST. LOUIS PLANT SITE
REV. 1

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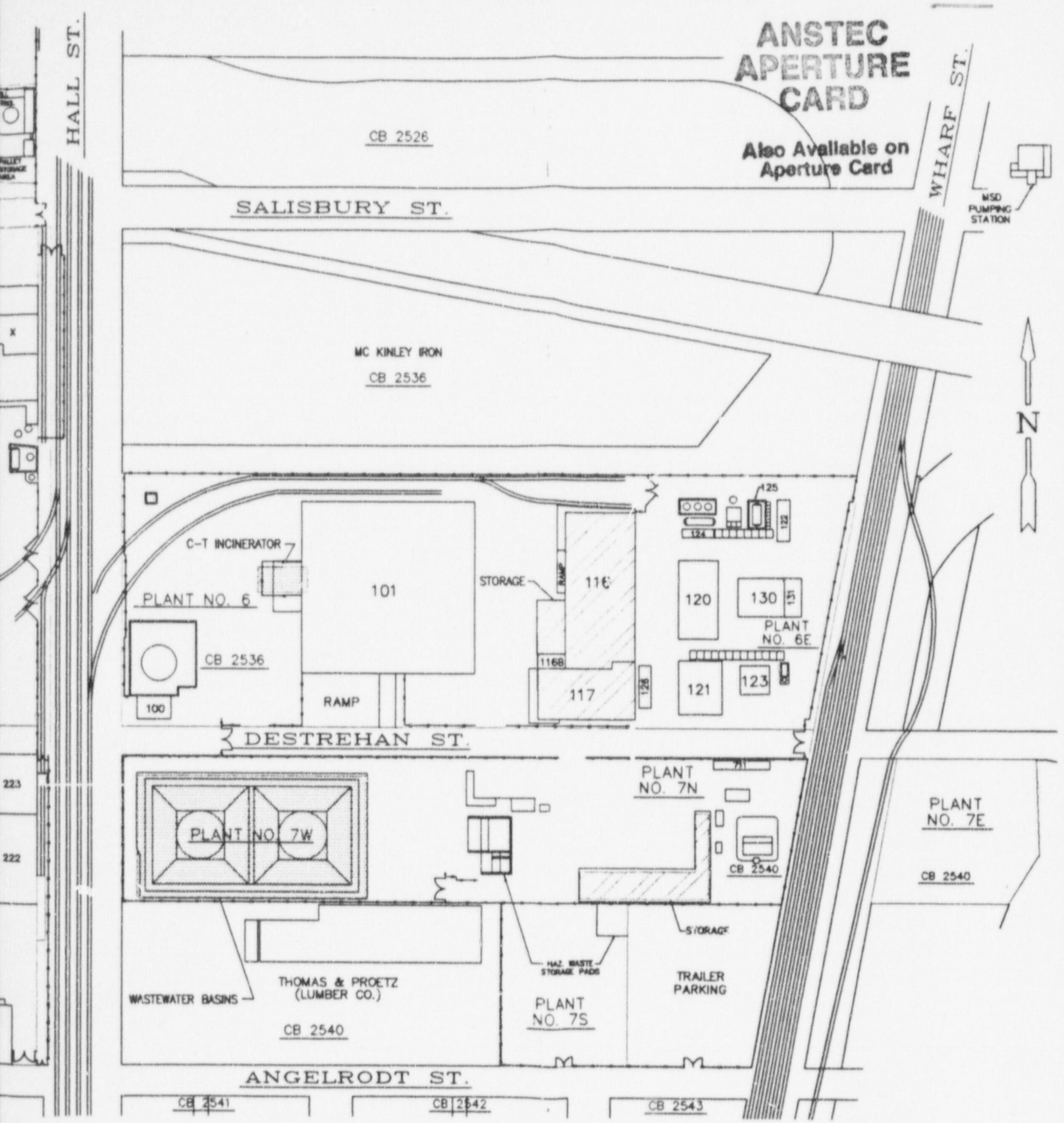


LEGEND

C-T PRODUCTION AND SUPPORT AREAS

CT SUPPORT AREAS REMOVED UNDER FUSRAP

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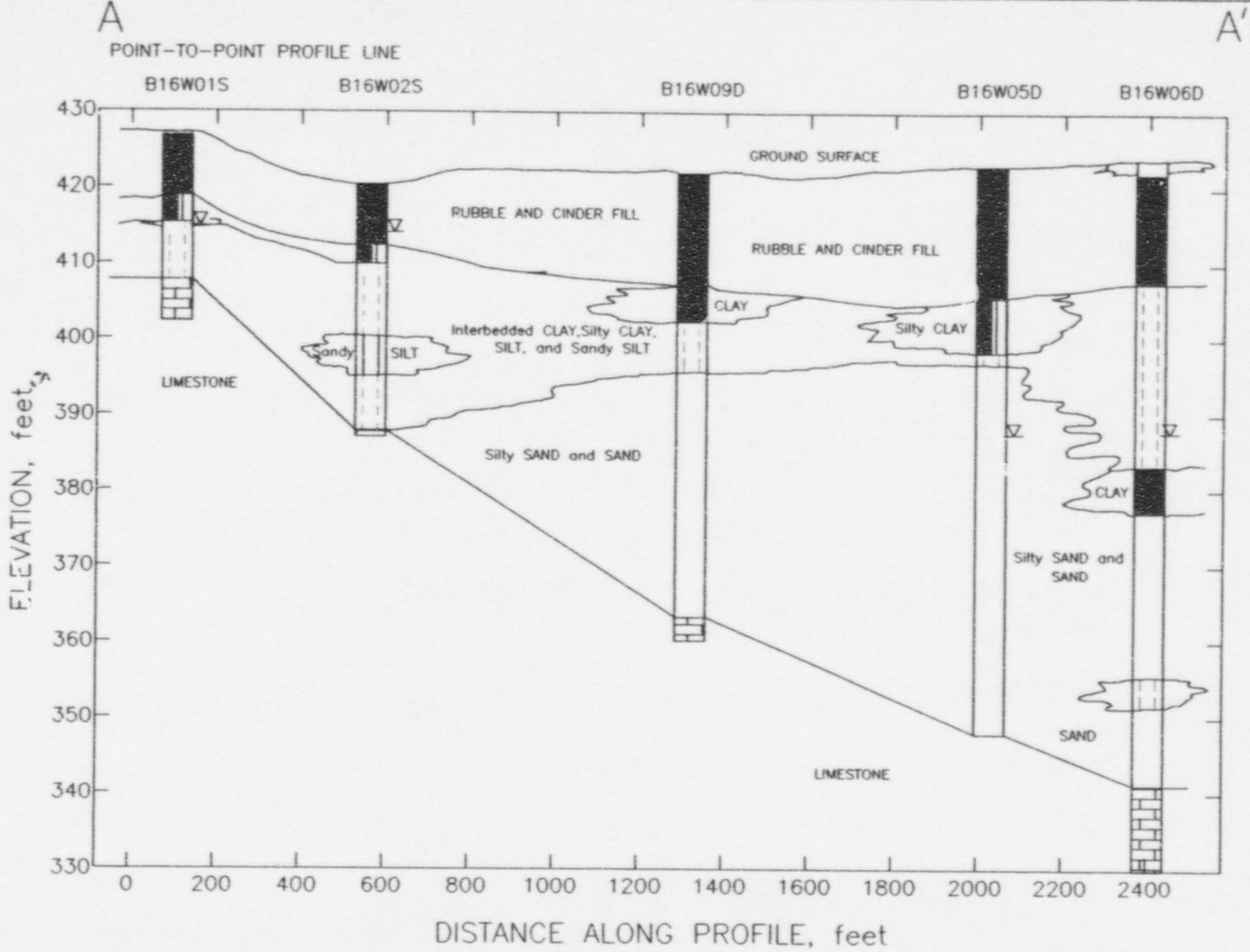
SCALE:

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98042300 17-02

FIGURE 1-3
C-T PRODUCTION PROCESS & SUPPORT AREAS.
REV. 1

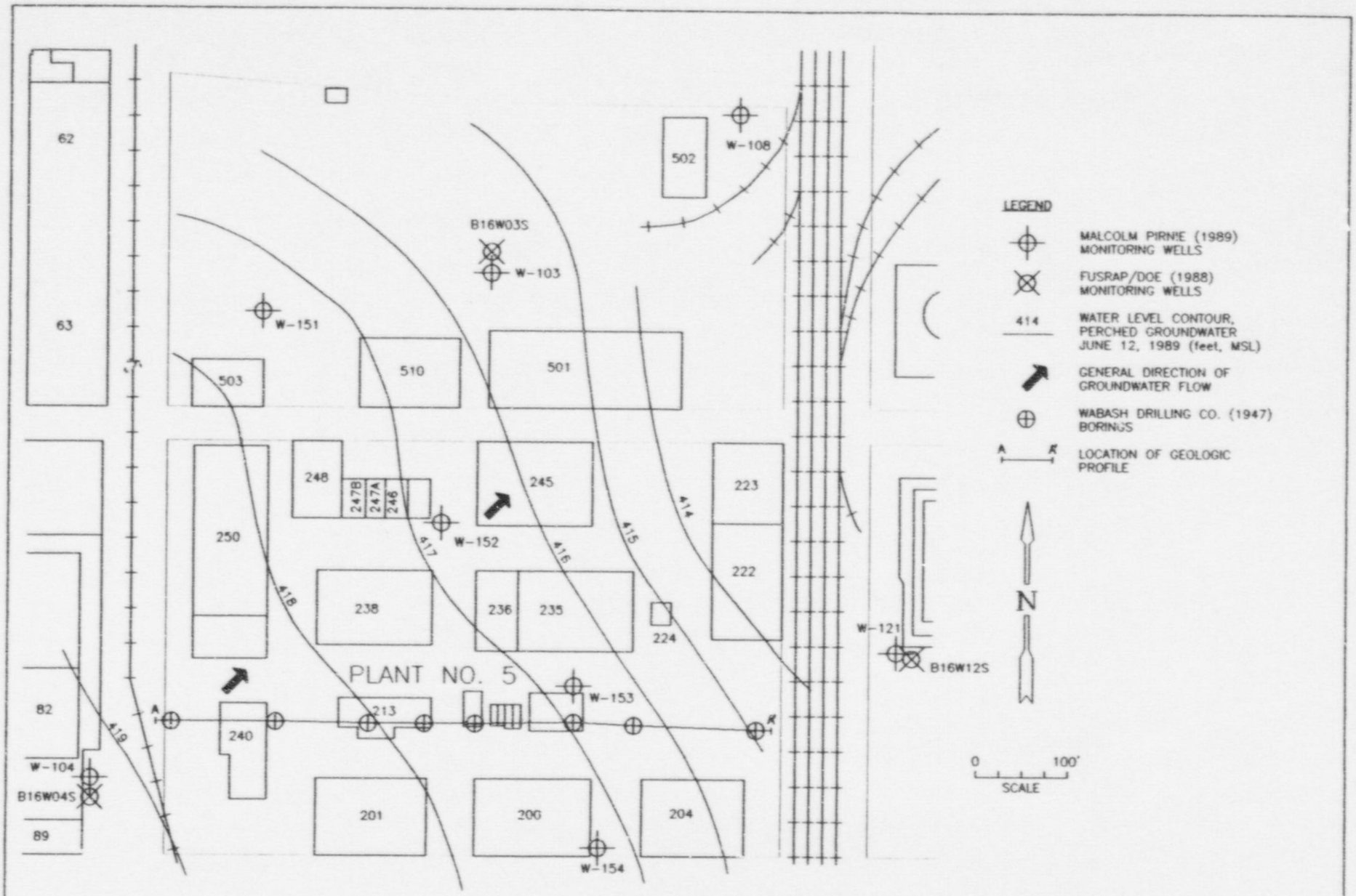
fn = F1-3 10/23/97



SOURCE:
Bechtel National, Inc. September 1990. Radiological, Chemical, and Hydrogeological Characterization Report for the St. Louis Downtown Site in St. Louis, MO. Formerly Utilized Sites Remediation Action Program (FUSRAP). Volumes I-III: Revision 1. DOE/OR/20722-258

FIGURE 1-4
HYDROGEOLOGICAL CROSS SECTION
REV. 1

fn = HYDRO_XN 10/16/97



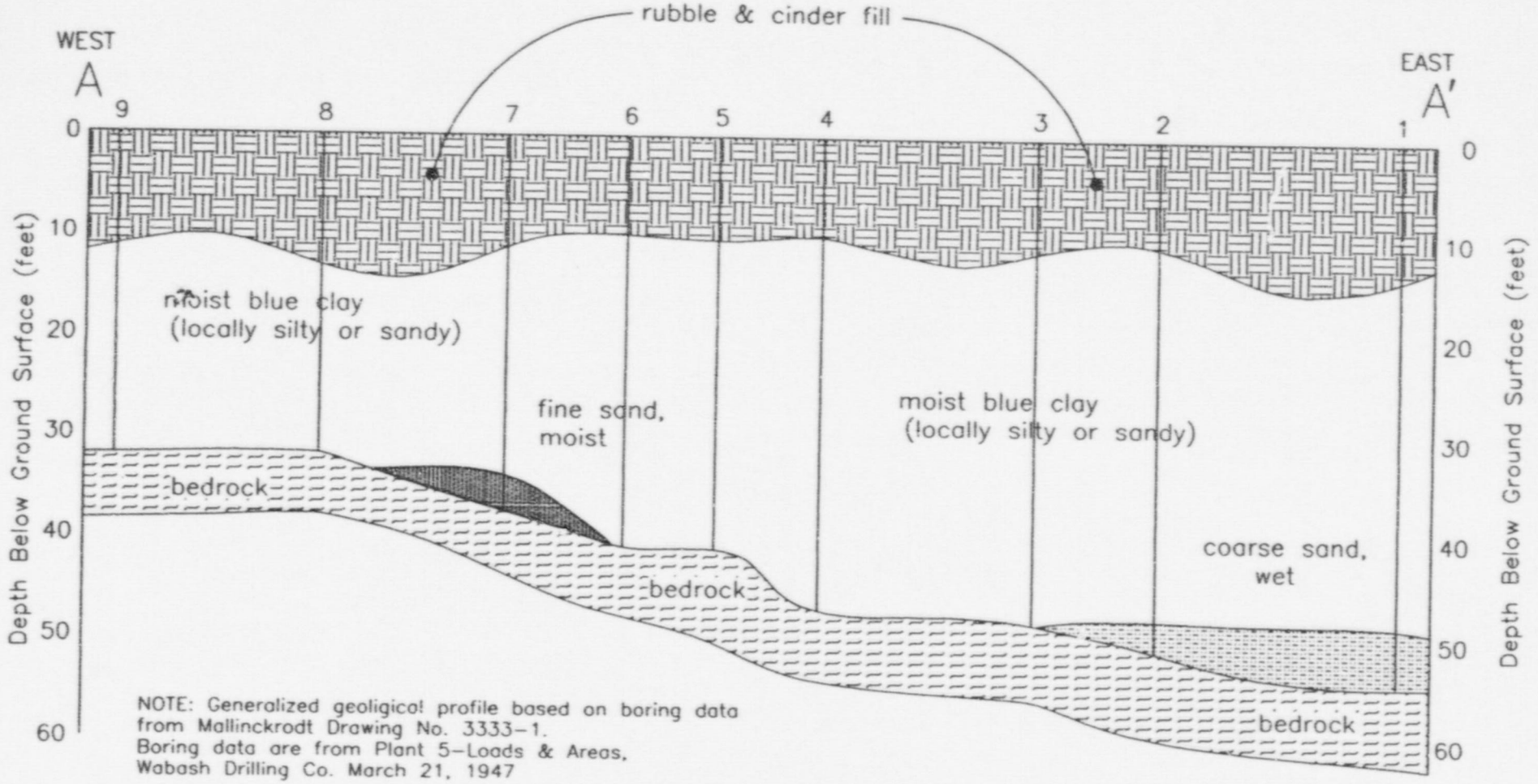
MONITORING WELL LOCATIONS AND FLOW DIRECTION OF PERCHED GROUNDWATER, PLANT 5

MAP REFERENCE

Bechtel National, Inc. September 1990. Radiological, Chemical, and Hydrogeological Characterization Report for the St. Louis Downtown Site In St. Louis, MO. Formerly Utilized Sites Remedial Action Program (FUSRAP). Volumes I–III: Revision 1. DOE/OR/20722–258

FIGURE 1–5
MONITORING WELL LOCATIONS
REV. 1

fn = F1-5 10/23/97



GENERALIZED GEOLOGICAL PROFILE
PLANT 5

FIGURE 1-6
GENERALIZED GEOLOGIC PROFILE
PLANT 5
REV. 1



LEGEND

- ⌑ MONITORING WELL
- 390 ELEVATION OF GROUNDWATER IN FEET ABOVE NATIONAL GEODETIC VERTICAL DATUM (NGVD). GRADE ELEVATION OF THE SITE RANGES FROM 419 TO 425 FEET ABOVE NGVD.
- CONTOUR OF GROUNDWATER ELEVATION

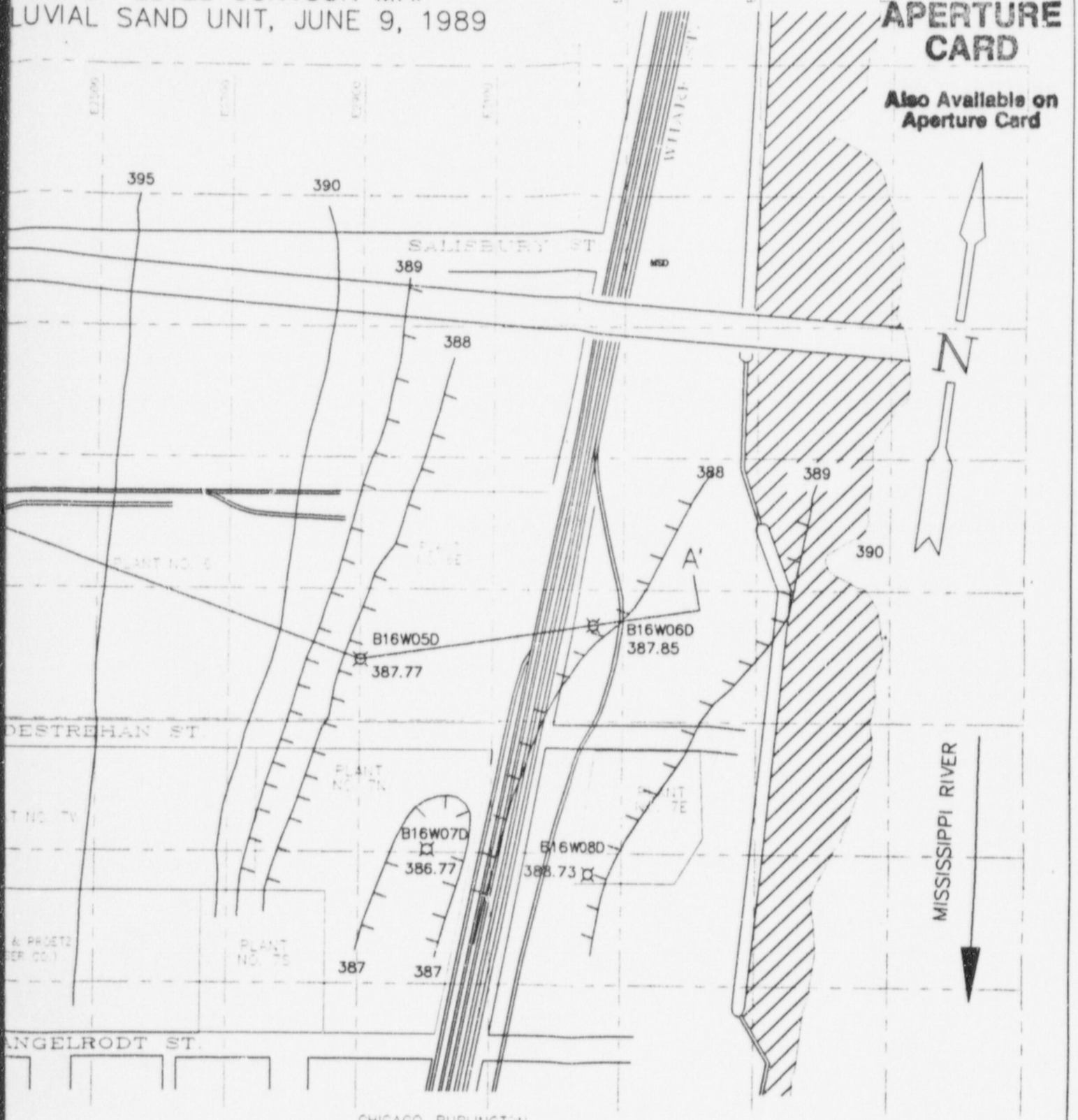
NOTE: WELL B16W09D WAS NOT COMPLETED UNTIL AFTER THE

SOURCE:
Bechtel National, Inc. September 1990. Radiological, Chem
Characterization Report for the St. Louis Downtown Site
Utilized Sites Remedial Action Program (FUSRAP). Volume
DOE/OR/20722-258

WATER LEVEL CONTOUR MAP
LUVIAL SAND UNIT, JUNE 9, 1989

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Also Available on Aperture Card



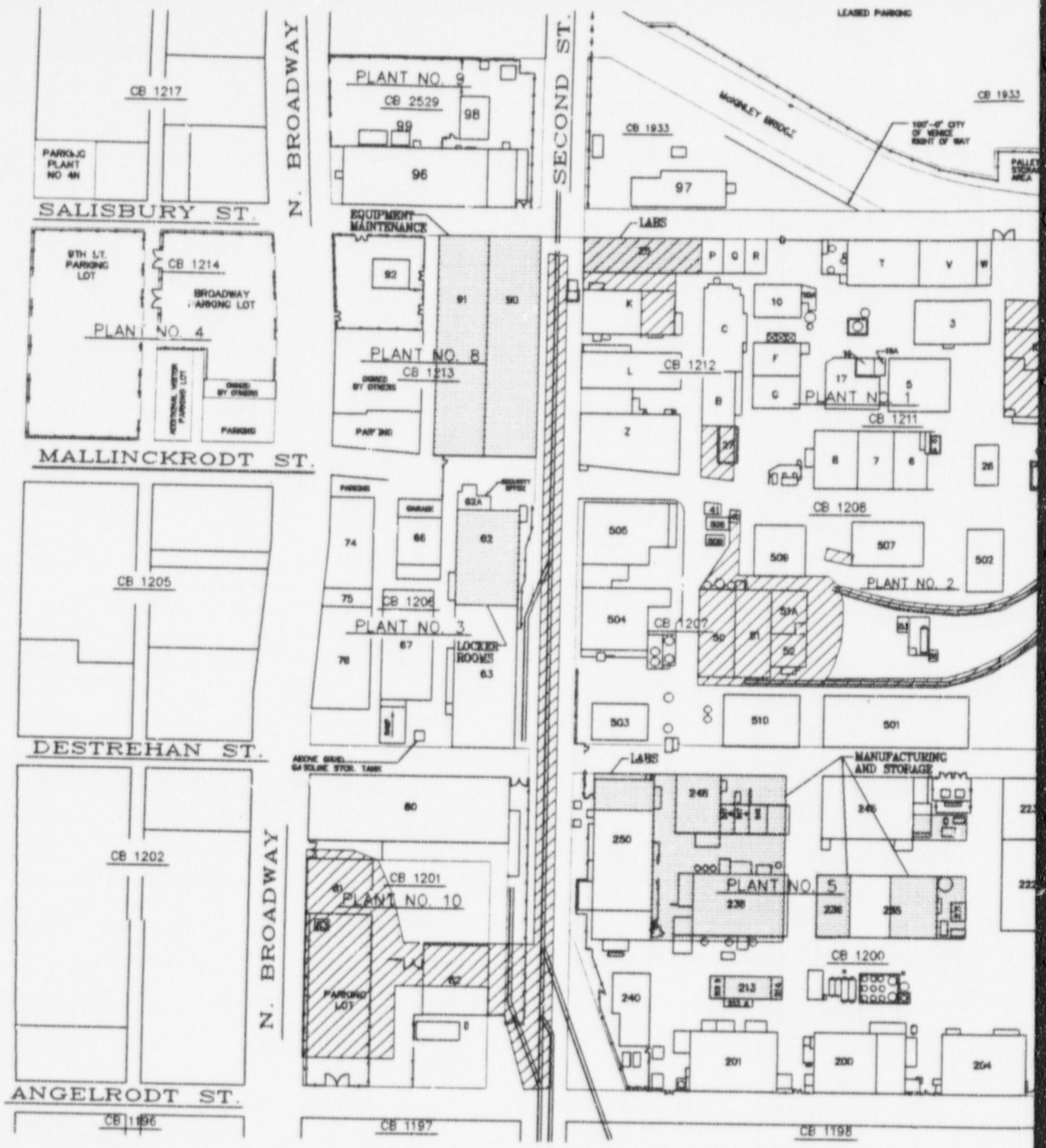
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READINGS WERE TAKEN

ical, and Hydrogeological
n St. Louis, MO. Formerly
s I–III: Revision 1.

SCALE:
0 100' 200' 300'

FIGURE 1-7
WATER LEVEL CONTOUR MAP
ALLUVIAL SAND UNIT
REV. 1

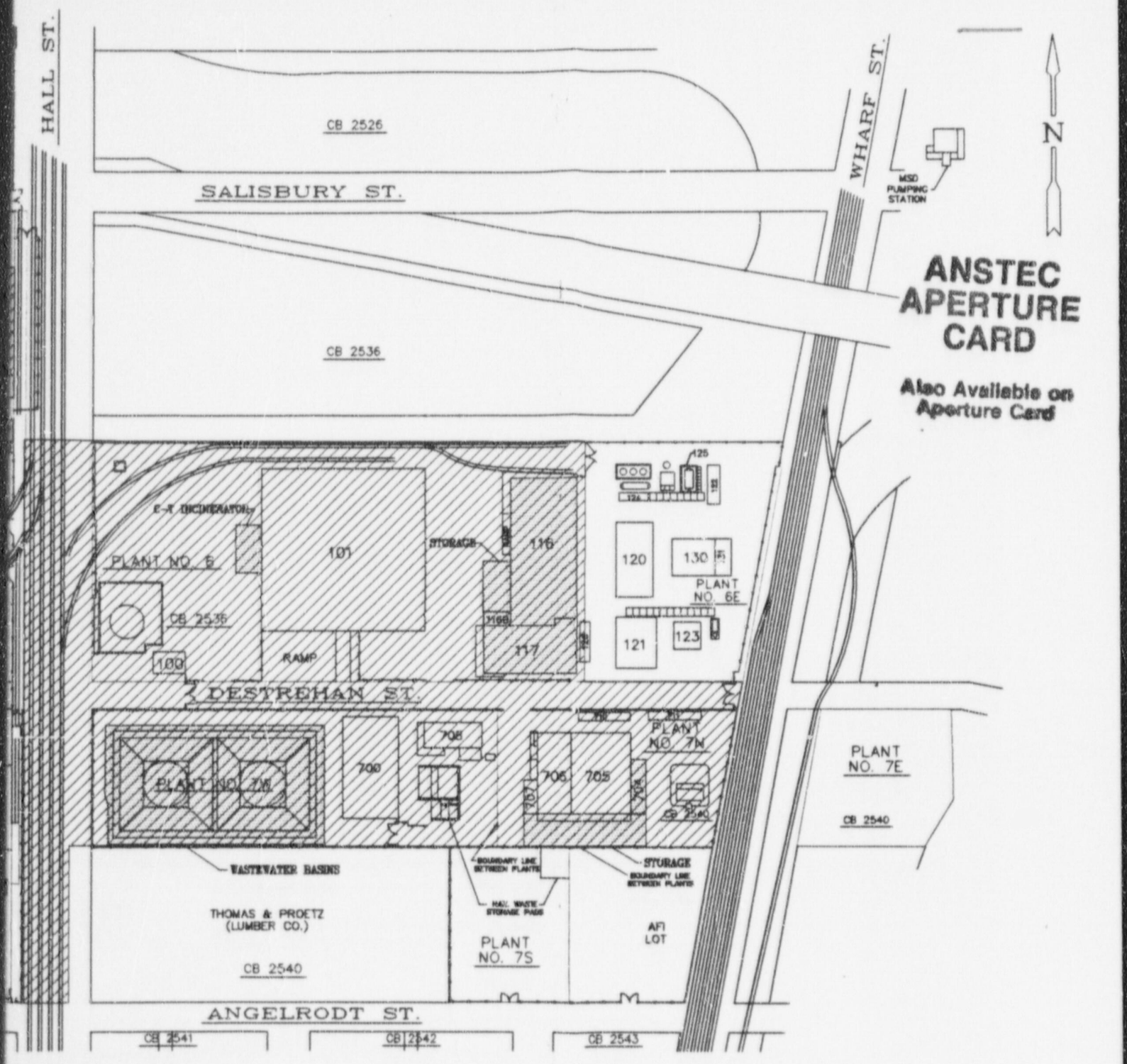


LEGEND

C-T PRODUCTION AND SUPPORT AREA

MED-AEC OPERATIONS

PLANT 1 & 2 (1943-1946)
PLANT 4 (1942-1961)
PLANT 6 (1946-1958)
PLANT 6E & 7 (1950-1958)

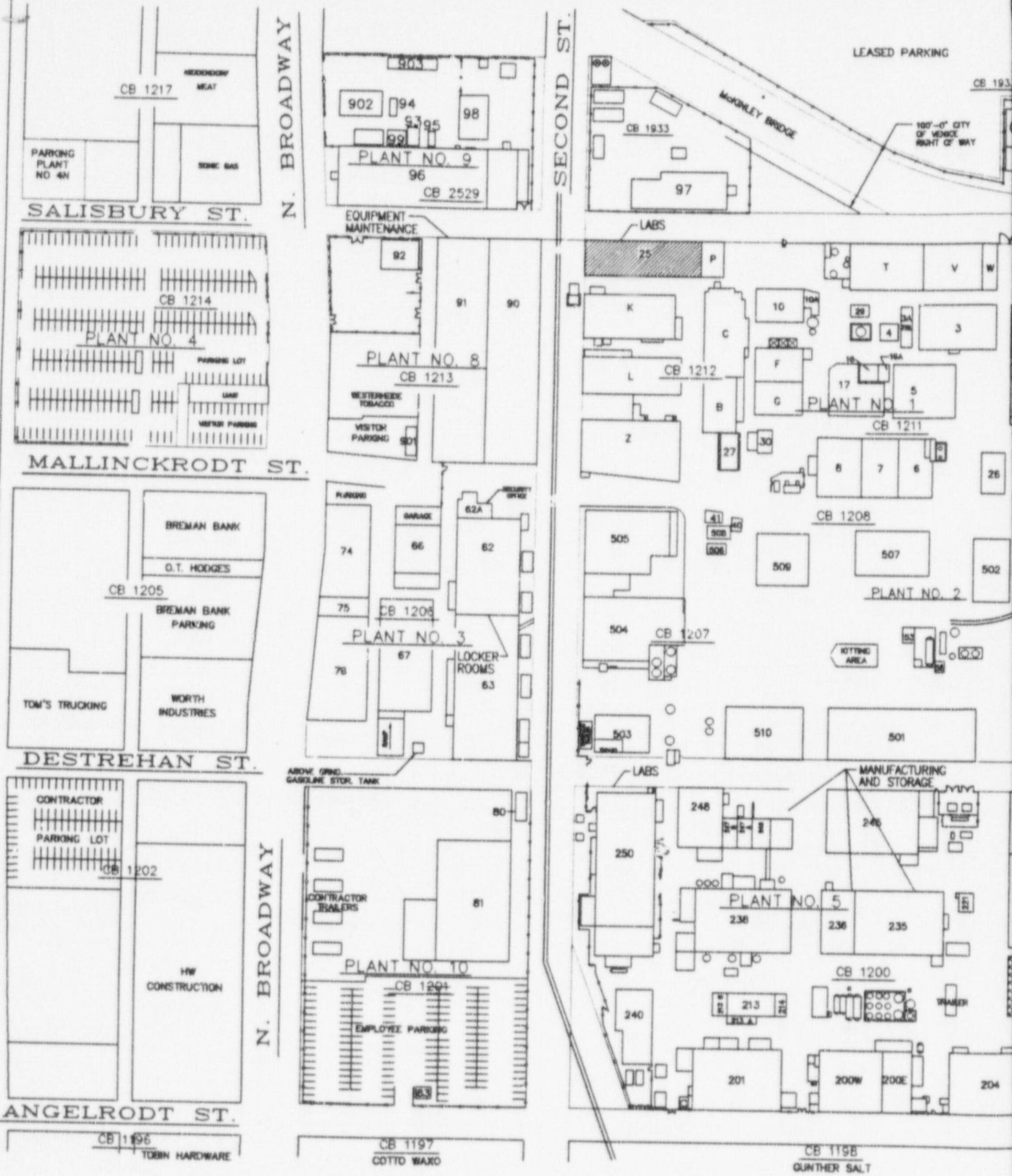


SCALE:

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FIGURE 1-8
MED-AEC PRODUCTION
AND SUPPORT AREAS
REV. 1

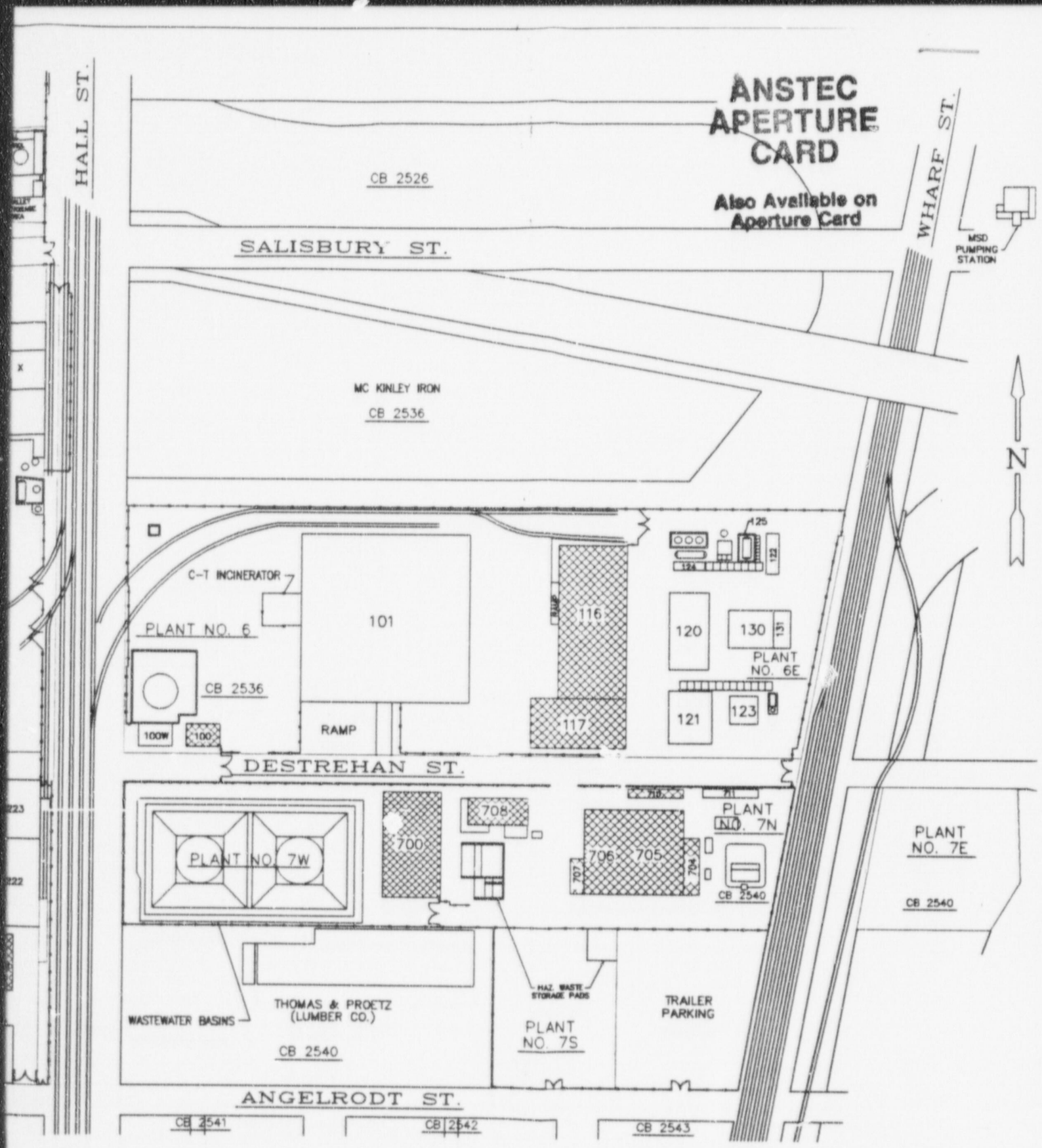


LEGEND

FUSRAP BUILDINGS

FUSRAP BUILDINGS REMOVED

ANSTEC APERTURE CARD
Also Available on Aperture Card



SCALE:

0 100' 200' 300'

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FIGURE 1-9
FUSRAP BUILDINGS
REV. 1

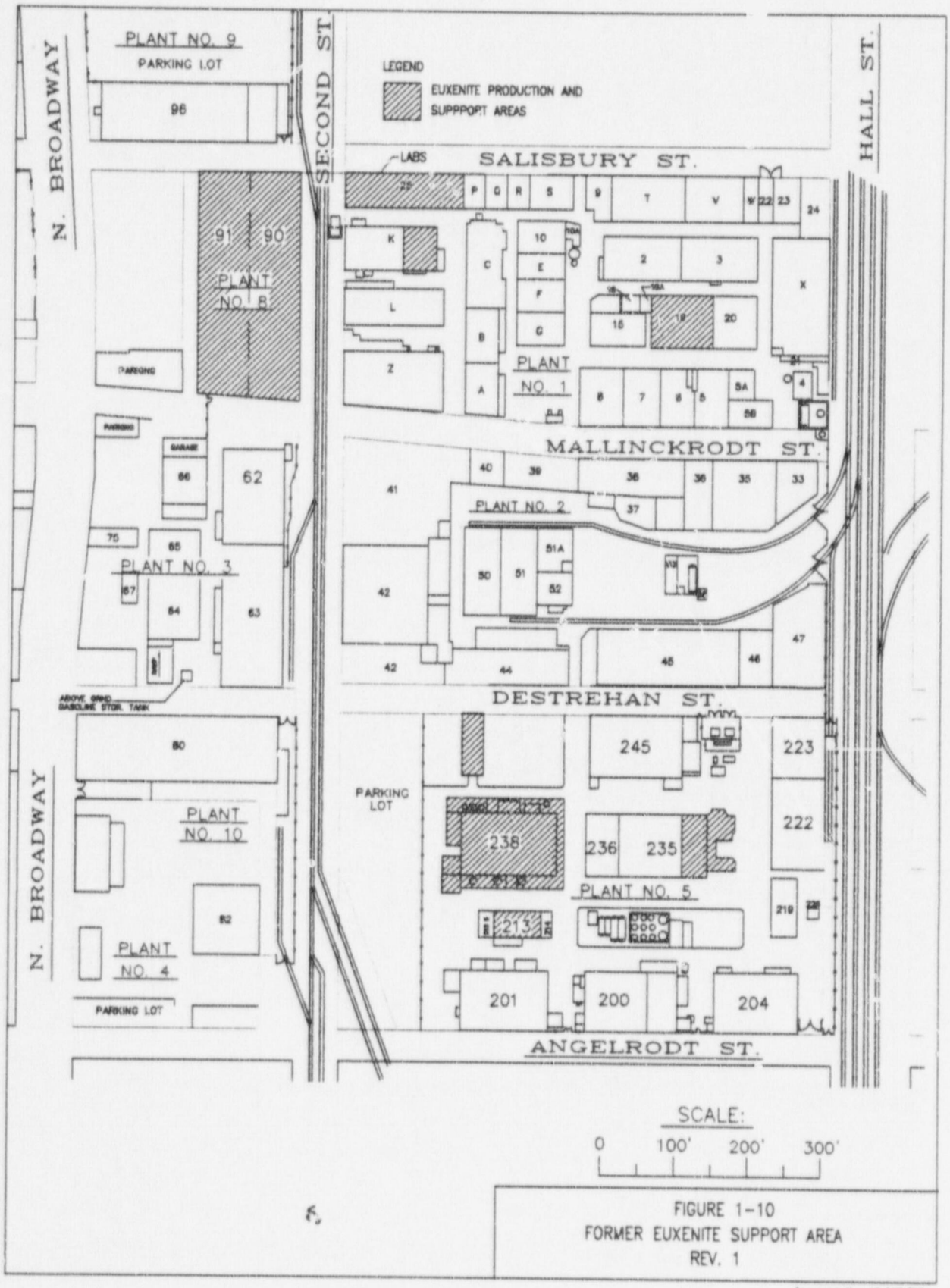
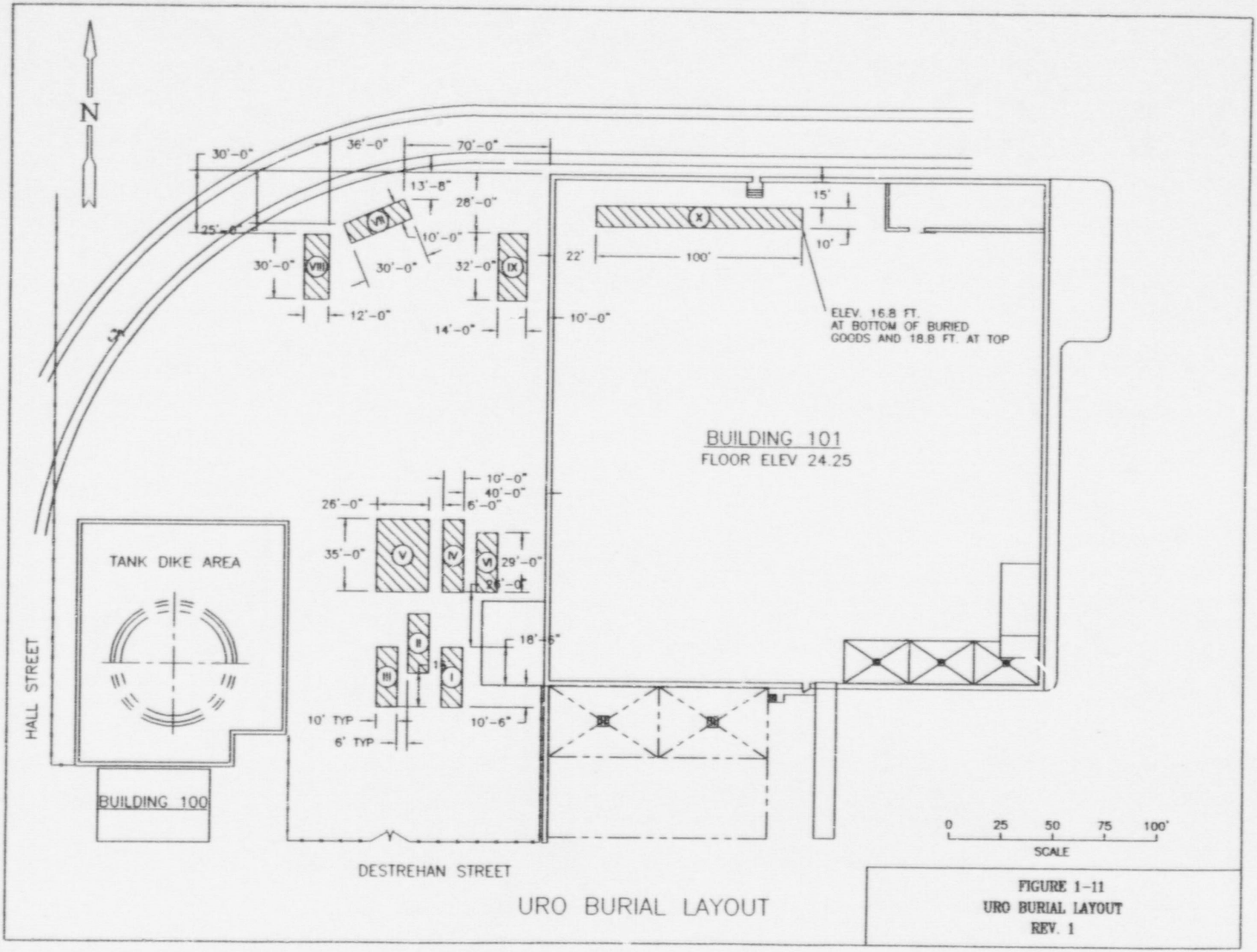


FIGURE 1-10
FORMER EUXENITE SUPPORT AREA
REV. 1

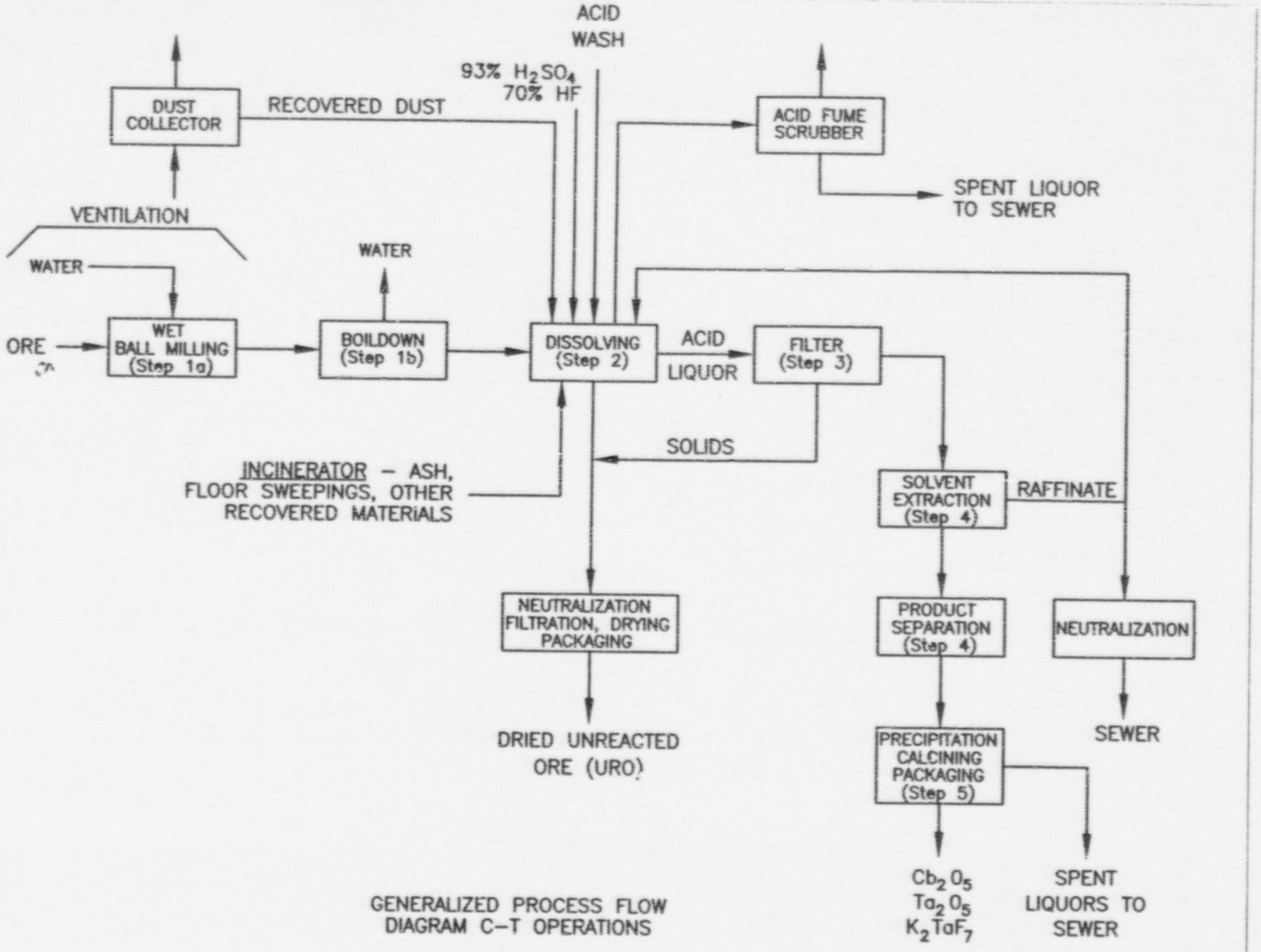
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URO BURIAL LAYOUT

FIGURE 1-11
URO BURIAL LAYOUT
REV. 1

fn = Uro_bury 8/7/97



GENERALIZED PROCESS FLOW DIAGRAM C-T OPERATIONS

FIGURE 1-12
GENERALIZED PROCESS FLOW DIAGRAM
C-T OPERATION
REV. 1

fn = Gen_proc 8/6/97

2. DECOMMISSIONING OBJECTIVES, ACTIVITIES AND TASKS 2-1

2.1 DECOMMISSIONING OBJECTIVE 2-1
2.2 DECOMMISSIONING CRITERIA 2-1
2.3 C-T PROJECT DECOMMISSIONING ACTIVITIES 2-4
2.3.1 Technical Approach to Decommissioning 2-6
2.3.2 Decommissioning Activities 2-7
2.4 PROJECT ORGANIZATION AND CONTROL 2-10
2.4.1 Project Organizational Structure 2-10
2.4.2 Managerial Responsibilities 2-11
2.4.3 Minimum Qualifications for Safety Related Positions 2-12
2.4.4 Work Controls 2-13
2.5 SCHEDULE 2-14

2. DECOMMISSIONING OBJECTIVES, ACTIVITIES AND TASKS

2.1 DECOMMISSIONING OBJECTIVE

The objective of the Phase I Plan is to remediate the C-T process equipment and buildings, and support buildings to obtain an unrestricted release. Equipment will be released in accordance with NRC "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material" dated November, 1983, as amended. Building surfaces will be released in accordance with 10 CFR 20, Subpart E.

