DECOMMISSIONING PLAN

CINTICHEM, INC.
RESEARCH REACTOR
AND
RADIOCHEMICAL PROCESSING
LABORATORY

DOCKET NO'S. 50-54 AND 70-687
NRC LICENSE NO'S. R-81
and SNM639
NYSDOL LICENSE NO. 0729-0322

CINTICHEM, INC.
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TUXEDO, NY 10987

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CINTICHEM INC.

DECOMMISSIONING PLAN

LICENSE NO. R-81
DOCKET 50-54

LICENSE NO. SNM 639
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CINTICHEM DECOMMISSIONING PLAN

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1.0 Background and Management

This plan outlines the dismantling and decommissioning of the Cintichem Reactor and Hot Laboratory facility and return of the facility to unrestricted use. This plan is being submitted as support to the application for authority to surrender licenses R-81 and SNM-639 voluntarily under 10 CFR 50.82 "Application for Termination of Licenses" and 10 CFR 70.38 "Expiration and Termination of Licenses" and to surrender voluntarily the New York State Department of Labor (NYSDOL) License 729-0322.

The outline of this Decommissioning Plan follows the suggested format prescribed in the U. S. Nuclear Regulatory Commission Standardization and Special Projects Branch paper entitled "Guidance and Discussion of Requirements for an Application to Terminate a Non-Power Reactor Facility License" Revision 1, dated September 15, 1984 and "Instructions for the Decontamination of Facilities and Equipment Prior to Release for Uncontrolled Use Pursuant to Section 38.29 of Industrial Code Rule 38."

1.1 Summary of Plan Content

Section One provides a brief description of the Cintichem Reactor and Hot Laboratory facility; the licenses to be terminated; the usage over the licensed period; the decommissioning approach; total costs involved in the decommissioning; the availability of funds; the schedules of major tasks; estimated total dose equivalent; contractor assistance; and the final radiation survey plan.

1.1.1 Description of Cintichem Reactor and Hot Laboratory Facility

The Cintichem Inc. nuclear reactor facility is located within the Town of Tuxedo, in Orange County, New York. Tuxedo is in the southeastern corner of Orange County. The plant site consists of 100 acres of land, owned by Cintichem, located on Long Meadow Road. It is in an industrial park area known as Sterling Forest and is about 3 1/4 miles northwest of the village of Tuxedo Park. A layout of the Cintichem complex is shown in Figure 1.1. Figures 1.2 and 1.3 show the layout of the reactor and hot laborator buildings.

The six ..incipal buildings ac the plant site are:

Building 1	Reactor
Building 2	Hot Laboratory (structurally joined to the reactor building)
Building 3	Maintenance, Warehouse and Engineering
Building 4	Administration & Manufacturing
Building 5	Heating Plant
Building 6	Class A Low Level Waste Storage

Only the Reactor Building (Building 1), the Hot Laboratory (Building 2) and the Class A Low Level Radioactive Waste Storage Building (Building 6) will be included in this Decommissioning Plan along with accessory structures and systems.

1.1.1.1 Reactor Building (Building 1)

The reactor building is a 70 x 92 x 57-ft-high (from the base floor) reinforced-concrete structure set into an excavation in the side of the adjacent rock mountain. Shielding and containment are provided on three sides of the building: by the native mountain rock against the west wall, and a combination of rock and fill on the north and south sides (Figure 1.4). The exposed portions of the walls and roof are reinforced concrete with a minimum thickness of 12 in. and 8 in., respectively. The volume of the reactor building is about 285,000 ft^3.

The area around the reactor is serviced by a 10-ton bridge crane traveling the length of the building. The reactor control room, several offices, and laboratories are provided inside of the reactor building.

The nuclear reactor is a pool-type research reactor and is licensed to operate at thermal power levels up to 5 MW. The reactor is a light-water-moderated, cooled, shielded, and reflected, solid-fuel reactor. It is typical of a number of NRC-licensed reactors based on the MTR design.

The reactor has a number of experimental facilities, including six beam tubes, a thermal column, and pneumatic rabbit tubes (Figures 1.5 and 1.6). The reactor core was composed of MTR type fuel assemblies inserted in the core grid plate. The reactor core grid plate, suspended from a movable bridge, is immersed in a 49x23x32-ft-high pool of demineralized water. The pool is divided into two sections separated by a 4-ft-wide opening that can be closed by a removable watertight gate. The narrower stall section contains the fixed experimental facilities such as the beam tubes and thermal column. The open end of the pool provides storage space for irradiated fuel and experiments. A 4' wide x 12' deep x 108' long canal connects the open pool with the hot cells to permit the transfer of irradiated material between the two facilities (see Figure 1.2).

Shielding in the stall area consists of a 5.8-ft-thick barytes concrete wall extending to a height of 15 ft above the pool floor. The wall thickness is then reduced to 3 ft to the top of the stall. The first 4 ft of the wall above the step is of barytes concrete and the remainder of regular 3000 psi concrete.

All sections of the pool and stall areas that are in contact with the reactor water are coated for ease of decontamination and to prevent interaction of the reactor water with the concrete. Areas normally exposed to high radiation are coated with glazed ceramic tile. Such areas include the floor of the pool and stall, the sides of stall up to and including the 15-ft step, and the sides of the pool behind the fuel storage racks.

The core support bridge, which is movable on rails mounted on top of the pool walls, is constructed of structural steel. The central section incorporates a superstructure to allow for the mounting of reactor control mechanisms and electrical equipment. The bridge is moved by manual rotation of a crank handle for positioning the reactor in either the pool or stall.

The core support tower, suspended from the bridge, is a structural aluminum frame.

The primary cooling system consists of demineralized water system plus heat exchangers and pumps. Primary coolant piping is aluminum; however, the portion of the reactor piping embedded in the concrete is constructed of stainless steel to eliminate the possibility of corrosive attack on the inaccessible piping. Core heat is transferred to the primary coolant. It is subsequently transferred to the secondary coolant through the heat exchangers. The secondary cooling system, in turn, transfers its heat to the atmosphere by the use of a cooling tower. During forced cooling, pool water flows via gravity downward in the pool or stall through the reactor core grid plate and plenum at a flow rate of approximately 2,200 gal/min and then to a holdup tank. Subsequently, the water is drawn from the holdup tank by a 2,200-gal/min main circulating pump, is pumped through the shell sides of the two stainless-steel heat exchangers in series, and then back into the pool.

A holdup tank (HUT) (30' x 15' x 10') is designed to provide a delay of the pool water in the primary system during normal operation to allow time for decay of N16 and other short-lived radioisotopes in the coolant before the water enters the pump room. The HUT is an underground concrete enclosure adjacent to the pump room and buried under 30 feet of soil. The total capacity is 33,000 gallons.

A 100,000-gallon (27' D x 23.75' H cylinder) aluminum storage tank is located south of the reactor building. This allows for drainage and subsequent storage of pool water while work is being done inside the pool. The bottom of the storage tank is about 8 feet above the pool surface so that the pool cannot be inadvertently drained by gravity. Pool water is transferred to the storage tank by means of the primary circulating pump.

The secondary cooling system transfers the heat from the heat exchanger to the cooling tower where the heat is dissipated to the atmosphere. A two-bay, forced-draft cooling tower, located near the reactor building, dissipates the heat in the secondary water transferred from the heat exchanger. Makeup water is supplied from the regular municipal water supply. The secondary system operates at a higher pressure than the primary system, thus preventing leakage of primary coolant to the secondary system. The secondary system pump delivers 2,300 gallons per minute.

The reactor primary water demineralizer system maintains the purity of the primary coolant system. It consists of anion and cation columns and accessories such as a pump, piping, valves and filters.

The reactor building ventilation system (Figure 1.7) is designed to provide these general features, (1) supply comfort heating and air conditioning; (2) maintain the pressure in the building negative; and (3) provide for building isolation in the event of a major radioactive airborne release. Approximately 19,000 ft^3/min of outside air is drawn into the facility fan room, filtered and cooled or heated as necessary and distributed throughout the reactor building by a supply fan. There are about 4 air changes per hour and no air is recirculated. The reactor building during reactor operation is normally maintained at a negative pressure of 1/4 in. water. There is an interlock that shuts the system down at a negative pressure of 1" of water. A pool sweep exhaust system takes 5,000 ft^3/min of air from above the reactor. This combines with the other exhaust streams in the reactor building. The air is exhausted by a 20,000 ft^3/min exhaust blower through a 28" diameter duct. This duct joins with the exhaust duct from the hot lab which extends into the 4-ft. diameter exhaust stack that is located high on a ridge overlooking the complex. The exhaust air flow is continuously monitored for indications of airborne radioactivity.

1.1.1.2 Hot Laboratory (Building 2)

The Hot Laboratory is a concrete structure 225 feet long by 57 feet wide by 37 feet high. It contains five hot cells, each having 4-foot thick walls of high density concrete (240 lbs/ft^3). The cells are separated from each other by 4-foot thick, high density concrete walls. A plan view of the Hot Lab is shown in Figure 1.2.

The cells are general purpose units designed to accommodate a variety of operations. A general description of the cells is presented below.

The internal dimensions of cell 1 are 16 feet wide by 10 feet long by 15 feet in height. This cell is equipped with a Remote Handling Arm (750 lb. capacity), one pair of heavy duty manipulators, and one pair of standard duty manipulators. Two 4-foot thick lead glass shielding windows are located in the front shielding wall of Cell 1.

Internally, cells 2, 3 and 4 are 6 feet wide by 10 feet long by 12 . feet in height. Cell 5 is 6 feet wide by 10 feet long by 25 feet in height. Cells, 2, 3, 4 and 5 are each equipped with a 4-foot thick lead glass shielding window and all cells are equipped with one pair of Master Slave Manipulators.

Major access to all the cells is possible through rear doors (7 feet wide by 6 feet high by 4 feet thick) which can be withdrawn utilizing electrical drives. Access to all cells is also possible via roof openings containing removable plugs. The roof and roof plugs of all cells are 3-1/2 foot thick magnetite concrete with a density of 240 lbs/ft^3.

A canal containing water connects Cell 1 with the reactor pool. Radioactive samples, specimens, isotopes, etc., can be transferred through this canal and brought into Cell 1 via a motorized elevator.

The area on the front side of the cells is the operating zone and is maintained as a clean area. The viewing windows, manipulator controls, intercell conveyor controls, and in-cell service controls are located in this area.

The charging area is located to the rear of the cells. Controls for the rear access doors to the cells are located in the cell operating area. Access to the decontamination room, exhaust fan room, waste treatment facility and conveyor loading station is from the charging area.

At the south end of the charging area, (cell rear door access area) swinging and sliding doors separate this area from the canal and the south loading dock. At the north end swinging doors separate the charging area from the waste drum storage area.

A fifteen ton capacity monorail hoist travels through this area from the north to the south loading dock. Heavy lifts can be maneuvered by this hoist throughout the charging area, north and south loading docks, gamma pit, waste processing area, and the waste drum storage area. A second 10-ton hoist can be operated remotely from the operating area.

A machine shop 29 feet by 25 feet is located in the south end of the Hot Lab. This shop contains a drill press, lathe, milling machine, band saw, electric and gas welding equipment, and a variety of hand tools.

The waste pit area, located at the north end of the hot laboratory building, consists of one hundred shielded individual waste drum storage cells (Figures 1.8 and 1.9). Each cell is 7 1/4 feet deep by 2 1/2 feet in diameter. The cells are arranged in a honeycomb fashion with additional shielding around the outer perimeter. Each cell has a removable shield plug and is internally vented to the hot cell exhaust ventilation filter system. Radioactive waste is stored in this facility prior to disposal.

The Hot Laboratory ventilation system (Figure 1.10) is designed to assure a continuous, positive flow of air from clean (non-radioactive) areas to contaminated areas. There are two major supply fans. One fan supplies 19,000 cu. ft/min of air to the first floor offices, loading dock, second floor offices, operating area, and the Radiochemical Lab. A second fan supplies 6,000 cu. ft/min of air to three laboratories on the second floor. The exhaust is continuously monitored for airborne radioactivity.

Liquid radwastes are collected in the 7200 gallon T1 tank installed in the existing underfloor space of Building 2. From the collection tank the liquid is pumped to a radwaste evaporator. The evaporator condensate is checked for radioactivity and recycled if radioactivity is found that exceeds regulatory free release requirements. See Figure 1.11 for the radioactive waste system diagram. This diagram is a general diagram and does not include all valves and piping.

1.1.1.3 Radioactive Waste Storage Building

The Radioactive Waste Storage Building is a single story concrete block building located 500 feet north of Building 2. It is approximately 24 feet x 60 feet and is utilized for storage of Class A waste, mostly 55 gallon drums.

1.1.1.4 Reactor Utilization

Planning and construction of the Cintichem Sterling Forest facility began in 1957. The 5 MW reactor had its initial criticality in 1961.

Total megawatt hours of usage from initial startup to final shutdown on February 9, 1990 is approximately 906,000 MW hours. Megawatt hours of reactor usage for the most recent ten years are listed below:

1990	4,615 MWH
1989	40,124 MWH
1988	39,076 MWH
1987	39,623 MWH
1986	40,380 MWH
1985	40,389 MWH
1984	40,007 MWH
1983	41,011 MWH
1982	38,858 MWH
1981	38,213 MWH

1.1.1.5 Utilities

Normal operating power is provided by a commercial utility company and it is a three-phase high voltage supply which is stepped down to 440 V by a dedicated transformer which is located in the motor control center in the Hot Laboratory.

Emergency power for the facility is provided by two auxiliary generators that are driven by internal combustion engines. When normal power is lost these auxiliary generators automatically start and when they attain their rated capacity an automatic transfer of the emergency load is accomplished. The buildings are also served by a non-radioactive sanitary system, 90 psig compressed air, 80 psig steam, natural gas, demineralized water, and city water.

1.1.2 Decommissioning Plan Synopsis

Cintichem has selected the DECON alternative for decommissioning the reactor, hot laboratory building, and Class A waste building. This option will require the removal of radioactive material to levels specified by NRC and NYSDOL. After removal of radioactivity, the buildings will be razed and the site backfilled. The estimated cost for implementing the DECON option is $20.5 million. These funds will be provided by Hoffmann-LaRoche. The schedule for performing the decommissioning is 26.5 months.

Cintichem Inc., intends to remove radioactivity within the reactor, hot lab building, and Class A waste building associated contaminated systems exterior to these buildings, and areas immediately adjacent to them, to permit unrestricted release and license termination. The decommissioning program will be conducted in two phases; (1) removal of contaminated and uncontaminated tools, equipment and fixtures and spent fuel under the current operating license authority and, (2) dismantling and/or decontamination of systems and structures under the plan provided herein. Phase One is currently underway and, in addition to removal of unneeded materials, detailed site radiological characterization and detailed decommissioning engineering is being conducted. The major decommissioning workscope will include:

- Removal of activated core structure, associated components and activated portions of the biological shield;
- Removal of contaminated equipment, components and fixtures;
- Decontamination of building structural surfaces;
- Removal of contaminated soil under and adjacent to the buildings;
- Removal of contaminated structures and equipment adjacent to and associated with the reactor and hot laboratory buildings;
- Perform Final Termination Radiological Survey to verify that radioactive material has been removed to NRC and NYSDOL release criteria;

- o Demolition of the remaining clean reactor and hot laboratory buildings to grade and backfilling the subsurface areas.

It is estimated that 128,000 cubic feet of radioactive waste will require disposal as a result of the decommissioning. Worker collective radiation exposure is estimated to be 366 person-rem with 75,545 exposure person-hours. Radiation exposure to the off-site general public is estimated to be less than 0.7 person-rem. This compares to a background radiation exposure of 2,595 person-rem for the same population.

1.2 Facility Operating History

The Cintichem facility, originally called the Union Carbide Nuclear Company (UCNC) Research Center, was originally designed and constructed to meet the joint needs of the Union Carbide Nuclear and Ore Companies. The facility, described earlier, consisted primarily of a 5MW pool type reactor linked via a four (4) foot wide by twelve (12) foot deep water filled canal to a bank of five adjacent and intercell-conveyor-connected hot cells.

Shortly after going into operation, the facilities operations expanded to provide various reactor based products and services to a multitude of research, production, medical, and educational groups outside the Union Carbide organization. In the late 1960's and early 1970's, the field of nuclear medicine grew and in 1978 the facility became the Medical Products Division of Union Carbide.

During the operating history of the Cintichem reactor, the following operational occurrences took place which could impact on decommissioning safety from a radiological standpoint:

1. From 1968 through 1972 the facility developed a process of separating isotopes from mixed fission products generated by irradiating enriched Uranium target capsules. By the late 70's, 20 to 30 capsules were being processed weekly and several isotopes such as Mo-99, I-131, and Xe-133 were being extracted for medical use. As an expected consequence of this production, the hot cells became contaminated with mixed fission products. Mixed fission products were also generated in the reactor primary cooling system due to cross contamination from the hot cells and tramp uranium in the coolant system. In addition, since the uranium electroplated targets were manufactured at the Cintichem facility, target preparation labs on the second floor of the hot laboratory were contaminated with U-235. This contamination did not cause operational problems.

2. In the mid 70's two events occurred which created a level of Ag-108/110m contamination throughout the reactor primary water system. During this period, the B_4C reactor control rods were replaced with AgInCd control rods to eliminate

the potential for a stuck rod accident. Activated silver from these new rods began leaching into the pool system. A second and unrelated event also began during this period. Algae began to grow in the reactor cooling system and was discovered plating out on pool surfaces and inside all primary piping. The algae had an affinity for the Ag-108/110m from the control rods and caused it to be trapped and concentrated inside the primary piping and pool surfaces. Although the activated silver leaching problem was brought under control in the late 70's and early 80's by replacing all control rods once again with a new set having a heavier outer nickel plating, the level of Ag-108/110m contamination is still noticeable throughout the system.

The floors and walls of the lower level pump room are expected to contain contamination from both Ag-108/110m and mixed fission products since these surfaces were frequently in contact with pool water. The inner surfaces of the heat exchanger and all primary piping is expected to contain algae residue that may contain small amounts of radioactive material that the algae removed from pool water.

3. The old evaporator room and T1 room floors and walls are expected to be contaminated with a combination of long lived isotopes from different sources. The old evaporator was used to process contaminated waste water until the mid 70's. Repairs and spills from this system would have contributed to contamination in these areas. The same is true of the T1 room since this is where all site radioactive waste water has always been collected for processing.

4. Late in 1989, an underground leak was discovered in the main hot cell bank ventilation system. Since this air contained contaminants from the hot cells, mixed fission products have been discovered in the soil. It is possible that bedrock, is also affected in the vicinity of this underground duct work.

5. Early in 1990, it was discovered that primary water leaks existed in the pool systems' hold up tank, canal, and gamma facility. Leaks had been discovered and repaired a few years earlier in the hold up tank. As a result of these leaks, it is anticipated that low levels of contamination may exist in the vicinity of the drainage system below the reactor building, areas immediately outside of the hold up tank, soil outside the holdup tank and pump room, soil outside of the canal and gamma pit structures, the hot lab building footings, and the north wall area of the reactor building.

Based on preliminary data from the site radiological characterization program, discussed in Section 1.3 of this plan, significant levels of Ag-108/110m and mixed fission products have been identified in algae removed from the internals of the reactor primary system heat exchanger. Mixed fission products have also been identified in sediment samples taken from hot cell number one.

1.3 Current Radiological Status of Facility

1.3.1 Introduction

A detailed site radiological characterization program, including activation analysis, was performed for the Cintichem reactor building, hot laboratory building and the immediately surrounding environs. The purpose of this characterization program centers on the need to obtain specific radiological data concerning areas of the plant that may have become internally or externally contaminated or activated during the reactor and hot lab operating history. This data is necessary for detailed decommissioning planning purposes, determination of effective and appropriate decontamination and dismantling techniques. This data is also needed for planning radioactive material disposal, assessing potential hazards during decommissioning and determining ALARA controls. The scope of this program addressed:

- o Structural surface contamination and area dose rates;
- o System and component contamination and contact dose rates;
- o Reactor structure and component activation;
- o Subsurface contamination;
- o Outdoor contamination (soil outside of the buildings).

The characterization program obtained the following information:

- o Locations, areas, and contamination levels on structural surfaces;
- o Depth of contamination penetration into surfaces;
- o Location, volume, and activity levels of contaminated soil;
- o Location, dimensions, volume and activity levels of contaminated equipment, ducts, fixtures, etc.;
- o Area and component radiation dose rates;
- o Calculation of activity levels induced by activation in the stall/pool structure and associated reactor components.

This characterization effort was started in June of 1990 and was completed in fall of 1990.

Based upon characterization data, the following items have been judged to be the major sources of radioactive material that will be dealt with during the decommissioning process:

Contaminated Structures

- Reactor biological shield (contaminated and activated);
- Miscellaneous reactor building surfaces (contaminated walls, floors, etc.);
- Primary water holdup tank and underlying soil/bedrock surfaces;
- Hot cells and surrounding support areas;
- Class "B" waste storage pits;
- T-1 and evaporator rooms;
- Uranium plating and solution labs;
- Miscellaneous hot lab building surfaces (floors, walls, etc.).

Contaminated Systems and Equipment

- Primary reactor cooling;
- Primary reactor cooling system purification;
- Reactor building air exhaust;
- Hot lab building air exhaust;
- Buried hot cells air exhaust;
- Exterior reactor and hot lab building exhaust air discharge ducting and stack;
- Waste water evaporator system and building waste collection drains.

Activated Components

- Reactor core support tower;
- Reactor core grid plate;
- Plenum;
- Core outlet assembly;
- Beam tubes;

- o Pneumatic rabbits;
- o Thermal column and thermal column shield assembly;
- o Activated biological shield concrete.

Subsurface Building Contamination

- o Exterior to below grade holdup tank and pump room;
- o Fill surrounding and beneath buried hot cell exhaust ducts and filter room;
- o Exterior to below grade T-1 and evaporator rooms;
- o Exterior to canal and gamma pit structures;
- o Hot Lab building footings (assumed);
- o North wall area reactor building (assumed).

Outdoor Contamination

- o Two 5,000 gallon buried holdup tanks (mall tanks);
- o Miscellaneous buried piping and associated collection basins.

1.3.2 Contaminated Structures

This section summarizes the radiological contamination status of general structural surfaces and area exposure rates within the reactor and hot lab buildings that have been preliminarily identified as contaminated. Additional minor contaminated areas are expected to be found within the facility and will be characterized by the ongoing characterization program.

General Structural Surfaces

As part of the ongoing characterization program the reactor and hot lab buildings were divided into manageable survey areas based upon the likelihood of similar contamination levels, deposition patterns and radionuclide mix. The location of these survey areas where structural contamination was found is shown in Figure 1.12. Table 1.1 presents a preliminary summary of the total surface direct beta contamination levels, removable beta and alpha contamination levels and general area exposure rates at one meter above the floor for areas shown to be contaminated, in Figure 1.12. Each set of determinations (mean and range values) was based upon 30 evenly spaced sample points within each sample population. In general, the highest total surface beta contamination levels were found on the charging area floor behind the hot cells (up to 533,000 dpm/100cm^2), in the hot cell conveyor station area (up to 420,000 dpm/100cm^2, in the transfer canal (up to 1,634,000 dpm/100cm^2), on the reactor pool surfaces (up to 621,000 dpm/100cm^2) and in the lower pump room (up to 242,000 dpm/100cm^2). The removable fraction of this surface contamination is generally less than five percent.

General building area (exclusive of the uranium labs), alpha contamination (as found on smear samples) was either nonexistent or very low, with the highest single removable alpha contamination level, of 4,310 dpm/100cm^2, being found on the transfer canal wall. As would be expected, highest surface contamination levels within an area were generally found on the floor and on miscellaneous horizontal surfaces.

Generally accessable area gamma dose rates, measured one meter above the floor surface ranged from background up to 3,600 uR/hr. The radionuclide mix was found to vary between the reactor and hot lab buildings. Table 1.2 presents a summary of gamma isotopic ratios for composite smear samples that were selected from the various contaminated areas, shown in Figure 1.12. It can be presumed that Sr-90 activity will be present in equal proportions to that of Cs-137.

As can be seen from Table 1.2, the isotopic mix comprising surface contamination varies considerably by location. The overall predominant radionuclides, in descending order are: Ce^{144}, Cs^{137}, K^{40}, Zr^{95}, Co^{60}, Nb^{95}, Ru^{103}, Zn^{65}, and Ag^{108}. Trace amounts of Sc^{46}, Mn^{54}, Co^{57}, Co^{58}, Sn^{113}, Sb^{124}, Sb^{125}, Cs^{134}, Eu^{152}, Eu^{155} and Ir^{192} are also occasionally present. In general, as can be expected, activation products are more prevalent in the reactor building and fission products more predominant when approaching the hot cell areas.

Hot Cells

The hot cells (#1 to 4) located in the hot lab building were found to have internal exposure rates of 100, 200, 1,000, and 60R/hr respectively, as of July 1990. Exposure rates in cell #5 are currently not available due to a clouded window. A small piece of a smear sample representing (1 cm^2) taken inside of hot cell #1 (7-16-90) was found to have 1.2uCi of Nb^{95}, 0.59uCi of Zr^{95}, 1.27uCi of Mo^{99}, 0.12uCi of Ru^{103}, 0.0041uCi of Sb^{125}, 0.0062uCi of Cs^{134}, 0.067uCi of Cs^{137}, 1.33uCi of Ce^{144}, 0.13uCi of Ir^{192} and 1.53uCi of Tc^{99m}. Obviously, only the Cs^{134}, Cs^{137} and Ce^{144} will be significantly present at the start of decommissioning due to radioactive decay of the short half-life radionuclides. Due to the very high radiation levels in the cells, it is impossible to determine surface contamination levels. However, a conservatively high estimate, based on the data above would be 2.2 x 10^9 dpm/100 cm^2 total surface contamination. The estimate is based on 6 months of decay with Sr-90 activity assumed equal to Cs-137, and a 10% smear retention factor.

Other Areas

Radioactive contamination has been preliminarily determined to be present beneath portions of the reactor and hot lab buildings. This contamination consists of soil, rubble backfill or possibly the underlying bedrock. These contaminated areas exist adjacent to the buried holdup tank, under or around the canal/gamma pit, and in soil surrounding the buried hot cell exhaust system and the subsurface T-1/evaporator rooms. This subsurface contamination is discussed in Section 1.3.5. As can be presumed, the backsides and/or undersides of structures in these areas may also be contaminated.

1.3.3 Contaminated Systems and Equipment

This section summarizes the preliminary radiological status of the systems and equipment within the reactor and hot laboratory facilities. Contact gamma dose rates were taken on each component and at periodic sections of pipe or duct runs. Components were opened for direct internal measurement of contamination levels where non-destructive access could be obtained. Samples of sediment were obtained when found inside of opened components and analyzed by gamma spectroscopy.

Primary Reactor Cooling System

Twenty contact gamma dose readings have been taken on the primary reactor cooling systems and equipment. The external gamma dose rates ranged from 14uR/hr to 2,900uR/hr on contact, with an average dose rate of 1,060uR/hr. Two components, the primary side of the heat exchanger and the primary pump were opened for direct measurement of internal contamination levels within the primary system. Respectively, 337,000 and 809,000 dpm/100 cm^2 beta was found. The secondary side of the heat exchanger was opened for direct contamination measurement to determine if primary to secondary leaks had occurred. Less than 1000 dpm/100 cm^2 beta was found when the primary side of the heat exchanger was opened and a sample of the internal sediment was taken along with a direct beta contamination measurement. This sediment sample was analyzed by gamma spectrometry. The radionuclides comprising this internal contamination and their relative contributions were found to be; Sc^{46}, 4%; Co^{60}, 12%; Zn^{65} 8%; Nb^{95}, 22%; Ag^{110m}, 10%, Ce^{144}, 34%; Eu^{152}, 2%; Ir^{192}, 3%; and the following radionuclides at less than 1% each; Mn^{54}, Fe^{59}, Ru^{103}, Ag^{108}, Sn^{113}, Sb^{124}, Sb^{125}, Cs^{134}, and Cs^{137}. This isotopic mix was determined to be present as of 27 July 1990.

Primary Reactor Cooling Purification System

Seven gamma dose readings have been taken on the primary reactor cooling purification system and equipment. The external gamma dose rates ranged from 500uR/hr to 4,000uR/hr on contact, with an average gamma dose rate of 1,600uR/hr. A flange cover plate was removed and a direct beta contamination measurement taken on the back side of the plate. A contamination level of 13,000 dpm/100 cm^2 beta was found.

Reactor Building Air Exhaust System

Twenty-one gamma dose readings have been taken on the reactor building air exhaust system. The external gamma dose rates ranged from 21uR/hr to 1,400uR/hr on contact, with an average gamma dose rate on contact of 340uR/hr. Two components were opened for direct measurement of internal surfac contamination. The total surface direct beta contamination levels were 1600 dpm/100cm^2 and 1900 dpm/100cm^2.

Hot Laboratory Building Air Exhaust System

Sixteen gamma dose readings have been taken on the hot laboratory building air exhaust system. The external gamma dose rates ranged from 34uR/hr to 2,200uR/hr on contact, with an average gamma dose rate on contact of 460uR/hr. Four exhaust system components were opened for direct measurement of internal surface contamination. Total surface direct beta contamination levels ranged from 1100 dpm/100cm^2 to 20,400 dpm/100cm^2.

Hot Cell Air Exhaust System (up stream of polishing filters)

One gamma dose reading has been taken on the hot laboratory building hot cell air exhaust system at a point down stream from the first filter bank, where surrounding soil had been previously excavated. The external gamma dose rate was 235uR/hr on contact. This system is in operation and therefore, the internal surfaces of the system were not characterized, however the internal contamination of this system can be expected to be similar to the inside of the hot cells.

Exterior Air Discharge Duct and Stack

Ten gamma dose readings have been taken on the building air exhaust systems located exterior to the reactor building and the hot laboratory. The external gamma dose rates ranged from 10uR/hr to 30uR/hr on contact, with an average gamma dose rate on contact of 18uR/hr. The base of the exhaust stack was opened for direct measurement of internal surface contamination. A total surface direct beta contamination level of 75,700 dpm/100cm^2 was found.

A small sediment sample was obtained from the base of the stack. Non-quantitative results (July 27, 1990) indicates the following proportions of gamma emitters being present in the stack and duct work: K-40, 80%; Co-60, 7%; Cs-137, 6%; Ag-108%, 4%; Ag110m, 2% and; Cs-134, Mn-54, Zn-65 and Nb-95 comprising the remaining 1%.

It can be presumed that Sr-90 is present in amounts approximately equal to that of Cs-137.

Waste Water Evaporator System

Systems and equipment within the T-1/Evaporator room were characterized for typical external contact exposure rate only because the system is in active use. The emergency surge tank was found to have a gamma dose rate of 200mR/hr to 500mR/hr. The equipment within the evaporator room was shown to generally exhibit a gamma dose rate of approximately 10mR/hr on contact.

Storage Wells Air Exhaust System

Three gamma dose rates were obtained on the storage wells air exhaust duct located within the crawl space beneath the radwaste laydown/shipping area. This duct discharges to the first filter bank of the hot cell exhuast system. The external gamma dose rates ranged from 5900uR/hr to 9500uR/hr on contact, with an average gamma dose rate on contact of 6800uR/hr. A 900uR/hr dose rate was found on the exhaust duct header sump. The sump was opened for direct beta contamination measurement. A contamination level of 132,700 dpm/100 cm2 was found.

1.3.4 Activated Components and Structures

Isotopic concentrations, curie content and radiation dose rates have been estimated for components and structures that received neutron irradiation during operation of the Cintichem reactor[1]. A neutron activation analysis was performed to provide an estimate of the neutron-induced radioactivity in core components, structures and the surrounding bioshield walls. The components that were addressed in this study are:

- Reactor core support tower;
- Reactor grid plate (and locator pins);
- Plenum;
- Core outlet assembly;
- Beam tubes;
- Thermal column and thermal column lead shield assembly;
- Pneumatic rabbit assembly;
- Concrete bio-shield.

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[1] "Radionuclide and Dose Rate Analysis for Neutron-induced Radioactivity in the Cintichem Reactor", prepared by TLG Engineering, Inc., October 198[illegible]

1.3.4.1 Approach

The calculated curie contents in these neutron-activated components were used as input to point-kernal shielding calculations, at various locations of interest, to determine radiation dose rates. These calculations were performed to provide data to support conceptual decommissioning engineering. For actual decommissioning work, calculational results will be benchmarked with actual gamma dose rate measurements and/or sampling to verify or refine these estimates prior to use for safety related purposes.

The ORIGEN-2[2] computer code was used to perform the neutron activation and depletion calculations. ORIGEN-2 was obtained through Oak Ridge National Laboratory's Radiation Shielding Information Center. Input to the code included material composition, the Cintichem reactor operating history, and calculated neutron fluxes. Wherever possible, data obtained from Cintichem (structural drawings, operating history, material compositions, etc.) were used. The balance of input data was obtained from available literature and is referenced in Radionuclide and Dose Rate Analysis for Neutron-Induced Radioactivity in the Cintichem reactor. Where needed by lack of available information, conservative assumptions were made.

The MICROSHIELD[3] computer code was used to calculate radiation dose rates at locations of interest. Some of the calculated dose rates will be used to benchmark the activation analysis results by comparing the MICROSHIELD calculations to actual measurements made in the field.

The radionuclides considered to be "of interest" in the activation analysis were:

- Those radionuclides listed in Tables 1 and 2 of 10CFR61, which establish radiological limits for waste that can be disposed of in a shallow land burial site.
- Those radionuclides which contribute greater than 1% towards the specific and total curie contents of individual components.
- Those radionuclides which contribute significantly to the total gamma flux emitted by the individual components.

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[2] ORNL/TM-7175, A User's Manual for the ORIGEN-2 Computer Code, Groff, A.G., Oak Ridge National Laboratory, July 1980

[3] MICROSHIELD 3 Manual, Software version 3.12, Grove Engineering, October 1987

1.3.4.2 Results

The following Table 1.3 presents a summary of the significant findings from reference (1). Included are a listing of the activiated items, material composition, material masses, total specific activities and total curies. Table 1.4 presents the calculated radionuclide inventory by individual radionuclide for each major activated item listed in Table 1.3. Table 1.5 presents the calculated radiation dose rates, from the individual items, at one meter in air and water.

1.3.5 Subsurface Building Contamination

Based upon events described in Section 1.2 a subsurface soil investigation was conducted in the areas where soil contamination could have occurred. Concrete core holes were made in the South wall of the reactor pump room East of the holdup tanks East exterior wall; in the South and East walls of the holdup tank; in the walls of the gamma pit/canal; and in the various walls of the subsurface T1 room which is adjacent to the buried hot cell exhaust duct. Additionally, a section of floor and soil was removed from above the buried exhaust duct as it leaves the filter room, thereby exposing the duct and nearby soil. At locations where soil was present, 400 to 500 grams of soil were removed for analysis.

At locations where soil was not present gross gamma NaI readings were taken to observe relative differences inside of each hole in an effort to screen for soil contamination.

From the vicinity of the hot cell exhaust system and the T-1/evaporator rooms three soil samples were obtained from core holes in the subsurface structures, and soil samples taken from 23 barrels of soil that had been removed from the vicinity of the buried exhaust duct. Soil removed from the core holes in the T-1/evaporator room ranged from 9 to 864 pCi/gm. The sample with the greatest activity was comprised of Ce-144, 69%; Cs-137, 26%; Co-60, 1%; and Nb-95, Ru-103, Sb-125, Cs-134 and Ir-192 the remaining 4% (August 10, 1990). Soil from the 23 barrels of soil removed from the vicinity of the hot cell exhaust duct ranged from no detectable activity up to 11,030 pCi/gm, with the activity comprised of Cs-137, 84%, Cs-134, 3%; Co-60, 2% and Ce-144, 11%. It is presumed that Sr-90 activity would be approximately equal to that of the Cs-137.

Seven soil samples were obtained from 7 of the 16 core holes through the walls of the canal/gamma pit structure. Of these, six had less than one pCi/gm of activity and one had 4 pCi/gm of activity (in the gamma pit). Gross gamma NaI readings in the 16 core holes indicate the possible presence of radioactive contamination in the location of five holes.

1.4 Decommissioning Alternative

In order to restore the areas now occupied by the reactor and hot lab facility to an unrestricted industrial use, Cintichem has selected the DECON alternative for decommissioning. This option will require the removal of radioactive material to levels specified below. The buildings will be razed and the site backfilled. Other decommissioning alternatives were considered inappropriate. It is Cintichem's intent to decommission the facility to allow termination of the NRC and New York State licenses.

The decommissioning plan presented here includes measures to reduce radiation to levels that permit unrestricted use as follows:

- o Removal of all radioactive materials, soil, neutron activated components and structures to meet the acceptance criteria of 5 uR/hr above background at 1 meter from the surface or 10 mrem/year above background, considering reasonable proximity and occupancy;

- o Decontamination of equipment, components and surfaces within the limits for acceptable residual surface contamination levels for unrestricted use as set forth in USNRC Regulatory Guide 1.86 or 12 NYCRR38, whichever is more limiting.

1.5 Decommissioning Organization and Responsibilities

The project organization for decommissioning the nuclear facilities is presented in Figure 1.13. The organization includes personnel with experience in reactor operation, radiochemical separations, health physics and environmental monitoring. It is intended to retain key personnel from the former facility operating staff to manage and implement many of the decommissioning tasks. A contractor with knowledge and experience in reactor decommissioning projects shall be retained as project co-manager. The minimum qualifications for safety related positions are given in Appendix A. Ultimate responsibility for decommissioning activities rests with Cintichem.

1.5.1 Cintichem LEVEL I Management

Cintichem, Inc. is a subsidiary of Hoffmann-La Roche (HLR). The Cintichem Plant Manager reports to the Sr. Vice President Finance, Human Resources and Administration of HLR. The Cintichem Plant Manager is responsible for the general management of all operations on site (i.e., Decommissioning, Conventional Plant Operations, Radiopharmaceutical Manufacturing Operations). The Decommissioning Project Manager, the Manager of Health Safety and Environmental Affairs and the D&D Consultant (Project Co-manager) report to the Plant Manager. The Plant Manager, D&D Project Manager, Manager of Health Safety and Environmental Affairs and the D&D consultant are at LEVEL I management of the project.

1.5.1.1 Plant Manager

The Plant Manager is directly responsible to Corporate Management and retains ultimate responsibility for all site activities, decommissioning and production. The Plant Manager sets general policies for decommissioning which are implemented by his subordinate, the Project Manager. The Plant Manager is the contact point for contested items from Quality Assurance, Health Safety and Environmental Affairs and Security. He also provides necessary liaison with regulatory agencies and the Nuclear Safeguards Committee.

1.5.1.2 Project Manager

The Project Manager shall have direct responsibility for the decommissioning project. During the decommissioning phase, he shall also be responsible for compliance with all NRC, NYSDOL, and other pertinent Federal, State, and local regulations. He shall direct all decommissioning activities including licensing, procurement, contractor activities, engineering, site services, Quality Assurance, and Security.

1.5.1.3 Manager of Health Safety and Environmental Affairs

The Manager of Health Safety and Environmental Affairs reports to the Plant Manager. He shall be responsible for implementation of the radiation protection, the industrial safety and the environmental protection programs. He shall implement these programs to assure that radiation exposures are maintained as low as reasonably achievable.

1.5.1.4 Project Co-manager (D&D Consultant)

The Project Co-manager shall be an organization capable of providing expertise and direct assistance in the areas of decommissioning, radiological health, environmental monitoring, engineering, radioactive waste handling and regulatory affairs. The Project Co-manager shall have a representative on site during most of the decommissioning work. Other qualified individuals from the D&D Consultant's organization shall be brought in as necessary to perform tasks where the Cintichem Staff needs assistance.

1.5.1.5 Nuclear Safeguards Committee

The Nuclear Safeguards Committee shall be organized and will function under the charter presented in Appendix "B". The Committee will generally review operations, procedures, proposed facility modifications and incident reports to assure that operations are conducted safely, that the occupational and non-occupational exposures are maintained ALARA and that physical facilities are preserved where it is necessary and feasible.

1.5.1.6 Radiation Safety/ALARA Committee

The Radiation Safety/ALARA Committee (RS/AC) shall be organized and will function under the Charter presented in Appendix C. The Committee reviews problem areas and/or operations for ways to reduce occupational radiation exposures. Planning and process procedures are under the surveillance of the Nuclear Safeguards Committee (NSC). ALARA considerations are formulated by the NSC and the RS/AC. ALARA for new projects is handled by the NSC, while the RS/AC is concerned with on-going operations.

1.5.2 Cintichem LEVEL II Management

The LEVEL II Management will consist of departments involved in carrying out the decommissioning operations. These Managers shall report to either the Decommissioning Project Manager, the Manager of Health Safety and Environmental Affairs, or the Project Co-manager as appropriate.

1.5.2.1 Staff Health Physicist

The Staff Health Physicist shall be responsible for monitoring results of routine surveillance programs, and program evaluation through special investigations or additional surveillance to assure the occupational and off-site exposures are maintained ALARA.

1.5.2.2 Decommissioning and Demolition Superintendent

The D&D Superintendent shall direct and coordinate specific decommissioning activities performed by the workforce. He shall assure that all decommissioning activities are carried out according to procedures and applicable requirements of the radiation safety program.

The D&D Superintendent shall be directly responsible for the safety and actions of all personnel involved in the dismantling activities. He shall be knowledgeable in the systems being dismantled and in dismantling techniques.

1.5.2.3 Manager of Waste Disposal

The Manager of Waste Disposal shall be responsible for processing gaseous, liquid and solid wastes generated by the decommissioning effort. He shall assure the gaseous effluent filtration systems are maintained within operating limits. He shall assure that liquid waste effluent is properly treated to assure discharges are ALARA. He shall assure that solid waste is packaged and disposed of properly and within applicable regulatory limits. Mixed waste shall be identified and disposed of appropriately.

1.5.2.4 Site Services Manager

The Manager of Site Services shall be responsible for providing support to the decommissioning effort in the areas of utilities, shop, electrical, instrumentation and security services.

1.5.2.5 Quality Assurance Manager

The Quality Assurance Manager shall maintain surveillance of project activities and perform audits as required to assure that decommissioning tasks are performed in accordance with applicable procedures, license conditions and regulatory requirements. The Quality Assurance Manager shall be the primary liaison with regulatory agencies for routine reporting and licensing applications.

1.5.3 Cintichem LEVEL III Management

The Cintichem LEVEL III Management shall consist of first line supervisors, shift foremen, and shop supervisors who shall be responsible for carrying out decommissioning tasks, radiological surveillance, environmental surveillance, waste processing, waste disposal and support functions on site.

1.5.4 Contractor Assistance

Several contractor assistance roles have been identified but actual selection of contractors, with the exception of Technical Support contractors, will take place after this plan is approved. Cintichem currently envisions the following contractor positions:

1. Technical Support Contractor to act as decommissioning Co-Managers assisting in the engineering, planning, and scheduling of decommissioning activities and enhancing the continuity of the overall effort.

   TLG Engineering, Inc.
   148 New Milford Road East
   Bridgewater, Connecticut 06752

2. Health Physics Support Contractor(s) to provide assistance in areas such as personnel monitoring, off site laundry services, final radiation survey, and other health physics tasks.

3. Radioactive waste contractors for:

   o Type A and B waste transportation;

   o Radioactive waste disposal or storage;

   o Radioactive waste recyclers;

- o Radioactive waste water treatment.

4. Heavy equipment operators such as excavators and lift operators for the removal or relocation of structural components, equipment, etc. as required by the plan.

5. Concrete contractors for:

   - o Concrete coring;

   - o Removal of activated and contaminated concrete via processes such as track drilling (and expanding grout), hydrolasing, demolition, and scabbling;

   - o Demolition of the hot lab and reactor buildings.

6. Asbestos contractors to assist in the removal and disposal of asbestos and/or to provide Cintichem employees with the necessary training to perform or assist in these tasks.

Schedules and specific tasks to be performed by contractors will be planned in advance and detailed work procedures written for significant operations. In addition, prerequisites such as health and safety precautions, protective clothing requirements, etc. will be defined in writing before work is started.

1.6 Regulations, Regulatory Guides and Standards

The Cintichem Decommissioning Plan has been written using the guidance and format specified in the NRC's "Guidance and Discussion of Requirements for an Application to Terminate a Non-Power Reactor Facility Operating License - Revision 1". The acceptance criteria used for the monitoring of unrestricted use will be as set forth in NRC Regulatory Guide 1.86 titled "Termination of Operating Licenses for Nuclear Reactors" or 12 NYCRR38, whichever is more limiting. These are the main documents Cintichem will utilize for its decommissioning effort. Other regulations, regulatory guides and standards Cintichem will utilize are listed below.

<u>Code of Federal Regulations</u>

10 CFR Part 19	Notices, Instructions and Reports to Workers; Inspections
10 CFR Part 20	Standards for Protection Against Radiation
10 CFR Part 30	Rules of General Applicability to Domestic Licensing of Byproduct Material
10 CFR Part 50	Domestic Licensing of Production and Utilization Facilities

10 CFR Part 51	Licensing and Regulatory Policy and Procedures for Environmental Protection
10 CFR Part 61	Licensing Requirements for Land Disposal of Radioactive Waste
10 CFR Part 70	Domestic Licensing of Special Nuclear Material
10 CFR Part 71	Packaging of Radioactive Material for Transport and Transportation of Radioactive Material under Certain Conditions
10 CFR Part 140	Financial Protection Requirements and Indemnity Agreements
29 CFR Part 1910	Occupational Safety and Health Standards
29 CFR Part 1926	Occupational Safety and Health Standards for Construction
49 CFR Parts 170 - 199	Department of Transportation Hazardous Material Regulations

NRC Regulatory Guides

4.15	Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment
8.2	Guide for Administrative Practices in Radiation Monitoring
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters
8.7	Occupational Radiation Exposure Records Systems
8.9	Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposure As Low As Reasonably Achievable
8.13	Instruction Concerning Prenatal Radiation Exposure
8.15	Acceptable Programs for Respiratory Protection

ANSI Standards

ANSI N13.13	Control of Radioactive Surface Contamination of Material, Equipment and Facilities to be Released for Uncontrolled Use (Draft)
ANSI Z88.2-1980	Practices for Respiratory Protection
ANSI N13.1	Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities
ANSI N323-1977	Radiation Protection Instrumentation Test and Calibration
ANSI/ANS 15.10-1981	Decommissioning of Research Reactors

Informal Guidance and Technical Reports

NUREG/CR 1756	Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors and addenda.
NUREG/CR 2082	Monitoring for Compliance with Decommissioning Termination Survey Criteria
NUREG 0586	Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities
NUREG/CR 2241	Technology and Cost of Termination Surveys Associated with Decommissioning of Nuclear Facilities

Agreement State Rules

State of New York Department of Labor Industrial Code Rule 38, "Ionizing Radiation Protection"

New York State Department of Labor Law Article #30

12 NYCRR Part 56

New York State Department of Health Title 10 Part 73

1.7 Training and Qualification

Training and qualifications requirements of personnel involved with the Cintichem facility decommissioning will be provided primarily by the Health Physics and Safety Department and the appropriate managerial staff. The training will be provided in Health Physics, Industrial Safety, and good work practices. The training records will be maintained by the Health Physics and Safety Department.

1.7.1 Radiation Safety Indoctrination

The basic goal of the Cintichem Radiation Safety Indoctrination program is to educate new employees and contractors as to the potential hazards and risks associated with occupational exposures to radiation and to teach them how to deal with the problems present in the work environment. This is accomplished through an extensive training period with Health Physics, followed by job specific training from the worker's supervisor.

All decommissioning workers will spend from one to five days with Health Physics, depending on their past experience and the degree to which their position involves working with radioactive materials and in radiation fields. An outline of the issues covered is in the "Check List for Health Physics Employee Indoctrination" (HP-A-06). As each item is covered, it is dated and initialed by the technician who provided the training. Once the Health Physics staff feels that the decommissioning worker has an adequate grasp of the potential hazards of, and their responsibilities regarding work with, radioactive materials and radiation fields they will be allowed to perform decommissioning work.

Some of the areas where training instruction is given are listed below.

- Instruction in Understanding Radiation;
- Instructions Concerning the Hazard and Risks from Occupational Exposure to Radiation;
- Instruction on Basic Radiation Physics;
- Instruction on Radiation Quantities and Units;
- Instruction on Time, Distance, and Shielding;
- Instruction on Contamination Control;
- Instruction on State and Federal Regulations 10CFR19, 10CFR20, 12 NYCRR38;
- Instruction on Health Risks to the Unborn;
- Instruction in Use of Radiation Instruments;
- Instruction in Use of Contamination Instruments;
- Instruction in Use of Anti-C Equipment;

- Instruction on Emergency Procedure - Access Control to Buildings 1 and 2.

Following the above training, instruction videos are viewed discussing "Risks Associated with Occupational Radiation Exposure", "Frisk Film", and "Access Procedures". A training quiz is then given.

After the training is completed, the decommissioning worker is issued a Radiation/General Safety Manual, film badges, and given a facility tour.

The following are Cintichem's Training and Qualification Procedures:

o HP-A-01	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable (HP/30E);
o HP-A-03	Personnel Radiation Protection (HP/33E);
o HP-A-04	Instructions Concerning the Hazard and Risks From Occupational Exposure to Radiation (HP/34E);
o HP-A-05	Health Risk to the Unborn Resulting From Radiation Exposure and Requirements for Notifying Management of Pregnancy (HP/156E);
o HP-A-09	Health Physics Technician Qualifications/Training Checklist (HP/159E);
o HP-A-10	Health Physics Technician Training Program (HP/160E).

Annual retraining for all radiation workers reinforces this initial training. This consists of instruction videos on radiation protection and relative risk which includes question and answer periods. This combination of indoctrination and periodic retraining provides an excellent means of maintaining an acceptable level of worker training and awareness with regard to work with radioactive materials.

1.7.2 General Safety Indoctrination

It is Cintichem's policy to provide all Cintichem decommissioning workers with the safest possible work environment and to insist on adherence to safe work practices.

To accommodate this all new decommissioning workers will be provided with initial training by the Cintichem Industrial Safety Specialist. This training is documented on the "Safety Indoctrination Form GSM-23".

Safety training will incorporate the following topics:

- Site safety rules and regulations;
- Emergency procedures;
- Reporting of injuries;
- Emergency equipment;
- Personal protective equipment;
- "Hazard Communication" Law and hazardous material handling;
- Safe work practices.

Cintichem

TABLE 1
Summary of Structural Co

AREA No.	Name	DOSE RATE uR/hr	BETA CONTAMINATION Direct Mean	Direct Range	R Mean
3-1	Cell Plug Area Floor	30-90	3200	0-24700	42
3-2	Cell Plug Area Lower Wall	N/A	460	0-9950	7
4-?	Upper Level Hot Lab Floor	ND	2100	0-12800	40
4-2	Upper Level Hot Lab Lower Wall	N/A	140	0-1100	5
4-3	Upper Level Hot Lab Upper Wall	N/A	30	0-520	4
4-4	Upper Level Hot Lab Ceiling	N/A	40	0-550	
6-1	Waste Pit Floor	150-3600	30000	0-135000	143
6-2	Waste Pit Lower Wall	N/A	620	0-1930	16
6-3	Waste Pit Upper Wall	N/A	800	0-4510	3
6-4	Waste Pit Ceiling	N/A	270	0-1840	2
6-5	Waste Pit Horizontal Surface	N/A	10330	0-69000	93
7-1	Shipping Area Hot Lab Floor	0-1340	22000	0-323000	54
7-2	Shipping Area Hot Lab Lower Wall	N/A	1740	0-20320	19
7-3	Shipping Area Hot Lab Upper Wall	N/A	13200	0-362000	12
7-4	Shipping Area Hot Lab Ceiling	N/A	1770	0-29360	2
7-5	Shipping Area Hot Lab Horizontal Surface	N/A	2340	0-8320	69
9-1	Charging Area Floor	490-3090	83620	0-533000	155
9-2	Charging Area Lower Wall	N/A	1320	0-16060	1
9-3	Charging Area Upper Wall	N/A	1030	0-12170	4
9-4	Charging Area Ceiling	N/A	710	0-4400	
9-5	Charging Area Horizontal Surface	N/A	7650	0-19320	40
10-1	Conveyor Station Hot Cell Floor	ND	38230	1900-420000	66
10-2	Conveyor Station Hot Cell Lower Wall	N/A	3900	750-26000	40
10-3	Conveyor Station Hot Cell Upper Wall	N/A	90	0-1160	6
12-1	Exhaust Fan Room Floor	70-180	7370	2230-21810	54
12-2	Exhaust Fan Room Lower Wall	N/A	610	0-8670	10
12-2	Exhaust Fan Room Upper Wall	N/A	200	0-1150	9
15-1	Oper Area Manipulator Enc Floor	130-1890	16000	0-187000	306
15-2	Oper Area Manipulator Enc Lower Wall	N/A	3750	0-18570	32
15-3	Oper Area Manipulator Enc Upper Wall	N/A	80	0-760	6
15-4	Oper Area Manipulator Enc Ceiling	N/A	150	0-870	
18-1	Canal Floor	310-1990	206100	0-553000	695
18-2	Canal Lower Wall	N/A	296700	5700-1634000	1162
21-1	Target Welding Shop Floor	10-35	0	0-0	
21-2	Target Welding Shop Lower Wall	N/A	70	0-1090	3
21-3	Target Welding Shop Upper Wall	N/A	60	0-1120	3
21-4	Target Welding Shop Ceiling	N/A	50	0-1400	1
21-5	Target Welding Shop Horizontal Surface	N/A	0	0-0	
21-6	Target Welding Shop	N/A	65	0-840	
23-1	Old Reactor Cave Area Floor	230-2990	5200	0-31590	18
26-1	Reactor 3rd Floor Floor	10-990	1320	0-5730	2
26-2	Reactor 3rd Floor Lower Wall	N/A	1600	0-18000	1
26-3	Reactor 3rd Floor Upper Wall	N/A	140	0-2140	
26-4	Reactor 3rd Floor Ceiling	N/A	460	0-4730	
26-7	Reactor Pool Gutter	ND	251000	63970-550000	N
26-8	Reactor Pool Walls	ND	485000	323000-621000	N
30-1	781.12' Elev Reactor Area Floor	60-1090	990	0-9580	18
30-2	781.12' Elev Reactor Area Lower Wall	N/A	1100	0-21760	1
30-3	781.12' Elev Reactor Area Upper Wall	N/A	100	0-560	2
32-1	Lower Pump Room Floor	240-790	52280	0-242000	95
32-2	Lower Pump Room Lower Wall	N/A	480	0-2140	4
32-3	Lower Pump Room Upper Wall	N/A	730	0-5050	2
32-4	Lower Pump Room Ceiling	N/A	190	0-1810	
32-5	Lower Pump Room Horizontal Surface	N/A	5790	0-54280	
33-1	Upper Pump Room Floor	210-1990	3090	0-17030	
33-2	Upper Pump Room Lower Wall	N/A	1750	0-8930	7
33-3	Upper Pump Room Upper Wall	N/A	770	0-7270	10
33-4	Upper Pump Room Ceiling	N/A	50	0-560	1

NOTE:

Direct instrument contamination levels corrected for n
Removal smear data corrected for instrument background
Dose rate data corrected for background from natual ro
N/A = Not Applicable
ND = Not Determined

1
tamination Levels

NET DPM/100 CM^2

movable	ALPHA CONTAMINATION			
	Direct		Removable	
Range	Mean	Range	Mean	Range
140-11[illegible]0	ND	ND	3	0-20
5-620	ND	ND	1	0-10
30-1500	ND	ND	16	3-50
0-410	ND	ND	4	0-50
5-160	ND	ND	2	0-7
0-30	ND	ND	1	0-10
0-10000	0	0-0	<1	0-3
0-3365	ND	ND	0	0-0
0-370	ND	ND	0	0-0
0-260	ND	ND	<1	0-20
0-540	0	0-0	10	0-70
0-3470	0	0-0	3	0-70
0-4180	0	0-0	7	0-150
0-2900	0	0-0	2	0-30
0-210	ND	ND	<1	0-3
2-5240	0	0-0	27	0-190
2-26850	ND	ND	20	0-310
0-180	ND	ND	1	0-10
0-250	ND	ND	<1	0-3
0-7	ND	ND	<1	0-7
70-1710	ND	ND	22	0-110
40-3060	ND	ND	14	0-60
2-5870	ND	ND	10	0-150
0-340	ND	ND	2	0-13
150-2500	ND	ND	11	0-50
0-730	ND	ND	4	0-30
20-220	ND	ND	2	0-10
340-18570	0	0-0	2	0-7
0-3140	ND	ND	<1	0-7
0-780	ND	ND	2	0-7
0-10	ND	ND	1	0-7
20-27060	0	0-0	316	3-1150
1920-79200	0	0-0	640	150-4310
0-0	309	0-5610	0	0-0
7-275	48	0-160	4	0-35
7-275	23	0-134	4	0-35
2-40	ND	ND	2	0-10
0-0	ND	ND	0	0-0
0-0	ND	ND	0	0-0
0-1780	ND	ND	2	0-20
2-60	ND	ND	1	0-10
0-170	ND	ND	1	0-7
0-0	ND	ND	0	0-0
0-0	ND	ND	0	0-0
ND	ND	ND	ND	ND
ND	ND	ND	ND	ND
10-2560	ND	ND	3	0-40
0-120	ND	ND	<1	0-7
0-60	ND	ND	<1	0-3
240-2580	ND	ND	14	0-40
0-240	ND	ND	1	0-7
0-490	ND	ND	1	0-10
0-40	ND	ND	1	0-7
0-0	ND	ND	[illegible]	0-0
0-0	ND	ND	[illegible]	0-0
5-670	ND	ND	1	0-7
2-1090	ND	ND	3	0-20
2-30	ND	ND	<1	0-3

tural rad material background.

k formation.

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9010230140-01

Cintichem

TABLE 1.2a
Isotopic Composition of Structural

AREA No.	Name	GAMMA EMITTING ISOTOPIC RATIO NA-22	K-40	SC-46	MN-54
3-1	Cell Plug Area Floor		85.42%		
3-2	Cell Plug Area Lower Wall				
4-1	Upper Level Hot Lab Floor				
4-2	Upper Level Hot Lab Lower Wall				
4-3	Upper Level Hot Lab Upper Wall				
4-4	Upper Level Hot Lab Ceiling				
6-1	Waste Pit Floor				
6-2	Waste Pit Lower Wall				
6-3	Waste Pit Upper Wall				
6-4	Waste Pit Ceiling				
6-5	Waste Pit Horizontal Surface		65.40%		
7-1	Shipping Area Hot Lab Floor				
7-2	Shipping Area Hot Lab Lower Wall				
7-3	Shipping Area Hot Lab Upper Wall				
7-4	Shipping Area Hot Lab Ceiling				
7-5	Shipping Area Hot Lab Horizontal Surface				
9-1	Charging Area Floor				
9-2	Charging Area Lower Wall				
9-3	Charging Area Upper Wall				
9-4	Charging Area Ceiling				
9-5	Charging Area Horizontal Surface				
10-1	Conveyor Station Hot Cell Floor				
10-2	Conveyor Station Hot Cell Lower Wall				
10-3	Coveyor Station Hot Cell Upper Wall				
12-1	Exhaust Fan Room Floor				
12-2	Exhaust Fan Room Lower Wall				
12-2	Exhaust Fan Room Upper Wall				
15-1	Oper Area Manipulator Enc Floor				
15-2	Oper Area Manipulator Enc Lower Wall				
15-3	Oper Area Manipulator Enc Upper Wall				
15-4	Oper Area Manipulator Enc Ceiling				
18-1	Canal Floor				
18-2	Canal Lower Wall	0.09%	15.05%	0.90%	1.09%
21-1	Target Welding Shop Floor				
21-2	Target Welding Shop Lower Wall				
21-3	Target Welding Shop Upper Wall				
21-4	Target Welding Shop Ceiling				
21-5	Target Welding Shop Horizontal Surface				
21-6	Target Welding Shop				
23-1	Old Reactor Cave Area Floor				
26-1	Reactor 3rd Floor Floor				
26-2	Reactor 3rd Floor Lower Wall				
26-3	Reactor 3rd Floor Upper Wall				
26-4	Reactor 3rd Floor Ceiling				
26-7	Reactor Pool Gutter		14.17%	2.67%	1.26%
26-8	Reactor Pool Walls	0.22%	12.16%	3.27%	1.38%
30-1	781.12' Elev Reactor Area Floor				
30-2	781.12' Elev Reactor Area Lower Wall				
30-3	781.12' Elev Reactor Area Upper Wall				
32-1	Lower Pump Room Floor				
32-2	Lower Pump Room Lower Wall				
32-3	Lower Pump Room Upper Wall				
32-4	Lower Pump Room Ceiling				
32-5	Lower Pump Room Horizontal Surface				
33-1	Upper Pump Room Floor				
33-2	Upper Pump Room Lower Wall				
33-3	Upper Pump Room Upper Wall				
33-4	Upper Pump Room Ceiling				

Contamination

E-59	CO-57	CO-58	CO-60	ZN-65	ZR-95	NB-95	RU-103
					1.35%	2.00%	0.15%
			2.35%		7.60%	8.53%	1.14%
					11.10%		
			0.42%	0.21%	1.95%	2.74%	0.19%
			7.01%		6.96%		
			4.20%		4.27%		1.51%
			0.70%		19.42%		15.16%
							2.86%
			7.41%		5.01%	7.40%	0.50%
	0.02%		6.85%	5.30%	1.00%	3.17%	0.54%
0.41%		0.10%	7.04%	15.18%	4.81%	6.24%	0.76%
			8.76%	4.38%	1.35%	4.60%	1.07%
			16.99%				

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GAMMA EMITTING ISOTOPIC RATIO

AREA No.	Name	AG-108	AG-110M	SN-113	SB-124	SB
3-1	Cell Plug Area Floor					
3-2	Cell Plug Area Lower Wall					
4-1	Upper Level Hot Lab Floor					
4-2	Upper Level Hot Lab Lower Wall					
4-3	Upper Level Hot Lab Upper Wall					
4-4	Upper Level Hot Lab Ceiling					
6-1	Waste Pit Floor					
6-2	Waste Pit Lower Wall					
6-3	Waste Pit Upper Wall					
6-4	Waste Pit Ceiling					
6-5	Waste Pit Horizontal Surface	0.13%				
7-1	Shipping Area Hot Lab Floor					
7-2	Shipping Area Hot Lab Lower Wall					
7-3	Shipping Area Hot Lab Upper Wall					
7-4	Shipping Area Hot Lab Ceiling					
7-5	Shipping Area Hot Lab Horizontal Surface	2.06%				
9-1	Charging Area Floor					
9-2	Charging Area Lower Wall					
9-3	Charging Area Upper Wall					
9-4	Charging Area Ceiling					
9-5	Charging Area Horizontal Surface					
10-1	Conveyor Station Hot Cell Floor					
10-2	Conveyor Station Hot Cell Lower Wall					
10-3	Coveyor Station Hot Cell Upper Wall					
12-1	Exhaust Fan Room Floor					
12-2	Exhaust Fan Room Lower Wall					
12-2	Exhaust Fan Room Upper Wall					
15-1	Oper Area Manipulator Enc Floor					
15-2	Oper Area Manipulator Enc Lower Wall					
15-3	Oper Area Manipulator Enc Upper Wall					
15-4	Oper Area Manipulator Enc Ceiling					
18-1	Canal Floor				0.13%	
18-2	Canal Lower Wall	3.85%	4.26%	0.32%	0.08%	
21-1	Target Welding Shop Floor					
21-2	Target Welding Shop Lower Wall					
21-3	Target Welding Shop Upper Wall					
21-4	Target Welding Shop Ceiling					
21-5	Target Welding Shop Horizontal Surface					
21-6	Target Welding Shop					
23-1	Old Reactor Cave Area Floor					
26-1	Reactor 3rd Floor Floor					
26-2	Reactor 3rd Floor Lower Wall					
26-3	Reactor 3rd Floor Upper Wall					
26-4	Reactor 3rd Floor Ceiling					
26-7	Reactor Pool Gutter	1.67%	8.61%	1.30%	0.26%	
26-8	Reactor Pool Walls	4.80%	21.41%	1.51%	0.23%	
30-1	781.12' Elev Reactor Area Floor					
30-2	781.12' Elev Reactor Area Lower Wall					
30-3	781.12' Elev Reactor Area Upper Wall					
32-1	Lower Pump Room Floor					
32-2	Lower Pump Room Lower Wall					
32-3	Lower Pump Room Upper Wall					
32-4	Lower Pump Room Ceiling					
32-5	Lower Pump Room Horizontal Surface					
33-1	Upper Pump Room Floor					
33-2	Upper Pump Room Lower Wall					
33-3	Upper Pump Room Upper Wall					
33-4	Upper Pump Room Ceiling					

TABLE 1.2b
Isotopic Composition of Structural Contamination

-175	CS-134	CS-137	CE-144	EU-152	EU-155	IR-192	TOTAL
		4.85%	6.23%				100.00%
		100.00%					100.00%
							0.00%
							0.00%
							0.00%
							0.00%
		8.65%	71.73%				100.00%
		9.12%	79.78%				100.00%
							0.00%
							0.00%
0.20%	0.12%	5.47%	22.97%			0.10%	100.00%
		29.19%	56.84%				100.00%
		72.12%	26.28%			1.60%	100.00%
		49.48%	50.52%				100.00%
							0.00%
		51.71%	36.24%				100.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
		9.35%	52.35%				100.00%
		54.27%	42.87%				100.00%
							0.00%
							0.00%
0.60%	0.16%	5.14%	72.57%			1.07%	100.00%
0.51%	0.11%	0.96%	54.20%	0.30%	0.30%	1.11%	100.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
0.62%	0.31%	0.67%	28.41%	1.67%		3.80%	100.00%
1.17%	0.21%	0.28%	29.78%	1.08%	0.27%	2.06%	100.00%
		4.69%	78.33%				100.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%
							0.00%

TABLE 1.3
SUMMARY OF ACTIVATED INVENTORY

Item	Material	Volume cubic cm	Specific Activity[a]
Fuel support plate (aluminum)	Al 6061	14,817.58	2.12E-02
Fuel support plate (aluminum)	Al 1100	745.37	1.48E-02
Fuel support plate alignment pins	304SS	21.71	2.19E+00
Core support tower	Al 6061	7,928.76	3.52E-02
Plenum	Al 6061	10,769.26	6.67E-04
Flapper valve	Al 1100	2,394.49	4.65E-04
Bellows	304SS	27,534.27	1.76E-05
Thermal column lead shield	Pb	55,052.97	1.65E-04
Thermal column lead shield liner	AL 6061	5,505.30	5.57E-02
Thermal column external liner	Al 1100	121,819.07	1.92E-03
Pool lead liner	Pb	42,475.27	1.27E-07
Beam tubes	Al 1100	8,648.89	1.01E-03

Totals		297,712.94	
Pool walls, first foot	Barytes	4,018,594.72	5.02E-06
Pool walls, second foot	Barytes	4,018,594.72	2.51E-07
Pool walls, third foot	Barytes	4,018,594.72	1.25E-08
Pool floor	Std conc.	5,266,933.47	8.45E-07
Reinforcing steel	Carbon stl	42,047.50	4.44E-06

Pool walls & floor		17,364,765	
TOTAL volume, metal and concrete		17,662,478	

(a) All curie estimates are based upon decay until January 1, 1991.

TABLE 1.4
ESTIMATED RADIONUCLIDE INVENTORY IN ACTIVATED COMPONENTS[a]

Item	H3	C14	Fe55	Co60	Ni59	Ni63	Zn65	Nb94	Tc99	Eu152	Eu154	Others	Total Curies
Fuel support plate (6061 aluminum)	---	---	287	327	---	6	230	---	---	---	---	---	849
Fuel support plate (1100 aluminum)	---	---	10	14	---	0	6	---	---	---	---	---	30
Fuel support plate alignment pins	---	0	234	142	---	4	---	---	---	---	---	---	381
Core support tower	---	---	254	289	---	6	204	---	---	---	---	---	753
Plenum	---	---	7	7	---	0	5	---	---	---	---	---	19
Flapper valve	---	---	3	4	---	0	2	---	---	---	---	0.01	9
Bellows	---	---	2	1	---	0	---	---	---	---	---	---	4
Thermal column lead shield	---	---	3	1	---	2	11	---	---	---	---	86[b]	103
Thermal column lead shield liner	---	---	281	320	---	6	225	---	---	---	---	---	832
Thermal column external liner	---	---	219	288	---	7	120	---	---	---	---	---	634
Pool lead liner	---	---	---	---	---	---	---	---	---	---	---	0.06	0
Beam tubes	---	---	8	11	---	0	4	---	---	---	---	---	24
Pool walls, first foot	9	---	60	0	---	---	---	---	---	---	---	1.05	71
Pool walls, second foot	0	---	3	0	---	---	---	---	---	---	---	0.05	4
Pool walls, third foot	0	---	0	---	---	---	---	---	---	---	---	---	0
Pool floor	6	---	1	0	---	---	---	---	---	---	---	0.76	10
Reinforcing steel	---	---	1	0	---	---	---	---	---	---	---	---	1
Total by isotope	17	0	1,375	1,404	<0.01	33	806	<0.01	<0.0	<0.01	<0.01	87	3,723

(a) All curie estimates are based upon decay until January 1, 1991.

(b) Includes 27.2 curies of ^{109}Cd, 27.2 curies of ^{109m}Ag, 13.6 curies of ^{108m}Ag, and 5.48 curies of ^{123m}Te, among other isotopes.

TABLE 1.5
CALCULATED RADIATION DOSE
RATES FROM INDIVIDUAL ACTIVATED COMPONENTS[a]

Item	Roetegen/hour at 1 meter (Air)	Roetegen/hour at 1 meter (Water)
Fuel support plate (6060 aluminum)	306.00	3.55
Fuel support plate (1100 aluminum)	306.00	3.55
Fuel support plate alignment pins	181.90	3.04
Core support tower	29.90	0.46
Plenum	34.15	0.51
Flapper valve	3.70	0.06
Bellows	1.27	0.02
Thermal column lead shield	0.8084	0.0135
Thermal column lead shield liner	397.20	6.36
Thermal column external liner	26.48	0.42
Pool lead liner	0.0093	0.0001
Beam tubes	29.90	0.46
Pool walls, first foot	15.8800	0.0109
Pool walls, second foot	0.3361	n/a
Pool walls, third foot	0.0074	n/a
Pool floor	0.1681	n/a
Reinforcing steel	1st foot	n/a

[a] All dose rate estimates are based upon decay until January 1, 1991.

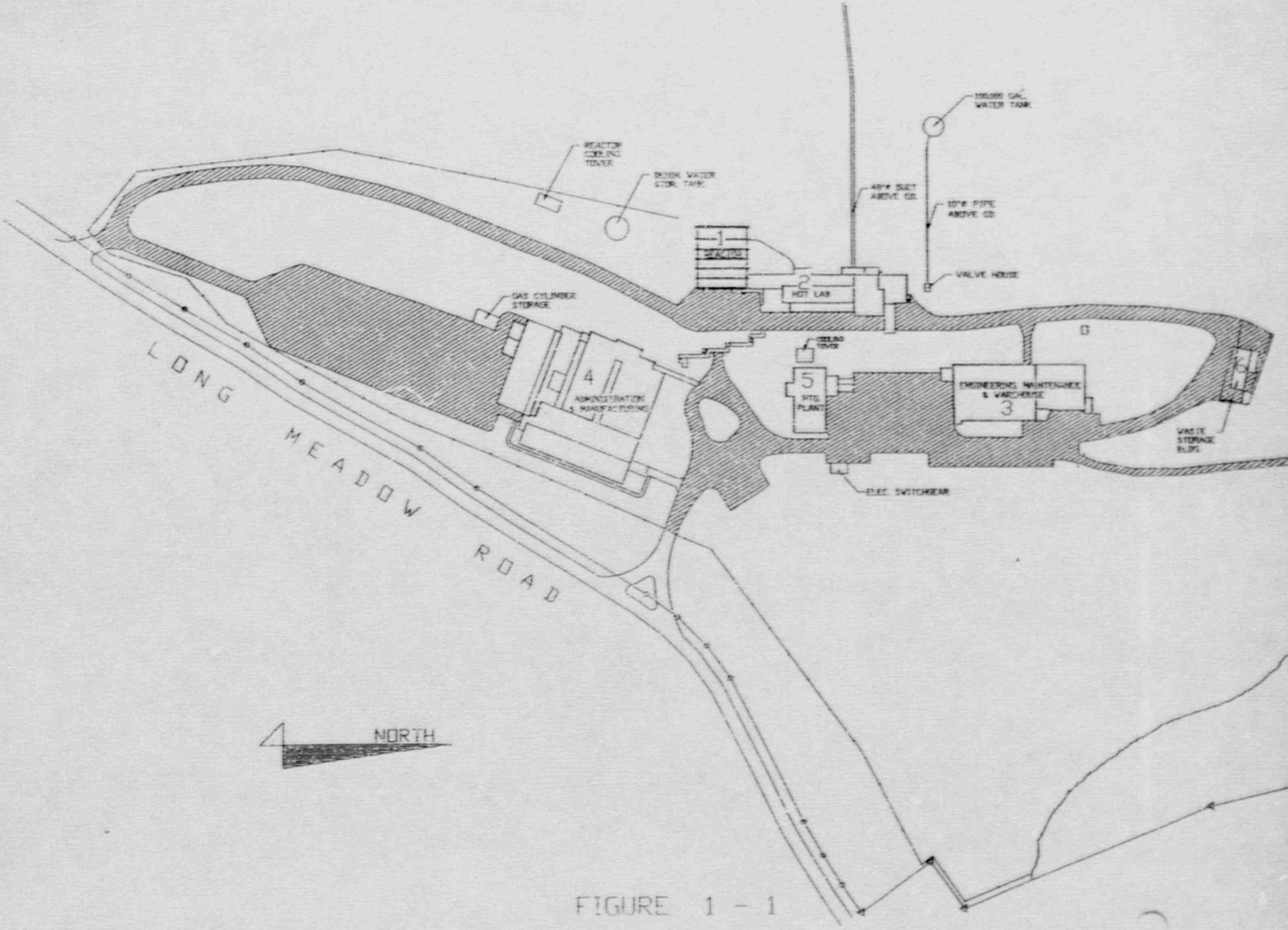

100,000 GAL WATER TANK
REACTOR COOLING TOWER
DEMIN. WATER STOR. TANK
48"Ø DUCT ABOVE GD.
10"Ø PIPE ABOVE GD.
1
REACTOR
2
HOT LAB
VALVE HOUSE
GAS CYLINDER STORAGE
COOLING TOWER
4
ADMINISTRATION & MANUFACTURING
5
HTG. PLANT
ENGINEERING MAINTENANCE & WAREHOUSE
3
6
WASTE STORAGE BLDG.
ELEC. SWITCHGEAR
LONG MEADOW ROAD
NORTH

FIGURE 1 - 1

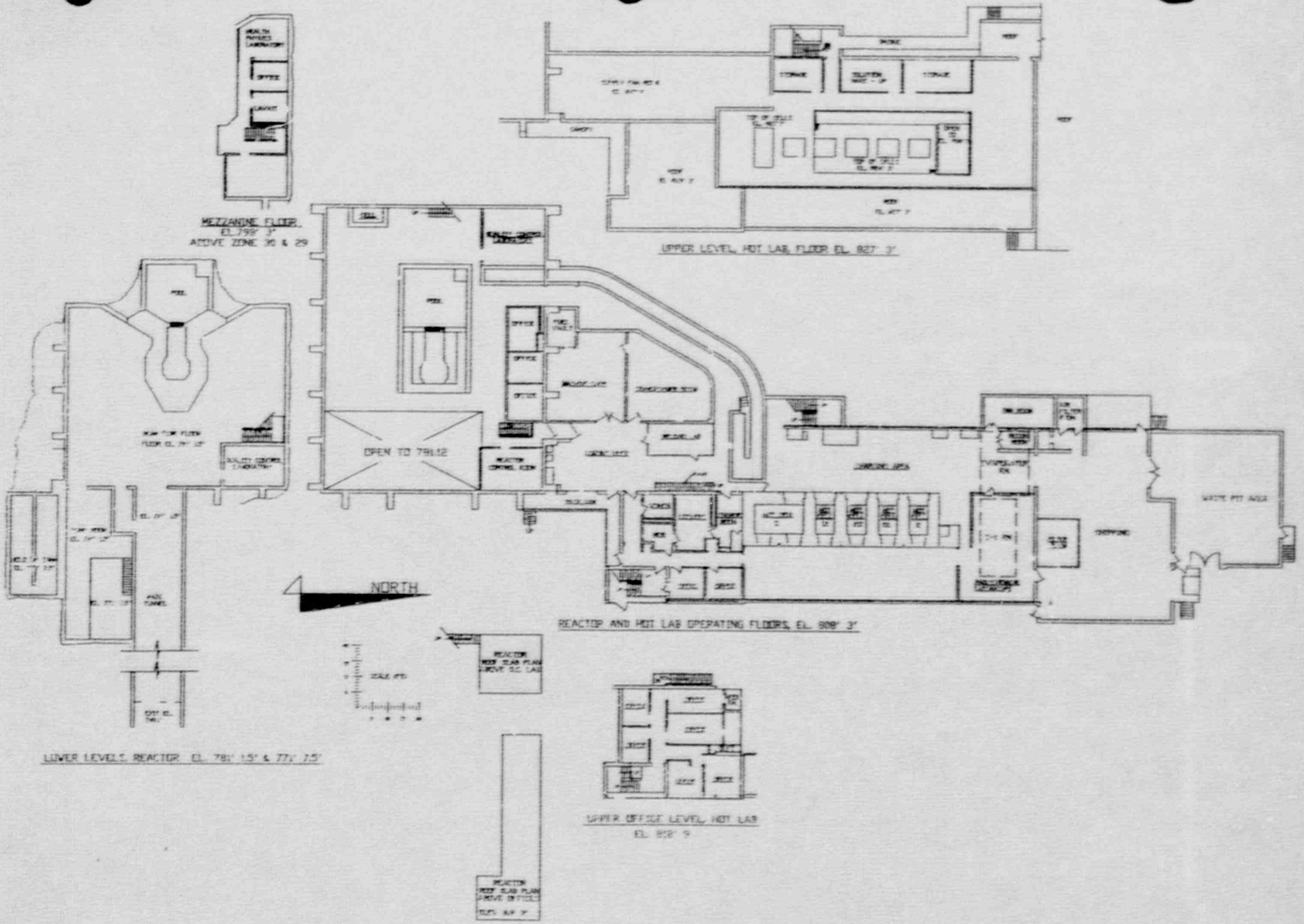
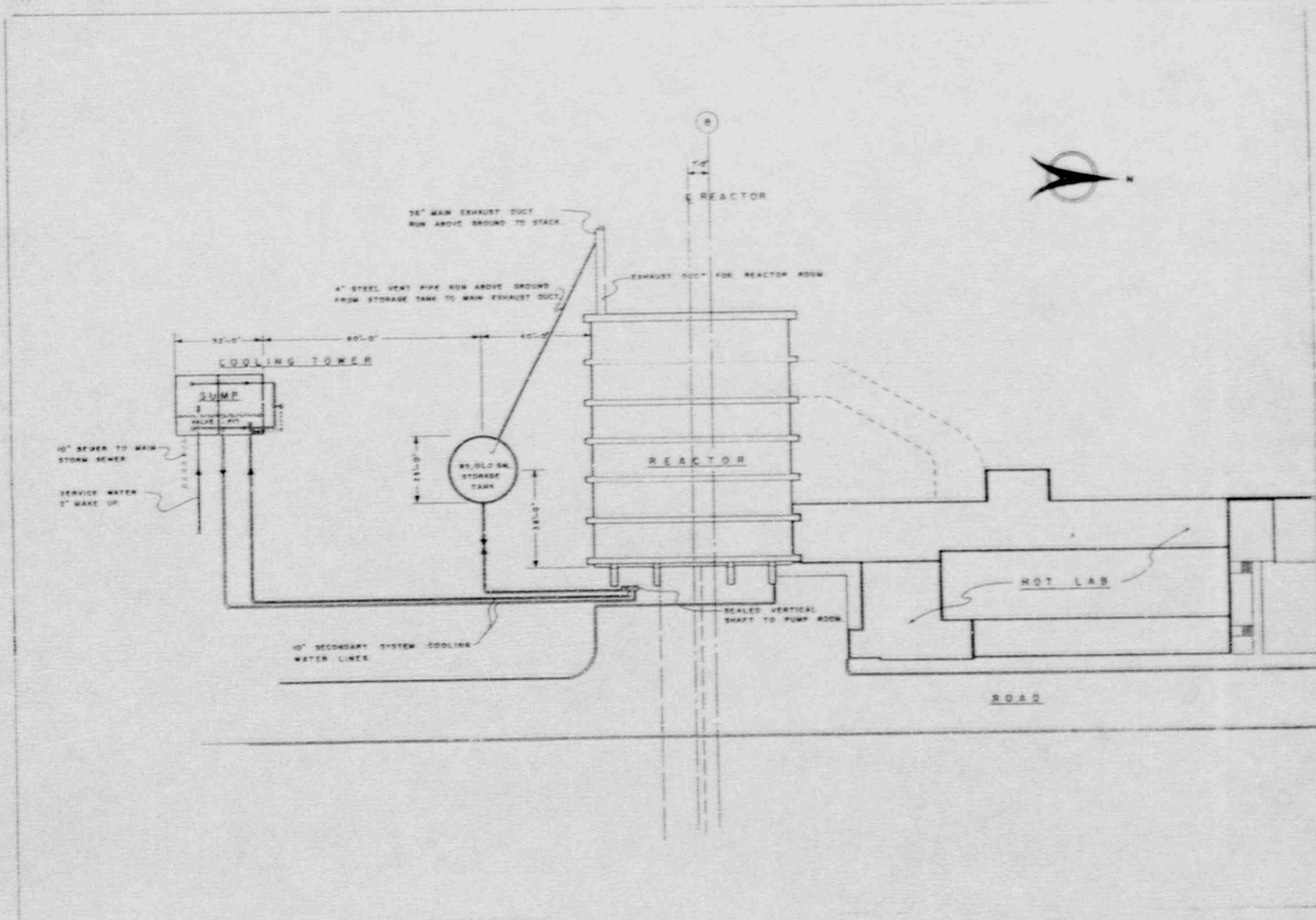
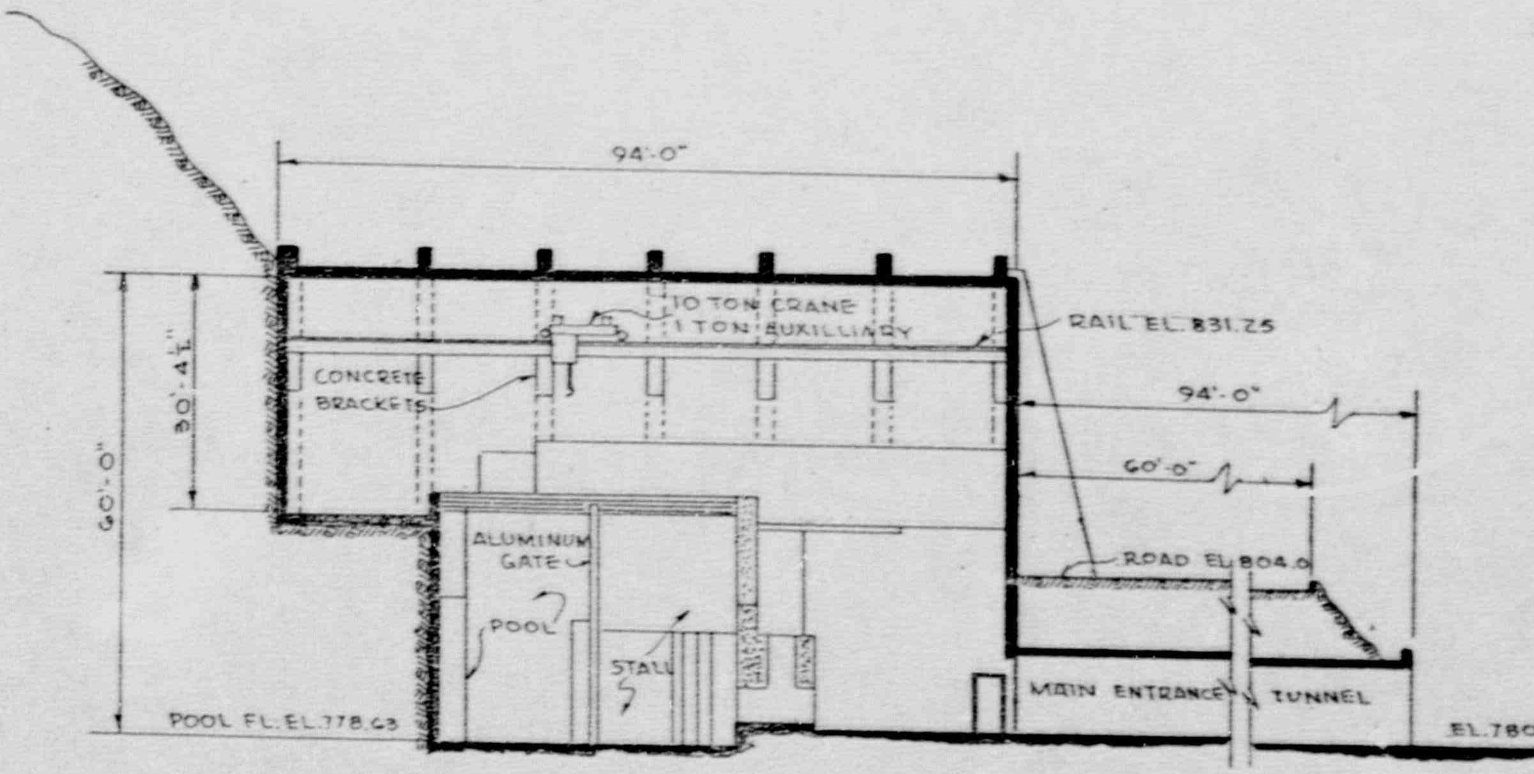

MEZZANINE FLOOR
EL 799' 3"
ABOVE ZONE 30 & 29
UPPER LEVEL, HOT LAB, FLOOR EL. 827' 3"
OPEN TO 791'12
NORTH
REACTOR AND HOT LAB OPERATING FLOORS, EL. 808' 3"
LOWER LEVELS REACTOR EL. 781' 1.5" & 771' 7.5"
UPPER OFFICE LEVEL, HOT LAB
EL. 812' 9