2.2 DECOMMISSIONING CRITERIA

Based on C-T process history and site characterization, the radiological constituents addressed in this Phase I Plan are:

- U-238, U-235, and U-234, all from naturally occurring uranium (and their progeny Th-230, Ra-226, and other short-lived isotopes); and

- Th-232 from naturally occurring thorium, and its progeny (Ra-228, Th-228, and other short-lived isotopes).

The radioactive constituents of interest and release limits proposed in this Phase I Plan are listed in Table 2-1 for equipment surfaces, Table 2-2 for building surfaces, and Table 2-3 for building rubble. Each building has a different criterion based on its radiological history and actual field measurements. Only beta emission release limits are proposed. The rationale for using release limits based on a single type of radiation emitter is three-fold. First, direct measurement of alpha particles can be unreliable if the surface of a contaminated area has been painted or otherwise covered, due to energy absorption by the covering. Second, direct measurement of the gamma emitter concentration on building and equipment surfaces will not be representative due to a significant gamma contribution from subsurface areas. Third, the MDA for beta measurements will be lower than for combined beta-gamma since the beta background is only one-third of the combined beta-gamma background. These criteria apply to residual regulated radionuclides and exclude background radioactivity.

Equipment

The proposed surface beta activity levels on equipment that will be released for unrestricted use have been derived to be consistent with NRC "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material" dated November 1983. Table 2-1 lists the equipment surface beta release limits. The derivation of the limits is presented in Appendix D.

Table 2-1 Equipment Surface Release Limits by Location			
Equipment Location	Average (β/min/100 cm^2)	Maximum (β/min/100 cm^2)	Removable (β/min/100 cm^2)
Building 90/91	1100	3300	220
Building 101 Incinerator Pad	1800	5400	360
Building 101 Roof	1800	5400	360
Building 200 Roof	1560	4680	312
Building 201 Roof	1560	4680	312
Building 204 Roof	400	1200	80
Building 213	500	1500	100
Building 213 Roof	660	1980	132
Building 214	500	1500	100
Building 235 Roof	120	360	24
Building 236 Roof	150	450	24
Building 238	1000	3000	200
Building 238 Roof	1800	5400	360
Building 245	1560	4680	312
Building 245 Roof	1560	4680	312
Building 246	230	690	46
Building 246 Roof	1070	3210	214
Building 247	230	690	46
Building 247 Roof	470	1410	94
Building 248	430	1290	86
Building 248 Roof	490	1470	98
Building 250	1200	3600	240

Buildings

The residual isotopic surface beta activity levels that will be considered acceptable for unrestricted use of buildings that will remain on-site following decommissioning are presented in Table 2-2. The average surface beta limit in Table 2-2 is derived from a dose limit of 25 mrem/y for a worker assumed to be exposed as defined for the Worker-Building Occupation Scenario in NUREG/CR-5512. The corresponding surface beta limit was derived in two stages. In the first stage, a surface activity limit corresponding to the dose limit of 25 mrem/y is derived for each nuclide of interest. This derivation is described in Appendix C. In the second stage, the nuclide mix representative of each building is used in conjunction with the beta yield and the nuclide activity limits derived in Appendix C to derive a surface beta limit for each building. This derivation is described in Appendix D.

Table 2-2 Surface Release Limits for Individual Buildings	
Location	Average (β/min/100 cm^2)
Building 62	1560
Building 91 Crane Service Pad	700
Building 101 Roof/Pad	730
Building 200 Roof	600
Building 201 Roof	600
Building 204 Roof	600
Building 222 Roof	920
Building 223 Roof	1510
Building 235	910
Building 235 Roof	910
Building 236	600
Building 236 Roof	600
Building 240	1100
Building 240 Roof	1100
Building 245	1230
Building 245 Roof	1010
Building 250	830
Building 250 Roof	580

Building Rubble

Mass concentration limits are the appropriate criteria for building rubble. Buildings to be demolished will first be decontaminated with the goal of removing material that could cause the rubble to exceed limits for unrestricted use. If decontamination is not likely to be successful, decontamination will be conducted only to the extent necessary for health and safety controls, and the rubble will be isolated from other building rubble. After decontamination, the remaining building structure will be demolished, segregated as necessary, and volume reduced. The resulting rubble will be analyzed for radionuclide content; if the rubble can be distinguished from background and cannot be released for unrestricted use it will be managed as waste (See Section 3.6). Otherwise, the rubble will be released for unrestricted use. Table 2-3 guides the disposal of the bulk material for unrestricted release. The mass concentration release limits were derived as described in Appendix E.

Table 2-3 Bulk Material Concentration Release Limits	
Radionuclide	Release Criteria (pCi/g)
Total U	100
Th-230	7.2
Ra-226	1.7
Th-232	1.5

Since the thorium series will grow to radioactive equilibrium, the Th-232 limit is the indicator limit for the series.

The limits or guideline values presented in Table 2-3 will be applied using the sum-of-ratio approach for mixed radionuclides. Samples of rubble and suitable material from a reference area will be analyzed for radionuclide content and the following index will be calculated for each sample.

$$\frac{C_1}{L_1} + \frac{C_2}{L_2} + \frac{C_3}{L_3} + \cdots \frac{C_n}{L_n}$$

where: C_n = is the concentration of a particular radionuclide
L_n = the concentration limit of a particular radionuclide

For indices so calculated, the single-valued derived concentration guideline level (DCGL), the value of the index corresponding to the release criterion for a mixture of nuclides, is 1. Compliance with the release criterion is tested by statistical comparison of a statistical sample of indices from the survey unit against a statistical sample of adjusted indices from the referenced area. The reference area indices are adjusted by adding 1, the DCGL, to each index in the statistical sample. If the statistical comparison allows inference that the central tendency of the population of survey unit indices is not greater that the central tendency of the population of reference area indices, as adjusted, the release criterion is met.

This Phase I Plan does not include guideline values for soil, other subsurface material, or groundwater because they will be addressed in Phase II of the Plan.

2.3 C-T PROJECT DECOMMISSIONING ACTIVITIES

Most of the activities required to decommission the C-T project site will take place within Plant 5. However, certain structures and areas outside Plant 5 were also involved in C-T work and will also be remediated. Within Plant 5, the C-T Project remediation area boundaries have been defined as:

- everything south of the south edge of Destrehan Street,
- everything north of a line drawn along the south sides of Buildings 201, 200 and 204,
- everything west of a line drawn on the east side of Bldgs. 222 and 223, and
- everything east of a line drawn along the east sides of Buildings 240 and 250.

See Figure 2-1 for a map of the Plant 5 C-T Project remediation areas.

The following C-T support areas outside of Plant 5 will also be surveyed and, if necessary, remediated during Phase I:

- Building 62 in Plant 3,
- Building 101 area in Plant 6 (specifically the CT incinerator), and
- Buildings 90 and 91 in Plant 8.

See Figure 2-2 for a map of the St. Louis Plant C-T support areas requiring surveying and possible remediation.

Radiological surveys performed in 1996 on Buildings 201, and 245 indicate that those buildings qualify, without further endeavor, for release for unrestricted use. The radiation survey data for Buildings 201, and 245 will be incorporated into a Final Status Survey Report for NRC review and approval prior to license termination.

During the planning stages, alternative methods for decommissioning the C-T project site were evaluated. Some of the alternatives evaluated included various options for decontaminating buildings as well as equipment decontamination. Mallinckrodt, after a careful review of site and facility characterization data, research program results, and engineering cost analyses, has decided to: (a) demolish buildings 213, 214, 238, 246A&B, 247A&B, and 248; and (b) survey and decontaminate (if necessary) the remainder of the building structures and the equipment.

Decontamination and volume reduction methods were selected to minimize the volume of radioactive waste requiring disposal. This Phase I Plan is based on:

- scabbling the surface of concrete floors, pavement, block, and brick;
- decontaminating process equipment when it is judged to be cost-effective compared to disposal;

- decontaminating or removing selected contaminated areas of concrete, block, and brick to reduce the average mass concentration activity below the release limits; and

- crushing contaminated concrete, block, and brick to obtain a homogeneous material type where practical for improved representative sampling, and to increase the bulk density, thereby lowering the shipping volume and transportation costs.

2.3.1 Technical Approach to Decommissioning

The technical approach for decommissioning during Phase I is based on:

- removing all equipment and services from Buildings 213, 214, 238, 246A&B, 247A&B, and 248 in preparation for demolition;

- roof sampling and decontaminating roof and exterior surfaces of Buildings 101, 200, 201, 204, 222, 223, 235, 236, 240, 245, and 250, if necessary to prepare for the final radiological survey for release for unrestricted industrial use;

- locally decontaminating and surveying for release for unrestricted industrial use selected interior areas in Building 250;

- surveying for release for unrestricted industrial use Building 62;

- perform surveying for release for unrestricted industrial use Buildings 90, 91;

- remove incinerator on pad near Building 101;

- demolishing the interior walls, elevated floors, mezzanines and the remaining shells (walls and roofs only) of Buildings 213, 214, 238, 246A&B, 247A&B, and 248.

- crushing concrete slabs from elevated floors, block, and brick in preparation for sampling and disposal;

- packaging and shipping for transfer to a licensed facility or disposal at an appropriate waste disposal facility; and,

- removing or fixing any surface contamination remaining on the concrete floors after building demolition. Building ground-floor slabs will be removed during Phase II.

2.3.2 Decommissioning Activities

The activities to be performed during Phase I decommissioning are grouped below in several categories based on similar work.

- **Utility relocation, termination, and temporary connections**

  The C-T Project is served by overhead gas, electric and cooling water, and underground city water and fire mains. Prior to the start of decommissioning, all services which feed through the C-T project site, will be relocated to minimize utility disruption for the rest of the site.

  As work commences in each building to be demolished, all services to that building will be terminated. Contractor feed points will be established to allow the decommissioning contractor access to and control of temporary electricity, water, and other utilities as appropriate within the area. Utilities were already terminated in Building 238 prior to the application for a possession-only license.

  A new construction power source will be installed to provide electrical power service to the main site during its decommissioning. This service will likely be a 480-volt, 600-amp service. The new construction power source will be available at two connection points, currently planned to be near building 213 and near building 247, although other locations may be selected during detailed planning. Construction power will be run from these points into each building via temporary lines and connected to power carts for contractor use.

  Buildings where limited decontamination or removal activities are to take place (e.g., Building 250) have adequate power to support the decontamination activity, although it may be necessary to locally de-energize the area to be remediated and run temporary lines from distribution points elsewhere in the building.

- **Removal of C-T process equipment**

  The removal of process equipment, whether located inside or exterior to the buildings, will be performed using standard equipment. C-T process equipment will either be transferred to another licensee, decontaminated and released for unrestricted use, or reduced in size for disposal. The C-T incinerator located in Plant 6 will also be removed.

- **Interior demolition of the C-T Project process buildings**

  During demolition of the internal walls and floors, the building will be modified to provide high efficiency or HEPA filtered containment as appropriate. After removal of the equipment, interior walls, and upper level floors, the building will consist of four walls, the grade level floor slab, and a roof. Support systems and services such as ventilation, air monitoring, fire protection, and temporary and emergency power will be maintained, as practical, until the building demolition is complete.

- **Decontamination of C-T support facilities and other areas**

  The only C-T process equipment outside of Plant 5 is the incinerator near Building 101 in Plant 6. The incinerator will be reduced in size to efficiently fit into standard shipping containers.

  Localized contamination in Building 250 will be removed by scrubbing or abrasive cleaning, or removal of the contaminated item - e.g. lab drain piping. No other decontamination of C-T support buildings (buildings 90, 91, 62) is required. Contaminated roofing and/or roof equipment will be removed from buildings 204, 222, and 223.

  Potential radiological hazards due to waste generation during decontamination activities are expected to be minor. Liquid effluent will be minimized during decommissioning. Dry abrasive cleaning, when used, will result in airborne particulate. In such instances, containment zones will be established by erecting temporary walls of plastic, cloth, or other material around the remediation area to minimize the spread of contamination. The air will be exhausted (as appropriate) through roughing filters, prefilters, and high efficiency or HEPA filters, as appropriate, prior to discharge into the main building, and a negative pressure will be maintained in the enclosure relative to the rest of the building. The pressure differential will be verified through the use of a differential pressure gauge or smoke testing to determine the direction of airflow.

- **Roof asbestos removal for C-T Project process buildings**

  All asbestos abatement work will be performed in accordance with the EPA and Occupational Safety and Health Administration (OSHA) regulations and guidelines, State of Missouri asbestos regulations, and City of St. Louis ordinances. The buildings within the C-T project site that will be demolished have limited amounts of asbestos containing material (ACM). Buildings 213, 214, 238, 246A&B, 247A&B, and 248 have asbestos-containing, built-up roofing. The built-up roofing will be removed from the roof decking using a

powered roofing cutter while keeping the roof surface wet. The roof sections and cutting waste will be bagged and lowered to the ground for disposal as contaminated asbestos waste. The remaining portions of the roof will be sprayed with a fixative to encapsulate any remaining asbestos and or radioactive material that may exist. The roof deck will stay in place until final demolition of the balance of the building.

- **Demolition of the C-T Project process buildings**

At the time of demolition, the buildings will consist of exterior walls, the grade-level concrete floor slab and the roof. The exterior walls are constructed of two or three course block and/or brick walls, approximately 12 inches thick. Structural steel columns support the roof, and formerly supported the interior upper floors. Demolition of the exterior walls and roof will be performed in sections. The order of demolition will be removal of the brick and block sections between the structural steel columns, followed by removing the roof and demolishing the structural steel. Finally, if the upper surface of the grade-level concrete floor slab is still contaminated, the surface will either be decontaminated or have a fixative applied to it. Removal of the floor slab is part of Phase II Plan.

Removal of the brick and block will be performed with appropriate dust control, such as a high efficiency or HEPA filtered, as appropriate, temporary enclosure. Such methods provide adequate containment of potentially contaminated airborne particulate by establishing inward airflow and lowering the airborne particulate inside the enclosure to minimize worker exposure.

The preferred temporary enclosure will be a lean-to design and constructed in-place. The enclosure will consist of a rigid frame and a plastic sheet contamination barrier. The enclosure will be attached to the wall section to be removed with additional plastic sheeting and sealed at the ground with sandbags or equivalent. In this way, the remaining portion of the building will act as part of the enclosure.

After the brick and block have been removed from the enclosed section of the building, operations will be suspended and the enclosure moved to an adjacent building section. The openings between the enclosure and the building will again be sealed with plastic sheeting, and the demolition process repeated. Upon completing the progressive removal of the block and brick, the plastic sheeting enclosure frame will be disassembled. The materials will be packaged for proper disposal when no longer needed for building demolition.

The nonmetallic wall and roof rubble will be removed using standard construction equipment and transported to an on-site crushing area for further

processing and disposition. Crushing the building rubble will decrease the disposal volume and facilitate sampling, packaging, shipment, and disposal. The crushing will be accomplished using standard bulk materials handling equipment, such as conveyors, crushers, and storage bins. Mock-up tests will be performed to determine the proper methods for crushing the wall and roof rubble while keeping the amount of dust to an acceptable level.

Demolition waste will be managed as described in Section 3.6.

The Phase II Plan activities will include removal of the grade-level concrete floor slabs under the demolished buildings, asphalt and concrete pavement, and subsurface material. These activities will be defined in detail in the Phase II Plan. Phase II decommissioning is anticipated to commence after all Phase I activities are completed and the NRC has approved the Phase II Plan.

2.4 PROJECT ORGANIZATION AND CONTROL

This section describes the project organization that will become effective upon NRC approval of the Phase I Plan. As the project moves forward, changes to the project organization may be justified in response to the varying level of site activities. Effective with NRC approval of this plan, the licensee may make changes to the organizational structure provided: (1) the safety and quality functions maintain an independent reporting relationship from that of operations; and, (2) the individuals responsible for safety functions satisfy the educational and experience qualifications in Section 2.4.3.

2.4.1 Project Organizational Structure

The C-T decommissioning organizational structure will consist of a management team from Mallinckrodt and the decommissioning contractor. The C-T project decommissioning organization is illustrated in Figure 2-3.

The Mallinckrodt Site Environmental Manager/C-T Project Manager will provide overall direction for the C-T project decommissioning. The Project Manager will be supported by managers responsible for contract administration, environmental affairs, quality assurance, project engineering, and legal.

While Mallinckrodt will be responsible for ensuring that NRC requirements are met, Mallinckrodt will use the services of a qualified, experienced, NRC-licensed contractor. The contractor will provide the radiological, occupational, and environmental safety programs and the procedures to implement the program requirements. The contractor will also provide an independent quality assurance program, and provide and manage a trained, experienced labor force to perform the decommissioning activities.

The contractor's Project Manager will provide day-to-day direction for the C-T Project decommissioning, but will report directly to the Mallinckrodt Project Manager. The contractor's Project Manager will be supported by operations and environmental, safety and health personnel who will provide technical advice, resources, and day-to-day management of the decommissioning workers.

The responsibilities of the primary managers are provided in section 2.4.2. Section 2.4.3 identifies the minimum qualifications for managers with safety-related responsibilities.

2.4.2 Managerial Responsibilities

Mallinckrodt Site Environmental Manager/C-T Project Manager is responsible for ensuring that the overall C-T decommissioning project, and the work being performed by all the contractors, is in accordance with applicable health, safety, quality, technical, and contractual requirements. The Mallinckrodt Project Manager is responsible for assuring that the requirements of the NRC are met. He also is responsible for coordinating activities between Plant operations and the decommissioning contractor. He shall use other Mallinckrodt staff as appropriate to perform this coordination. The Mallinckrodt Project Manager has full authority to halt any operation which he believes has the potential to threaten the health and safety of personnel, the public, or the environment or is not meeting the requirements of the NRC. He is also responsible for ensuring that established environmental programs and contractor environmental programs are in compliance with applicable and relevant laws and regulations. The Site Environmental Manager/C-T Project Manager is the designated contact point with the NRC.

Mallinckrodt Site Radiation Safety Officer (SRSO) is responsible for ensuring that established programs and contractor radiation programs are in compliance with applicable and relevant laws and regulations and for auditing the contractor's programs. The SRSO advises the Mallinckrodt C-T Project Manager on matters pertaining to radiation safety. The SRSO reports to the Site Environmental Manager. The Mallinckrodt SRSO has the authority to halt any operation which he believes has the potential to threaten the health and safety of personnel, the public, or the environment.

Contractor's Project Manager is responsible for the execution of all of the decommissioning activities. He is the primary interface with the Mallinckrodt C-T Project Manager. The contractor's Project Manager is directly responsible for all field work being performed by the contractor. As such, he is responsible for field work being conducted in accordance with applicable health, safety, quality, and technical requirements, including Mallinckrodt procedures. The contractor's Project Manager has full authority to stop any activity when he believes these requirements are not being met.

Contractor's Environment, Safety and Health Representative is responsible for establishing and ensuring implementation of the contractor environmental and safety programs. This function reports directly to the contractor's Project Manager and is

functionally independent of Operations, thus assuring independence of action in matters pertaining to health and safety. The contractor's ES&H Representative has the authority to stop work on any decommissioning operation based on safety concerns.

Contractor Operations Manager reports directly to the contractor's Project Manager and receives program and task directives directly from the contractor's Project Manager. The contractor's Operations Manager is responsible for nuclear materials accounting, field engineering, waste tracking, daily work assignments for all field personnel and the physical execution of the decontamination and decommissioning activities for the C-T decommissioning project.

Contractor's Quality Assurance Representative is responsible for establishing and assuring implementation of the contractor quality assurance program. This function is independent of Operations and will report directly to the Mallinckrodt C-T Project Manager with copies of audits provided to the contractor's Project Manager. The details about how Quality Assurance will be incorporated into Phase I decommissioning are provided in Section 5.0.

2.4.3 Minimum Qualifications for Safety Related Positions

Mallinckrodt Project Manager must hold a baccalaureate degree in science or engineering and have a minimum of five years of project management experience.

Mallinckrodt SRSO must hold a baccalaureate degree in science or engineering and have a minimum of five years experience in radiation safety. The SRSO must also be knowledgeable in C-T chemistry and familiar with the C-T manufacturing process.

Contractor's Project Manager must hold (a) a baccalaureate degree in science or engineering and have a minimum of five years of experience in the nuclear industry, including five years of project management experience or (b) a baccalaureate degree in science or engineering and have a minimum of five years of construction experience with at least five years of experience in nuclear activities as well as five years of project management experience.

Contractor's ES&H Representative must hold a baccalaureate degree in science or engineering and have a minimum of five years experience in nuclear safety, health physics, industrial safety, or environmental safety, and five years of management experience.

Quality Assurance Representative must hold (a) a baccalaureate degree with at least two years experience in quality assurance or quality control.

2.4.4 Work Controls

Decommissioning activities for the C-T Project will be performed in accordance with written instructions. There will be five general types of written instructions in use for the C-T Project: Plans, Procedures, Field Instructions, Radiation Work Permits, and Daily Hot Work Permits. These written instructions will be reviewed and approved by the contractor operations representative and the ES&H representative, as well as being reviewed and approved by the Mallinckrodt Project Manager or his designated representative where applicable, e.g. when health, safety, and quality of the decommissioning issues are addressed.

Plans are broad-based documents that provide management guidance for operations in the field. Plans include decommissioning contractor developed plans such as a Health and Safety Plan, ALARA Plan, Waste Management Plan, Final Survey Plan, Quality Assurance Plan, and the Industrial Safety Plan. All plans affecting the health, safety, and quality of the decommissioning will be reviewed by Mallinckrodt. Specific procedures, field instructions, and radiation work permits, as discussed below implement these programs and plans.

Procedures are essential, written instructions and specifications to provide the controls needed to ensure safety and other objectives of the procedure are achieved. A procedure is ordinarily appropriate for repetitive activities such as defining how to operate equipment, calibration methods, or other routine work activities.

Field Instructions are the primary written instructions for decontamination and decommissioning work for the C-T Project. They may be as short as a simple job order (e.g. run temporary electricity into Building XX), or as detailed as necessary to safely accomplish the work. The detailed field instructions provide specific instructions in a logical and sequenced manner for one-time or short duration activities requiring a disciplined approach to ensure that health and safety requirements are met.

Radiation Work Permits (RWPs) specify necessary radiation safety controls, including personnel monitoring, monitoring devices, protective clothing, respiratory protection equipment, special air sampling, and additional precautionary measures. RWPs are issued for non-routine activities where there is a need to prescribe the conditions under which the work may be done in order to assure adequate protection of workers and the public from the potential radiological hazards that may be encountered. RWPs required for decommissioning activities are usually identified during the preparation and review of procedures and field instructions. RWPs will be prepared in accordance with the contractors written procedure that will be reviewed by Mallinckrodt.

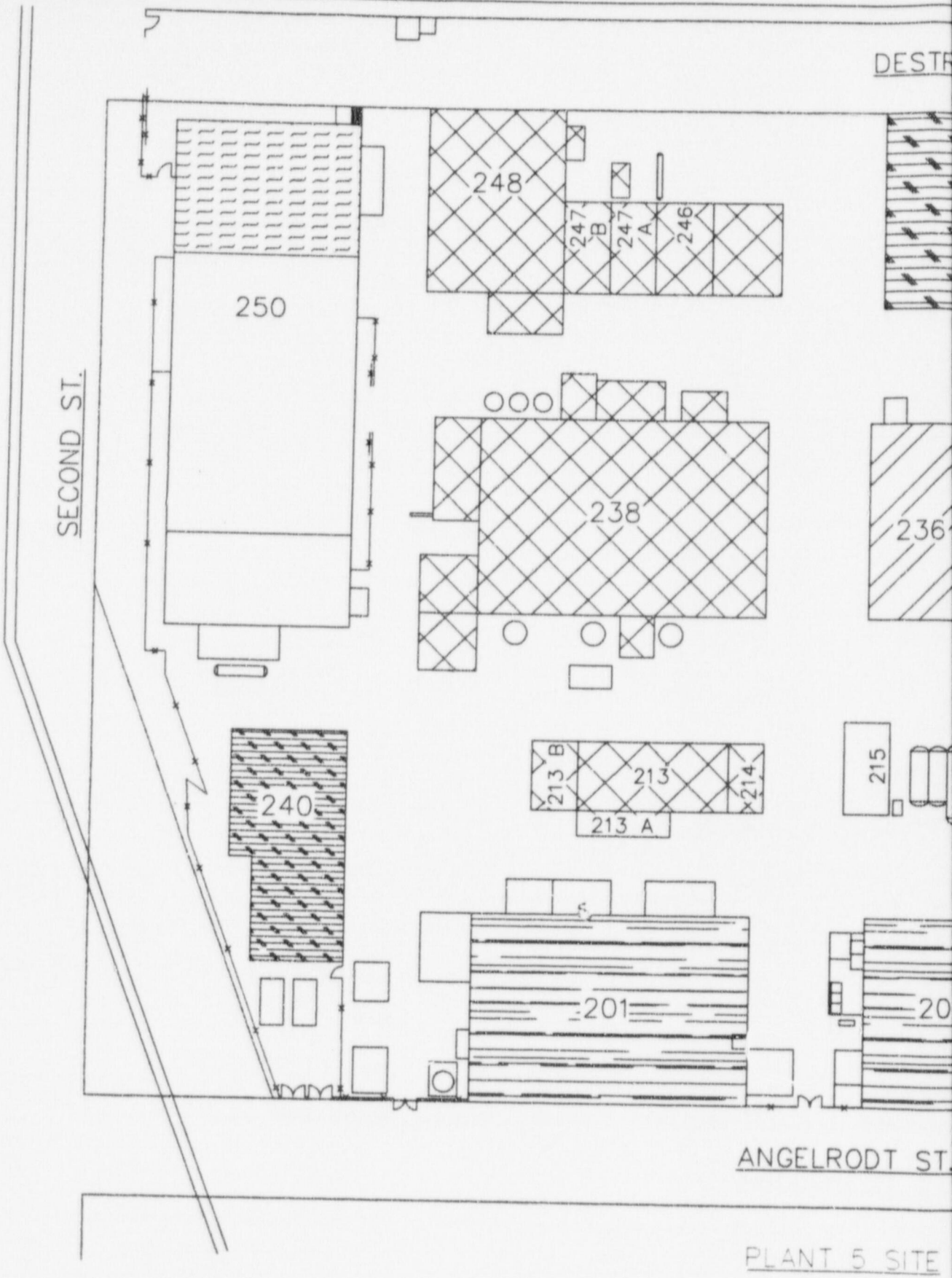
Daily Hot Work Permits check for hazardous conditions, allow the use of spark generating tools and equipment, ensure adequate ventilation, etc.

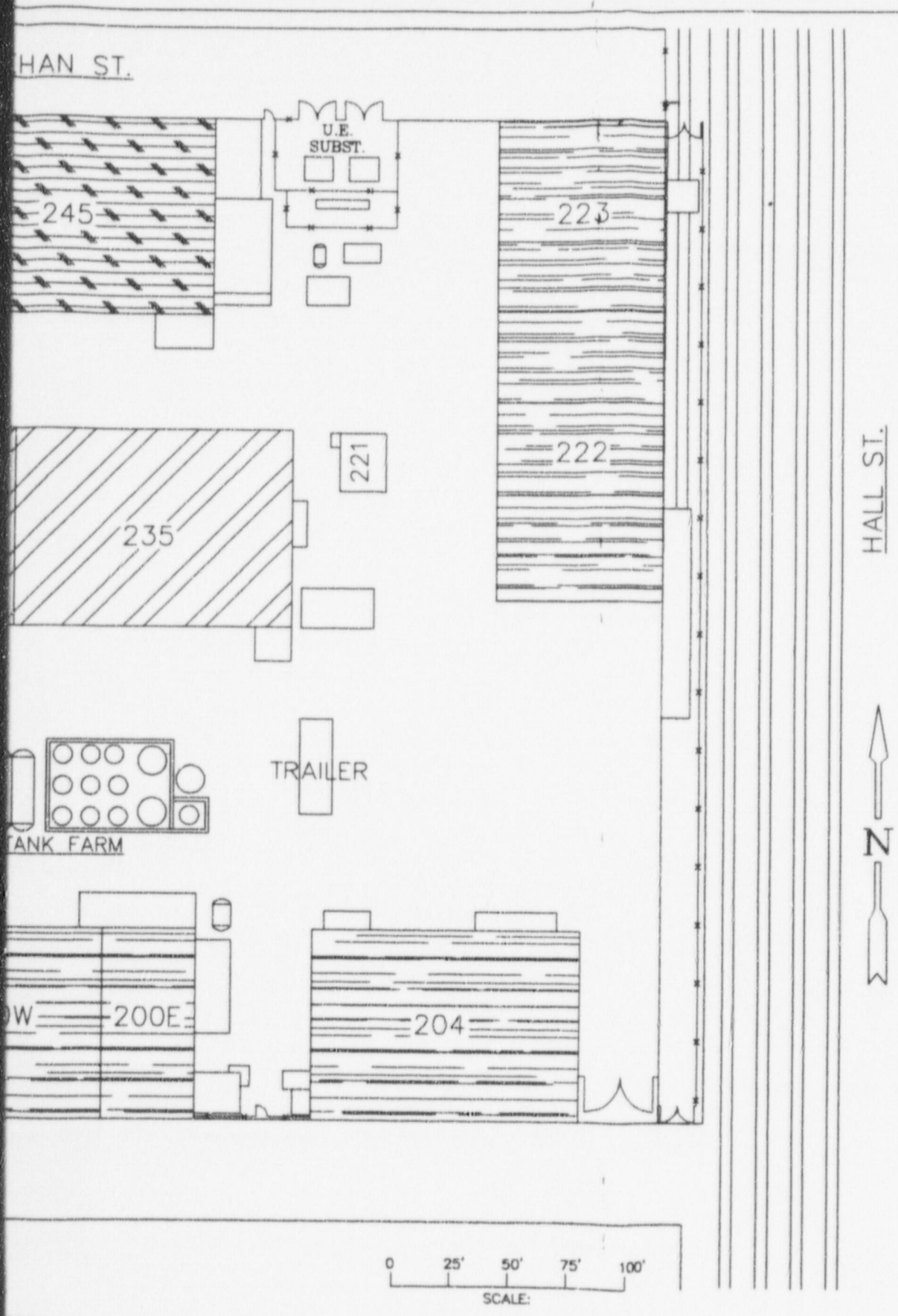
2.5 SCHEDULE

A preliminary schedule has been developed for the Phase I decommissioning activities described in Section 2.3. This schedule is presented in Appendix F.

The key elements forming the bases for the schedule are:

- removing the equipment from the C-T project site buildings will take approximately 8 months;
- demolishing the buildings in Plant 5 will take approximately 12 months;
- relocating the utilities will take approximately 6 months and certain utility relocation must be completed prior to the start of demolition; and
- completing the final radiation surveys and NRC verification that the residual radioactivity limits have been met will occur concurrent with other activities and not impact the length of the Phase I schedule.


DESTR
SECOND ST.
250
248
247 B
247 A
246
238
236
240
213 B
213
214
213 A
215
201
20
ANGELRODT ST.
PLANT 5 SITE
DEMOLISH
SURFACE DECON AND EQUIPMENT REMOVAL
ROOF SAMPLING AND DECONTAMINATION
SURVEY AND RELEASE
LOCAL
SURVEY



ANSTEC APERTURE CARD

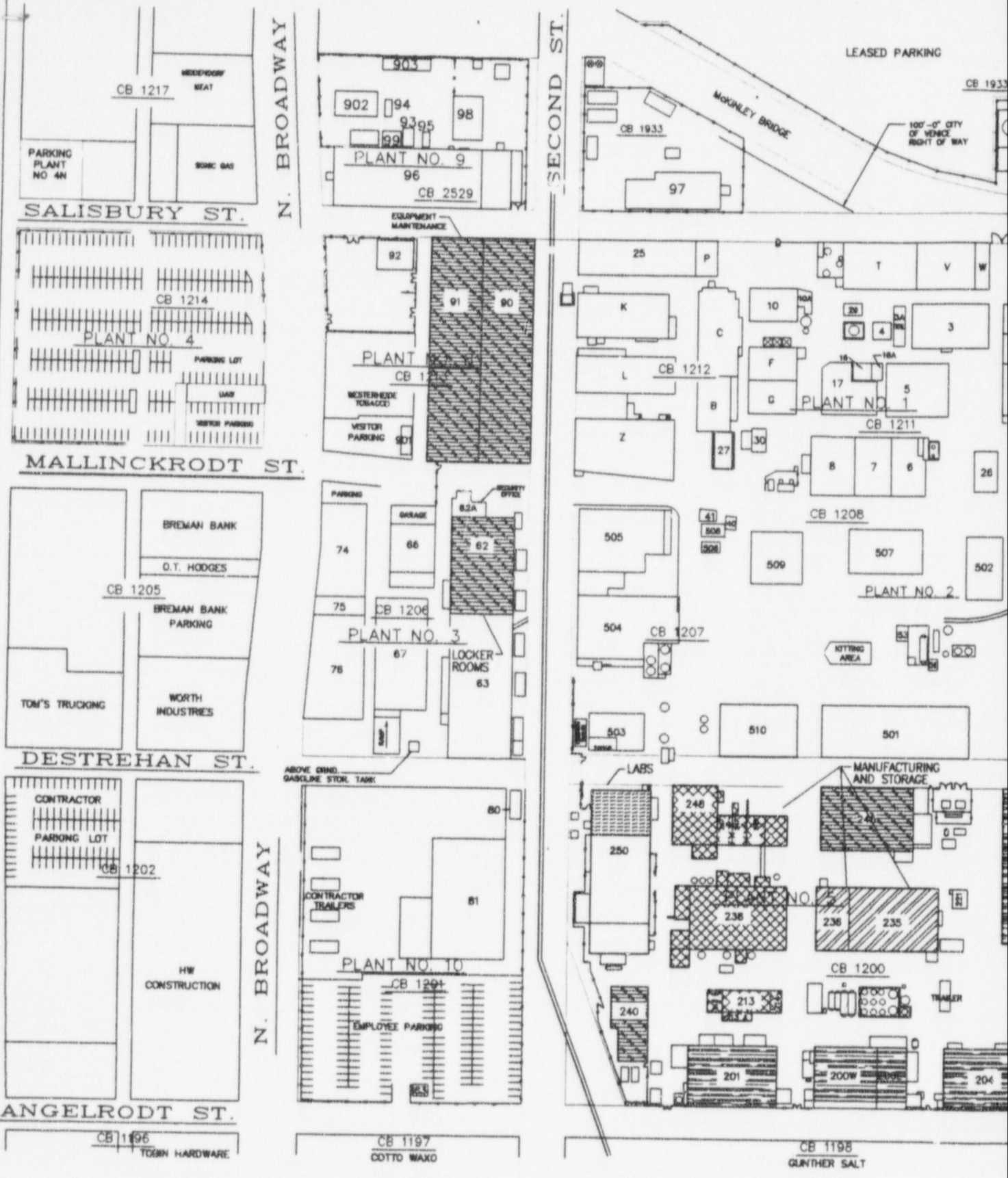
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ECON
AND RELEASE

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FIGURE 2-1
C-T PRODUCTION PROCESS & SUPPORT AREAS
REV. 1

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SALISBURY ST.
MALLINCKRODT ST.
DESTREHAN ST.
ANGELRODT ST.
N. BROADWAY
N. BROADWAY
SECOND ST.
CB 1217
PARKING PLANT NO 4N
MIDDENDORF MEAT
SOMC GAS
CB 1214
PLANT NO. 4
PARKING LOT
LOAD
VISITOR PARKING
903
902
94
93
95
98
99
PLANT NO. 9
96
CB 2529
EQUIPMENT MAINTENANCE
92
91
90
PLANT NO. 9
CB 1213
WESTERNHEIDE TOBACCO
VISITOR PARKING
BREMAN BANK
O.T. HODGES
CB 1205
BREMAN BANK PARKING
TOM'S TRUCKING
WORTH INDUSTRIES
PARKING
GARAGE
SECURITY OFFICE
82A
74
66
62
75
CB 1206
PLANT NO. 3
76
67
LOCKER ROOMS
63
ABOVE GRND. GASOLINE STOR. TANK
CONTRACTOR
PARKING LOT
CB 1202
HW CONSTRUCTION
80
CONTRACTOR TRAILERS
81
PLANT NO. 10
CB 1201
EMPLOYEE PARKING
CB 1196
TOBIN HARDWARE
CB 1197
COTTO WAXO
LEASED PARKING
CB 1933
McKINLEY BRIDGE
100'-0" CITY OF VENICE RIGHT OF WAY
97
25
P
T
V
W
K
C
10
10A
20
4
3A
3
L
CB 1212
F
G
B
Z
27
30
18
18A
17
5
PLANT NO. 1
CB 1211
8
7
6
26
41
40
506
508
505
509
507
502
CB 1208
PLANT NO. 2
504
CB 1207
KITTING AREA
503
510
501
LABS
MANUFACTURING AND STORAGE
250
246
248
PLANT NO. 5
238
236
235
CB 1200
240
213
TRAILER
201
200W
204
DEMOLISH
SURFACE DECON AND EQUIPMENT REMOVAL
ROOF SAMPLING AND DECONTAMINATION
SURVEY AND RELEASE
LOCAL DECON SURVEY AND RELEASE
EQUIPMENT REMOVAL LOCAL DECON


SALISBURY ST.
N. BROADWAY
SECOND ST.
MALLINCKRODT ST.
DESTREHAN ST.
ANGELRODT ST.
N. BROADWAY
LEASED PARKING
McKINLEY BRIDGE
CB 1217
CB 1214
CB 1933
CB 2529
CB 1212
CB 1211
CB 1208
CB 1207
CB 1206
CB 1205
CB 1202
CB 1201
CB 1200
CB 1196
CB 1197
CB 1198
PLANT NO. 4
PLANT NO. 9
PLANT NO. 1
PLANT NO. 2
PLANT NO. 3
PLANT NO. 10
PARKING PLANT NO 4N
PARKING LOT
VISITOR PARKING
EQUIPMENT MAINTENANCE
BREMAN BANK
O.T. HODGES
BREMAN BANK PARKING
TOM'S TRUCKING
WORTH INDUSTRIES
CONTRACTOR PARKING LOT
H/W CONSTRUCTION
GARAGE
LOCKER ROOMS
ABOVE GRND. GASOLINE STOR. TANK
CONTRACTOR TRAILERS
EMPLOYEE PARKING
LABS
MANUFACTURING AND STORAGE
TRAILER
TOBIN HARDWARE
COTTO WAXO
GUNTHER SALT
DEMOLISH
SURFACE DECON AND EQUIPMENT REMOVAL
ROOF SAMPLING AND DECONTAMINATION
SURVEY AND RELEASE
LOCAL DECON SURVEY AND RELEASE
EQUIPMENT REMOVAL LOCAL DECON