FIGURE 1 - 2

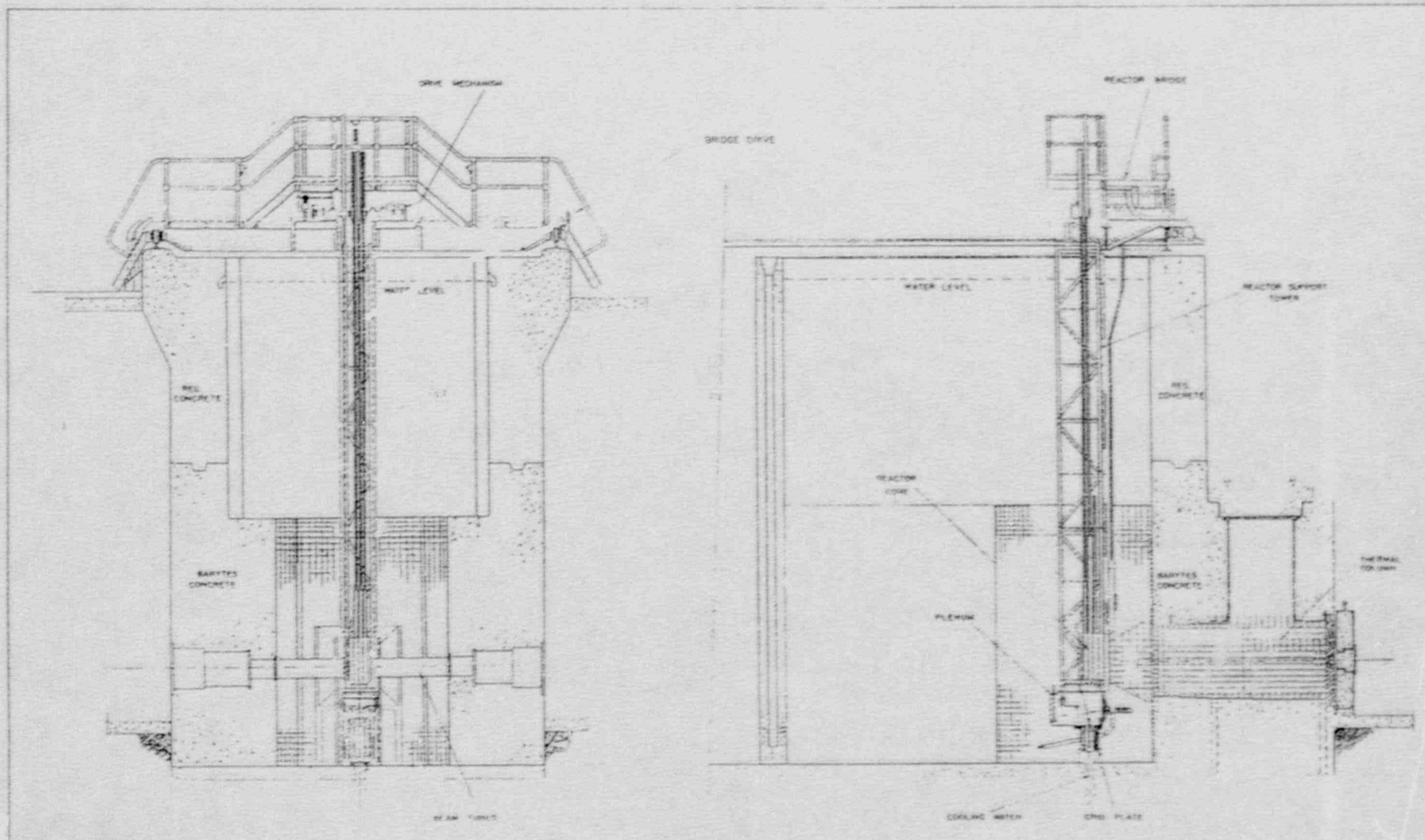


REACTOR BUILDING SITE PLAN
FIGURE 1.3



REACTOR BUILDING

FIGURE 1.4



VERTICAL SECTIONS THRU REACTOR POOL

FIGURE 1.5

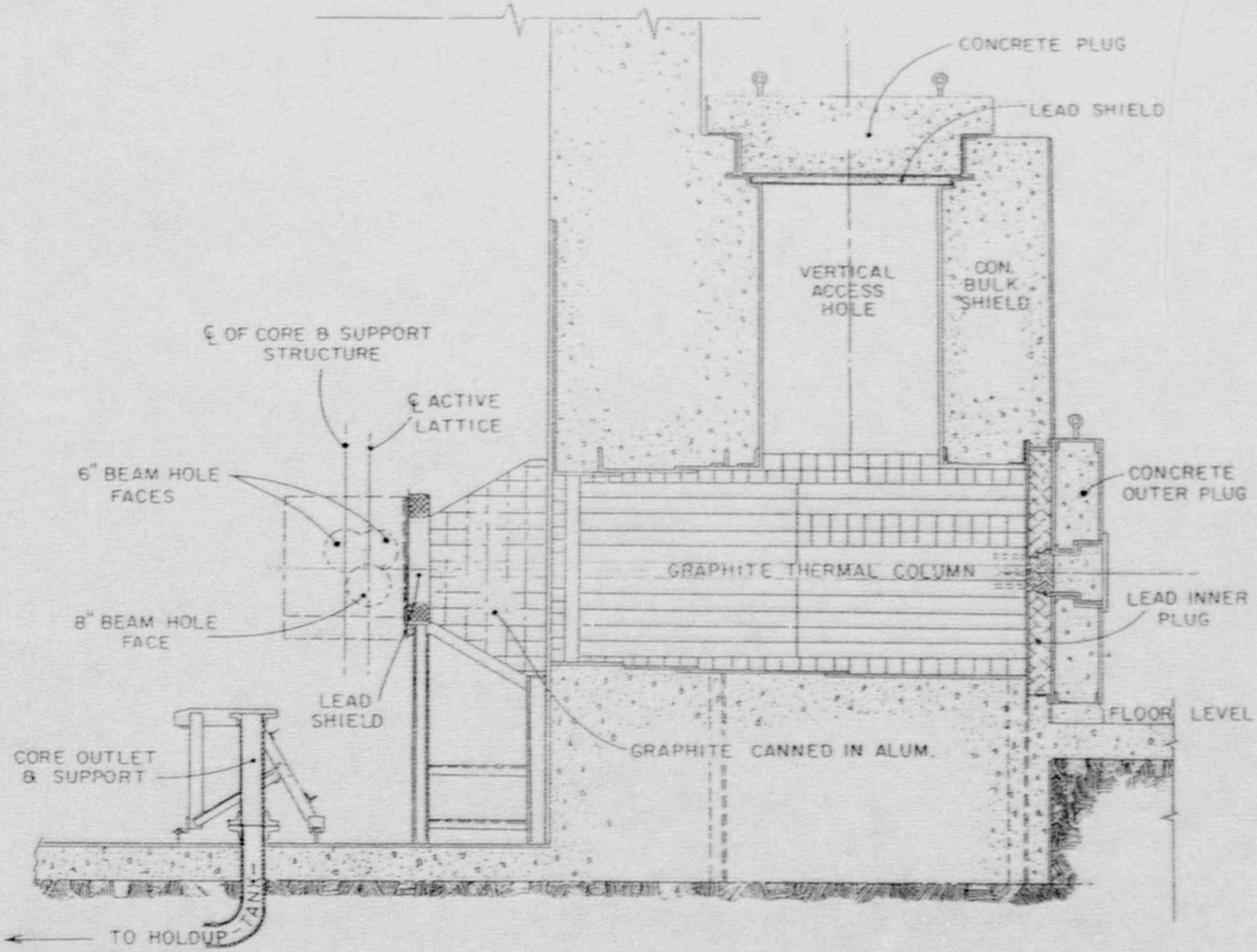


FIGURE 1.6

VERTICAL SECTION THRU THERMAL COLUMN

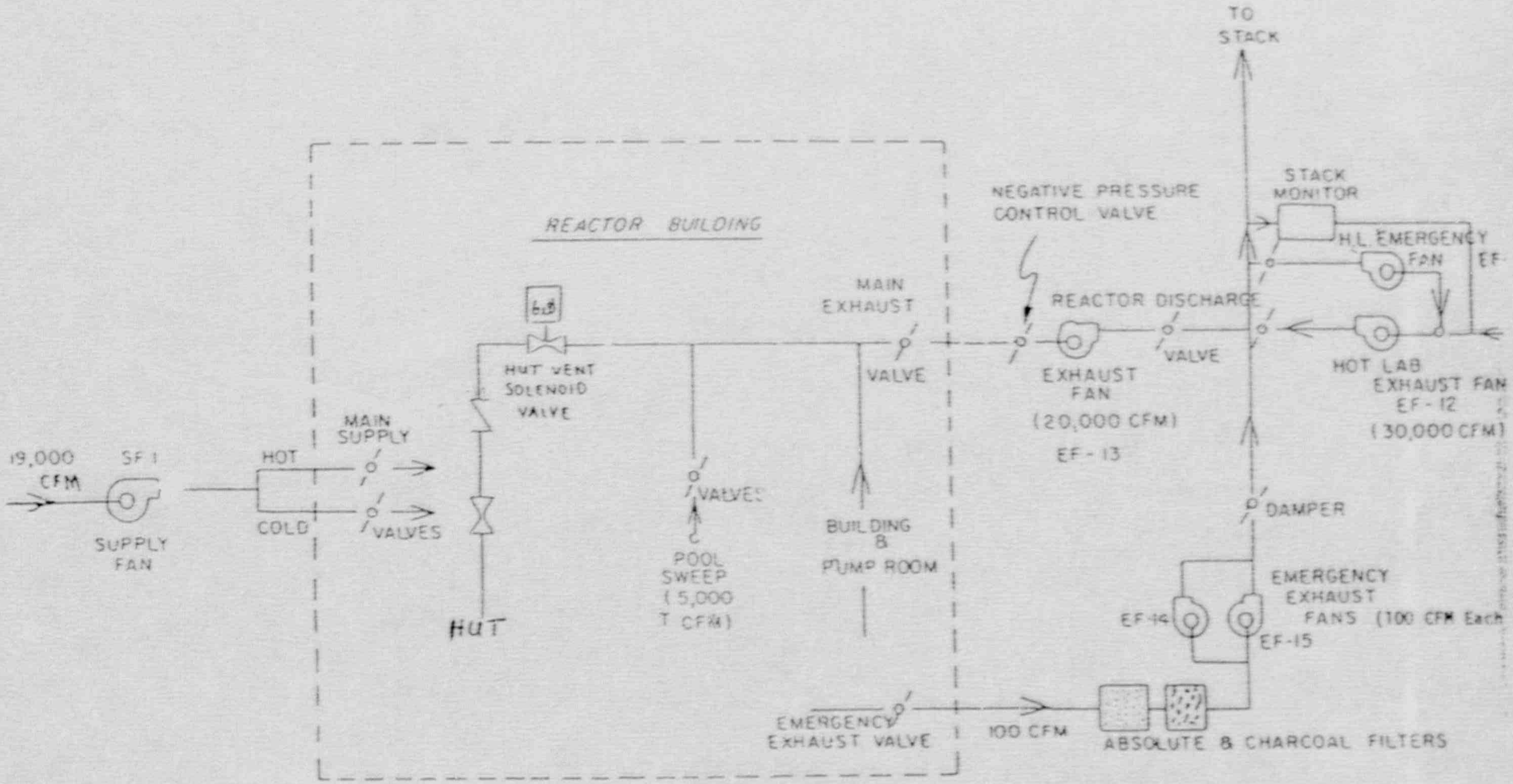
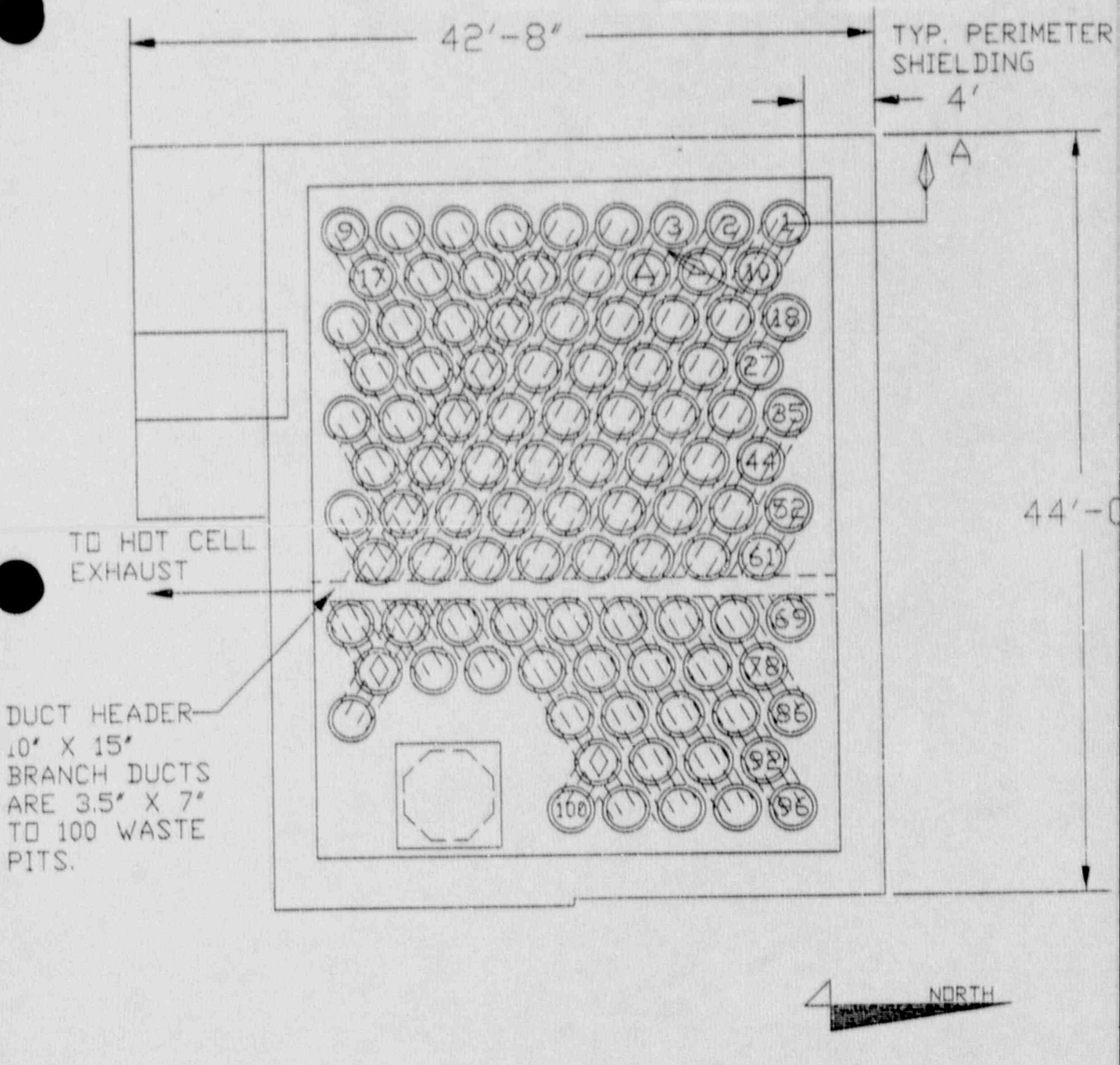
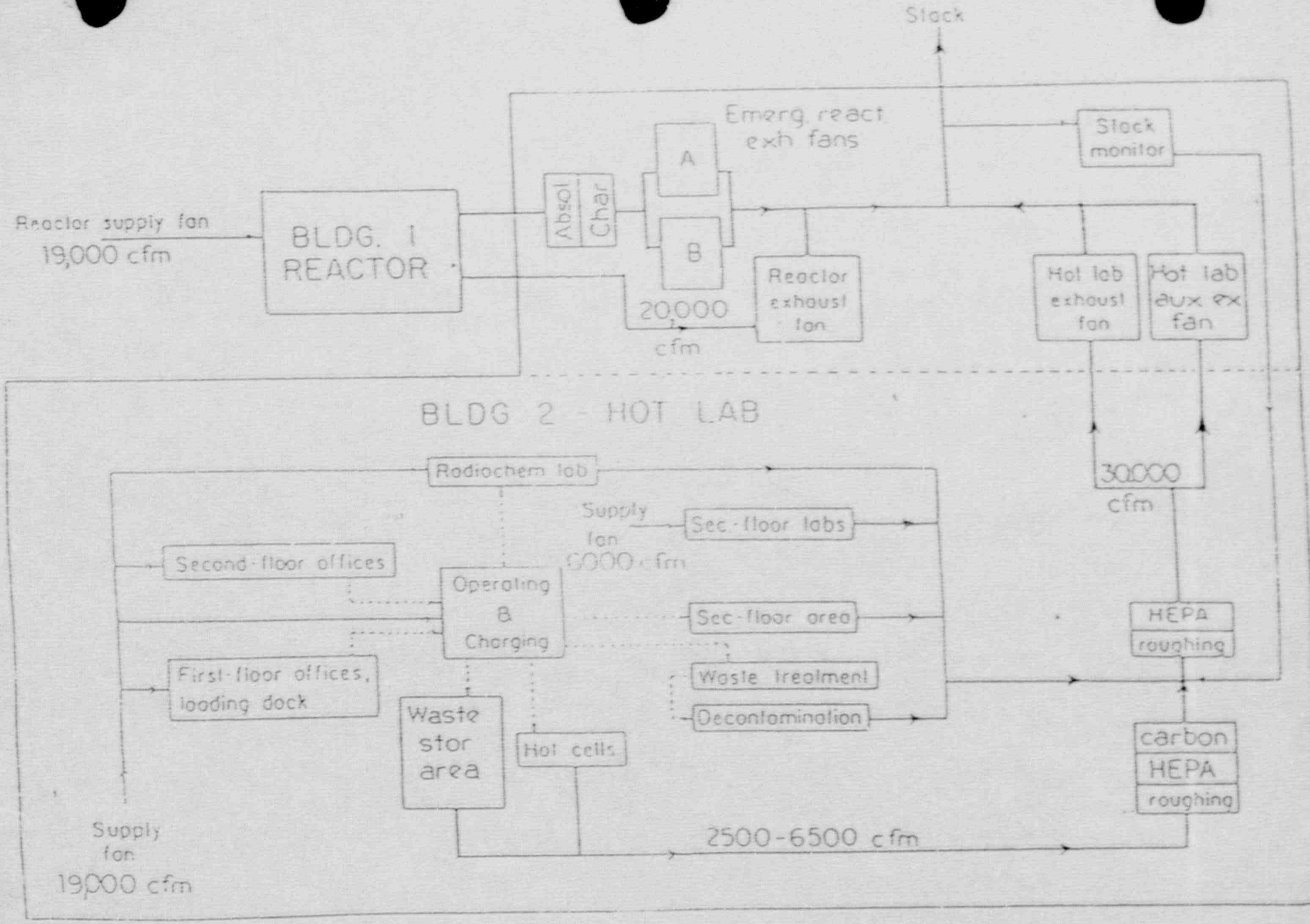


Figure 1.7 Ventilation system



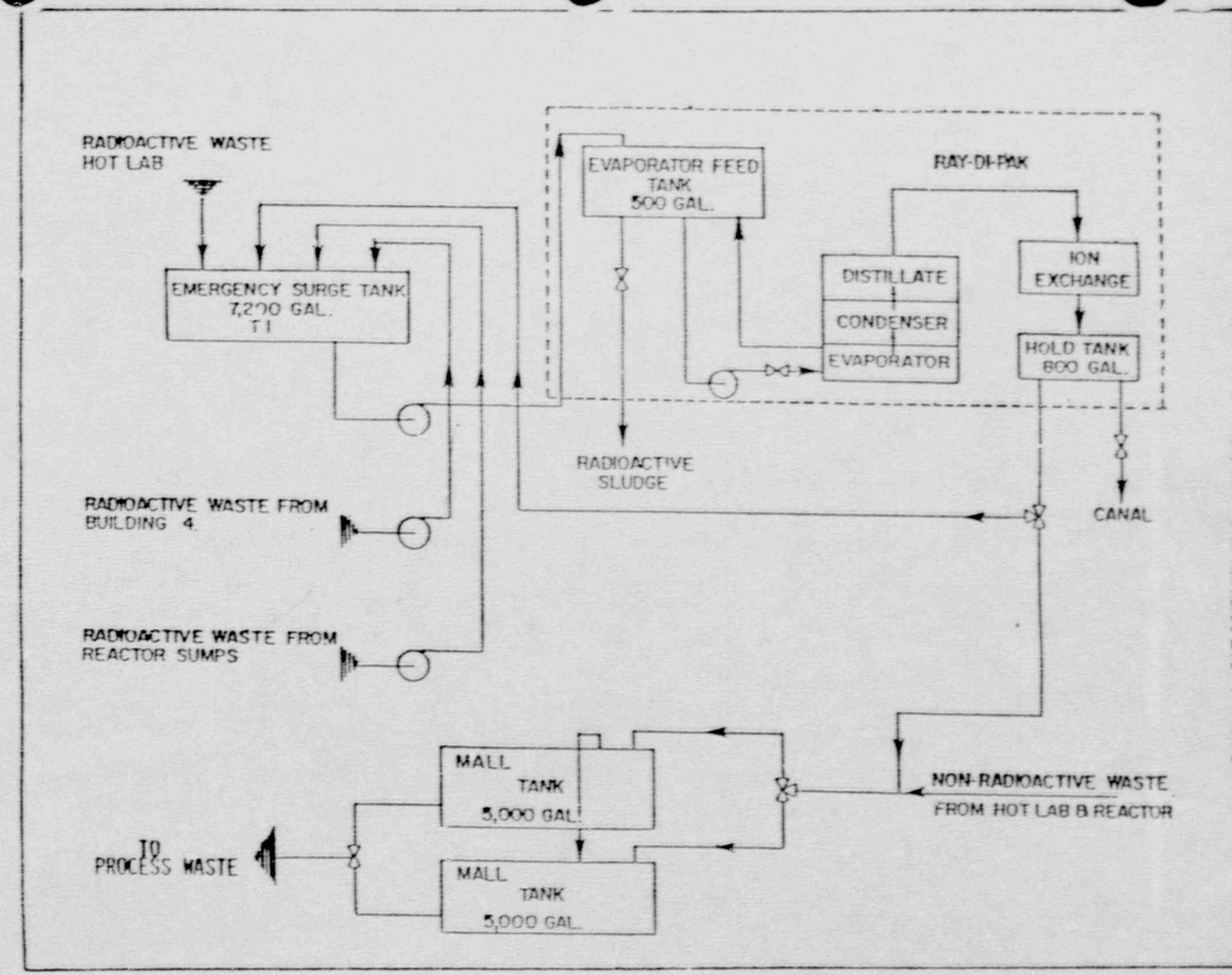
WASTE PIT AREA

FIGURE 1 - 8



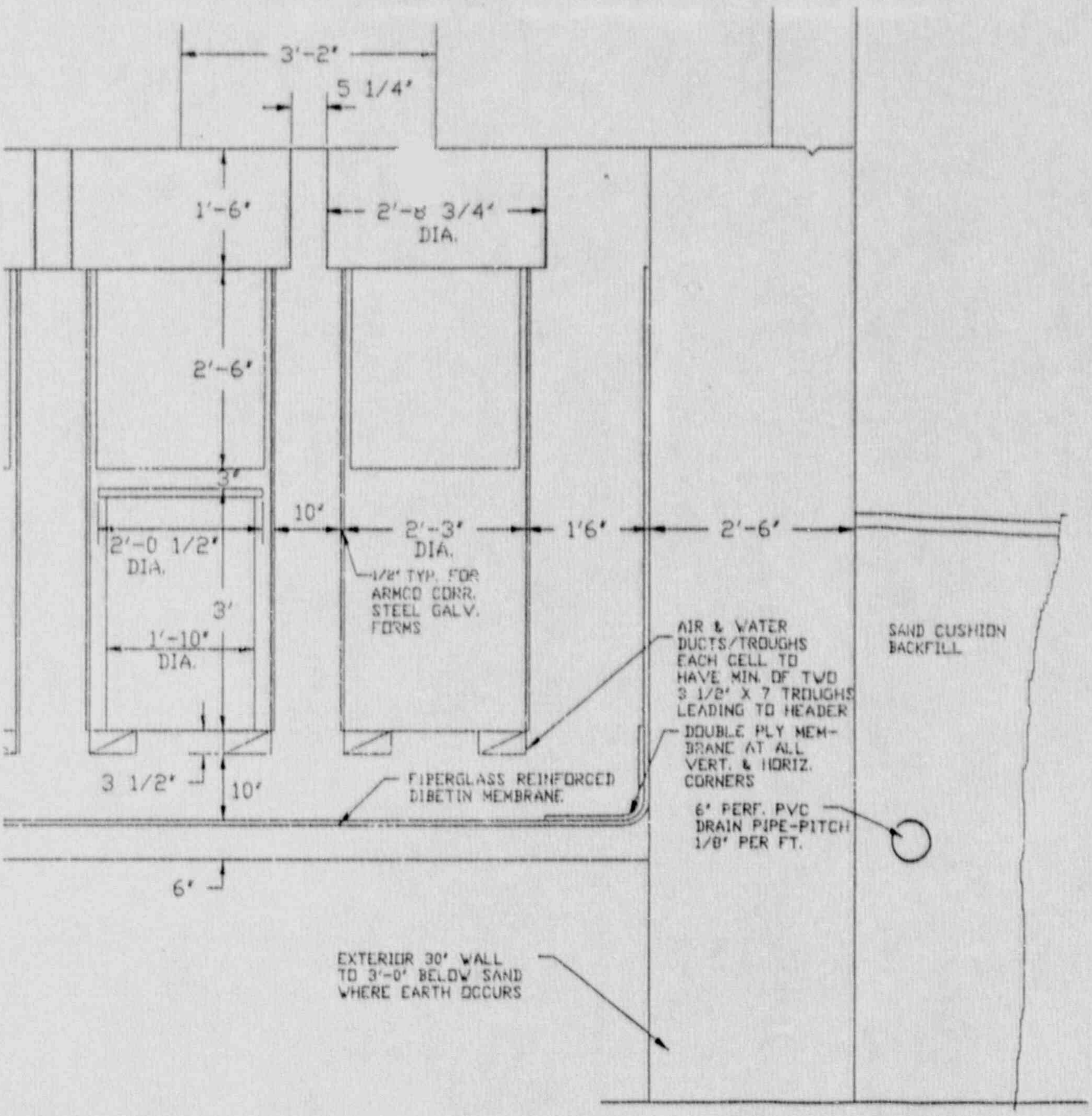
SCHEMATIC of REACTOR, HOT LAB AIR FLOW

FIGURE 1.10



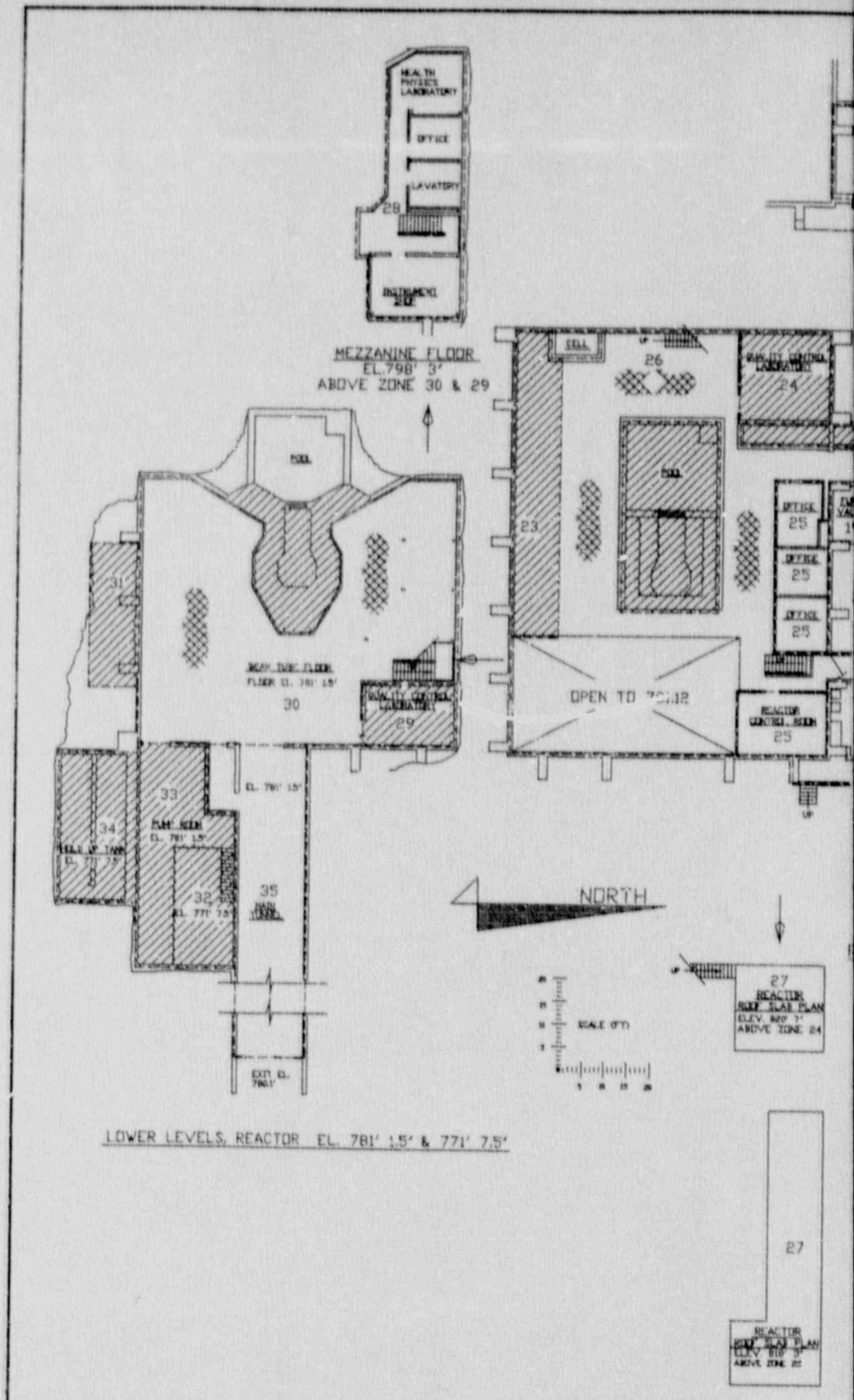
Radioactive Liquid Waste Disposal System

Figure 1.11



SECTION A - A

FIGURE 1 - 9



HEALTH PHYSICS LABORATORY
OFFICE
LAVATORY
28
MEZZANINE FLOOR
EL.798' 3'
ABOVE ZONE 30 & 29
POOL
31
BEAM TUBE FLOOR
30
QUALITY CONTROL LABORATORY
29
33
34
PUMP ROOM
EL. 781' 1.5'
HOLD UP TANK
EL. 771' 7.5'
32
EL. 771' 7.5'
35
EL. 781' 1.5'
CELL
26
QUALITY CONTROL LABORATORY
24
23
POOL
OFFICE
25
OFFICE
25
OFFICE
25
REACTOR CONTROL ROOM
25
NORTH
UP
27
REACTOR ROOF SLAB PLAN
ABOVE ZONE 24
SCALE (FT)
27
REACTOR
ROOF SLAB PLAN
ABOVE ZONE 22
LOWER LEVELS, REACTOR EL. 781' 1.5' & 771' 7.5'

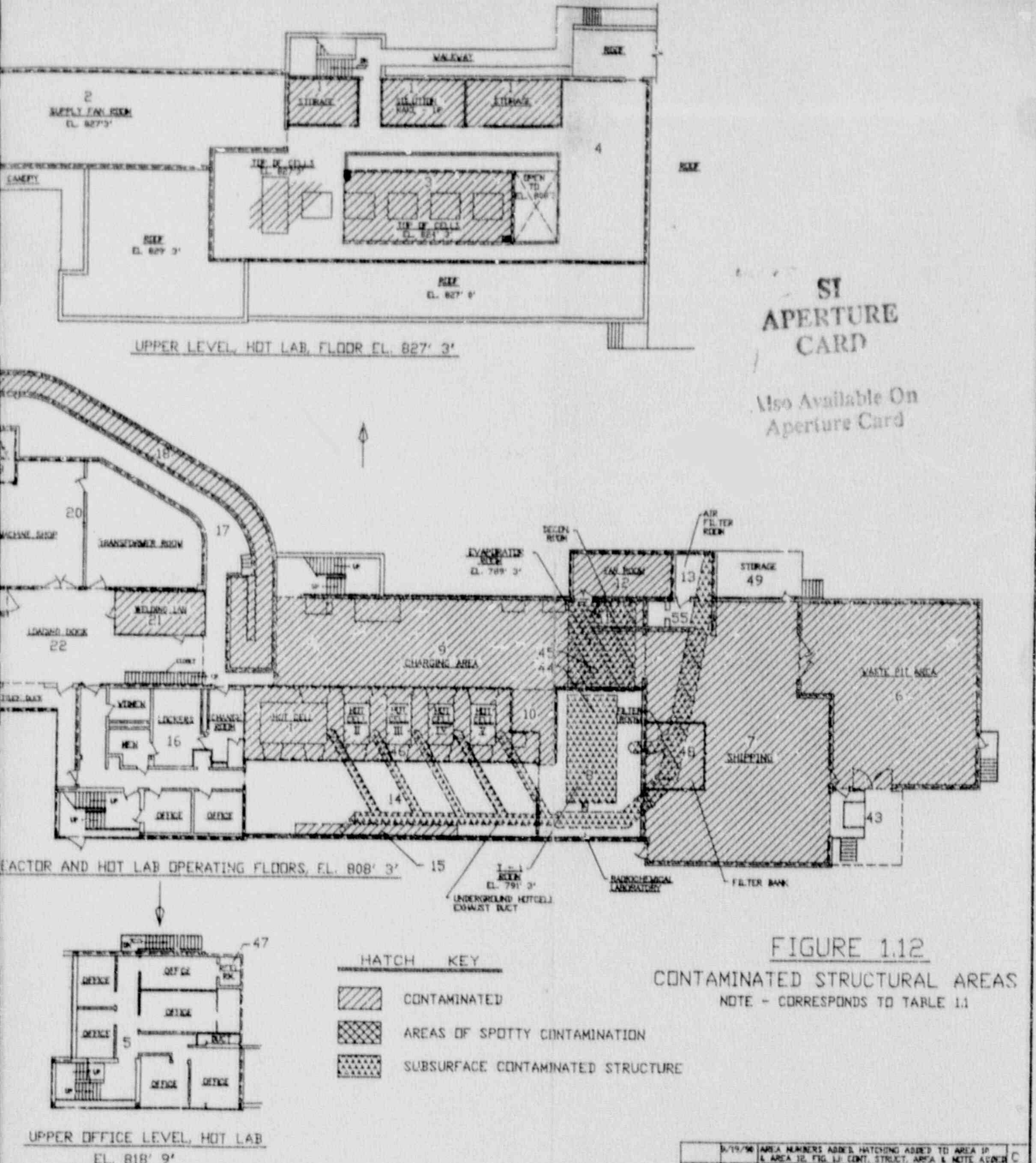


FIGURE 1.12
CONTAMINATED STRUCTURAL AREAS
NOTE - CORRESPONDS TO TABLE 1.1

SI
APERTURE
CARD

Also Available On
Aperture Card

APPR.	DATE	DESCRIPTION	REV.
	9/19/90	AREA NUMBERS ADDED, HATCHING ADDED TO AREA 19 & AREA 18, FIG. 1.1 CONT. STRUCT. AREA & NOTE ADDED	C
	8/13/90	HATCH KEY ADDED, UNDERGROUND CONTAMINATED STRUCTURES ADDED	B
	6/5/90	INSTRUMENT SHOP WAS UTILITY ROOM, FUEL VAULT ADDED TO AREA 19, ELEV. 820' 4' WAS 818' 3', AREA 47 ADDED TO RECL ROOM, HATCHING ADDED	A

CAD FILE # 136E2328 TLO — REVISIONS — 1/8 SCALE FACTOR

DRAWN: CJG 6/14/90

CINTICHEM, INC.

P.O. BOX 816, TUXEDO, NEW YORK 10987

REACTOR HOT LAB OPERATING FLOORS — SIZE E — DRAWING NO. 136E2328

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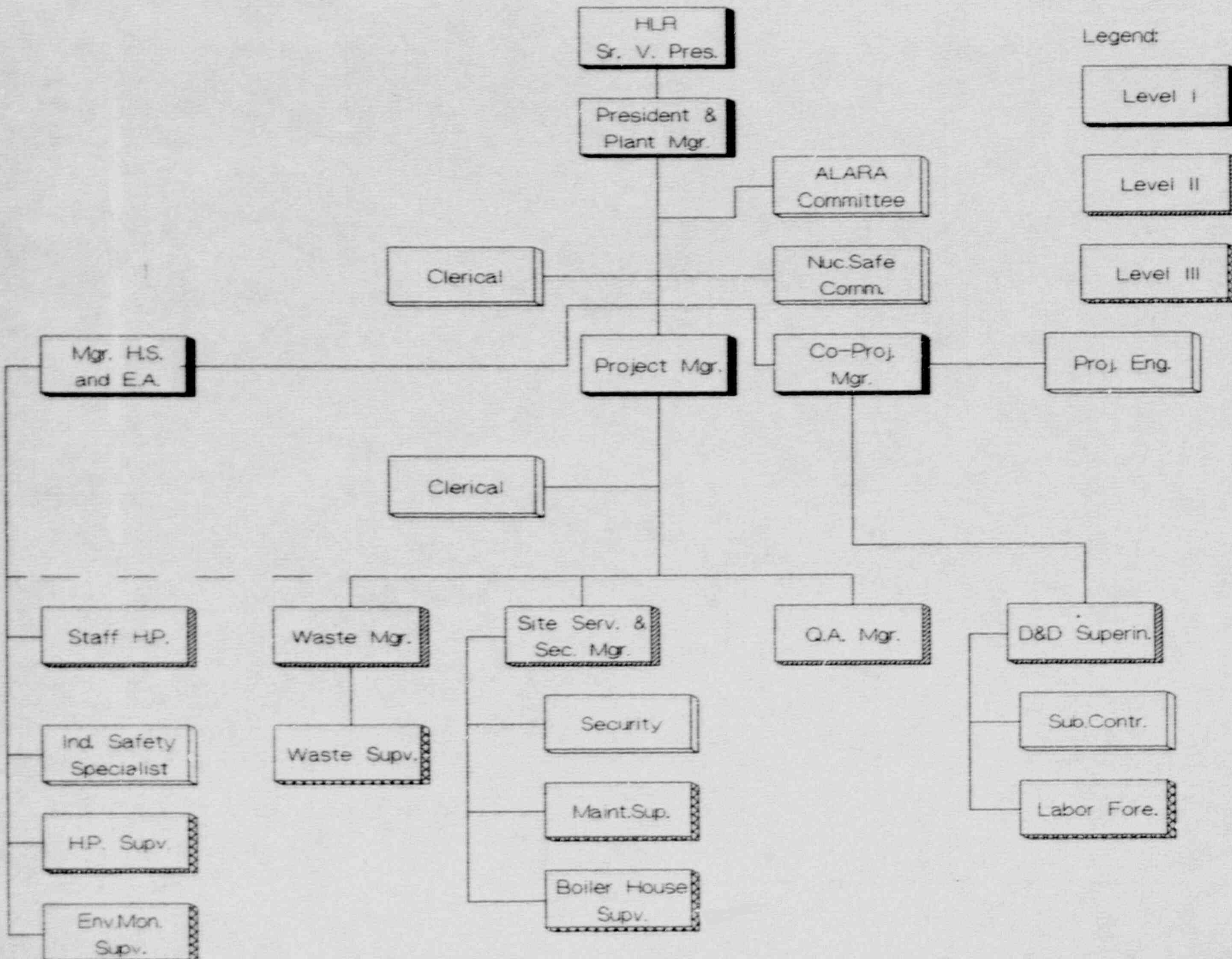


HLR
Sr. V. Pres.
President & Plant Mgr.
ALARA Committee
Clerical
Nuc.Safe Comm.
Mgr. H.S. and E.A.
Project Mgr.
Co-Proj. Mgr.
Proj. Eng.
Clerical
Staff H.P.
Waste Mgr.
Site Serv. & Sec. Mgr.
Q.A. Mgr.
D&D Superin.
Security
Sub.Contr.
Ind. Safety Specialist
Waste Supv.
Maint.Sup.
Labor Fore.
H.P. Supv.
Boiler House Supv.
Env.Mon. Supv.
Legend:
Level I
Level II
Level III

FIGURE 1.13

2.0 Occupational and Radiation Protection Programs

2.1 Radiation Protection Program

Cintichem has a radiation safety program with a Health Physics staff equipped with radiation detection equipment to determine, control, and document radiation exposures within the facility and offsite. The Health Physics Department ensures that occupational radiation exposures meet ALARA requirements through a monitoring program of air sampling, radiation monitoring, and contamination control measures. Control of dismantling and decommissioning work will be maintained by Health Physics Radiation Work Permits.

The Health Physics Program addresses radiation protection through employee training, specification of engineered safeguards, and implemtation of a monitoring program for employees, work areas, and the environment. The objectives of this program include:

- o Minimizing the exposure of the general public and the environment to the radioactive effluents that may be released during the decommissioning activities;
- o Ensuring the health and safety of personnel by providing protection programs which include a commitment to the principles of maintaining exposures As Low As Reasonably Achievable (ALARA);
- o Control of radioactive contamination and sources;
- o Control of transfer of radioactive contamination;
- o Certification that release criteria are met.

2.1.1 Health Physics Organization

a. Manager of Health Safety and Environmental Affairs

The Manager of Health Safety and Environmental Affairs is the site Radiation Safety Officer. Reporting to this position are a Staff Health Physicist, and the Health Physics Supervisor. The Health Physics Technicians report to the Health Physics Supervisor.

The Manager of Health Safety and Environmental Affairs manages the on-site and off-site environmental monitoring program per license and regulatory requirements, and ALARA principles. He coordinates activities of all operations groups involving health safety, and environmental protection against radiological and other, more common hazards. Day-to-day operations and training are carried out through delegation of specific duties to the Health Physics Supervisor and the Staff Health Physicist.

b. Health Physics Supervisor

The Health Physics Supervisor supervises the health physics staff per regulatory and license requirements and ALARA principles. The Supervisor implements the Health Physics program in the field. Specific duties include:

- o Planning and scheduling of health physics technicians;
- o Approval of Radiation Work Permits;
- o Coordinating with the D & D Superintendent;
- o Assuring that the personnel monitoring program is maintained;
- o Assuring the area surveys are conducted per the program;
- o Assuring that routine surveillance is performed in acccordance with the program;
- o Maintaiing adequate control of radiological hazards;
- o Ensuring that the equipment calibration program is maintained.

c. Staff Health Physicist

The Staff Health Physicist is responsible for the development, implementation, evaluation, investigation, follow-up, and/or performance of site radiation programs/operations regarding ALARA principles, regulatory requirements, radiation health safety, and environmental considerations. Specific duties include:

- o Responsibility for radiation worker training;
- o Radiological dose assessments;
- o Evaluating the radiation monitoring program and recommending improvements and changes as appropriate;
- o Conducting special investigations into unusual events or exposures;
- o Serving as Chairman of the ALARA Committee;
- o Evaluating the effectiveness of the environmental program;
- o Performing technical and hazard summary evaluations for regulatory and license applications.

d. Health Physics Technician

Because of the nature of dismantling and decommissioning work (large work force on site with varying degrees of experience, increased contamination control problems, and changing airborne and radiation conditions), all work crews will be under the surveillance of a health physics technician. Some jobs require more technicians and they are assigned as specified on the Radiation Work Permit. The technician(s) also assure proper protective clothing and equipment is provided and access to contaminated areas is limited to workers who are needed on the job.

2.1.2 Control Programs

2.1.2.1 Radiation Work Permits

Control of dismantling and decommissioning work is maintained by Radiation Work Permits. Because of the nature of dismantling and decommissioning work, a system is needed to allow Health Physics to manage the tasks. A Radiation Work Permit and an assigned personnel list will be issued for each job. The Radiation Work Permit specifications include:

- Location;
- General description of work;
- Protective clothing;
- Equipment (tents, HEPA ventilation, respirators, etc.);
- Dosimetry;
- Personnel monitoring (frisking);
- Health Physics coverage;
- Dose limits;
- Remarks/recommendations;
- Expiration date.

2.1.2.2 Contamination Control Program

A contamination control program will be implemented during the decommissioning project to minimize the potential contamination of workers or adjacent facilities, and protect the public. The decommissioning work at the Cintichem facility must be performed in a controlled manner, so that any hazards created by decommissioning are eliminated or sufficiently controlled.

2.1.2.2.1 Surveys

Contamination surveys are an integral part of the Contamination Control Program. Appropriate smears will be taken to ensure that no contamination has spread outside of controlled surface contamination areas. Survey results will be reviewed by the Health Physics Department and appropriate action will be taken to minimize recurrance of contamination incidents.

2.1.2.2.2 Access Control

The Cintichem Health Physics Department will monitor decontamination access control via:

- Personnel monitors capable of detecting alpha and/or beta-gamma radiation as appropriate shall be provided at the routine exits of the facility until the buildings are breached. At that point, whole body frisks will be performed at an alternate access point. In addition to Health Physics support, employees shall be required to monitor themselves and report contamination levels above the alarm set points to a Health Physics Technician;

- Areas requiring special protective clothing will be posted by Health Physics and specified in the Radiation Work Permit.

  A changing room and shower facility shall be provided for all personnel. The shower facility shall be used by decommissioning workers for removal of body contamination as recommended by Health Physics.

2.1.2.2.3 Contamination Containment

a. Contamination Control Areas

   The facility will be divided into as many different work areas as needed to meet contamination control objectives. Typically there are three frequently used areas. Each work area will be further subdivided into three distinct areas:

- Contaminated Area: where contamination does or could occur;

- Transition Area: the area where personnel or equipment ingress and egress from the contaminated work area;

- Clean Area: the non-contaminated area where workers should not be exposed to hazardous conditions.

  Delineation of these three areas will be based on survey results and on an evaluation of potential routes and the amount of contaminant dispersion that could take place in the event of a release. Movement of personnel and equipment among these areas will be minimized and restricted to specific Access Control Points to prevent cross-contamination from contaminated areas to clean areas.

The Contaminated Area is the area where contamination does or could occur. The primary activities performed in the Contaminated Areas are:

- o decontamination and dismantling activities;
- o preliminary waste treatment and packaging.

The outer boundary of the Contaminated Area will be clearly marked by lines, placards, hazard tape and/or signs; or enclosed by physical barriers, such as chains, fences, ropes or containment structures. Access control points will be established at the periphery of this area to regulate the flow of personnel and equipment. This will help to maintain the proper procedures for entering and exiting.

The Contaminated Area will be subdivided into different areas of contamination or dose rate based on the known or expected degree of hazard. This allows more flexibility in safety requirements, operation of decontamination procedures, and use of resources.

All personnel within the Contaminated Area will wear a proper level of protective equipment as is required by the Radiation Work Permit which will be posted at the entrance to the area. Within the area, different levels of protection will be justified based on the degree of hazard present.

The Transition Area is between the contaminated work area and the Clean Area. This area is designed to reduce the probability that the Clean Area will become contaminated by the decommissioning work or affected by site hazards. The distance between the Contaminated and Clean Areas provided by the Transition Area, together with decontamination and contamination monitoring of workers and equipment, limits the physical transfer of radioactive material into clean areas. The degree of contamination in the Transition Area decreases as one moves from the Contaminated Area to the Clean Area, due both to the distance and the decontamination procedures.

The Transition Area is divided into two sub-areas. The first area upon leaving the Contaminated Area is where personnel remove personal protective equipment (coveralls, gloves, etc.). The second sub-area, closer to the clean areas, is where personnel check themselves or equipment for radioactive contamination prior to leaving this area for clean areas.

The Clean Area will be the location of the administrative and other support functions needed to keep the operations in the Contamination and Transition Areas running smoothly. Any function that need not or cannot be performed in the Contaminated or Transition Areas will be performed here. Personnel may wear normal work clothes within this area. Any potentially contaminated clothing, equipment, and samples will remain in the Transition Area until decontaminated or properly transferred.

b. Contamination Control Enclosures

Work areas will require special consideration for set up due to the nature of the work being conducted in that area, the hazard present in the area, and the physical layout of the area. Due to the dusty nature of demolition work, where appropriate, the Contaminated Area for some work areas will be located within containment structures with an inward flow of air.

The containment structure is a barrier system that prevents radioactive material from contaminating areas outside the work area. It consists of two parts: the enclosed work area (Contaminated Area) and anteroom. The barriers are constructed of two layers of plastic sheeting.

The anteroom is constructed of plastic sheeting and has overlapping drapes. When needed, the plastic sheeting can be supported by a framework such as wooden 2x4's, 2x6's, or by a PVC pipe frame. The plastic sheeting can be attached using duct tape and/or furring strips with staples. In some cases it may be necessary to also use plywood sheeting where puncture hazards or structural weakness exists.

Inward air movement within the containment structure is an important feature which sets up air flow from clean to contaminated areas. Therefore, if airborne contamination is generated, it cannot leak out of the containment structure. A ventilation unit draws the air from within the work area through HEPA filters and exhausts it to the interior of the building or directly to the building exhaust system.

The ventilation unit intake duct will be positioned nearest to the contaminated work area. Thus, air taken from the interior of the work area will be replaced by air moving into it from the anteroom; that air, in turn, will be replaced by clean air moving in from the environment outside the containment structure. Ventilation units will be designed to provide a minimum of one complete change in the air within the structure every 5 to 10 minutes, (6 to 12 air changes per hour).

Roughing filters will be placed at the beginning of the intake duct to collect most of the airborne debris and thereby prolong the functioning life of the HEPA filter. This will also reduce the internal contamination of the air duct and HEPA ventilation unit.

Prior to actual decommissioning work, the proper functioning of the containment structure and HEPA ventilation system must be verified. Upon initial installation of HEPA filters or vacuum cleaners within a ventilation unit, the integrity of the filter and the filter-unit seals must be checked. Systems containing HEPA filters should be tested with 0.7 micron (or smaller) DOP particle aerosol and be at least 99.95% efficient. Additionally, any HEPA filters used must be certified in writing by the filter manufacturer as meeting 99.97% efficiency for 0.3 micron DOP particles.

Prior to use, the containment structure's function will be tested with smoke generating tubes. With the HEPA ventilation units operating, smoke is generated outside the containment near any potential leakage points or penetrations to make sure that any leakage is inward.

During actual performance of decontamination tasks, the containment structures performance should be verified and documented with an air sampling program. That is, air samples for airborne contamination should be taken in the Clean Area at potential release points. These areas would include both outside the anteroom and the HEPA ventilation air discharge area. Because a contamination release could occur at any time, air sampling will be performed as long as the release potential exists. Typically, a sample will be taken at each location over the duration of the work shift and another taken over the duration of non-work hours. In addition to the air sampling program, a routine program of contamination surveys will be performed on the floor areas within the Transition and Clean Areas to determine if loose contamination is migrating from the Contaminated Area.

2.1.2.3 Dose Control Program

The purpose of the Dose Control Program is to maintain exposures ALARA by means of administrative limits and controls, procedural cutoffs, monitoring and surveys, and postings.

In order to assure that employees are not exposed to levels of radiation in excess of federal and state limits, and in an attempt to keep all exposure as low as reasonable achievable, guidelines are established to assist in the management of personnel radiation exposure. These guidelines are shown below.

	Four Weeks	Eight Weeks	Ten Weeks
Whole Body	500 mRem	800 mRem	1,000 mRem
Extremities	7,500 mRem	10,000 mRem	12,500 mRem

If an individual's cumulative exposure reaches either guideline, it will be the responsibility of the R.S.O. to plan a restricted exposure schedule for that individual.

Reports of exposure are listed for each individual in an exposure history record, which clearly shows his total lifetime and permitted occupational exposure. All radiation workers (Cintichem and contractors) will receive a medical examination (including whole body counting) before starting and upon completion of work at Cintichem. Special whole body counts are obtained whenever it appears that an individual may have been exposed to excessive airborne radioactivity.

An integral part of the Dose Control Program are the radiation surveys in areas where radiation levels above background could exist. Routine and/or continuous radiation surveys will be taken to help maintain exposures ALARA.

Personnel are monitored for gamma and beta dose using TLD and/or film dosimetry. The personnel monitoring program is designed to comply with 10 CFR 20.202 and New York State Code Rule 38. In addition, personnel wear gamma pocket chambers that are evaluated daily. In this way, exposure can be maintained ALARA by lowering any exposure trends as they develop.

For high dose rate situations personnel are provided with dosimeters of a range appropriate to the dose rates encountered. For certain applications, electronic digital dosimeters are used.

Access to high radiation areas is controlled by Health Physics via special authorization, using standardized radiation protection requirements or through lock and key control and mandatory Health Physics surveillance. Stay-time and number of workers exposed is determined by good ALARA practices (i.e., criteria of ALARA for total person-rems).

Cintichem utilizes a formal posting and labeling program. This program is designed to comply with the requirements of 10 CFR 20.203 and New York State Code Rule 38.

2.1.2.4 ALARA Control Program

Cintichem's policy requires that all radiation exposures, which result from plant decommissioning operations, will be kept as low as is reasonably achievable below regulatory limits. The program is implemented by the Radiation Safety/ALARA Committee. The committee organization and ALARA goals are delineated in Appendix C.

The measures which will be taken to mitigate and monitor typical hazards which may arise during the decommissioning are listed below:

HAZARD	MITIGATION	MONITORING
airborne dust/ radionuclides	°water fogger °respirators °HEPA filtration units °supplied air system	°whole-body counts °continuous air sampling °grab air sampling
beta-gamma exposure rates	°limited access °shielding, distance, time	°personnel dosimetry °surveys
loose surface contamination	°anti-contamination clothing °cleanup/decontamination	°frisking °swipe surveys
airborne hazardous chemicals/vapors	°respirators °cleanup/decontamination °supplied air system	°air sampling

2.1.2.5 Environmental Program

Cintichem conducts a continuous radiological Environmental Monitoring Program to assess the impact of site operations on the surrounding environment. Samples of the surrounding environment and site effluents are collected on a routine basis and analyzed to determine if radiation and radioactive materials are present in the exposure pathways. The program is conducted as specified in Table 7.1. It is directed by the Environmental Monitoring Supervisor, who reports to the Manager of Health Safety and Environmental Affairs. His specific duties include:

- Planning and scheduling of the Health Physics Technicians;
- Assuring that surveys are taken in accordance with the program;
- Assures that the minimum detectable limits for radioisotopes are met;
- Assures that the measurable concentrations of radioisotopes and levels of radiation are not higher than expected on the basis of effluent measurements and modeling of the environmental exposure pathways;
- Provides quantitative analysis of the data collected.

2.1.3 Health Physics Instrumentation

Health Physics has fixed and portable instrumentation and a facility to calibrate them.

The maintenance facility for instrumentation is a fully equipped electronics shop staffed by an electronics technician.

The Health Physics laboratory facility is a room fully dedicated to the evaluation of air, water, soil, and surface contamination radioactivity. Instruments include those for measurement of alpha, beta, and gamma radioactivity including a multichannel analyzer with a sodium iodide crystal or germanium lithium crystal for gamma spectroscopy.

The following radiation detectors and monitors (or their equivalents) are typical. They will be calibrated periodically by written procedures.

- Portable Ionization Chambers, 0-5000 mr/hr
- Extendable Geiger-Mueller (GM) exposure rate meter, 4-13 feet, 0-1000 R/hr
- Sodium-Iodide Micro-R meter, 0-5000 micro-R/hr
- Shielded thin window GM pancake detectors with portable count rate meters (0-500K cpm, typical sensitivity approximately 3000-4000 beta dpm/100 cm^2)
- Zinc-sulfide alpha scintillation detectors (75 cm^2) with portable count rate meters (0-500k cpm, typical sensitivity approximately 70 alpha dpm/100 cm^2)
- Area radiation alarm-meters with energy compensated GM detectors 0.1-1000 mR/hr
- Portable alarming rate meters with shielded thin window GM pancake detectors (for personnel frisking stations) (0-500k cpm, typical sensitivity 3000-4000 beta dpm/100 cm^2)
- Portal monitor
- High volume Air Grab sampler, 20 cfm
- Low volume continuous air samplers, variable 0-5 cfm
- Breathing Zone personnel air samples, variable 0-5 liters/min
- High purity germanium detection system with MCA, printer and associated electronics and software
- Low background alpha-beta proportional counter
- Self-reading pocket dosimeters, 0-200 mR and 0-5 R, with chargers
- TLD badges for personnel beta-gamma exposure monitoring, (with service from an outside vendor)

- o Whole body counter services for personnel internal monitoring (service from outside contractor)

2.1.4 Records and Reports

The Health Physics records that will be taken are indicated below:

- o All releases of radioactivity;
- o Health Physics audits;
- o Instrument calibration;
- o ALARA reviews;
- o Employee training;
- o Personnel exposure;
- o Radiation surveys;
- o Environmental surveys.

2.2 Industrial Safety and Hygiene Program

The Industrial Safety Specialist maintains a strong industrial safety and hygiene program and will take all necessary precautions to assure the safe completion of the dismantling and decommissioning operation. The Industrial Safety Specialist duties include:

- o Ensuring the proper use of personal protection devices;
- o Managing the Hearing Conservation Program;
- o Managing the Respiratory Protection Program;
- o Ensuring fire protection and prevention for the facility;
- o Ensuring the proper use of hand and power tools, cutting equipment, and lifting equipment;
- o Ensuring that excavations, work in confined areas, and asbestos removal are performed properly.

The Industrial Safety and Hygiene Program during decommissioning activities will be implemented in accordance with 29 CFR Part 1910, "Occupational Safety and Health Standards", and 29 CFR Part 1926, "Safety and Health Regulations for Construction", as well as Cintichem's and Hoffmann-La Roche's Safety Standards.

All significant dismantling operations will be controlled by written procedures. The procedures will be reviewed by the Nuclear Safeguards Committee and/or the ALARA Committee. These operations will be approved by Level II management. Special Hazards Work Permits will be issued where required for such operations as burning, welding, cutting and electrical lockout.

Industrial hazards will include the handling of decontaminating chemical agents, cutting with oxy-acetylene and arc-type torches, heavy rigging for component removal, and the routine industrial hazards normally associated with construction or dismantlement. Industrial hazards will be minimized by close administrative control and supervision of all work, training, and written procedures.

In the event of an accident, the site First Aid Squad and certain members of Health Physics and Safety personnel are trained to provide first-aid and prompt response to an accident situation.

The following sections present typical provisions for industrial safety and hygiene which will be implemented during decommissioning activities. Several of these provisions were already considered in the task/activity concepts presented in this decommissioning plan and will be further applied during the detailed procedures development.

2.2.1 Occupational Health and Environmental Control

The occupational health of workers during decommissioning activities will be protected by providing adequate facilities and systems as follows:

- o First aid supplies within work area;
- o Emergency shower and eye wash facilities;
- o Trained first aid personnel or ambulance service shall be on site or readily available;
- o Environmental controls in the work space to include adequate ventilation and dust control, temperature control, illumination, noise control, portable water and sanitary facilities;
- o Fire protection in accordance with Section 2.2.5.

2.2.2 Personal Protective Devices

Protective devices provided for workers involved in decommissioning activities will include:

- o Helmets and safety shoes for protection against impact and penetration of falling or flying objects;

- Hearing protection devices;

- Eye and face protection for workers exposed to injury from physical, chemical or radiation agents;

- Respiratory protection devices.

2.2.3 Hearing Conservation Program

A hearing conservation program will be established by Cintichem for all workers who are exposed to noise levels of 85 DBA or greater (as an 8-hour, time-weighted average exposure). It will be in accordance with 29CFR1910.95. Noise control measures, including the requirement to wear hearing protection equipment, will be determined by the Manager of Health Safety and Environmental Affairs. Personnel who are assigned tasks in known noise hazardous areas ($\geq$ 90 DBA) will be required to use hearing protection devices. Records will be maintained that document all noise monitoring conducted, employee training done, control measures implemented, and protective equipment issued.

2.2.4 Respiratory Protection Program

Cintichem's "Respiratory Protection Program" (Safety Standard #3/Health P'ysics Procedure #HP-N-03) will be followed to prevent employee exposure to occupational dusts, fumes, mists, radionuclides, gases and vapors above OSHA and USNRC limits.

Respiratory protection measures, including the requirement to wear respirators, will be determined by the Manager of Health Safety and Environmental Affairs.

Only respirators approved by NIOSH or MSHA will be used.

Records will be maintained that document all air monitoring conducted, employee training done, medical monitoring done, control measures implemented and protective equipment used.

2.2.5 Fire Protection and Prevention

The Tuxedo Fire Company is located 1/4 mile from the site and has been made aware of the decommissioning program and its hazards.

Fire protection devices will be made available during decommissioning tasks. Portable fire extinguishers will be strategically located throughout the site to serve areas for the various decommissioning activities. An appropriate number of Cintichem employees will be trained in the use of fire extinguishers to assure that each shift is manned with trained people. Site wide fire drills will be conducted annually.

Fire prevention measures will be implemented to avoid ignition hazards from electrical wiring and equipment and from combustible materials. No smoking will be permitted in areas where a potential fire hazard is present.

Cintichem's "Permit for Burning, Welding and Cutting" (Safety Standard #2/Health Physics Procedure #HP-N-02) will be followed for burning, welding, cutting and other fire potential operations.

2.2.6 Hand and Power Tools and Cutting Equipment

The condition of the hand and power tools used during decommissioning activities will be routinely checked for proper operation and for utilization in compliance with the applicable provisions of 29 CFR 1926 Subpart I, "Tools-Hand and Power", 29 CFR 1926 Subpart J, "Welding and Cutting" and 29 CFR 1910 Subpart P, "Hand and Portable Powered Tools and Other Hand-held Equipment".

2.2.7 Lifting Equipment

Lifting equipment used in the decommissioning activities will comply with the applicable provisions of 29 CFR 1926 Subpart N, "Cranes, Derricks, Hoists, Elevators and Conveyors", and 29 CFR 1910 Subpart N, "Materials Handling and Storage". We will follow our existing maintenance program.

2.2.8 Excavations

Excavations required during decommissioning activities will comply with applicable provisions of 29 CFR 1926, Subpart P, "Excavations, Trenching and Shoring".

2.2.9 Working in Confined Space Areas

Operations required in the confined spaces will comply with applicable provisions of 29 CFR 1910.94.

2.2.10 Asbestos Removal

Asbestos removal operations required during decommissioning activities will comply with applicable provisions of OSHA Standards 29 CFR 1926.58; New York State Department of Labor, Article #30 and 12 NYCRR Part 56; and New York State Department of Health Title 10 Part 73.

2.3 Contractor Safety

Due to the specific expertise of the contractors they will supply the personnel and equipment necessary to accomplish their individual tasks as outlined in Section 1.5.4. Cintichem will, however, retain responsibility for health and safety during all aspects of decommissioning and will assure that contractors, their personnel, and equipment meet necessary standards.

In addition, Cintichem will provide industrial safety indoctrinations and, when applicable, radiation safety indoctrinations for contractors and their personnel. Cintichem will also require medical examinations and, when applicable, whole body counts for contractors and their personnel as indicated in Section 2.1.2.3.

2.4 Cost Estimate and Funding

The estimated cost associated with the decommissioning program described in this plan is $20,482,000 in 1990 dollars excluding contingency (Ref. 1). Table 2-1 provides a breakdown of the estimated costs. This cost estimate has been prepared in accordance with guidelines given in reference 2.

The decommissioning funds will be provided by Hoffmann-LaRoche (HLR), Cintichem's parent company. Included in Appendix D is a letter from HLR indicating their commitment to funding the decommissioning and a financial statement.

TABLE 2.1

SUMMARY OF CINTICHEM DECOMMISSIONING COSTS

Cost Category	Cost (Thousands $)	Percent of Total Cost
Activity Dependent (1)		
o Decontamination-Dismantling	4,321	21.10
o Disposal-Burial	5,781	28.22
o Waste Transportation	311	1.52
o Waste Packages	662	3.23
Period Dependent (2)		
o Project Management - Technical Support	4,017	19.61
o Support Labor	1,668	8.14
o Equipment Rental - Services	2,251	10.99
Collateral Costs (3)		
o Site Preparations	597	2.91
o Equipment	658	3.21
o Engineering/Technical Services	216	1.07
SUBTOTAL	20,482	100.00
Contingency (4)		
o Activity Dependent	2,125	10.37
o Period Dependent	2,226	10.87
o Collateral Costs	147	0.72
TOTAL	24,980	

TABLE 2-1 FOOTNOTES:

(1) Activity-Dependent Costs

Activity-dependent costs are those that are directly related to discrete activities, e.g., decontamination, removal, packaging, shipping and disposal. They include all labor, materials, energy, equipment and services (shipping and disposal) associated with the "hands-on" activities.

(2) Period-Dependent Costs

Period-dependent costs are those associated with project management, administration, routine maintenance, radiological, environmental and industrial safety, security and can include special support equipment rental and operators. They are not directly assignable to any one activity, but are essentially period-dependent, i.e., these costs continue for the duration of the decommissioning program or period.

(3) Collateral Costs

Collateral or special item costs are typically one-time costs that are neither attributable directly to a specific decommissioning activity, nor period-dependent. This category includes such items as heavy equipment purchase, health and safety supplies and facility/site preparations.

(4) Contingency Costs

These costs are allocated to account for unforeseen increases in costs that cannot be identified prior to conduct of the work and which are beyond the control of project management. Typical reasons for contingency include changes in project schedule due to weather, labor, or regulatory delays, greater than estimated contamination levels or areas, increased work difficulty, or changes of disposal, transportation or packaging cost.

References:

(1) "Cintichem Reactor and Hot Laboratory Decommissioning Cost Estimate", prepared by TLG Engineering, Inc., Bridgewater, CT, September 1990

(2) National Environmental Studies Project - "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates", AIF/NESP-036, May 1986, TLG Engineering, Inc.

3.0 Dismantling and Decontamination (D&D) Tasks and Schedules

Cintichem Inc., intends to remove radioactivity within the reactor and hot lab buildings, associated contaminated systems exterior to these buildings, and areas immediately adjacent to them, to permit unrestricted release and license termination. The decommissioning program will be conducted in two phases; (1) removal of contaminated and uncontaminated tools, equipment and fixtures and spent fuel under the current operating license authority and, (2) dismantling and/or decontamination of systems and structures under the plan provided herein. Phase One is currently underway and, in addition to removal of unneeded materials, detailed site radiological characterization and detailed decommissioning engineering is being conducted. After license termination it is not expected that the buildings will be left in a usable condition. All above grade building structures will be demolished, with clean rubble being disposed of in the building's subsurface void. Where applicable, subsurface structures found to meet release criteria will remain.

3.1 Phase One Activities

3.1.1 Items to be Removed Prior to Dismantling and Decontamination

Unneeded equipment and material not associated with support of the facilities licensed condition will have been removed prior to the start of D&D activities. Items will be removed only if they are judged to be not necessary for decommissioning, safety, or the licensed condition of the facility. Primarily these items consist of laboratory equipment, glove boxes and unneeded supplies and tools. Removal of such equipment will allow unencumbered site radiological characterization and site preparations for Phase Two activities. The removal of these items is being accomplished by Cintichem staff. Contaminated materials are being removed under current license authority. These items will be disposed of at a licensed disposal site by burial or at a licensed off-site decontamination-recycle/disposal facility center (such as Quadrex's Knoxville, Tennessee facility).

3.1.2 Site Preparations for Decommissioning

The decommissioning program described herein will require that several tasks be performed prior to start of Phase Two decommissioning activities, to prepare the facility for decommissioning. These tasks will be performed as permitted under Cintichem's current license authority. These tasks will include the following:

- Spent fuel removal;
- Construction of a Radwaste laydown - shipping area;

- Installation of security fences and personnel/equipment portals;

- Installation of site water control and divert runoff from future decommissioning excavation areas;

- Installation of HEPA filtration in the currently unfiltered exhaust air systems and modify reactor building instrument air system to allow control of HVAC system;

- Contamination reduction by nondestructive means where significant levels of loose radioactive surface contamination exists, where needed;

- Relocate emergency generators out of active decommissioning areas and re-route essential utility services that will be disrupted by decommissioning activities.

3.2 Phase Two: Systems and Structures to be Dismantled and/or Decontaminated During Decommissioning

This section of the decommissioning plan describes typical methods that will be used to dismantle and decontaminate the reactor and hot lab facilities. Decommissioning activities will be implemented in accordance with provisions of specific work packages for each individual activity. Approved work packages will specify the sequence of activities to be accomplished and stipulate acceptable methods and procedures for completion of specific tasks. Decommissioning work packages and procedures will be reviewed and approved in accordance with Cintichem administrative practices. The underlying purpose of all the decommissioning procedures will be successful completion of the work activities in a controlled and safe manner.

Equipment, systems and structures to be decontaminated and/or dismantled and disposed of as part of the decommissioning work scope are listed in Table 3.1. Items were listed in Table 3.1 based upon operational history, preliminary radiological characterization data, and conservative assumptions based upon prior experience of other facilities having undergone decommissioning. As such, it is possible that the content or extent of this list could change somewhat based on the results of the ongoing radiological characterization and results found during the decommissioning process. The physical location and a description of the items listed in Table 3.1 is given in Section 1.1.1 "Cintichem Reactor and Hot Laboratory Facility".

TABLE 3.1
MAJOR DECOMMISSIONING WORK SCOPE ITEMS

REACTOR BUILDING

- o Reactor core structure and components (Activated);
- o Biological shield (activated concrete and steel embedments);
- o Activated experimental facilities (i.e. beam tubes, pneumatic rabbits, thermal column);
- o Contaminated holdup tank and surrounding soil/backfill;
- o Reactor pool/stall contaminated surfaces;
- o Contaminated canal/gamma pit structures and underlying soil/backfill;
- o Contaminated reactor building structural surfaces;
- o Contaminated reactor building systems:
  - o HVAC exhaust;
  - o Primary coolant;
  - o Coolant purification;
  - o Floor drain and pool gutter collection;
- o Primary water storage tank, piping and surrounding soil;
- o Storage tubes (embedded in reactor building South wall);
- o Storage tubes (embedded in reactor building West wall).

HOT LABORATORY BUILDING

- o Hot cells (five);
- o Contaminated structural surfaces;

TABLE 3.1
MAJOR DECOMMISSIONING WORK SCOPE ITEMS (Continued)

- o Class "B" waste storage facility;
- o Liquid waste evaporator;
- o Buried hot cell exhaust ducting, filter room and surrounding contaminated soil;
- o Hot lab HVAC exhaust ducts;
- o T-1 and evaporator rooms contaminated surfaces and surrounding soil;
- o Hot lab floor drain and waste collection systems.

<u>OUTSIDE AREAS</u>

- o 5,000 gallon waste water holding tanks (two);
- o Class A storage building surfaces;
- o Exterior exhaust ducts, blowers and discharge stack;
- o Miscellaneous buried piping, manholes and collection basins and surrounding soil.

3.3 Phase Two: Decommissioning Task Descriptions and Analysis

3.3.1 Decommissioning Objectives, Tasks and Activities

The decommissioning objectives are to remove the radioactivity in the Cintichem reactor and hot laboratory buildings to meet US Nuclear Regulatory Commission, and New York State Department of Labor decommissioning criteria. Where criteria between these agencies differ, the more stringent criteria will be used. To accomplish these objectives, 37 major tasks are planned. Task numbers are assigned by relationship in the Pert chart (Figure 3.1) and are not numbered consecutively in an order in which they would be expected to be accomplished. For these relationships consult the Gantt chart (see Figure 3.2). These tasks are summarized as follows:

3.3.1.1 Remove Core and Systems - Task #1

This decommissioning task will entail removal of activated and/or contaminated reactor core components. The major items to be removed include:

- Reactor Core Support Tower;
- Core Grid Plate;
- Cooling Water Plenum and Flapper Valve Assembly;
- Core Outlet Assembly (at the stall operating position);
- Shim-rod, Regulating Rod and Fission Chamber Drive Mechanisms and Accessories;
- Pneumatic Rabbit Tubes;
- Thermal Column Shield Assembly.

Due to the high radiation levels associated with the reactor components they will be disassembled remotely underwater. Initial disassembly of the reactor components will take place in the reactor pool. Individual components or portions of them will be transferred via the canal to the gamma pit where a segmentation and cask loading station will be set up. This will be performed such that components can be segmented to fit into Cintichem's Model Nos. B-3 and BMI-1 shipping casks (or equivalents) which will be loaded at the gamma pit. Some slightly activated or only contaminated materials, which have lower associated dose rates may be packaged in standard LSA containers. Expected gamma dose rates on components will be verified and components will be sorted for appropriate packaging.

The cooling water plenum assembly will be grappled, rigged to the overhead crane and remotely unbolted from the core grid plate. The plenum assembly will be moved to the transfer canal, rerigged to the canal hoist and transferred to the gamma pit.

At the gamma pit, the plenum/flapper valve assembly will be segmented, as needed, to be efficiently and appropriately packaged into Cintichem's shipping casks.

The reactor core grid plate will be grappled, rigged to the overhead crane, remotely unbolted from the core support tower and moved to the transfer canal. At the transfer canal, the grid plate will be rerigged to the canal hoist and moved to the gamma pit area where it will be loaded into Cintichem's B-3 shipping cask without segmentation.

The drive mechanisms and accessories, and rabbit tubes will be grappled and cut free using remote hydraulic shears, saws and/or plasma arc thermal cutting. These will be transferred to the gamma pit in manageable sized sections, and segmented to fit into shipping containers.

The activated sections of the core support tower will be grappled and rigged to the overhead crane and segmented by hydraulic shearing or other mechanical means into manageable sized sections. These sections will be transferred to the gamma pit where further segmentation will take place to fit the tower's structural members efficiently into shipping casks.

The core outlet assembly at the stall position will be unbolted from the stall floor remotely, grappled and transferred to the gamma pit for packaging.

The core outlet assembly at the pool position was never used and is therefore not activated and will be dealt with when the pool is decontaminated.

The thermal column shield assembly consists of a lead and canned graphite structure, which is supported by an aluminum frame, between the reactor core and the thermal column. The activated lead shield will be grappled and then unbolted from the supporting frame. The lead shield will be moved to the canal/gamma pit for temporary storage. Dependent upon dose rates actually found on the lead shield, it will be placed into an LSA container or a shielded shipping cask for storage until a mixed waste disposal site becomes available. The canned graphite assembly will be rigged and transferred to the canal/gamma pit. If needed, to accommodate packaging constraints, it will be cut open under water using a circular saw, hydraulic powered chisel and/or plasma torch. The resulting pieces of aluminum casing and graphite blocks will be sorted and placed into shipping containers.

The thermal column shield assembly stand will be removed, segmented, and packaged in a manner similar to the core support tower.

3.3.1.2 Remove Storage Tubes-South Wall Reactor Building - Task #2

The storage tubes in the South wall of the reactor building will be removed in the following sequence:

- o Plug each tube;
- o Core drill around outside of each tube;
- o Remove tube;
- o Package contaminated tubes for proper disposal.

The tubes are encased in a monolithic concrete block extending outside the reactor building wall. The cover plate will be removed and an expandable pipe plug inserted in the end of the tube. A core will be taken around the outside of the tube for its entire length. A plate/shaft will next be welded to the end of the tube with the shaft being torqued to break the concrete at the far end of the tube. The freed tube will be removed and packaged in an LSA container for disposal. This sequence will be repeated until all storage tubes in the South wall have been removed.

The concrete surrounding the removed storage tubes is not expected to be contaminated. However, it will be radiologically surveyed. If found to be contaminated, it will be hydraulically split and packaged for disposal as contaminated concrete rubble prior to removal of the South wall during Task #36.

3.3.1.3 Remove Thermal Column Graphite Blocks - Task #3

The following activities are expected to occur during this decommissioning task:

- o Remove Thermal Column Door Assembly;
- o Remove Steel Shielding Assembly;
- o Remove Lead Shielding Blocks;
- o Remove Graphite Blocks;
- o Remove Concrete Vertical Access Door;
- o Remove Vertical Access Lead Shield;
- o Remove Face Plate;
- o Remove Face Plate Lead Gasket.

HEPA ventilated contamination control containments will be erected at the end of the horizontal thermal column access and above the vertical access door, and temporary shielding will be installed as necessary to protect workers. The thermal column door assembly and the concrete vertical access door assembly will be removed and if found to contain areas of contaminants will be decontaminated by scabbling, as necessary, and removed for disposal as clean material.

The lead shield wall (stacked blocks which are expected to be activated), and the plug assembly, will be removed and loaded into a shipping cask for future disposal (as a mixed waste). The graphite blocks will be removed individually and packaged for disposal in steel LSA boxes. Following completion of graphite block removal, the vertical access lead shield door will be removed and packaged.

After the reactor pool has been drained, the activated inner aluminum face plate will be removed by unbolting and packaged for disposal. The activated lead gasket will be removed and packaged separately with the lead blocks from the outer shield wall.

3.3.1.4 Remove Misc. Piping (Stall/Pool) - Task #4

The following activities are planned to occur during this decommissioning task:

- o Remove Exterior Bio-Shield Wall Gutter Drain System;
- o Remove External Piping from Stall/Pool Bio-Shield;
- o Remove Reactor Building Cold Water Piping;
- o Remove Reactor Building Demineralized Water Piping;
- o Remove Reactor Building Instrument Air Piping;
- o Remove Reactor Building High Pressure Instrument Air Piping;
- o Remove Pool Water Piping;
- o Remove Pool Beam Tube Recirculation Pump/Piping.

Utilizing HEPA contamination control containments erected during other tasks of decommissioning, the miscellaneous piping installed on the bio-shield will be removed using mechanical cutting methods (see Section 3.3.2). After draining and drying of these systems, the reactor pool gutter and drain system will be removed and packaged for shipment to an off-site decontamination-recycle facility (such as the Quadrex recycle center in Knoxville, Tennessee). The external piping (cold, demineralized, pool water pipes, and the instrument and high pressure instrument air) will be segmented and containerized for shipment to the off-site decontamination-recycle/disposal facility.

The beam tube pool water recirculation piping and pump will be completely removed to the penetration into the shield wall and the pipes capped. The embedded portions of this piping system will be removed during other tasks cf decommissioning.

3.3.1.5 Remove Activated Concrete - Task #20

The following activities are planned for this decommissioning task:

- o Remove Activated Concrete in Bio-Shield Stall Walls;
- o Remove Activated Concrete in Floor of Reactor Stall;
- o Remove Activated Concrete around Beam Tubes;
- o Remove Activated Concrete around Thermal Column.

A HEPA ventilated contamination control containment will be erected above the pool/stall prior to commencing removal activities within the pool/stall.

Activated wall concrete surrounding the core region will be removed by use of expanding grout (such as "Bristar") methods or hydraulic rock splitting methods, to fracture the activated concrete. The wall areas of the stall will be drilled completely from above using track drilling methods to form the holes for insertion of expanding grout or splitters before removal of activated concrete commences. To preclude excessive personnel exposure during drilling of vertical holes in the reactor stall, pool water will be lowered only enough to expose the top of the reactor stall shelf, thereby shielding the drill operator. The resulting activated rubble will be further fractured and removed from the pool floor using a ramhoe and clamshell bucket attached to a small hydraulic excavator.

The floor will also be drilled using track drilling methods and the activated concrete rubblized by expanding grout, or hydraulic splitting methods.

Activated concrete surrounding the beam tubes and the thermal column will be removed (after the pool has been drained) utilizing horizontal track drilling methods in patterns which will allow removal in the pattern of the embedments being removed. Horizontal holes will be drilled through the bio-shield from outside of the bio-shield to allow the drill operator to work in a low radiation area. Hydraulic splitting or expanding grout methods will be used to rubblize the concrete from the shield wall and any remaining steel embedments (primary steel waterseal, embedded piping, reinforcing steel, etc.) will be segmented using thermal torch cutting methods.

3.3.1.6 Remove Beam Tubes - Task #11

The following activities are planned to occur during this decommissioning task:

- o Erect Contamination Control Containments;

- Remove Lead and/or Concrete Block Shielding;
- Remove Four 6" Beam Tubes;
- Remove Two 8" Beam Tubes;
- Remove Four 6" Beam Tube Liners;
- Remove Two 8" Beam Tube Liners.

Following erection of HEPA ventilated contamination control containments, the plugs will be removed from the beam tubes and packaged for disposal. The beam tubes will be unbolted for removal, removed from the beam tube liners, and segmented for packaging and disposal.

Two of the six inch beam tube assemblies have square outer ports. These will be segmented by torch cutting and removed mechanically, exposing the inner circular liners. The outer ports for the remaining four liners will be removed by torch cutting and mechanical means. The six circular liners will be removed by concrete core drilling methods. A hollow concrete diamond core drill slightly larger than tube diameter will be used to over-drill the concrete that surrounds the liner along its length until penetration of the bio-shield occurs. The liner will be pulled out of the bio-shield and packaged for disposal. The beam tube associated piping will be removed in Task #19 "Embedded Pipe Pool/Stall".

3.3.1.7 Remove Primary and Secondary Reactor Systems - Task #12

The following activities are planned to occur during this decommissioning task:

- Remove Inlet/Outlet Embedments in Pool/Stall Floor;
- Remove 10" Stainless Steel Inlet and Outlet Piping;
- Remove 4" Embedded Gutter Drain System Piping;
- Remove Concrete Floor above Primary Piping (Outside of Bio-Shield at 781' elevation);
- Remove Primary Piping (Concrete Encased to Pipe Flange Pit);
- Remove Concrete Floor above Primary Piping (Pit to Pump Room);
- Remove Sand Fill Surrounding Piping in Trench;
- Remove Aluminum Piping from Pipe Flange Pit to Pump Room.

Contamination control containments erected for tasks within the pool/stall and the removal of beam tube and other embedments will be utilized for this decommissioning task.

The embedments for the inlet and outlet piping in the floor of the pool/stall will be removed by mechanical demolition means (such as drilling and using hydraulic rock splitting, ramhoe breaking or expanding grout methods). The concrete above the primary inlet and outlet pipe runs will be removed down to the steel waterstop liner, and the liner cut by thermal cutting methods. Unless activated during reactor operations the removed steel liner will be surveyed and released for disposal as non-contaminated material. Concrete below the steel waterstop liner and surrounding the inlet and outlet piping will be removed with drilling and splitting methods. The exposed stainless steel piping will be removed in sections suitable for packaging and shipped to the off-site decontamination-recycle facility.

The abandoned gutter drains will be removed by exposing the pipes embedded in the bio-shield wall by removing the surrounding clean concrete, thermal cutting the piping, and packaging for disposal. Following removal of embedded piping within the pool/stall, the floor above the primary inlet and outlet piping exterior to the bio-shield wall will be removed to expose the concrete encased pipe that runs below grade. The concrete floor surrounding the inlet and outlet piping will be removed from the bio-shield to the pipe pit. The piping will be removed and packaged for shipment to a off-site decontamination-recycle facility.

The concrete floor above the pipe trench from the pit to the pump room wall will be segmented, removed and radiologically characterized. The sand fill surrounding the piping in the trench will be removed and packaged for disposal, if found to be contaminated. The aluminum piping in the trench will be removed by mechanical or thermal cutting and packaged for shipment to a decontamination-recycle facility. The pipe trench structure will be radiologically characterized and if needed, decontaminated by concrete scabbling.

3.3.1.8 Remove Thermal Column Liner - Task #13

The following activities are expected to occur during this decommissioning task:

o Remove Vertical Accessway Compartment Steel Liner;

o Remove Horizontal Aluminum/Steel Liner.

The vertical accessway steel liner will be segmented from the interior using a cutting torch to free the steel panels. The liner will be pried free and packaged for disposal. Following the removal of the vertical portion of the thermal column, the horizontal portion of the liner will be segmented, pried free, and packaged for removal. Activated portions of the steel support structure will be removed by torch cutting after removal of surrounding concrete (in Task #20).

3.3.1.9 Remove Embedded Piping (Pool/Stall) - Task #19

The following activities are planned for this decommissioning task:

- o Remove Beam Tube Vents/Drains;
- o Remove Thermal Column Vents;
- o Remove Beam Tube Pool Circulation;
- o Remove Primary System;
- o Remove Pneumatic Rabbit Tubes.

The HEPA ventilated contamination control containment over the top of the reactor pool used for other decommissioning work will be used for this task.

The embedded piping systems in the pool and stall bio-shield will be removed by mechanical means such as core drilling or hydraulic concrete splitting. The areas of the bio-shield containing embedded pipe will be drilled and the concrete removed with hydraulic splitters. The piping passing through the steel liner will be cut using thermal cutting methods to remove contaminated or activated sections of the piping system and any activated sections of the steel liner. All header systems and vents and drains carrying or in contact with contaminated fluids or gasses will be removed from the shield wall.

3.3.1.10 Remove West Wall Storage Tubes (Below Canal Structure) - Task #16

The following activities are planned to occur during this decommissioning task:

- o Remove Concrete Surrounding Storage Tube Face Plate Embedments;
- o Remove Storage Tube Door/Face Embedments;
- o Core Drill Storage Tubes;
- o Remove Storage Tube Assemblies.

The clean concrete surrounding the front face of the storage tubes will be removed by jackhammering or by other standard concrete demolition means. The storage tube doors will be removed, an expandable plug inserted into the open storage tube, and the circular front flange plates cut from the storage tubes. The balance of the storage tubes will be removed by core over-drilling using a hollow diamond core drill slightly larger than the storage tubes. When the core drill depth reaches the end of the storage tubes, the core drill will be withdrawn. The concrete/storage tube assemblies will be placed in angular tension to fracture the storage tube from the remaining-in-place concrete. The storage tubes will be withdrawn, segmented if required, and packaged for disposal.

3.3.1.11 Decontaminate Reactor Pump Room - Task #17

The following activities are planned to occur during this decommissioning task:

- o Drain piping systems (primary cooling, secondary cooling and primary purification);
- o Remove demineralizer resins;
- o Remove pipe;
- o Remove lead bricks from around demineralizer;
- o Remove demineralizer, heat exchanger and segment;
- o Remove all other equipment (pumps, valves, motors, gauges, etc.);
- o Remove lead shield in floor above valve room;
- o Cap floor drains in reactor/pump room after potential for spills is past (drain in piping will be removed in a separate sub-task);
- o Remove sump pumps and associated piping;
- o Remove sump stainless steel sump liner
- o Scabble concrete wall/floor surfaces;
- o Remove embedded floor drains/piping;
- o Remove HVAC duct;
- o Remove pump room roof and South wall (common wall with holdup tank);
- o Remove embedded drain line in footing under South wall.

The activities will start with draining of piping fluids and removal of the resins in the demineralizers. Next, all piping is removed (with the exception of the sump discharge pipe) to open work area for equipment removal. Clean and contaminated pipe will be segregated and placed in appropriate container for shipping. The lead bricks around the demineralizer will then be removed to expose the demineralizer tank. The two demineralizers, two open tanks and the primary/secondary heat exchanger will be segmented and packaged as contaminated as required. The other equipment (pumps, valves, motors, etc.) in the pump room will then be removed.

A lead shield in the floor above the valve room will be rigged, cut and removed. If contaminated, this lead shield will be sent to an off-site recycle center for decontamination. The sump in the Southeast corner of the pump room will remain operational until such time that the reactor building drains can be capped off without affecting other decommissioning activities. Next, removal of the sump pumps, piping, and stainless steel sump liner will occur. Decontamination of the concrete floors, walls, and ceiling will be performed by surface removal (scabbling). After scabbling, plastic sheets will be placed over clean areas to protect them from recontamination and the floor cut and removed to expose embedded pipe. The embedded drain pipe under floor and a concrete filled abandoned primary pump suction line will be removed. As a final removal step, all contaminated ducting will be removed. Clean material removed as an encumbrance and contaminated material will be segregated in different containers for disposal.

Soil will be excavated above the pump room to expose the roof. Soil and exposed concrete will be radiologically surveyed during the excavation to verify that they have not been contaminated from HUT leaks. If found contaminated, scabbling of concrete and removal of soil for proper disposal will occur. Stitch core drilling around contaminated pipe/electrical penetrations will be performed to remove embedments. The pump room roof and walls will now be removed to expose potentially embedded contaminated drain lines in the walls and footings. These drain lines will be removed and packaged for proper disposal. Coring through remaining floor to check for residual contamination will occur. If conditions suggest migration of contaminants beneath the pump room floor, the floor will be segmented, lifted, and the underside radiologically surveyed. The underlying soil/fill will be surveyed and if necessary removed for disposal.

3.3.1.12 Decontaminate Pool/Stall - Task #18

Decontamination of the reactor pool/stall bio-shield will commence after activated concrete and components have been removed. The following activities are planned for this decommissioning task:

- o Remove Ceramic Tile;
- o Remove Wall/Floor Embedments;
- o Scarify Concrete Pool Wall/Floor Surfaces.

The remaining ceramic tile in the reactor stall will be removed using mechanical means such as jackhammer chiseling. The underlying grout will be removed completely until the exposed concrete can be radiologically characterized and any small areas of contaminants will be removed by a surface removal scabbling process. The spent fuel storage racks and other embedded hangers will be removed by jackhammering the surrounding concrete.

The reactor pool/stall contaminated concrete walls and floor remaining after the removal of activated materials will be scarified using a scabbler, and vacuumed. A multi-piston scabbler head will be mounted on a small backhoe or attached to a similar mechanically operated manipulator placed in the reactor pool/stall. Corners, edges, or other inaccessible areas will be manually scarified with a single or triple head scabbler. Dust control will be accomplished using a HEPA vacuum shroud system around the scabbler head, and/or a fog spray over the area to be scarified. The exposed surface will be vacuumed to remove any residual contaminated dust.

3.3.1.13 Remove Storage Tank - Task #9

The storage tank will be removed in the following sequence:

- o Drain tank-Hydrolase interior of tank;
- o Remove above ground pipe (fill/supply and vent line);
- o Remove abandoned below ground pipe (fill/supply and heating water lines);
- o Remove tank and foundation;
- o Remove vent line.

The tank, fill/supply line and vent line will be removed as contaminated material. The tank foundation, underground pipes near old abandoned fill/supply line and soil will be radiologically evaluated, and if found to be contaminated, removed and packaged for disposal.

The tank will be drained through the demineralizer system to the mall tanks. An air lock will next be installed for access to the tank. The interior walls and heating coil will be hydrolazed to remove surface contamination. If the walls are still contaminated, vacuum grit blasting may be used to remove contamination, or if vacuum blasting is determined to not be feasible, a containment tent will be erected and the tank walls segmented for removal.

Contaminated piping will be segmented in manageable sections and packaged for disposal. The pipe stubs in the wall of the reactor building will be capped and the walls stitch cored to remove the embedded pipe stubs. The wall will be patched with concrete to maintain reactor building containment integrity. Removal of the embedded pipe stubs will not be done during potentially airborne generating activities.

The tank foundation will be checked for contamination. If found contaminated, the foundation will be scarified. The vent line from the storage tank to the 28 inch exhaust header from the reactor building will be removed in manageable lengths, segmented and placed in LSA boxes for shipment to an off-site recycle center. The vent line will remain active during its removal to supply inward air flow for contamination control during the removal procss. The air flow in the vent line will be throttled to maintain air flow balance in the remainder of the system.

3.3.1.14 Remove Storage Tank Soil - Task #10

Following removal of the storage tank, the soil under and around the tank will be radiologically surveyed. If found to be contaminated, it will be packaged in LSA containers as contaminated soil and shipped for proper disposal.

3.3.1.15 Remove Hold-up Tank - Task #14

The holdup tank will be removed by the following activities:

- Drain water from tank;
- Excavate soil overburden above tank;
- Remove concrete roof of tank;
- Remove primary and vent pipes;
- Decontaminate interior/exterior of tank;
- Remove tank;
- Remove contaminated soil.

Any standing water in the tank will be drained with a temporary sump pump while the liquid radioactive waste system is still operational. Approximately 30 feet of clean soil will be excavated to expose the top and sides of the tank. The existing storm drain system in the area will be modified, sheet piling installed, and temporary dikes installed to divert surface runoff water. A containment tent with portable HEPA filters will be constructed over the tank area. The top of the tank will be removed for access. The roof slab that is removed will be scarified on one or both sides to remove surface contamination as needed. The contaminated primary and vent pipes in the tank will be removed. Pipe stubs that pass through the common hold-up tank-pump room wall will be capped and removed during pump room removal.

The interior portions of the tank's concrete walls will be scarified to decontaminate the tank interior. The exterior portions of the walls will next be scarified as required by radiation survey.

The decontaminated walls and roof of the holdup tank will be removed. The portion of the walls and floor formed by bedrock if found contaminated, will be broken into rubble and placed in LSA boxes for disposal as contaminated concrete. During removal of the tank floor, the old pump suction line from the pipe pit to the reactor building South wall will be removed. A sump pump will be placed in the old pipe pit to collect any ground water that may enter the tank excavation and will be treated by the site waste treatment system.

3.3.1.16 Remove Hold-up Tank Soil/Fill - Task #15

The soil/concrete fill/bedrock under the tank floor will be characterized during floor removal. Contaminated material will be packaged in LSA boxes for proper disposal. Excavated material that meets release criteria, will be stored on site and used as fill for site restoration.

3.3.1.17 Decontaminate Building #1 Structural Surfaces (Reactor Building) - Task #22

The following activities are planned to occur during this decommissioning task:

- o General Area Floors, Walls, Ceiling in Reactor Building;
- o Decontaminate Radiochemical Lab in Building el 808'-3";
- o Decontaminate Counting Room;
- o Decontaminate Q.C. Lab Elev. 798'3".

Activities listed for this task may be scheduled during other tasks or activities as necessary to maintain continuity of the decommissioning process within the reactor building.

Equipment, externally contaminated, will be decontaminated by vacuuming and/or wiping and radiologically surveyed. If areas of contaminants are found remaining, the equipment will be dismantled, packaged as appropriate, and shipped to an off-site decontamination-recycle facility.

Where possible, offices and laboratories will be utilized as their own contamination containments, and anterooms installed at the entrances and suction applied utilizing HEPA ventilation units for inward air flow. The surface areas will be vacuumed, wiped, and any areas of remaining contamination remaining scarified. Following decontamination activities within each office or laboratory, any contaminated air exhaust system serving the areas will be disabled and removed, and packaged for shipment to the recycling facility (see Task #24). Decontaminated surfaces will be covered with plastic sheeting to prevent re-contamination during other tasks and activities.

Where decontamination must occur in open areas, nonairborne producing decontamination techniques will be preferably employed or localized HEPA filtered containment enclosures erected to control airborne radioactivity that could be generated by more destructive decontamination techniques.

3.3.1.18 Decontaminate/Dismantle Reactor Building Exhaust Ventilation - Task #24

The following activities are planned to occur during this decommissioning task:

- o Remove Exhaust System from Storage Tube Area;
- o Remove Exhaust System from Offices/Laboratories/Fume Hoods;
- o Remove Exhaust System from Reactor Bridge (Stall);
- o Remove Concrete Embedded Exhaust Ducts from Reactor Bridge (Pool);
- o Remove Vertical Sections of Exhaust System;
- o Remove Horizontal Sections of Exhaust System (Suspended from Reactor Building Ceiling).

During conduct of this task the exhaust system for the reactor building will remain functional until the last section of exhaust duct is removed where the system penetrates the reactor building wall. The exhaust system will be dismantled in the direction of airflow to maintain inward air flow within the system for contamination control during dismantling. Removal will be accomplished utilizing small glove-bag type containment enclosures where deemed practical and needed to prevent re-contamination of the reactor building structure. Removed material will be volumetrically reduced, either by flattening or segmentation, and packaged for shipment to a decontamination-recycle facility while in a contamination control enclosure.

Prior to removal of main exhaust ducts, the smaller exhaust systems connected to the main exhaust system will be removed first, and the connection at the main exhaust system closed to maintain inward air flow in the system.