ANSTEC APERTURE CARD
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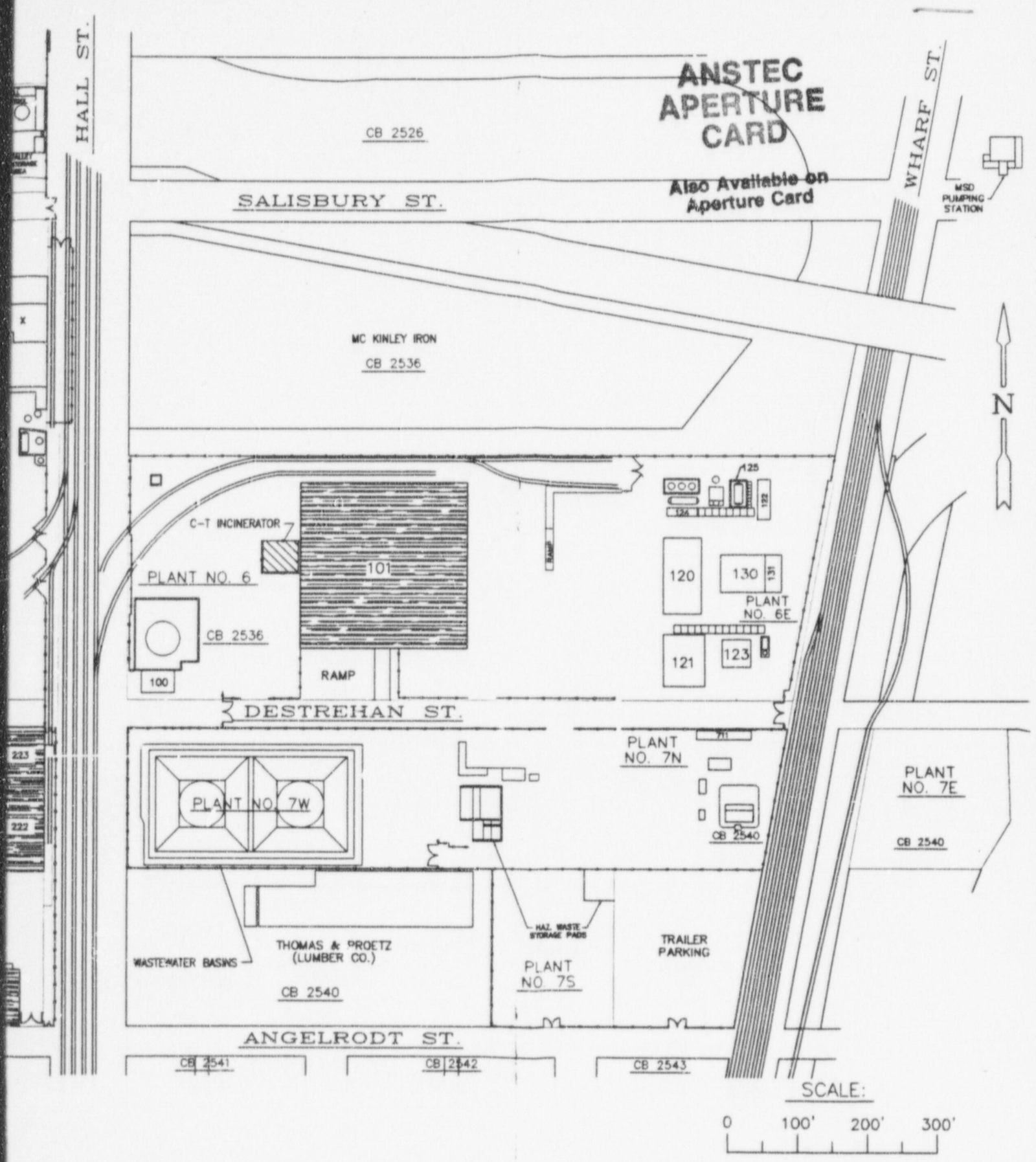


FIGURE 2-2
C-T PROJECT SITE SUPPORT AREAS
TO BE ADDRESSED IN PHASE I
REV. 1

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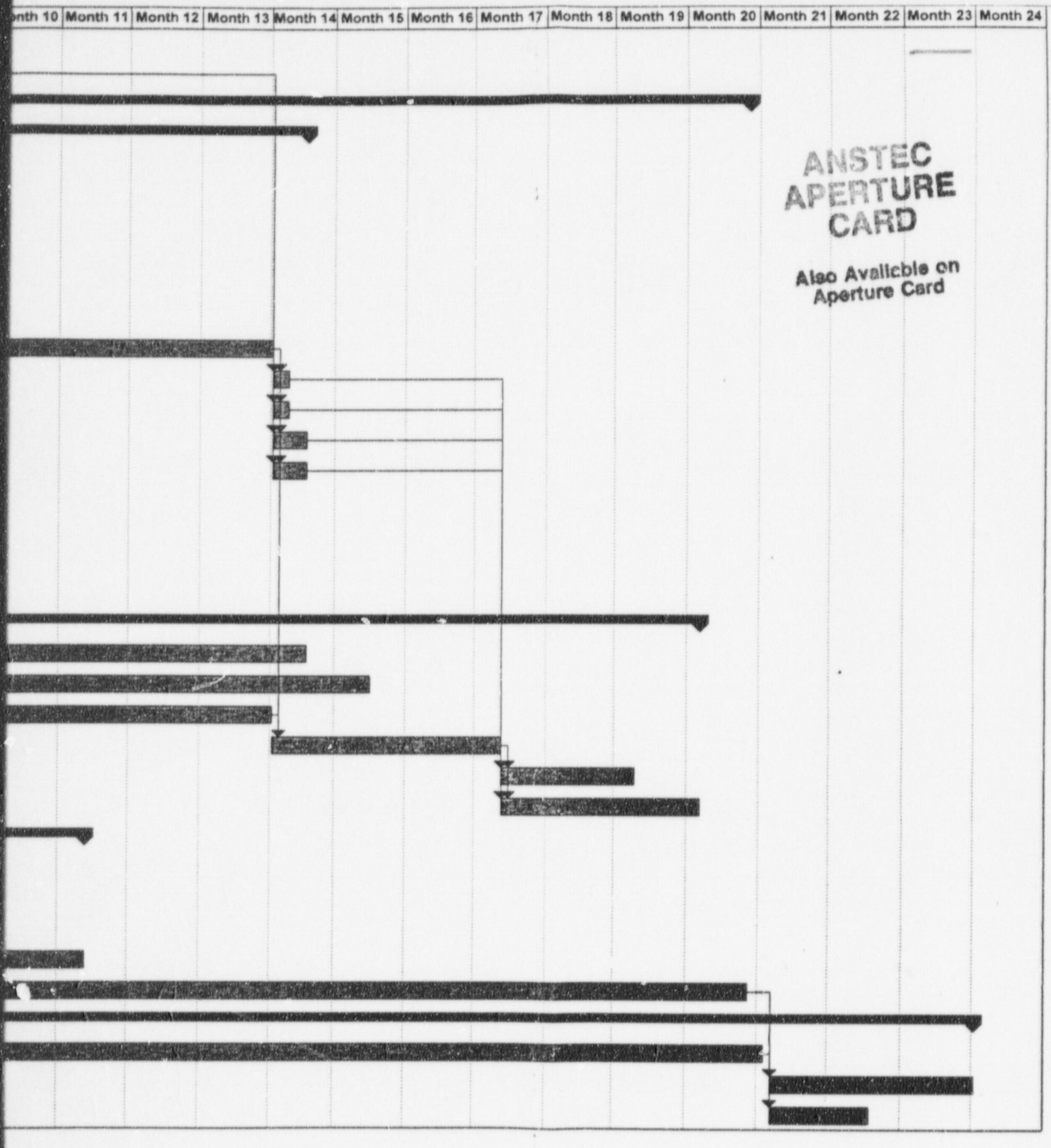
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ID	Task Name	Duration
1	Site Preparation	60d
2	Contractor Mobilization	30d
3	Site Activities	325d
4	Equipment Removal	190d
5	Non Buildings	30d
6	Equipment: Building 248	30d
7	Equipment: Building 247	20d
8	Equipment: Building 247A	20d
9	Equipment: Building 246A	20d
10	Equipment: Building 246	10d
11	Equipment: Building 238	80d
12	Equipment: Building 214	5d
13	Equipment: Building 213B	5d
14	Equipment: Building 213	10d
15	Equipment: Building 213A	10d
16	Equipment: Group C Buildings	20d
17	Equipment: Group D Buildings	30d
18	Equipment: Group E Buildings	70d
19	Equipment: Group F Buildings	30d
20	Demolition	210d
21	Demo: Building 248	90d
22	Demo: Building 247	110d
23	Demo: Building 246	80d
24	Demo: Building 238	70d
25	Demo: Building 214	40d
26	Demo: Building 213	60d
27	Decontamination	110d
28	Decon: Group C Buildings	10d
29	Decon: Group D Buildings	10d
30	Decon: Group E Buildings	30d
31	Decon: Group F Buildings	30d
32	Waste Shipping	225d
33	Final Survey	360d
34	Final Surveys	300d
35	Final Survey Report	60d
36	Demobilization	30d



Project: C-T D&D Phase I Schedule 1
Date: Wed 11/19/97

Task
Progress
Milestone
Summary
Rolled Up Task
Rolled Up Milestone



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Rolled Up Progress

ACRONYM LIST

Acronym	Definition
ACM	asbestos containing material
AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
C-T	Columbium-Tantalum
DCGL	Derived Concentration Guideline Level
DEA	Drug Enforcement Agency
DOE	Department of Energy
EMC	Elevated Measurement Comparison
EPA	Environmental Protection Agency
ERDA	Energy Research and Development Agency
ES&H	Environment, Safety and Health
FFA	Federal Facilities Agreement
FSSR	Final Status Survey Report
FUSRAP	Formerly Utilized Site Remedial Action Program
GSA	General Services Administration
LLRW	low level radioactive waste
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MCW	Mallinckrodt Chemical Works
MDA	minimum detectable activity
MDNR	Missouri Department of Natural Resources
MED	Manhattan Engineer District
MIBK	methyl isobutylketone
NIST	National Institute of Science and Technology
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
PRI	Preliminary Radiological Investigation
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	quality control
QIP	Quality Implementing Procedure
RCRA	Resource, Conservation, and Recovery Act
RWP	Radiation Work Permit
SRSO	Site Radiation Safety Officer

TLD	thermoluminescent dosimeter
TSCA	Toxic Substances Control Act
URO	unreacted ore
WRS	Wilcoxon Rank Sum

Mallinckrodt
C-T Decommissioning
Organization

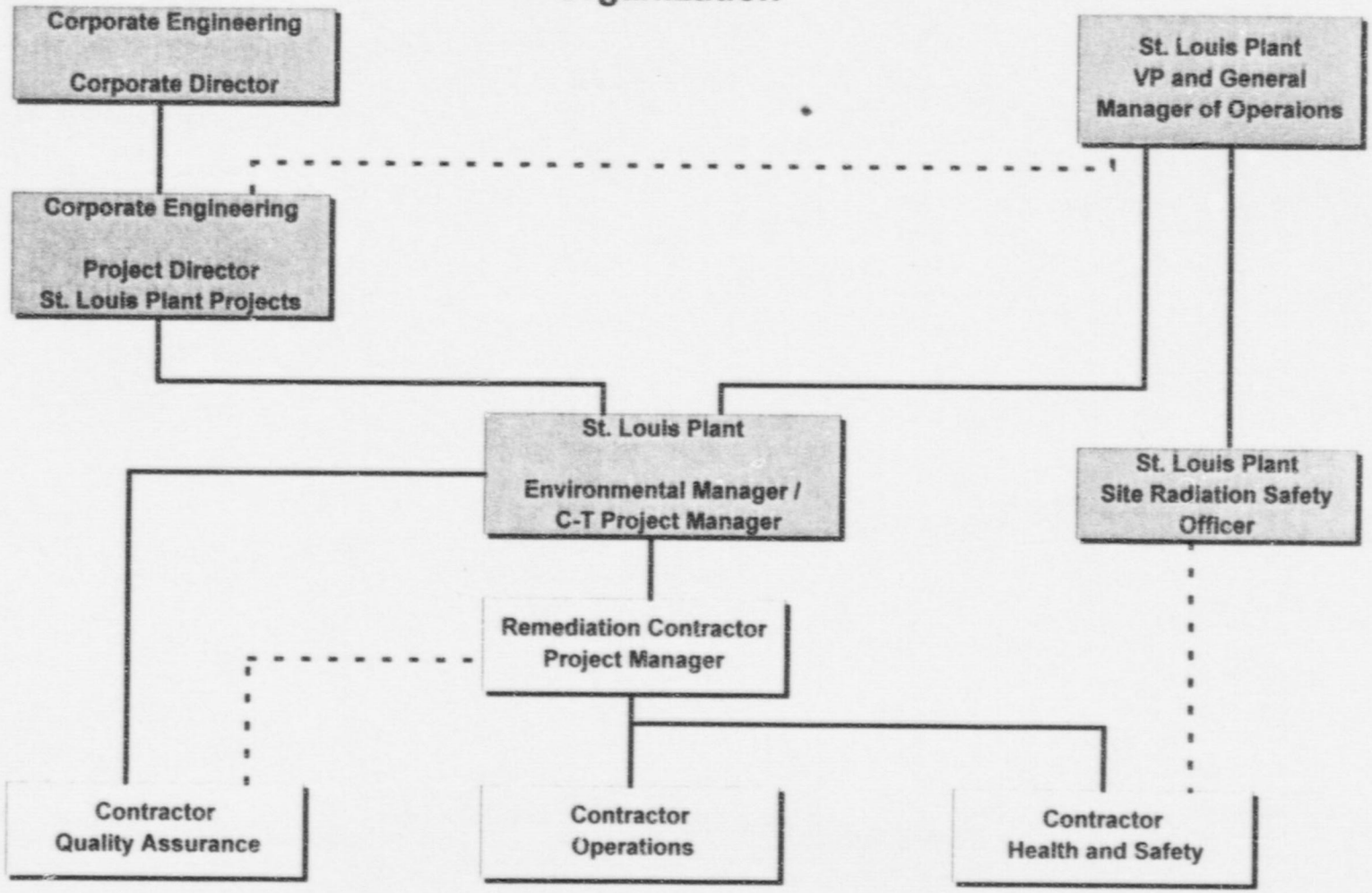


Figure 2-3
C-T Project
Decommissioning Organization

3. PROTECTION OF WORKERS, THE PUBLIC AND THE ENVIRONMENT AND RADIOACTIVE WASTE MANAGEMENT 3-1

3.1 HEALTH AND SAFETY PROGRAM 3-1
3.2 INDUSTRIAL SAFETY PROGRAM 3-1
3.2.1 Industrial Safety Training 3-2
3.3 RADIATION PROTECTION PROGRAM 3-4
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3.6 RADIOACTIVE WASTE MANAGEMENT 3-10
3.6.1 Management of Radioactive Waste 3-10
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3.6.6 Mixed Waste 3-11
3.6.7 Records 3-11

3. PROTECTION OF WORKERS, THE PUBLIC AND THE ENVIRONMENT AND RADIOACTIVE WASTE MANAGEMENT

3.1 HEALTH AND SAFETY PROGRAM

This section describes the measures to protect workers, the public, and the environment during remediation. A Health and Safety Program for the C-T Project will be developed to ensure the safety of all contractors and Mallinckrodt employees, visitors, and members of the public during decommissioning. In recognition that both the amount of radioactivity and the general safety hazards will be reduced as the project progresses, the Health and Safety Program may be modified to be commensurate with the activities being performed. Mallinckrodt will review and approve the Health and Safety Program, and any revisions that are made during the project.

The remediation contractor's ES&H Representative may authorize modifications to the Health and Safety Program under the following conditions:

- The modifications or changes will not reduce the effectiveness of the Health and Safety Program for the activities being performed.
- The modification remains compliant with Mallinckrodt's NRC license STB-401.
- A Safety Evaluation for the safety area to be modified or changed has been performed and approved by the remediation contractor's ES&H Representative, and Project Manager, and Mallinckrodt's Radiation Safety Officer and Project Manager.
- Copies of the Safety Evaluations will be maintained on-site for review by representatives of the NRC and other regulatory agencies.

The Health and Safety Program will consist of the following three parts:

- Industrial Safety Program
- Radiation Protection Program
- Environmental Safety Program.

3.2 INDUSTRIAL SAFETY PROGRAM

An industrial safety program will be adopted for the C-T Project that will augment the existing Mallinckrodt Health, Safety and Environmental Guidelines Manual by adding procedures specific to decommissioning. The remediation contractor will review the

existing Mallinckrodt procedures and supplement them, as necessary, with additional procedures. The total package of procedures will then form the Industrial Safety Program for the C-T Project. Table 3-1 lists industrial safety procedures that will be required for the planned remediation activities.

3.2.1 Industrial Safety Training

The St. Louis Plant site-wide industrial safety program will be used for training all unescorted individuals involved in decommissioning activities at the C-T project. The purpose of the program is to promote an awareness of the potential risks, and to provide knowledge and proficiency in industrial safety consistent with the assigned tasks. Training takes place on a continuing basis.

Personnel involved in C-T Project will be trained to perform their assigned responsibilities safely. On-the-job training and equipment-specific training will supplement the Mallinckrodt site-wide training program. Training in the proper use of specialized equipment is given before the individual uses that equipment. Credit may be given for applicable training received off-site.

The primary objectives of the industrial safety training program for the C-T Project.

- provide information on the industrial safety and hygiene hazards associated with working at the C-T Project ,and the steps to be taken to provide a safe work environment including those hazards unique to building demolition;

- enable each person to comply with plant rules and respond properly to warnings and alarms under normal and accidental conditions; and,

- enable individuals to recognize potential site specific hazards and to take appropriate measures to prevent personal injury or damage to facilities and equipment.

Table 3-1 Required Industrial Safety Procedures
Accident Investigation, Reporting, and Recordkeeping
Safety Color Code for Marking Physical Hazards
Working with Hazardous Chemicals
Welding and Thermal Cutting
Posting Requirements
Safety Training
Selection and Use of Personal Protective Equipment
Aerial Work Platforms
Scaffolds
Housekeeping
Equipment Lockout/Tagout
Operation of Lifting and Handling Equipment
Hand and Portable Power Tools
Electrical
Fall Protection
Guarding Floor Holes and Openings, Wall Holes and Openings
Fire Protection Program
Permit-Required Confined Space Entry Program
Excavation, Trenching, and Shoring
Drum Handling Procedure
Operation of Motorized Vehicles and Mechanized Equipment
Handling, Use, and Storage of Compressed Gas Cylinders
Manual Lifting
Job Safety Analysis Procedure
Heat Stress Program

The industrial safety training program will be reviewed and revised as needed to meet changing conditions and ensure that instructions are sufficiently well understood to permit practical application. The status and extent of the training of each individual will be documented to verify that workers are adequately trained for each assigned job.

The industrial safety training program includes:

- weekly shop-talks on pertinent industrial safety information, injury statistics, and specific safety issues;

- specific training on specialized equipment including the use of cranes, forklift trucks, front-end loaders, and scissors lifts;

- general industrial safety training including proper lifting, hearing conservation, eye protection, slips and falls, hazardous material handling, and use of power tools; and,

- specialized training including first aid, CPR, fire fighting, use of respirators, and HAZWOPER.

- "safe" work permits such as confined space entry, asbestos removal, lock-out/tag-out, etc.

3.3 RADIATION PROTECTION PROGRAM

The Radiation Protection Program will consist of procedures to protect workers, the public, and the environment from ionizing radiation, and an ALARA Plan that will provide guidance in keeping radiation exposures "As Low As Reasonably Achievable."

A radiation protection program will be adopted for the C-T Project that addresses the following topics. The contractor will be required to implement the program with oversight by the Site RSO.

- health and safety protection measures and policies;

- the ALARA Plan;

- instrumentation, calibration, and equipment;

- use of lapel air samplers, monitoring policy methods, frequency, and procedures;

- contamination control and personnel decontamination;

- internal and external exposure control;

- bioassay program;

- airborne releases and monitoring;

- respiratory protection;

- Radiation Work Permits;

- engineering controls;

- transportation

- the General Emergency Plan;
- posting and labeling;
- records and reports; and,
- potential sources of contamination exposure.

3.3.1 ALARA Plan

The remediation contractor will be required to develop and implement an ALARA Plan to ensure that radiation exposures to workers, the public and the environment are maintained ALARA. In the ALARA Plan, the remediation contractor will commit to provide an ALARA Coordinator who is trained in health physics, radiation protection, and has experience as an ALARA Coordinator. The ALARA Coordinator will review and approve all plans, procedures, field instructions, and RWPs and review environmental effluent releases for trends. The plan will reflect a strong management commitment to training workers on radiation protection principles, to monitoring and controlling occupational exposure, and monitoring and controlling environmental effluents.

The ALARA Plan will require the remediation contractor to provide for and maintain a trained work force. This is the responsibility of project management, EH&S, the ALARA Coordinator and all personnel. The roles assumed by each include:

- **Contractor Project Manager** is required to a) use only trained and qualified personnel; b) identify those persons to be employed at jobs or in areas requiring safety training, qualification training, or procedural training; c) make personnel available for training, or renewal/refresher training as required; d) provide task and daily safety review briefings to work crews; e) identify possible "lessons learned" topics for training; and f) provide specialized training as identified in the ALARA and RWP review and approval process.

- **Contractor ES&H** representative will ensure that personnel identified by the Project Manager as performing tasks requiring safety training will have current and appropriate training for the area or task. Contractor ES&H will also be responsible for identifying additional training needs during the ALARA, RWP, incident investigation, and permit review processes, including "lessons learned" topics. This person or their designee will also serve as the **ALARA Coordinator** to review and approve all plans, procedures, field instructions, and RWPs to ensure that remediation activities are performed such that exposures are maintained "As Low As Reasonably Achievable."

- **All personnel** are responsible for applying training to their work, notifying management when training is lacking for an assigned task, and identifying safety issues for "lessons learned" training to the project manager, training coordinator, or ES&H Manager.

The ALARA Plan will require the contractor to monitor the operations to identify needed upgrades in engineering controls (including equipment, containment, remote handling systems, and operating systems) or administrative controls (procedures, etc.) which, when implemented, will reduce the potential for radiation exposures to workers, the public and the environment. The management positions responsible for radiation protection and maintaining occupational exposure ALARA will be described in the plan.

The contractor's commitment to the ALARA concept will be further emphasized in radiation management control programs, such as using the RWP program with its interrelationship with the ALARA Plan, which are responsive to early indications of potential problems. In addition, implementing instructions to workers will stress the importance of continuous effective exposure control.

Management attention also focuses upon operating conditions that require modification for reduced exposure. Major facility changes and equipment and process development programs during the C-T Project are subject to safety reviews by Mallinckrodt to assure that ALARA has been addressed in proposed project designs.

Finally, the ALARA Plan will require monitoring of environmental effluents. The contractor's ES&H personnel will develop a system to monitor, organize, and report exposure and effluent data for measuring and assessing trends. By setting action levels and investigating exposures and effluents above these levels, problem areas will be readily identified and remedied.

3.3.2 Radiation Safety Training

All unescorted individuals involved in decommissioning activities for the C-T Project will be required to complete the Mallinckrodt radiation safety training course or the contractor equivalent course. The purpose of the training is to increase awareness of the potential radiation risks during decommissioning, and to provide a level of proficiency in personal radiation protective measures consistent with assigned tasks. On-the-job training, as deemed necessary by the contractor ES&H personnel, will be used to complement the formal radiation safety training.

All individuals will be trained before entering a controlled area to perform work. The safety performance of each individual will be reviewed annually, and workers will be retrained every two years. Credit may be given for applicable training received off-site, but plant-specific training is required for all decommissioning personnel. Training and examination results will be formally documented.

The primary objectives of the radiation safety training program are to:

- provide information on the biological effects of ionizing radiation, the potential risks associated with radiation exposure, and the basis for biological risk estimates;

- enable each person to comply with plant rules and respond properly to warnings and alarms under normal and emergency conditions; and,

- enable individuals to keep their own radiation exposures ALARA and to effectively apply ALARA considerations in making decisions that affect the radiation exposure of other workers, the public and the environment.

The radiation safety training will be reviewed and revised as appropriate to meet changing conditions and ensure that instructions are sufficiently well understood to permit practical application.

The radiation safety training program includes the following topics:

- radiation fundamentals - basic characteristics of radiation and contamination;

- radiation exposure limits, administrative control levels, and controls - external radiation exposure control methods, procedures, and equipment;

- radiation contamination limits and controls - contamination and internal radiation exposure control methods, procedures, and equipment;

- contaminated materials associated with decommissioning work - potential radiological problems;

- radiological work planning - integrating radiation safety and operational requirements to ensure safe conduct of work;

- emergency procedures and systems - work related information and actions;

- biological effects of radiation - basic understanding of biological effects and methods of assessment; and,

- the Radiation Protection Program.

3.4 ENVIRONMENTAL SAFETY PROGRAM

An Environmental Safety Program will be developed and implemented which will be designed to monitor air and water effluents discharged from the C-T Project during decommissioning. The program will be reviewed and approved by Mallinckrodt prior to implementation. Samples will be routinely collected or measurements routinely made at on-site and off-site locations to determine the extent of environmental discharges. Monitoring locations will be chosen commensurate with remediation activities.

In recognition that both the amount of radioactivity and the general environmental hazards will be reduced as decommissioning progresses, the Environmental Safety Program can be modified to be commensurate with the activities being performed. Following a similar procedure as described in Section 3.3, the contractor's ES&H manager may modify or change elements of the Environmental Safety Program with the approval of the remediation contractor's Project Manager and the Mallinckrodt RSO and Project Manager.

3.4.1 Effluent Air Monitoring

Operating exhaust stacks that ventilate C-T remediation work areas will be representatively sampled on a continuous basis at a point prior to discharge. The samples will be analyzed weekly. To be commensurate with on-going remediation activities, this sampling frequency may be modified in accordance with section 3.4. The measurement from each stack will be used to determine the activity concentration and total activity discharged from the stack. Radioactive releases in the airborne effluents will be summarized monthly, including pertinent information for each stack and for the total site. Any monthly average of effluent samples exceeding 10% of 10 CFR 20, Appendix B, Table II levels, will be evaluated by ES&H and approved by the ALARA Coordinator for necessary supplemental action to assure that releases of radioactive materials are ALARA.

3.4.2 Environmental Air Monitoring

Environmental sampling stations will be provided during remediation to verify there are no adverse impacts to on-site workers and the public. Each environmental sampling station will be equipped with a fallout collector, air sampler and thermoluminescent dosimeter (TLD). The fallout collector collects airborne materials such as precipitation, dust, and insects that are indigenous to the sampling area. The air sampler draws air through a filter via an opening in the sampler holder. The TLD will contain thermoluminescent chips capable of measuring gamma radiation dose.

Collection and analysis of the continuous air samples will be performed on a weekly basis. The samples will be analyzed for gross alpha and gross beta activity. The

analytical instruments will be calibrated using standards traceable to the National Institute of Science & Technology (NIST).

3.4.3 Liquid Effluent Monitoring

It will be the policy of Mallinckrodt during the C-T Project to minimize the production of contaminated aqueous liquids. There are three possible sources of contaminated aqueous liquids: sink and shower water, decontamination fluids, and water used to wet down asbestos containing materials. All site liquids are expected to be suitable for disposal under the conditions of the Metropolitan St. Louis Sewer District.

3.4.4 Direct Radiation Monitoring

The Radiation Protection Program is designed to assure that direct radiation in unrestricted areas does not exceed limits in 10 CFR 20.1301. The objective of direct radiation monitoring is to verify the effectiveness of the Radiation Protection Program in meeting the limits.

The monitoring of penetrating radiation will be performed using standard environmental thermoluminescent dosimeters that are placed at various locations around the perimeter of the restricted remediation area. These dosimeters will be collected by Health and Safety personnel and analyzed quarterly by a qualified contract vendor to measure the integrated gamma dose for each location.

3.4.5 Action Levels

Action levels will be established based on 10 CFR 20, Appendix B, Tables 2 and 3. If an action level is exceeded, the contractor ES&H manager will be notified and corrective action will be implemented as appropriate.

3.5 SECURITY DURING DECOMMISSIONING

Extensive security is provided at the Mallinckrodt St. Louis Plant. The entire perimeter is fenced, and all entrances are controlled by card-readers or the site security force. Normal Mallinckrodt security will be maintained at the St. Louis Plant during decommissioning activities. In addition, temporary fences will be employed around the areas undergoing remediation. These fences will be relocated as remediation progresses and will remain in place until the activities are completed and the area is surveyed, and made safe for general access. The fences will control personnel access and vehicle access to the portion of the site under remediation. All working personnel will enter and exit the remediation site through a controlled access gate and proceed to a clean change area to check in and out, and don or shed protective equipment.

3.6 RADIOACTIVE WASTE MANAGEMENT

Radioactive waste from the C-T project decommissioning activities will be managed in accordance with the C-T Project Waste Management Plan. The plan ensures that radioactive waste from the C-T project will be handled, stored and disposed of in accordance with applicable regulatory requirements. A summary of the plan is described below.

3.6.1 Management of Radioactive Waste

Three general categories of radioactive waste will exist during decommissioning: equipment, building material, and surface/subsurface.

- ***Equipment***

  Equipment, including piping, wiring, and associated components, will be segregated based on contamination level. Equipment may be released for unrestricted use or disposed of as described in 3.6.4.

- ***Building Material***

  Building material, including roofing material, block, brick, structural steel and concrete, will be segregated based on contamination level. Building material will be released for unrestricted use or disposed of as described in 3.6.4.

- ***Surface/Subsurface***

  Work involving pavement and streets, soil, building slabs, and underground structures and utilities, will not be performed during Phase I decommissioning. The Phase II Plan will discuss surface/subsurface waste management.

3.6.2 Regulatory Requirements

Processing and disposal of radioactive waste will be performed in accordance with the relevant requirements of 10 CFR 20, 10 CFR 40, 10 CFR 71, and the applicable disposal site waste acceptance criteria.

3.6.3 Projected Quantities of Radioactive Waste

Table 3-2 provides a summary level breakdown of the estimated volume of radioactive waste that is expected to be generated during decommissioning of the buildings and equipment of the C-T process and support areas.

Table 3-2 Estimated Phase I Radioactive Waste Volumes (Buildings and Equipment)	
Type	Volume (cubic feet)
Building Structure	36,000
Equipment	41,000
Total	**77,000**

3.6.4 Waste Disposition

Equipment and other materials known to be contaminated will be disposed of by transferring to a licensed facility, or will be decontaminated, surveyed and released under Annex C of the existing license. Other materials will be analyzed for radioactivity prior to final disposition. These materials will be managed and disposed of at either a licensed facility, a special waste facility, or released for unrestricted use, as appropriate. Table 2-3 in Section 2.0 of this plan guides the disposal of the bulk material for unrestricted release.

3.6.5 Temporary On-Site Storage of Radioactive Waste Prior to Shipping

Contaminated material maybe staged on-site temporarily to (a) stage material for sampling and analysis; (b) accumulate sufficient quantities for economical shipment; and (c) coordinate shipments between the carrier and the disposal site.

Building material and equipment may be stored in designated staging areas. Materials stored outside will be covered with tarpaulins or an equivalent heavy-duty material at all times, except when work must be performed on the material or equipment.

3.6.6 Mixed Waste

Based upon an evaluation of site characterization results, Mallinckrodt anticipates that a limited amount of materials containing both radiological and chemical constituents at levels sufficient to require management as hazardous waste may be identified during decommissioning. Management and disposal of mixed waste generated during the C-T Project will be performed in accordance with all applicable federal and state regulations. Mallinckrodt will manage this anticipated, small quantity of mixed waste in an environmentally responsible manner, and will explore viable treatment and disposal alternatives for the mixed waste during the decommissioning period.

3.6.7 Records

Mallinckrodt will maintain records of waste material released from the C-T decommissioning area or controlled areas.

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4. PLANNED FINAL STATUS SURVEY

This section will provide a summary of the planned Final Status Survey. This survey will demonstrate that the three types of media, equipment, building surfaces, and building rubble, meet the criteria for unrestricted release.

Table 1-6, "Equipment Status," lists the buildings which contain equipment to be included in the final status survey. Section 1.6.4, "Current Radiological Status," further describes the current radiological condition of the equipment. Equipment which will be released for unrestricted use will be surveyed in accordance with Section 4.4.6.1, "Equipment Surveys." Equipment which is located in buildings slated for demolition and equipment which is suspect or known to contain internal contamination will be disposed of in accordance with Section 3.6.4, "Waste Disposition."

Table 1-5, "Disposition Of C-T Process And Support Area Buildings," lists the buildings which were used for the C-T Process and support areas. Section 1.6.4, "Current Radiological Status," further describes the current radiological status of the buildings and Appendix A, "Surface Survey Area Results," classifies these buildings according to contamination levels consistent with Section 4.4.1.2, "Building Surfaces." Final Status Survey results for all buildings other than those addressed under FUSRAP will be included in the Final Status Survey Report.

The third medium which will be included in the Final Status Survey is building rubble resulting from the crushing of the building material. Buildings slated for demolition are listed in Table 1-5, "Disposition Of C-T Process And Support Area Buildings."

To the extent practical, guidance from the Draft Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) has been incorporated in this Final Status Survey. First, release criteria (denoted "derived concentration guideline level" or "DCGL" in MARSSIM) have been established and are described in Section 4.1, "Release Criteria." Next, proper instrumentation has been chosen and described in Section 4.2, "Instrumentation." The background determination is described in Section 4.3, "Background." This section describes prior activities to determine background values and the remaining activities to be performed. Finally, Section 4.4, "Survey Methodology" describes the approach which will be taken to perform the Final Status Survey. MARSSIM guidance was adapted as appropriate to provide instruction in survey design for equipment and building rubble.

The final section describes the Quality Assurance Program as it applies to the Final Status Survey.

4.1 RELEASE CRITERIA

Residual radioactive material release criteria (DCGL) have been determined for each of the three types of media. Residual radioactive material is material which is regulated as a radioactive material and does not include naturally occurring radioactive material.

The equipment release criteria are based on the limits found in NRC FC83-23, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material." To facilitate surveying, beta DCGL were calculated from the FC83-23 guidance. This derivation is further described in Appendix D, "Surface Beta Radioactivity Limit Derivation."

The building surface DCGLs were derived using guidance from NUREG/CR-5512 assuming a dose limit of 25 mrem/y to a worker assumed to be exposed as defined for the Worker-Building Occupation Scenario. This derivation is described in Appendix C, "Derivation of Nuclide Limits for Building Surfaces." As with equipment, beta DCGLs were calculated from the derived limits.

The DCGLs for the building rubble were derived utilizing the DOE code RESRAD assuming a dose limit of 25 mrem/y to a worker or potential future resident. This derivation is described in Appendix E, "Derivation of Mass Concentration Limits."

4.2 INSTRUMENTATION

The Final Status Survey will consist of scanning, direct stationary surveying, and analysis of material samples. Therefore, scanning instruments, direct measuring instruments and laboratory instruments will be used in conducting the final radiation surveys and sample analysis. All instruments will be appropriate for the type of survey and the concentration of radioactivity to be measured. QA/QC procedures to be used with instrumentation described in Section 5.7, "Equipment Maintenance and Calibration."

Typical instrumentation is listed in Tables 4-1 and 4-2. Other instrumentation meeting requisite detection capabilities may be used. Table 4-1 lists the instrumentation to be used for survey activities, along with typical parameters and detection sensitivities for the instrumentation and survey technique. The combination of instrumentation and technique will be chosen to provide a detection sensitivity below the release criteria shown in Section 2.2. Sensitivities for scanning techniques are based on movement of the detector over the surface at one detector-width per second and use of audible indicators to sense changes in instrument count rate. Experience has shown that the Ludlum E600 gas proportional probe can achieve detection sensitivities as low as 100 dpm/100 cm^2 with count times of 5 minutes for direct surveys (stationary) for typical brick or concrete and lower for lower background materials.

Table 4-1 Typical Direct Measurement Instruments for Performing Final Radiological Surveys			
Instrument/ Type	Radiation Type	Application	Typical Detection Sensitivity
Gas Proportional Probe	Beta	Scanning	680-4560 dpm/100 cm^2 depending on background
Gas Proportional Probe	Beta	Surface Surveying (Stationary)	100-260 dpm/100 cm^2
End Window or Pancake Probe	Beta	Surface Surveying (Stationary)	500 dpm/100 cm^2
Bicron AB-100 Scintillation Probe	Beta	Surface Surveying (Stationary)	150-370 dpm/100 cm^2

Table 4-2 Typical Laboratory Instruments for Performing Final Radiological Surveys			
Instrument Method	Radiation Type	Application	Typical Sensitivity
HPGe Spectrometer	Gamma	Sample Analysis	<0.1 pCi/g
Alpha Spectrometer	Alpha	Sample Analysis	0.1 pCi/g
Gas Proportional Counter	Beta, Alpha	Gross Radioactivity	1-3 dpm beta or alpha
NaI Scintillator	Gamma	Sample Analysis	Variable

Direct surface measurements for gross beta activity will be performed using scintillation probes or gas proportional probes. Special consideration will be given to the use of large area probes on the order of 100 cm^2. Scanning shall be performed using scintillation probes or gas proportional probes such as the Bicron Surveyor M with A100 or B100 probes and Eberline gas proportional floor monitors for beta activity. Removable activity measurements for equipment shall be performed using a low background gas flow proportional counter such as the Tennelec LB 5100.

4.3 BACKGROUND

Because the nuclides of interest are naturally occurring and are measurable in most materials, background values will be needed for the three media to be surveyed under the Final Status Survey, equipment, building surfaces, and building rubble. As part of the characterization, background surveys were made for equipment and building surfaces. Background measurements were taken on site or in the immediate vicinity of the site in areas that were not affected by site operations. To determine brick backgrounds, bricks were removed from areas not affected by site operations, cut in half to expose fresh clay material, and surveyed. Table 4-3 shows the background levels which have been

determined. Background values which must still be determined are denoted by TBD and values which are assumed to be 0 are also noted.

Table 4-3
Background Values of Materials of Construction

Material	Number of Samples	Average Background (β/min/100 cm^2)	Standard Deviation (β/min/100 cm^2)
Asphalt	42	254	166
Asbestos Tile	N/A	Assumed 0	N/A
Brick	90	638	140
Concrete	70	180	79
Concrete Block	51	299	62
Concrete Block Bldg 101	21	560	62
Ceramic Tile	33	591	44
Counter Top - Bldg 250 Lab	10	403	77
Ceiling Tile	30	510	49
Expansion Joint	N/A	Assumed 0	N/A
Foam	N/A	Assumed 0	N/A
Fiberglass	N/A	Assumed 0	N/A
Glass	N/A	Assumed 0	N/A
Gypsum Board	N/A	Assumed 0	N/A
Liner	N/A	Assumed 0	N/A
Metal	N/A	Assumed 0	N/A
Other Non-Metal	N/A	Assumed 0	N/A
Plastic	N/A	Assumed 0	N/A
Roofing	N/A	TBD	N/A
Red Clay Tile	1	638	N/A
Tar/Roofing	N/A	Assumed 0	N/A
Vinyl Tile	N/A	Assumed 0	N/A
Wood	N/A	Assumed 0	N/A

In general, the sensitivity of a survey is greatest when the number of measurements is split between the background and the survey unit. In the event that the number of background data points acquired during characterization is insufficient, additional background data points will be taken as needed and as practicable.

During characterization, background values for building rubble were not established and will therefore be determined prior to the initiation of the Final Status Survey. Building rubble background will be obtained by removing construction materials from a location not affected by site operations, crushed and analyzed to determine the background. Alternatively, background values for cinder fill at the site may be used to represent background for crushed building material.

4.4 SURVEY METHODOLOGY

The Final Status Survey will be designed to detect elevated areas and determine the distribution of radionuclides in accordance with the Draft Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) to the extent practical. First, areas will be classified according to potential for residual contamination or known contamination level. Next, these areas will be grouped into survey units as described in Section 4.4.2. Then, the required number of measurements and the survey locations will be determined. Finally, the surveys will be performed and the data will be analyzed.

4.4.1 Area Classification

The intent of the plan is to focus most of the survey effort on areas where the likelihood of exceeding the DCGL is greatest. Therefore, for the purposes of establishing the sample density and the sampling pattern, equipment, building surfaces, and building rubble will be classified according to the potential for residual contamination.

4.4.1.1 Equipment

Equipment is defined as material that is not a structural component of the building or integral to the building. For example, duct work and piping will be treated as equipment.

Equipment will first be classified as either Non-Impacted or Impacted. Non-Impacted equipment is equipment which has no reasonable potential for containing residual radioactive material and will not need any level of survey coverage. Equipment will be considered Impacted if it has been used or stored in areas where it may have been in contact with uncontained radioactive material.

Impacted equipment will be further subdivided into one of two categories:

- **Category 1 Equipment**: Equipment which has become contaminated, or is highly suspect, requiring comprehensive or full survey.

- **Category 2 Equipment**: Equipment which is possibly contaminated but for which there is no direct evidence of contamination. At least a confirmatory/verification-type survey is required for unrestricted release of equipment in this category.

Category 1 Equipment located in buildings slated for demolition, other equipment classified initially as Category 1, and equipment ultimately reclassified as Category 1 will be dispositioned of in accordance with Section 3.6.4, "Radioactive Waste Management" and will not be included in the Final Status Survey.

For the purposes of establishing the sample density and sampling pattern, the equipment has been classified according to the potential for residual contamination as delineated in

Table 4-4. If the equipment is classified impacted, the level of impact is noted. Appendix A, "Surface Survey Area Results" identifies the equipment within each building and its category.

Table 4-4
Equipment Area Categories

Equipment Location	Category
Building 90/91	Non-Impacted
Building 101 Roof	Impacted, Category 1 and Category 2
Building 101 Incinerator	Impacted, Category 1
Building 200 Roof	Impacted, Category 2
Building 201 Roof	Impacted, Category 2
Building 204 Roof	Impacted, Category 2
Building 235 Roof	Impacted, Category 2
Building 236 Roof	Impacted, Category 2
Building 245	Impacted, Category 2
Building 245 Roof	Impacted, Category 2
Building 250	Non-Impacted and Impacted, Category 1 and Category 2
Building 250 Lab 202	Impacted, Category 2

4.4.1.2 Building Surfaces

Building surfaces will first be classified as either Non-Impacted or Impacted. Non-Impacted areas are areas that have had no reasonable potential for containing residual radioactive material and do not need any level of survey coverage. These areas have not been radiologically impacted from site operations. Impacted areas are areas which have potential for containing residual radioactive material.

Impacted areas will be further subdivided into one of three classifications:

- **Class 1 Areas**: Areas that have, or had, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys). Areas containing contamination in excess of the DCGL prior to remediation shall be classified as Class 1 areas. Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or spills are known to have occurred, 3) waste storage sites, and 4) areas with contaminants in discrete solid pieces of material having high specific activity.

- **Class 2 Areas**: These areas have, or had, a potential for radioactive contamination or known contamination, but are not expected to exceed the DCGL. To justify changing an area's classification from Class 1 to Class 2, the existing data shall provide a high degree of confidence that no individual measurement will exceed the DCGL. Examples of areas that might be classified as Class 2 f the final status survey

include: 1) locations where radioactive materials were present in an unsealed form (*e.g.*, process facilities), 2) potentially contaminated transport routes, 3) upper walls and ceilings of buildings or rooms subjected to airborne radioactivity, 4) areas where low concentrations of radioactive materials were handled, and 5) areas on the perimeter of former contamination control areas.

- **Class 3 Areas**: Any Impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a very small fraction of the DCGL, based on site operating history and previous radiological surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a Non-Impacted classification.

For the purposes of establishing the sample density and sampling pattern, the buildings have been classified according to the potential for residual contamination as delineated in Table 4-5.

Table 4-5
Building Surface Area Classes

Location	Class
Building 62	Non-Impacted
Building 91 Crane Service Pad	Impacted, Class 3
Building 90/91 (Balance of Bldg)	Non-Impacted
Building 101 Roof	Impacted, Class 1
Building 101 External Walls	Non-Impacted and Impacted, Class 2
Building 101 Incinerator Pad	Impacted, Class 1
Building 200 Roof	Impacted, Class 3
Building 201 Roof	Impacted, Class 3
Building 204 Roof	Impacted, Class 3
Building 222 Roof	Impacted, Class 3
Building 223 Roof	Impacted, Class 3
Building 235	Impacted, Class 1, Class 2 and Class 3
Building 235 Roof	Impacted, Class 2
Building 236	Impacted, Class 1, Class 2 and Class 3
Building 236 Roof	Impacted, Class 2
Building 240	Impacted, Class 3
Building 240 Roof	Impacted, Class 3
Building 245	Impacted, Class 1 and Class 3
Building 245 Roof	Impacted, Class 3
Building 250	Impacted, Class 1 and Class 2
Building 250 Roof	Impacted, Class 3

4.4.1.3 Building Rubble

Prior to demolition, building areas will be classified as either Non-Impacted or Impacted. Non-Impacted building areas are areas that have no reasonable potential for residual radioactive material. These areas have not been radiologically impacted from site operations. Impacted building areas are areas which have potential for containing residual radioactive material.

Impacted building areas will be further subdivided into one of three levels:

- **Level 1 Rubble**: Building areas that have, or had, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys) and are expected to exceed the mass DCGL after crushing.

- **Level 2 Rubble**: Building areas that have, or had, a potential for radioactive contamination or known contamination, but are not expected to exceed the mass DCGL after crushing.

- **Level 3 Rubble**: Building areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a very small fraction of the DCGL, based on site operating history and previous radiological surveys.

For the purposes of establishing the sample density and sampling pattern, the building rubble from buildings slated for demolition has been classified according to the potential for residual contamination as delineated in Table 4-6. If the building rubble is classified Impacted, the level of impact is noted. There are no buildings which will be demolished producing rubble to be classified as Level 1 rubble. During the remediation process, selected removal of elevated surface activity will be performed prior to demolition to assure that the mass concentrations of the residual CT nuclides will be below the mass release criteria.