The exposed exhaust duct from the reactor bridge location in the stall position will be removed and packaged for shipment. The concrete floor area above the embedded exhaust duct for the reactor bridge's pool position will be removed. Under the floor is concrete fill in which the exhaust duct is embedded. This fill will be removed using standard demolition methods until the exhaust duct is exposed. The concrete rubble from the fill will be radiologically surveyed and if necessary will be packaged in LSA containers and sent for disposal. After completion of the exhaust duct removal, the remaining concrete fill will be vacuumed and radiologically surveyed. If contaminants are not found, the remaining fill will be removed during the demolition of the building structure. If contamination is found to exist, the concrete fill will be removed as radioactive waste.

Once the branch ducts, filters, dampers, etc. have been removed, the remaining vertical and horizontal main exhaust duct system inside the reactor building will be removed and packaged for disposal.

3.3.1.19 Remove Canal/Gamma Pit - Task #5

The following activities are expected to occur during this decommissioning task:

- o Erect Contamination Control Containments;
- o Remove Canal Water;
- o Remove Ceramic Tile in Gamma Pit;
- o Remove Crane Rail, Gates, Elevator Mech, Etc.;
- o Remove Reactor Chem Lab Wall (to provide access);
- o Vacuum, Wipe, Scabble Surfaces;
- o Remove Passageway Floor (flooring next to canal structure);
- o Remove Compacted Fill;
- o Remove Inner Canal/Gamma Pit Wall;
- o Remove Canal/Gamma Pit Floor;
- o Reroute/Repair Footing Drain (Beneath RB/Canal Floor).

Contamination control containment barriers will be erected at the reactor Building passageway, the hot laboratory accessway, and the hot cell charging area passageway and HEPA ventilation units installed. The canal water will be removed and processed in the same manner as the pool water. The ceramic tile located in the gamma pit will be removed by mechanical chiseling. The miscellaneous contaminated equipment in the area such as the canal gates, hot cell and elevator mechanisms will be removed and packaged for shipment to a decontamination-recycle facility. The surface areas of the canal and passageways will be vacuumed, wiped, and scarified as necessary to remove areas of contamination.

The passageway floor will be segmented and removed and the underside radiologically characterized, and if found, surface contaminants will be removed by scabbling. The compacted fill beneath the passageway floor will be removed, and if found to be contaminated, packaged for disposal. If contamination is found on the outer surface of the inner canal/gamma pit wall, the surfaces will be decontaminated (scabbled) as necessary before removal. The inner building wall will be removed and sent for disposal as clean rubble.

The canal/gamma pit concrete floor will be cut into manageable sized segments, lifted and removed from the trench. The underside surfaces will be radiologically characterized. If contamination is found, the concrete slabs will be decontaminated (scarified). The canal/gamma pit floor was poured on concrete fill which was poured directly on bedrock. If the interface between the concrete pours is non-distinct, and proper cleavage of the floor does not occur, it may be necessary to remove the floor of the canal with demolition techniques which will render the concrete to rubble form. Should this be necessary, the canal/gamma pit floor rubble will be packaged in LSA boxes and sent for disposal.

Following removal of the canal/gamma pit floor, the reactor building footing drains which pass below the canal structure inside the reactor building, will be examined to make certain that they are still functional, and if necessary will be replaced or rerouted to maintain their functional status.

The roof above the canal structure will be removed and disposed of as clean material. The outer building wall (which forms one of the canal's wall) will be segmented for removal and the outer surfaces, which form the canal, radiologically surveyed. Contamination, if found, will be removed by scabbling processes. A portion of the outer canal wall was poured directly against bedrock. This area may require some demolition and the concrete, if contaminated, rubblized for removal. Concrete fill located behind these walls will be removed under controlled conditions until bedrock is encountered. Compacted fill will be radiologically surveyed and removed until bedrock is encountered. Compacted fill, or removed concrete fill, that is found contaminated will be packaged in LSA containers for disposal. The roof and outer building wall of the canal will be removed only after the reactor building exhaust line has been removed. This exhaust line is supported by the outer canal wall, which prevents removal of canal roof and outer wall any earlier in the project schedule.

3.3.1.20 Remove Canal Soil/Fill - Task #8

Following removal of the canal structure and the associated concrete or compacted fill behind the walls, the remaining concrete fill below the floor areas will be removed as determined by radiological survey, and sent for disposal as required.

3.3.1.21 Decontamination/Dismantling Hot Cells - Task #6

The following activities are expected to occur during this decommissioning task:

- Erect Contamination Control Containments;

- Preliminary Decontamination of Cell Surfaces;
- Removal of Steel Floor and Platforms;
- Decontamination/Scabble Hot Cell Concrete Surfaces;
- Remove Embedments;
- Remove Exhaust ducts.

Airborne contamination control envelope enclosures will be constructed around the ceiling plug access area, above the hot cell facilities. The underground exhaust duct will remain in operation during decommissioning activities, used to create inward air flow.

The surfaces will be decontaminated by high pressure water rinsing (hydrolased) to reduce the dose rate within the cells to levels acceptable for decommissioning activities. This will entail removal of the 8-12 inch diameter roof plugs above the hot cells and the insertion of high pressure water lances through the plug openings. The walls, ceiling, platforms, and any other accessible surfaces will be hydrolased with high pressure water lances. Liquids will be collected in the existing stainless steel floor liners and transferred to the waste treatment facility.

Upon completion of the hydrolasing activities, and confirmation that dose rates within the cells are at acceptable levels (reduced from the tens or hundreds of R/hr to a few mR/hr range), the charging doors will be opened and the elevated floor platforms removed.

After closing the charging door, the large roof plugs will be removed, scarified, and disposed of as clean material. The walls of each hot cell will be scarified to remove surface contamination. The stainless steel floor liners will be decontaminated and removed.

The embedments (exhaust ducts, storage wells, conveyor ducts, windows, drains, etc.) will be exposed for removal by removing the surrounding concrete covering. The windows will be removed and the oil between the window segments drained and disposed of as non-contaminated material.

The underground exhaust duct (vitreous clay) directly below the hot cells (surrounded by concrete fill) will be exposed for removal by removing structural concrete using standard demolition methods.

3.3.1.22 Decontamination/Dismantling Storage Wells (Hot Lab Building) - Task #7

The following activities are expected to occur during this task of decommissioning:

- Erect Contamination Control Containments;
- Decontaminate Upper Horizontal Surfaces;
- Decontaminate Floor Areas;
- Decontaminate Main Storage Pit;
- Remove and Decontaminate Storage Well Plugs;
- Decontaminate Well Surfaces;
- Decontaminate Duct System;
- Remove Filter Room Duct Header.

Contamination control containment barriers will be erected around the storage well area at the openings to the room area. Using portable HEPA ventilation units inward air flow will be created. The embedded exhaust duct system will be deactivated prior to the decommissioning activities. A sump pump system for collection of decontamination water will be installed in the exhaust duct header sump. A drain line from the header to a temporary waste treatment system will be installed. The upper horizontal surfaces of this building area will be decontaminated using vacuuming and wiping methods. The surface of the floors and the top of the storage well plugs will be scarified and vacuum cleaned.

The plugs will be removed and the concrete surfaces below the floor level (axial and radial) scarified as needed. The bare concrete interior well surfaces where the well plugs mate will be scarified below the floor line, but above the storage well liner.

The main storage pit will be opened, vacuumed and wiped. Any areas with remaining contamination will be identified and decontaminated by the scabbling process.

The lower portion of the storage wells are lined with a galvanized steel liner. The steel liner surfaces of the storage wells will be cleaned using high pressure water lances. The rinse water will be collected through the embedded duct system. After all storage well liners have been decontaminated, the embedded duct system will be hydrolased from the outermost point toward the central header sump. The vertical portion of the central header will be hydrolased, and the drain system removed from the sump area of the central duct header.

If it is found that decontamination of a portion of the storage wells proves to be ineffective, the steel liners will be removed from the storage wells, the imbedded ducts removed by drilling and splitting the concrete so that duct work can be removed for disposal.

3.3.1.23 Dismantling/Removal Underground Exhaust (Hot Cells to Primary Filter Room) - Task #21

The following activities are planned for this decommissioning task:

- o Erect Contamination Control Containments;
- o Remove Floor in Hot Cell Operating Area;
- o Remove Floor in Radiochemistry Laboratory;
- o Remove Overburden;
- o Remove Vitreous Clay Exhaust Duct.

The decontamination of surfaces within the hot cell operating area, and the radiochemistry laboratory will have been accomplished prior to commencing with this task. Protective wall/floor/ceiling coverings will be erected in the hot cell operating area, radiochem lab and on the front exposed face of the hot cells to prevent recontamination of those areas. Contamination control barriers will be erected in the exits of the hot cell operating area, and the radiochemistry laboratory. The underground exhaust duct will remain active during its removal and serve to maintain inward air flow into the enclosed area.

The floor areas immediately above the underground exhaust duct, in the hot cell operating area and radiochemistry laboratory, will be segmented using a concrete saw. The slabs will be lifted and the underside characterized for radiological contamination. If the underside of the floor is contaminated, the slabs' underside will be decontaminated via the scabbling process.

Following the removal of the floor slabs for this task, the soil overburden will be excavated and the exhaust duct exposed using a combination of a small loader and hand excavation methods. The soil will be radiologically surveyed during the removal process, and removed until soil contamination has been eliminated or until structural concerns limit further removal. (If required additional soil will be removed via Task #35).

The exhaust duct will be removed in sections and packaged for disposal. During this process the North, East, and South exterior walls of the underground (T-1) room will also be exposed and if found contaminated will be decontaminated via a scabbling process. The floor slab and soil overburden above the evaporator room will be removed, and if found contaminated, the soil will also be packaged for disposal and the concrete surfaces scarified.

The exhaust duct will be removed from the hot cells to where the duct passes under the building footings under the North wall of the Radiochem Lab. Removal of the remaining pipe section, primary filter room, the underground duct between the primary and polishing filter tank is discussed separately under Task #25

3.3.1.24 Remove Primary Filter Room (Hot Cell Exhaust System) - Task #23

The following activities are planned to occur during this decommissioning task:

- o Erect Contamination Control Containments;
- o Remove Filter Room Internal Equipment;
- o Decontaminate/Scabble Surfaces;
- o Remove Ceilings/Walls;
- o Remove Surrounding Floor Areas (Radwaste Laydown);
- o Remove/Scabble Manhole in Base of Filter Room.

Contamination control containments will be erected surrounding the filter room to facilitate contamination control during the decontamination and dismantling activities. The physical removal of the filter room will occur after removal of internal equipment and the decontamination of its internal surfaces. Decontaminated structural material will be removed and disposed of as clean material and used as concrete rubble backfill during site restoration. The removal of portions of the filter room is necessary to provide access for removal of underground exhaust ducts and the manhole under the filter room.

The filter room's internal equipment (filters, holding brackets, etc.) will be removed and packaged either for disposal, or shipment to a decontamination-recycle facility. The exterior roof and wall of the filter room will be decontaminated by vacuuming, wiping, and/or surfaces removed by a scabbling processes. The internal surfaces of the filter room which are concrete will be decontaminated by scabbling. Once deemed noncontaminated, the structure of the filter room will be removed to the level of the radwaste laydown floor.

The floor of the radwaste laydown area immediately above the underground exhaust duct areas will be removed. The concrete foundation of the filter room will be decontaminated and removed as necessary.

The remaining vitreous clay hot cell exhaust pipe entering the manhole will be excavated and removed. Soil will be radiologically surveyed and disposed of as indicated by its contamination or lack thereof. If found, contaminated soil will be removed. The manhole will be removed intact and the surfaces scarified as necessary.

3.3.1.25 Dismantle Underground Hot Cell Exhaust Duct (Between Primary and Polishing Filter Rooms) - Task #25

The following activities are planned to occur during this decommissioning task:

- o Remove Floor Above Duct;
- o Remove Soil;
- o Remove Concrete Duct.

The concrete floor above the exhaust duct (from the removed filter room area to the polishing filter room lower plenum) will be cut into manageable sized slabs and removed. The underside will be radiologically surveyed and if necessary decontaminated. If the underside of the floor slab is found to be contaminated, additional floor areas will be removed and decontaminated as is necessary.

A contamination control containment enclosure will be erected above the buried exhaust duct. Soil will be removed using a small loader and/or hand excavation, thereby exposing the concrete duct from the removed filter room area to the polishing filter room lower plenum. Soil will be radiologically surveyed during removal. Soil removal will continue until contamination has been eliminated. The concrete duct will be segmented using standard concrete cutting methods and removed in sections and then packaged for disposal.

3.3.1.26 Decontaminate Building #2 Structural Surfaces (Hot Laboratory) - Task #26

The following activities are planned to occur during this decommissioning task:

- o Hot Cell Roof Plug Area Decontamination;
- o Uranium Plating Laboratories Decontamination;
- o Hot Cell Charging Area Decontamination;
- o Radwaste Laydown Area Decontamination;
- o Decontamination Booth Decontamination;
- o Polishing Filter Room/Plenum Decontamination;
- o Weld Shop (Target Lab) Decontamination;
- o PAR Manipulator Reel Room Decontamination.

Activities listed for this task may be scheduled during other tasks or activities as necessary to maintain continuity of the decommissioning process within the building.

The Uranium Plating laboratories will be decontaminated concurrently with decontamination activities within the hot cells and the area above the hot cells. At each laboratory entrance, anterooms will be installed as contamination control barriers and HEPA filtered ventilation applied to maintain inward air flow, within each work area. Exhaust ducts will be removed and packaged for shipment to the decontamination-recycling facility. The remaining fixtures will be removed and packaged for processing either to recycle or disposal. The wall, floor and ceiling surfaces will be vacuumed, wiped, and/or scarified as required.

The exhaust duct system in the radiochemistry lab will be removed prior to decontamination activities. The radiochemistry laboratory surface's will be decontaminated before the process for removing the floor begins (see Task #21). In the radiochemistry lab only small areas may need to be scarified. The decontaminated surfaces will be covered with plastic to prevent recontamination.

In the hot cell charging area, small containments will be erected on the floor along the charging door area of the hot cells for decontamination activities in this area. Containment barriers will also be erected on the North boundary of the charging area (to isolate it from the rad waste laydown area) and at the exits on the South side entrances to the gamma pit area and second floor stair well, and exhaust HEPA ventilation applied.

The charging doors of the hot cells and the nearby floors will be scarified, and the charging doors removed. Contaminated equipment will be removed and packaged for either recycle or disposal as required. Upper horizontal surfaces will be decontaminated by vacuuming and/or wiping of the surfaces. Upper and lower walls will be vacuumed, wiped, and scarified as necessary. The floor drains will be disconnected, grating and drain lines removed, and the concrete scarified. The charging area floor will be scarified as necessary. The overhead exhaust duct system will be removed and packaged for shipment to a off-site decontamination-recycle/disposal facility.

In the radwaste laydown area (North of the filter rooms), containment barriers will be erected to isolate this area. Contaminated equipment will be removed and packaged for disposal. The upper horizontal areas will be decontaminated by vacuuming and/or wiping. Upper and lower walls will be vacuumed, wiped, and scarified as necessary. The floor area will be scarified where contaminated.

Following the completion of decontamination of surfaces in the hot cells (a separate task) removal of waste line piping will commence. For each area of the building requiring removal of contaminated piping, the material will be removed during the remedial activities for that particular area.

The decontamination booth located West of the hot cell charge area will be vacuumed and wiped down to remove contaminants. Small areas of the stainless steel liner surface that may remain contaminated will be removed with power grinders to remove contaminants. Piping and drains will be removed.

After the underground concrete exhaust duct pipe is removed, equipment, components, filters etc. from the filter room #2 will be removed and sent for disposal or recycling. The surfaces remaining will be vacuumed, wiped, and scarified as necessary to remove remaining contaminants. The plenum below the floor will be scarified and vacuum cleaned as necessary.

The filters and equipment within the gas filter pit behind hot cell number five will be removed. Embedded piping (hot off-gas, drains, etc.) will be removed and the exposed concrete surfaces scarified as necessary.

Equipment in the target welding shop will be removed, and if contaminated, packaged for shipment to the decontamination-recycling facility. Upper surfaces will be vacuumed and wiped, and small contaminated areas will be scarified. Lower surfaces and the concrete floor will be scarified to remove contaminants as required.

3.3.1.27 Decontaminate/Dismantle Emergency Storage (T-1)/Waste Treatment Facility - Task #27

The following activities are planned to occur during this decommissioning task:

- Decontaminate Outer Surfaces;
- Remove Floor Plug of Waste Treatment Facility;
- Remove Piping/Equipment/Tanks;
- Remove Sump Pumps/Equipment;
- Decontaminate/Scabble Interior Surfaces;
- Scabble Exterior Walls of Emergency Storage/Waste Treatment Facility.

Inward air flow will be applied using portable HEPA ventilation units, and an anteroom constructed over the floor plug opening. (The interior of the emergency storage (T-1) and the evaporator room will be utilized as containments.)

The waste collection and emergency surge tank (WT-100) will be drained, desludged, segmented in place, and packaged for transport to a recycling facility. The remaining tanks, evaporators, condensers, etc. will be drained, segmented in place and packaged for transport to a decontamination recycling facility. The piping, sump pumps, etc. will be removed and packaged.

The interior surfaces of the evaporator room and the emergency storage facility will be vacuumed, wiped, and scarified as necessary. The exterior surfaces of the evaporator and emergency storage rooms will be exposed by excavating the surrounding fill (which will be radiologically monitored and removed as contaminated material as necessary in Task #21) and scarified as necessary to remove any areas of contamination. The load bearing footing/floor structure for these rooms will not be removed until after the building demolition phase of decommissioning. When the floor slabs are removed, the underside of these slabs will be scarified if small areas of contamination are found to exist.

3.3.1.28 Decontaminate/Dismantle Hot Laboratory Building Exhaust Ventilation System (EVS) - Task #28

The following activities are planned to occur during this decommissioning task:

- o Remove EVS in Operating Area;
- o Remove EVS in Charging Area;
- o Remove EVS in Maintenance/Canal Area;
- o Remove EVS in Radwaste Handling Area;
- o Remove EVS in Above Storage Well Area;
- o Remove Exhaust Fans (Fan Room).

The equipment, heaters, fans, ducts etc. for the operating area, charging area, maintenance/canal area, radwaste handling area, and the storage well addition will be vacuumed, wiped, and removed. If found to be contaminated, the items will be packaged for shipment to the off-site decontamination-recycle/disposal facility. The exhaust fans located within the hot lab fan room will be segmented, removed and packaged for shipment to the off-site decontamination-recycle/disposal facility. After the systems from inside the building have been removed the air handling units, duct work, etc. on the roof of the hot lab building will be segmented, removed, and packaged for shipment. Glove-bag type containments or small containment enclosures will be used to isolate contaminated EVS ducts and equipment will be removed in the direction of air flow so that the EVS can serve to maintain inward air flow within the system.

3.3.1.29 Remove 5000 Gallon Tanks - Task #29

The removal of the contaminated two 5000 gallon holdup tanks for buildings #1 and 2 process water will be performed as follows:

- o drain tanks;
- o excavate surrounding soil to expose tanks;

- o segment tanks;
- o check if soil is releasable.

The tanks will be drained by pumping the liquids if necessary to a temporary water treatment system. The interiors of the tanks will be hydrolazed and the bottom sludge slurried out for treatment if necessary.

The soil will be excavated to expose the tanks. The soil will be radiologically surveyed. If the soil is found contaminated, it will be packaged and sent for proper disposal. The access manhole will be surveyed, then removed. If the manhole is found contaminated, it will be decontaminated by scarification or broken into rubble, placed in LSA boxes, and shipped as contaminated concrete.

The tanks will be placed in a contamination control containment tent with portable HEPA ventilation units. The tanks will be segmented by thermal cutting and placed in LSA boxes for shipment to the off-site decontamination-recycle center/disposal facility. The soil around and below the tanks will be radiologically surveyed. If the soil is found to be contaminated, it will be placed in shipping containers and sent for disposal.

3.3.1.30 Remove Outside Ducts and Exhaust Stack - Task #30

The outside exhaust stack and duct removal work will encompass the removal of the following items:

- o 28 inch exhaust from reactor building to exhaust fan room;
- o 4 inch emergency exhaust from reactor building to exhaust fan room;
- o 48 inch duct from exhaust fan room to and including the stack.

The two ducts from the reactor building are outside of the building. In order to contain the contamination, portable HEPA filtered ventilation units will be applied to the down stream end of the ducts to minimize spread of contamination during cutting. Portable containment tents or glove-bags around the duct will be used to isolate the contaminated ducts. Duct sections will be rigged to a crane, cut free, both ends sealed, and transfered to a segmentation/volume reduction area. Within the segmentation/volume reduction area, while in a containment tent, the ducts will be segmented lengthwise and placed into LSA boxes for shipment to off-site decontamination-recycle/disposal facility.

The duct from the fan room to the stack will be removed by first removing the stack then transferring capped duct segments to an area for further segmentation. Removal equipment will be brought over the road to the stack. A portable HEPA filtered ventilation unit will be connected to the stack, the stack rigged for removal. The stack will be cut, layed down and both ends capped. Within a contamination tent the stack will be segmented lengthwise and packaged in LSA boxes.

The concrete duct supports/stack foundation will be surveyed and if contaminated they will be scarified and abandoned in place. The soil under and around the duct will be surveyed and if contaminated, removed and packaged for proper disposal.

The duct penetrations in the building walls will be capped, removed, and placed in an LSA box for disposal.

3.3.1.31 Remove Manholes - Task #31

The following manholes are suspected of being contaminated and may need to be decontaminated and/or removed:

- Manhole 16 and catch basin;
- Manhole 15 (process portion).

The soil surrounding the manholes will be excavated and monitored for radioactivity. If contaminated, manholes' surfaces will be scarified or removed and packaged for proper disposal.

The interior of the sumps will be drained by pumping any liquids to a temporary water treatment system. The pipes entering the manholes will be cored and removed. The manhole surfaces will be surveyed to determine if they are releasable. If found contaminated, they will be scarified to remove contamination after contamination control tents are installed. The walls will be removed as clean concrete. The floor slabs will be removed and bottom of the slab radiologically surveyed. If contaminated, they will be scarified until releasable or packaged whole. As a final step the soil under the former manholes will be monitored for releasability.

If it is determined that the process line from manhole 10 to manhole 8 is contaminated, manholes 10, 11 and the monitoring manhole will be removed in the manner stated above.

3.3.1.32 Remove Yard Piping - Task #32

The following yard pipes are currently suspected of being contaminated and may be removed:

- Hot laboratory storm drains to manhole #16, from this manhole to a catch basin and from the basin to its discharge point;
- Portions of the reactor building footing drains that connect to the hot lab storm drain between manhole 16 and a catch basin;

- o Reactor pump room discharge pipe to the two 5000 gallon holdup tanks;
- o Process pipe from hot lab to the 5000 gallon holdup tanks;
- o Reactor pump room discharge to hot laboratory;
- o Building #4 holdup tank pipe to reactor building.

The soil above the pipes will be excavated to expose the pipe. The soil will be radiologically monitored during excavation. If the soil is releasable it will be used to backfill the trench or, if it is not releasable, it will be packaged and disposed of properly. The floor of the main tunnel will be cut to expose the pipe from the reactor pump room discharge to the 5,000 gallon holdup tanks. The manholes and catch basin concrete will be radiologically surveyed. If they are found to be contaminated, they will be dealt with as described in Task #31.

The pipe line from the building #4 holdup tank to the reactor building will be removed. The pipe and valves between the 5,000 gallon holdup tanks and manhole 10 will be removed and capped. The process line from manhole 10 to monitoring manhole at building #3 and to manhole 8 will be checked for contamination. If found to be contaminated, it will be excavated, removed and the soil checked for contamination.

3.3.1.33 Survey Structures - Task #33

The remaining structures will be subjected to a Phase I final termination survey as described in Section 8.0 of this plan. At this point some contaminated structures may still be present, that cannot be decontaminated until the buildings have been razed. Therefore, a second phase survey will be performed after the buildings have been razed and any remaining contamination removed to release limits.

3.3.1.34 Raze Non-contaminated Building Structures - Task #34

The following activities are planned to occur during this decommissioning task:

- o Demolish Building #2 Hot Lab Structure;
- o Demolish Reactor Structure;
- o Demolish Reactor Building Structure;
- o Demolish Tunnel Structure.

Prior to the demolition and removal of the structures, all will have been verified to meet release criteria. Below grade contaminated building foundations and soil/ bedrock may still be present. Any water collected during demolition will be monitored to determine that it meets regulatory release criteria.

The removal of building structures will be accomplished as conventional industrial demolition. This demolition will not include any inaccessable contaminated foundations or areas that may still be present. These will be removed under controlled conditions and radiological monitoring (see Task #35).

The demolition process will be radiologically monitored for airborne contaminants during this task as a conservative measure. Soil removed during demolition will be radiologically characterized and stockpiled for site restoration backfill following the release of the site.

The non-concrete debris from demolition will be removed from the site and transported to local landfill areas. Concrete rubble may be retained on site for backfill purposes. Non-contaminated structural steel will be disposed of as scrap.

3.3.1.35 Decontaminate/Remove Remaining Contaminated Structure and Soil - Task #35

The following items may have to be removed to gain access to potentially contaminated areas.

- o Maintenance/Electrical Shops Floors/Footings;
- o Foundation/Footings Building #2 East Wall;
- o Floor T-1/Evaporator Room;
- o Building #1 East Wall and Buttress Footings;
- o West Wall Building #1 (North of pool);
- o North Wall Building #1;
- o Ribs and Footings Building #1 North Wall.

The removal of these items, while not contaminated, will be accomplished under controlled conditions with airborne and radiological monitoring. As sections of these areas are removed, each will be radiologically surveyed. If and when contamination is found to exist on either the structural surfaces or the surrounding soil/bedrock, containments will be erected if deemed necessary by contamination levels, prior to proceeding with further removal of concrete or soil/bedrock.

The following represents the footings/areas which may be expected to reflect some level of contamination due to periodic breaches of systems, components, or areas:

- o Building #2-Areas beneath applicable columns;

- o Building #2-Area beneath T-1/Evaporator room floor;
- o Building #1 South-Footings near Hold-up Tank;
- o Building #1 East-Buttresses adjacent to drain tile areas;
- o Building #1-West Wall/Monolithic Concrete;
- o Building #1 North-Ribs as applicable and drain tile areas.

If contamination has migrated from the canal, as determined from the canal removal, the floor of the maintenance/electrical shops will be segmented and removed. The underside will be surveyed, and any areas of contaminants will be removed by scabbling. Any pillars or footings beneath this section of building #2 will be removed and radiologically surveyed and decontaminated as necessary.

If contamination has migrated from the buried exhaust duct, as determined by removal of that duct, the foundation and footings of the East wall of building #2 will be excavated and surveyed. If areas of contamination are found, they will be removed by scabbling, and the foundation will be removed using conventional demolition methods. The footings will be lifted out and will be radiologically surveyed. If contamination exists and it is removable by the scabbling process, the footings will be scarified and used for site backfill. Should the footings be found to be deeply contaminated, they will be rubbled as needed and packaged for disposal.

The floor surface of the evaporator/T-1 room will be rubbled and removed. The underside of each slab will be surveyed, and if contamination is found will be removed by scabbling. Should the irregularity of the surface prohibit scabbling, the concrete will be further rubbled and packaged for disposal.

Soil exterior to the east wall of building #1 will be removed as needed to expose the drain tile. The drain tile will be removed and packaged for disposal if found to be contaminated. The wall and buttress foundation will be surveyed, and should contamination exist, it will be removed by scabbling. The cast iron drain piping through the buttresses will be surveyed and if found contaminated, will be removed by core over-drilling methods, and packaged for disposal. If soil contamination is found in the area of the drain tile, the soil/fill will be removed as needed.

The West wall of the reactor building North of the pool and below elevation 808' will be removed to expose the concrete fill located behind the wall. The concrete monolith will be removed using a method such as an expanding grout until the bedrock wall is exposed. As each monolithic piece of concrete is removed, it will be radiologically surveyed, and if small areas of contamination exist the surfaces will be scarified.

The North wall of building #1 will be removed as needed to expose the canal drain tile. The wall and ribs will be removed and the drain tile, soil/bedrock side surveyed. Sections of the wall poured directly against bedrock, if contaminated, will be further rubbled or scarified and packaged in LSA containers for disposal. The footings for each rib will be excavated and surveyed. If contamination is not found the footings will be used for backfill. The areas below the footings which contained the drain tile will be excavated. The drain tile will be packaged in LSA containers for disposal if necessary.

3.3.1.36 Decontaminate Class A Storage Building - Task #36

When the Class A storage building is no longer needed, the structure will be surveyed and if necessary decontaminated by vacuuming, wiping, or scabbling.

3.3.1.37 Final Termination Survey - Task #37

The Phase II final termination survey of the remaining Cintichem site will be conducted according to details in Section 8.0 of this decommissioning plan. After termination of the licenses by NRC and NYS, the site will be backfilled.

3.3.2 Generic Description of D&D Methods

D&D work procedures will be developed for the following generic activities:

- o Contaminated pipe and metal cutting;
- o Contaminated concrete decontamination;
- o Equipment and structural decontamination and dismantling;
- o Activated component cutting;
- o Activated concrete removal;
- o Contaminated soil bedrock removal;
- o Radioactive waste packaging and disposal.

The typical methods which are envisioned for use are described below. Specific procedures for application of these D&D methods will be developed during detailed engineering.

3.3.2.1 Pipe and Metal Cutting

This section describes acceptable methods for sectioning all metal components, parts, and equipment. This includes the large bore primary piping attached to the reactor pool and the small bore piping of the various process systems, as well as other equipment such as tanks, filter assemblies, heat exchangers, etc.

As a prerequisite for any cutting activity, all materials being disassembled will be considered to be contaminated until determined otherwise by survey. This will mean the issuance of radiation work permits and implementation of area and airborne radiation surveillance and contamination control procedures as a part of any cutting activity. Draining of each system will be implemented to assure that systems contain no liquid before dismantling begins, as required. Low points of piping systems will be drained by drilling and use of saddle taps to allow for controlled draining of fluids.

In general, cutting of ferrous materials will be accomplished using a plasma arc torch. This technique produces very fast cuts and requires little or no equipment set up time, thereby reducing personnel exposures. Non-ferrous materials, such as aluminum, will be cut using portable power saws or other mechanical methods. Use of grinding, or abrasive methods will be discouraged.

Whenever cutting techniques which could generate loose or airborne contamination are used, suitable contamination control techniques will be employed. These techniques will be deployed on a case by case basis as is warranted by contamination/activation levels and the dispersal potential of the cutting techniques used. Control techniques may include, but not be limited to, use of HEPA ventilated containment tents, glove boxes, or localized use of HEPA vacuum cleaners/air filtration units to catch generated particulates, fumes or vapors.

All equipment or pipe openings will be capped or temporarily sealed until the contamination levels on interior surfaces have been determined. The procedure also will call for good housekeeping in the work area, especially for collection of any debris produced in the cutting process.

3.3.2.2 Contaminated Concrete Surface Removal

Concrete surfaces will be decontaminated by scarification using equipment such as a Scabbler™. A SCABBLER is a commercially available tool that is used to remove up to a 1/4 inch layer of concrete surface per pass. Scabbler(s) are available with one to eleven pneumatically operated pistons with carbide tipped bits to impact concrete surfaces, thereby pulverizing the concrete surface. Handheld single piston models are used to decontaminate near walls, corners, or encumbrances. Multi-piston models (in configurations similar to that of a lawn mower) are available for open areas. The commercially available scabblers will be modified to include HEPA ventilated shrouds to preclude airborne release of contaminated dust during the scarification process. After the floor surface have been scarified the resulting pulverized concrete will be removed from the floor using high powered HEPA vacuums which can directly place the concrete dust into disposal containers.

Small deeper "hot spots" or contaminated cracks or seams will be cut out using standard concrete pavement breakers. Hydraulic units are preferred to prevent airborne dispersion problems associated with the air exhaust from pneumatic devices. Localized HEPA ventilation units or vacuums would be used to preclude release of contaminated dust.

3.3.2.3 Equipment Decontamination

In general contaminated equipment will be disposed of as radioactive waste or packaged for shipment to a decontamination-recycling facility. However, when feasible, contaminated equipment may be decontaminated to reduce radioactive waste volume. The choice of complete disposal versus in-situ decontamination will be made on a case by case basis given financial, safety and ALARA factors.

Dry type decontamination processes that may be used are vacuuming, wire brushing, dry wiping, and similar techniques either singularly or in combination to achieve the desired decontamination. Wiping employing solvents with a sorbent material dampened with detergent or chemical cleaners may also be used. Using damp wiping processes, only cleaning agents which are non-hazardous and are compatible with radioactive waste processing and disposal site requirements will be used.

Contaminated walls, steel decking and underside of floors may be decontaminated using a vacuum shot blasting technique. This technique uses steel shot or glass beads in a manner similar to sand blasting to remove surface paint or other coverings. The Vac-U-Blast™ system, however, has a vacuum shroud which collects the used steel shot and the surface debris mixture. The steel shot is magnetically separated and recycled while the contaminated material removed from the decontaminated surfaces is directly deposited into disposal containers. The Vac-U-Blast system is preferable to sandblasting in that (1) clean up is accomplished simultaneous to the decontamination process, and (2) only a very small volume of contaminated shot is generated compared to tons of spent sandblast sand.

3.3.2.4 Concrete Removal

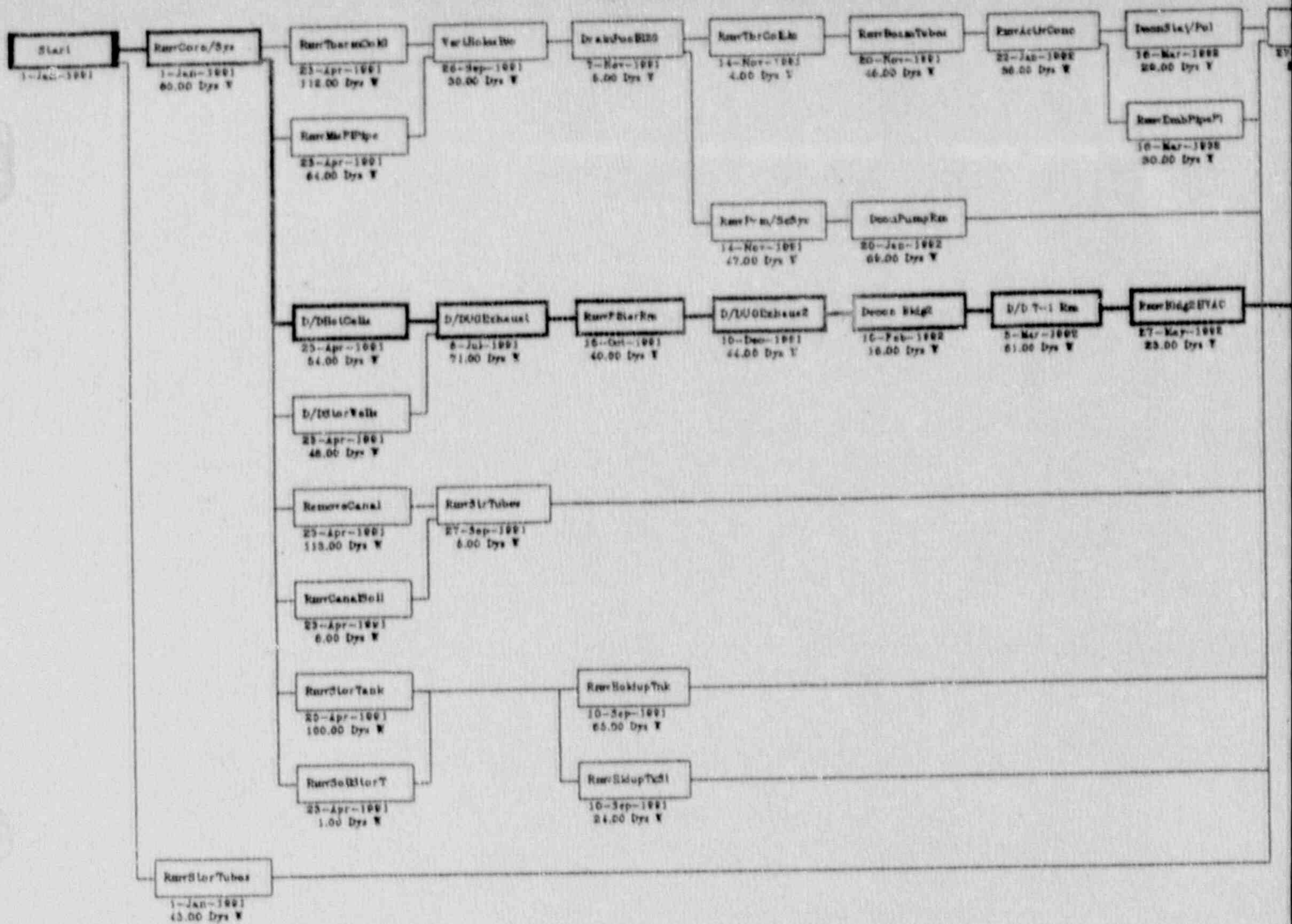
Areas of the reactor pool and stall containing concentrations of radioactive material produced by neutron activation in excess of releasable limits will be removed by scarifying, or other standard demolition techniques such as rock splitting, ramhoe breaking, cutting, or expanding grout techniques.

3.4 Schedule of Major Activities

The sequence of decommissioning tasks is shown in Figure 3.1 (PERT).

The schedule for accomplishing the phase two decommissioning workscope is shown on Figure 3.2 (Gantt). As indicated the total duration of the decommissioning tasks is 26.5 months.

PERTChart


Start
1-Jan-1991
RmvCore/Sys
1-Jan-1991
60.00 Dys
23-Apr-1991
118.00 Dys
23-Apr-1991
64.00 Dys
26-Sep-1991
30.00 Dys
7-Nov-1991
5.00 Dys
14-Nov-1991
4.00 Dys
20-Nov-1991
46.00 Dys
22-Jan-1992
36.00 Dys
16-Mar-1992
20.00 Dys
16-Mar-1992
30.00 Dys
14-Nov-1991
47.00 Dys
DeconPumpRm
20-Jan-1992
69.00 Dys
D/DHotCells
23-Apr-1991
54.00 Dys
D/DUGExhaust
8-Jul-1991
71.00 Dys
16-Oct-1991
40.00 Dys
D/DUGExhaust2
10-Dec-1991
44.00 Dys
Decon Bldg2
16-Feb-1992
16.00 Dys
5-Mar-1992
61.00 Dys
27-May-1992
25.00 Dys
23-Apr-1991
48.00 Dys
RemoveCanal
23-Apr-1991
115.00 Dys
RmvStrTubes
27-Sep-1991
5.00 Dys
23-Apr-1991
6.00 Dys
RmvStorTank
20-Apr-1991
100.00 Dys
RmvHoldupTnk
10-Sep-1991
65.00 Dys
23-Apr-1991
1.00 Dys
10-Sep-1991
24.00 Dys
RmvStorTubes
1-Jan-1991
43.00 Dys

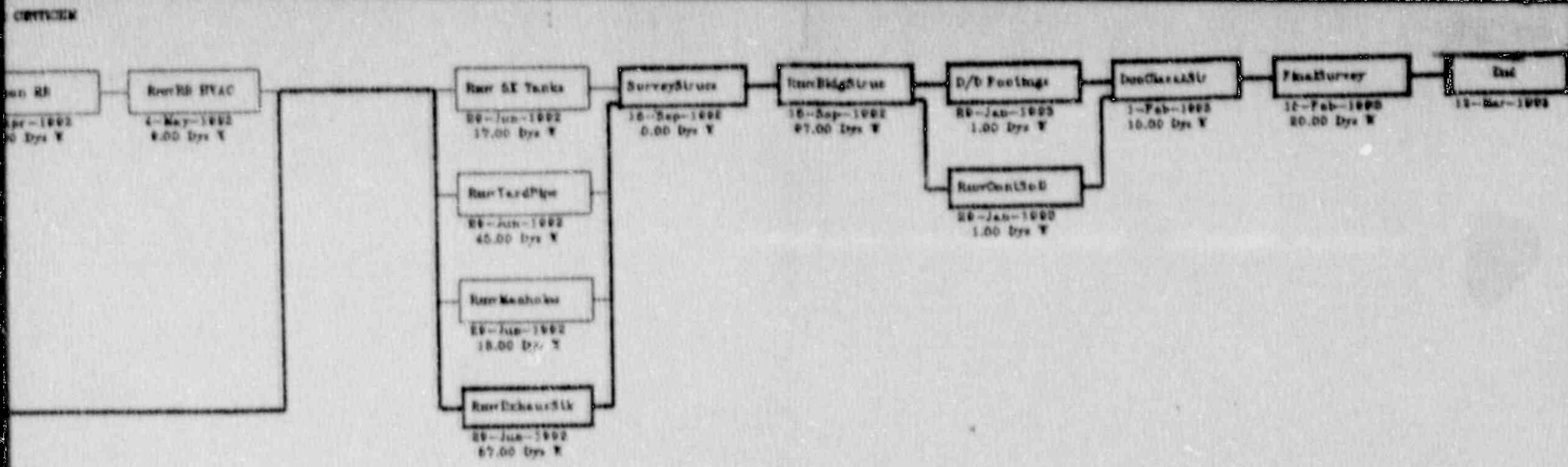


Figure 3.1

Gantt Chart

RmvCore/Sys
1-Jan-1991
80.00 Dys Y

Start
1-Jan-1991

RmvStorTubes
1-Jan-1991
45.00 Dys Y

RmvThermoCold
23-Apr-1991
112.00 Dys Y

RmvCanalSoil
23-Apr-1991
6.00 Dys Y

RmvSoilStorT
23-Apr-1991
1.00 Dys Y

RmvMiscPipe
23-Apr-1991
64.00 Dys Y

D/DStorTank
23-Apr-1991
46.00 Dys Y

D/DHotCells
23-Apr-1991
64.00 Dys Y

RmvStorTank
23-Apr-1991
100.00 Dys Y

RemoveCanal
23-Apr-1991
113.00 Dys Y

D/DOGExhaust
6-Jul-1991
71.00 Dys Y

RmvHoldupTnk
10-Sep-1991
65.00 Dys Y

RmvHldupTkSl
10-Sep-1991
24.00 Dys Y

VerifHolesRm
26-Sep-1991
30.00 Dys Y

RmvStrTubes
27-Sep-1991
5.00 Dys Y

RmvFilterRm
15-Oct-1991
40.00 Dys Y

DrainPoolH2O
7-Nov-1991
6.00 Dys Y

RmvThrCoRm
14-Nov-1991
4.00 Dys Y

RmvPrim/SdSys
14-Nov-1991
47.00 Dys Y

RmvBeamTubes
20-Nov-1991
40.00 Dys Y

D/DOGExhaus2
10-Dec-1991
44.00 Dys Y

DconPumpRm
20-Jan-1992
69.00 Dys Y

RmvActivConc
22-Jan-1992
36.00 Dys Y

Decon Bldg2
10-Feb-1992
16.00 Dys Y

D/D T-1 Rm
3-Mar-1992
61.00 Dys Y

DeconStal/Pol
16-Mar-1992
29.00 Dys Y

RmvEmbPipePl
16-Mar-1992
30.00 Dys Y

Dcon RB
27-Apr-1992
5.00 Dys Y

RmvRB HVAC
4-May-1992
9.00 Dys Y

RmvBldg2HVAC
27-May-1992
23.00 Dys Y

RmvExhaustStk
29-Jun-1992
57.00 Dys Y

RmvYardPipe
29-Jun-1992
45.00 Dys Y

Rmv UG Tanks
27-Jun-1992
17.00 Dys Y

RmvManholes
29-Jun-1992
15.00 Dys Y

SurveyStrucs
16-Sep-1992
0.00 Dys Y

RmvBldgStruc
16-Sep-1992
97.00 Dys Y

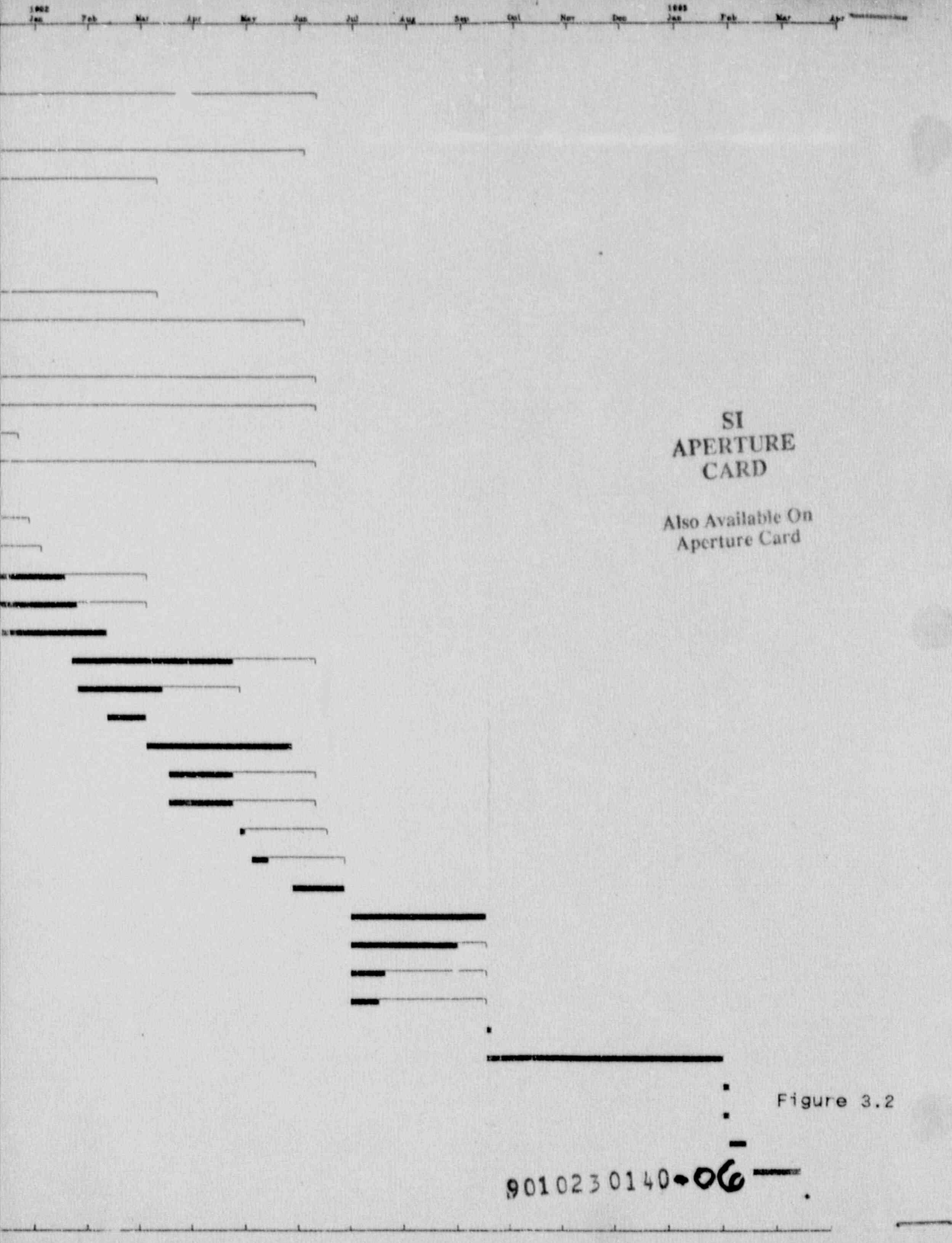
D/D Footings
20-Jan-1993
1.00 Dys Y

RmvContSoil
24-Jan-1993
1.00 Dys Y

DcnClassAStr
1-Feb-1993
10.00 Dys Y

FinalSurvey
15-Feb-1993
20.00 Dys Y

End
12-Mar-1993



SI
APERTURE
CARD

Also Available On
Aperture Card

Figure 3.2

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4.0 Physical Security Plan Description

The removal of all special nuclear materials from the facility will occur under the current operating license before the initiation of decommissioning and dismantling. After removal of the SNM, changes to the physical security of the site will be conducted to the following plan description.

A security program will be enforced to provide control of personnel and materials during decommissioning. Perimeter barriers will be established requiring personnel access authorization and control of vehicles. The activities of the security force will be governed by Site Management and provided by existing security personnel.

4.1 Security Organization

The site security organization consists of the site operations/security services manager and security supervisor. The security services manager is responsible for implementing the security plan and providing overall supervision of the security organization. The security supervisor provides the day-to-day administration and coordination of the security force. The security services manager reports to the project manager.

4.2 Security Personnel

Appropriately trained personnel shall implement the provisions of this security plan. During abnormal security conditions, as detected by security intelligence, the security force shall be appropriately augmented and local law enforcement agencies alerted if necessary, as per the security plan.

The duties of the site security personnel shall normally include:

- Control of the perimeter portals;
- Response to intrusion conditions;
- Search functions;
- Verify area access authorizations;
- Perform security patrols;
- Perform inspections and maintain records.

4.2.1 Training

Security personnel shall not be assigned security duties unless such individuals have been properly trained and demonstrate an understanding of the plant security procedures.

Initial training will include the following:

- o Responsibilities of security personnel;
- o Access authorization and control;
- o Facility security policies and procedures;
- o Perimeter barrier requirements;
- o Patrolling of perimeters;
- o Fire control;
- o Recording and reporting observations;
- o Methods of search and seizure.

4.2.2 Retraining

Security personnel shall be requalified every 12 months. Requalification shall be documented.

4.3 Access Authorization

Instructions and/or procedures, in addition to existing security procedures, shall be established for authorization of access to the facility and for the authorization and identification of equipment and materials which enter or leave secured areas.

4.3.1 Departure Clearance

Health Physics approval will be verified by security prior to equipment and materials being allowed to depart from the secured area.

4.4 Site Perimeter

4.4.1 Perimeter Barrier

The perimeter barrier will utilize the existing 84" chain link fence as shown in Figure 4-1. This fence will provide perimeter control of the Eastern boundary adjacent to Long Meadow Road, the Southern boundary and approximately half of the Western boundary. The inaccessible boundaries provided by the steep vertical drop of exposed rock and the Indian Kill reservoir provide the remaining perimeter barrier."No Trespassing" signs shall be conspicuously placed in the wooded areas of the Western boundary.

4.4.2 Portals

During working hours the Southern and Eastern gates will be open allowing access through the outer perimeter to site operations and activities not related to decommissioning. During non-working hours the Southern gate will be secured by key operated lock. The Eastern gate will be electrically operated utilizing programmable access cards and the existing security system.

4.5 Secured Area

A typical secured area layout is shown in Figure 4-2. The boundaries shown are approximate and will be moved as the need arises.

Exterior areas shall be illuminated to the extent practical.

4.5.1 Escorts

All personnel not authorized unescorted access to the secured area shall be escorted by an individual authorized access to the secured area.

Personnel permitted escort access to the secured area shall be restricted to those authorized by Level 1 and 2 management or designated alternate, per Section 1.5 - Organizational Chart.

4.5.2 Buildings One and Two Secured Area

The physical structure and existing card key system of buildings one and two will serve as a controlled area during the initial phases of equipment, systems and structure decontamination. This building barrier will not be considered a secured area boundary.

The building barrier will be breached during structure dismantling. The secured area perimeter and the surveillance procedures will be adjusted accordingly to provide proper control of personnel and materials.

4.5.3 Secured Area Perimeter Patrol

Patrols shall be performed by members of the security force to verify the integrity of the secured area physical barrier. Patrolling personnel shall have communications capabilities with security control. These patrols shall be performed per plant security procedures.

4.6 Response To Security Contingencies

The security procedures manual will be updated to include written procedures which shall be utilized to the extent applicable for abnormal security conditions including the following:

- o Disruption of Internal order;
  - o fire;
  - o site evacuation;
  - o personnel disturbance.
- o Stated or Perceived Threat of Sabotage;
- o Civil Disturbance;
- o Suspected or Confirmed Intrusion;
  - o Intrusion;
  - o Discovery of breached secured area barrier;
  - o Discovery of unidentified person in secured area.

4.7 Inspection and Maintenance

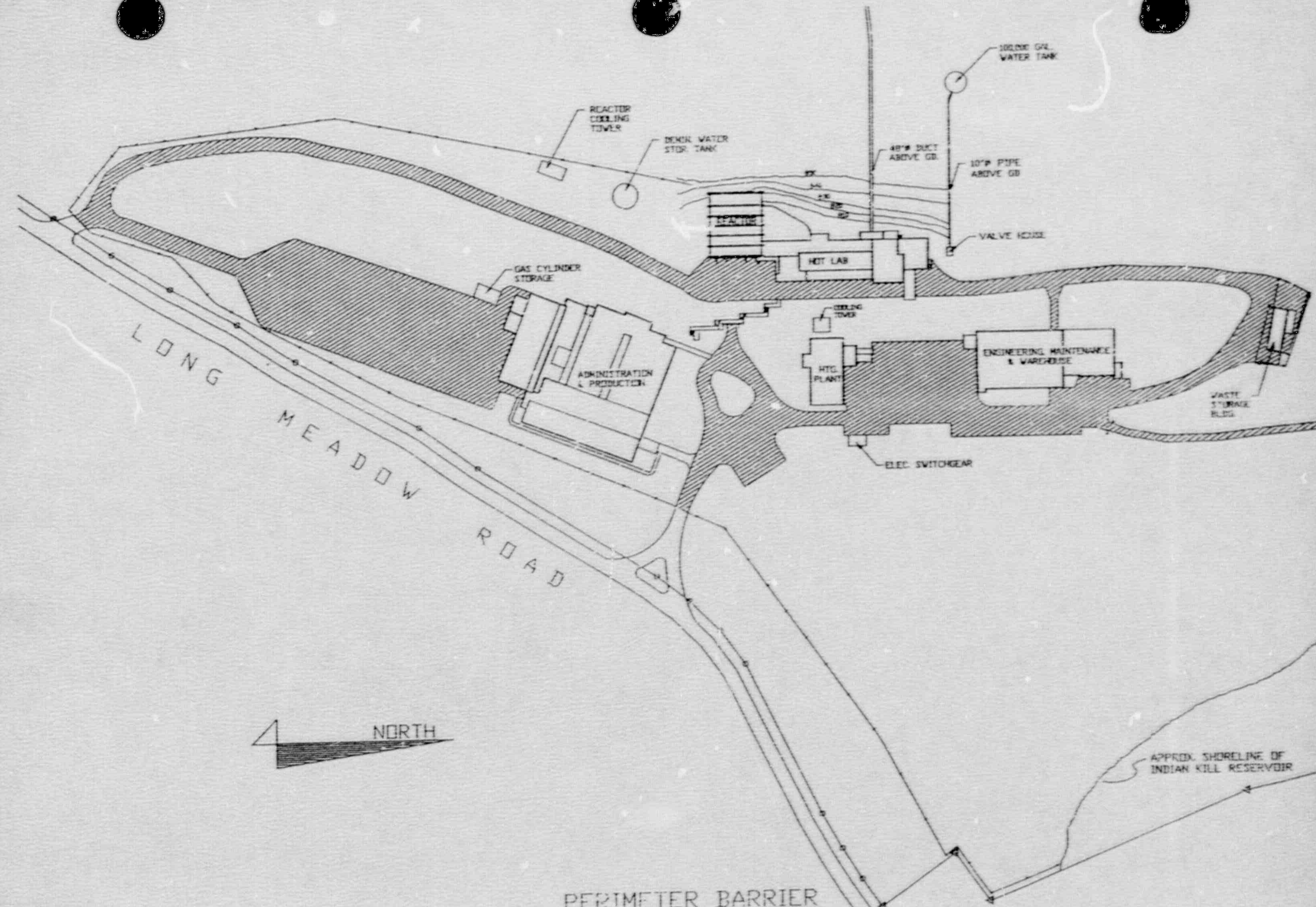
Required physical barriers and access points shall be maintained in operable condition. Inspections shall be performed in accordance with appropriate procedures. Identified significant degradation shall be corrected, or appropriate compensatory action taken in a timely manner.

On-site security communication equipment shall be tested for performance at the beginning of each shift.

4.8 Security Records

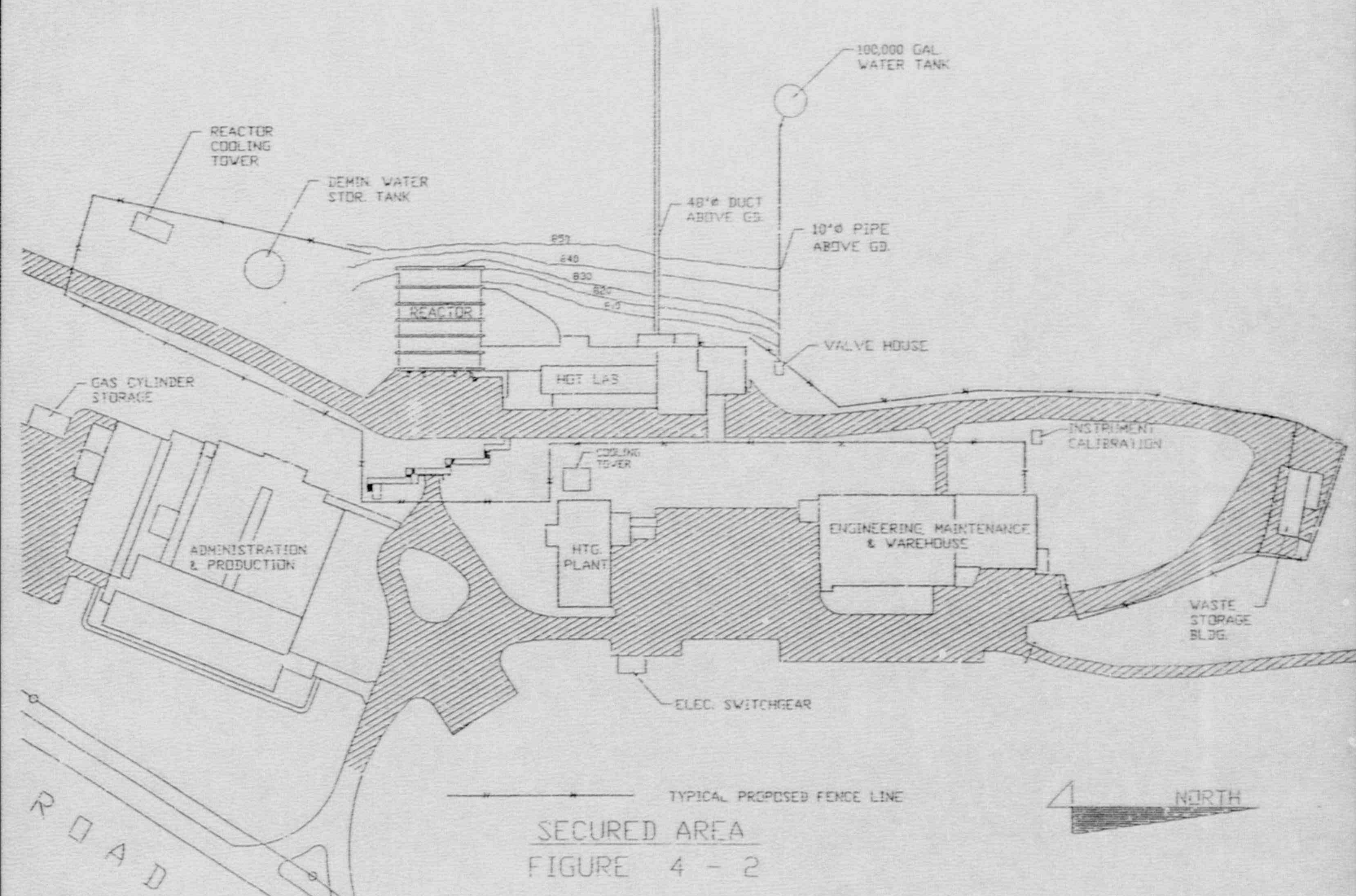
Security records shall be maintained to include the following and any additional information as required:

- o barrier inspection reports;
- o training records;
- o abnormal occurrence reports.



PERIMETER BARRIER

FIGURE 4 - 1



SECURED AREA

FIGURE 4 - 2

5.0 Radiological Accident Analysis

The Cintichem Decommissioning Plan has been written based on the guidance of the NRC's "Guidance and Discussion of Requirements for an Application to Terminate a Non-Power Reactor Facility Operating License". This guide under Section 5.0 "Radiological Accident Analysis" states that "if no fuel is present on site, radiological accidents need not be considered except for those that would fall within the general categories of (Decommissioning Tasks)". Because Cintichem plans to have all reactor fuel and other Special Nuclear Material off site when the decommissioning starts, this section will only contain non-fuel related hazards.

A number of accidents were examined that could occur during the decommissioning and dismantling of the facility. The potential hazards during the dismantling period are orders of magnitude less than during the operating period. The frequency and overall consequences of accidents resulting in radiation exposure to the public would be much less for dismantling activities than for accidents during the normal facility operations. Inventories of radioactive materials are orders of magnitude less than during operations. All new and spent fuel, and other special nuclear material other than residual contamination, will have been removed prior to the start of the dismantling and decommissioning phase.

The hazards which may be associated with the Cintichem decommissioning are broken into two general categories, natural hazards and accidents directly associated with the decommissioning and dismantlement.

5.1 Natural Hazards

<u>Windstorm</u>

Since the reactor and hot laboratory buildings are built into the side of a hill, and are ruggedly built, windstorm damage to it can be precluded. The site is generally not subject to severe weather or windstorms and given the steel and masonary construction of the reactor and hot laboratory facility, significant damage from a windstorm is thought to be remote.

The buildings will not be removed until after the majority of radioactive material is removed and only a minor amount of contamination could be subject to wind effects. The radiological consequences are estimated to be much less severe than the drum drop accident analyzed in Section 5.2.

<u>Forest Fire</u>

The location of the buildings and its concrete and steel construction should protect the radioactive material inventory from damage due to a forest fire in the area. Also, the Sterling Forest Fire Department is located only 1/4 mile from the facility. It is not likely that a radiation hazard would result from a forest fire.

Earthquake

This area has a long record of freedom from violent earthquakes. No strong earthquakes have occurred since 1878. The combination of earthquake rarity, the low seismic intensities of any local earthquake, the rock foundations for the reactor and hot laboratory buildings and the structural design of the facility makes a significant hazard from seismic events remote. The potential hazards from earthquakes will be orders of magnitude less during the decommissioning due to a greatly reduced radioactive material inventory and lack of a critical array.

5.2 Decommissioning Hazards

The potential hazards of decommissioning due to accidents are principally associated with loss of control of radioactive materials. The consequences of these accidents has been analyzed and found to be minimal. The following summarizes the results of the hazard analysis for worst case decommissioning accidents.

The following accidents are the worst conceivable decontamination accidents:

- o Accidental cutting of activated reactor component in the pool without controls;
- o Dropping of a 7.5 ft^3 drum containing contaminated concrete dust outside of the buildings;
- o Flooding after the building is breeched.

5.2.1 Summary of Results

Accidental Cutting of Activated Reactor Component in the Pool Without Controls by Plasma Torch.

The most activated part of the most activated component (core support tower) is assumed to be cut without a containment tent. This component would normally be cut by hydraulic shear which would not generate radioactive vapors. Under worst case conditions, the estimated equivalent total body dose to a child at the nearest residential development is 0.0014 Rem (see Appendix E). This compares to an annual limit of 0.50 Rem total body dose permitted for unrestricted areas. The worst case estimated equivalent total body dose to a worker in the building is 0.36 Rem. This compares to the annual limit of 5.0 Rem total body dose for occupational workers. See Appendix E for calculations.

Dropping 7.5 ft^3 drum of contaminated concrete outside

This drum is assumed to contain the most contaminated concrete at the facility (hot cell walls and floors). The total body effective dose equivalent to a teenager at the site boundary is estimated to be 0.0032 Rem. This compares to an annual limit of 0.50 Rem total body dose permitted for unrestricted areas. The total body effective dose equivalent to a worker standing nearby is estimated to be 0.010 Rem. This compares to the annual limit of 5.0 Rem total body dose for occupational workers. See Appendix F for calculations.

Flooding After Buildings 1 and 2 are Breeched

All potentially contaminated soil around buildings 1 and 2 is assumed to be washed into Indian Kill Reservoir as a consequence of a severe rainstorm. Conservatively, this would put the drinking water at 11% of the unrestricted area MPC (Maximum Permissible Concentration). See Appendix G for calculations. In the event that a real threat to the reservoir was eminent, Cintichem would contact the Sterling Forest Water Company and advise them to shut down the use of the reservoir.

In summary, the scenarios listed above pose no major hazards to the public or assigned Cintichem personnel under the proposed decommissioning activities.

6.0 Radioactive Materials and Waste Management

During the decommissioning activities radioactive materials (radwaste) in liquid, solid and gaseous forms will be generated. Management of these wastes is an integral part of the decommissioning plan which includes provisions for minimizing the amount of waste generated, waste collection, treatment, packaging and shipment off-site for disposal.

6.1 SNM Disposal

As indicated in earlier sections of this plan, Cintichem intends to remove all on site SNM including the fuel from its 5 MW MTR research reactor. It will be done under the current operating license. Therefore, at the beginning of decommissioning operations the only SNM on site will be residual contamination.

Uranium survey data will be used to predict potential accumulations of SNM. If there is an indication that significant quantities of SNM could be concentrated or accumulated in decommissioning operations, then such operations will be managed to prevent accumulation greater than one-half the single parameter mass limit in any one location.

6.2 Liquid Radwaste

Liquid radwastes generated during decommissioning activities will be collected, monitored and solidified prior to shipment to an approved disposal site.

Liquid radwastes will be collected in the 7200 gallon T-1 tank in the existing underfloor space of Building 2 or may be collected in 55 gallon drums for solidification. From the collection tank the liquid will be pumped to a radwaste evaporator. The evaporator condensate will be checked for radioactivity and recycled if radioactivity is found that exceeds free release requirements. Evaporator bottom sludge will be concentrated, solidified and sent to a radwaste burial site. See Figure 1.11 for the radioactive waste system diagram. This diagram is a general diagram and does not include all valves and piping.

As the decommissioning process proceeds, the 7200 gallon T-1 collection tank may be removed and a mobile wastewater treatment system used in its place.

Sources of liquid radwaste are:

- Decontamination of components, parts, etc.;
- Decontamination of structures, floors, etc.;
- Hot sink wash;
- Reactor primary water;

o Run-off through contaminated soil.

Efforts will be made throughout the decommissioning activities to minimize the generation of liquid waste. Whenever possible scrubbing with swabs will be used instead of spraying. Liquid waste will be absorbed or solidified using absorbant materials or solidification agents.

Liquid radwaste is controlled by SP-18 (Radioactive Waste Evaporator Operating Procedures) and SP-02 (Radioactive Waste Disposal Procedures).

6.3 Solid Radwaste

The solid radwaste generated during decommissioning activities will be packaged on site in containers suitable for shipping and disposal. Based upon past operating experience, the Cintichem Characterization Study, and TLG Engineering Study, wastes generated during decommissioning activities will fall under the 10CFR61 "A", "B", and "C" classifications. Classification of these wastes will be verified by determining radionuclide concentrations in each waste stream according to the requirements of 10CFR61.55.

Eight major types of waste streams will be generated as indicated below and in Table 6.1. Two of these types are classified as mixed waste. The radionuclide distribution and subsequent classification will be determined in a different manner for each waste stream.

6.3.1 Contaminated Concrete Debris and Soil

Sample aliquots will be taken from batches of containerized waste that originate from a discrete work area and therefore would exhibit radionuclide distributions similar to those batches. The samples will be analyzed by gamma spectroscopy. Based on the results of sampling, an average radionuclide concentration will be assigned to that batch of waste. Decay corrected study data will be used to (1) determine that non-gamma emitters are not present or (2) determine the ratio (scaling factors) of non-gamma emitters to gamma emitters, e.g. Co-60 to other corrosion/activation producers and Cs-137 to fission product and transuranic elements.

6.3.2 Dry Active Waste

Dry active waste (such as contaminated plastic, paper, wood, and asbestos) will be classified using radionuclide distributions and concentrations estimated from smear sample data obtained from sample locations where the waste is generated. Based upon the results of the smear gamma spectral analysis, conversion factors will be established to convert smear results (dpm/100cm^2 of a particular radionuclide) to microcuries per mass of waste. Containers of waste will be weighed and the total quantity of each radionuclide calculated. The calculated quantity of each radionuclide would then be divided by the volume of the package and the volumetric concentrations determined.

6.3.3 Activated Components

Activated core components will be classified using radionuclide concentration data provided in the TLG Engineering Study. This study performed an activation analysis of various core components. Direct measurements of dose rates will be used to confirm this data as waste is generated.

6.3.4 Contaminated Piping and Equipment

Decay corrected radionuclide concentration data from the Cintichem Characterization Study will be used to classify piping and equipment wastes. This study provides internal surface contamination data on a system by system basis. Estimates of internal surface area per unit mass of pipe or equipment will be made. The contaminated surface area per waste's mass factors will be used in conjunction with waste package mass, volume, and Cintichem Characterization Study data to obtain a volumetric radionuclide concentration for each waste package. Items to be placed in waste containers will be tracked by system and component geometry such that appropriate mass-to-surface area factors and characterization surface concentrations are used for each waste package. As a system is opened, direct contamination measurement and/or sampling and analysis will be used to benchmark characterization study data.

6.3.5 Activated Concrete

Waste packages containing activated concrete rubble will be classified using methods discussed under 6.3.3 above for activated core components. Once again, direct measurements of dose rate will be used to confirm this data as waste is generated.

6.3.6 Activated Lead

Activated lead is present at the face of the thermal column and on the north and south walls of the stall section of the reactor pool. These components will be classified using radionuclide concentration data provided in the TLG Engineering Study. Direct measurements will be used to confirm this data as waste is generated. Packages containing activated lead are mixed waste and will be appropriately separated from non-mixed waste packages to assure that they are disposed of at a suitable facility when one becomes available.

6.3.7 Contaminated Lead

Whenever practicable, contaminated lead will be decontaminated and disposed of as hazardous waste. In other cases, samples will be taken from batches of mixed waste that originate from discrete waste streams. Samples will then be analyzed by gamma spectroscopy and, based on these results, average radionuclide concentrations will be assigned to that batch of waste. Contaminated lead packages are mixed waste and will be appropriately separated from non-mixed waste as indicated in 6.3.6 above.

6.3.8 Solidified Evaporator Concentrates

Samples will be taken from each batch of evaporator concentrates and analyzed by gamma spectroscopy to determine radionuclide concentrations. Scaling factors will be used to determine the concentration of non-gamma emitting radionuclides. The calculated quantity of each radionuclide would then be divided by the volume of the package and the volumetric concentrations determined. The generation and origin of contaminated waste water will be closely monitored during decommissioning activities so that significant changes in radionuclide mixtures from this waste stream can be anticipated.

TABLE 6.1

SOLID RADWASTE QUANTITIES AND PACKAGING

Material/Component	(a) Disposal Volume ft^3	(b) Average Concentration Ci/ft^3	Total Curies	Highest 10CFR61 Classification
Contaminated Concrete Debris and Soil	79,906	0.0028	223.3	A
Dry Active Waste	8,247	Nil	Nil	A
Activated Components	369	9.58	3,534	C
Contaminated Piping and Equipment	37,367	0.0012	44.8	A
Activated Concrete	1,490	0.0577	86	A
Activated Lead	3.5(c)	(c) 29.45	103.06	B
Contaminated Lead	149	Nil	Nil	A
Solidified Evaporator Concentrates	60	3.4	204	A
TOTALS:	127,592		4,195.06	

(a) Packaged disposal volume
(b) Average concentration is derived from packaged disposal volume and total estimated activity
(c) Actual volume not packaged

6.4 Mixed Waste

The following types of mixed waste are present or will be generated:

- o Lead shielding that is contaminated;
- o Activated lead;
- o Contaminated hazardous chemicals.

The mixed waste will be handled through discussions with the EPA, New York State, and burial site regulatory bodies. It is anticipated that a subcontractor will be utilized to dispose of or store these wastes.

6.5 Packing and Shipping of Radwaste

Packaging and shipping of the waste will be carried out under the supervision of Cintichem personnel. Each waste container will be packaged, sealed, surveyed, labeled, classified, and loaded in accordance with applicable regulations and procedures.

The 10CFR61.55 classification of the waste that will be placed in each container is given (see Table 6.1 for the types of waste that will be included under each classification). The following is a list of the waste containers that will be used:

- o Low Specific Activity box/typically 4' x 4' x 6' - Low Specific Activity Class A waste. Typically used for bulk shipments of things like structural steel to a recycling facility;

- o Large LSA box/typically 8' x 8' x 20' or 8' x 8' x 40' - large container for the same material as a. above;

- o U.S.D.O.T. specification 7A/4 ft^3 - Class A wastes;

- o U.S.D.O.T. specification 7A/7.5 ft^3 - Class A wastes;

- o U.S.D.O.T. specification 7A/118 ft^3 - Class A wastes;

- o High Integrity Container/7.5 ft^3 - Class B and/or C wastes.

6.6 Volume Conservation Techniques

Cintichem will minimize the generation of radioactive waste volume. The measures listed below will be undertaken to help minimize the waste volume generated.

- o Some waste will be sent to an intermediate waste processor (i.e. Quadrex or SEG) for decontamination and/or volume reduction as appropriate. This will be mostly contaminated metallic waste;

- Cintichem will segregate radioactive waste from non-radioactive waste as practical;
- Limit contamination of clean materials;
- Items will be segmented where feasible to yield maximum utilization of container space;
- Container voids will be filled with non-compactible waste such as concrete rubble, soil, or scabble dust whenever possible;
- Where applicable, items will be compressed to reduce volume.

6.7 On-site Storage of Wastes

During decommissioning, the generated radioactive wastes will be shipped off-site for disposal as soon as is practical after its generation. Cintichem intends to ship waste on a continuous basis and, as such does not intend to store wastes on-site any longer than it takes to prepare waste containers, arrange shipping lots and load trucks. Therefore, long-term storage of wastes on-site is not anticipated for normal conditions.

7.0 Technical and Environmental Specifications

The Decommissioning Plan encompasses all operations to be performed following issuance of a dismantlement order through ultimate release of the facility from licensed conditions. The following Technical Specifications have been developed and included to support this plan. They will replace the current R-81 and SNM 639 License Technical Specifications. Since the only SNM on site will be residual contamination, and there will be less than 350 grams, no 10CFR70 license requirements are necessary. Residual SNM contamination will be covered under the by-product materials license. In addition, an Environmental Report has been written in order to allow the NRC to assess the environmental impacts associated with the decommissioning project. It has been included as Appendix H to this plan.

The Technical and Environmental Specifications will control conditions and set limits so that during decommissioning activities the industrial and radiation exposure to workers and the public shall be maintained ALARA and are some small fraction of the respective limits and guidelines (10 CFR 20, NIOSH, OSHA, etc.).

These Technical Specifications have been divided into the following categories:

1. Safety Limits
2. Limiting Safety System Settings
3. Engineered Safety Features
4. Surveillance Requirements
5. Administrative Controls

7.1 Safety Limits

Safety limits for the decommissioning are bounds on certain parameters important to safety which must be maintained for adequate control of the decommissioning activities. In some cases, Cintichem administrative limits may be lower than the maximum safety limits and these administrative limits will be applied to ensure that safety limits are not exceeded. The ultimate goal of the safety limits shall be to control individual and collective doses.

7.1.1 External Exposure

External Exposure for individuals in restricted areas during decommissioning shall not exceed the limits specified in 10 CFR 20.101 or 12NYCRR38, whichever is more limiting.

7.1.2 Internal Exposure

Internal Exposure from inhalation of radioactive material in air in restricted areas shall not exceed that which would result from the inhalation of the limiting quantities specified in 10 CFR 20.103.

7.1.3 Liquid Effluent Release

Liquid waste released from the site shall comply with 10 CFR 20.106.

7.1.4 Concentration of Airborne Radioactive Material in Restricted Areas:

Concentration of airborne radioactive material shall comply with 10 CFR 20.103.

7.1.5 Concentration of Airborne Radioactive Material in Unrestricted Areas:

Concentration of airborne radioactive material shall comply with 10 CFR 20.106.

7.1.6 Concentration of Airborne Non-Radioactive Contaminants:

Concentration of such contaminants shall not exceed the limits specified in the applicable industrial hygiene regulations.

7.2 Limiting Safety System Settings

Limiting Safety System Settings allow sufficient time for corrective action to ensure Safety Limits are not exceeded.

7.2.1 External Exposure:

In order to assure that employees are not exposed to levels of radiation in excess of federal and state limits, and in an attempt to keep all exposure as low as reasonably achievable, guidelines are established to assist in the management of personnel radiation exposure.

	Four Weeks	Eight Weeks	Ten Weeks
Whole Body	0.5 Rem	0.8 Rem	1.0 Rem
Extremities	7.5 Rem	10.0 Rem	12.5 Rem

If an individual's cumulative exposure reaches any guideline, steps will be taken to assure that any further exposures to that individual during that calendar quarter will occur on a planned basis only.

7.2.2 Internal Exposure:

a. Airborne Radioactivity

Radiation Work Permits (see Section 2) will be required when work is performed in areas with airborne radioactivity. Respiratory protection will be used when the airborne concentration exceeds 25% of the total MPC. Dust suppression/collection systems (as specified in the Radiation

Work Permit) will normally be required during decommissioning activities prone to generate radioactive airborne particulates.

b. Airborne Radioactivity Effluent

The limiting safety system setpoint for radioactive airborne effluent is 25% of MPC at the point of discharge.

c. Liquid Effluent Releases

The limiting safety system setting for radioactive liquid effluent is 25% of MPC at the point of discharge.

7.3 Engineered Safety Features

These are features which if modified could have a significant effect on the safety of workers or the public.

7.3.1 Radiation Monitoring Systems

For maintenance or repair, required radiation monitors may be replaced by portable or substitute instruments for periods of up to 24 hours provided the function will still be accomplished. Interruption for brief periods to permit checking or calibration is permissible.

a. Exhaust Duct ("Stack") Monitor

A stack monitor, capable of detecting particulate radioactivity, will be used to measure the total radioactive airborne concentration leaving Buildings 1 and 2. The stack monitor will be operational until the ventilation system is disabled at which time airborne radioactive effluent will be monitored as per Table 7.1.

b. Building Continuous Air Monitor

Each building in which decommissioning activities are taking place will have a continuous air monitor until the buildings are breached. It will be used to detect particulate activity. If the continuous air monitor is not operational, a portable air sampler will be operated continuously with individual samples taken every four hours. The basis of this requirement is to ensure that radioactive particulates are not present in the general area.

c. Outside Work Area Monitoring

Continuous air sampling will be conducted at each active decommissioning work area when a potential for airborne radioactivity exists.

d. Area Radiation Monitors

Area radiation monitors (fixed or portable) shall be provided to give warning for higher than expected radiation dose rates for jobs where the dose rate could exceed 100 mR/hr.

7.3.2 Emergency Electric Generator

Upon loss of commercial power, a generator will start automatically and supply emergency power to the building ventilation system and portable HEPA ventilation units.

7.3.3 Building Exhaust HEPAs

While operational, the building 1 and 2 exhaust systems will have HEPA filters that are certified by the manufacturer to be 99.97% efficient.

7.4 Surveillance Requirements

Surveillance requirements during decommissioning activities comprise the test, calibration, and inspection activities necessary to ensure that systems, components, and instruments important to safety are operating in such manner that the monitored parameters and/or variables are maintained within the safety limits specified in 7.1.

7.4.1 Radiation Monitoring Systems

a. Definitions

<u>Channel Calibration</u>: A channel calibration is an adjustment of the channel so that its output responds, within acceptable range and accuracy, to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip.

<u>Channel Check</u>: A channel check is a qualitative verification of acceptable performance by observation of channel behavior.

<u>Channel Test</u>: A channel test is the introduction of a signal into the channel to verify that it is operable.

<u>Reporting Interval</u>: In all instances where the specified frequency is annual, the interval between tests is not to exceed 14 months; when semiannual, the interval should not exceed 7 months; when monthly, the interval shall not exceed 6 weeks; when weekly, the interval shall not exceed 10 days; and when daily, the interval shall not exceed 3 days.

b. While operational, the stack, continuous air monitors, and area radiation monitors shall be calibrated annually.

c. While operational, the stack, continuous air monitors, and area radiation monitors shall receive a channel test and alarm test monthly.

d. While operational, the stack, continuous air monitors, and area radiation monitors shall receive a channel check and a set point verification daily.

7.4.2 Emergency Electric Generator

While required, the ability of an emergency generator to start and run normally shall be checked weekly.

7.4.3 Radiological Environmental Monitoring

The radiological environmental monitoring program shall be conducted as specified in Table 7.1.

Basis: This section provides measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from the facility decommissioning.

7.4.4 Building Exhaust HEPA

While operational, the pressure drop across the building 1 and 2 exhaust system HEPA filters will be checked weekly.

7.5 Administrative Controls

Administrative Controls during decommissioning are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure completion of decommissioning the facility in a safe manner.

a. Responsibility

The Plant Manager shall be responsible for the completion of the decommissioning of the facility.

b. Organization

The organizational structure for management and performance of the decommissioning activities is shown in Section 1.5. The functions, responsibilities, and minimum required qualifications/experience of each position in Level I and Level II management are also detailed in Section 1.5 of the Decommissioning Plan.

c. Records and Reports

Accurate and complete records shall be maintained by Cintichem of the exposure of workers or the public to radiation in accordance with Section 2 of the Decommissioning Plan. The records will be maintained by Hoffmann-La Roche after decommissioning.

Reports pertaining to decommissioning activities shall be written and submitted to the proper authorities pursuant to Regulatory Guide 1.86.

d. Review

Responsibility for review of procedures, practices, and performance shall rest with the appropriate individuals and/or Level I or Level II management.

TABLE 7.1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE	1 Sample from Laurel Ridge Area (Station D)	Continuous Sampling*	Particulate sampler
	1 Sample from perimeter at gas house (Station A)	Continuous Sampling*	Particulate sampler
	1 Sample from Tuxedo Park residential area (Station C)	Continuous Sampling*	Particulate sampler
	1 Sample from Sterling Forest/ Long Meadow Rd area at Sterling Forest Maintenance Shed (Station B)	Continuous Sampling*	Particulate sampler
	1 Sample from NE site perimeter adjacent to Indian Kill Reservoir (Station E)	Continuous Sampling*	Particulate sampler
*Weekly filter change			Perform gamma isotopic analysis on each weekly particulate sample.

At the beginning of decommissioning operations, two additional monitoring stations will be located in a southerly and westerly direction from buildings 1 and 2 within the site perimeter.

2. DIRECT RADIATION	Same as No. 1 above plus the Clinton Woods residential area	Continuous**	Gamma dose. At least once per 92 days.

**With TLG change/readout at least once per 92 days.

TABLE 7.1 (continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
3. INGESTION			
Water	Indian Kill Inlet	Quarterly	Mixed Gamma Fission Products
	Indian Kill Outlet	Quarterly	Mixed Gamma Fission Products
	Indian Kill Outlet below sewer plant outlet	Quarterly	Mixed Gamma Fission Products
	Warwick Brook	Quarterly	Mixed Gamma Fission Products
	Sterling Lake Outlet	Quarterly	Mixed Gamma Fission Products
	Jones Spring	Quarterly	Mixed Gamma Fission Products
	Ramapo River	Quarterly	Mixed Gamma Fission Products
	Holding Pond Outlet	Quarterly	Mixed Gamma Fission Products
	Indian Kill Reservoir Water Intake	Quarterly	Mixed Gamma Fission Products
Fish	Indian Kill Reservoir Number determined at time of sampling	Spring of year after spring overturn	Mixed fission products
Benthos	Indian Kill Reservoir Number determined at time of sampling	Annually	Mixed fission products

TABLE 7.1 (continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
4. OTHER			
Sediment	Indian Kill Reservoir	Annually	Mixed fission products
	Number determined at time of sampling		
Sewage Sludge	Treatment Plant	Annually	Mixed fission products
Soil	Selected sites surrounding plant	Annually	Mixed fission products

8.0 Proposed Termination Radiation Survey Plan

The goal of decommissioning under the DECON option is to release the site for unrestricted use. The termination radiation survey will be conducted in order to ensure that the residual radioactivity on the site is within the release limits specified in Section 8.1. The termination radiation survey will be conducted after decontamination and dismantlement work detailed in Section 3 has been completed. As described in Section 3, the final radiation survey may be performed incrementally as the decommissioning progresses.

The detailed plan for the termination radiation survey will depend on:

- The initial site characterization study and radiological history of the site;
- The details and sequence of the decontamination and dismantling process;
- The results of incremental surveys conducted during the decontaminating/dismantling process.

8.1 Regulatory Criteria

8.1.1 Unrestricted Release on Site

The Nuclear Regulatory Commission (NRC), and New York State Department of Labor (DOL) as the licensing authorities for Cintichem, have the authority to define requirements for releasing materials, structures, and the site for unrestricted use. To date, the NRC has no official "de minimus" levels authorizing releasable limits. However, the NRC has published Regulatory Guide 1.86 which details acceptable surface contamination levels below which a radioactive material license may be terminated. This is a guide, not a law, and is thus subject to interpretation contingent upon NRC approval on a case-by-case basis. New York State provides instructions for the decontamination of facilities and equipment in Section 38.29 of Industrial Code Rule 38 Table 5.

NRC Regulatory Guide (RG) 1.86 criteria, shown in Table 8.1 and Table 5 of Section 38.29 of 12NYCRR38, shown in Table 8.2, do not indicate agreement between the regulating authorities for allowable residual contamination limits. For example, if soil contamination is detected, the dose rate limit at 1 meter of 5 uR/hr is suggested by NRC's NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities", and the NRC's Draft Guide for Decommissioning Plans for Research and Test Reactors; NYS, on the other hand, uses specific isotopic concentration limits for soil (12NYCRR 38, Table 2, Col. 2). The final release criteria applied in the Cintichem decommissioning will be the most limiting of the USNRC and NYSDOL specifications presented in Tables 8.1 and 8.2

8.1.2 Unrestricted Off Site Release Criteria

Any materials to be released off-site will have no detectable radioactive material other than what is naturally occurring.

8.2 Sampling Protocol

This section describes the methods that will be used to document the radiological status of the structures, equipment and the exterior environment of the Cintichem reactor and hot laboratory facilities during the final radiation survey.

This section is divided into four subsections:

- o Structural Survey;
- o System and Equipment Survey;
- o Activated Material Survey;
- o Environmental Survey.

8.2.1 Structural Surveys

8.2.1.1 Data Needs

The desired survey results, achieved by various survey methods, include:

- o The amount of surface contamination (total/removable) present;
- o The isotopic distribution of nuclides present if previously unknown;
- o Radiation dose rates from potentially activated structures.

8.2.1.2 Applicable Areas and Items

The objective of the Cintichem decommissioning is to raze buildings 1 and 2 and associated structures and systems. Prior to demolition of all or sections of these structures, acceptable survey results of major areas and components will be required. The termination survey will include the interior and exterior surfaces of the reactor and hot lab buildings. The survey will also include exterior surfaces adjacent to these buildings that could also have been affected by operations. These will include items such as ramps to doors, sidewalks, sumps, manholes, soil adjacent to foundations and basement rooms or other building structures below grade.

8.2.1.3 Selection of Sampling Locations

8.2.1.3.1 Unbiased Survey

This survey type is characterized by a systematic placement of sample/measurement locations that enables an unbiased representative survey of areas which may or may not contain pockets of contamination. Selection of these areas will be achieved by review of historical data; that is, by selecting those areas from routine area radiation surveys and prior characterization surveys that have no known surface contamination or are likely to be homogenous in radiological condition. The unbiased structural survey requires a measurement regime that provides a 95 percent confidence for determining the true mean surface contamination levels (both total surface and removable) to within plus or minus 20 percent. This degree of confidence will be achieved by making 30 evenly spaced initial measurements on a surface. These initial measurements will be used to determine whether the population of measurements meet the desired confidence and error limits. These measurement results will also be used to estimate the number of additional measurements that will be needed to achieve the required accuracy in determining the true mean surface contamination level.

Structures will be divided into manageable survey areas (e.g., a section of a building elevation, the outside wall of a building, etc.) to make data handling more manageable. These survey areas are further broken down into survey populations that would be expected to have similar modes of contamination deposition patterns within the survey area. (See Figures 8.1a and 8.1b and Table 8.3 for a typical presentation of survey points and data). Typically areas will be divided into the following survey populations, as applicable:

- o FLOORS - areas of potential spills, areas of heavy traffic;
- o WALLS - less than 4 feet high - potential for splashes from nearby equipment;
- o WALLS - greater than 4 feet high - settling of dust, sprays or pipe leaks;
- o CEILINGS - duct leaks, contaminated air circulation.

Each survey area will be divided initially into no less than 30 equal sections, evenly spaced over the entire surface area where possible. Measurements taken in each population will consist of (1) total surface beta/gamma open and closed window contamination measurements with a Geiger-Mueller (GM) shielded "pancake" type detector, then (2) smear samples counted for gross beta and gross alpha activity. Additionally, total surface alpha contamination measurements will be made in areas where alpha contamination could exist. Gamma dose rate measurements will be made at one meter above potentially activated survey points. Once all 30 sections of the survey unit have been surveyed and the results tabulated, the sample results will be corrected by a calculated background value at the 95% confidence level for the background distribution for the type of material being surveyed (i.e., concrete, metal, etc.) (see Appendix I). If the results are not equal to or less than background, the area will be considered potentially contaminated and additional decommissioning work will be performed.

After the initial 30 measurements are made for floor, wall and ceiling survey populations, a number of additional measurements will be made as based upon the requirements specified at the beginning of this section.

The actual number of survey points in each area shall be systematically determined. Floors, walls and ceilings will be divided by a rectangular grid system. The blocks formed by this grid system will be referred to as "survey blocks" and the corners of the survey blocks shall be called "grid points". The factors which will guide the set up of a grid system are as follows:

o No survey block will have a side measuring less than 1 meter;

o No survey block will have a side measuring more than 3 meters;

o There will be at least N survey blocks as determined by the following equation[1] (unless this violates the minimum 1 meter grid above).

$N = (ts/rx)^2$, where

N = required number of measurements,

t = the t statistic for 95 percent confidence and 29 (N-1) degrees of freedom (1.699),

r = acceptable relative error (0.20),

x = the average surface radiation level,

s = standard deviation of average surface radiation level.

The radiological conditions to be characterized will include surface beta-gamma contamination levels by direct instrument survey, and removable beta-gamma and alpha contamination levels.

At 1 meter above the center of each survey block where activation could have occurred, the gamma exposure rate will be taken and compared to the NRC 5 uR/hr at one meter criteria. Within each survey block, five direct instrument beta-gamma contamination measurements will be made, at the corners and center of a one meter

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[1] NUREG/CR-2082, Monitoring for Compliance with Decommissioning Termination Survey Criteria

square centered in the middle of the survey block. If the entire survey block is only 1 square meter, then the corner measurement points will be moved 30 cm towards the center of the block. The five measurement values are averaged, and will represent an average level over 1 square meter; this is considered an "unbiased" measurement for that survey block. The "unbiased" measurement results will be compared to NRC "average area" criteria.