Table 4-6
Building Rubble Classification

Location	Classification
Building 213	Impacted, Level 2 and Level 3
Building 214	Impacted, Level 2 and Level 3
Building 238	Impacted, Level 2
Building 246	Impacted, Level 2 and Level 3
Building 247	Impacted, Level 2 and Level 3
Building 248	Impacted, Level 2 and Level 3

4.4.2 Survey Unit Definition

To facilitate survey design and assure that the number of survey data points for the media are relatively uniformly distributed within areas of similar contamination potential, and to focus most of the survey effort on areas where the likelihood of exceeding the DCGL is greatest, the media will be divided into survey units which have a common history or other similar characteristics, or are naturally distinguishable from other portions of the site.

4.4.2.1 Equipment

An equipment survey unit will only include pieces of equipment from the same equipment category. The equipment in the survey unit will be from contiguous locations and/or exhibit similar characteristics.

4.4.2.2 Building Surfaces

A building surface survey unit will not include areas that have different classifications. The survey unit characteristics will be generally consistent with exposure pathway modeling that was used to convert dose into the DCGL. For indoor areas, where rooms are classified as Class 1 areas, each room may be designated as a survey unit. Indoor areas may also be subdivided into several survey units of different classification, such as separating floors and lower walls from upper walls and ceilings (and other upper horizontal surfaces) or subdividing a large warehouse based on floor area.

Survey units will be limited in size based on classification, exposure pathway modeling assumptions, and site-specific conditions. The suggested maximum areas for survey units are as follows:

Table 4-7
Maximum Areas for Structure Survey Units

Area Classification	Typical Maximum
Class 1	100 m^2
Class 2	100 to 1000 m^2
Class 3	no limit

4.4.2.3 Building Rubble

A building rubble survey unit will not include building rubble of differing classification levels or differing material. The survey unit may consist of material from different buildings but with the same classification level.

4.4.3 Hypothesis Formulation

The decision that the DCGLs are met is based on a hypothesis test. The null hypothesis will be that the survey unit exceeds the release criterion. This will require that significant evidence exists such that the residual radioactivity in the survey unit is less than the release criterion to reject the null hypothesis (and pass the survey unit). If the evidence is not significant at level α, the null hypothesis of a non-complying survey unit will be accepted (and the survey unit will fail). The target for α will be 0.05 but may be higher as determined by the practical limitations on the number of measurements as described in Section 4.4.4.

4.4.4 Determination of Number of Stationary Measurements

The number of measurements required for each survey unit will be determined by three considerations. The first is the sensitivity required by statistical analysis to support the hypothesis test, as discussed in Section 4.4.1 below. The second, applicable only to Category 1 equipment and Class 1 building surfaces, is the sampling density required for elevated area detection, discussed in Section 4.4.4.2 below. Finally, because, in some instances, the DCGL are such a small increment of background concentrations, which makes detection of the increment very difficult, a sampling density cap, based on practicality, is imposed, as discussed in Section 4.4.4.3 below. The number of stationary measurements for a given survey unit will be the larger of the two numbers determined in Sections 4.4.4.1 and 4.4.4.2, subject to limitations in Section 4.4.4.3, if applicable.

4.4.4.1 Number of Samples Required for Central Tendency Analysis

Two different approaches are used for central tendency analysis. The first applies to the situation in which the radionuclides of interest are present in the background. The second applies to the situation in which they are not present in the background (i.e., where the background concentration is zero or is assumed to be zero). These situations are discussed separately below.

Nuclides of Interest Present in Background

The number of data points required when the contaminant is present in the background will be determined in accordance with section 5.5.2.2 of MARSSIM:

1. Estimate the relative shift, Δ/σ, where σ is the expected standard deviation of the survey unit measurements and Δ is the width of the "gray region." The gray region, as used in MARSSIM, can be considered as a region of central tendency nuclide concentration that corresponds to dose. Within the gray region, the probabilities of either a Type I decision error (deciding the unit meets DCGL when it does not) or a Type II decision error (deciding the unit exceeds DCGL when it does not) exceed desired limits. The upper bound of the gray region necessarily corresponds to the

release criterion concentration. The lower bound, and thus the width, is, to some extent, arbitrary. However, setting the boundary too low drives down the central tendency concentration below which one is highly confident in a decision that a unit exceeds DCGL. Setting the value too high drives up the number of measurements required to achieve the desired decision error probabilities. For purposes of estimating the number of data points, the value of σ will be estimated from characterization survey data or from background measurements whichever is representative of the survey unit (i.e., if licensed radioactive material is not expected to be present in the survey unit). The determination of Δ/σ may be iterative. On the first iteration, a value of 1 will be selected as a reasonable compromise.

2. Determine P_r, the probability that a measurement performed at a random location in the survey unit will result in a larger value than a measurement performed at a random location in the reference area. P_r is determined from Table 5.1 of MARSSIM using the estimated relative shift, Δ/σ, determined in step 1.

3. Determine the decision error percentiles corresponding to the desired error probabilities α and β using Table 5.2 of MARSSIM. On the first iteration, a value of 0.05 will be used for α and β. Larger values may be used on subsequent iterations.

4. Calculate the total number of measurement points (survey unit plus reference area) for the WRS Test using Equation 4-1 (Equation 5-1 from MARSSIM) with input parameter values determined in steps 1 through 3.

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{3\left(P_r - 0.5\right)^2} \qquad \text{[Equation 4-1]}$$

Where $Z_{1-\alpha}$ and $Z_{1-\beta}$ are the percentiles represented by the selected decision error levels; P_r is the probability that a measurement performed at a random location in the survey unit will result in a larger value than a measurement performed at a random location; and N is the number of data points to be obtained from each reference area/survey unit.

Equation 4-1 assumes equal numbers of background and survey measurements. As noted in the discussion on background in section 4.3, the number of background measurements may be constrained. There will be some advantage in sensitivity achievable by collecting a larger number of measurements from the survey unit, although there will be diminishing returns as the number of measurements from the survey unit increases. To calculate the number of samples in this case, Equation 4-2 (based on NUREG-1507, Equation 5-1) is first used:

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{12c(1-c)\left(P_r - 0.5\right)^2} \qquad \text{[Equation 4-2]}$$

where $Z_{1-\alpha}$ and $Z_{1-\beta}$ are the percentiles represented by the selected decision error levels; P_r is the probability that a measurement performed at a random location in the survey unit will result in a larger value than a measurement performed at a random location; c is the fraction of samples from the reference area; and N is the number of data points to be obtained from each reference area/survey unit.

In any survey there will be some missing or unusable data. The rate of missing or unusable measurements, R, expected to occur in survey units or reference areas will be accounted for during survey planning. To assure sufficient data points to attain the desired power level with the statistical tests and allow for possible lost or unusable data, the number of data points will be increased by 20% (R=0.2), and rounded up, over the values calculated in the final step.

The required number of measurements determined in the first iteration may exceed reasonable bounds. The process can be repeated using more suitable values of Δ, α, and β as appropriate.

Nuclides of Interest Not Present in Background

The general approach to determining the number of required data points when the contaminant is not present in the background parallels, to some extent, the approach used for the situation when the contaminant is present in the background, described above. However, because background concentrations need not be considered, the formulation for the number of measurements differs somewhat.

1. First, the relative shift Δ/σ is determined as described above.

2. Next, Sign p is determined. Sign p is the probability that a random measurement from the survey unit will be less than shift, Δ. Given the relative shift, Sign p is determined using the previously determined value of Δ/σ and Table 5.4 of MARSSIM.

3. Determine the decision error percentiles corresponding to the desired error probabilities α and β using Table 5.2 of MARSSIM. On the first iteration, a value of 0.05 will be used for α and β. Larger values may be used on subsequent iterations.

4. Finally, the number of data points is calculated using Equation 4-3 (equation 5-2 from MARSSIM)

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{4\left(Sign\ p - 0.5\right)^2} \qquad \text{[Equation 4-3]}$$

The number of anticipated data points will be increased by 20% to assure sufficient power of the tests and to allow for possible data losses. Because the presence of nuclides of interest in background are not a consideration for this case, detection difficulties should not arise. However, if they do, the required number of measurements determined in the first iteration can be adjusted by repeating the process using more suitable values of Δ, α, and β as appropriate.

4.4.4.2 Determining Data Points for Small Areas of Elevated Activity

The measurements described in section 4.4.4.1 are designed to test whether the DCGL are met based on measures of central tendency. Scanning measurements are the primary method for detecting small areas of elevated contamination. Scanning will be performed for survey units containing equipment categories 1 (as needed for disposition) and 2, and building surface classes 1 and 2. For survey units containing equipment category 2 and building surface category 2, the scanning is confirmatory only. For survey units containing building equipment category 1 and building surface class 1, individual measurements that may exceed DCGL are anticipated. The interest in these units is to assure that the concentrations and areal extent of elevated contamination are sufficiently constrained to assure that unrestricted release dose criteria are met. To the extent that scanning measurements are not sufficiently sensitive to assure this, they are supplemented by stationary measurements. The number of measurements required for this purpose is determined following the procedure in section 5.5.2.4 of MARSSIM.

4.4.4.3 Caps on Numbers of Stationary Measurements

The unrestricted release dose limit of 25 mrem/y established as one of the criteria for decommissioning, is so low that it is in the range of doses that result from naturally occurring radionuclides in the environment. Consequently, the DCGL, surface or mass concentration limits, are frequently a small increment of the concentration of naturally occurring radioactive material present in the material of interest. This situation complicates the process of assuring with a high level of confidence that licensed radioactive material present does not exceed the small permissible increment. Demonstration of that condition with a high level of confidence could require an unreasonably large number of measurements. Consequently, for these situations, a sampling density cap is applied to each survey unit, which effectively assures that a reasonable upper bound is placed on the sampling effort. The sampling density cap to be used for equipment and building surfaces is $1/m^2$ and the cap for building rubble is 30 samples per survey unit. The effect of the cap is to limit the level of confidence associated with a decision that the permissible concentration increment is not exceeded. However, the sampling caps established are high enough to provide reasonable assurance that the total concentration of radioactive material present (background plus licensed radioactive material) will not substantially exceed background concentrations.

4.4.5 Survey Locations

The number of data points required and the area of the survey unit will determine the location of the data points. Differing approaches will be taken for the equipment, building surfaces, and building rubble.

4.4.5.1 Equipment Survey Locations

Equipment to be surveyed in the Final Status Survey is classified as Category 2 and requiring a confirmatory/verification survey. Equipment classified as Category 1 equipment will be dispositioned in accordance with Section 3.6, "Radioactive Waste Management."

The number of data points will be distributed approximately uniform among the pieces of equipment of the survey units. Equipment with larger surface areas may receive more than one sample or measurement.

A equal number of direct (stationary) and removable measurements will be taken.

4.4.5.2 Building Surfaces Survey Locations

A scale drawing of the survey unit will be prepared, along with the overlying planar reference coordinate system or grid system. Any location within the survey area is thus identifiable by a unique set of coordinates. The maximum length, X, and width, Y, dimensions of the survey unit are then determined.

All structure surfaces for a specific survey unit will be included on a single reference grid system for purposes of identifying survey locations. Measurements and samples in Class 3 survey units and reference areas will be taken at random locations. These locations will be determined by generating sets of random numbers (2 values, representing the X axis and Y axis distances). Each set of random numbers will be used to provide coordinates, relative to the origin of the survey unit reference grid pattern. Coordinates identified in this manner, which do not fall within the survey unit area or which cannot be surveyed, due to site conditions, will be replaced with other survey points determined in the same manner.

Class 2 areas will be surveyed on a random-start systematic pattern. The number of calculated survey locations, based on the statistical tests, will be used to determine the spacing, L, of a systematic pattern by the following equations, Equation 4-4 for a triangular grid or Equation 4-5 for a square grid. In the equations, L is the spacing, A is the area of the survey unit and n is the number of calculated survey locations.

$$L = \sqrt{\frac{A}{0.866n}} \qquad \text{[Equation 4-4]}$$

$$L = \sqrt{\frac{A}{n}}$$ [Equation 4-5]

The choice of grid pattern will be made by considering elevated area contamination potential and general shape of the survey unit. The grid which is most practical to survey and evaluate will be chosen.

After L is determined, a random coordinate location will be identified, as described previously, for a survey pattern starting location. Beginning at the random starting coordinate, a row of points will be identified, parallel to the X axis, at intervals of L. For a triangular grid, a second row of points will then be developed, parallel to the first row, at a distance of 0.866•L from the first row. Survey points along that second row will be midway (on the X-axis) between the points on the first row. This process will be repeated to identify a pattern of survey locations throughout the survey unit. If identified points fall outside the survey unit or at locations which cannot be surveyed, additional points will be determined using the random process described above, until the desired total number of points is identified.

For Class 1 areas, a systematic pattern will be installed on the survey unit. The starting point for this pattern will be selected at random, as described above for Class 2 areas. The same process as described above for Class 2 areas applies to Class 1, only the estimated number of samples may be different.

In addition to the survey locations identified for statistical evaluations and elevated measurement comparisons, it is likely that data will also be obtained from judgment locations, selected due to unusual appearance, location relative to contamination areas, high potential for residual activity, general supplemental information, *etc.* These data points selected based on professional judgment will not be included with the data points from the random-start triangular grid for statistical evaluations; instead they will be compared to the investigation levels described in section 4.4.7.2. Measurement locations selected based on professional judgment violate the assumption of unbiased measurements used to develop the statistical tests.

4.4.5.3 Building Rubble Sample Collection

The required number of data points for each survey unit will be collected following crushing of the building material. Each survey unit will be crushed separately. Samples of the crushed material will be removed manually or automatically at random or preset intervals. These samples will be tagged and stored for analysis. The required number of samples will be chosen randomly from this set of samples. Materials from a survey unit that cannot be crushed practically, such as concrete attached to rebar, will be considered to have the same radiological status as the crushed material.

4.4.6 Surveys

After the number of required samples has been established and the location of the samples is determined, the survey strategy will be developed using the following guidelines. Both scans and stationary measurements will be performed.

4.4.6.1 Equipment Surveys

The equipment to be surveyed for the Final Status Survey is classified as Category 2 and will receive a 10% scan. If these scans indicate possible contamination greater than the DCGL, the equipment will be reclassified and disposed of in accordance with section 3.6. At each survey location , a direct and removable measurement will be taken.

4.4.6.2 Building Surface Surveys

Class 1 Areas

Surface scans will be performed over 100% of structure surfaces for beta radiation which might be emitted from the potential radionuclide contaminants. Locations of direct radiation, above the DCGL, will be identified and evaluated. Results of initial and follow-up direct measurements and sampling at these locations will be recorded and documented in the Final Status Survey Report. Measurements of total contamination will be performed at locations identified by scans and at previously determined locations (Section 4.4.5.2).

Class 2 Areas

Surface scans will be performed for β radiation over 10 to 100% of structure surfaces. Generally, floors will receive 100% surface scans, walls up to 2 m will receive 25% surface scan and upper wall surfaces above 2 m and ceilings will receive 10% surface scan.

Class 3 Areas

Scans of Class 3 area surfaces will be performed for beta radiations at designated locations. These locations will be determined through the use of historical knowledge and contractor experience. Direct (stationary) measurements of total beta contamination will be performed at locations identified by the scans and at the randomly selected locations, chosen in accordance with Section 4.4.5.2.

4.4.6.3 Building Rubble Survey

Gamma spectroscopy will be performed on the individual samples and alpha spectroscopy will be performed on a composite sample to determine the nuclide

distribution. Nuclides which do not appear under gamma spectroscopy will be determined using the alpha spectroscopy.

4.4.7 Data Analysis

The evaluation of survey results is performed in four stages. The first stage will consist of a preliminary review of the data. The second stage of evaluation will consist of an evaluation of elevated measurements against investigation levels. The third stage of data evaluation will consist of statistical analysis to determine if DCGL have been satisfied. The last stage of data evaluation will consist of concluding whether the results of the survey meet the design objectives. Based on results from the first three stages, resurvey, reclassification, remedial action, or some combination of these measures may be required. The survey will not be complete until the conclusion that survey objectives have been met can be supported. Each of these stages is discussed in greater detail in the following subsections.

4.4.7.1 Conduct a Preliminary Data Review

The first stage of data evaluation will consist of a preliminary review of the data to check quality and reasonableness. Data conversion to DCGL units may also occur in this stage. For equipment and building surfaces, the units are βpm/100cm^2. For building rubble, the unit for each measurement point will be an index calculated as the sum of the ratios of the measured mass concentration of each nuclide of interest to its mass concentration release limit in Table 2-3. The calculation of this index and the expression of DCGL in terms of this index are described in Section 2.2.

4.4.7.2 Conduct Elevated Measurement Test

The second stage of evaluation will consist of an evaluation of elevated measurements against investigation levels. Investigation levels are radioactivity measurement levels or indices calculated from radioactivity measurements that are used to indicate when additional investigations may be necessary. For example, a measurement that exceeds the investigation level may indicate that the survey unit has been improperly classified. The investigation level is intended to flag potential failures in the remediation process. Exceeding an investigation level will not be considered the primary means to identify whether or not a unit meets the release criterion.

Individual beta measurements from equipment and building surfaces and indices calculated from nuclide concentration measurements in rubble samples will be compared to investigation levels for evidence of small areas of elevated activity. Scan results for those units subject to scanning will also be compared to investigation levels. The levels established for investigation will depend upon the survey unit classification. Lower investigation thresholds will be set for those units having lower potential for elevated areas. Depending on the outcome of the elevated measurement test and other tests, resurvey, reclassification, partial or complete remediation, or some combination of these

measures may be required. (If only partial remediation is required, resurvey of some portion of the unit after supplementary remediation will also be required. To the extent practical and appropriate, original survey data from portions of the unit outside the supplementary remediation area will be used in conjunction with new survey data from the supplementary remediation area in new tests to determine whether the unit meets release criteria.) The results of all investigations will be documented in the final status survey report. Investigation levels are described in greater detail below.

In equipment category 3, building surface class 3 and building rubble level 3 units, no residual nuclides are expected. Therefore, investigation levels are set to flag measurements that are just above the range expected for background levels or just above detection limits for the measurement method, whichever is greater.

In equipment category 2, building surface class 2 and building rubble levels 1 and 2 survey units, measurements of net levels above the DCGL are not expected. Therefore, investigation levels are set to flag measurements exceeding the DCGL on a net basis. If the scanning MDA exceeds the DCGL on a net basis, any scanning result exceeding the MDA will also be investigated.

For equipment category 1 and building surface class 1 survey units, measurements above the DCGL are not unexpected, so a special derived concentration guideline for elevated measurements, $DCGL_{EMC}$, is used as the basis for investigation levels. The derived concentration guideline level for the EMC (elevated measurement comparison) is determined by using Equation 4-6

$$DCGL_{EMC} = (A_m)(DCGL_W) \qquad \text{[Equation 4-6]}$$

where A_m is the area factor, as determined following the procedure in section 5.5.2.4 of MARSSIM, and $DCGL_w$ is the DCGL as specified in Section 2.2.

The table below summarizes the investigation levels.

Table 4-8
Summary of Investigation Levels

Survey Unit Classification	Flag Direct Measurement or Sample Result When:	Flag Scanning Measurement Result When:
Equipment Category 1	Gross Beta>($DCGL_{EMC}$ + $\bar{x}_{ref}$[a] + $3s_{ref}$[a])	Gross Beta>maximum of ($DCGL_{EMC}$+ $\bar{x}_{ref}$[a] + $3s_{ref}$[a]) or MDA
Equipment Category 2	Gross Beta>($DCGL_W$+ $\bar{x}_{ref}$ + $3s_{ref}$)	Gross Beta>maximum of ($DCGL_W$+ $\bar{x}_{ref}$ + $3s_{ref}$) or MDA
Equipment Category 3	Gross Beta>maximum of ($\bar{x}_{ref}$ + $3s_{ref}$) or MDA	Gross Beta>maximum of ($\bar{x}_{ref}$ + $3s_{ref}$) or MDA
Building Surface Class 1	Gross Beta>($DCGL_{EMC}$ + $\bar{x}_{ref}$[a] + $3s_{ref}$)	Gross Beta>maximum of ($DCGL_{EMC}$+ $\bar{x}_{ref}$[a] + $3s_{ref}$) or MDA
Building Surface Class 2	Gross Beta>($DCGL_W$+ $\bar{x}_{ref}$ + $3s_{ref}$)	Gross Beta>maximum of ($DCGL_W$+ $\bar{x}_{ref}$ + $3s_{ref}$) or MDA
Building Surface Class 3	Gross Beta>maximum of ($\bar{x}_{ref}$ + $3s_{ref}$) or MDA	Gross Beta>maximum of ($\bar{x}_{ref}$ + $3s_{ref}$) or MDA
Building Rubble Level 1	Index>($DCGL_{EMC}$+ $\bar{x}_{ref}$ + $3s_{ref}$)	No Scanning
Building Rubble Level 2	Index>($DCGL_W$+ $\bar{x}_{ref}$ + $3s_{ref}$)	No Scanning
Building Rubble Level 3	Index> ($DCGL_W$+ $\bar{x}_{ref}$ + $3s_{ref}$)	No Scanning

[a] $\bar{x}_{ref}$ and $3s_{ref}$ are the mean and standard deviation of the reference or background measurements or indices calculated from measurements

4.4.7.3 Conduct Statistical Analysis

The third stage of data evaluation will consist of statistical analysis of results from direct measurements or samples from specific locations to determine if DCGL have been satisfied. This stage will also include reassessment of the power of the hypothesis test, based on survey data. Consideration of values Δ, α, and β more suitable than those used in the survey design may be appropriate.

Different forms of statistical analysis are required depending on whether the nuclides of interest are present in the background. These two forms are discussed separately below. If criteria are not met, resurvey or remedial action may be required.

Nuclides Present in Background

For situations in which the nuclides of interest are present in the background, the following logic will be used to judge compliance with the DCGL:

Table 4-9
Summary of Statistical Tests
(Radionuclide in background or non-radionuclide-specific measurement made)

Survey Result	Conclusion
All survey unit measurements are less than $DCGL_W$ on a net basis	Survey unit meets release criteria
Difference between any survey unit measurement and any reference area measurement greater than $DCGL_W$ (not to be used for survey units with less than 5 measurements)	Conduct WRS test and elevated measurement comparison

If statistical analysis is necessary for a survey unit, the two-sample Wilcoxon Rank Sum (WRS) test (also called the Mann-Whitney test) will be conducted. The WRS test assumes that the reference area and survey unit data distributions are similar except for a possible shift in the medians, and is applied as follows:

1. Obtain the adjusted reference area measurements, Z_i, by adding the DCGLw to each reference area measurement, X_i as shown in equation 4-7.

$$Z_i = X_i + DCGL_W \qquad \text{[Equation 4-7]}$$

2. The m adjusted reference sample measurements, Z_i, from the reference area and the n sample measurements, Y_i, from the survey unit are pooled and ranked in order of increasing size from 1 to N, where $N = m+n$.

3. If several measurements are tied (have the same value), they are all assigned the average rank of that group of tied measurements.

4. If there are t "less than" values, they are all given the average of the ranks from 1 to t. Therefore, they are all assigned the rank $t(t+1)/(2t) = (t+1)/2$, which is the average of the first t integers. All observations below the detection limit should be treated as "less than" values. If more than 40 percent of the data from either the reference area or survey unit are "less than," the WRS test will be used. (The use of "less than" values in data reporting will be minimized to the extent practical.)

5. Sum the ranks of the adjusted measurements from the reference area, W_r.

6. Compare W_r with the critical value given in Table I.4 of the MARSSIM manual for

the appropriate values of n, m, and . If W_r is greater than the tabulated value, reject the hypothesis that the survey unit exceeds the release criterion.

Contaminant Not Present in Background

For situations in which the contaminant of interest is not present in the background, the following logic will be used to judge central tendency compliance with the release criteria:

Table 4-10
Summary of Statistical Tests
(Radionuclide not in background and radionuclide-specific measurements made)

Survey Result	Conclusion
All measurements less than $DCGL_W$	Survey unit meets release criterion
Any measurement greater than $DCGL_W$ (not to be used for survey units with less than 5 measurements)	Conduct Sign test and elevated measurement comparison

If statistical analysis is required for a survey unit, the one-sample Sign test will be used. The Sign test evaluates whether the median of the data is above or below the DCGL, and is applied as follows:

1. List the survey unit measurements, X_i , $i = 1, 2, 3..., N$.

2. Subtract each measurement, X_i, from the DCGLw to obtain the differences using equation 4-8. where $i = 1, 2, 3..., N$.

$$DCGL_W - X_i \qquad \text{[Equation 4-8]}$$

3. If any difference is exactly zero, discard it from the analysis, and reduce the sample size, N, by the number of such zero measurements.

4. Count the number of positive differences. The result is the test statistic S+. A positive difference corresponds to a measurement below the DCGLw and contributes evidence that the survey unit meets the release criterion.

5. Large values of S+ indicate that the null hypothesis (that the survey unit exceeds the release criterion) is false. The value of S+ is compared to the critical values in Table I.3 of the MARSSIM manual. If S+ is greater than the critical value, k, in that table, the null hypothesis is rejected. Otherwise, the null hypothesis is accepted.

4.4.7.4 Draw Conclusions and Document Survey

The last stage of data evaluation will consist of a conclusion whether the results of the survey meet the design objectives. If a conclusion that design objectives have been met cannot be supported, resurvey, reclassification, remedial action, or some combination of those measures may be required. The survey will not be complete until the conclusion that design objectives have been met can be supported.

Documentation of the Final Status Survey will provide a complete and unambiguous record of the radiological status of the survey unit, relative to the established DCGLs. Sufficient data and information should be provided to enable an independent re-creation and evaluation at some future time. To the extent practicable, this report will be a stand-alone document with minimum information incorporated by reference. The report will be independently reviewed and will be approved by a person, designated by the project manager, who is capable of evaluating all aspects of the report prior to release, publication, or distribution.

4.5 FINAL STATUS SURVEY QA/QC

Section 5.0 describes the Quality Assurance Program (QAP) which will ensure that the results of the Final Status Survey are accurate and that the uncertainties have been considered.

5. QUALITY ASSURANCE 5-1

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5. QUALITY ASSURANCE

Decommissioning activities will be performed in a manner to ensure the results are accurate and that uncertainties have been adequately considered. The quality assurance program will operate in all stages of decommissioning through the final survey, validation of the data, and the interpretation of the results to verify that this has occurred.

Persons or organizations responsible for ensuring that the quality assurance program has been established and verifying that activities affecting quality have been correctly performed will have sufficient authority, access to work areas, and organizational freedom to:

- identify quality problems;
- initiate, recommend, or provide solutions to quality problems through designated channels;
- verify implementation of solutions; and,
- ensure that further decommissioning activities are controlled until proper disposition of a nonconformance or deficiency has occurred.

Such persons or organizations will have direct access to responsible management at a level where appropriate action can be taken. Such persons or organizations will report to a management level such that required authority and organizational freedom are provided, including sufficient independence from cost and schedule considerations.

The major aspects of the quality assurance program for the decommissioning activities are discussed in the following sections.

5.1 QUALITY ASSURANCE PROJECT PLAN

For execution of decommissioning activities at the Mallinckrodt C-T Project, a Quality Assurance Project Plan (QAPP), consistent with applicable guidelines will be developed . The QAPP will be reviewed and approved by Mallinckrodt prior to implementation. The objective of the QAPP is to ensure confidence in the sampling, analysis, interpretation and use of radiological data generated during the decommissioning project.

The QAPP will ensure collection of reliable data by serving as the instrument of control for field and analytical activities associated with the project. Stated within the QAPP are the quality assurance policies, quality control criteria, and reporting requirements that must be followed by all site and contractor personnel when carrying out their assigned responsibilities on this project. The QAPP describes the functional activities and quality

assurance/quality control (QA/QC) protocols necessary to collect data of adequate quality.

5.2 PROCEDURES

Supporting Quality Implementing Procedures (QIPs) will provide step-by-step details for complying with project QA requirements. The final radiological survey, including development of sampling plans, direct measurements, sample analysis, instrument calibration, daily functional checks of instruments, and sampling methods will be performed according to written procedures. These written procedures will be reviewed and approved by the Mallinckrodt project manager.

5.3 SUBCONTRACTORS

The activities to be conducted during decommissioning will require the services of a decommissioning contractor and various specialty subcontractors such as a qualified drilling contractor or a licensed surveyor. Contractor activities will be under the direct supervision of Mallinckrodt personnel in accordance with the QAPP. Subcontractor activities will be under the direct supervision of the decommissioning contractor personnel, also in accordance with the QAPP. Vendors providing quality related goods or services for this project would be selected from an Approved Suppliers List maintained by the decommissioning contractor's Quality Assurance personnel or by Mallinckrodt.

5.4 LABORATORY SERVICES

A qualified laboratory recommended by the decommissioning contractor and approved by Mallinckrodt will perform the radiological analytical laboratory services for the project. The laboratory will be responsible for all bench level QA/QC, data reduction, data reporting, and analytical performance monitoring. Laboratory accuracy will be evaluated by the analysis of blank and spiked samples. Sample handling protocols, analytical procedures, and reporting procedures employed by the analytical laboratory will be described in the laboratory's Quality Assurance Plan.

The laboratory will be responsible for assuring that all appropriate laboratory personnel are thoroughly familiar with the Quality Assurance Project Plan and good laboratory practices, and that all appropriate laboratory personnel meet the requisite qualifications for their positions within the laboratory. The laboratory Director, or his equivalent/representative, will review and approve all reports. The Director will also be responsible for assuring laboratory personnel have appropriate training to perform assigned responsibilities, and for daily management of the laboratory and its staff.

The laboratory will have a QA designee who will be responsible for assuring that the QA/QC requirements of the QAPP, the laboratory Quality Assurance Plan, and its

associated operating procedures are strictly followed. The QA designee will be responsible for review of data, alerting the Mallinckrodt decommissioning Project Manager and the Contractor Project Manager of the need for corrective action (when necessary), performing internal audits as specified by the QAPP, and maintenance of the QC records. The QA designee will also be responsible for preparing project specific QA/QC plans, as necessary.

5.5 SURVEYS AND SAMPLING ACTIVITIES

Trained individuals following written procedures will perform surveys using properly calibrated instruments. The custody of samples will be tracked from collection to analysis. Final survey data will be retained for a minimum of five years. The designated analytical laboratory will collect a split sample when desired by the NRC to obtain samples that are duplicates of those to be analyzed. When this operation is performed, the procedure for obtaining duplicate samples will be followed.

QC hold points will be utilized as necessary to ensure quality of surveys and sampling. Hold points will also be used to ensure that debris is moved only after QA has verified that the proper sampling and survey information for the debris in question has been obtained.

5.6 DOCUMENTATION

Data will be recorded and documented in a data management system. Entries will include the location of the survey or sampling point on the appropriate building grid. Data management personnel will also ensure that chain-of-custody and data management procedures are followed for decommissioning-related samples. The decommissioning contractor's procedures for proper handling, shipping and storage of samples will be used.

Both direct measurements and analytical results will be documented. The results for each survey measurement and sample analysis will be placed on the building map at the appropriate grid block location, and will also be listed in tabular form (i.e., result versus sample or survey location).

Data will be recorded in an orderly and verifiable way and reviewed for accuracy and consistency. Every step of the decommissioning process, from training personnel to calculating and interpreting the data, shall be documented in a way that lends itself to audit. Records of training to demonstrate qualification will also be maintained.

5.7 EQUIPMENT MAINTENANCE AND CALIBRATION

Measuring equipment will be maintained, calibrated, and tested according to Regulatory Guides 4.15 and 4.16 recommendations. Further, the procedures, responsibilities, and schedules for calibrating and testing equipment will be documented. Procedures will

meet the requirements of ANSI N323-1978. Calibration and maintenance of laboratory instrumentation will also meet the requirements of INPO 83-016.

Proper maintenance of equipment varies, but maintenance information and use limitations are provided in the vendor documentation. Measuring and analyzing equipment will be tested and calibrated before initial use and will be recalibrated if maintenance or modifications could invalidate earlier calibrations. Field and laboratory equipment, specifically used for obtaining final radiological survey data, will be calibrated based on standards traceable to NIST. In those cases where NIST-traceable standards are not available, standards of an industry-recognized organization (for example, the New Brunswick Laboratory for various uranium standards) will be used. Minimum frequencies for calibrating equipment will be established and documented.

Measuring equipment will be tested at least once on each day the equipment is used. Test results will be recorded in tabular or graphic form and compared to predetermined, acceptable performance ranges. Equipment that does not conform to the performance criteria will be promptly removed from service until the deficiencies can be resolved.

5.8 DATA MANAGEMENT

5.8.1 Laboratory Data

Data reduction, QC review, and reporting will be the responsibility of the analytical laboratory. Data reduction includes all automated and manual processes for reducing or organizing raw data generated by the laboratory. The laboratory will provide a data package for each set of analyses that will include a copy of the raw data in electronic format, and any other information needed to check and recalculate the analytical results.

Once a data package is received from the laboratory, the analytical results and pertinent QA/QC data will be compiled onto standardized data formats. The data packages will serve as basic reference sheets for data validation, as well as for project data use.

5.8.2 Field Survey Data

The generation, handling, computations, evaluation and reporting of final radiological survey data will be as specified in the decommissioning contractor's procedures. Included in these procedures will be a system for data review and validation to ensure consistency, thoroughness and acceptability. Qualified health and safety, operations, and/or engineering personnel will review and evaluate survey data.

5.8.3 Data Evaluation

Prior to releasing data for use by project staff, selected data will undergo data evaluation based on intended end use of the data. Data points chosen for evaluation will be

examined to determine compliance with QA requirements and other factors that determine the quality of the data.

If sample data are rejected or data omissions are identified during the data validation, this data will be evaluated to judge the impact on the project. Other corrective action may include re-sampling and analyzing, evaluating and amending sampling and analytical procedures and accepting data acknowledging the level of uncertainty.

5.9 SAMPLE CHAIN-OF-CUSTODY

One of the most important aspects of sample management is to ensure that the integrity of the sample is maintained; that is, that there is an accurate record of sample collection, transport, analysis, and disposal. This ensures that samples are neither lost nor tampered with and that the sample analyzed in the laboratory is actually and verifiably the sample taken from a specific location in the field.

Sample custody will be assigned to one individual at a time. This will prevent confusion of responsibility. Custody is maintained when (1) the sample is under direct surveillance by the assigned individual, (2) the sample is maintained in a tamper-free container, or (3) the sample is within a controlled-access facility.

The individual responsible for sample collection will initiate a chain-of-custody record using a standard form provided by the decommissioning contractor. A copy of this form will accompany the samples throughout transportation and analyses; and any breach in custody or evidence of tampering will be documented.

5.10 AUDITS

Periodic audits will be performed to verify that decommissioning activities comply with established procedures and other aspects of the QAPP and to evaluate the overall effectiveness of the QA program. Mallinckrodt and Contractor Quality Assurance personnel will verify that qualified personnel are used to conduct audits to ensure that the applicable procedures are being properly implemented. The audits will be conducted on at least a quarterly basis, in accordance with written guidelines or checklists. Health and safety personnel will also conduct semiannual audits in their area of concern. External program audits may also be used at the discretion of either Mallinckrodt or contractor management. Audit results will be reported to both Mallinckrodt and contractor management in writing, and actions to resolve identified deficiencies will be tracked and appropriately documented.

6. FINANCIAL ASSURANCE 6-1

6. FINANCIAL ASSURANCE

The work required to reduce the level of residual radioactive material within the area of the Mallinckrodt St. Louis Plant where C-T operations were performed has been divided into two phases. Phase I is the above-grade decommissioning of the buildings and equipment. Phase II completes the decommissioning of the C-T Project by addressing the grade and below-grade structures and soil.

Mallinckrodt presently has in place financial assurance in the amount of $750,000. It is estimated that the cost of decommissioning activities described by this Phase I Plan is $10,655,000. This cost is based upon a detailed estimate as presented in Table 6-1.

Table 6-1
Decommissioning Costs

Activity	Labor Costs	Radioactive Waste Disposal Cost	Transportation Costs	Miscellaneous Expenses	Totals
Building Decontamination	$33,000	$16,000	$4,000	$19,000	$72,000
Building Demolition	$1,888,000	$1,052,000	$481,000	$953,000	$4,374,000
Equipment Decontamination	$876,000	$865,000	$1,791,000	$456,000	$3,988,000
Building and Equipment Surveys	$312,000	$0	$0	$202,000	$514,000
Project Management	$943,000	$189,000	$63,000	$63,000	$1,258,000
Regulatory Interface	$116,000	$144,000	$123,000	$66,000	$449,000
Totals	$4,168,000	$2,266,000	$2,462,000	$1,759,000.00	$10,655,000

Mallinckrodt will provide a total of $10,655,000 in the form of (Letter of Credit/Standby Trust Agreement, Certificate of Deposit, Surety Bond, etc.) within 90 days after approval of the Phase I Plan. Mallinckrodt may reduce the amount of financial assurance required as the project progresses. Financial assurance for the Phase II Plan will be provided within 90 days of approval of the Phase II Plan.

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
1	62	Tile Floors - Men's Area	77/106	1560	Clean	Non Impacted
2	62	Tile Floors - Women's Area	-80/-40	1560	Clean	Non Impacted
3	90	1st Floor Machinery	0/42	1100	Clean	Non Impacted
4	90	1st Floor and Drains, Etc.	TBD[1]	1100	Assumed Clean	Non Impacted
5	90	2nd Floor and Drains, Etc.	TBD	1100	Assumed Clean	Non Impacted
6	91	1st Floor Machinery	0/42	1100	Clean	Non Impacted
7	91	1st Floor and Drains, Etc.	TBD	1100	Assumed Clean	Non Impacted
8	91	2nd Floor and Drains, Etc.	TBD	1100	Assumed Clean	Non Impacted
9	91	2nd Floor Crane Room	343/389	700	Clean	Non Impacted
10	101	Former Gas Heaters/Roof Equipment	93/148	1800	Clean	Impacted Category 2
11	101	Incinerator Pad Area	553/868	600	Contaminated	Impacted Class 1 (Phase II)
12	101	Incinerator Unit/Equipment	935/1777	1800	Contaminated	Impacted Category 1
13	101	External Walls	TBD	600	Assumed Clean	Impacted Class 2
14	101	Roof Decking	318/368	600	Clean	Impacted Class 2
15	101	Roof ß/Gamma Conversion FC/NUREG=2000/640	N/A	N/A	N/A	N/A
16	200E	Roof	TBD	600	Assumed Clean	Impacted Class 3
17	200E	Roof Equipment	211/409	1560	Clean	Impacted Category 2
18	200W	Roof	TBD	600	Assumed Clean	Impacted Class 3
19	200W	Roof Equipment	8/136	1560	Clean	Impacted Category 2
20	201	Roof	TBD	600	Assumed Clean	Impacted Class 3
21	201	Roof Equipment	TBD	1560	Assumed Clean	Impacted Category 2
22	204	Roof	TBD	600	Assumed Clean	Impacted Class 3
23	204	Roof Equipment	116/624	400	Contaminated	Impacted Category 1
24	213	Pump Room - Interior Walls	1/78	1010	Clean	N/A - Demolish
25	213	Pump Room - Floor	310/509	1010	Clean	N/A - Demolish
26	213	Pump Room - Ceiling	129/272	1010	Clean	N/A - Demolish
27	213	Pump Room - Equipment	-126/18	500	Clean	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
28	213	North Room - Interior Walls	386/562	1010	Clean	N/A - Demolish
29	213	North Room - Floor	1748/3415	1010	Contaminated	N/A - Demolish
30	213	North Room - Ceiling	63/183	1010	Clean	N/A - Demolish
31	213	North Room - Equipment/Pipe	6500/1300	500	Contaminated	N/A - Demolish
32	213	North Foyer - Interior Walls	716/1017	1010	Contaminated	N/A - Demolish
33	213	North Foyer - Floor	698/1852	1010	Contaminated	N/A - Demolish
34	213	North Foyer - Ceiling	28/205	1010	Clean	N/A - Demolish
35	213	South Foyer - Interior Walls	-153/-38	1010	Clean	N/A - Demolish
36	213	South Foyer - Floor	182/287	1010	Clean	N/A - Demolish
37	213	South Foyer - Equipment	TBD	500	Assumed Contaminated	N/A - Demolish
38	213	South Foyer - Ceiling	79/299	1010	Clean	N/A - Demolish
39	213	South Foyer - Ceiling Pipe	1500/5400	500	Contaminated	N/A - Demolish
40	213	Women's BR - Interior Walls	-34/103	1010	Clean	N/A - Demolish
41	213	Women's BR - Floor	226/395	1010	Clean	N/A - Demolish
42	213	Women's BR - Ceiling	-11/149	1010	Clean	N/A - Demolish
43	213	Women's BR - Pipe/Equipment/Ledge	5723/1100	500	Contaminated	N/A - Demolish
44	213	Men's BR - Interior Walls	173/377	1010	Clean	N/A - Demolish
45	213	Men's BR - Ledge/Pipe/Equipment	1845/2023	500	Contaminated	N/A - Demolish
46	213	Men's BR - Floor	336/724	1010	Clean	N/A - Demolish
47	213	Men's BR - Ceiling	139/451	1010	Clean	N/A - Demolish
48	213	Breakroom - Interior Walls	TBD	1010	Assumed Clean	N/A - Demolish
49	213	Breakroom - Ledges/Pipe	TBD	1010	Assumed Clean	N/A - Demolish
50	213	Breakroom - Floor	96/126	1010	Clean	N/A - Demolish
51	213	Breakroom - Ceiling	TBD	1010	Assumed Clean	N/A - Demolish
52	213	Roof	TBD	660	Assumed Contaminated	N/A - Demolish
53	213	Cooling Tower	TBD	660	Assumed Contaminated	N/A - Demolish
54	213	South Exterior Wall	111/185	1010	Clean	N/A - Demolish
55	213	North Exterior Wall	2400/3300	1010	Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
56	213	West Exterior Wall	155/256	1010	Clean	N/A - Demolish
57	213	213/238 Pipe Rack	TBD	1010	Assumed Contaminated	N/A - Demolish
58	213	213/238 Steam Line Expansion	TBD	1010	Assumed Contaminated	N/A - Demolish
59	214	Interior Walls	232/528	1010	Clean	N/A - Demolish
60	214	Floor	1245/1984	1010	Contaminated	N/A - Demolish
61	214	Above Ceiling Tile Support	1344/4775	1010	Contaminated	N/A - Demolish
62	214	Exterior Walls	496/1202	1010	Contaminated	N/A - Demolish
63	214	Equipment	TBD	500	Assumed Clean	N/A - Demolish
64	214	East Side Pipe Rack	TBD	1010	Assumed Contaminated	N/A - Demolish
65	214	Roof	TBD	580	Assumed Contaminated	N/A - Demolish
66	219	Foam/Metal	825/1063	N/A	FUSRAP	N/A - Demolish
67	219	Concrete Supports	136/484	N/A	FUSRAP	N/A - Demolish
68	219	Roof Steel and Material	309/654	N/A	FUSRAP	N/A - Demolish
69	222	Roof	TBD	580	Assumed Clean	Impacted Class 3
70	223	Roof	TBD	400	Assumed Clean	Impacted Class 3
71	235	East Room - Interior Walls	22/81	910	Clean	Impacted Class 3
72	235	East Room - Floor	-41/9	910	Clean	Impacted Class 3
73	235	East Room - Ceiling	160/287	910	Clean	Impacted Class 3
74	235	West Room - Interior Walls	-163/113	910	Clean	Impacted Class 3
75	235	West Room - Floor	197/280	910	Clean	Impacted Class 3
76	235	West Room - Expansion Joint	341/571	910	Clean	Impacted Class 3
77	235	West Room - Ground Rod Holes	884/18000	910	Contaminated	Impacted Class 1
78	235	West Room - Ceiling/ I Beams	172/253	910	Clean	Impacted Class 3
79	235	North Exterior Wall	-46/13	910	Clean	Impacted Class 3
80	235	South Exterior Wall	159/328	910	Clean	Impacted Class 3
81	235	East Exterior Wall	1306/1720	910	Contaminated	Impacted Class 1
82	235	Roof Piping and Equipment	TBD	120	Assumed Contaminated	Impacted Category 2
83	235	Roof	TBD	910	Assumed Contaminated	Impacted Class 2