The survey block will next be scanned with an open thin window GM detector and ratemeter to locate the maximum beta-gamma contamination point. At this point, each type of measurement will be made, including a smear sample for removable beta and alpha contamination. These measurements are considered the maximum level within the sample block and is considered a biased measurement. This will be compared to NRC "maximum" surface contamination criteria.

8.2.1.3.2 Biased Survey

Biased sample locations are those that will be selected for the survey based upon there being a likelihood of hidden contamination that was removed during decontamination work. Examples of areas where biased surveys will be conducted include:

- structural surfaces upon which contamination was found in;
- floor drains;
- floors under contaminated components;
- cracks in structural surfaces;
- spill areas;
- horizontal surfaces.

Horizontal and vertical overhead surfaces will typically show uniform surface contamination with horizontal surfaces showing higher contamination levels than vertical surfaces. This is because deposition on upper surfaces tends to be due to settling of contaminated dust particles rather than by spills or leaks as on floors and lower walls. For this reason, the standard procedure will be to use the 30 initial "point" measurements and smears for each type of measurement on both vertical and horizontal surfaces.

Analytical Methods

.owing describes the analytical techniques that will be used ey potential structural contamination and the criteria under hey will be implemented at a sample location.

Removable Surface Contamination Determinations:

Smear samples will be taken at all selected sampling locations. All smear samples will be analyzed for gross alpha and beta activity.

If activity is detected, smears with the highest gross beta and/or alpha activity from each survey population will be selected for quantitative and qualitative isotopic analysis. These smears will be initially analyzed by gamma spectroscopy. Additional radiochemical analysis for non-gamma emitting radionuclides may be performed if gross smear results do not correlate well with gamma spectral analysis results. A smear with the highest gross alpha activity from the reactor, hot lab and hot cells will be analyzed for uranium and transuranic isotopes.

Instruments used for contamination determination will be sensitive enough to provide assurance the NRC or NYSDOL criteria for contamination limits can be measured.

Total Surface Beta Contamination Determination:

Total surface beta contamination (fixed and loose) levels will be determined at each sample location selected for the survey. These determinations will be made by direct instrument measurements on the surfaces of interest, using hand held GM detectors and ratemeters. Each measurement will be made with the detector held one centimeter or less from the surface of interest. Shielded thin-window (1 - 2 mg/cm^2) pancake type detectors, or equivalent, will be used whenever feasible for the total surface beta measurements. The GM detection equipment in general will have a sensitivity of about 3000 - 4000 $dpm/100cm^2$ for betas having an average energy of approximately 100 keV. Open and closed window count rates will be determined at each location to obtain beta-gamma and gamma count rates respectively. These separate count rates will be used to separate the surface beta count rate from the count rate caused by area gamma activity. The net beta count rate obtained will be corrected for instrument background, detector area and efficiency to yield measurement results in betas/minute per $100cm^2$. The total surface beta results will be corrected for natural radioactive material background (concrete, cinder block, wood, steel etc., see Appendix H).

Bulk Sampling and Analysis

If or when encountered, sediment deposits will be collected on a case by case basis for isotopic analysis. Types of analysis to be specified will be the same as described for smear samples.

Gamma Exposure Rate

Gamma exposure rates for the structural survey will be taken at one meter above measurement locations. Measurements will be made with a gamma scintillation exposure rate meter, such as an Eberline Micro R meter.

Total Surface Alpha Contamination Determinations

Total surface alpha contamination determinations will be made by direct measurement with Zinc-sulfide alpha scintillation detectors and rate meters. Direct alpha measurements will be taken at the following locations:

- o At survey locations that had positive removable alpha results;
- o At survey locations that had the highest total surface beta activity within a survey population;
- o At all survey locations where alpha emitting material was known to have been present (e.g. Uranium solution plating lab).

8.2.2 System and Equipment Characterization

Prior characterization activities determined the radiological characteristics of contaminated systems which are within the Reactor and Hot Lab buildings. The scope of these characterization activities included the interior surfaces of components. Such components would have been removed during decommissioning work. In the event components are to be salvaged and released for unrestricted use they will be deemed to be potentially contaminated.

Potentially contaminated systems will be surveyed by selecting representative components for direct measurement and sampling for radioactive contamination. The components that are selected will be opened to expose the interior surfaces for survey. The survey will include determinations of total surface and removable gross beta contamination and, gross alpha contamination, if required.

8.2.2.1 Data Needs

This section sets forth the methodology by which potentially contaminated systems and equipment will be surveyed for release. The data that will be obtained as part of the system and equipment survey will include and be used for:

- o Determination of total surface and removable contaminant activity levels on external and internal surfaces of systems;
- o Comparison of survey data with NRC and/or NYSDOL criteria to assure limits are not exceeded.

8.2.2.2 Applicable Areas and Items

Any remaining equipment and systems within or adjacent to the reactor and hot lab buildings that must be released and that could conceivably contain radioactive materials will be surveyed.

Contamination determinations should be made on the internal side wall and on the internal bottom surface where contamination levels could differ because of gravitational settling of potential contaminants.

8.2.2.3 Analytical Methods

The following describes the analytical techniques that will be used to survey systemic contamination and the criteria under which they will be implemented at a sample location.

Removable Surface Contamination Determination:

Smear samples will be taken from the internal surfaces of equipment or components to be surveyed. All smear samples will be analyzed for gross beta activity and at least one smear sample with the highest detectable beta activity from each system will also be counted for gross alpha activity.

If activity is detected, smears with the highest gross beta activity of contamination from each system will be selected for quantitative and qualitative isotopic analysis, such that appropriate criteria can be used. These smears will be analyzed by high resolution gamma spectroscopy. Additional radiochemical analysis for non-gamma emitting radionuclides may be performed if gross beta results do not correlate well with gamma spectral analysis results. Smears with gross alpha activity above release criteria will be analyzed for uranium and transuranic isotopes.

Instruments used for contamination determination will be sensitive enough to provide assurance that release criteria limits can be measured.

Total Surface Contamination Determinations:

Total surface contamination levels will be determined on internal surfaces of equipment and components. These determinations will be made by direct instrument measurements on the surfaces of interest, using hand held GM detectors and ratemeters. Each measurement will be made with the detector held as close to the surface of interest with "open" and "closed window" count rates to separate surface beta activity from area gamma activity.

Thin-window (1 - 2 mg/cm^2) shielded pancake type detectors, or equivalent will be used whenever feasible. This detection equipment in general will have a sensitivity of about 3000 - 4000 dpm/100cm^2 for betas having an average energy of approximately 100keV. The count rate obtained will be corrected for instrument background, detector area and efficiency to yield measurement results in betas per minute per $100cm^2$. The beta emission rate will be converted to $dpm/100cm^2$ based upon the isotopic mix identified for the system and the associated decay schemes.

Bulk Sampling and Analysis:

If or when encountered, sediment deposits will be collected, on a case by case basis, for isotopic analysis. Types of analysis to be specified will be the same as was described for smear samples.

8.2.3 Environmental Survey

The scope of the Environmental Survey will include existing surface soil of the site and soil surfaces exposed during decommissioning. There will be two general categories of environmental surveys done in the final survey plan. Biased surveys will be conducted in those areas where contamination was removed during decommissioning and where contamination is liable to exist as a result of decommissioning operations. Unbiased surveys will be conducted on site in a systematic random manner. The survey will entail measurement for USNRC soil criteria for gamma emitters (5 uR/hour at one meter).

8.2.3.1 Biased Environmental Survey

The biased environmental survey will consist of direct radiation measurements at one meter from the soil (1) under any remaining reactor and hot lab structures, and (2) where effluent discharge points are known or likely to occur. (Refer to Figure 8.2).

8.2.3.1.1 Soil Under Remaining Structures

Three sources of sub-building soil contamination are currently known or suspected. They are:

- Hold up tank;
- Gamma pit/canal;
- Buried hot cell HVAC duct.

These structures will be removed during decommissioning along with any remaining contaminated soil. The remaining soil will be surveyed for gamma exposure rates at 1 meter.

8.2.3.1.2 Effluent Discharge Points

Surface soil and/or sediments will be surveyed on a biased basis i.e., where sources of discharge to the ground surface are known or likely to have occurred. These areas are:

- The perimeter of the primary storage tank and supply pipeline;
- The ground area surrounding the exhaust discharge stack;
- The ground surface in or adjacent to the Class "A" waste storage building.

8.2.3.2 Unbiased Environmental Survey

The unbiased environmental survey will consist of a systematic random radiation survey of the entire site area. The site will be divided into discrete areas by a 50' by 50' grid network. The grid network will be referenced to site survey benchmarks for accurate location of survey points. (Refer to Figure 8.3).

At each accessable grid point a gamma measurement will be taken at 1 meter from the soil surface with a uR meter. Measurement results will be compared to the NRC 5 uR/hr criteria.

8.3 Final Survey Report

The radiological survey report is written in two sections. The first section consists of an overview of the radiological condition of the site given in the text of the report with figures illustrating specific radiological conditions. The second section consists of a detailed presentation of data in the form of tables or figures. The original data will be retained.

8.4 Quality Assurance/Quality Control Requirements

Quality Control will be maintained utilizing existing Cintichem procedures[2]. Data will be reviewed in accordance with existing requirements to ensure reasonable interpretation of results. Samples and measurements at all locations will be collected using accepted documented procedures to ensure interpretable results and consistent collection techniques. On-site sampling and measurement instruments will be subject to daily operational checks and periodic calibration to ensure both accuracy and precision of results. Records and calculations will be checked for accuracy and appropriate recording and calculation techniques. Independent laboratories will verify data quality per existing procedural requirements. Existing chain-of-custody procedures will be implemented to maintain the integrity of samples and corresponding analytical results. This program will ensure that the survey data can be used to accurately evaluate site conditions.

To ensure measurement quality, the following controls will be implemented:

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2 Radiological Measurements Quality Assurance Manual

- Five percent of soil samples will be split and analyzed by a separate laboratory;
- Five percent of swipes (system & structure) will be reanalyzed;
- Five percent of direct measurement of total surface contamination will be retaken by an independent survey team;
- Quality control replicate sample results should fall within the two sigma interval of the original result. Non-conformance will require investigation and resolution.

TABLE 8.1

U.S. NUCLEAR REGULATORY COMMISSION
REGULATORY GUIDE 1.86

Acceptable Surface Contamination Levels

Nuclides[a]	Average[b,c,f]	Maximum[b,d,f]	Removable[b,e,f]
U-nat, U-235, U-238, and associated decay products	5,000dpm α/100cm^2	15,000dpm α/100cm^2	1,000dpm α/100cm^2
Transuranics, Ra-226, Ra-228 Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100dpm/100cm^2	300dpm/100cm^2	20dpm/100cm^2
Th-nat, Th-232, Sr-90 Ra-223, Ra-224, U-232, I-126 I-131, I-133	1,000dpm/100cm^2	3,000dpm/100cm^2	200dpm/100cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and other noted above.	5,000dpm $\beta\gamma$/100cm^2	15,000dpm $\beta\gamma$/100cm^2	1,000dpm $\beta\gamma$/100cm^2

[a]Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than one square meter. For objects of less surface area, the average should be derived for each such object.

[d]The Maximum contamination level applies to an area of not more than 100cm^2.

[e]The amount of removable radioactive material per 100cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f]The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1cm and 1.0 mrad/hr at 1cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

TABLE 8.2

NEW YORK STATE RADIOLOGICAL RELEASE CRITERIA*

A) Surface Contamination Limits

1. Alpha Emitters

(i)	Removable:	33dpm/100cm^2	Average
		100dpm/100cm^2	Maximum
(ii)	Total (fixed):	1,000dpm/100cm^2	Average
		5,000dpm/100cm^2	Maximum
		0.25 mrem/hr at 1cm	

2. Beta-Gamma Emitters

(i)	Removable:		
	- all except H-3	222dpm/100cm^2	Average
		1,000dpm/100cm^2	Maximum
	- H-3	2,220dpm/100cm^2	Average
		11,100dpm/100cm^2	Maximum
(ii)	Total (fixed):	0.25 mrem/hr at 1cm	

B. Air and Water:

Table 6, Schedule II (Industrial Code, Rule 38, June 25, 1985)

C. Soil and Other Materials:

1. Table 2, Col. 2 (Industrial Code, Rule 38, June 25, 1985) except source material.

2. Source Material: 0.05% by weight.

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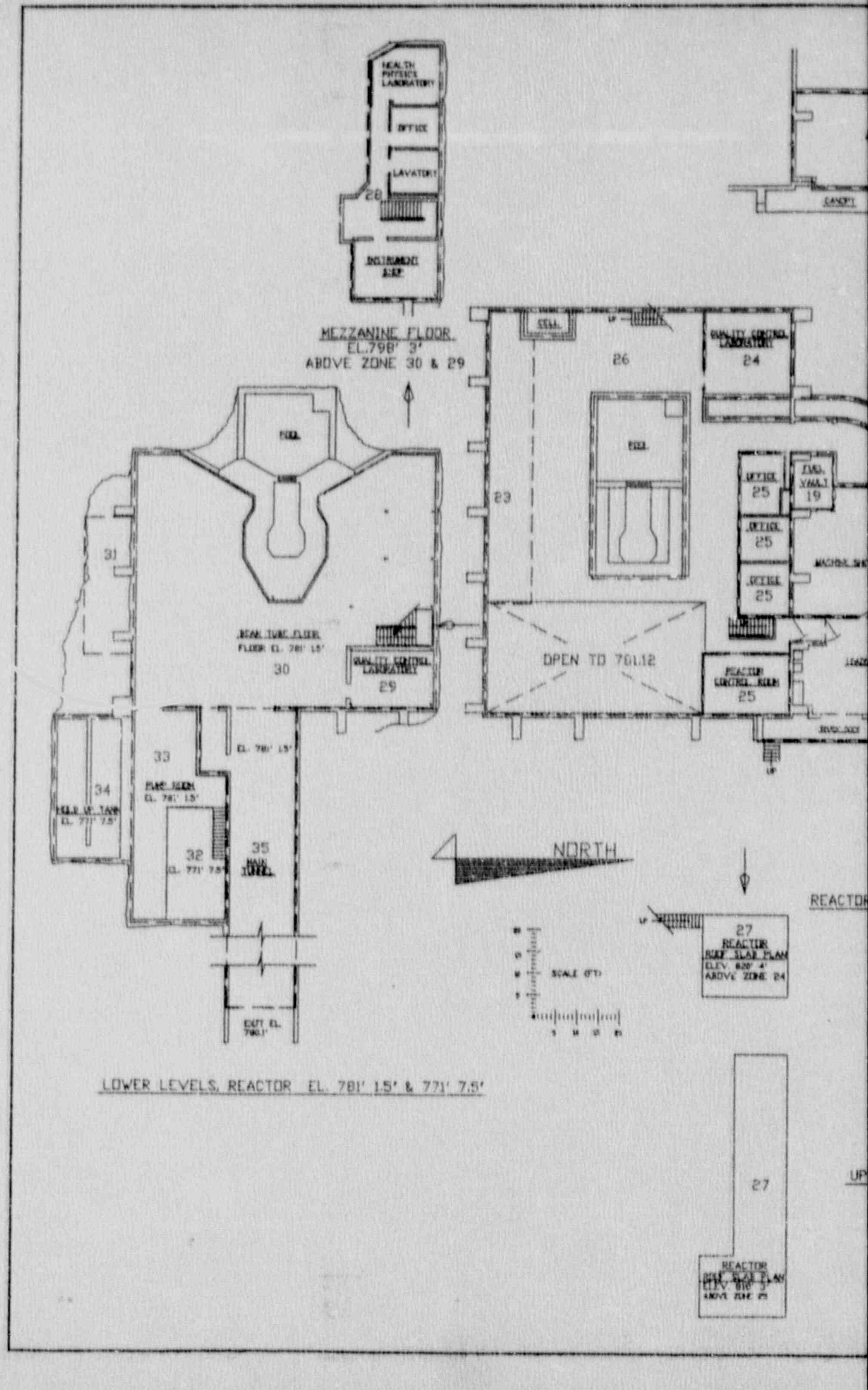
*Facility Remains a Commercial/Industrial Facility.

TABLE 8.3

WASTE PIT BUILDING - 1ST FLOOR

- - - - -dpm per 100 cm2- - - - - -

SAMPLE NUMBER	DIRECT ALPHA	DIRECT BETA	REMOVABLE ALPHA	REMOVABLE BETA	uR/hr AT ONE METER
1	0	0	0	443	149
2		20264	0	953	
3		0	0	0	
4	0	109285	0	1023	
5		25119	0	2995	459
6		12171	0	1542	
7		28356	0	765	
8		12171	0	526	
9		0	3	1735	
10	0	135182	0	2646	
11	0	83388	0	2636	
12		28356	0	1647	
13		21882	0	1430	
14		25119	0	1806	
15		0	0	653	
16		25119	0	709	3589
17		12171	0	733	
18	0	31594	0	802	
19	0	31594	0	984	
20	0	63965	0	536	1089
21	0	80151	0	1172	
22		18645	0	9954	
23	0	63965	0	110	
24	0	31594	0	305	
25		5696	0	383	
26		8934	0	492	
27		12171	0	3124	
28		2459	0	1233	
29		5696	0	517	
30		0	0	1011	2789
n =	30	30	30	30	
Min. =	0	0	0	0	
Max. =	0	135182	3	9954	
x =	0.0	29835	0.1	1429	
St.Dev. =	0.0	34013	0.6	1810	


HEALTH PHYSICS LABORATORY
OFFICE
LAVATORY
28
INSTRUMENT SHOP
MEZZANINE FLOOR
EL.798' 3'
ABOVE ZONE 30 & 29
CELL
26
QUALITY CONTROL LABORATORY
24
POOL
23
OFFICE
25
FUEL VAULT
19
MACHINE SHOP
OPEN TO 781.12
REACTOR CONTROL ROOM
25
31
BEAM TUBE FLOOR
FLOOR EL. 781' 1.5'
30
QUALITY CONTROL LABORATORY
29
33
EL. 781' 1.5'
PUMP ROOM
EL. 781' 1.5'
34
HOLD UP TANK
EL. 771' 7.5'
32
EL. 771' 7.5'
35
MAIN TUNNEL
NORTH
REACTOR
27
REACTOR ROOF SLAB PLAN
ELEV. 820' 4'
ABOVE ZONE 24
SCALE (FT)
EXIT EL.
780.1'
LOWER LEVELS, REACTOR EL. 781' 1.5' & 771' 7.5'
27
REACTOR
ROOF SLAB PLAN
ELEV. 810' 3'
ABOVE ZONE 29
CANOPY
UP

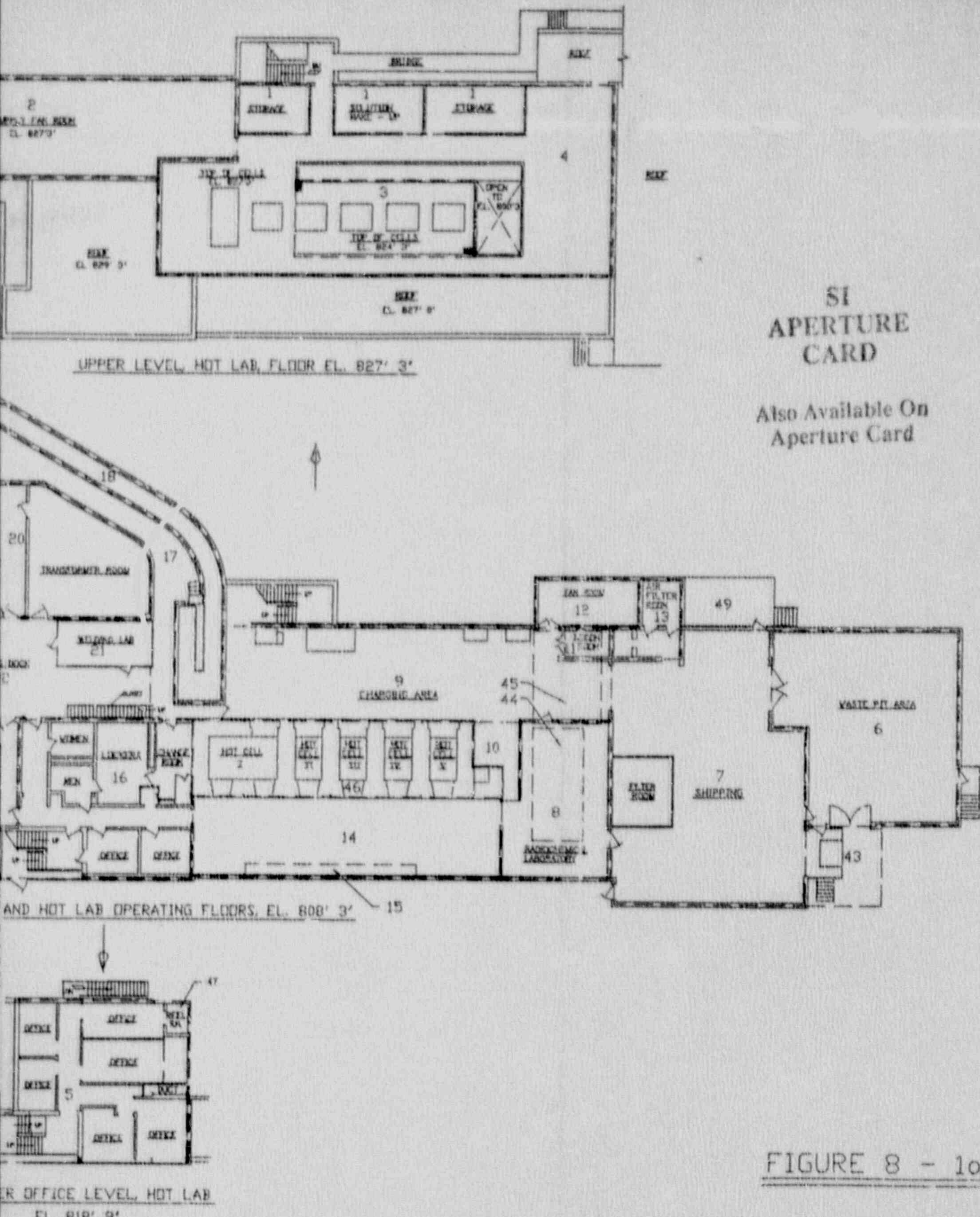


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FIGURE 8 - 1a.

APPR	DATE	DESCRIPTION	REV
	9/18/90	WASTE PIT AREA WAS WASTE STORAGE, AREA 49 ADDED	B
	8/9/90	INSTRUMENT SHOP WAS UTILITY ROOM, FUEL VAULT ADDED TO AREA 15 ELEV. 809' 4" WAS 818' 3", AREA 47 ADDED TO REEL ROOM	A

REVISIONS

DRAWN C.G.

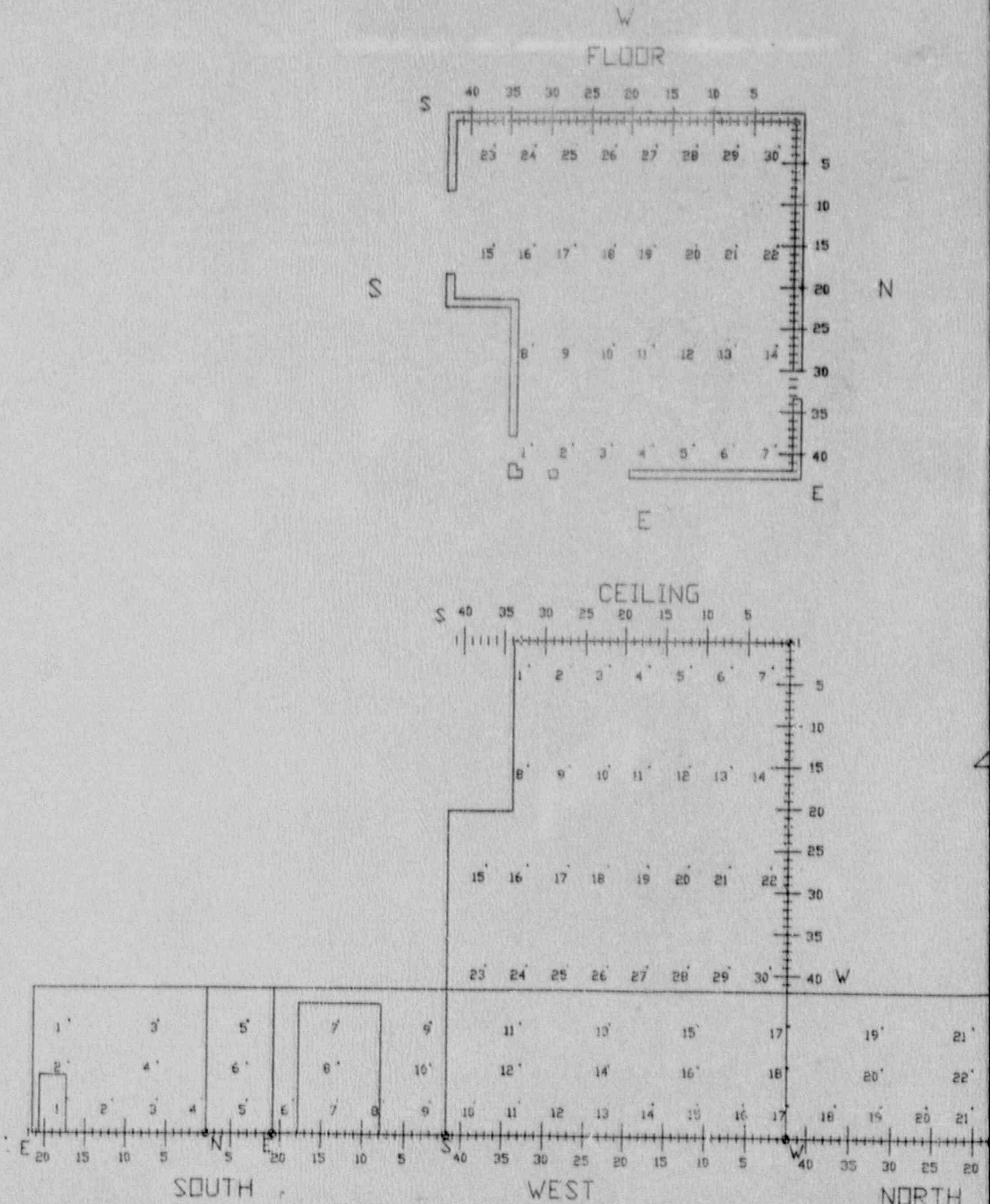
CINTICHEM, INC.

REACTOR HOT LAB OPERATING FLOORS

E

136E2328

9010230140-07



AREA 6

WASTE PIT BLDG.

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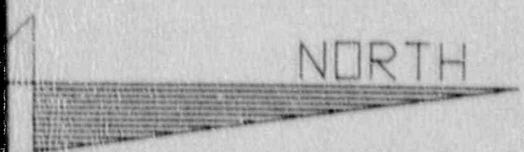
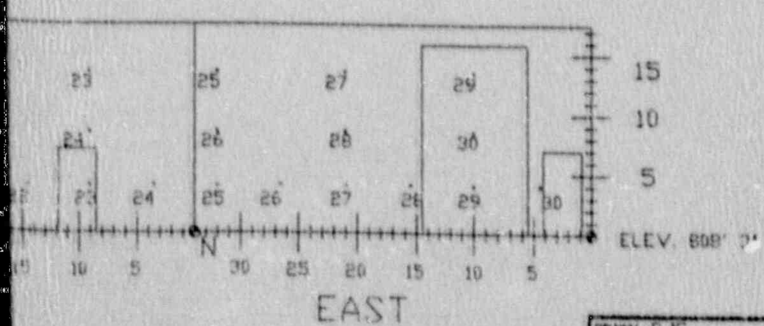


FIGURE 8.1 b

(NTS)



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CAD FILE # DWGS\ 2354		REVISIONS	

DRAWN CJG 8/29/90

DESIGN

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EQUIP. AT

UNLESS OTHERWISE NOTED: TOLERANCE +/-

FRAC DEC ANGLE

CINTICHEM, INC.

P.O. BOX 816, TUXEDO, NEW YORK 10987

WASTE PIT BLDG. AREA 6

SIZE D

DRAWING NO. 136D2354

SCALE

DIMENSIONS IN

SHEET

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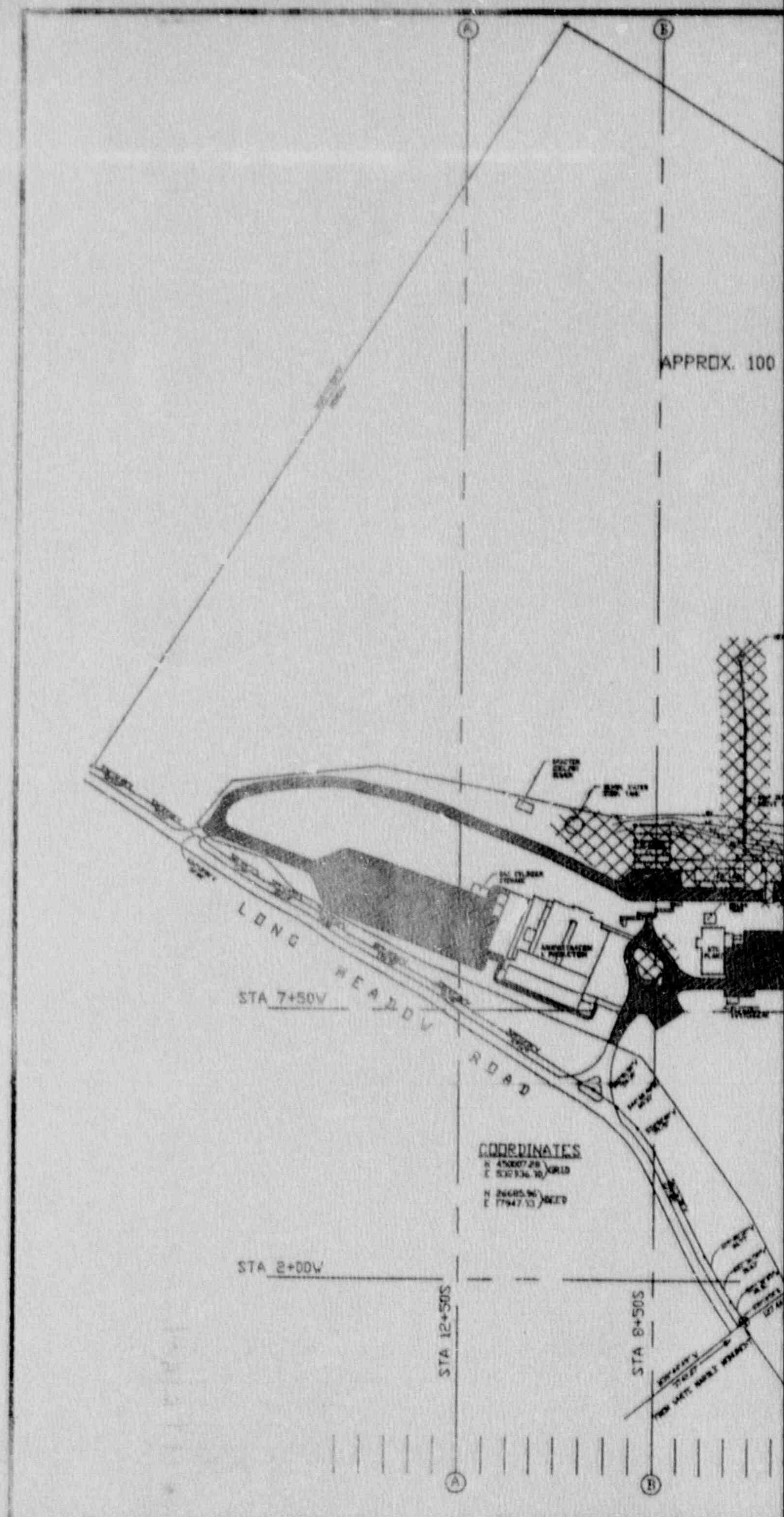
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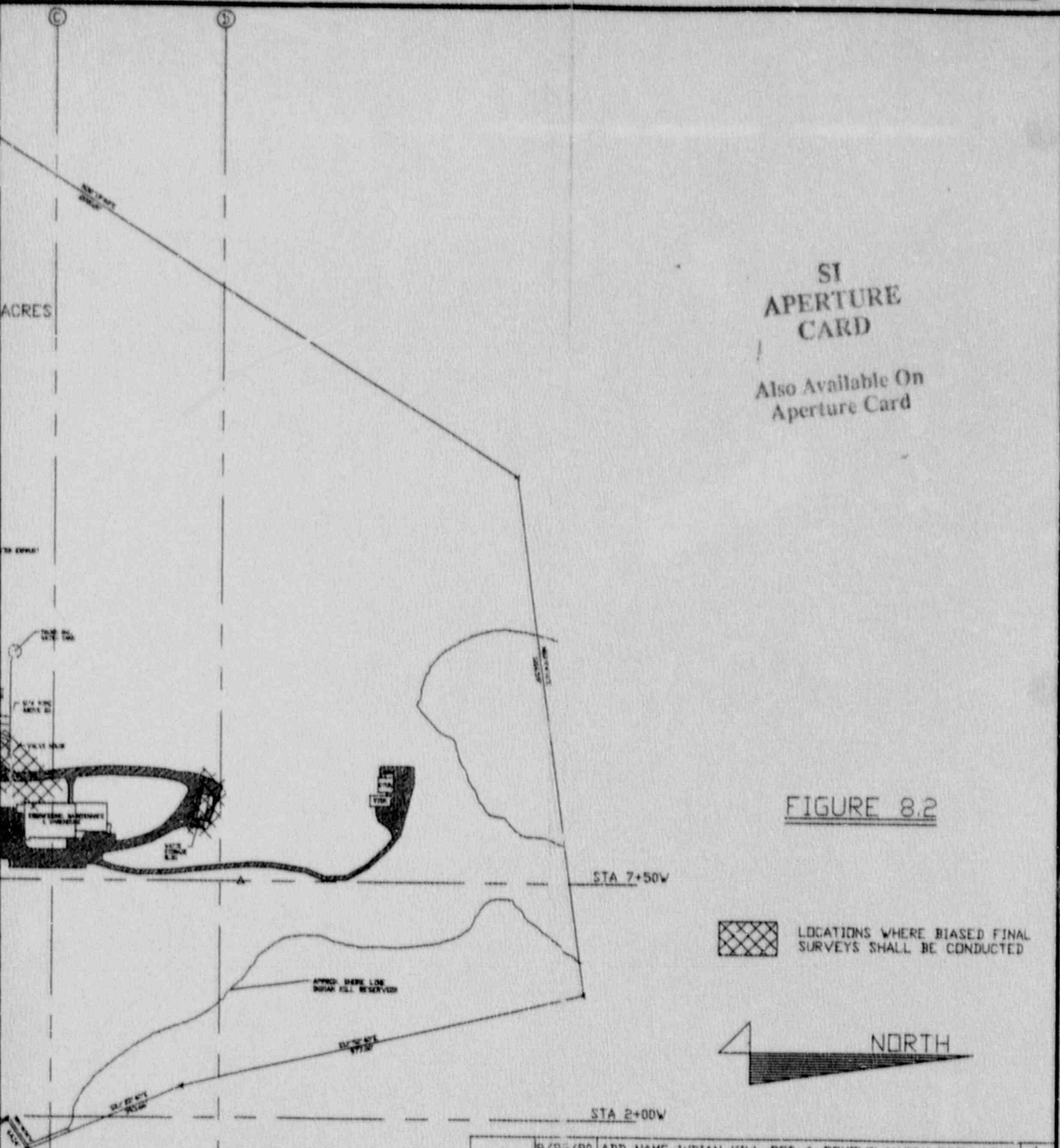
STA 12+50S
STA 8+50S
COORDINATES



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	9/25/90	ADD NAME INDIAN KILL RES. & REMOVE HATCH FROM POND	A
APPR.	DATE	DESCRIPTION	REV
CAD FILE # DWGS\ 2353		REVISIONS	2400 S.F.

DRAWN CJG	8/29/90	CINTICHEM, INC.
DESIGN		
CHK./APPR.		P.O. BOX 816, TUXEDO, NEW YORK 10987
EQUIP. AT	UNLESS OTHERWISE NOTED TOLERANCE +/- FRAC DEC ANGLE	FINAL BIASED SURVEY PLAN · SIZE C · DRAWING NO. 136C2353
		SCALE 1" = 200FT · SHEET

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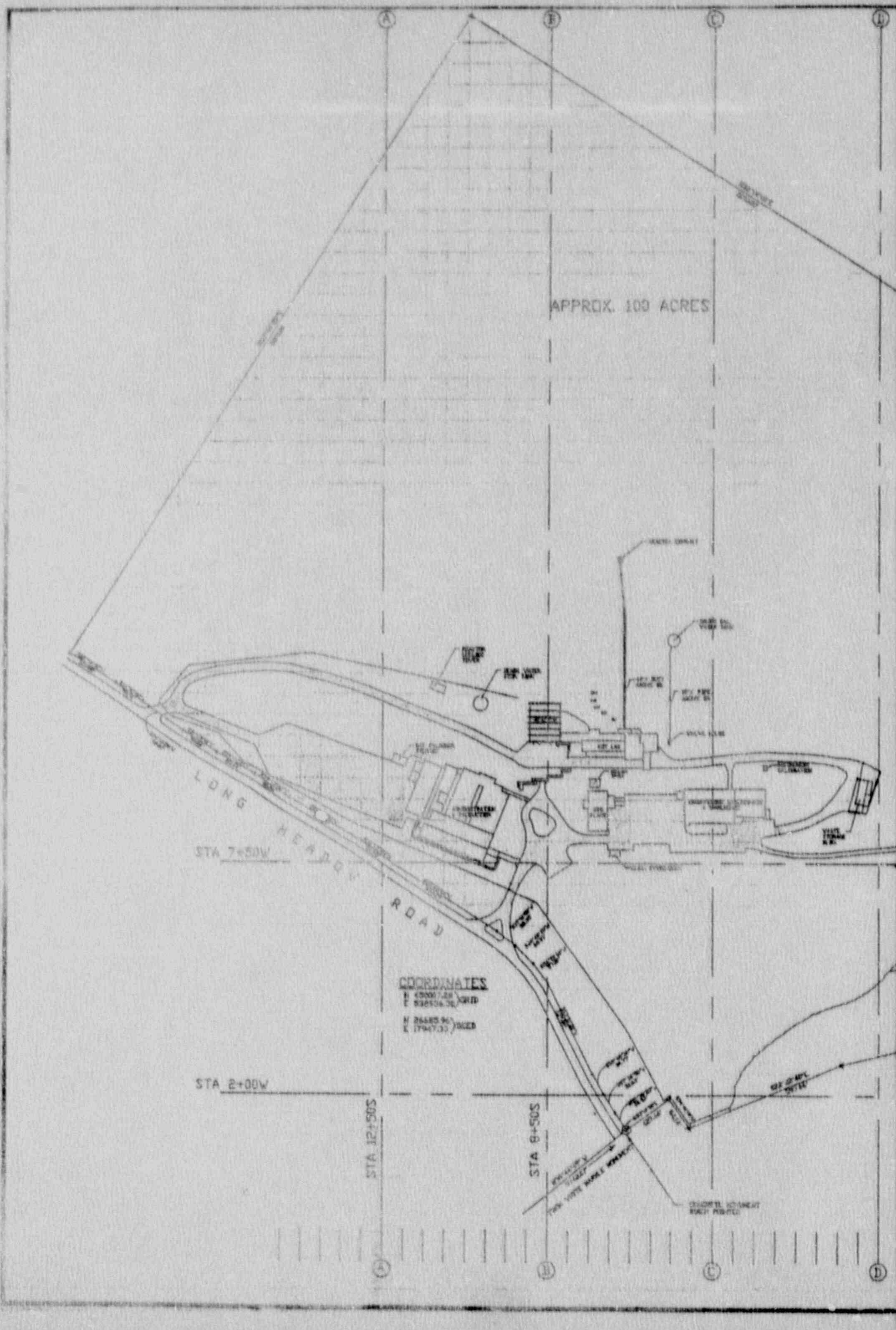

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STA 12+50S
STA 8+50S
COORDINATES

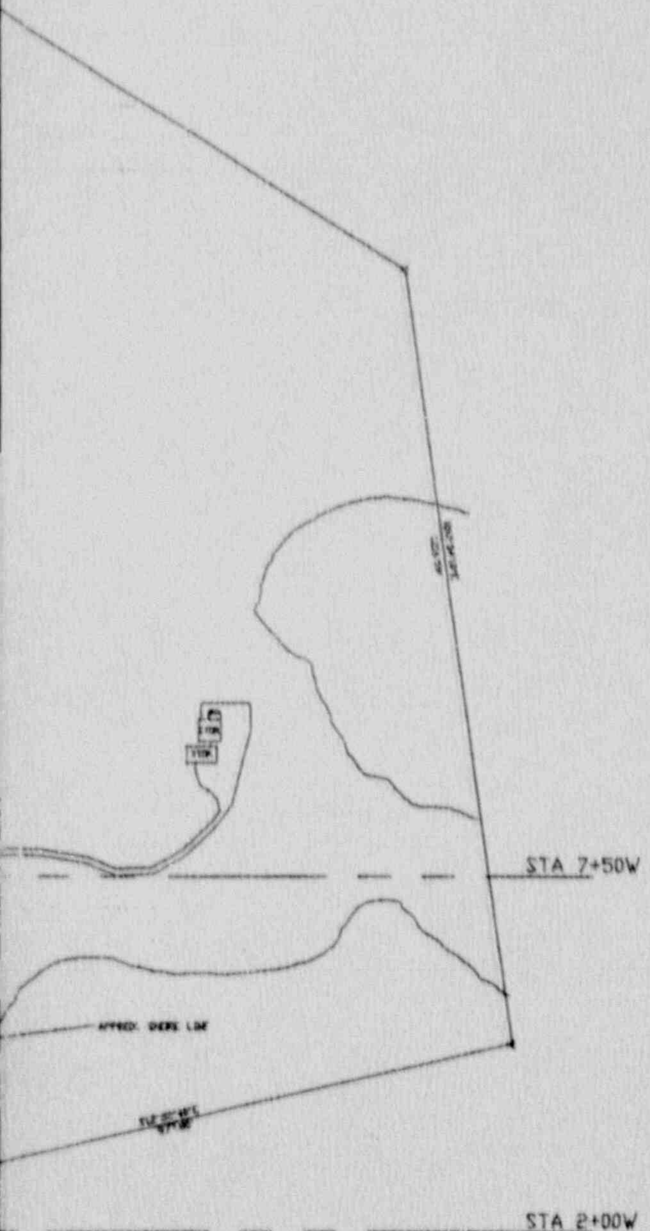
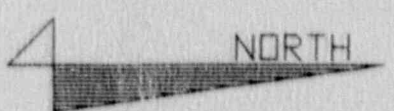


FIGURE 8.3

REPRESENTS 50' X 50' AREA



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DESIGN		
CHK./APPR.		P.O. BOX 816, TUXEDO, NEW YORK 10987
EQUIP. AT	UNLESS OTHERWISE NOTED: TOLERANCE +/− FRAC DEC ANGLE	FINAL UNBIASED SURVEY PLAN
SIZE C	DRAWING NO. 136C2352	
SCALE 1" = 200FT		SHEET

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APPENDIX A

QUALIFICATIONS OF DECOMMISSIONING STAFF

Position Title	Qualifications/Prerequisites
Plant Manager	B.S. Degree in scientific or engineering discipline and approximately 10 years of related nuclear operations and production experience.
Project Manager	B.S. Degree in a scientific and engineering discipline, and approximately 10 years experience in nuclear engineering or operations.
Manager, Health, Safety and Environmental Affairs	B.S. Degree in physics, biology, or science and approximately 10 years experience in the field of Health Physics and General Safety work.
Project Co-Manager (D & D Consultant)	B.S. Degree in scientific or engineering discipline and approximately 10 years of related experience in management of decommissioning at comparable facility.
Staff Health Physicist	B.S. Degree in scientific or engineering discipline and approximately 5 years experience in the field of Health Physics and/or Environmental Monitoring.
D & D Superintendent	B.S. Degree in scientific or engineering discipline or related experience in management of decommissioning at a comparable facility.

Manager of Waste Disposal	B.S. Degree in scientific or engineering discipline or at least 5 years work experience in radioactive and other hazardous waste disposal.
Site Services Manager	B.S. Degree in a scientific or engineering discipline or equivalent work experience in conventional utility plant operations and/or a related engineering field.
Quality Assurance Manager	B.S. Degree in a scientific or engineering discipline or equivalent work experience in operations or engineering related to nuclear reactor operations or fuel cycle facility operations and with specific knowledge of regulatory requirements and guidelines.

APPENDIX B

NUCLEAR SAFEGUARDS COMMITTEE

The independent review and audit of decommissioning operations shall be performed by the Nuclear Safeguards Committee.

A. Composition and Qualifications

The Nuclear Safeguards Committee shall be composed of a minimum of five members. The members shall collectively provide a broad spectrum of expertise in the safety of decommissioning operations. Members and alternates shall be appointed by and report to the Level I authority. Qualified and approved alternates may serve in the absence of regular members. The Nuclear Safeguards Committee shall be composed as follows:

- A senior technically qualified person in a position of responsibility for the operation of the facility. (Level I or II.);
- An engineer or physicist knowledgeable in decommissioning. (Level II or III in Management);
- A Health Physicist. (Level II or III in Management);
- A technically qualified person knowledgeable in radioactive and other hazardous waste disposal. (Level II or III in Management);
- A senior technical person not associated with facility decommissioning but knowledgeable in the field of decommissioning. (This person should have an advanced degree in Science or Engineering or equivalent work experience. The work experience shall be at least 5 years in decommissioning activities).