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
84	236	Interior Walls/Ledges	1326/1935	600	Contaminated	Impacted Class 1
85	236	Floor	23/180	600	Clean	Impacted Class 3
86	236	Ceiling/Equipment	107/192	150/600	Clean	Impacted Class 3 / Category 2
87	236	North Exterior Wall	312/578	600	Clean	Impacted Class 3
88	236	South Exterior Wall/Ledges	1107/1939	600	Contaminated	Impacted Class 1
89	236	West Exterior Wall/Ledges	1246/1973	600	Contaminated	Impacted Class 1
90	236	Roof	TBD	600	Assumed Contaminated	Impacted Class 2
91	236	Roof Equipment	TBD	150	Assumed Contaminated	Impacted Category 2
92	238	Main Room - North Interior Wall (PRI)	5123/7113	1100	Contaminated	N/A - Demolish
93	238	Main Room - South Interior Wall (PRI)	8461/1200	1100	Contaminated	N/A - Demolish
94	238	Main Room - East Interior Wall (PRI)	3520/5000	1100	Contaminated	N/A - Demolish
95	238	Main Room - West Interior Wall (PRI)	8800/1150	1100	Contaminated	N/A - Demolish
96	238	Main Room - Floor (PRI)	3000/1420	1100	Contaminated	N/A - Demolish
97	238	Main Room - Overheads (PRI)	4000/5500	1100	Contaminated	N/A - Demolish
98	238	Main Room - 1st Floor Equipment (PRI)	3240/5800	1000	Contaminated	N/A - Demolish
99	238	Mezzanine - North Interior Wall (PRI)	109/322	1100	Assumed Contaminated	N/A - Demolish
100	238	Mezzanine - South Interior Wall (PRI)	1969/2931	1100	Contaminated	N/A - Demolish
101	238	Mezzanine - East Interior Wall (PRI)	1053/2082	1100	Contaminated	N/A - Demolish
102	238	Mezzanine - West Interior Wall (PRI)	1334/2224	1100	Contaminated	N/A - Demolish
103	238	Mezzanine - Floor (PRI)	5470/6359	1100	Contaminated	N/A - Demolish
104	238	Mezzanine - Equipment (PRI)	4211/7000	1000	Contaminated	N/A - Demolish
105	238	Mezzanine - Overheads (PRI)	2628/4000	1100	Contaminated	N/A - Demolish
106	238	Mezzanine - Ceiling (PRI)	524/788	1100	Assumed Contaminated	N/A - Demolish
107	238	S. Switch Gear Room - Interior Walls (PRI)	3627/5068	1100	Contaminated	N/A - Demolish
108	238	S. Switch Gear Room - Floor (PRI)	4754/7614	1100	Contaminated	N/A - Demolish
109	238	S. Switch Gear Room - Equipment	TBD	1000	Assumed Clean	N/A - Demolish
110	238	S. Switch Gear Room - Roof	519/968	740	Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
111	238	S. Switch Gear Room - Exterior Walls, East	308/1031	1100	Assumed Contaminated	N/A - Demolish
112	238	S. Switch Gear Room - Exterior Walls, West	63/230	1100	Clean	N/A - Demolish
113	238	S. Switch Gear Room - Exterior Walls, South	128/128	1100	Clean	N/A - Demolish
114	238	S. Switch Gear Room - Ceiling (PRI)	749/1620	1100	Contaminated	N/A - Demolish
115	238	N. Switch Gear Room - Interior Walls (PRI)	2395/3979	1100	Contaminated	N/A - Demolish
116	238	N. Switch Gear Room - Floor	3360/2080	1100	Contaminated	N/A - Demolish
117	238	N. Switch Gear Room - Equipment	TBD	1000	Assumed Clean	N/A - Demolish
118	238	N. Switch Gear Room - Roof	598/1404	740	Contaminated	N/A - Demolish
119	238	N. Switch Gear Room - Exterior Walls	2326/1140	1100	Contaminated	N/A - Demolish
120	238	N. Switch Gear Room - Ceiling (PRI)	262/486	1100	Clean	N/A - Demolish
121	238	W. Switch Gear Room - Exterior Walls	92/220	1100	Clean	N/A - Demolish
122	238	Ball Mill Room, 1st Floor - Interior Walls (PRI)	3360/4726	1100	Contaminated	N/A - Demolish
123	238	Ball Mill Room, 1st Floor - Floor (PRI)	1300/1350	1100	Contaminated	N/A - Demolish
124	238	Ball Mill Room, 1st Floor - Equipment (PRI)	1240/1707	1000	Contaminated	N/A - Demolish
125	238	Ball Mill Room, Mezzanine - Interior Walls (PRI)	2130/3679	1100	Contaminated	N/A - Demolish
126	238	Ball Mill Room, Mezzanine - Floor (PRI)	5756/8432	1100	Contaminated	N/A - Demolish
127	238	Ball Mill Room, Mezzanine - Exterior Walls	866/2063	1100	Contaminated	N/A - Demolish
128	238	Ball Mill Room, Mezzanine - Ceiling	TBD	1100	Assumed Contaminated	N/A - Demolish
129	238	Ball Mill Room, Mezzanine - Roof/Equipment	TBD	1800	Assumed Contaminated	N/A - Demolish
130	238	Ball Mill Room, Mezzanine - Equipment	1240/1710	1000	Contaminated	N/A - Demolish
131[2]	238	Blender Room, 1st Flr - Interior Walls (PRI)	2368/3900	1100	Contaminated	N/A - Demolish
132[2]	238	Blender Room, 1st Flr - Floor (PRI)	6587/8466	1100	Contaminated	N/A - Demolish
133[2]	238	Blender Room, 1st Flr - Ceiling/Beams (PRI)	1491/2543	1100	Contaminated	N/A - Demolish
134[2]	238	Blender Room, 1st Flr - Equipment-All (PRI)	1139/1534	1000	Contaminated	N/A - Demolish
135[2]	238	Blender Room, 2nd Flr - West Wall (PRI)	320/496	1100	Clean	N/A - Demolish
136[2]	238	Blender Room, 2nd Flr - Floor (PRI)	2979/9715	1100	Contaminated	N/A - Demolish
137[2]	238	Blender Room, 3rd Flr - West Wall (PRI)	238/238	1100	Clean	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
138[2]	238	Blender Room, 3rd Flr - Floor/Landing (PRI)	1070/1070	1100	Assumed Contaminated	N/A - Demolish
139[2]	238	Blender Room, 3rd Flr - Roof/Equipment	TBD	1800	Assumed Contaminated	N/A - Demolish
140[2]	238	Blender Room, 3rd Flr - Exterior Walls	4870/8060	1100	Contaminated	N/A - Demolish
141	238	Metal Diss. Room, 2nd Flr - S/E Walls (PRI)	2228/4323	1100	Contaminated	N/A - Demolish
142	238	Metal Diss. Room, 2nd Flr - Floor (PRI)	1986/3551	1100	Contaminated	N/A - Demolish
143	238	Metal Diss. Room, 2nd Flr - Interior Walls (PRI)	544/957	1100	Assumed Contaminated	N/A - Demolish
144	238	Metal Diss. Room, 2nd Flr - Floor (PRI)	1193/2208	1100	Contaminated	N/A - Demolish
145	238	Metal Diss. Room, 2nd Flr - Ceiling	TBD	1100	Assumed Contaminated	N/A - Demolish
146	238	Metal Diss. Room, 2nd Flr - Roof	TBD	740	Assumed Contaminated	N/A - Demolish
147	238	Metal Diss. Room, 2nd Flr - Exterior Walls	359/1098	1100	Assumed Contaminated	N/A - Demolish
148	238	Metal Diss. Room, 2nd Flr - Equipment (PRI)	551/828	1000	Assumed Contaminated	N/A - Demolish
149	238	Calciner Room, 1st Flr - Interior Walls (PRI)	259/370	1100	Clean	N/A - Demolish
150	238	Calciner Room, 1st Flr - Floor (PRI)	766/1366	1100	Contaminated	N/A - Demolish
151	238	Calciner Room, 1st Flr - Equipment (PRI)	351/380	1000	Assumed Contaminated	N/A - Demolish
152	238	Calciner Room, 2nd Flr - Interior Walls (PRI)	48/157	1100	Clean	N/A - Demolish
153	238	Calciner Room, 2nd Flr - Floor (PRI)	667/934	1100	Assumed Contaminated	N/A - Demolish
154	238	Calciner Room, 2nd Flr - Ceiling (PRI)	84/84	1100	Clean	N/A - Demolish
155	238	Calciner Room, 2nd Flr - Equipment (PRI)	348/374	1000	Assumed Contaminated	N/A - Demolish
156	238	Calciner Room, 2nd Flr - Exterior Walls	194/323	1100	Clean	N/A - Demolish
157	238	Calciner Room, 2nd Flr - Roof	TBD	740	Assumed Contaminated	N/A - Demolish
158	238	North Exterior Wall	4471/8411	1100	Contaminated	N/A - Demolish
159	238	South Exterior Wall	4029/5865	1100	Contaminated	N/A - Demolish
160	238	East Exterior Wall	5394/1016	1100	Contaminated	N/A - Demolish
161	238	West Exterior Wall	1214/3418	1100	Contaminated	N/A - Demolish
162	238	East Pipe Rack	367/643	1100	Assumed Contaminated	N/A - Demolish
163	238	Roof (PRI)	TBD	740	Assumed Contaminated	N/A - Demolish
164	238	Roof Equipment	0586/1623	1800	Contaminated	N/A - Demolish
165	238	Outside Equipment East Side	450/970	1800	Assumed Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
166	238	Outside Equipment North Side	TBD	1800	Assumed Contaminated	N/A - Demolish
167	238	Outside Equipment South Side (PRI)	0900/2150	1800	Contaminated	N/A - Demolish
168	240	East Exterior Wall/Ledges	-143/-38	1100	Clean	Impacted Class 3
169	240	North Exterior Wall/Ledges	-53/84	1100	Clean	Impacted Class 3
170	240	Floor/Doorways	43/110	1100	Clean	Impacted Class 3
171	240	Roof	TBD	1100	Assumed Clean	Impacted Class 3
172	245	Interior Walls	40/159	1000	Clean	Non Impacted
173	245	Floor/Drains	72/253	1000	Clean	Non Impacted
174	245	Ceiling	TBD	1000	Assumed Clean	Non Impacted
175	245	Windows	611/736	1000	Removed	N/A
176	245	North Exterior Wall	180/247	1000	Clean	Impacted Class 3
177	245	South Exterior Wall	TBD	1000	Assumed Clean	Impacted Class 3
178	245	Switch Gear, South Wall	TBD	1000	Removed	N/A
179	245	East Exterior Wall	217/328	1000	Clean	Impacted Class 3
180	245	West Exterior Wall/Equipment	53/688	1560	Assumed Clean	Impacted Category 2
181	245	West Exterior Pad	983/2058	1000	Contaminated	Impacted Class 1 (Phase II DP)
182	245	Exterior Pipe, NW Corner	173/229	1000	Clean	Impacted Category 2
183	245	Switch Gear, West Wall	301/378	1000	Removed	N/A
184	245	Roof	180/221	1000	Removed	N/A
185	245	Roof, Tannic Acid Shed	391/491	1000	Removed	N/A
186	245	Roof Equipment, Tannic Acid Shed	83/122	1000	Removed	N/A
187	245	Roof, Utility Room	501/665	1000	Removed	N/A
188	245	Roof, Spray Dryer Room	434/477	1000	Removed	N/A
189	245	Roof, Gallic Acid	N/A	1000	Removed	N/A
190	245	Roof, Equipment	225/267	1000	Removed	N/A
191	246A	Tile Floor	197/230	390	Assumed Contaminated	N/A - Demolish
192	246A	Interior Walls & Ceiling	-289/251	390	Clean	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
193	246A	Roof, Equipment	TBD	1070	Assumed Contaminated	N/A - Demolish
194	246A	Roof Metal Shed, Equipment	TBD	1070	Assumed Contaminated	N/A - Demolish
195	246A	East Wall Pipe Rack	TBD	230	Assumed Contaminated	N/A - Demolish
196	246A	East Exterior Wall	-106/102	390	Assumed Contaminated	N/A - Demolish
197	246A	North Exterior Wall	252/404	390	Assumed Contaminated	N/A - Demolish
198	246A	South Exterior Wall	233/964	390	Assumed Contaminated	N/A - Demolish
199	246A	246A/238 Pipe Rack	TBD	390	Assumed Contaminated	N/A - Demolish
200	246A	246A/238 Steam Line Expansion	TBD	390	Assumed Contaminated	N/A - Demolish
201	246B	1st Floor - Interior Walls (PRI)	614/844	390	Contaminated	N/A - Demolish
202	246B	1st Floor - Floor (PRI)	334/427	390	Contaminated	N/A - Demolish
203	246B	1st Floor - Ceiling	TBD	390	Assumed Contaminated	N/A - Demolish
204	246B	1st Floor - Equipment (PRI)	619/881	230	Contaminated	N/A - Demolish
205	246B	2nd Floor - Interior Walls/Ceiling (PRI)	1370/2634	390	Contaminated	N/A - Demolish
206	246B	2nd Floor - Floor (PRI)	1289/1802	390	Contaminated	N/A - Demolish
207	246B	2nd Floor - Equipment (PRI)	1428/2737	230	Contaminated	N/A - Demolish
208	246B	North Exterior Wall	35/228	390	Assumed Contaminated	N/A - Demolish
209	246B	South Exterior Wall	165/298	390	Assumed Contaminated	N/A - Demolish
210	246B	Roof	TBD	110	Assumed Contaminated	N/A - Demolish
211	246B	Roof Equipment	283/537	1070	Contaminated	N/A - Demolish
212	246B	246B/238 Elevated Walkway	TBD	1070	Assumed Contaminated	N/A - Demolish
213	246B	Elevated Walkway Roof	282/716	110	Contaminated	N/A - Demolish
214	247A	1st Floor - Interior Walls (PRI)	328/595	390	Contaminated	N/A - Demolish
215	247A	1st Floor - Floor (PRI)	627/776	390	Contaminated	N/A - Demolish
216	247A	1st Floor - Ceiling (PRI)	TBD	390	Assumed Contaminated	N/A - Demolish
217	247A	1st Floor - Equipment (PRI)	542/922	230	Contaminated	N/A - Demolish
218	247A	2nd Floor - Interior Walls (PRI)	214/407	390	Assumed Contaminated	N/A - Demolish
219	247A	2nd Floor - Ceiling	TBD	390	Assumed Contaminated	N/A - Demolish
220	247A	2nd Floor - Floor (PRI)	1915/2676	390	Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
221	247A	2nd Floor - Equipment (PRI)	1207/1900	230	Contaminated	N/A - Demolish
222	247A	North Exterior Wall	-55/65	390	Assumed Contaminated	N/A - Demolish
223	247A	South Exterior Wall	88/170	390	Assumed Contaminated	N/A - Demolish
224	247A	247 A Roof	TBD	660	Assumed Contaminated	N/A - Demolish
225	247A	247 A Roof Equipment	82/147	470	Assumed Contaminated	N/A - Demolish
226	247B	1st Floor - Interior Walls (PRI)	131/204	390	Assumed Contaminated	N/A - Demolish
227	247B	1st Floor - Floor (PRI)	322/387	390	Assumed Contaminated	N/A - Demolish
228	247B	1st Floor - Ceiling (PRI)	TBD	390	Assumed Contaminated	N/A - Demolish
229	247B	1st Floor - Equipment (PRI)	264/363	230	Contaminated	N/A - Demolish
230	247B	2nd Floor - Interior Walls (PRI)	49/110	390	Assumed Contaminated	N/A - Demolish
231	247B	2nd Floor - Floor (PRI)	209/271	390	Assumed Contaminated	N/A - Demolish
232	247B	2nd Floor - Equipment (PRI)	244/340	230	Contaminated	N/A - Demolish
233	247B	2nd Floor - Ceiling (PRI)	374/457	390	Contaminated	N/A - Demolish
234	247B	North Exterior Wall	-57/102	390	Assumed Contaminated	N/A - Demolish
235	247B	South Exterior Wall	54/189	390	Assumed Contaminated	N/A - Demolish
236	247B	247 B Roof	TBD	660	Assumed Contaminated	N/A - Demolish
237	247B	247B Roof Equipment	125/437	470	Assumed Contaminated	N/A - Demolish
238	246/7 N. Pad	N. Switch Gear Room - Interior Walls	TBD	390	Assumed Contaminated	N/A - Demolish
239	246/7 N. Pad	N. Switch Gear Room - Floor	TBD	390	Assumed Contaminated	N/A - Demolish
240	246/7 N. Pad	N. Switch Gear Room - Equipment	TBD	230	Assumed Contaminated	N/A - Demolish
241	246/7 N. Pad	N. Switch Gear Room - Ceiling	TBD	390	Assumed Contaminated	N/A - Demolish
242	246/7 N. Pad	N. Switch Gear Room - Exterior Walls	TBD	390	Assumed Contaminated	N/A - Demolish
243	246/7 N. Pad	N. Switch Gear Room - Roof	TBD	660	Assumed Contaminated	N/A - Demolish
244	248A	1st Floor - Interior Walls (PRI)	1.100386	770	Clean	N/A - Demolish
245	248A	1st Floor - Floor (PRI)	210/242	770	Clean	N/A - Demolish
246	248A	1st Floor - Ceiling (PRI)	57/63	770	Clean	N/A - Demolish
247	248A	1st Floor - Equipment (PRI)	345/466	430	Assumed Contaminated	N/A - Demolish
248	248A	2nd Floor - Interior Walls (PRI)	-317/-297	770	Clean	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
249	248A	2nd Floor - Floor (PRI)	119/150	770	Clean	N/A - Demolish
250	248A	2nd Floor - Ceiling (PRI)	119/204	770	Clean	N/A - Demolish
251	248A	2nd Floor - Equipment (PRI)	735/1354	430	Contaminated	N/A - Demolish
252	248B	1st Floor - Interior Walls (PRI)	-159/-106	770	Clean	N/A - Demolish
253	248B	1st Floor - Floor (PRI)	100/147	770	Clean	N/A - Demolish
254	248B	1st Floor - Ceiling	54/54	770	Clean	N/A - Demolish
255	248B	1st Floor - Equipment	193/210	430	Assumed Contaminated	N/A - Demolish
256	248B	2nd Floor - Interior Walls (PRI)	-132/-82	770	Clean	N/A - Demolish
257	248B	2nd Floor - Floor (PRI)	100/130	770	Clean	N/A - Demolish
258	248B	2nd Floor - Ceiling (PRI)	121/200	770	Clean	N/A - Demolish
259	248B	2nd Floor - Equipment (PRI)	202/238	430	Assumed Contaminated	N/A - Demolish
260	248E. Pad	Switch Gear Room - Interior Walls/Floor	TBD	770	Assumed Clean	N/A - Demolish
261	248E. Pad	Switch Gear Room - Equipment	TBD	430	Assumed Clean	N/A - Demolish
262	248E. Pad	Switch Gear Room - Exterior Walls/Roof	TBD	770	Assumed Clean	N/A - Demolish
263	248E. Pad	West Exterior Wall	131/244	770	Assumed Clean	N/A - Demolish
264	248E. Pad	South Exterior Wall	18/80	770	Clean	N/A - Demolish
265	248E. Pad	East Exterior Wall	-33/26	770	Clean	N/A - Demolish
266	248A&B	North Exterior Wall	24/116	770	Clean	N/A - Demolish
267	248A&B	Roof	TBD	1100	Assumed Clean	N/A - Demolish
268	248A&B	North Wall Pipe Rack	TBD	770	Assumed Clean	N/A - Demolish
269	248A&B	Outside Equipment North Side	TBD	490	Assumed Clean	N/A - Demolish
270	248A&B	Outside Equipment South Side	TBD	770	Assumed Clean	N/A - Demolish
271	248A&B	248B/250 Pipe Line	TBD	490	Assumed Clean	N/A - Demolish
272	248A&B	Roof Equipment	27/61	490	Clean	N/A - Demolish
273	250	1st Floor - Rm 101 Closet	17/103	830	Clean	Impacted Class 2
274	250	1st Floor - Rm 103/A/B Doors	53/197	830	Clean	Impacted Class 2
275	250	1st Floor - Rm 110 Desk Drawer	-38/-38	1200	Clean	Impacted Class 2
276	250	1st Floor - Rm 115 Doorways	78/253	830	Clean	Impacted Class 2

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
277	250	1st Floor - Rm 119 Lockers	88/384	830	Clean	Impacted Class 2
278	250	1st Floor - Rm 120 Sink	626/1462	830	Contaminated	Impacted Category 1
279	250	1st Floor - Rm 122 Sink	1060/2173	830	Contaminated	Impacted Class 1
280	250	1st Floor - East Stairwell	1349/2185	830	Contaminated	Impacted Class 1
281	250	1st Floor - West Stairwell	716/1262	830	Contaminated	Impacted Class 1
282	250	1st Floor - CT Lead Well	-42/130	830	Clean	Impacted Class 2
283	250	1st Floor CT Lab Office - Interior Walls	-138/-112	830	Clean	Impacted Class 2
284	250	1st Floor CT Lab Office - Floor	25/80	830	Clean	Impacted Class 2
285	250	1st Floor CT Lab Office - Ceiling	-128/-12	830	Clean	Impacted Class 2
286	250	1st Floor CT Lab Hall - Interior Walls	26/50	830	Clean	Impacted Class 2
287	250	1st Floor CT Lab Hall - Floor	30/90	830	Clean	Impacted Class 2
288	250	1st Floor CT Lab Hall - Ceiling	-143/91	830	Clean	Impacted Class 2
289	250	1st Floor CT Lab Count - Interior Walls	-21/2	830	Clean	Impacted Class 2
290	250	1st Floor CT Lab Count - Floor	40/58	830	Clean	Impacted Class 2
291	250	1st Floor CT Lab Count - Ceiling	69/109	830	Clean	Impacted Class 2
292	250	1st Floor CT Lab Count - Ceiling Drain Pipes	783/1506	830	Contaminated	Impacted Class 1
293	250	2nd Floor Rm 201 - Countertops/Sinks	800/1749	830	Contaminated	Impacted Category 1
294	250	2nd Floor Rm 201 - East Hood	131/383	830	Clean	Impacted Category 2
295	250	2nd Floor Rm 201 - NE Hood	24322	830	Contaminated	N/A - Demolish
296	250	2nd Floor Rm 201 - West Hood	425/1123	830	Contaminated	N/A - Demolish
297	250	2nd Floor Rm 201 - North Wall	166	830	Clean	Impacted Class 2
298	250	2nd Floor Rm 202 - Floors	338/664	830	Remediated	Impacted Class 1
299	250	2nd Floor Rm 202 - Interior Walls	49/96	830	Remediated	Impacted Class 1
300	250	2nd Floor Rm 202 - East Hood	610/6157	830	Remediated	N/A
301	250	2nd Floor Rm 202 - West Hood	2661/6157	830	Remediated	N/A
302	250	2nd Floor Rm 202 - Countertops	44/202	830	Remediated	N/A
303	250	2nd Floor Rm 205 - East Wall	78/875	830	Assumed Clean	Impacted Class 2
304	250	2nd Floor Rm 207 - Sinks	698/1778	830	Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

Appendix A

Area	Bldg.	Location / Surface	Result*	Criteria**	Evaluation	Classification
305	250	2nd Floor Rm 207 - Staging Area	43/118	830	Clean	Impacted Class 2
306	250	2nd Floor Rm 209 - West Hood	1994/6464	830	Contaminated	N/A - Demolish
307	250	2nd Floor Rm 209 - East Hood	760/3452	830	Contaminated	N/A - Demolish
308	250	2nd Floor Rm 209 - Countertops	-117/3	830	Clean	Impacted Category 2
309	250	2nd Floor Rm 209 - Drains	-2/468	830	Clean	Impacted Category 2
310	250	2nd Floor Rm 209 - Floors	11	830	Clean	Impacted Class 2
311	250	2nd Floor Rm 210 - Floors	-59/-16	830	Clean	Impacted Class 2
312	250	2nd Floor Rm 210 - North and West Walls	1/2	830	Clean	Impacted Class 2
313	250	2nd Floor Rm 210 - Mechanical Room Equipment	447/945	1200	Assumed Contaminated	N/A - Demolish
314	250	2nd Floor Rm 211 - West Hood	109/158	1200	Clean	Impacted Category 2
315	250	2nd Floor Rm 211 - Countertops	-71/9	1200	Clean	Impacted Category 2
316	250	2nd Floor Rm 211 - Drain	-92	1200	Clean	Impacted Category 2
317	250	2nd Floor Rm 211 - Cabinets	69/269	1200	Clean	Impacted Category 2
318	250	East Exterior Wall	52/88	830	Clean	Impacted Class 2
319	250	South Exterior Wall	TBD	830	Assumed Clean	Impacted Class 2
320	250	Roof Decking	52/62	580	Clean	Impacted Class 2
321	250	East Pad	TBD	830	Assumed Contaminated	Impacted Class 2

1 Data will be collected during decommissioning.

2 The results include Potassium 40.

* Mean/95 % Confidence (βpm/100 cm^2)

**βpm/100 cm^2

APPENDIX B
BUILDING SURFACE MAPS

1.0 OVERVIEW

This appendix contains selected examples of building surface maps. These surfaces were selected to depict typical results for various surface materials (concrete, brick, steel, etc.), surface location (interior or exterior wall, floor, roof, etc.), and degree of contamination (highly contaminated, clean, etc.).

The map depicts a grid system of the area surveyed. The origin of the grid is the south west corner of the surface. Gamma and beta/gamma measurements were taken at each grid intersection. All maps reference the 25 mrem/yr industrial worker criteria as defined in Table 2-2.

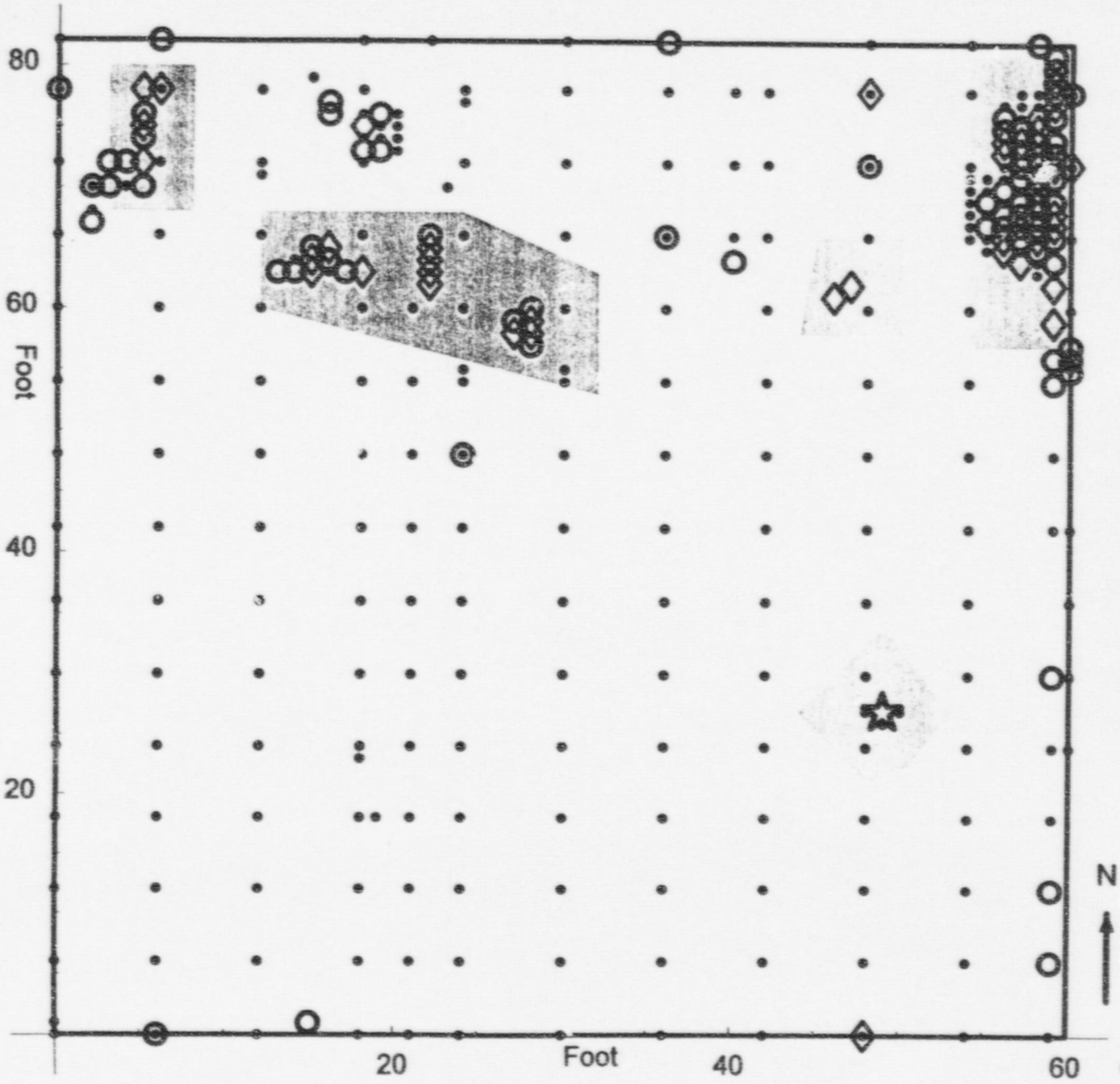
The maps also give the following information:

- number of data points
- square footage of survey area
- average value of data points in βpm/100 cm^2
- maximum value of data points in βpm/100 cm^2
- 95% confidence level
- number of points greater than criteria
- number of points greater than three times the criteria
- whether the data was corrected for background

The grid map has four symbols that mark data points that fall into one of the four following categories:

1. greater than three times the criteria
2. greater than the criteria
3. greater than half of the criteria
4. less than half of the criteria

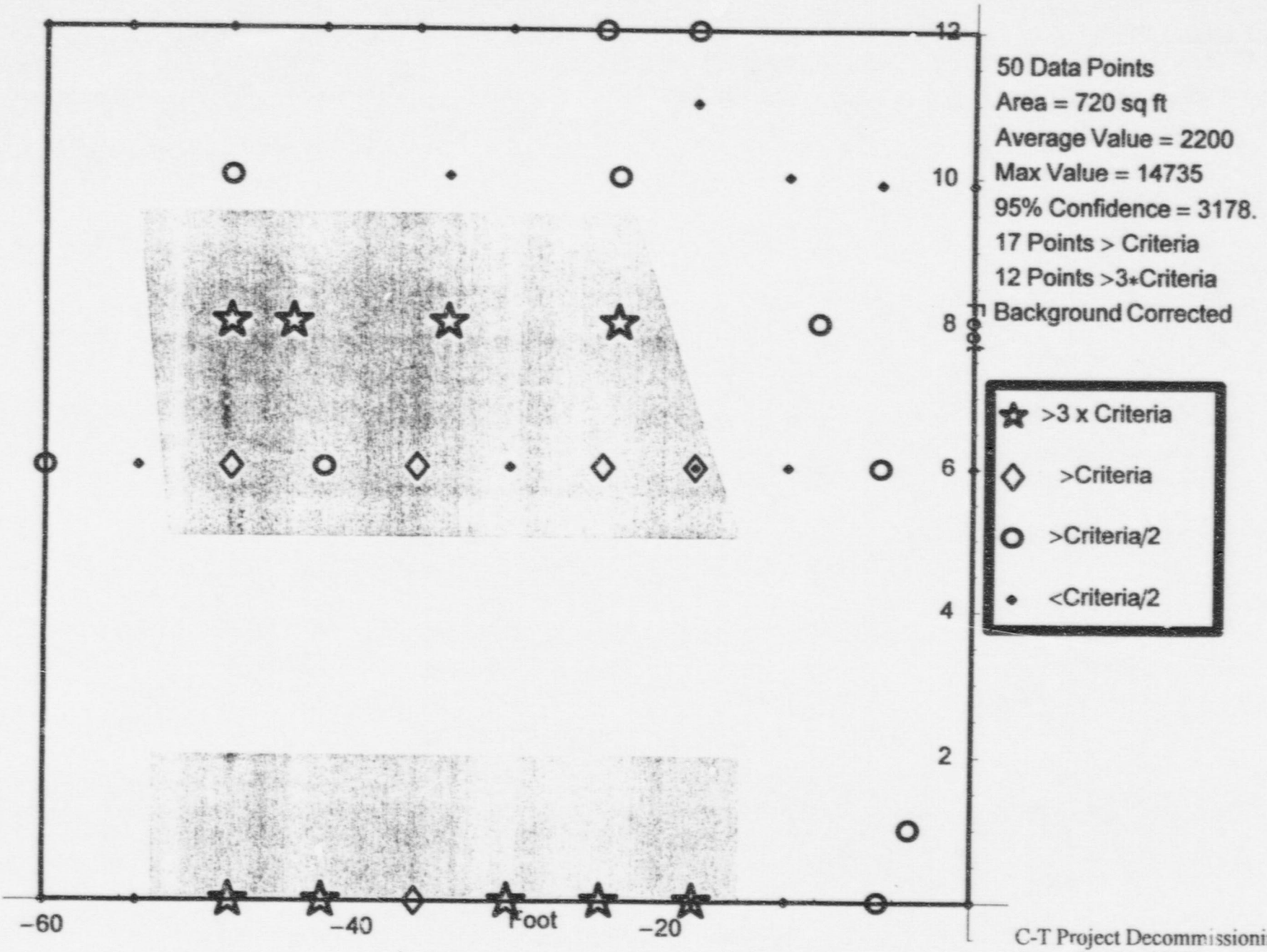
Building 248 First Floor



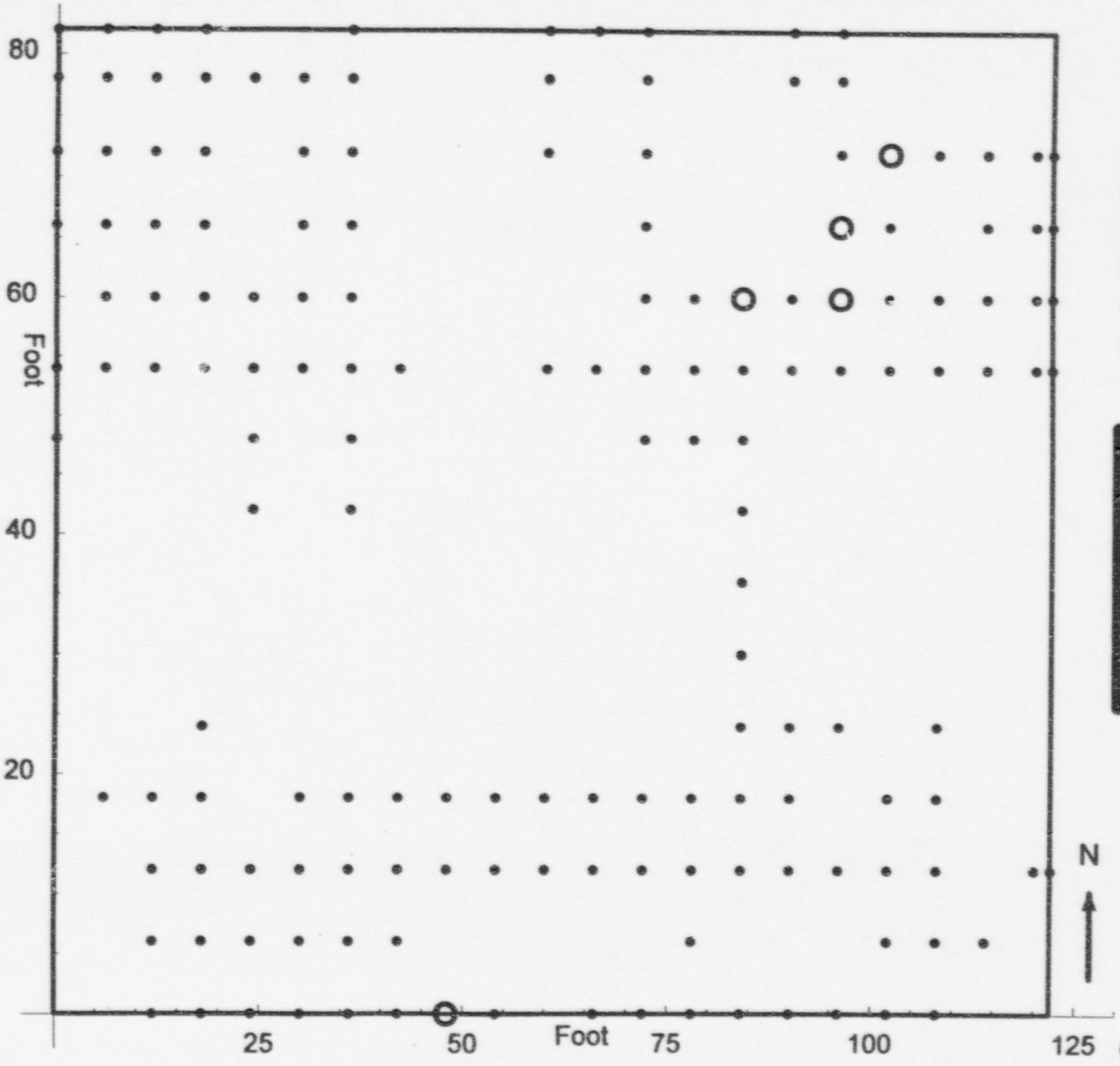
456 Data Points
Area = 4920 sq ft
Average Value = 274
Max Value = 2377
95% Confidence = 305.
43 Points > Criteria
1 Points >3*Criteria
Background Corrected

- ☆ >3 x Criteria
- ◇ >Criteria
- ○ >Criteria/2
- • <Criteria/2

Building 213 North Exterior Wall



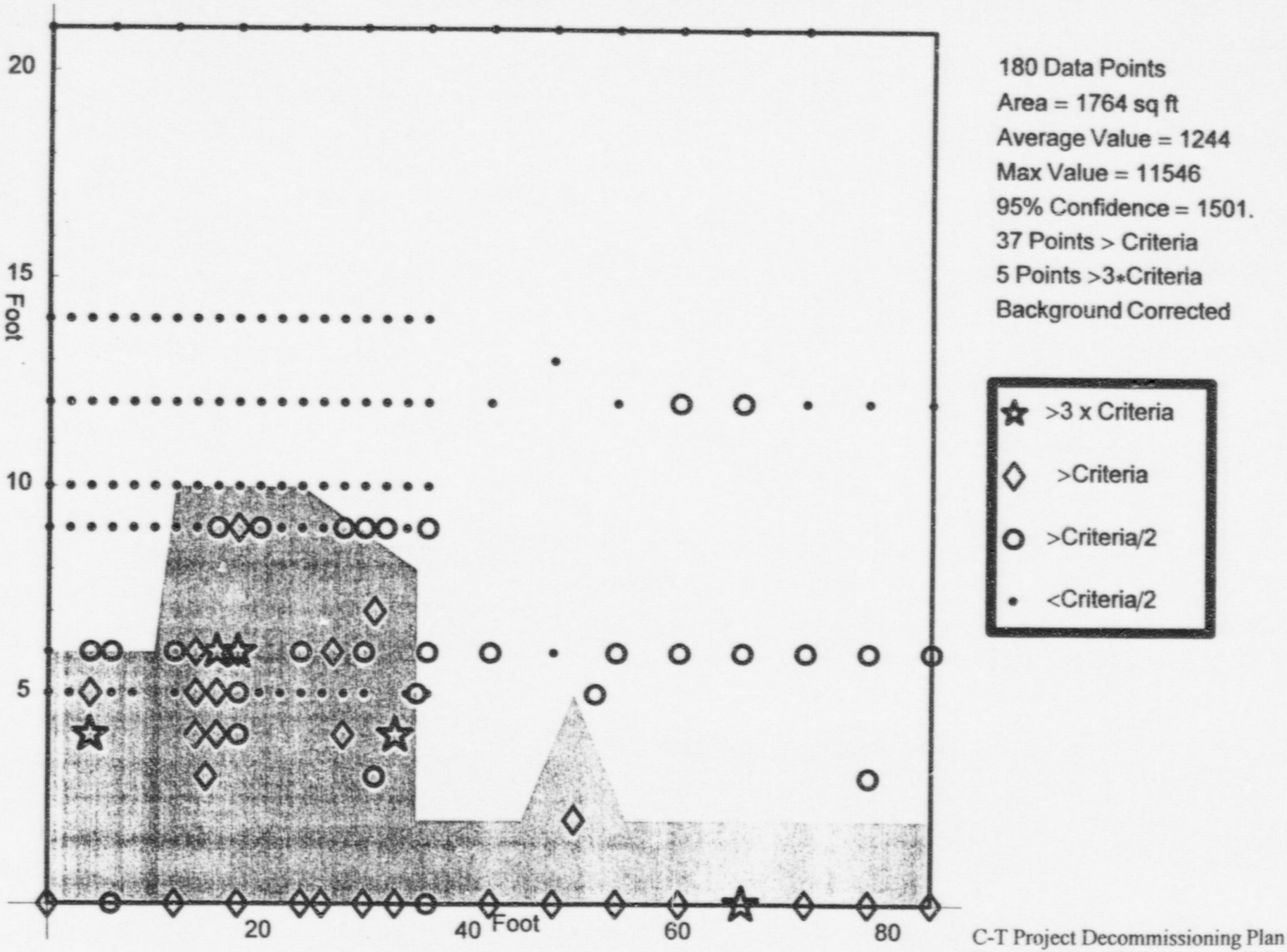
Building 238 Mezzanine



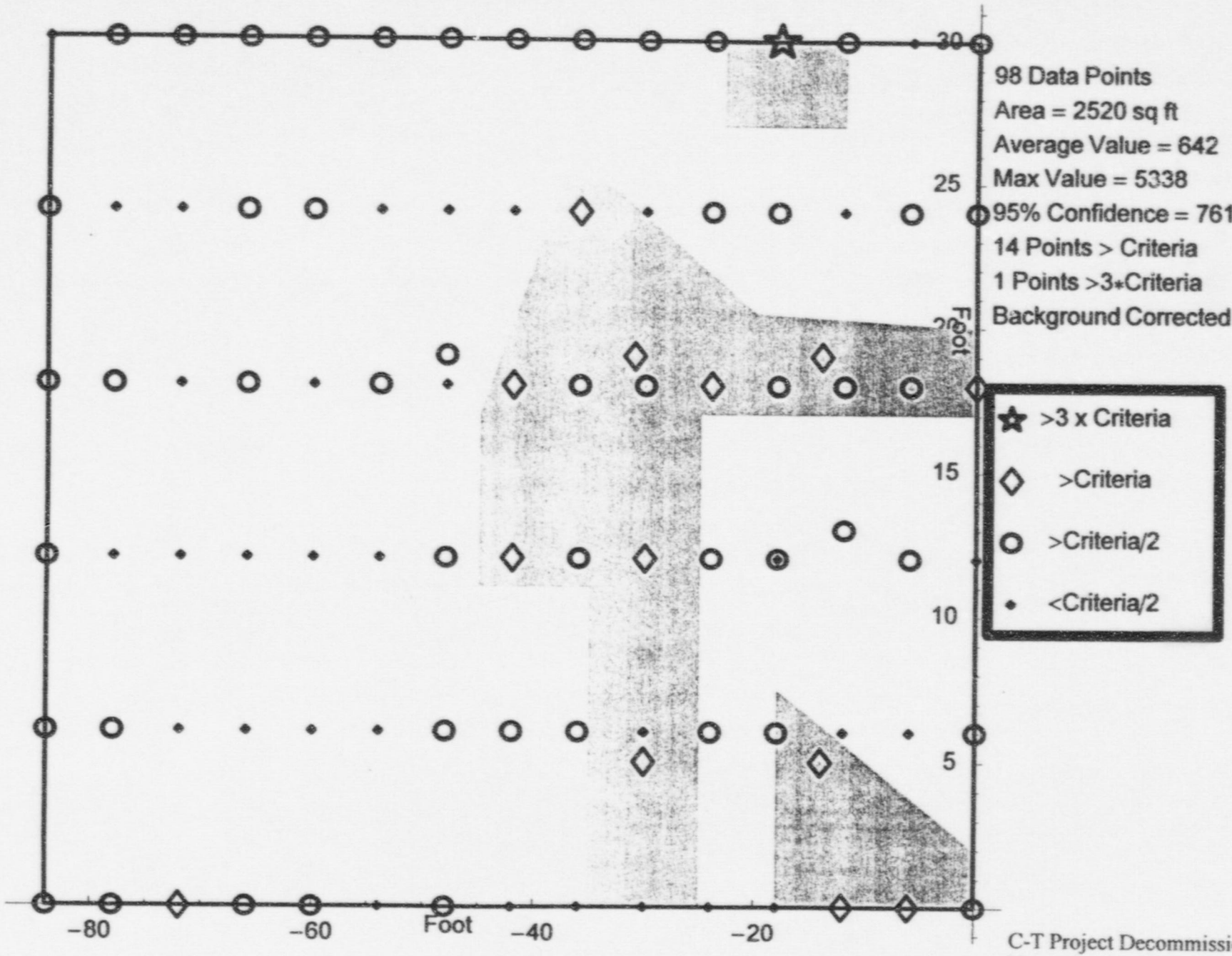
160 Data Points
Area = 10004 sq ft
Average Value = 146
Max Value = 1058
95% Confidence = 170.
0 Points > Criteria
0 Points >3*Criteria
Background Corrected

- >3 x Criteria
- >Criteria
- >Criteria/2
- <Criteria/2

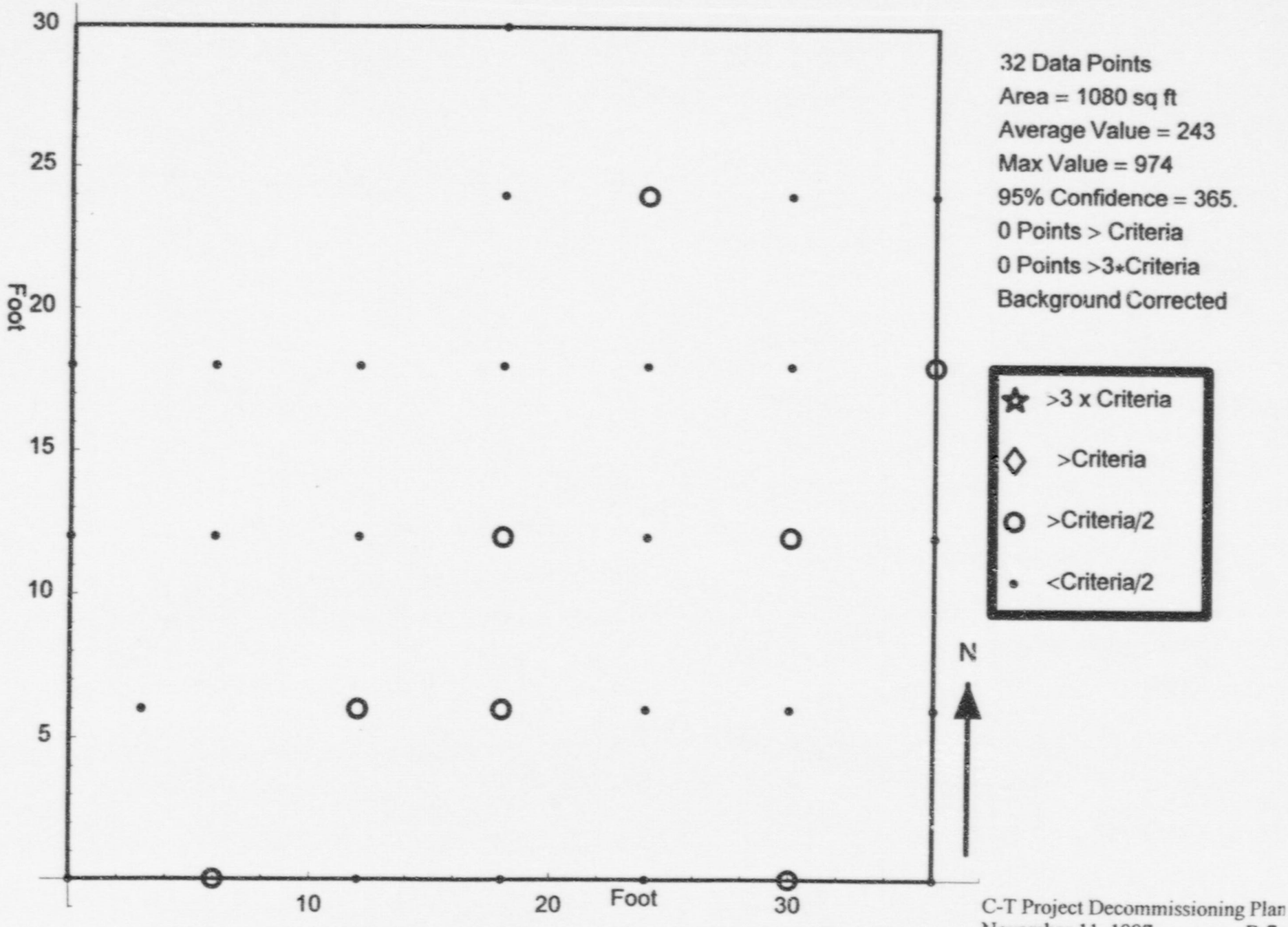
Building 235 East Wall



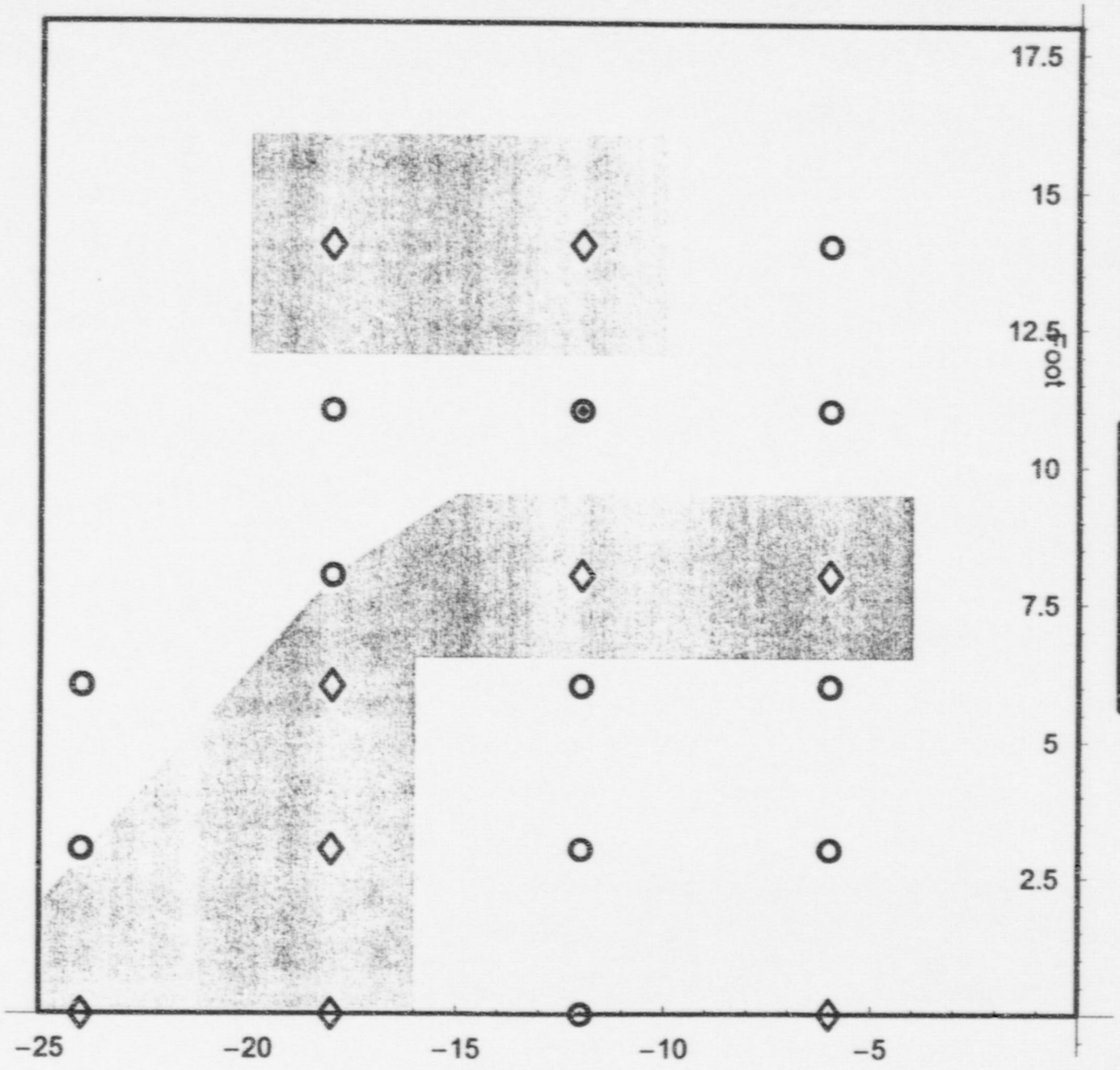
Building 248 West Exterior Wall



Building 213 Break Room Ceiling



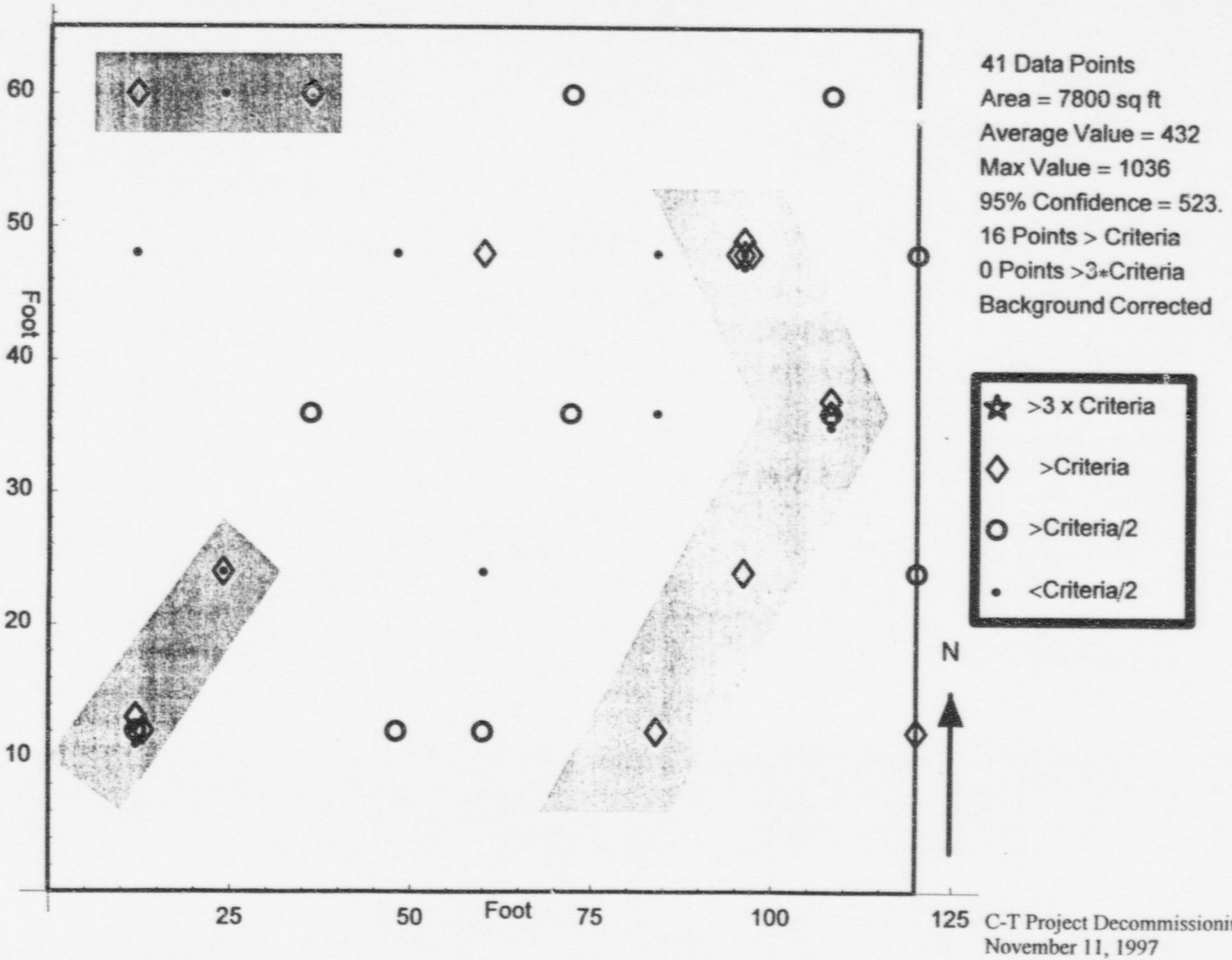
Building 246B South Interior Wall



22 Data Points
Area = 432 sq ft
Average Value = 381
Max Value = 1170
95% Confidence = 478.
9 Points > Criteria
1 Points >3*Criteria
Background Corrected

- ★ >3 x Criteria
- ◇ >Criteria
- ○ >Criteria/2
- ◆ <Criteria/2

Building 204 Roof



APPENDIX C
DERIVATION OF NUCLIDE LIMITS FOR BUILDING SURFACES

1.0 OVERVIEW

The purpose of this appendix is to describe the derivation of nuclide limits for building surfaces. Under newly adopted NRC regulations, 10 CFR 20, Subpart E, limits for radioactive material remaining after license termination are expressed in terms of radiation dose (effective dose equivalent) that might reasonably be anticipated from exposure to the residual radionuclides. However, the anticipated dose cannot be measured directly. Therefore, to implement the regulation, the dose limit must be translated into a corresponding limit of some related quantity that can be measured directly. In this decommissioning plan, that is done in two major steps. First, the dose limit is translated into nuclide limits in terms of nuclear transformations per unit time per unit area of building surface. That translation is described in this appendix. The second major step is the use of nuclide limits (derived as described in this appendix) in conjunction with building-specific nuclide mixes to derive limits expressed in terms of the total (over the entire nuclide mix) emission rate of a particular type of radiation. The derivation in the second major step is described in Appendix D.

One objective of this decommissioning plan is to remediate the buildings so that those remaining on the site after license termination may be used without restrictions. The dose limit specified in 10 CFR Part 20, Subpart E, for license termination without restrictions on use is 25 millirem per year. Therefore, the dose basis for the nuclide limits derived in this appendix is 25 millirem per year.

The derivation of the nuclide limits described in this appendix also occurs in a series of steps. The first step is the identification of an exposure scenario to serve as the basis for the nuclide limits to be derived. The exposure scenario specification describes how persons might come to be exposed to residual radioactive materials and includes values for parameters important in determining dose, such as exposure times. The ideal basis exposure scenario would be a reasonably anticipated exposure scenario that would result in the maximum dose from exposure to residual radioactive material after termination of the license. The exposure scenario definition process is described in Section 2.0 of this appendix.

Radiation dose assessment models are usually used to calculate radiation doses from input nuclide concentrations in some medium. However, because radiation dose from a nuclide scales directly to the nuclide concentration, other factors remaining constant, the results can be used to compute the nuclide concentration that corresponds to any particular dose limit. The radiation dose assessment model input and output and the computation of nuclide limits from the output are described in Section 3.0 of this appendix.

Nuclide concentration limits derived in Section 3.0 of this appendix account for radiation dose from progeny nuclides, but are expressed in terms of the concentration of the first nuclide in the decay chain. Some concentration limits in NRC guidance documents, such as FC 83-23 (USNRC, 1983) and NUREG-1500 (USNRC 1994) are expressed on a different basis, such as total activity in the decay chain. Because FC 83-23 surface nuclide limits are applied to equipment, nuclide limits for buildings have been reformulated in FC 83-23 format. This process is described in Section 4.0 of this appendix. The product of the work in that section of this appendix is a table of nuclide limits that serve as input to the second major step in the translation of dose limit to radiation emission limit.

2.0 EXPOSURE SCENARIO

This section specifies the exposure scenario to serve as the basis for derivation of nuclide limits on building surfaces. The exposure scenario specification describes how persons might come to be exposed to residual radioactive materials and would includes values for parameters important in determining dose, such as exposure times.

As noted in the overview, the ideal basis exposure scenario would be a reasonably anticipated exposure scenario that would result in the maximum dose from exposure to residual radioactive material after termination of the license. For purposes of this decommissioning plan, the scenario assumed for purposes of deriving nuclide limits for building surfaces is the building occupancy scenario from NUREG\CR-5512 (Kennedy, 1992). As explained in NUREG\CR-5512, this scenario was developed for generic pathway modeling. Consequently, the scenario specification incorporates some exposure assumptions that tend to be somewhat unrealistic and tend to cause the dose associated with this scenario to be substantially higher than might be expected for more likely scenarios specifically tailored to the C-T situation after license termination. A large fraction of the exposure scenarios that might be reasonably anticipated after termination of the license for the C-T areas would lead to substantially lower doses. It is considered likely that no other reasonably anticipated exposure scenario would lead to substantially higher radiation dose.

The building occupancy scenario involves chronic exposure to an individual for a full work year in a building that is part of a typical commercial facility. With regard to use as part of a commercial facility, the scenario fits the C-T situation well. The C-T buildings to remain after license termination have been used only for commercial purposes and most likely will continue to be for the remainder of their useful lives. The building occupancy scenario exposure pathways considered sufficiently important for analysis include external exposure to penetrating radiation from surface sources, inhalation of resuspended surface contamination, and inadvertent ingestion of surface contamination. Other exposure pathways exist, but are considered to be much less important in terms of potential dose (NUREG\CR-5512, Section 3.1.2).

The building occupancy scenario incorporates conservative assumptions regarding potential exposure characteristics. As noted above, the occupation time is assumed to be the full working year, 2,000 hours per year, equivalent to 50 weeks at 40 hours per week. The extent of surfaces bearing radionuclides at the limiting concentration is assumed to be sufficiently large as to be infinite in extent for purposes of estimating direct radiation dose. The extent is also sufficiently large to support a resuspension factor (ratio of air concentration to surface concentration) of 1×10^{-6} m^{-1}, as specified in NUREG\CR-5512. The value of the resuspension factor itself is considered conservative for application in a situation in which readily removable radionuclides have already been removed, as will be the case in the C-T situation. Furthermore, the nuclide inhalation dose factor typically specified is that for the chemical form likely to lead to the highest dose. Finally, the ingestion factor is assumed to be 10^{-4} m^2 per hour of exposure, and, for purposes of estimating dose from ingestion, the nuclide is assumed to be present in its most soluble form to maximize the calculated ingestion dose.

3.0 COMPUTER MODEL RESULTS

As noted in the overview, radiation dose assessment models are usually used to calculate radiation doses from input nuclide concentrations in some medium. However, because radiation dose from a nuclide scales directly to the nuclide concentration, other factors remaining constant, the results can be used to compute the nuclide concentration that corresponds to any particular dose limit. The RESRAD-BUILD computer code (Yu, 1994), developed by the United States Department of Energy, and commonly used in radiation dose assessments of this type, was used for this analysis. The radiation dose assessment model input and output and the computation of nuclide limits from the output are described in this section.

The mathematical model upon which RESRAD-BUILD is based differs in some substantial ways from the model developed in NUREG\CR-5512. Major differences are related to the inhalation and ingestion pathways. However, by careful specification of RESRAD-BUILD input parameter values one can used RESRAD-BUILD to simulate the NUREG\CR-5512 model. For example, the NUREG\CR-5512 inhalation model assumes a simple resuspension factor approach to calculation of the concentration of a nuclide in air. The nuclide concentration in air is equal to the surface concentration times a resuspension factor. RESRAD-BUILD uses a different approach. In its simplest implementation, which was used in this analysis, the building is modeled as a single compartment. A specified area is assumed to contain a specified radionuclide surface concentration. The rate of removal to air is determined by user-input values for the fraction of the surface concentration that is removable at all, the time over which that fraction is removed, and the fraction of the removed material that is transported to room air. The concentration in air is calculated by assuming that the removed material is mixed in air flowing through the room at a rate determined by user-input dimensions and a user input ventilation rate (volume fraction per unit time). Table C-1 shows how the

RESRAD-BUILD inhalation model parameters were adjusted to simulate the NUREG\CR-5512 model.

The RESRAD-BUILD and NUREG\CR-5512 ingestion pathway models also differ. In the NUREG\CR-5512 model, the ingestion rate is the product of the nuclide surface concentration, the exposure time, and a factor of 10^{-4} m^2 per hour of exposure. The model incorporates an implicit assumption that all of the radioactive material on the surface is available for removal. In RESRAD-BUILD, the ingestion rate is the product of the nuclide surface concentration, the exposure time, the fraction of surface nuclide considered actually removable, and an uptake factor expressed in terms of the fraction of removable nuclide inventory per unit exposure time. To make the RESRAD-BUILD calculation equivalent to the NUREG\CR-5512 calculation, it is necessary only to specify a value of 1 for the fraction removable and an uptake factor equal to the NUREG\CR-5512 factor divided by the source area. For the 36 m^2 source area specified in NUREG\CR-5512, the required uptake factor is 3×10^{-6}/h.

The input and output for the RESRAD-BUILD run used in this analysis (MI1) are listed in Attachment 1. The RESRAD-BUILD output for each nuclide can be interpreted as a dose factor (mrem/y per dpm/100 cm^2). The concentration limit corresponding to 25 mrem/y can be calculated by dividing 25 by the dose factor.

4.0 REFORMULATION OF RESULTS

Nuclide concentration limits derived in Section 3.0 of this appendix account for radiation dose from progeny nuclides, but are expressed in terms of the concentration of the first nuclide in the decay chain. Some concentration limits in NRC guidance documents, such as FC 83-23 (USNRC, 1983) and NUREG-1500 (USNRC, 1994) are expressed on a different basis, such as total activity in the decay chain. Because FC 83-23 surface nuclide limits are applied to equipment, nuclide limits for buildings have been reformulated in FC 83-23 format. Nuclide limits have also been reformulated in NUREG-1500 format for comparison purposes. Data and intermediate results are included in Table C-2 for the thorium-232 chain nuclides and in Table C-3 for the uranium-238 chain nuclides. The ultimate results are included as the last column in Table C-4. Table C-4 also includes FC 83-23 and NUREG-1500 limits (scaled to 25 mrem/y and recast in FC 83-23 format) for comparison. The values calculated using RESRAD-BUILD are quite close to NUREG-1500 values. The last column in Table C-4 is product of the work described in this appendix, and serves as input to the second major step in the translation of dose limit to radiation emission limit, described in Appendix D.

5.0 REFERENCES

Kennedy, 1992. Kennedy, W.E., and Strenge, D.L., "Residual Radioactive Contamination from Decommissioning: Volume 1, Technical Basis for Translating

Contamination Levels to Annual Total Effective Dose Equivalent," NUREG\CR-5512, Pacific Northwest Laboratory, Richland, Washington, September, 1992.

USNRC, 1983. "Termination of Byproduct, Source, and Special Nuclear Material Licenses," Policy and Guidance Directive FC-83-23, U.S. Nuclear Regulatory Commission, Washington, D.C., June 23, 1983.

USNRC, 1994. "Working Draft Regulatory Guide on Release Criteria for Decommissioning: NRC Staff's Draft for Comment," NUREG-1500, U.S. Nuclear Regulatory Commission, Washington, D.C., August, 1994.

Yu, 1994. Yu, C., LePoire, D.J., Jones, L.G., and Chen, S.Y., "RESRAD-BUILD: A Computer Model for Analyzing the Radiological Doses Resulting from the Remediation and Occupancy of Buildings Contaminated with Radioactive Material," ANL/EAD/LD-3, Argonne National Laboratory, Argonne, Illinois, November, 1994

Table C-1
Relation between NUREG/CR-5512 and RESRAD-B Airborne Concentration Models

NUREG/CR-5512 MODEL	
INPUT	
1	SC, source concentration, pCi/m2
1.00E-06	RF, resuspension factor, 1/m
OUTPUT	
1.00E-06	CA, air concentration, pCi/m3, SC*RF
RESRAD-B MODEL (1 COMPARTMENT, IGNORING DEPOSITION AND RESUSPENSION)	
INPUT	
1	SC, source concentration, pCi/m2
0.8	LAM, ventilation rate, 1/h
36	AR, area, m2
2.5	HT, height, m
0.5	FAR, fraction of removed material to air
1	FREM, fraction removable
10950	TR, removal time, d
24	UC1, h/d
OUTPUT	
7.20E+01	FLO, ventilation flow rate, m3/h, LAM*AR*HT
6.85E-05	RMR, removal rate, pCi/h, SC*AR*FAR*FREM/TR/UC1
9.51E-07	CA, concentration in air, pCi/m3, RMR/FLO

Table C-2
NUREG-1500 and C-T Surface Concentration Limits for Nuclides and Nuclide Chains
Th-232 Chain, 25 millirem per year, NUREG/CR-5512 Building Occupancy Scenario

Nuclide/Nuclide Chain	NUREG-1500 Dose Conversion Factor (DCF) mrem/y per dpm/100 cm2	ALPHA 1=yes 0=no	BETA 1=yes 0=no	NUREG-1500 Concentration Limit dpm/100 cm2		RESRAD-BUILD Run MI1 Dose Conversion Factor (DCF) mrem/y per dpm/100 cm2	C-T Concentration Limit Derived from RESRAD-BUILD dpm/100 cm2
Nuclide							
Thorium-232	2.03E-01	1	0	123		2.00E-01	125
Radium-228	2.09E-02	0	1	1196		1.30E-02	1923
Actinium-228	1.53E-05	0	1	1.6E+06		0	
Thorium-228	3.87E-02	1	0	646		3.20E-02	781
Radium-224	8.33E-05	1	0	3.0E+05		0	
Radon-220	--	1	0				
Polonium-216	--	1	0				
Lead-212	3.70E-06	0	1	6.8E+06		0	
Bismuth-212	2.51E-07	0.36	0.64	1.0E+08		0	
Polonium-212 (64%)	--	1	0				
Thallium-208 (36%)	--	0	1				
Lead-208	--	0	0				
Nuclide Chains							
Th-228_C (7 TRANSFORMS)	5.54E-03			4512		4.57E-03	5469
Ra-228_C (9 TRANSFORMS)	6.63E-03			3769		5.00E-03	5000
Th-232_C (10 TRANSFORMS)	2.63E-02			952		2.45E-02	1020

NUREG-1500 individual nuclide DCF values from NUREG-1500 Table A-2

Nuclide chain values computed from individual nuclide values, assuming equilibrium

DCF=total of individual nuclide DCFs divided by number of nuclides in chain (down to stable isotope)

Calculated values for NUREG-1500 agree with chain values tabulated in NUREG-1500 Table A-2

DCF is total dose per unit activity in the entire chain

Table C-3

NUREG-1500 and C-T Surface Concentration Limits for Nuclides and Nuclide Chains

U-238 Chain, 25 millirem per year, NUREG/CR-5512 Building Occupancy Scenario

Nuclide/Nuclide Chain	NUREG-1500 Dose Conversion Factor mrem/y per dpm/100 cm2	ALPHA 1=yes 0=no	BETA 1=yes 0=no	NUREG-1500 Concentration Limit dpm/100 cm2		RESRAD-BUILD Run MI1 Dose Conversion Factor (DCF) mrem/y per dpm/100 cm2	C-T Concentration Limit Derived from RESRAD-BUILD dpm/100 cm2
Nuclide							
Uranium-238	1.50E-02	1	0	1667		1.50E-02	1667
Thorium-234	1.52E-05	0	1	1.6E+06		0.00E+00	
Protactinium-234m	--	0	1				
Protactinium-234 (0.13%)	--	0	0				
Uranium-234	1.69E-02	1	0	1479		1.70E-02	1471
Thorium-230	4.01E-02	1	0	623		4.00E-02	625
Radium-226	1.57E-02	1	0	1592		2.10E-02	1190
Radon-222	--	1	0				
Polonium-218	--	1	0				
Lead-214 (99.98%)	--	1	0				
Astatine-218 (0.02%)	--	0	0				
Bismuth-214	--	1	0				
Polonium-214 (99.98%)	--	1	0				
Thallium-210 (0.02%)	--	0	0				
Lead-210	5.85E-02	0	1	427		6.90E-02	362
Bismuth-210	3.01E-04	0	1	8.3E+04		0.00E+00	
Polonium-210 (99.9+%)	8.31E-03	1	0	3008		0.00E+00	
Thallium-206 (0.00013%)	--	0	0				
Lead-206							
Nuclide Chains							
U-238_C (14 TRANSFORMS)	1.11E-02			2261		1.16E-02	2169
U-234_C (11 TRANSFORMS)	1.27E-02			1967		1.34E-02	1871
Th-230_C (10 TRANSFORMS)	1.23E-02			2034		1.30E-02	1923
Ra-226_C (9 TRANSFORMS)	9.20E-03			2717		1.00E-02	2500

NUREG-1500 individual nuclide DCF values from NUREG-1500 Table A-2

Nuclide chain values computed from individual nuclide values, assuming equilibrium

DCF=total of individual nuclide DCFs divided by number of nuclides in chain (down to stable isotope)

Calculated values agree with chain values tabulated in NUREG-1500 Table A-2

DCF is total dose per unit activity in the entire chain

Table C-4
Surface Release Limits--Building Occupancy Scenario
dpm/100cm^2

Nuclide	FC 83-23	NUREG-1500 (Building occupancy scenario--scaled to 25 mrem/y)	C-T (Building occupancy scenario--25 mrem/y)
U-238	5000 (625)[1]	2250 (160)[3]	2170 (150)[3]
Th-230	100	620	620
Ra-226	100	2700 (300)[3]	2500 (280)[3]
Th-232	1000 (1000)[2]	950 (95)[3]	1020 (100)[3]
Ra-228	100	1200	1920
Th-228	100	4530 (650)[3]	5470 (780)[3]

[1]Listed limit applies to natural uranium and associated decay products that are alpha-emitters (8 in total decay chain). This corresponds to a limit of 625 dpm/100 cm^2 for U-238 and each of its progeny in equilibrium, as shown in parentheses.

[2]Listed limit applies to thorium-232. This corresponds to a limit of 1000 dpm/100 cm^2 for Th-232 each of its progeny, as shown in parentheses.

[3]Values listed are nuclear transformations per minute per 100 cm^2 over the entire decay chain. The number of chain transformations per parent transformation is 14 for U-238, 9 for Ra-226, 19 for Th-232, and 7 for Th-228. Corresponding individual limit for parent nuclide, assuming the presence of progeny in equilibrium, is shown in parentheses.

APPENDIX C
ATTACHMENT 1

RESRAD INPUT AND OUTPUT

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1

RESRAD-BUILD Table of Contents

Input Parameters..... 0-1
For Each Time (I) :.....
Time Specific Parameters..... I-1
Receptor-Source Dose Summary..... I-2
Dose by Pathway Detail..... I-3
Dose by Nuclide Detail..... I-4
Full Summary..... F-1

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1

RESRAD-BUILD Input Parameters

Number of Sources : 1
Number of Receptors: 1
Total Time : 2.000000E+02 days
Fraction Inside : 4.200000E-01

Receptor Information

Receptor	Room	x [m]	y [m]	z [m]	FracTime	Inhalation [m3/day]	Ingestion(Dust) [m2/hr]
1	1	3.000	3.000	1.000	1.000	3.00E+01	1.00E-04

Receptor-Source Relationship

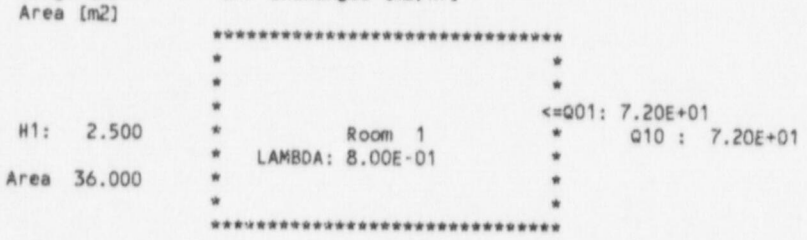
Receptor	Source 1	
	Density [g/cm3]	Thickness [cm]
1	2.400E+00	0.000E+00

!!!!!!!! Building Information !!!!!!!!

Building Air Exchange Rate: 8.00E-01 1/hr

Height [m]
Area [m2]

Air Exchanges [m3/hr]



Deposition velocity: 1.00E-02 [m/s] Resuspension Rate: 5.00E-07 [1/s]

!!!!!!!! Source Information !!!!!!!!

Source: 1
Location:: Room : 1 x: 3.00 y: 3.00 z: 0.00[m]
Geometry:: Type: Area Area:3.60E+01 [m2] Direction: z
Pathway ::
Direct Ingestion Rate: 3.000E-06 [1/hr]
Fraction released to air: 5.000E-01
Removable fraction: 1.000E+00
Time to Remove: 1.095E+04 [day]

Radon Release Fraction: 0.000E+00

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/m2]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
U-238	4.505E+01	2.500E-04	1.200E-01	3.530E-06	9.510E-08	1.600E-04
U-235	4.505E+01	2.500E-04	1.200E-01	1.950E-05	4.740E-07	9.030E-04
U-234	4.505E+01	2.600E-04	1.300E-01	8.750E-08	2.520E-10	8.930E-07
PA-231	0.000E+00	1.100E-02	1.300E+00	4.760E-06	1.190E-07	2.010E-04
TH-232	4.505E+01	2.800E-03	1.600E+00	6.450E-08	3.260E-10	1.020E-06
TH-230	4.505E+01	5.300E-04	3.200E-01	8.780E-08	7.570E-10	2.040E-06
TH-228	4.505E+01	7.500E-04	3.100E-01	1.650E-04	6.390E-06	9.410E-03
AC-227	0.000E+00	1.500E-02	6.700E+00	4.530E-05	1.260E-06	2.160E-03
RA-228	4.505E+01	1.200E-03	4.500E-03	1.090E-04	3.740E-06	5.590E-03
RA-226	4.505E+01	1.100E-03	7.900E-03	1.940E-04	7.000E-06	1.040E-02
PB-210	4.505E+01	6.700E-03	2.100E-02	4.140E-07	3.820E-09	1.430E-05

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 0.000000 years

!!!

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 0.000000 years

RESRAD-BUILD Dose Tables

Receptor Point-Source Doses
(mrem)

		Source 1	Total
Receptor	1	4.1E-01	4.1E-01
Total		4.1E-01	4.1E-01

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 0.000000 years

Pathway Detail of Doses
[mrem]

Source: 1

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	1.37E-03	2.28E-05	2.28E-07	2.75E-01	1.55E-09	1.38E-01
Total	1.37E-03	2.28E-05	2.28E-07	2.75E-01	1.55E-09	1.38E-01

Nuclide Detail of Doses
(mrem)

Source: 1

Nuclide	Receptor 1	Total
U-238		
U-238	1.55E-02	1.55E-02
U-235		
U-235	1.55E-02	1.55E-02
U-234		
U-234	1.66E-02	1.66E-02
TH-232		
TH-232	2.01E-01	2.01E-01
TH-230		
TH-230	3.98E-02	3.98E-02
TH-228		
TH-228	3.17E-02	3.17E-02
RA-228		
RA-228	1.27E-02	1.27E-02
RA-226		
RA-226	1.24E-02	1.24E-02
PB-210		
PB-210	6.90E-02	6.90E-02

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 1.00000 years

!!!

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 1.00000 years

RESRAD-BUILD Dose Tables

Receptor Point-Source Doses
[mrem]

	Source 1	Total
Receptor 1	4.1E-01	4.1E-01
Total	4.1E-01	4.1E-01

Pathway Detail of Doses
[mrem]

Source: 1

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	1.33E-03	2.28E-05	2.28E-07	2.75E-01	1.55E-09	1.34E-01
Total	1.33E-03	2.28E-05	2.28E-07	2.75E-01	1.55E-09	1.34E-01

Nuclide Detail of Doses
[mrem]

Source: 1

Nuclide	Receptor 1	Total
U-238		
U-238	1.54E-02	1.54E-02
U-234	4.68E-08	4.68E-08
TH-230	5.06E-13	5.06E-13
RA-226	1.96E-17	1.96E-17
PB-210	2.61E-19	2.61E-19
U-235		
U-235	1.54E-02	1.54E-02
PA-231	5.21E-06	5.21E-06
AC-227	2.82E-07	2.82E-07
U-234		
U-234	1.66E-02	1.66E-02
TH-230	3.57E-07	3.57E-07
RA-226	2.34E-11	2.34E-11
PB-210	1.34E-12	1.34E-12
TH-232		
TH-232	2.00E-01	2.00E-01
TH-228	5.87E-04	5.87E-04
RA-228	1.39E-03	1.39E-03
TH-230		
TH-230	3.97E-02	3.97E-02
RA-226	5.20E-06	5.20E-06
PB-210	4.46E-07	4.46E-07
TH-228		
TH-228	2.19E-02	2.19E-02
TH-228		
TH-228	8.99E-03	8.99E-03
RA-228	1.09E-02	1.09E-02
RA-226		
RA-226	1.20E-02	1.20E-02
PB-210	2.05E-03	2.05E-03
PB-210		
PB-210	6.48E-02	6.48E-02

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 10.0000 years

Assessment for Time: 3
Time =1.00E+01 yr

Source Information

Source: 1
Location:: Room : 1 x: 3.00 y: 3.00 z: 0.00 [m]
Geometry:: Type: Area Area:3.60E+01 [m2] Direction: z
Pathway ::
Direct Ingestion Rate: 3.000E-06 [1/hr]
Fraction released to air: 5.000E-01
Removable fraction: 1.000E+00
Time to Remove: 1.095E+04 [day]

Contamination::

Nuclide	Concentration [pCi/m2]
U-238	3.002E+01
U-235	3.002E+01
U-234	3.002E+01
PA-231	6.347E-03
TH-232	3.002E+01
TH-230	3.002E+01
TH-228	3.002E+01
AC-227	9.144E-04
RA-228	3.002E+01
RA-226	3.002E+01
PB-210	3.002E+01

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 10.0000 years

RESRAD-BUILD Dose Tables

Receptor Point-Source Doses

[mrem]

	Source 1	Total
Receptor 1	3.7E-01	3.7E-01
Total	3.7E-01	3.7E-01

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 10.0000 years

Pathway Detail of Doses
[mrem]

Source: 1

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	9.14E-04	2.28E-05	2.28E-07	2.75E-01	1.55E-09	9.28E-02
Total	9.14E-04	2.28E-05	2.28E-07	2.75E-01	1.55E-09	9.28E-02

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 10.0000 years

Nuclide Detail of Doses
[mrem]

Source: 1

Nuclide	Receptor 1	Total
U-238		
U-238	1.46E-02	1.46E-02
U-234	4.47E-07	4.47E-07
TH-230	4.86E-11	4.86E-11
RA-226	1.58E-14	1.58E-14
PB-210	6.34E-15	6.34E-15
U-235		
U-235	1.47E-02	1.47E-02
PA-231	4.53E-05	4.53E-05
AC-227	2.43E-05	2.43E-05
U-234		
U-234	1.58E-02	1.58E-02
TH-230	3.43E-06	3.43E-06
RA-226	1.68E-09	1.68E-09
PB-210	8.83E-10	8.83E-10
TH-232		
TH-232	1.92E-01	1.92E-01
TH-228	1.64E-02	1.64E-02
RA-228	6.06E-03	6.06E-03
TH-230		
TH-230	3.81E-02	3.81E-02
RA-226	3.72E-05	3.72E-05
PB-210	2.87E-05	2.87E-05
TH-228		
TH-228	7.71E-04	7.71E-04
TH-228		
TH-228	1.19E-02	1.19E-02
RA-228	2.60E-03	2.60E-03
RA-226		
RA-226	8.57E-03	8.57E-03
PB-210	1.26E-02	1.26E-02
PB-210		
PB-210	3.45E-02	3.45E-02

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 29.0000 years

!!!

RESRAD-BUILD Dose Tables

Receptor Point-Source Doses
(mrem)

	Source 1	Total
Receptor 1	2.8E-01	2.8E-01
Total	2.8E-01	2.8E-01

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 29.0000 years

Pathway Detail of Doses
[mrem]

Source: 1

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	4.48E-05	2.28E-05	2.28E-07	2.75E-01	1.55E-09	6.72E-03
Total	4.48E-05	2.28E-05	2.28E-07	2.75E-01	1.55E-09	6.72E-03

Title : MI NUREG/CR-5512 BLDG OCCUPANCY
Input File : MI1 Evaluation Time: 29.0000 years

Nuclide Detail of Doses
[mrem]

Source: 1

Nuclide	Receptor 1	Total
U-238		
U-238	1.31E-02	1.31E-02
U-234	1.16E-06	1.16E-06
TH-230	3.73E-10	3.73E-10
RA-226	6.37E-14	6.37E-14
PB-210	4.65E-14	4.65E-14
U-235		
U-235	1.31E-02	1.31E-02
PA-231	8.94E-05	8.94E-05
AC-227	1.51E-04	1.51E-04
U-234		
U-234	1.42E-02	1.42E-02
TH-230	9.09E-06	9.09E-06
RA-226	2.33E-09	2.33E-09
PB-210	2.17E-09	2.17E-09
TH-232		
TH-232	1.74E-01	1.74E-01
TH-228	2.31E-02	2.31E-02
RA-228	9.78E-04	9.78E-04
TH-230		
TH-230	3.48E-02	3.48E-02
RA-226	1.78E-05	1.78E-05
PB-210	2.33E-05	2.33E-05
TH-228		
TH-228	6.46E-07	6.46E-07
TH-228		
TH-228	1.10E-03	1.10E-03
RA-228	3.07E-05	3.07E-05
RA-226		
RA-226	1.41E-03	1.41E-03
PB-210	3.22E-03	3.22E-03
PB-210		
PB-210	2.21E-03	2.21E-03

RESRAD-BUILD Dose (Time) Tables

Receptor Doses By Time

[mrem]

	Time [yr]			
	0.00E+00	1.00E+00	1.00E+01	2.90E+01
1	4.14E-01	4.09E-01	3.68E-01	2.82E-01

Receptor Doses Per Year

[mrem/yr]

	Time [yr]			
	0.00E+00	1.00E+00	1.00E+01	2.90E+01
1	7.56E-01	7.48E-01	6.73E-01	5.14E-01

APPENDIX D
DERIVATION OF SURFACE BETA RADIOACTIVITY LIMIT

1.0 INTRODUCTION

The application of the release criteria to building surfaces and equipment at the C-T project site requires an innovative approach. The following discussion enumerates the complicating factors present at the site and the approach which has been utilized to accommodate these difficulties. The methodology developed is a technique to measure beta disintegrations per minute as a means of demonstrating compliance with the release criteria.

1.1 RADIOLOGICAL CONSTITUENTS OF INTEREST

The radiological constituents of interest present on the C-T project site are:

- U-238, U-235, and U-234, all from naturally occurring uranium (and their progeny Th-230, Ra-226, and other short-lived isotopes);
- Th-232 from naturally occurring thorium, Th-228, and their progeny (Ra-228 and other short-lived isotopes); and,
- K-40, an unregulated isotope in naturally occurring potassium, of interest only because of its potential to interfere with the measurement of uranium and thorium series nuclides.

1.2 COMPLICATING FACTORS

Complicating factors associated with the radionuclides of interest are:

- all radionuclides of interest are present naturally in most materials;
- all radionuclides of interest were also processed in nearby areas within the St. Louis Plant during MED-AEC operations;
- the uranium and thorium progeny are not always present in equilibrium due the use of a variety of chemical processing and extraction methods;
- background levels of some radionuclides are near the limits for release for unrestricted use;

- distinguishing licensed material from background material in the range of unrestricted release limits is problematic; and

- the presence of K-40 due to the production and use of potassium salts on site complicates background radiation measurements.

1.3 APPROACH TO MONITORING

An ideal monitoring technique would accommodate these complicating factors. The challenge therefore is to accommodate the difficulties presented by the mix of radionuclides on site as well as translate the residual radioactivity limits to beta limits. Beta monitoring is the best option for the following reasons.

- Direct measurement of alpha particles is unreliable due to self adsorption; contamination of painted or other coated areas would not be detected.

- Direct measurement of beta/gamma activity would not provide results representative of actual contamination; contribution of "shine' from subsurface and other areas containing natural gamma emitters (e.g., brick and concrete).

- Beta activity measurements can achieve lower MDAs as beta background is approximately 1/3 that of background beta/gamma activity.

- NRC required that beta activity be measured in other licensee characterization programs that preceded the C-T characterization plan submission.

The following describes the methodology used to derive the average beta radiation limit accounting for the specific mixtures of nuclides on site, non-equilibrium conditions, and contributions from non-regulated K-40 materials present at the Mallinckrodt C-T project site.

Equipment release criteria is based on Policy and Guidance Directive FC 83-23. Building surface isotopic release criteria are based on the 25 mrem/yr dose limit for unrestricted industrial use and are derived using the computer code RESRAD. Further information on this derivation may be found in appendix C. While the regulatory basis for the release criteria for equipment differs from building surfaces, the same methodology will apply to the derivation of beta limits for each. The following tables show the basis for equipment and building surface release criteria.

Table D-1 Equipment Isotopic Surface Release Limits			
Radionuclide	Average dpm/100 cm^2	Maximum dpm/100 cm^2	Removable dpm/100 cm^2
U-238	5000 α	15000 α	1000 α
Th-230	100	300	20
Ra-226	100	300	20
Th-232	1000	3000	200
Ra-228	100	300	20
Th-228	100	300	20

Table D-2 Isotopic Building Surface Release Limits	
Radionuclide	Average dpm/100 cm^2
U-238	2170
Th-230	620
Ra-226	2500
Th-232	1020
Ra-228	1920
Th-228	5470

2.0 METHODOLOGY

This section describes the methodology used to derive beta limits from the above guidelines. The equipment surface release limit is the average surface limit. The maximum and removable equipment limits are determined by multiplying or dividing by the appropriate factor. The equipment maximum limit is three times the average limit and the removable limit is 1/5 the average limit following the convention of FC 83-23. The building surface limit derived is the average surface release limit.

2.1 STEP 1 - DETERMINE RELATIVE CONCENTRATIONS

First, the relative fractions of U-238, Th-230, and Ra-226 in the U-238 family and the relative fractions of Th-232, Ra-228, and Th-228 in the Th-232 family were calculated. The history of each area was considered to allow for evaluation of variability. The areas were scanned and the locations of highest activity were marked. Then, a beta activity measurement was performed. The area was then scabbled and the concentration of the seven radionuclides of interest were determined using either alpha or gamma analysis.

These biased samples were then used to represent the radionuclide spectrum used to derive the surface release value for that area.

2.2 STEP 2 - DETERMINE EQUILIBRIUM STATUS

The average relative activity fractions in scabble samples were examined for both the U-238 and Th-232 families. If the relative activity fractions were similar for both the parent and progeny nuclides, the sample was treated as containing only the parent in equilibrium. However, if the progeny fractions were greater than the parent, the parent fraction was considered in equilibrium and the excess was treated separately. To account for any inaccuracy in the measurement techniques, the chain was considered out of equilibrium if the progeny fraction was greater than 1.3 times the parent.

2.3 STEP 3 - DETERMINE GROSS ACTIVITY GUIDELINE

The gross activity guideline for the surface was calculated using equation A-2 from NUREG/CR-5849.

$$Gross\ Activity\ Guideline = \frac{1}{\frac{f_1}{G_1} + \frac{f_2}{G_2} + \ldots + \frac{f_n}{G_n}} \qquad \text{[Equation D-1]}$$

where f_n is the relative fraction of the total activity contributed by each radionuclide group and G_n is the guideline value for each radionuclide group. For ease of calculation, the following equivalent of this formula was used.

$$Gross\ Activity\ Guideline = \frac{Activity_1 + Activity_2 + \ldots + Activity_n}{\frac{Activity_1}{G_1} + \frac{Activity_2}{G_2} + \ldots + \frac{Activity_n}{G_n}} \qquad \text{[Equation D-2]}$$

where G_n is the guideline value for each radionuclide group and $Activity_n$ is the activity value for each radionuclide.

2.4 STEP 4 - DETERMINE β TO DISINTEGRATION RATIO

The beta-to-disintegration (β/δ) ratio was calculated by averaging the contributions of the materials present (equation D-3) where f_n is the relative fraction of each nuclide.

$$\frac{\beta}{\delta}\ Ratio = f_1\left(\frac{\beta}{\delta}\right)_1 + f_2\left(\frac{\beta}{\delta}\right)_2 + \ldots + f_n\left(\frac{\beta}{\delta}\right)_n \qquad \text{[Equation D-3]}$$

The beta to disintegration (δ) ratio of each radionuclide was calculated from the decay chain of U-238 and Th-232 and each is shown in the following table.

Table D-3 β to δ Ratio		
Nuclide	β emitted per disintegration	β/Disintegration
U-238	5.4	.39
Th-230	0	0
Ra-226	3.3	.37
Th-232	3.6	.36
Ra-228	1.1	1.1
Th-228	2.5	.36
K-40	1	

2.5 STEP 5 - DETERMINE β LIMIT

The disintegration limit calculated in step 3 was multiplied by the beta-to-disintegration ratio calculated in step 4:

$$\beta\ Limit = (Gross\ Activity\ Guideline) * \frac{\beta}{\delta}\ Ratio\ .$$ [Equation D-4]

2.6 STEP 6 - ACCOUNT FOR K-40 CONTRIBUTION

Finally, the K-40 contribution was added to the value calculated in step 5. The percent of activity due to K-40 was determined from the measured total value at the location where the sample was taken. Equations D-5 and D-6 were used to determine the K-40 contribution.

$$^{40}K = \frac{Activity_{^{40}K} * 1}{Activity_1 * \beta_1 + Activity_2 * \beta_2 + \ldots + Activity_{^{40}K}}$$ [Equation D-5]

$$^{40}K\ Contribution = {}^{40}K * \beta\ measured$$ [Equation D-6]

Due to the complexity of the calculations, a computer spreadsheet was generated to perform calculations for each area of the site. The radionuclide spectrum are entered in the spreadsheet and the formulas are used to calculate the beta limits. The logic delineated in the six steps flows downward in the spreadsheet as follows. The average of the relative fractions is determined. A decision is made whether the fractions are in excess of equilibrium. A gross activity guideline is calculated. The beta-per-disintegration value is then used to compute the net beta emission limit. Finally, a K-40 value is added to complete the calculation.

3.0 CALCULATION EXAMPLE

Table D-4 shows the derivation of beta limits for the surface of the Building 235 East pad. This was a storage area for drums and for other materials. There is some evidence that the drums may have leaked at one time. The surface was scanned and the six points with the highest activity were scabbled for analysis. Since the concentrations varied in the samples, the relative fraction of each of the radionuclides was calculated. The relative fractions were averaged to determine the equilibrium status of the progeny. The Ra-226 fraction was approximately 250% of the U-238 and was the only one of the average ratios to be in excess of equilibrium and therefore is the only one to have "TRUE" in the excess column. The K-40 component was between 3%-9% of the total activity and therefore is taken into account in the derived beta limit. The derived beta limit is therefore 1751 beta/min/100 cm^2.

4.0 CONCLUSIONS

Utilizing this methodology, beta limits were derived for equipment based on the limits shown in Table D-1 and for each building based on the limits shown in Table D-2. The use of beta limits will accommodate survey difficulties as well as account for the difficulties encountered due to the radionuclide mix present on the site. Equipment beta limits are based on the FC 83-23 limits for the building in which they are housed. These limits are shown in Table D-5. The beta limits for each building are shown in Table D-6.

Table D-4 - Surface Beta Radioactivity Limit Derivation

Building 235 East Pad

Sample ID	Concentration (pCi/g)							Total beta observed per min per 100 cm2
	U_238	Th_230	Ra_226	Th_232	Ra_228	Th_228	K_40	
1315	17.8	51.3	32.9	7.5	9.8	6.3	11.7	52400
1317	28.4	12.4	73.6	2.2	2.4	3	13.1	18800
1318	18.1	14.6	40.7	8.1	9.9	8.1	18.1	5870
1319	33.5	17.5	46.8	4.8	4.3	5.2	16.1	10600
1320	16.9	36.6	104.4	5.6	5.6	5.9	11.7	5860
1321	26.5	21.7	65.5	9.7	10.5	12.1	14.1	10300
Blank	0.21	0.56	0	0.03	0.14	0.25		1.19

Relative Fraction of U+Th Series Total Concentration With Blank Subtraction

	U_238	Th_230	Ra_226	Th_232	Ra_228	Th_228
1315	0.141	0.408	0.264	0.060	0.078	0.049
1317	0.233	0.098	0.609	0.018	0.019	0.023
1318	0.182	0.143	0.414	0.082	0.099	0.080
1319	0.300	0.153	0.422	0.043	0.038	0.045
1320	0.096	0.207	0.601	0.032	0.031	0.033
1321	0.182	0.146	0.452	0.067	0.072	0.082
Arith. Avg.	0.189	0.192	0.460	0.050	0.056	0.052

Fraction In Excess of Radioactive Equilibrium

0.003	0.271		0.006	0.001	

Is fraction in excess by more than 30% ?

FALSE	TRUE		FALSE	FALSE

Detectable Beta per Disintegration

0.39	0	0.37	0.36	1.1	0.36

Total Activity Limit by Radionuclide (dpm/100 cm2)

2170	620	2500	1020	1920	5470

Surface activity limit for the mixture (a dpm/100 cm2) =	2085
Number of beta emissions per disintegration (b/dis)=	0.38
Net beta emission limit (b/min 100 cm2)=	785

Beta Radiation Contributed by Potassium-40

Beta per Disintegration

U_238 series	Th_230	Ra_226	Th_232 series	Ra_228	Th-228	K_40
5.4	0	3.3	3.6	1.1	2.5	1

Sample	beta per min per 100 cm2	fraction of total activity
1315	3275	0.063
1317	757	0.040
1318	484	0.082
1319	661	0.062
1320	166	0.028
1321	453	0.044
Arith. Average	966	0.053

Net beta radiation allowed from surface ((beta/(min 100cm2)) **1751**

Table D-5
Equipment Surface Release Limits per Building

Equipment Location	Average (β/min/100 cm^2)	Maximum (β/min/100 cm^2)	Removable (β/min/100 cm^2)
Building 90/91	1100	3300	220
Building 101 Incinerator Pad	1800	5400	360
Building 101 Roof	1800	5400	360
Building 200 Roof	1560	4680	312
Building 201 Roof	1560	4680	312
Building 204 Roof	400	1200	80
Building 213	500	1500	100
Building 213 Roof	660	1980	132
Building 214	500	1500	100
Building 235 Roof	120	360	24
Building 236 Roof	150	450	24
Building 238	1000	3000	200
Building 238 Roof	1800	5400	360
Building 245	1560	4680	312
Building 245 Roof	1560	4680	312
Building 246	230	690	46
Building 246 Roof	1070	3210	214
Building 247	230	690	46
Building 247 Roof	470	1410	94
Building 248	430	1290	86
Building 248 Roof	490	1470	98
Building 250	1200	3600	240

Table D-6 Surface Release Limits for Individual Buildings	
Location	Average (β/min/100 cm^2)
Building 62	1560
Building 91 Crane Service Pad	700
Building 101 Roof/Pad	730
Building 200 Roof	600
Building 201 Roof	600
Building 204 Roof	600
Building 222 Roof	920
Building 223 Roof	1510
Building 235	910
Building 235 Roof	910
Building 236	600
Building 236 Roof	600
Building 240	1100
Building 240 Roof	1100
Building 245	1230
Building 245 Roof	1010
Building 250	830
Building 250 Roof	580

APPENDIX E
DERIVATION OF MASS CONCENTRATION LIMITS FOR UNRESTRICTED RELEASE OF BUILDING RUBBLE

1.0 Introduction

Background information on the site, operations at the site, and the decommissioning in general are provided in the main body of the decommissioning plan. As explained in Section 2 of the decommissioning plan, disposition of building rubble from buildings to be dismantled will be determined based on mass concentration limits (pCi/g or equivalent). Mass concentration limits will be applied in a two-tiered system. Rubble containing radionuclide concentrations exceeding the uppermost tier will be sent to a licensed radioactive waste disposal facility. Rubble containing radionuclide concentrations below the lowermost tier will be released for unrestricted use. Rubble containing radionuclide concentrations between the two tiers will be temporarily retained on site pending the approval of the Phase II decommissioning plan which will address this type of material. The extent of building decontamination to be conducted prior to dismantling will also be based on anticipated compliance with the same mass concentration limits. Areas will be assessed to determine the need for decontamination and the likelihood that decontamination will reduce mass concentrations in rubble to levels at least below the upper tier limits. Expected mass concentrations can be estimated based on measured surface concentrations and surface density (mass per unit area of surface). If decontamination is needed to meet mass concentration limits and is judged likely be successful, the area will be decontaminated. If decontamination is not likely to be successful, decontamination will be conducted only to the extent necessary for health and safety controls, and the rubble will be isolated from other building rubble for disposal at a licensed radioactive waste disposal facility. The purpose of this appendix is to describe the basis for the lowermost tier of mass concentration limits—that set of limits defining the boundary between release of building rubble for unrestricted use and disposition at regulated waste facilities.

The US Nuclear Regulatory Commission recently adopted new radiological criteria for decommissioning (FR, 1997). The new criteria are expressed in terms of radiation dose that might reasonably be expected from residual radioactive material after decommissioning. At any particular site at which decommissioning is planned, this dose would depend upon concentrations of residual radioactive materials in soils, other remaining bulk materials, and on remaining surfaces. The dose would also depend on site-specific factors that might control potential resource use, potential migration of radioactive materials, and potential access to radioactive materials. Finally, this dose would also depend on potential activities of future users of the site. Because the potential radiation dose from residual radioactive material is not a quantity that can be measured directly, application of these dose-based criteria presupposes application of measurable site-specific concentration criteria. These concentration limits can be derived from the

regulatory dose limits in a way that reasonably incorporates site-specific features and potential activities of future users of the site.

Prior to promulgation of the new radiological criteria for decommissioning, NRC applied radionuclide surface concentration limits (dpm/100 cm^2 or equivalent) for decommissioning management of structures, even when the structures were to be dismantled as part of the decommissioning process (USNRC, 1983). Application of derived surface concentration limits continues to be a reasonable approach for decommissioning of structures that will remain in use after decommissioning because resulting doses will be related most directly to surface concentration. However, no building surfaces would remain after dismantling. Doses from residual radioactive material in building rubble would be most directly related to bulk or mass concentration. Because application of surface concentration limits to structures to be dismantled is inconsistent with the rationale underlying the new decommissioning criteria, mass concentration limits were derived for application in this plan. Because the building rubble can be best considered as a soil-like material radiological assessment methods commonly used for deriving concentration limits for radionuclides in soils were used in this derivation.

Derivation of dose-based concentration limits is a simple extension of the conventional radiological dose assessment process. In that process, known or expected radionuclide concentrations in a primary medium, such as soil, are used in conjunction with environmental transport modeling to calculate concentrations in various environmental media and the associated radiation dose. However, if other factors are held constant, radiation dose is directly proportional to radionuclide concentration in the primary medium, which permits computation of concentration limits from dose limits.

The radiation dose assessment process includes the estimation of the maximum radiation dose that might be received by a typical member of a small group of people that could be expected to receive the highest doses from use of the site as far as 1,000 years into the future (FR, 1997). Thus, the assessment considers not only the expected conditions at the site, soon after remediation, but conditions projected for the more distant future, as well. The assessment evaluates potential uses of the site and potential migration of radioactive materials through the environment over time, taking account of both natural processes and human activities that could be expected to alter the patterns or rates of constituent movement.

In general, the dose assessment process consists of two steps: 1) development of representations of site physical conditions and potentially exposed populations, and expression of these representations in mathematical terms; and 2) use of a mathematical model with input from the representations and/or technical literature to estimate future exposures and radiation doses as a function of time. The dual objective in the development of simplified representations is that the representations be realistic and not result in underestimation of exposures and doses. The following sections describe the

representations of the radioactive material source, the site environs, and potential exposure scenarios (step 1), and the dose assessment methodology and results (step 2).

2.0 Radioactive Material Source

Potential radiation doses from residual radioactive materials in building rubble remaining on the site after decommissioning will depend upon the types and inventories of radionuclides present, the concentrations of nuclides in wastes and soils, and the physical distribution of the wastes and soils on the site. This section describes the development of the site source representation for dose assessment purposes. This development consists of the definition, in spatial terms, of source units within which nuclide distributions and/or physical characteristics can be considered reasonably uniform and at the same time reasonably representative of site conditions. Such a representation of the site is necessary to permit description of the site conditions in a mathematical form.

The radionuclides of interest for evaluation are identified in the main body of the decommissioning plan. They are naturally occurring uranium (U-238, U-234, and 235) and naturally occurring thorium (Th-232 and Th-228) and their radioactive progeny.

For purposes of evaluation, it is assumed that the concentration of radionuclides in building rubble is uniform and that building rubble occupies an area of 10,000 square meters to a depth of 2 meters below the surface. No cover is assumed to be present over this material. These numerical values are default values for exposure scenarios developed for screening radiological assessment in Policy and Guidance Directive PG-8-08 (USNRC, 1994). However, they are sufficiently large that assuming more extensive distributions would not result in higher dose estimates. For this reason, results based on this analysis can be considered in a conservative way to be independent of the quantity of building rubble generated. Concentration limits would not be lower for much larger quantities. Concentration limits could be higher for much smaller quantities.

The key output from the analysis is the ratio of radiation dose to nuclide concentration in the primary medium, building rubble. For this reason, the absolute concentration of each nuclide assumed present in the building rubble is arbitrary, and was chosen to be 5 pCi/g for purposes of this analysis.

3.0 Potential Exposure Scenarios

People could conceivably receive radiation doses from radioactive materials on the site through either internal or external exposure. In external exposure, the body absorbs radiation emitted by radioactive material outside the body. For example, radioactive materials deposited on the ground surface can cause external exposure of a person standing on the surface if the radiation emitted is sufficiently penetrating to reach internal body tissues. In internal exposure, the body absorbs radiation emitted by radioactive material that has been introduced into the body by inhalation of radioactive material

constituents in air, or by ingestion of radioactive material constituents in food or water. In general, the level of the radiation dose received decreases as the quantity of radioactive material available for exposure decreases, but the exact relationships are sometimes complex. External radiation exposure can be mitigated by dilution of the source material, by increasing the distance between the receptor and the source, by limiting exposure time, and by shielding, i.e., the placement of radiation-absorbing material between the receptor and the source. Internal exposure can be best mitigated by isolation or dilution of the source material to minimize inhalation or ingestion of radioactive material.

People could conceivably be exposed to radioactive materials from unrestricted use of the site in a number of ways, depending on potential uses of land and groundwater resources. The site and the area in which it is located have long been used for industrial purposes, and most likely will continue to be so used for the foreseeable future. However, over the 1,000-year period of interest, use of the site for residential and agricultural purposes would be the feasible uses most likely to maximize radiation exposure. Use of groundwater for domestic and agricultural purposes or for irrigation is considered not feasible, because of poor quality (STEP, 1997).

Given potential land and water uses, exposure to residual radioactive material in building rubble on the site could possibly occur through one or more of three terrestrial pathways (water independent pathways). Exposure could occur from radiation emitted directly from radioactive material in place in the rubble, from inhalation of dusts containing residual radioactive material (resuspended from rubble at the ground surface), or from ingestion of residual radioactive material taken up from soil-like rubble into garden products produced on the site. Because radon-222 and radon-220 are progeny of the uranium-238 and thorium-232 decay chains, respectively, inhalation of radon daughter products is a possible radiation exposure pathway at this site. However, because radiation doses from these nuclides are best controlled by measures commonly incorporated in new structures, radiation doses from these nuclides have been excluded from the new radiological criteria for decommissioning, and are not included in this assessment.

A comprehensive set of possible general exposure pathways for consideration in dose assessment was developed based on the nuclides of interest, the anticipated distribution of the nuclides on the site, potential uses of the site, and potential environmental migration pathways. This compilation is listed below:

1. Ground

    - Direct radiation from material in soil

2. Dust

    - Resuspension of surface particulate material--air--inhalation

3. Plant (water independent)

   - Resuspension of surface particulate material--air--deposition or uptake in edible plant tissue--ingestion
   - Uptake of material in soil through roots of edible plants--ingestion

4. Meat or Milk (water independent)

   - Resuspension of surface particulate material--air--deposition or uptake in edible plant tissue--ingestion by animal--ingestion of animal product
   - Uptake of material in soil through roots of edible plants--ingestion by animal--ingestion of animal product

5. Soil

   - Ingestion of soil

A suitable subset of this list can be taken as a complete list of exposure pathways for any particular exposure scenario. A hypothetical scenario that incorporates all of the pathways listed above would be a resident farmer. Because this scenario incorporates all of the potential exposure pathways, it is most likely to result in the highest dose, and, for that reason, is selected as the scenario that represents maximum potential impact from use of the site. More realistic exposure scenarios would exclude some potential pathways and would result in lower calculated doses. For example, agricultural use of the site is considered highly unlikely.

4.0 Dose Assessment Methodology

Potential future radiation doses are computed from estimates of potential intake rates and exposure rates. Radiation doses from internal exposure (inhalation or ingestion) are computed using dose factors developed using current methodology, specifically those in Federal Guidance Report 11 (USEPA, 1988). In conformance with this guidance, the term "dose," as it is used in this report, means "committed effective dose equivalent" (CEDE) in reference to doses from internal exposure, "deep dose" in reference to external exposure, and "total effective dose equivalent" (TEDE) in reference to combined internal and external exposure. Doses totaled over all applicable exposure pathways are computed as a function of time after license termination (over a period up to 1,000 years) for each potentially important receptor. The peak value for each receptor is retained.

The computation of nuclide concentrations in media and radiation doses associated with exposure to those media is complex, and is usually performed using computer codes designed for the purpose. The RESRAD code (Gilbert, 1989 and Yu, 1993) was selected

as suitable for evaluation of all pathways in the analysis. The version used was Version 5.62, the latest available at the time the analysis was performed.

For purposes of analysis, the rubble is conservatively assumed to be soil-like. Exposure parameter values for the resident farmer exposure scenario selected for evaluation were drawn from Scenario C, the resident farmer scenario, in NRC Policy and Guidance Directive PG-8-08 (USNRC, 1994). However, for reasons described above, the water-dependent and radon pathways included in the PG-8-08 screening scenario have been excluded from this analysis. Default RESRAD parameter values were used where recommended values are not specified in PG-8-08. Complete lists of parameter values are provided in PG-8-08 (USNRC, 1994) and in the RESRAD documentation (Yu, 1993) and are only summarized here. The resident farmer is assumed to spend 55% of the time indoors in a structure built on the building rubble, 21% of the time outdoors on the rubble, and 24% of his time away from the site. Only modest dose reduction factors are applied for indoor exposure through the ground (0.67) and inhalation (0.5) pathways. Half of the grain, vegetables, and fruit in the farmer's diet is assumed to be grown in a garden assumed present in the rubble. All of the farmer's meat and milk is assumed to be produced from animals living on the rubble and consuming feed produced on the rubble. Dust levels in the outdoor air in the vicinity of the garden are representative for earth moving areas because of tilling, planting, harvesting, and other activities that may increase suspension of soil particles in the air relative to locations that are more residential in character. Complete listings of the input data file for the RESRAD run are provided in Attachment 1.

5.0 Results

Detailed results of the RESRAD analysis are tabulated in Attachment 1. Concentration limits for each principal nuclide were computed by scaling the 5 pCi/g input concentration by the ratio of 25 mrem/y divided by the peak calculated dose for the principal nuclide. These results are summarized in Table E.1.

Table E.1
Rubble Concentration Limits for Unrestricted Release

Radionuclide	Concentration Limit (pCi/g)
Total U	100
Th-230	7.2
Ra-226	1.7
Th-232	1.5

6.0 References

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APPENDIX E
ATTACHMENT 1

RESRAD INPUT AND OUTPUT

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Dose Conversion Factor (and Related) Parameter Summary
File: DOSFAC.BIN

Menu	Parameter	Current Value	Default	Parameter Name
B-1	Dose conversion factors for inhalation, mrem/pCi:			
B-1	Ac-227+D	6.720E+00	6.720E+00	DCF2(1)
B-1	Pa-231	1.280E+00	1.280E+00	DCF2(2)
B-1	Pb-210+D	2.320E-02	2.320E-02	DCF2(3)
B-1	Ra-226+D	8.600E-03	8.600E-03	DCF2(4)
B-1	Ra-228+D	5.080E-03	5.080E-03	DCF2(5)
B-1	Th-228+D	3.450E-01	3.450E-01	DCF2(6)
B-1	Th-230	3.260E-01	3.260E-01	DCF2(7)
B-1	Th-232	1.640E+00	1.640E+00	DCF2(8)
B-1	U-234	1.320E-01	1.320E-01	DCF2(9)
B-1	U-235+D	1.230E-01	1.230E-01	DCF2(10)
B-1	U-238+D	1.180E-01	1.180E-01	DCF2(11)
D-1	Dose conversion factors for ingestion, mrem/pCi:			
D-1	Ac-227+D	1.480E-02	1.480E-02	DCF3(1)
D-1	Pa-231	1.060E-02	1.060E-02	DCF3(2)
D-1	Pb-210+D	7.270E-03	7.270E-03	DCF3(3)
D-1	Ra-226+D	1.330E-03	1.330E-03	DCF3(4)
D-1	Ra-228+D	1.440E-03	1.440E-03	DCF3(5)
D-1	Th-228+D	8.080E-04	8.080E-04	DCF3(6)
D-1	Th-230	5.480E-04	5.480E-04	DCF3(7)
D-1	Th-232	2.730E-03	2.730E-03	DCF3(8)
D-1	U-234	2.830E-04	2.830E-04	DCF3(9)
D-1	U-235+D	2.670E-04	2.670E-04	DCF3(10)
D-1	U-238+D	2.690E-04	2.690E-04	DCF3(11)
D-34	Food transfer factors:			
D-34	Ac-227+D , plant/soil concentration ratio, dimensionless	2.500E-03	2.500E-03	RTF(1,1)
D-34	Ac-227+D , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	2.000E-05	2.000E-05	RTF(1,2)
D-34	Ac-227+D , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	2.000E-05	2.000E-05	RTF(1,3)
D-34	Pa-231 , plant/soil concentration ratio, dimensionless	1.000E-02	1.000E-02	RTF(2,1)
D-34	Pa-231 , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	5.000E-03	5.000E-03	RTF(2,2)
D-34	Pa-231 , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	5.000E-06	5.000E-06	RTF(2,3)
D-34	Pb-210+D , plant/soil concentration ratio, dimensionless	1.000E-02	1.000E-02	RTF(3,1)
D-34	Pb-210+D , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	8.000E-04	8.000E-04	RTF(3,2)
D-34	Pb-210+D , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	3.000E-04	3.000E-04	RTF(3,3)
D-34	Ra-226+D , plant/soil concentration ratio, dimensionless	4.000E-02	4.000E-02	RTF(4,1)
D-34	Ra-226+D , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	1.000E-03	1.000E-03	RTF(4,2)
D-34	Ra-226+D , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	1.000E-03	1.000E-03	RTF(4,3)
D-34	Ra-228+D , plant/soil concentration ratio, dimensionless	4.000E-02	4.000E-02	RTF(5,1)
D-34	Ra-228+D , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	1.000E-03	1.000E-03	RTF(5,2)
D-34	Ra-228+D , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	1.000E-03	1.000E-03	RTF(5,3)
D-34	Th-228+D , plant/soil concentration ratio, dimensionless	1.000E-03	1.000E-03	RTF(6,1)
D-34	Th-228+D , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	1.000E-04	1.000E-04	RTF(6,2)
D-34	Th-228+D , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	5.000E-06	5.000E-06	RTF(6,3)

Dose Conversion Factor (and Related) Parameter Summary (continued)
File: DOSFAC.BIN

Menu	Parameter	Current Value	Default	Parameter Name
D-34	Th-230 , plant/soil concentration ratio, dimensionless	1.000E-03	1.000E-03	RTF(7,1)
D-34	Th-230 , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	1.000E-04	1.000E-04	RTF(7,2)
D-34	Th-230 , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	5.000E-06	5.000E-06	RTF(7,3)
D-34				
D-34	Th-232 , plant/soil concentration ratio, dimensionless	1.000E-03	1.000E-03	RTF(8,1)
D-34	Th-232 , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	1.000E-04	1.000E-04	RTF(8,2)
D-34	Th-232 , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	5.000E-06	5.000E-06	RTF(8,3)
D-34				
D-34	U-234 , plant/soil concentration ratio, dimensionless	2.500E-03	2.500E-03	RTF(9,1)
D-34	U-234 , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	3.400E-04	3.400E-04	RTF(9,2)
D-34	U-234 , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	6.000E-04	6.000E-04	RTF(9,3)
D-34				
D-34	U-235+D , plant/soil concentration ratio, dimensionless	2.500E-03	2.500E-03	RTF(10,1)
D-34	U-235+D , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	3.400E-04	3.400E-04	RTF(10,2)
D-34	U-235+D , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	6.000E-04	6.000E-04	RTF(10,3)
D-34				
D-34	U-238+D , plant/soil concentration ratio, dimensionless	2.500E-03	2.500E-03	RTF(11,1)
D-34	U-238+D , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	3.400E-04	3.400E-04	RTF(11,2)
D-34	U-238+D , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	6.000E-04	6.000E-04	RTF(11,3)
D-5	Bioaccumulation factors, fresh water, L/kg:			
D-5	Ac-227+D , fish	1.500E+01	1.500E+01	BIOFAC(1,1)
D-5	Ac-227+D , crustacea and mollusks	1.000E+03	1.000E+03	BIOFAC(1,2)
D-5				
D-5	Pa-231 , fish	1.000E+01	1.000E+01	BIOFAC(2,1)
D-5	Pa-231 , crustacea and mollusks	1.100E+02	1.100E+02	BIOFAC(2,2)
D-5				
D-5	Pb-210+D , fish	3.000E+02	3.000E+02	BIOFAC(3,1)
D-5	Pb-210+D , crustacea and mollusks	1.000E+02	1.000E+02	BIOFAC(3,2)
	Ra-226+D , fish	5.000E+01	5.000E+01	BIOFAC(4,1)
D-5	Ra-226+D , crustacea and mollusks	2.500E+02	2.500E+02	BIOFAC(4,2)
D-5				
D-5	Ra-228+D , fish	5.000E+01	5.000E+01	BIOFAC(5,1)
D-5	Ra-228+D , crustacea and mollusks	2.500E+02	2.500E+02	BIOFAC(5,2)
D-5				
D-5	Th-228+D , fish	1.000E+02	1.000E+02	BIOFAC(6,1)
D-5	Th-228+D , crustacea and mollusks	5.000E+02	5.000E+02	BIOFAC(6,2)
D-5				
D-5	Th-230 , fish	1.000E+02	1.000E+02	BIOFAC(7,1)
D-5	Th-230 , crustacea and mollusks	5.000E+02	5.000E+02	BIOFAC(7,2)
D-5				
D-5	Th-232 , fish	1.000E+02	1.000E+02	BIOFAC(8,1)
D-5	Th-232 , crustacea and mollusks	5.000E+02	5.000E+02	BIOFAC(8,2)
D-5				
D-5	U-234 , fish	1.000E+01	1.000E+01	BIOFAC(9,1)
D-5	U-234 , crustacea and mollusks	6.000E+01	6.000E+01	BIOFAC(9,2)
D-5				
D-5	U-235+D , fish	1.000E+01	1.000E+01	BIOFAC(10,1)
D-5	U-235+D , crustacea and mollusks	6.000E+01	6.000E+01	BIOFAC(10,2)
D-5				

Dose Conversion Factor (and Related) Parameter Summary (continued)
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Menu	Parameter	Current Value	Default	Parameter Name
D-5	U-238+D , fish	1.000E+01	1.000E+01	BIOFAC(11,1)
D-5	U-238+D , crustacea and mollusks	6.000E+01	6.000E+01	BIOFAC(11,2)

Site-Specific Parameter Summary

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R011	Area of contaminated zone (m**2)	1.000E+04	1.000E+04	---	AREA
R011	Thickness of contaminated zone (m)	2.000E+00	2.000E+00	---	THICK0
R011	Length parallel to aquifer flow (m)	1.000E+02	1.000E+02	---	LCZPAQ
R011	Basic radiation dose limit (mrem/yr)	2.500E+01	3.000E+01	---	BRDL
R011	Time since placement of material (yr)	0.000E+00	0.000E+00	---	TI
R011	Times for calculations (yr)	1.000E+00	1.000E+00	---	T(2)
R011	Times for calculations (yr)	3.000E+00	3.000E+00	---	T(3)
R011	Times for calculations (yr)	1.000E+01	1.000E+01	---	T(4)
R011	Times for calculations (yr)	3.000E+01	3.000E+01	---	T(5)
R011	Times for calculations (yr)	1.000E+02	1.000E+02	---	T(6)
R011	Times for calculations (yr)	3.000E+02	3.000E+02	---	T(7)
R011	Times for calculations (yr)	5.000E+02	1.000E+03	---	T(8)
R011	Times for calculations (yr)	7.000E+02	0.000E+00	---	T(9)
R011	Times for calculations (yr)	1.000E+03	0.000E+00	---	T(10)
R012	Initial principal radionuclide (pCi/g): Ra-226	5.000E+00	0.000E+00	---	S1(4)
R012	Initial principal radionuclide (pCi/g): Th-230	5.000E+00	0.000E+00	---	S1(7)
R012	Initial principal radionuclide (pCi/g): Th-232	5.000E+00	0.000E+00	---	S1(8)
R012	Initial principal radionuclide (pCi/g): U-234	5.000E+00	0.000E+00	---	S1(9)
R012	Initial principal radionuclide (pCi/g): U-235	2.300E-01	0.000E+00	---	S1(10)
R012	Initial principal radionuclide (pCi/g): U-238	5.000E+00	0.000E+00	---	S1(11)
R012	Concentration in groundwater (pCi/L): Ra-226	not used	0.000E+00	---	W1(4)
R012	Concentration in groundwater (pCi/L): Th-230	not used	0.000E+00	---	W1(7)
R012	Concentration in groundwater (pCi/L): Th-232	not used	0.000E+00	---	W1(8)
R012	Concentration in groundwater (pCi/L): U-234	not used	0.000E+00	---	W1(9)
R012	Concentration in groundwater (pCi/L): U-235	not used	0.000E+00	---	W1(10)
R012	Concentration in groundwater (pCi/L): U-238	not used	0.000E+00	---	W1(11)
R013	Cover depth (m)	0.000E+00	0.000E+00	---	COVER0
R013	Density of cover material (g/cm**3)	not used	1.500E+00	---	DENSCV
R013	Cover depth erosion rate (m/yr)	not used	1.000E-03	---	VCV
R013	Density of contaminated zone (g/cm**3)	1.630E+00	1.500E+00	---	DENSCZ
R013	Contaminated zone erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCZ
R013	Contaminated zone total porosity	3.000E-01	4.000E-01	---	TPCZ
R013	Contaminated zone effective porosity	2.000E-01	2.000E-01	---	EPCZ
R013	Contaminated zone hydraulic conductivity (m/yr)	1.000E+01	1.000E+01	---	HCCZ
R013	Contaminated zone b parameter	5.300E+00	5.300E+00	---	BCZ
R013	Humidity in air (g/cm**3)	not used	8.000E+00	---	HUMID
R013	Evapotranspiration coefficient	5.000E-01	5.000E-01	---	EVAPTR
R013	Precipitation (m/yr)	1.000E+00	1.000E+00	---	PRECIP
R013	Irrigation (m/yr)	0.000E+00	2.000E-01	---	RI
R013	Irrigation mode	overhead	overhead	---	IDITCH
R013	Runoff coefficient	2.000E-01	2.000E-01	---	RUNOFF
R013	Watershed area for nearby stream or pond (m**2)	1.000E+06	1.000E+06	---	WAREA
R013	Accuracy for water/soil computations	1.000E-03	1.000E-03	---	EPS
R014	Density of saturated zone (g/cm**3)	1.630E+00	1.500E+00	---	DENSAQ
R014	Saturated zone total porosity	3.000E-01	4.000E-01	---	TPSZ
R014	Saturated zone effective porosity	2.000E-01	2.000E-01	---	EPSZ
R014	Saturated zone hydraulic conductivity (m/yr)	1.000E+02	1.000E+02	---	HCSZ
R014	Saturated zone hydraulic gradient	2.000E-02	2.000E-02	---	HGWT

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R014	Saturated zone b parameter	5.300E+00	5.300E+00	---	BSZ
R014	Water table drop rate (m/yr)	0.000E+00	1.000E-03	---	VWT
R014	Well pump intake depth (m below water table)	1.000E+01	1.000E+01	---	DWIBWT
R014	Model: Nondispersion (ND) or Mass-Balance (MB)	ND	ND	---	MODEL
R014	Well pumping rate (m**3/yr)	2.500E+02	2.500E+02	---	UW
R015	Number of unsaturated zone strata	1	1	---	NS
R015	Unsat. zone 1, thickness (m)	1.000E+00	4.000E+00	---	H(1)
R015	Unsat. zone 1, soil density (g/cm**3)	1.500E+00	1.500E+00	---	DENSUZ(1)
R015	Unsat. zone 1, total porosity	3.000E-01	4.000E-01	---	TPUZ(1)
R015	Unsat. zone 1, effective porosity	2.000E-01	2.000E-01	---	EPUZ(1)
R015	Unsat. zone 1, soil-specific b parameter	5.300E+00	5.300E+00	---	BUZ(1)
R015	Unsat. zone 1, hydraulic conductivity (m/yr)	1.000E+01	1.000E+01	---	HCUZ(1)
R016	Distribution coefficients for Ra-226				
R016	Contaminated zone (cm**3/g)	7.000E+01	7.000E+01	---	DCNUCC(4)
R016	Unsaturated zone 1 (cm**3/g)	7.000E+01	7.000E+01	---	DCNUCU(4,1)
R016	Saturated zone (cm**3/g)	7.000E+01	7.000E+01	---	DCNUCS(4)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	1.749E-03	ALEACH(4)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(4)
R016	Distribution coefficients for Th-230				
R016	Contaminated zone (cm**3/g)	6.000E+04	6.000E+04	---	DCNUCC(7)
R016	Unsaturated zone 1 (cm**3/g)	6.000E+04	6.000E+04	---	DCNUCU(7,1)
R016	Saturated zone (cm**3/g)	6.000E+04	6.000E+04	---	DCNUCS(7)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.045E-06	ALEACH(7)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(7)
R016	Distribution coefficients for Th-232				
R016	Contaminated zone (cm**3/g)	6.000E+04	6.000E+04	---	DCNUCC(8)
5	Unsaturated zone 1 (cm**3/g)	6.000E+04	6.000E+04	---	DCNUCU(8,1)
5	Saturated zone (cm**3/g)	6.000E+04	6.000E+04	---	DCNUCS(8)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.045E-06	ALEACH(8)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(8)
R016	Distribution coefficients for U-234				
R016	Contaminated zone (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCC(9)
R016	Unsaturated zone 1 (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCU(9,1)
R016	Saturated zone (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCS(9)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.447E-03	ALEACH(9)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(9)
R016	Distribution coefficients for U-235				
R016	Contaminated zone (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCC(10)
R016	Unsaturated zone 1 (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCU(10,1)
R016	Saturated zone (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCS(10)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.447E-03	ALEACH(10)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(10)

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R016	Distribution coefficients for U-238				
R016	Contaminated zone (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCC(11)
R016	Unsaturated zone 1 (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCU(11,1)
R016	Saturated zone (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCS(11)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.447E-03	ALEACH(11)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(11)
R016	Distribution coefficients for daughter Ac-227				
R016	Contaminated zone (cm**3/g)	2.000E+01	2.000E+01	---	DCNUCC(1)
R016	Unsaturated zone 1 (cm**3/g)	2.000E+01	2.000E+01	---	DCNUCU(1,1)
R016	Saturated zone (cm**3/g)	2.000E+01	2.000E+01	---	DCNUCS(1)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	6.091E-03	ALEACH(1)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(1)
R016	Distribution coefficients for daughter Pa-231				
R016	Contaminated zone (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCC(2)
R016	Unsaturated zone 1 (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCU(2,1)
R016	Saturated zone (cm**3/g)	5.000E+01	5.000E+01	---	DCNUCS(2)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.447E-03	ALEACH(2)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(2)
R016	Distribution coefficients for daughter Pb-210				
R016	Contaminated zone (cm**3/g)	1.000E+02	1.000E+02	---	DCNUCC(3)
R016	Unsaturated zone 1 (cm**3/g)	1.000E+02	1.000E+02	---	DCNUCU(3,1)
R016	Saturated zone (cm**3/g)	1.000E+02	1.000E+02	---	DCNUCS(3)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	1.225E-03	ALEACH(3)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(3)
R016	Distribution coefficients for daughter Ra-228				
R016	Contaminated zone (cm**3/g)	7.000E+01	7.000E+01	---	DCNUCC(5)
5	Unsaturated zone 1 (cm**3/g)	7.000E+01	7.000E+01	---	DCNUCU(5,1)
6	Saturated zone (cm**3/g)	7.000E+01	7.000E+01	---	DCNUCS(5)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	1.749E-03	ALEACH(5)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(5)
R016	Distribution coefficients for daughter Th-228				
R016	Contaminated zone (cm**3/g)	6.000E+04	6.000E+04	---	DCNUCC(6)
R016	Unsaturated zone 1 (cm**3/g)	6.000E+04	6.000E+04	---	DCNUCU(6,1)
R016	Saturated zone (cm**3/g)	6.000E+04	6.000E+04	---	DCNUCS(6)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.045E-06	ALEACH(6)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(6)
R017	Inhalation rate (m**3/yr)	1.051E+04	8.400E+03	---	INHALR
R017	Mass loading for inhalation (g/m**3)	2.000E-04	2.000E-04	---	MLINH
R017	Dilution length for airborne dust, inhalation (m)	3.000E+00	3.000E+00	---	LM
R017	Exposure duration	3.000E+01	3.000E+01	---	ED
R017	Shielding factor, inhalation	5.000E-01	4.000E-01	---	SHF3
R017	Shielding factor, external gamma	6.700E-01	7.000E-01	---	SHF1
R017	Fraction of time spent indoors	5.500E-01	5.000E-01	---	FIND
R017	Fraction of time spent outdoors (on site)	2.100E-01	2.500E-01	---	FOTD
R017	Shape factor flag, external gamma	1.000E+00	1.000E+00	1 shows circular AREA.	FS

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R017	Radii of shape factor array (used if FS = -1):				
R017	Outer annular radius (m), ring 1:	not used	5.000E+01	---	RAD_SHAPE(1)
R017	Outer annular radius (m), ring 2:	not used	7.071E+01	---	RAD_SHAPE(2)
R017	Outer annular radius (m), ring 3:	not used	0.000E+00	---	RAD_SHAPE(3)
R017	Outer annular radius (m), ring 4:	not used	0.000E+00	---	RAD_SHAPE(4)
R017	Outer annular radius (m), ring 5:	not used	0.000E+00	---	RAD_SHAPE(5)
R017	Outer annular radius (m), ring 6:	not used	0.000E+00	---	RAD_SHAPE(6)
R017	Outer annular radius (m), ring 7:	not used	0.000E+00	---	RAD_SHAPE(7)
R017	Outer annular radius (m), ring 8:	not used	0.000E+00	---	RAD_SHAPE(8)
R017	Outer annular radius (m), ring 9:	not used	0.000E+00	---	RAD_SHAPE(9)
R017	Outer annular radius (m), ring 10:	not used	0.000E+00	---	RAD_SHAPE(10)
R017	Outer annular radius (m), ring 11:	not used	0.000E+00	---	RAD_SHAPE(11)
R017	Outer annular radius (m), ring 12:	not used	0.000E+00	---	RAD_SHAPE(12)
R017	Fractions of annular areas within AREA:				
R017	Ring 1	not used	1.000E+00	---	FRACA(1)
R017	Ring 2	not used	2.732E-01	---	FRACA(2)
R017	Ring 3	not used	0.000E+00	---	FRACA(3)
R017	Ring 4	not used	0.000E+00	---	FRACA(4)
R017	Ring 5	not used	0.000E+00	---	FRACA(5)
R017	Ring 6	not used	0.000E+00	---	FRACA(6)
R017	Ring 7	not used	0.000E+00	---	FRACA(7)
R017	Ring 8	not used	0.000E+00	---	FRACA(8)
R017	Ring 9	not used	0.000E+00	---	FRACA(9)
R017	Ring 10	not used	0.000E+00	---	FRACA(10)
R017	Ring 11	not used	0.000E+00	---	FRACA(11)
R017	Ring 12	not used	0.000E+00	---	FRACA(12)
R018	Fruits, vegetables and grain consumption (kg/yr)	1.660E+02	1.600E+02	---	DIET(1)
R018	Leafy vegetable consumption (kg/yr)	1.100E+01	1.400E+01	---	DIET(2)
9	Milk consumption (L/yr)	1.000E+02	9.200E+01	---	DIET(3)
3	Meat and poultry consumption (kg/yr)	6.300E+01	6.300E+01	---	DIET(4)
R018	Fish consumption (kg/yr)	not used	5.400E+00	---	DIET(5)
R018	Other seafood consumption (kg/yr)	not used	9.000E-01	---	DIET(6)
R018	Soil ingestion rate (g/yr)	1.825E+01	3.650E+01	---	SOIL
R018	Drinking water intake (L/yr)	not used	5.100E+02	---	DWI
R018	Contamination fraction of drinking water	not used	1.000E+00	---	FDW
R018	Contamination fraction of household water	not used	1.000E+00	---	FHHW
R018	Contamination fraction of livestock water	0.000E+00	1.000E+00	---	FLW
R018	Contamination fraction of irrigation water	0.000E+00	1.000E+00	---	FIRW
R018	Contamination fraction of aquatic food	not used	5.000E-01	---	FR9
R018	Contamination fraction of plant food	-1	-1	0.500E+00	FPLANT
R018	Contamination fraction of meat	-1	-1	0.500E+00	FMEAT
R018	Contamination fraction of milk	-1	-1	0.500E+00	FMILK
R019	Livestock fodder intake for meat (kg/day)	6.800E+01	6.800E+01	---	LFI5
R019	Livestock fodder intake for milk (kg/day)	5.500E+01	5.500E+01	---	LFI6
R019	Livestock water intake for meat (L/day)	5.000E+01	5.000E+01	---	LWI5
R019	Livestock water intake for milk (L/day)	1.600E+02	1.600E+02	---	LWI6
R019	Livestock soil intake (kg/day)	5.000E-01	5.000E-01	---	LSI
R019	Mass loading for foliar deposition (g/m**3)	1.000E-04	1.000E-04	---	MLFD

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R019	Depth of soil mixing layer (m)	1.500E-01	1.500E-01	---	DM
R019	Depth of roots (m)	9.000E-01	9.000E-01	---	DROOT
R019	Drinking water fraction from ground water	0.000E+00	1.000E+00	---	FGWDW
R019	Household water fraction from ground water	not used	1.000E+00	---	FGWHH
R019	Livestock water fraction from ground water	not used	1.000E+00	---	FGWLW
R019	Irrigation fraction from ground water	0.000E+00	1.000E+00	---	FGWIR
C14	C-12 concentration in water (g/cm**3)	not used	2.000E-05	---	C12WTR
C14	C-12 concentration in contaminated soil (g/g)	not used	3.000E-02	---	C12CZ
C14	Fraction of vegetation carbon from soil	not used	2.000E-02	---	CSOIL
C14	Fraction of vegetation carbon from air	not used	9.800E-01	---	CAIR
C14	C-14 evasion layer thickness in soil (m)	not used	3.000E-01	---	DMC
C14	C-14 evasion flux rate from soil (1/sec)	not used	7.000E-07	---	EVSN
C14	C-12 evasion flux rate from soil (1/sec)	not used	1.000E-10	---	REVSN
C14	Fraction of grain in beef cattle feed	not used	8.000E-01	---	AVFG4
C14	Fraction of grain in milk cow feed	not used	2.000E-01	---	AVFG5
STOR	Storage times of contaminated foodstuffs (days):				
STOR	Fruits, non-leafy vegetables, and grain	1.400E+01	1.400E+01	---	STOR_T(1)
STOR	Leafy vegetables	1.000E+00	1.000E+00	---	STOR_T(2)
STOR	Milk	1.000E+00	1.000E+00	---	STOR_T(3)
STOR	Meat and poultry	2.000E+01	2.000E+01	---	STOR_T(4)
STOR	Fish	not used	7.000E+00	---	STOR_T(5)
STOR	Crustacea and mollusks	not used	7.000E+00	---	STOR_T(6)
STOR	Well water	1.000E+00	1.000E+00	---	STOR_T(7)
STOR	Surface water	1.000E+00	1.000E+00	---	STOR_T(8)
STOR	Livestock fodder	4.500E+01	4.500E+01	---	STOR_T(9)
R021	Thickness of building foundation (m)	not used	1.500E-01	---	FLOOR
R021	Bulk density of building foundation (g/cm**3)	not used	2.400E+00	---	DENSFL
1	Total porosity of the cover material	not used	4.000E-01	---	TPCV
1	Total porosity of the building foundation	not used	1.000E-01	---	TPFL
R021	Volumetric water content of the cover material	not used	5.000E-02	---	PH2OCV
R021	Volumetric water content of the foundation	not used	3.000E-02	---	PH2OFL
R021	Diffusion coefficient for radon gas (m/sec):				
R021	in cover material	not used	2.000E-06	---	DIFCV
R021	in foundation material	not used	3.000E-07	---	DIFFL
R021	in contaminated zone soil	not used	2.000E-06	---	DIFCZ
R021	Radon vertical dimension of mixing (m)	not used	2.000E+00	---	HMIX
R021	Average annual wind speed (m/sec)	not used	2.000E+00	---	WIND
R021	Average building air exchange rate (1/hr)	not used	5.000E-01	---	REXG
R021	Height of the building (room) (m)	not used	2.500E+00	---	HRM
R021	Building interior area factor	not used	0.000E+00	---	FAI
R021	Building depth below ground surface (m)	not used	-1.000E+00	---	DMFL
R021	Emanating power of Rn-222 gas	not used	2.500E-01	---	EMANA(1)
R021	Emanating power of Rn-220 gas	not used	1.500E-01	---	EMANA(2)

Summary of Pathway Selections

Pathway	User Selection
1 -- external gamma	active
2 -- inhalation (w/o radon)	active
3 -- plant ingestion	active
4 -- meat ingestion	active
5 -- milk ingestion	active
6 -- aquatic foods	suppressed
7 -- drinking water	suppressed
8 -- soil ingestion	active
9 -- radon	suppressed

Contaminated Zone Dimensions

Area:	10000.00 square meters
Thickness:	2.00 meters
Cover Depth:	0.00 meters

Initial Soil Concentrations, pCi/g

Ra-226	5.000E+00
Th-230	5.000E+00
Th-232	5.000E+00
U-234	5.000E+00
U-235	2.300E-01
U-238	5.000E+00

Total Dose TDOSE(t), mrem/yr
Basic Radiation Dose Limit = 25 mrem/yr
Total Mixture Sum M(t) = Fraction of Basic Dose Limit Received at Time (t)

t (years):	0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	5.000E+02	7.000E+02	1.000E+03
TDOSE(t):	6.969E+01	7.632E+01	8.953E+01	1.251E+02	1.571E+02	1.612E+02	1.409E+02	1.269E+02	1.179E+02	1.099E+02
M(t):	2.788E+00	3.053E+00	3.581E+00	5.003E+00	6.284E+00	6.447E+00	5.637E+00	5.078E+00	4.716E+00	4.397E+00

Maximum TDOSE(t): 1.631E+02 mrem/yr at t = 64.70 ± 0.06 years

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 64.70 years
Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	2.650E+01	0.1625	1.252E-01	0.0008	0.000E+00	0.0000	4.527E+01	0.2776	1.428E+00	0.0088	1.224E+00	0.0075	4.675E-01	0.0029
Th-230	8.008E-01	0.0049	1.615E+00	0.0099	0.000E+00	0.0000	1.340E+00	0.0082	3.897E-02	0.0002	3.247E-02	0.0002	4.787E-02	0.0003
Th-232	4.344E+01	0.2664	9.822E+00	0.0602	0.000E+00	0.0000	2.676E+01	0.1641	7.473E-01	0.0046	9.497E-01	0.0058	3.429E-01	0.0021
U-234	1.189E-03	0.0000	5.584E-01	0.0034	0.000E+00	0.0000	2.683E-01	0.0016	8.701E-03	0.0001	2.316E-02	0.0001	1.677E-02	0.0001
U-235	8.204E-02	0.0005	2.522E-02	0.0002	0.000E+00	0.0000	1.463E-02	0.0001	9.095E-04	0.0000	1.007E-03	0.0000	7.966E-04	0.0000
U-238	3.190E-01	0.0020	4.986E-01	0.0031	0.000E+00	0.0000	2.547E-01	0.0016	8.263E-03	0.0001	2.201E-02	0.0001	1.593E-02	0.0001
Total	7.115E+01	0.4363	1.264E+01	0.0775	0.000E+00	0.0000	7.391E+01	0.4532	2.232E+00	0.0137	2.252E+00	0.0138	8.917E-01	0.0055

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 64.70 years

0
0 Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	7.502E+01	0.4600
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.875E+00	0.0238
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.206E+01	0.5032
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.765E-01	0.0054
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.246E-01	0.0008
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.119E+00	0.0069
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.631E+02	1.0000

0*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 0.000E+00 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	3.051E+01	0.4378	4.256E-02	0.0006	0.000E+00	0.0000	2.354E+01	0.3379	6.749E-01	0.0097	8.982E-01	0.0129	9.224E-02	0.0013
Th-230	3.363E-03	0.0000	1.613E+00	0.0232	0.000E+00	0.0000	2.441E-01	0.0035	4.919E-03	0.0001	3.812E-04	0.0000	3.800E-02	0.0005
Th-232	1.457E-03	0.0000	8.116E+00	0.1165	0.000E+00	0.0000	1.216E+00	0.0174	2.450E-02	0.0004	1.899E-03	0.0000	1.893E-01	0.0027
U-234	1.127E-03	0.0000	6.533E-01	0.0094	0.000E+00	0.0000	3.139E-01	0.0045	1.018E-02	0.0001	2.713E-02	0.0004	1.963E-02	0.0003
U-235	9.589E-02	0.0014	2.800E-02	0.0004	0.000E+00	0.0000	1.362E-02	0.0002	4.419E-04	0.0000	1.177E-03	0.0000	8.518E-04	0.0000
U-238	3.737E-01	0.0054	5.840E-01	0.0084	0.000E+00	0.0000	2.984E-01	0.0043	9.679E-03	0.0001	2.578E-02	0.0004	1.866E-02	0.0003
Total	3.098E+01	0.4446	1.104E+01	0.1584	0.000E+00	0.0000	2.563E+01	0.3678	7.246E-01	0.0104	9.546E-01	0.0137	3.587E-01	0.0051

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 0.000E+00 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.576E+01	0.8001
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.904E+00	0.0273
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.549E+00	0.1370
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.025E+00	0.0147
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.400E-01	0.0020
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.310E+00	0.0188
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.969E+01	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+00 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	3.044E+01	0.3989	4.598E-02	0.0006	0.000E+00	0.0000	2.462E+01	0.3226	7.221E-01	0.0095	9.187E-01	0.0120	1.074E-01	0.0014
Th-230	1.656E-02	0.0002	1.613E+00	0.0211	0.000E+00	0.0000	2.545E-01	0.0033	5.226E-03	0.0001	7.769E-04	0.0000	3.805E-02	0.0005
Th-232	2.368E+00	0.0310	8.151E+00	0.1068	0.000E+00	0.0000	4.126E+00	0.0541	1.072E-01	0.0014	1.113E-01	0.0015	2.017E-01	0.0026
U-234	1.124E-03	0.0000	6.517E-01	0.0085	0.000E+00	0.0000	3.131E-01	0.0041	1.016E-02	0.0001	2.706E-02	0.0004	1.958E-02	0.0003
U-235	9.566E-02	0.0013	2.794E-02	0.0004	0.000E+00	0.0000	1.364E-02	0.0002	4.506E-04	0.0000	1.174E-03	0.0000	8.504E-04	0.0000
U-238	3.728E-01	0.0049	5.825E-01	0.0076	0.000E+00	0.0000	2.976E-01	0.0039	9.655E-03	0.0001	2.572E-02	0.0003	1.861E-02	0.0002
Total	3.330E+01	0.4363	1.107E+01	0.1451	0.000E+00	0.0000	2.962E+01	0.3882	8.548E-01	0.0112	1.085E+00	0.0142	3.862E-01	0.0051

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+00 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.686E+01	0.7450
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.929E+00	0.0253
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.507E+01	0.1974
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.023E+00	0.0134
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.397E-01	0.0018
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.307E+00	0.0171
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	7.632E+01	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+00 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	3.031E+01	0.3385	5.245E-02	0.0006	0.000E+00	0.0000	2.638E+01	0.2947	7.817E-01	0.0087	9.480E-01	0.0106	1.363E-01	0.0015
Th-230	4.288E-02	0.0005	1.613E+00	0.0180	0.000E+00	0.0000	2.766E-01	0.0031	5.877E-03	0.0001	1.586E-03	0.0000	3.815E-02	0.0004
Th-232	8.383E+00	0.0936	8.336E+00	0.0931	0.000E+00	0.0000	9.002E+00	0.1005	2.456E-01	0.0027	2.932E-01	0.0033	2.265E-01	0.0025
U-234	1.119E-03	0.0000	6.485E-01	0.0072	0.000E+00	0.0000	3.116E-01	0.0035	1.011E-02	0.0001	2.693E-02	0.0003	1.948E-02	0.0002
U-235	9.519E-02	0.0011	2.782E-02	0.0003	0.000E+00	0.0000	1.366E-02	0.0002	4.675E-04	0.0000	1.169E-03	0.0000	8.478E-04	0.0000
U-238	3.710E-01	0.0041	5.797E-01	0.0065	0.000E+00	0.0000	2.962E-01	0.0033	9.608E-03	0.0001	2.560E-02	0.0003	1.852E-02	0.0002
Total	3.920E+01	0.4379	1.126E+01	0.1257	0.000E+00	0.0000	3.628E+01	0.4052	1.053E+00	0.0118	1.297E+00	0.0145	4.398E-01	0.0049

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+00 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.861E+01	0.6546
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.978E+00	0.0221
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.649E+01	0.2958
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.018E+00	0.0114
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.392E-01	0.0016
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.301E+00	0.0145
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.953E+01	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+01 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	2.985E+01	0.2387	7.179E-02	0.0006	0.000E+00	0.0000	3.162E+01	0.2528	9.589E-01	0.0077	1.034E+00	0.0083	2.226E-01	0.0018
Th-230	1.341E-01	0.0011	1.613E+00	0.0129	0.000E+00	0.0000	3.649E-01	0.0029	8.528E-03	0.0001	4.597E-03	0.0000	3.870E-02	0.0003
Th-232	2.694E+01	0.2154	9.092E+00	0.0727	0.000E+00	0.0000	1.920E+01	0.1535	5.339E-01	0.0043	6.711E-01	0.0054	2.902E-01	0.0023
U-234	1.106E-03	0.0000	6.376E-01	0.0051	0.000E+00	0.0000	3.063E-01	0.0024	9.936E-03	0.0001	2.647E-02	0.0002	1.915E-02	0.0002
U-235	9.359E-02	0.0007	2.743E-02	0.0002	0.000E+00	0.0000	1.376E-02	0.0001	5.254E-04	0.0000	1.149E-03	0.0000	8.395E-04	0.0000
U-238	3.647E-01	0.0029	5.699E-01	0.0046	0.000E+00	0.0000	2.912E-01	0.0023	9.445E-03	0.0001	2.516E-02	0.0002	1.820E-02	0.0001
Total	5.738E+01	0.4588	1.201E+01	0.0960	0.000E+00	0.0000	5.180E+01	0.4142	1.521E+00	0.0122	1.762E+00	0.0141	5.898E-01	0.0047

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+01 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.376E+01	0.5098
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.164E+00	0.0173
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.672E+01	0.4535
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.001E+00	0.0080
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.373E-01	0.0011
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.279E+00	0.0102
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.251E+02	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+01 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	2.858E+01	0.1819	1.059E-01	0.0007	0.000E+00	0.0000	4.068E+01	0.2589	1.267E+00	0.0081	1.175E+00	0.0075	3.763E-01	0.0024
Th-230	3.872E-01	0.0025	1.614E+00	0.0103	0.000E+00	0.0000	6.832E-01	0.0043	1.834E-02	0.0001	1.426E-02	0.0001	4.137E-02	0.0003
Th-232	4.200E+01	0.2673	9.758E+00	0.0621	0.000E+00	0.0000	2.611E+01	0.1662	7.291E-01	0.0046	9.260E-01	0.0059	3.383E-01	0.0022
U-234	1.099E-03	0.0000	6.074E-01	0.0039	0.000E+00	0.0000	2.918E-01	0.0019	9.464E-03	0.0001	2.521E-02	0.0002	1.825E-02	0.0001
U-235	8.917E-02	0.0006	2.650E-02	0.0002	0.000E+00	0.0000	1.409E-02	0.0001	6.793E-04	0.0000	1.095E-03	0.0000	8.210E-04	0.0000
U-238	3.473E-01	0.0022	5.427E-01	0.0035	0.000E+00	0.0000	2.773E-01	0.0018	8.994E-03	0.0001	2.396E-02	0.0002	1.734E-02	0.0001
Total	7.141E+01	0.4545	1.265E+01	0.0805	0.000E+00	0.0000	6.805E+01	0.4332	2.034E+00	0.0129	2.165E+00	0.0138	7.924E-01	0.0050

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+01 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	7.218E+01	0.4594
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.758E+00	0.0176
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	7.987E+01	0.5083
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.532E-01	0.0061
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.324E-01	0.0008
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.218E+00	0.0077
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.571E+02	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+02 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	2.454E+01	0.1523	1.248E-01	0.0008	0.000E+00	0.0000	4.440E+01	0.2755	1.406E+00	0.0087	1.177E+00	0.0073	4.718E-01	0.0029
Th-230	1.191E+00	0.0074	1.616E+00	0.0100	0.000E+00	0.0000	2.029E+00	0.0126	6.074E-02	0.0004	5.088E-02	0.0003	5.509E-02	0.0003
Th-232	4.346E+01	0.2697	9.822E+00	0.0609	0.000E+00	0.0000	2.676E+01	0.1661	7.475E-01	0.0046	9.500E-01	0.0059	3.429E-01	0.0021
U-234	1.395E-03	0.0000	5.126E-01	0.0032	0.000E+00	0.0000	2.466E-01	0.0015	7.995E-03	0.0000	2.125E-02	0.0001	1.540E-02	0.0001
U-235	7.539E-02	0.0005	2.406E-02	0.0001	0.000E+00	0.0000	1.506E-02	0.0001	1.101E-03	0.0000	9.252E-04	0.0000	7.738E-04	0.0000
U-238	2.926E-01	0.0018	4.574E-01	0.0028	0.000E+00	0.0000	2.337E-01	0.0014	7.580E-03	0.0000	2.019E-02	0.0001	1.461E-02	0.0001
Total	6.956E+01	0.4316	1.256E+01	0.0779	0.000E+00	0.0000	7.369E+01	0.4573	2.231E+00	0.0138	2.221E+00	0.0138	9.006E-01	0.0056

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+02 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	7.212E+01	0.4475
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.002E+00	0.0310
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.209E+01	0.5093
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.052E-01	0.0050
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.173E-01	0.0007
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.026E+00	0.0064
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.612E+02	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+02 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	1.586E+01	0.1126	8.366E-02	0.0006	0.000E+00	0.0000	2.954E+01	0.2097	9.369E-01	0.0066	7.760E-01	0.0055	3.182E-01	0.0023
Th-230	2.909E+00	0.0206	1.622E+00	0.0115	0.000E+00	0.0000	5.213E+00	0.0370	1.617E-01	0.0011	1.347E-01	0.0010	8.922E-02	0.0006
Th-232	4.344E+01	0.3083	9.818E+00	0.0697	0.000E+00	0.0000	2.675E+01	0.1899	7.472E-01	0.0053	9.496E-01	0.0067	3.428E-01	0.0024
U-234	3.960E-03	0.0000	3.164E-01	0.0022	0.000E+00	0.0000	1.566E-01	0.0011	5.068E-03	0.0000	1.316E-02	0.0001	9.540E-03	0.0001
U-235	4.672E-02	0.0003	1.810E-02	0.0001	0.000E+00	0.0000	1.497E-02	0.0001	1.597E-03	0.0000	5.724E-04	0.0000	6.281E-04	0.0000
U-238	1.794E-01	0.0013	2.805E-01	0.0020	0.000E+00	0.0000	1.433E-01	0.0010	4.649E-03	0.0000	1.239E-02	0.0001	8.962E-03	0.0001
Total	6.244E+01	0.4431	1.214E+01	0.0861	0.000E+00	0.0000	6.183E+01	0.4387	1.857E+00	0.0132	1.886E+00	0.0134	7.693E-01	0.0055

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+02 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.752E+01	0.3372
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.013E+01	0.0719
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.205E+01	0.5823
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.047E-01	0.0036
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.259E-02	0.0006
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.292E-01	0.0045
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.409E+02	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 5.000E+02 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	1.025E+01	0.0808	5.407E-02	0.0004	0.000E+00	0.0000	1.910E+01	0.1504	6.056E-01	0.0048	5.015E-01	0.0040	2.057E-01	0.0016
Th-230	4.015E+00	0.0316	1.624E+00	0.0128	0.000E+00	0.0000	7.273E+00	0.0573	2.270E-01	0.0018	1.888E-01	0.0015	1.113E-01	0.0009
Th-232	4.343E+01	0.3421	9.814E+00	0.0773	0.000E+00	0.0000	2.674E+01	0.2107	7.469E-01	0.0059	9.492E-01	0.0075	3.426E-01	0.0027
U-234	7.475E-03	0.0001	1.961E-01	0.0015	0.000E+00	0.0000	1.050E-01	0.0008	3.389E-03	0.0000	8.300E-03	0.0001	5.990E-03	0.0000
U-235	2.895E-02	0.0002	1.315E-02	0.0001	0.000E+00	0.0000	1.268E-02	0.0001	1.542E-03	0.0000	3.541E-04	0.0000	4.791E-04	0.0000
U-238	1.100E-01	0.0009	1.721E-01	0.0014	0.000E+00	0.0000	8.792E-02	0.0007	2.852E-03	0.0000	7.598E-03	0.0001	5.497E-03	0.0000
Total	5.784E+01	0.4556	1.187E+01	0.0935	0.000E+00	0.0000	5.332E+01	0.4200	1.587E+00	0.0125	1.656E+00	0.0130	6.716E-01	0.0053

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 5.000E+02 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.071E+01	0.2419
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.344E+01	0.1059
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.202E+01	0.6461
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.263E-01	0.0026
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.716E-02	0.0005
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.859E-01	0.0030
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.269E+02	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 7.000E+02 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	6.625E+00	0.0562	3.495E-02	0.0003	0.000E+00	0.0000	1.234E+01	0.1047	3.914E-01	0.0033	3.242E-01	0.0027	1.329E-01	0.0011
Th-230	4.725E+00	0.0401	1.624E+00	0.0138	0.000E+00	0.0000	8.596E+00	0.0729	2.690E-01	0.0023	2.235E-01	0.0019	1.255E-01	0.0011
Th-232	4.341E+01	0.3682	9.810E+00	0.0832	0.000E+00	0.0000	2.673E+01	0.2267	7.466E-01	0.0063	9.488E-01	0.0080	3.425E-01	0.0029
U-234	1.088E-02	0.0001	1.225E-01	0.0010	0.000E+00	0.0000	7.579E-02	0.0006	2.434E-03	0.0000	5.382E-03	0.0000	3.840E-03	0.0000
U-235	1.794E-02	0.0002	9.317E-03	0.0001	0.000E+00	0.0000	9.914E-03	0.0001	1.289E-03	0.0000	2.190E-04	0.0000	3.511E-04	0.0000
U-238	6.741E-02	0.0006	1.056E-01	0.0009	0.000E+00	0.0000	5.393E-02	0.0005	1.750E-03	0.0000	4.660E-03	0.0000	3.372E-03	0.0000
Total	5.485E+01	0.4653	1.171E+01	0.0993	0.000E+00	0.0000	4.781E+01	0.4055	1.412E+00	0.0120	1.507E+00	0.0128	6.085E-01	0.0052

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 7.000E+02 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.985E+01	0.1684
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.556E+01	0.1320
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.199E+01	0.6954
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.208E-01	0.0019
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.903E-02	0.0003
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.367E-01	0.0020
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.179E+02	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+03 years
Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	3.442E+00	0.0313	1.816E-02	0.0002	0.000E+00	0.0000	6.412E+00	0.0583	2.034E-01	0.0018	1.684E-01	0.0015	6.906E-02	0.0006
Th-230	5.340E+00	0.0486	1.622E+00	0.0148	0.000E+00	0.0000	9.743E+00	0.0886	3.054E-01	0.0028	2.537E-01	0.0023	1.378E-01	0.0013
Th-232	4.338E+01	0.3946	9.804E+00	0.0892	0.000E+00	0.0000	2.672E+01	0.2430	7.462E-01	0.0068	9.483E-01	0.0086	3.423E-01	0.0031
U-234	1.498E-02	0.0001	6.184E-02	0.0006	0.000E+00	0.0000	5.413E-02	0.0005	1.724E-03	0.0000	3.044E-03	0.0000	2.096E-03	0.0000
U-235	8.745E-03	0.0001	5.370E-03	0.0000	0.000E+00	0.0000	6.291E-03	0.0001	8.650E-04	0.0000	1.065E-04	0.0000	2.096E-04	0.0000
U-238	3.236E-02	0.0003	5.072E-02	0.0005	0.000E+00	0.0000	2.592E-02	0.0002	8.409E-04	0.0000	2.239E-03	0.0000	1.620E-03	0.0000
Total	5.222E+01	0.4750	1.156E+01	0.1052	0.000E+00	0.0000	4.296E+01	0.3908	1.258E+00	0.0114	1.376E+00	0.0125	5.530E-01	0.0050

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+03 years
Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ra-226	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.031E+01	0.0938
Th-230	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.740E+01	0.1583
Th-232	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.194E+01	0.7454
U-234	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.378E-01	0.0013
U-235	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.159E-02	0.0002
U-238	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.137E-01	0.0010
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.099E+02	1.0000

*Sum of all water independent and dependent pathways.

Dose/Source Ratios Summed Over All Pathways
Parent and Progeny Principal Radionuclide Contributions Indicated

Parent (i)	Product (j)	Branch Fraction	DSR(j,t) (mrem/yr)/(pCi/g) t= 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	5.000E+02	7.000E+02	1.000E+03
Ra-226	Ra-226	1.000E+00	1.115E+01	1.113E+01	1.108E+01	1.091E+01	1.045E+01	8.965E+00	5.794E+00	3.745E+00	2.420E+00	1.258E+00
Ra-226	Pb-210	1.000E+00	0.000E+00	2.434E-01	6.430E-01	1.841E+00	3.991E+00	5.459E+00	3.709E+00	2.398E+00	1.550E+00	8.052E-01
Ra-226	SDSR(j)		1.115E+01	1.137E+01	1.172E+01	1.275E+01	1.444E+01	1.442E+01	9.504E+00	6.143E+00	3.970E+00	2.063E+00
0Th-230	Th-230	1.000E+00	3.808E-01	3.808E-01	3.808E-01	3.808E-01	3.807E-01	3.804E-01	3.796E-01	3.787E-01	3.779E-01	3.766E-01
Th-230	Ra-226	1.000E+00	0.000E+00	4.828E-03	1.445E-02	4.779E-02	1.403E-01	4.338E-01	1.062E+00	1.466E+00	1.725E+00	1.950E+00
Th-230	Pb-210	1.000E+00	0.000E+00	6.015E-05	4.461E-04	4.287E-03	3.068E-02	1.863E-01	5.847E-01	8.435E-01	1.010E+00	1.154E+00
Th-230	SDSR(j)		3.808E-01	3.857E-01	3.957E-01	4.328E-01	5.516E-01	1.000E+00	2.026E+00	2.688E+00	3.113E+00	3.480E+00
0Th-232	Th-232	1.000E+00	1.910E+00	1.910E+00	1.910E+00	1.910E+00	1.910E+00	1.909E+00	1.909E+00	1.908E+00	1.907E+00	1.906E+00
Th-232	Ra-228	1.000E+00	0.000E+00	9.872E-01	2.633E+00	6.048E+00	8.352E+00	8.569E+00	8.565E+00	8.562E+00	8.558E+00	8.553E+00
Th-232	Th-228	1.000E+00	0.000E+00	1.160E-01	7.549E-01	3.387E+00	5.712E+00	5.939E+00	5.937E+00	5.935E+00	5.932E+00	5.928E+00
Th-232	SDSR(j)		1.910E+00	3.013E+00	5.297E+00	1.134E+01	1.597E+01	1.642E+01	1.641E+01	1.640E+01	1.640E+01	1.639E+01
0U-234	U-234	1.000E+00	2.050E-01	2.045E-01	2.035E-01	2.001E-01	1.905E-01	1.605E-01	9.833E-02	6.024E-02	3.691E-02	1.770E-02
U-234	Th-230	1.000E+00	0.000E+00	3.466E-06	1.029E-05	3.390E-05	9.918E-05	3.039E-04	7.270E-04	9.850E-04	1.142E-03	1.269E-03
U-234	Ra-226	1.000E+00	0.000E+00	2.173E-08	1.949E-07	2.141E-06	1.869E-05	1.866E-04	1.247E-03	2.607E-03	3.896E-03	5.432E-03
U-234	Pb-210	1.000E+00	0.000E+00	2.023E-10	4.233E-09	1.334E-07	2.947E-06	6.473E-05	6.303E-04	1.431E-03	2.214E-03	3.163E-03
U-234	SDSR(j)		2.050E-01	2.045E-01	2.035E-01	2.001E-01	1.906E-01	1.610E-01	1.009E-01	6.526E-02	4.416E-02	2.756E-02
0U-235	U-235	1.000E+00	6.086E-01	6.072E-01	6.042E-01	5.939E-01	5.656E-01	4.765E-01	2.921E-01	1.791E-01	1.098E-01	5.269E-02
U-235	Pa-231	1.000E+00	0.000E+00	2.745E-04	8.143E-04	2.662E-03	7.598E-03	2.132E-02	3.912E-02	3.989E-02	3.416E-02	2.335E-02
U-235	Ac-227	1.000E+00	0.000E+00	4.057E-06	3.364E-05	3.338E-04	2.301E-03	1.217E-02	2.785E-02	2.957E-02	2.575E-02	1.782E-02
U-235	SDSR(j)		6.086E-01	6.074E-01	6.050E-01	5.969E-01	5.755E-01	5.100E-01	3.591E-01	2.485E-01	1.697E-01	9.386E-02
0U-238	U-238	1.000E+00	2.620E-01	2.614E-01	2.601E-01	2.557E-01	2.435E-01	2.052E-01	1.258E-01	7.709E-02	4.726E-02	2.268E-02
U-238	U-234	1.000E+00	0.000E+00	5.872E-07	1.738E-06	5.679E-06	1.621E-05	4.551E-05	8.366E-05	8.545E-05	7.332E-05	5.025E-05
U-238	Th-230	1.000E+00	0.000E+00	4.973E-12	4.388E-11	4.792E-10	4.168E-09	4.134E-08	2.719E-07	5.599E-07	8.258E-07	1.132E-06
U-238	Ra-226	1.000E+00	0.000E+00	2.055E-14	5.522E-13	2.019E-11	5.262E-10	1.723E-08	3.292E-07	1.089E-06	2.155E-06	3.927E-06
U-238	Pb-210	1.000E+00	0.000E+00	1.578E-16	9.434E-15	9.725E-13	6.531E-11	5.035E-09	1.531E-07	5.680E-07	1.183E-06	2.238E-06
U-238	SDSR(j)		2.620E-01	2.614E-01	2.601E-01	2.557E-01	2.435E-01	2.052E-01	1.258E-01	7.718E-02	4.734E-02	2.274E-02

Branch Fraction is the cumulative factor for the j'th principal radionuclide daughter: CUMBRF(j) = BRF(1)*BRF(2)* ... BRF(j).
The DSR includes contributions from associated (half-life = 0.5 yr) daughters.

Single Radionuclide Soil Guidelines G(i,t) in pCi/g
Basic Radiation Dose Limit = 25 mrem/yr

Nuclide (i)	t= 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	5.000E+02	7.000E+02	1.000E+03
Ra-226	2.242E+00	2.199E+00	2.133E+00	1.960E+00	1.732E+00	1.733E+00	2.631E+00	4.070E+00	6.297E+00	1.212E+01
Th-230	6.565E+01	6.482E+01	6.318E+01	5.776E+01	4.532E+01	2.499E+01	1.234E+01	9.301E+00	8.032E+00	7.183E+00
Th-232	1.309E+01	8.297E+00	4.719E+00	2.204E+00	1.565E+00	1.523E+00	1.523E+00	1.524E+00	1.525E+00	1.526E+00
U-234	1.219E+02	1.222E+02	1.228E+02	1.249E+02	1.311E+02	1.552E+02	2.477E+02	3.831E+02	5.661E+02	9.070E+02
U-235	4.108E+01	4.116E+01	4.132E+01	4.188E+01	4.344E+01	4.902E+01	6.962E+01	1.006E+02	1.473E+02	2.664E+02
U-238	9.541E+01	9.564E+01	9.611E+01	9.777E+01	1.027E+02	1.218E+02	1.987E+02	3.239E+02	5.281E+02	1.099E+03

0

Summed Dose/Source Ratios DSR(i,t) in (mrem/yr)/(pCi/g)
and Single Radionuclide Soil Guidelines G(i,t) in pCi/g
at tmin = time of minimum single radionuclide soil guideline
and at tmax = time of maximum total dose = 64.70 ± 0.06 years

Nuclide (i)	Initial pCi/g	tmin (years)	DSR(i,tmin)	G(i,tmin) (pCi/g)	DSR(i,tmax)	G(i,tmax) (pCi/g)
Ra-226	5.000E+00	58.07 ± 0.06	1.503E+01	1.664E+00	1.500E+01	1.666E+00
Th-230	5.000E+00	1.000E+03	3.480E+00	7.183E+00	7.750E-01	3.226E+01
Th-232	5.000E+00	91.08 ± 0.09	1.642E+01	1.523E+00	1.641E+01	1.523E+00
U-234	5.000E+00	0.000E+00	2.050E-01	1.219E+02	1.753E-01	1.426E+02
U-235	2.300E-01	0.000E+00	6.086E-01	4.108E+01	5.418E-01	4.615E+01
U-238	5.000E+00	0.000E+00	2.620E-01	9.541E+01	2.237E-01	1.118E+02

Individual Nuclide Dose Summed Over All Pathways
Parent Nuclide and Branch Fraction Indicated

Nuclide (j)	Parent (i)	BRF(i)	DOSE(j,t), mrem/yr									
		t=	0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	5.000E+02	7.000E+02	1.000E+03
Ra-226	Ra-226	1.000E+00	5.576E+01	5.564E+01	5.540E+01	5.456E+01	5.223E+01	4.483E+01	2.897E+01	1.872E+01	1.210E+01	6.288E+00
Ra-226	Th-230	1.000E+00	0.000E+00	2.414E-02	7.224E-02	2.389E-01	7.014E-01	2.169E+00	5.308E+00	7.328E+00	8.625E+00	9.748E+00
Ra-226	U-234	1.000E+00	0.000E+00	1.087E-07	9.743E-07	1.071E-05	9.343E-05	9.328E-04	6.236E-03	1.303E-02	1.948E-02	2.716E-02
Ra-226	U-238	1.000E+00	0.000E+00	1.028E-13	2.761E-12	1.010E-10	2.731E-09	8.615E-08	1.646E-06	5.443E-06	1.078E-05	1.963E-05
Ra-226	SDOSE(j):		5.576E+01	5.566E+01	5.547E+01	5.480E+01	5.293E+01	4.700E+01	3.429E+01	2.607E+01	2.075E+01	1.606E+01
0Pb-210	Ra-226	1.000E+00	0.000E+00	1.217E+00	3.215E+00	9.205E+00	1.996E+01	2.730E+01	1.855E+01	1.199E+01	7.748E+00	4.026E+00
Pb-210	Th-230	1.000E+00	0.000E+00	3.008E-04	2.231E-03	2.143E-02	1.534E-01	9.314E-01	2.923E+00	4.218E+00	5.049E+00	5.770E+00
Pb-210	U-234	1.000E+00	0.000E+00	1.011E-09	2.116E-08	6.672E-07	1.474E-05	3.237E-04	3.152E-03	7.155E-03	1.107E-02	1.581E-02
Pb-210	U-238	1.000E+00	0.000E+00	7.892E-16	4.717E-14	4.863E-12	3.265E-10	2.518E-08	7.657E-07	2.840E-06	5.914E-06	1.119E-05
Pb-210	SDOSE(j):		0.000E+00	1.217E+00	3.217E+00	9.227E+00	2.011E+01	2.823E+01	2.147E+01	1.621E+01	1.281E+01	9.812E+00
0Th-230	Th-230	1.000E+00	1.904E+00	1.904E+00	1.904E+00	1.904E+00	1.903E+00	1.902E+00	1.898E+00	1.894E+00	1.889E+00	1.883E+00
Th-230	U-234	1.000E+00	0.000E+00	1.733E-05	5.144E-05	1.605E-04	4.959E-04	1.520E-03	3.635E-03	4.925E-03	5.709E-03	6.345E-03
Th-230	U-238	1.000E+00	0.000E+00	2.486E-11	2.194E-10	2.396E-09	2.084E-08	2.067E-07	1.359E-06	2.799E-06	4.129E-06	5.660E-06
Th-230	SDOSE(j):		1.904E+00	1.904E+00	1.904E+00	1.904E+00	1.904E+00	1.903E+00	1.901E+00	1.899E+00	1.895E+00	1.890E+00
0Th-232	Th-232	1.000E+00	9.549E+00	9.549E+00	9.549E+00	9.549E+00	9.549E+00	9.547E+00	9.543E+00	9.540E+00	9.536E+00	9.530E+00
0Ra-228	Th-232	1.000E+00	0.000E+00	4.936E+00	1.316E+01	3.024E+01	4.176E+01	4.284E+01	4.283E+01	4.281E+01	4.279E+01	4.277E+01
0Th-228	Th-232	1.000E+00	0.000E+00	5.799E-01	3.775E+00	1.693E+01	2.856E+01	2.970E+01	2.968E+01	2.967E+01	2.966E+01	2.964E+01
0U-234	U-234	1.000E+00	1.025E+00	1.023E+00	1.018E+00	1.000E+00	9.526E-01	8.025E-01	4.916E-01	3.012E-01	1.845E-01	8.849E-02
U-234	U-238	1.000E+00	0.000E+00	2.936E-06	8.692E-06	2.840E-05	8.105E-05	2.276E-04	4.183E-04	4.273E-04	3.666E-04	2.512E-04
U-234	SDOSE(j):		1.025E+00	1.023E+00	1.018E+00	1.000E+00	9.526E-01	8.027E-01	4.921E-01	3.016E-01	1.849E-01	8.874E-02
0U-235	U-235	1.000E+00	1.400E-01	1.396E-01	1.390E-01	1.366E-01	1.301E-01	1.096E-01	6.719E-02	4.119E-02	2.525E-02	1.212E-02
0Pa-231	U-235	1.000E+00	0.000E+00	6.314E-05	1.873E-04	6.123E-04	1.748E-03	4.904E-03	8.998E-03	9.174E-03	7.856E-03	5.370E-03
0Ac-227	U-235	1.000E+00	0.000E+00	9.331E-07	7.738E-06	7.677E-05	5.293E-04	2.799E-03	6.406E-03	6.801E-03	5.924E-03	4.100E-03
0U-238	U-238	1.000E+00	1.310E+00	1.307E+00	1.301E+00	1.278E+00	1.217E+00	1.026E+00	6.288E-01	3.855E-01	2.363E-01	1.134E-01

BRF(i) is the branch fraction of the parent nuclide.

Individual Nuclide Soil Concentration
Parent Nuclide and Branch Fraction Indicated
S(j,t), pCi/g

Nuclide (j)	Parent (i)	BRF(i)	t= 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	5.000E+02	7.000E+02	1.000E+03
Ra-226	Ra-226	1.000E+00	5.000E+00	4.989E+00	4.967E+00	4.892E+00	4.683E+00	4.020E+00	2.598E+00	1.679E+00	1.085E+00	5.638E-01
Ra-226	Th-230	1.000E+00	0.000E+00	2.164E-03	6.477E-03	2.143E-02	6.289E-02	1.945E-01	4.759E-01	6.571E-01	7.734E-01	8.741E-01
Ra-226	U-234	1.000E+00	0.000E+00	9.734E-09	8.734E-08	9.600E-07	8.378E-06	8.364E-05	5.592E-04	1.169E-03	1.747E-03	2.435E-03
Ra-226	U-238	1.000E+00	0.000E+00	9.200E-15	2.474E-13	9.051E-12	2.359E-10	7.725E-09	1.476E-07	4.881E-07	9.662E-07	1.761E-06
Ra-226	SS(j):		5.000E+00	4.991E+00	4.974E+00	4.913E+00	4.746E+00	4.214E+00	3.074E+00	2.337E+00	1.860E+00	1.440E+00
0Pb-210	Ra-226	1.000E+00	0.000E+00	1.528E-01	4.429E-01	1.313E+00	2.875E+00	3.943E+00	2.680E+00	1.732E+00	1.120E+00	5.817E-01
Pb-210	Th-230	1.000E+00	0.000E+00	3.328E-05	2.928E-04	3.009E-03	2.196E-02	1.343E-01	4.221E-01	6.091E-01	7.293E-01	8.334E-01
Pb-210	U-234	1.000E+00	0.000E+00	1.001E-10	2.653E-09	9.226E-08	2.099E-06	4.659E-05	4.549E-04	1.033E-03	1.599E-03	2.284E-03
Pb-210	U-238	1.000E+00	0.000E+00	7.054E-17	5.661E-15	6.625E-13	4.627E-11	3.618E-09	1.105E-07	4.100E-07	8.539E-07	1.616E-06
Pb-210	SS(j):		0.000E+00	1.528E-01	4.432E-01	1.316E+00	2.897E+00	4.078E+00	3.103E+00	2.343E+00	1.851E+00	1.417E+00
0Th-230	Th-230	1.000E+00	5.000E+00	5.000E+00	5.000E+00	4.999E+00	4.998E+00	4.994E+00	4.983E+00	4.972E+00	4.961E+00	4.945E+00
Th-230	U-234	1.000E+00	0.000E+00	4.495E-05	1.345E-04	4.446E-04	1.302E-03	3.990E-03	9.545E-03	1.293E-02	1.499E-02	1.666E-02
Th-230	U-238	1.000E+00	0.000E+00	6.370E-11	5.714E-10	6.277E-09	5.468E-08	5.426E-07	3.569E-06	7.351E-06	1.084E-05	1.486E-05
Th-230	SS(j):		5.000E+00	5.000E+00	5.000E+00	5.000E+00	5.000E+00	4.998E+00	4.993E+00	4.985E+00	4.976E+00	4.962E+00
0Th-232	Th-232	1.000E+00	5.000E+00	5.000E+00	5.000E+00	5.000E+00	5.000E+00	4.999E+00	4.997E+00	4.995E+00	4.993E+00	4.990E+00
0Ra-228	Th-232	1.000E+00	0.000E+00	5.673E-01	1.514E+00	3.478E+00	4.803E+00	4.928E+00	4.926E+00	4.924E+00	4.922E+00	4.918E+00
0Th-228	Th-232	1.000E+00	0.000E+00	9.316E-02	6.205E-01	2.806E+00	4.739E+00	4.928E+00	4.926E+00	4.924E+00	4.922E+00	4.918E+00
0U-234	U-234	1.000E+00	5.000E+00	4.988E+00	4.963E+00	4.879E+00	4.646E+00	3.914E+00	2.398E+00	1.469E+00	9.000E-01	4.316E-01
U-234	U-238	1.000E+00	0.000E+00	1.414E-05	4.221E-05	1.383E-04	3.951E-04	1.110E-03	2.040E-03	2.084E-03	1.788E-03	1.225E-03
U-234	SS(j):		5.000E+00	4.988E+00	4.963E+00	4.879E+00	4.646E+00	3.915E+00	2.400E+00	1.471E+00	9.018E-01	4.328E-01
0U-235	U-235	1.000E+00	2.300E-01	2.294E-01	2.283E-01	2.244E-01	2.137E-01	1.801E-01	1.104E-01	6.767E-02	4.148E-02	1.991E-02
0Pa-231	U-235	1.000E+00	0.000E+00	4.854E-06	1.449E-05	4.748E-05	1.356E-04	3.806E-04	6.985E-04	7.121E-04	6.099E-04	4.168E-04
0Ac-227	U-235	1.000E+00	0.000E+00	7.637E-08	6.681E-07	6.738E-06	4.680E-05	2.481E-04	5.680E-04	6.031E-04	5.253E-04	3.636E-04
0U-238	U-238	1.000E+00	5.000E+00	4.988E+00	4.963E+00	4.879E+00	4.546E+00	3.915E+00	2.400E+00	1.471E+00	9.018E-01	4.328E-01

BRF(i) is the branch fraction of the parent nuclide.

APPENDIX F
PROJECT SCHEDULE

1.0 OVERVIEW

This appendix contains the proposed schedule for the C-T Decommissioning Project. The schedule in Attachment 1 is a Microsoft Project version of the expected general task schedule for the Phase I project. It shows a project completion 23 months after Plan approval.

The following assumptions and constraints were used in generating the schedule:

- .Job site mobilization can commence 3 months after approval.
- The schedule includes contractor and Mallinckrodt activities.
- The schedule is based on a 5 day work week, 8 hours per day.
- Labor overtime is not scheduled (considered contingency).
- 11 Holidays per year are factored into the schedule.
- Parallel activities can be performed in all the affected areas.
- Equipment will be removed as required before decontamination or demolition activities.
- Buildings 246, 246A, 247, 247A, and 248 will be demolished together.

The remediation contractor's project manager will create the actual schedule and will maintain the detailed schedule, control resource changes, and coordinate with Mallinckrodt to meet all reporting requirements. The sequence of site activities may change from the attached schedule to minimize on-site time and interference with ongoing production activities in nearby buildings.

Building groupings have been made for ease of schedule presentation. The buildings have been grouped according to their decommissioning paths, i.e. buildings with similar decommissioning activities have been grouped together.

- Building Groups A and B consist of buildings scheduled to be demolished (razed) and include Bldgs 213, 214, 238, 246, 247, and 248.
- Building Group C consists of Buildings 90 and 91. Some equipment in these buildings needs to be surveyed prior to final release.
- Building Group D consists of Buildings 62 and 240. These buildings will be surveyed for final release only.
- Building Group E consists of Buildings 235 and 236. Some surface decontamination, equipment removal, and roof replacement and surveys are necessary for release.
- Building Group F consists of Buildings 101, 200E, 200W, 201, 204, 222, and 223. These roofs will have the equipment surveyed, decontaminated if needed and then the roofs will be subjected to a final survey.

Mallinckrodt

C-T Project

Decommissioning Plan

Part I

November 11, 1997

188 pages
14 maps

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