B. Charter and Rules

The committee shall function under the following operating rules:

- Meetings shall be held not less than semiannually or more frequently as circumstances warrant, consistent with effective monitoring of decommissioning activities;
- A quorum shall consist of not less than one-half the membership;
- Subgroups may be appointed to review specific items;

- Minutes shall be kept, and shall be disseminated to members and to the Plant Manager within 1 month after the meeting.
- The committee shall appoint one or more qualified individuals to perform the audits.

C. Review Function

The following items shall be reviewed by the Nuclear Safeguards Committee, or a subgroup thereof:

- Determinations that proposed decommissioning operations do not involve an unreviewed safety question;
- All new decommissioning procedures and major revisions thereto having safety significance;
- All new systems and equipment or changes thereto relating to safety;
- Proposed changes in technical specifications, license, or charter;
- Violations of technical specifications, license, or charter; violations of internal procedures or instructions having safety significance;
- Audit reports;
- Reportable occurrences listed in Section 7.5.

D. Audit Function

An audit of the Health Physics, Environmental Monitoring and General Safety Programs shall be conducted annually.

The audit function shall include selective (but comprehensive) examination of decommissioning records, logs, and other documents. Where necessary, discussions with responsible personnel shall take place. In no case shall the individual or individuals conducting the audit be immediately responsible for the area being audited.

Deficiencies shall immediately be reported to the Level II authority. A written report of the findings of the audit shall be submitted to the Level I authority and to the Nuclear Safeguards Committee members within 90 days after the audit has been completed.

APPENDIX C

Radiation Safety/ALARA Committee

The ALARA process is implemented by the Radiation Safety/ALARA Committee. The Committee is composed of Level I and Level II managers and the Staff Health Physicist, who is the Chairman. The Committee meets at least six times per year. ALARA recommendations are discussed by Committee members and delegated for implementation. The goals of ALARA are accomplished through:

- Plant policy that requires that all radiation exposures will be kept as low as is reasonably achievable;
- All employees who routinely work in a radiation controlled area must successfully complete a formal radiation safety training course;
- All major work in a controlled area must be reviewed and approved by Health Physics;
- New Health Physics or Safety procedures having safety significance or significant changes to existing procedures must be reviewed by the Nuclear Safeguards Committee for conformance with ALARA concepts. Appropriate recommendations will be made to assure compliance;
- Entrance into a high radiation area requires Health Physics approval;
- All radiation exposures in excess of action levels shall be reported to management;
- All jobs with projected radiation exposures exceeding 100 mRem or airborne exposures exceeding 10 MPC-hours to an employee shall require approval from senior Health Physics staff;
- At the conclusion of major jobs requiring senior Health Physics approval the job shall be reviewed by Health Physics to identify areas requiring improvement;
- Use of whole body and extremity exposure control procedures.

The Radiation Safety/ALARA Committee functions under the following operating rules:

- Meetings shall be held not less than every two months;
- Each major operating group shall be represented at meetings;
- Sub-groups may be appointed to attend to specific items;

- Minutes of proceedings shall be published;
- Monthly safety audits shall be conducted by persons designated by the Committee.

The audit function of the Radiation Safety/ALARA Committee includes selective (but comprehensive) examination of operating records, logs, and other documents. Where necessary, discussions with responsible personnel takes place. An individual conducting the audit shall not be immediately responsible for the area being audited.

A written report of the findings of the audit is submitted to Level I management and the Radiation Safety/ALARA Committee members within 90 days after the audit has been completed. Deficiencies that affect safety are reported promptly to the Level I management.

APPENDIX D

FINANCIAL ASSURANCE PLAN

Hoffmann-La Roche

Hoffmann-La Roche Inc
340 Kingsland Street
Nutley, New Jersey 07110-1199

Irwin Lerner
President and Chief Executive Officer
(201) 235-2011

October 15, 1990

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

I am the President and Chief Executive Officer of Hoffmann-La Roche Inc., which maintains its principal place of business at 340 Kingsland Street, Nutley New Jersey 07110, a New Jersey Corporation. This letter is in support of this Company's use of the financial test as set forth in Paragraph A.1 of Section II of Appendix A to 10 CFR Part 30, as referenced by 10 CFR Parts 50 and 70, to demonstrate financial assurance, as specified in 10 CFR Parts 50 and 70 on behalf of Cintichem, Inc., our subsidiary.

I hereby certify that Hoffmann-La Roche Inc. is currently a going concern, and that it possesses positive tangible net worth in the amount of $ * .

This Company is not required to file a Form 10K with the U. S. Securities and Exchange Commission for the latest fiscal year. The fiscal year of this Company ends on December 31.

I hereby certify that the content of this letter is true and correct to the best of my knowledge.

Very truly yours,

Irwin Lerner

Irwin Lerner
President and
Chief Executive Officer

1012c

*WITHHELD AND EXEMPTED FROM
PUBLIC DISCLOSURE PURSUANT
TO 10 C.F.R. SECTION 2.790

Hoffmann-La Roche

Hoffmann-La Roche Inc
340 Kingsland Street
Nutley, New Jersey 07110-1199

Martin F. Stadler
Senior Vice President
Finance, Human Resources
and Administration
(201) 235-2022

October 15, 1990

U.S. Nuclear Regulatory Commission
Washington D.C. 20555

I am the Senior Vice President-Finance, Human Resources and Administration of Hoffmann-La Roche Inc., which maintains its principal place of business at 340 Kingsland Street, Nutley, NJ 07110, a New Jersey Corporation. This letter is in support of this Company's use of the financial test set forth in Paragraph A.1 of Section II of Appendix A to 10 CFR Part 30, as referenced by 10 CFR Parts 50 and 70, to demonstrate financial assurance, as specified in 10 CFR Parts 50 and 70 on behalf of Cintichem, Inc., our subsidiary.

This Company guarantees, through the Parent Company Guarantee submitted to demonstrate compliance under 10 CFR Parts 50 and 70, the decommissioning of the following facility owned or operated by Cintichem Inc., a subsidiary of this Company. The current cost estimates or certified amounts for decommissioning, so guaranteed, are shown for each facility:

U.S. NUCLEAR REGULATORY COMMISSION LICENSE NUMBER	NAME AND ADDRESS OF LICENSEE	ADDRESS OF LICENSED ACTIVITY	COST OF ESTIMATES FOR REGULATORY ASSURANCES DEMONSTRATED BY THIS AGREEMENT
R-81 SNM-639	Cintichem Inc. Long Meadow Road Tuxedo, NY 10987	Cintichem Inc. Long Meadow Road Tuxedo, NY 10987	$20,482,000

The Company is not required to file a Form 10K with the U.S. Securities and Exchange Commission for the latest fiscal year.

This fiscal year of this Company ends on December 31. The figures for the following items marked with an asterisk on Appendix 1, "Financial Test:

Alternative A.1," attached hereto, are derived from this Company's independently audited, year-end financial statements and footnotes for the latest completed fiscal year, ended December 31, 1989.

I hereby certify that the content of this letter and Appendix 1 hereto are true and correct to the best of my knowledge.

Very truly yours,

Martin F. Stadler
Senior Vice President

Appendix 1

Hoffmann-La Roche Inc.

Financial Test: Alternative A.1 ($ Thousands)

1.	Decommissioning cost estimates for facility (R-81 SNM-639) total of all cost estimates shown in paragraphs above	$ **
*2.	Total liabilities (if any portion of the cost estimates for decommissioning is included in total liabilities on your firm's financial statement, deduct the amount of that portion from this line and add that amount to lines 3 and 4)	$ **
*3.	Tangible net worth	$ **
*4.	Net worth	$ **
*5.	Current assets	$ **
*6.	Current liabilities	$ **
*7.	Net working capital (line 5 minus line 6)	$ **
*8.	The sum of net income plus depreciation, depletion, and amortization	$ **
*9.	Total assets in United States (required only if less than 90 percent of firm's assets are located in the United States)	$ **

		Yes	No
10.	Is line 3 at least $10 million?	**	**
11.	Is line 3 at least 6 times line 1?	**	**
	Is line 7 at least 6 times line 1?	**	**
	Are at least 90 percent of firm's assets located in the United States? If not, complete line 14.	**	**
14.	Is line 9 at least 6 times line 1?	**	**
	(Guarantor must meet two of the following three ratios)		
15.	Is line 2 divided by line 4 less that 2.0?	**	**
16.	Is line 8 divided by line 2 greater than 0.1?	**	**
17.	Is line 5 divided by line 6 greater than 1.5?	**	**

---

*Denotes figures derived from financial statements.

(A.1) As set forth in Paragraph A.1 of Section II of Appendix A to 10 CFR Part 30, as referenced by 10 CFR Parts 50 and 70.

** WITHHELD AND EXEMPTED FROM PUBLIC DISCLOSURE PURSUANT TO 10 C.F.R. SECTION 2.790

KPMG Peat Marwick

Certified Public Accountants

New Jersey Headquarters
150 John F. Kennedy Parkway
Short Hills, NJ 07078

Telephone 201 467 9650
Telex 136584

Telecopier 201 467 7930

Independent Auditor's Report

The Board of Directors
Hoffmann-La Roche Inc.:

We have audited, in accordance with generally accepted auditing standards, the consolidated balance sheet of Hoffmann-La Roche Inc. and subsidiaries (the Company) as of December 31, 1989 and the related consolidated statements of earnings and retained earnings, and cash flows for the year then ended, and have issued our report thereon dated January 26, 1990.

At the Company's request, we performed the procedures enumerated below to assist the Company in complying with a request from the U.S. Nuclear Regulatory Commission (the Commission) for information regarding the Company's use of a financial test, as set forth in paragraph A.1 of Appendix A to 10 CFR Part 30, as referenced by 10 CFR Parts 50 and 70, to demonstrate its financial assurance, as outlined in Mr. Martin Stadler's October 15, 1990 letter and the attachments thereto to the Commission and guided by 10 CFR Parts 50 and 70. It is understood that this report is intended solely for the information and use of the Board of Directors and management of the Company and the Commission and should not be used for any other purpose. With respect to the schedule attached to this letter reconciling amounts contained in Mr. Stadler's October 15, 1990 letter with amounts contained in the 1989 audited consolidated financial statements of the Company as of and for the year ended December 31, 1989, we have:

- Agreed the amounts contained on the attached schedule in the column "Per Consolidated Financial Statements" to the respective amounts contained in the Company's audited consolidated financial statements as of and for the year ended December 31, 1989;
- Agreed the amounts contained on the attached schedule in the column "Per CFO's Letter" to the Company's "Financial Test: Alternative A.1" included with Mr. Stadler's October 15, 1990 letter;
- Determined that there are no reconciling items;
- Recomputed the subtotals and totals contained in Mr. Stadler's October 15, 1990 letter and Alternative A.1 attachment to the subtotals and totals contained in the attached schedule.

Because the above procedures do not constitute an audit made in accordance with generally accepted auditing standards, we express no opinion on any of the specified accounts or items referred to above. In connection with the procedures referred to above, no matters came to our attention that caused us to believe that the specified accounts or items should be adjusted. This report relates only to the accounts or items specified above and does not extend to any financial statements of Hoffmann-La Roche Inc. taken as a whole.

KPMG Peat Marwick

October 15, 1990

Schedule Reconciling Amounts Contained in Mr. Martin Stadler's October 15, 1990 Letter and Alternative I Attachment With Amounts in Audited 1989 Consolidated Financial Statements

WITHHELD AND EXEMPTED FROM PUBLIC DISCLOSURE PURSUANT TO 10 C.F.R. SECTION 2.790

PARENT COMPANY GUARANTEE

Guarantee made this 15th day of October, 1990 by Hoffmann-La Roche Inc., a corporation organized under the laws of the State of New Jersey, with a principal place of business at 340 Kingsland Street, Nutley, New Jersey 07110, (herein referred to as "Guarantor"), to the U.S. Nuclear Regulatory Commission, an agency of the United States Government, with a principal office at 1717 H Street, N.W., Washington, D.C. 20555 (herein referred to as "NRC") on behalf of Guarantor's subsidiary Cintichem, Inc., a corporation organized under the laws of the State of Delaware, with its principal place of business at Long Meadow Road, Tuxedo, New York 10987 (herein referred to as "Licensee").

Recitals

1. The Guarantor has full authority and capacity to enter into this guarantee under its bylaws, articles of incorporation, and the laws of the State of New Jersey, its State of incorporation. Guarantor has approval from the Executive Committee of its Board of Directors to enter into this guarantee.

2. This guarantee is being issued to comply with regulations issued by the NRC, an agency of the United States Government, pursuant to the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974. NRC has promulgated regulations in Title 10,

Chapter I of the Code of Federal Regulations, Parts 50 and 70, which require that a holder of, or an applicant for, an operating license issued pursuant to 10 CFR Part 50 and a special nuclear material license issued pursuant to 10 CFR Part 70 provide assurance that funds will be available when needed for required decommissioning activities.

3. The guarantee is issued to provide financial assurance for decommissioning activities for NRC Reactor Operator's License No. R-81 and Special Nuclear Materials License No. SNM-639 issued in the name of Cintichem, Inc. for its facility located at Long Meadow Road, Tuxedo, New York 10987, as required by 10 CFR Parts 50 and 70. The decommissioning costs for which are as follows: Twenty Million Four Hundred Eighty Two Thousand Dollars ($20,482,000).

4. The Guarantor meets or exceeds the financial test criteria set forth in Paragraph A.1 of Section II of Appendix A to 10 CFR Part 30, as referenced by 10 CFR Parts 50 and 70, in that Guarantor (i) has net working capital and tangible net worth each at least six times the current decommissioning cost estimate; and (ii) has assets located in the United States amounting to at least 90 percent of its total assets or at least six times the amount of the current decommissioning

cost estimates; and (iii) meets two of the following three ratios: a ratio of total liabilities to net worth less than 2.0; a ratio of the sum of net income plus depreciation, depletion, and amortization to total liabilities that is greater than 0.1; and a ratio of current assets to current liabilities that is greater than 1.5; and (iv) has tangible net worth of at least $10 million and agrees to comply with all notification requirements as specified in 10 CRF Parts 30, 50 and 70.

5. The Guarantor has majority control of the voting stock for the following Licensee covered by this guarantee.

U.S. NUCLEAR REGULATORY COMMISSION LICENSE NUMBERS	NAME AND ADDRESS OF LICENSEE	ADDRESS OF LICENSED FACILITY
R-81 SNM-639	Cintichem, Inc. Long Meadow Road Tuxedo, NY 10987	Cintichem, Inc. Long Meadow Road Tuxedo, NY 10987

6. Decommissioning activities as used below refers to the activities required by 10 CFR Parts 50 and 70 for decommissioning of facility identified above.

7. For value received from Cintichem and pursuant to the authority conferred upon the Guarantor by the unanimous resolution of the Executive Committee of its Board of Directors, a certified copy of which is attached, the

Guarantor guarantees to the NRC that if the Licensee fails to perform the required decommissioning activities, as required by Reactor Operator's License No. R-81 and Special Nuclear Material License No. SNM-639, the Guarantor shall:

(a) carry out the required activities, or

(b) set up a trust fund in favor of the NRC, as beneficiary, in the amount of the current cost estimate for these activities.

8. The Guarantor agrees to submit revised financial statements, financial test data, and a special auditor's report and reconciling schedule annually within 90 days of the close of the parent Guarantor's fiscal year.

9. The Guarantor agrees that if, at the end of any fiscal year before termination of this guarantee, it fails to meet the financial test criteria, the Licensee shall send within 90 days of the end of the fiscal year, by certified mail, notice to the NRC that the Licensee intends to provide alternative financial assurance as specified in 10 CFR Parts 50 and 70. Within 120 days after the end of the fiscal year, the Guarantor shall establish such financial assurance if the Licensee has not done so.

10. The Guarantor also agrees to notify the beneficiary promptly if the ownership of the Licensee or the parent firm is transferred and to maintain this guarantee until the new parent firm or the Licensee provides alternative financial assurance acceptable to the beneficiary.

11. The Guarantor agrees that within 30 days after it determines that it no longer meets the financial test criteria or it is disallowed from continuing as a Guarantor for the facility under Reactor Operator's License No. R-81 and Special Nuclear Material License No. SNM-639, it shall establish an alternative financial assurance as specified in 10 CFR Parts 50 and 70 as applicable, in the name of Licensee unless Licensee has done so.

12. The Guarantor as well as its successors and assigns agree to remain bound jointly and severally under this guarantee notwithstanding any or all of the following: amendment or modification of Reactor Operator's License No. R-81 and Special Nuclear Material License No. SNM-639 issued to Licensee or NRC-approved decommissioning funding plan for Licensee's facility, the extension or reduction of the time of performance of required activities, or any other modification or alteration of an obligation of the Licensee pursuant to 10 CFR Parts 50 or 70.

13. The Guarantor agrees that all bound parties shall be jointly and severally liable for all litigation costs incurred by the beneficiary, NRC, in any successful effort to enforce the agreement against the Guarantor.

14. The Guarantor agrees to remain bound under this guarantee for as long as Licensee must comply with the applicable financial assurance requirements of 10 CFR Parts 50 and 70, for the previously listed facility, except that the Guarantor may cancel this guarantee by sending notice by certified mail to the NRC and to Licensee, such cancellation to become effective no earlier than 120 days after receipt of such notice by both the NRC Licensee as evidenced by the return receipts.

15. The Guarantor agrees that if Licensee fails to provide alternative financial assurance as specified in 10 CFR Parts 50 and 70, as applicable, and obtain written approval of such assurance from the NRC within 90 days after a notice of cancellation by the Guarantor is received by both the NRC and Licensee from the Guarantor, the Guarantor shall provide such alternative financial assurance in the name of Licensee or make full payment under this guarantee.

16. The Guarantor expressly waives notice of acceptance of this guarantee by the NRC or by Licensee. The Guarantor also expressly waives notice of amendments or modification of the decommissioning requirements and of amendments or modifications of the license.

17. The Guarantor is not required to file financial reports with the U.S. Securities and Exchange Commission ("SEC"). Should Guarantor be required to file financial reports with the SEC prior to the termination of this guarantee, then it shall promptly submit them to the NRC during each year in which this guarantee is in effect and the Guarantor is required to make such filings.

I hereby certify that this guarantee is true and correct to the best of my knowledge.

Effective date: October 15, 1990

HOFFMANN-LA ROCHE INC.

By: Martin Stadler
Martin F. Stadler
Senior Vice President
Finance

Sworn to before me
this 15th day of October, 1990

Kathleen Dragos
Notary Public

2830C

KATHLEEN DRAGOS
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 12/28/92

Hoffmann-La Roche

Hoffmann-La Roche Inc.
340 Kingsland Street
Nutley, New Jersey 07110-1199

David E. Alpert
Assistant Secretary
(201) 235-2287

CERTIFICATION

I, David E. Alpert, Assistant Secretary of Hoffmann-La Roche Inc., a New Jersey corporation, hereby certify that the resolution as attached was unanimously adopted by the Executive Committee of the Board of Directors of said Corporation on October 15, 1990, which Executive Committee has the authority of the Board of Directors, pursuant to the by-laws of the Corporation, to adopt said resolution.

IN WITNESS WHEREOF, I have set my hand and the seal of the Corporation this 17th day of October 1990.

David E. Alpert

David E. Alpert

1053c

RESOLVED, that the President, any Senior Vice President, any Vice President, the Secretary or the Treasurer be, and each of them hereby is, authorized and directed to execute any and all documents providing a Parent Company Guarantee to the United States Nuclear Regulatory Commission ("NRC") in support of the Financial Assurance Plan of Cintichem, Inc., a wholly owned subsidiary of this Corporation, for the decommissioning of Cintichem's nuclear reactor and hot laboratory facilities located at its Tuxedo, New York facility pursuant to the rules and regulations of the NRC, and to execute such other agreements and instruments, and to perform all such other acts and things, for and on behalf of the Corporation as may be necessary, advisable or proper in order to provide such Parent Company Guarantee, including the filing of an application and supporting affidavit requesting that the proprietary and confidential financial information contained in the Parent Company Guarantee be withheld and exempt from public disclosure by the NRC through its Public Document Room, under the Freedom of Information Act and otherwise.

1009c

STANDBY TRUST AGREEMENT

TRUST AGREEMENT, the Agreement entered into as of October 15, 1990 by and between Cintichem, Inc., a Delaware corporation, herein referred to as the "Grantor, " and Swiss Bank Corporation, New York Branch, 10 East 50th Street, New York, New York 10022, the "Trustee."

WHEREAS, the U.S. Nuclear Regulatory Commission ("NRC"), an agency of the United States Government, pursuant to the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, has promulgated regulations in Title 10, Chapter I of the Code of Federal Regulations, Parts 50 and 70. These regulations, applicable to the Grantor, require that a holder of, or an applicant for, a Part 50 and 70 license provide assurance that funds will be available when needed for required decommissioning activities.

WHEREAS, the Grantor has elected to use a Parent Company Guarantee, to provide all of such financial assurance for the facilities identified herein; and

WHEREAS, when payment is necessary and made under the Parent Company Guarantee this standby trust shall be used for the receipt of such payment; and

WHEREAS, the Grantor, acting through its duly authorized officers, has selected the Trustee to be the trustee under this Agreement, and the Trustee is willing to act as trustee,

NOW, THEREFORE, the Grantor and the Trustee agree as follows:

Section 1. Definitions. As used in this Agreement:

(a) The term "Grantor" means the NRC licensee (Cintichem, Inc.) who enters into this Agreement and any successors or assigns of the Grantor.

(b) The term "Trustee" means the trustee who enters into this Agreement and any successor Trustee.

Section 2. Costs of Decommissioning. This Agreement pertains to the costs of decommissioning the materials and activities identified in NRC License Numbers R-81 and SNM-639 issued pursuant to 10 CFR Parts 50 and 70 as shown in Schedule A. The decommissioning costs are also intended to cover the decommissioning of facilities where byproduct material is licensed by New York, an agreement state.

Section 3. Establishment of Fund. The Grantor and the Trustee hereby agree to establish a standby trust fund (the Fund) for the benefit of the NRC when payment becomes necessary and is made pursuant to the terms of the Parent Company Guarantee. The Grantor and the Trustee intend that no third party have access to the Fund except as provided herein.

Section 4. Payments Constituting the Fund. Payments when made to the Trustee for the Fund shall consist of cash, securities, or other liquid assets (the "Property") acceptable to the Trustee. The Fund shall be established initially as consisting of Property which is acceptable to the Trustee. Such Property and any other Property subsequently transferred to the Trustee are referred to as the "Fund," together with all earnings and profits thereon, less any payments or distributions made by the Trustee pursuant to this Agreement. The Fund shall be held by the Trustee, IN TRUST, as hereinafter provided. The Trustee shall not be responsible nor shall it undertake any responsibility for the amount of, or adequacy of the Fund, nor any duty to collect from the Grantor, any payments necessary to discharge any liabilities of the Grantor established by the NRC.

Section 5. Payment for Required Activities Specified in the Plan. The Trustee shall make payments from the Fund to the Grantor upon presentation to the Trustee of the following:

a. A certificate duly executed by the Secretary of the Grantor attesting to the occurrence of the events, and in the form set forth in the attached Specimen Certificate, (attached hereto as Exhibit I) and

b. A certificate attesting to the following conditions;

(1) that decommissioning is proceeding pursuant to an NRC-approved plan.

(2) that the funds withdrawn will be expended for activities undertaken pursuant to that Plan, and

(3) that the NRC has been given 30 days' prior notice of Cintichem, Inc.'s intent to withdraw funds from the escrow fund.

No withdrawal from the fund can exceed 100 percent of the outstanding balance of the Fund or $20,482,000 Dollars, whichever is greater, unless NRC approval is attached.

In the event of the Grantor's default or inability to direct decommissioning activities, the Trustee shall make payments from the Fund as the NRC shall direct, in writing, to provide for the payment of the costs of required activities covered by this Agreement. The Trustee shall reimburse the Grantor or other persons as specified by the NRC from the Fund for expenditures for required activities in such amounts as the NRC shall direct in writing. In addition, the Trustee shall refund to the Grantor such amounts as the NRC specifies in writing. Upon refund, such funds shall no longer constitute part of the Fund as defined herein.

Section 6. Trust Management. The Trustee shall invest and reinvest the principal and income of the Fund and keep the Fund invested as a single fund, without distinction between principal and income, in accordance with general investment policies and guidelines which the Grantor may communicate in writing to the Trustee from time to time, subject, however, to the provisions of this section. In investing, reinvesting, exchanging, selling, and managing the Fund, the Trustee shall discharge its duties with respect to the Fund solely in the interest of the beneficiary and with the care, skill, prudence, and diligence under the circumstances then prevailing which persons of prudence, acting in a like capacity and familiar with such matters, would use in the conduct of an enterprise of a like character and with like aims; except that:

(a) Securities or other obligations of the Grantor, or any other owner or operator of the facilities, or any of their affiliates as defined in the Investment Company Act of 1940, as amended (15 U.S.C. 80a-2(a)), shall not be acquired or held, unless they are securities or other obligations of the Federal or a State government;

(b) The Trustee is authorized to invest the Fund in time or demand deposits of the Trustee; and

(c) For a reasonable time, not to exceed 60 days, the Trustee is authorized to hold uninvested cash, awaiting investment or distribution, without liability for the payment of interest thereon.

Section 7. Commingling and Investment. The Trustee is expressly authorized in its discretion:

(a) To transfer from time to time any or all of the assets of the fund to any common, commingled, or collective trust fund created by the Trustee in which the Fund is eligible to participate, subject to all of the provisions thereof, to be commingled with the assets of other trusts participating therein; and

(b) To purchase shares in any investment company registered under the Investment Company Act of 1940 (15 U.S.C. 80a-1 at seq.), including one that may be created, managed, underwritten, or to which investment advice is rendered, or the shares of which are sold by the Trustee. The Trustee may vote such shares in its discretion.

Section 8. Express Powers of Trustee. Without in any way limiting the powers and discretion conferred upon the Trustee by the other provisions of this Agreement or by law, the Trustee is expressly authorized and empowered:

(a) To sell, exchange, convey, transfer, or otherwise dispose of any property held by it, by public or private sale, as necessary for prudent management of the Fund;

(b) To make, execute, acknowledge, and deliver any and all documents of transfer and conveyance and any and all other instruments that may be necessary or appropriate to carry out the powers herein granted;

(c) To register any securities held in the Fund in its own name, or in the name of a nominee, and to hold any security in bearer form or in book entry, or to combine certificates representing such securities with certificates of the same issue held by the Trustee in other fiduciary capacities, to reinvest interest payments and funds from matured and redeemed instruments, to file proper forms concerning securities held in the Fund in a timely fashion with appropriate government agencies, or to deposit or arrange for the deposit of such securities in a qualified central depository even though, when so deposited, such securities may be merged and held in bulk in the name of the nominee or such depository with other securities deposited therein by another person, or to deposit or arrange for the deposit of any securities issued by the U.S. Government, or any agency or instrumentality thereof, with a Federal Reserve bank, but the books and records of the Trustee shall at all times show that all such securities are part of the Fund;

(d) To deposit any cash in the Fund in interest-bearing accounts maintained or savings certificates issued by the Trustee, in its separate corporate capacity, or in any other banking institution affiliated with the Trustee; and

(e) To compromise or otherwise adjust all claims in favor of or against the Fund.

Section 9. Taxes and Expenses. All taxes of any kind that may be assessed or levied against or in respect of the Fund and all brokerage commissions incurred by the Fund shall be paid from the Fund. All other expenses incurred by the Trustee in connection with the administration of this Trust, including fees for legal services rendered to the Trustee, the compensation of the Trustee to the extent not paid directly by the Grantor, and all other proper charges and disbursements of the Trustee shall be paid from the Fund.

Section 10. Annual Valuation. After payment has been made into this standby trust fund, the Trustee shall annually, at least 30 days before the anniversary date of receipt of payment into the standby trust fund, furnish to the Grantor and to the NRC a statement confirming the value of the Trust. Any securities in the Fund shall be valued at market value as of no more than 60 days before the anniversary date of the establishment of the Fund. The failure of the Grantor to object in writing to the Trustee within 90 days after the statement has been furnished to the Grantor and the NRC shall constitute a conclusively binding assent by the Grantor, barring the grantor from asserting any claim or liability against the Trustee with respect to the matters disclosed in the statement.

Section 11. Advice of Counsel. The Trustee may from time to time consult with counsel with respect to any question arising as to the construction of this Agreement or any action to be taken hereunder. The Trustee shall be fully protected, to the extent permitted by law, in acting on the advice of counsel.

Section 12. Trustee Compensation. The Trustee shall be entitled to reasonable compensation for its services as agreed upon in writing with the Grantor. (See Schedule B.)

Section 13. Successor Trustee. Upon 90 days notice to the NRC, the Trustee may resign; upon 90 days notice to NRC and the Trustee, the Grantor may replace the Trustee; but such resignation or replacement shall not be effective until the Grantor has appointed a successor Trustee and this successor accepts the appointment. The successor Trustee shall have the same powers and duties as those conferred upon the Trustee hereunder. Upon the successor Trustee's acceptance of the appointment, the Trustee shall assign, transfer, and pay over to the successor Trustee the funds and properties then constituting the Fund. If for any reason the Grantor cannot or does not act in the event of the resignation of the Trustee, the Trustee may apply to a court of competent jurisdiction for the appointment of a successor Trustee or for instructions. The successor Trustee shall specify the date on which it assumes administration of the trust in a writing sent to the Grantor, the NRC and the present Trustee by certified mail 10 days before such change becomes effective. Any expenses incurred by the Trustee as a result of any of the acts contemplated by this section shall be paid as provided in Section 9.

Section 14. Instructions to the Trustee. All orders, requests, and instructions by the Grantor to the Trustee shall be in writing, signed by such persons as are signatories to this agreement or such other designees as the Grantor may designate in writing. The Trustee shall be fully protected in acting without inquiry in accordance with the grantor's orders, requests, and instructions. If the NRC issues orders, requests, or instructions to the Trustee these shall be in writing, signed by the NRC or their designees, and the Trustee shall act and shall be fully protected in acting in accordance with such orders, requests, and instructions. The Trustee shall have the right to assume, in the absence of written notice to the contrary, that no event constituting a change or a termination of the authority of any person to act on behalf of the Grantor or the NRC hereunder has occurred. The Trustee shall have no duty to act in the absence of such orders, requests, and instruction from the Grantor and/or the NRC except as provided for herein.

Section 15. Amendment of Agreement. This Agreement may be amended by an instrument in writing executed by the Grantor, the Trustee and the NRC or by the Trustee and the NRC if the Grantor ceases to exist.

Section 16. Irrevocability and Termination. Subject to the right of the parties to amend this Agreement as provided in Section 15, this trust shall be irrevocable and shall continue until terminated at the written agreement of the Grantor, the Trustee, and the NRC, or by the Trustee and the NRC if the Grantor ceases to exist. Upon termination of the trust, all remaining trust property, less final trust administration expenses, shall be delivered to the Grantor or its successor.

Section 17. Immunity and Indemnification. The Trustee shall not incur personal liability of any nature in connection with any act or omission, made in good faith, in the administration of this trust, or in carrying out any directions by the Grantor or the NRC issued in accordance with this Agreement. The Trustee shall be indemnified and saved harmless by the Grantor or from the trust fund, or both, from and against any personal liability to which the Trustee may be subjected by reason of any act or conduct in its official capacity, including all expenses reasonably incurred in its defense in the event the Grantor fails to provide such defense.

Section 18. This Agreement shall be administered, construed, and enforced according to the laws of the State of New York.

Section 19. Interpretation and Severability. As used in this Agreement, words in the singular include the plural and words in the plural include the singular. The descriptive headings for each section of this Agreement shall not affect the interpretation or the legal efficacy of this Agreement. If any part of this agreement is invalid, it shall not affect the remaining provisions which will remain valid and enforceable.

IN WITNESS WHEREOF the parties have caused this Agreement to be executed by the respective officers duly authorized and the Grantor's corporate seal to be hereunto affixed and attested as of the date first written above.

ATTEST:

CINTICHEM, INC.

By: William [signature]

Title: Treasurer

By: [signature]

Title: Asst. Treasurer & Controller

[Seal]

SWISS BANK CORPORATION

By: [signature]
Reto Jenal

Title: Vice President

By: George W. Lamberton

Title: ASSISTANT VICE PRESIDENT

Exhibit I

Specimen Certificate of Events

Swiss Bank Corporation
New York Branch
10 East 50th Street
New York, New York 10022

Attention: Trust Division

Gentlemen:

In accordance with the terms of the Standby Trust Agreement with you dated October 15, 1990, I, ____________, Secretary of Cintichem, Inc. hereby certify that the following events have occurred:

1. Cintichem, Inc. is required to commence the decommissioning of its facility located at Long Meadow Road, Tuxedo, NY 10987 (hereinafter called the Decommissioning).

2. The plans and procedures for the commencement and conduct of the Decommissioning have been approved by the United States Nuclear Regulatory Commission, or its successor, on ________________ (copy of approval attached).

3. The Board of Directors of Cintichem, Inc. has adopted the attached resolution authorizing the commencement of the Decommissioning.

____
Secretary of Cintichem, Inc.

____
Date

SCHEDULE A

This Agreement demonstrates financial assurance for the following cost estimates for the following licensed activities:

U.S. NUCLEAR REGULATORY COMMISSION LICENSE NUMBER	NAME AND ADDRESS ADDRESS OF LICENSEE	ADDRESS OF LICENSED ACTIVITY	COST ESTIMATES FOR REGULATORY ASSURANCES DEMONSTRATED BY THIS AGREEMENT
R-81 SNM-639	Cintichem Inc. Long Meadow Road Tuxedo, NY 10987	Cintichem Inc. Long Meadow Road Tuxedo, NY 10987	$20,482,000*

*The Cost estimate is included as part of the "Decommissioning Plan for Cintichem, Inc. Research Reactor and Radiochemical Processing Laboratory" filed with the U.S. Nuclear Regulatory Commission on or about October 18, 1990. The Decommissioning cost estimate is also intended to cover decommissioning of facilities where by product material is licensed by New York, an agreement state.

SCHEDULE B

Trustee Fee: $250 establishment fee, payable upon execution of the Standby Trust Agreement; Grantor shall pay Swiss Bank Corporation such reasonable fees for its services as Trustee under the Trust Agreement dated as of October 15, 1990 relating to the letter of Credit as Swiss Bank Corporation shall from time to time specify.

ACKNOWLEDGEMENT

State of New Jersey)
)
County of Essex)

On this 15th day of October, 1990, before me personally came William Hennrich and John Taylor to me known, who being duly sworn, did depose and say that they have an office at 340 Kingsland Street, Nutley, New Jersey 07110, that they are, respectively, Treasurer and Assistant Treasurer and Controller of the Grantor, the corporation described in and which executed the Trust Agreement in accordance with regulations issued under the authority of the U. S. Nuclear Regulatory Commission; that he knows the seal of said corporation; that the seal affixed to such instruments is such corporate seal; that it was so affixed by order of the Board of Directors of said corporation, and that they signed their names thereto by like order.

Kathleen Dragos
(Notary Public)

KATHLEEN DRAGOS
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 12/20/92

1016c

ACKNOWLEDGEMENT

State of New York)
)
County of New York)

On this 15th day of October, 1990, before me personally came Reto Jenal and George W. Lambertson, to me known, who, being by me duly sworn, did depose and say that they have an office at 10 East 50th Street, New York, New York; that they are, respectively a Vice President and an Assistant V.P., of Swiss Bank Corporation, New York Branch, the corporation described in and which executed the above instrument; and that they signed their names thereto by order of the Board of Directors of said corporation.

Patricia A. Squasoni
(Notary Public)

PATRICIA A. SQUASONI
Notary Public, State of New York
No. 31-4914982 New York County
Term Expires January 11, 1992

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APPENDIX E

ESTIMATION OF RADIATION DOSE DUE TO ACCIDENTAL CUTTING OF ACTIVATED REACTOR COMPONENTS

Co-60 was found to be the principle contributor to the dose. This is shown by multiplying the exponents of worst case activation of the core support towe: by the lung inhalation dose factors.

Isotope	Activation (Ci/gm)	Dose Factor (mRem/pCi) Adult	Dose Factor (mRem/pCi) Child	Exponent Adult	Exponent Child
H-3	2.196×10^{-9}	1.58×10^{-7}	3.04×10^{-7}	-16	-16
C-14	4.22×10^{-18}	4.26×10^{-7}	1.82×10^{-6}	-25	-24
Fe-55	5.28×10^{-3}	9.01×10^{-6}	3.00×10^{-5}	- 9	- 8
Co-60	7.26×10^{-2}	7.46×10^{-4}	1.91×10^{-3}	- 6	- 5
Ni-59	8.99×10^{-7}	1.2×10^{-5}	3.59×10^{-6}	-12	-13
Ni-63	2.53×10^{-4}	2.2×10^{-5}	7.43×10^{-5}	- 9	- 9
Zn-65	3.46×10^{-3}	1.08×10^{-4}	2.69×10^{-4}	- 7	- 7

This analysis assumes:

- The most activated section (as determined by TLG's activation analysis) of the most activated component (core support tower) is cut and vaporized by plasma torch. It is planned to cut this section by hydraulic shearing or other mechanical means;
- The 2" x 2" x 1/4" angled aluminum is cut completely through;
- The cut is by plasma torch at its' maximum width, 0.25";
- The kerf material is 100% vaporized;
- The cut is done under water. No activity is assumed to be trapped in the water;
- There is no HEPA filtered containment tent around the pool;
- The HEPA filters in the building exhaust are only 99% efficient;
- It takes 1 minute to evacuate the building;
- The most conservative wind type is used (Pasquill-Gifford Type F, no vertical dispersion, wind speed 1/2 of site average = 1 m/s);

- The activity is dispersed in only one-half of the building. This results in a higher airborne concentration;
- The worker is not wearing a respirator;
- The "effective dose equivalent" weighting factor for stochastic effects is 0.12 for the lung (Ref. ICRP Publication 26, 1977, p. 21).

The airborne activity is assumed to only occupy only one-half of the reactor building (to give a higher concentration).

$$1/2 \text{ building 1 volume} = \frac{285{,}000 \text{ ft}^3}{2} = 142{,}500 \text{ ft}^3$$

Building 1 exhaust rate = 20,000 ft^3/min

$$\frac{142{,}500 \text{ ft}^3}{20{,}000 \text{ ft}^3/\text{min}} = 7.125 \text{ minutes} = 428 \text{ seconds}$$

This is how long it will take to exhaust the activity to the child.

Next, the grams of aluminum vaporized are calculated.

Angle is 2" x 2" x 1/4"

Assume maximum width of plasma torch (3.1 -> 6.4 x 10^{-3}m from NUREG/CR-1756 p.N-12) $\underline{6.4 \times 10^{-3}\text{m}} = \underline{1/4"}$

Volume of Aluminum Vaporized

$$\underset{\text{length}}{(2 \text{ In} + 2 \text{ In})}\ \underset{\text{width}}{(0.25 \text{ In})}\ \underset{\text{depth}}{(0.25 \text{ In})} = \underline{0.25 \text{ In}^3}$$

$$0.25 \text{ In}^3 \times \frac{(2.54\text{cm})^3}{\text{In}^3} = \underline{4.1 \text{ cm}^3}$$

Grams of Aluminum Vaporized

$$4.1 \text{ cm}^3 \times \frac{2.7 \text{ g}}{\text{cm}^3} = \underline{11.1 \text{ g}}$$

CASE #1 - Child at the Nearest Residential Development

The basic equation used to estimate dispersion in an airborne plume as it is blown downwind from a stack is the Gaussian plume equation of Pasquill (1961) as modified by Gifford (1961)[1].

where: $$X = \frac{Q}{2\pi\sigma_y\sigma_z u} e^{-1/2(\frac{y}{\sigma_y})^2}\left(e^{-1/2(\frac{z-H}{\sigma_z})^2} + e^{-1/2(\frac{z+H}{\sigma_z})^2}\right)$$

X = concentration in air at x meters downwind, y meters crosswind, and z meters above ground (Ci/m^3),

Q = uniform emission rate from the stack (Ci/sec),

u = mean wind speed (m/sec),

σ_y = horizontal dispersion coefficient (m),

σ_z = vertical dispersion coefficient (m),

H = effective stack height (physical stack height, h, plus the plume rise),

y = crosswind distance (m),

z = vertical distance (m).

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1 Gifford, F.A., Jr. 1961. Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion. Nucl. Saf. 2(4): 14-15 as quoted in RSIC Computer Code Collection-357 AIRDOS-EPA Estimation of Radiation Doses Caused by Airborne Radionuclides in Areas Surrounding Nuclear Facilities ORNL-5532 UC-41,11. June 1979.

Setting y = 0, H = 0, and z = 0, (worst case), one obtains the relationship used for the analysis.

$$X = \frac{Q}{\pi \sigma_y \sigma_z u}$$

or, $\frac{X}{Q} = \frac{1}{\pi \sigma_y \sigma_z u}$ as a general case

Using the most conservative wind type (Class F),

σ_y = 18m (at 457 m; see Slade, D.H. ed., Meteorology and Atomic Energy, July 1968, pp. 102 - 103).

σ_z = 7.8 m (at 457 m; see Slade, D.H. ed., Meteorology and Atomic Energy, July 1968, pp. 102 - 103).

u = 1 m/sec

$$\frac{X}{Q} = \frac{1}{(3.14)(18m)(7.8m)(1 \text{ m/sec})}$$

$$\frac{X}{Q} = 0.00227 \text{ sec/m}^3$$

$$X = \frac{X}{Q} (Q)$$

X = (0.00227 sec/m^3)Q = airborne concentration at Laurel Ridge.

Q = curies/sec in the ventilation duct (before the filter) is calculated for Co-60 and Zn-65 below

$$Q(Co\text{-}60) = \frac{(7.26 \times 10^{-2}\,Ci/gm)(11.1gm)}{427.5\ secs} = 0.001885\ Ci/sec$$

$$Q(Zn\text{-}65) = \frac{(3.46 \times 10^{-3}\,Ci/gm)(11.1gm)}{427.5\ secs} = 0.0000898\ Ci/sec$$

$$X(Co\text{-}60) = (0.001885\ Ci/sec)(0.00227\ sec/m^3)(10^{12}\ pCi/Ci) = 4.28 \times 10^6\ pCi/m^3$$

$$X(Zn\text{-}65) = (0.0000898\ Ci/sec)(0.00227\ sec/m^3)(10^{12}\ pCi/Ci) = 2.04 \times 10^5\ pCi/m^3$$

The radiation dose equivalent via inhalation is calculated using:

$$R = (X\ B\ T\ D)\ 0.01$$

where:

R = radiation dose equivalent to the lung, mrem

X = airborne concentration, pCi/m^3

B = ventilation rate of exposed individual

T = time of exposure to the airborne radionuclide concentration, sec

D = radiation dose equivalent factor, mrem/pCi.

0.01 = assumes 1% of the activity passes through the HEPA filter.

The inhalation dose model presented in the above equation is consistent with the ICRP Task Group Lung Model.

where:

B = $3.3 \times 10^{-4}\ m^3/sec$

T = 428 sec

D for a child is obtained from USNRC Regulatory Guide 1.109 for each isotope (see the first page of this appendix).

$R(Co\text{-}60) = (4.28 \times 10^6\ pCi/m^3)(3.3 \times 10^{-4} m^3/sec)(428\ sec)(1.91 \times 10^{-3} mRem/pCi)(0.01) = 11.5\ mRem = 0.0115\ Rem$

$R(Zn\text{-}65) = (2.04 \times 10^5\ pCi/m^3)(3.3 \times 10^{-4} m^3/sec)(428\ sec)(2.69 \times 10^{-4} mRem/pCi)(0.01) = 0.08\ mRem = 0.00008\ Rem$

Total Estimated Lung Dose = 0.012 Rem

Since the "effective dose equivalent" weighting factor is 0.12 for the lung, the estimated equivalent total body dose is:

$$0.012\ Rem \times 0.12 = \underline{0.0014\ Rem}$$

This value is 0.3 percent of the applicable limit of 0.5 Rem.

CASE #2 - A Worker in the Reactor Building

It is assumed that the air the worker is breathing has not been filtered by a portable HEPA filter unit. The airborne activity in the building is:

$$\frac{(7.26 \times 10^{-2}\ Ci/gm)(11.1\ gm)(10^{12}\ pCi/Ci)}{(142{,}500\ ft^3)(0.3048\ m/ft)^3} = 2.00 \times 10^8\ pCi/m^3 \text{ from Co-60}$$

$$\frac{(3.46 \times 10^{-3}\ Ci/gm)(11.1\ gm)(10^{12}\ pCi/Ci)}{(142{,}500\ ft^3)(0.3048\ m/ft)^3} = 9.52 \times 10^6\ pCi/m^3 \text{ from Zn-65}$$

The dose to the worker is:

$(2.00 \times 10^{8}$ pCi/m^3)(3.3 x 10^{-4} m^3/sec)(60 sec)(7.46 x 10^{-4} mRem/pCi)
= 2950 mRem = 2.95 Rem lung dose from Co-60

$(9.52 \times 10^{6}$ pCi/m^3)(3.3 x 10^{-4} m^3/sec)(60 sec)(1.08 x 10^{-4} mRem/pCi)
= 20.4 mRem = 0.020 Rem lung dose from Zn-65

Total estimated lung dose to the worker is 2.97 Rem.

Since the "effective dose equivalent" weighting factor is 0.12 for the lung, the equivalent total body dose estimate is <u>0.36 Rem</u>.

2.97 Rem x 0.12 = 0.36 Rem

This value represents 7.2 percent of the annual limit of 5.0 Rem.

APPENDIX F

DOSE ANALYSIS FOR RESUSPENSION OF CONCRETE DUST REMOVED FROM HOT CELL WALLS AND FLOORS

Case 1

Estimated dose to a worker operating a fork lift while transporting a 7.5 cubic foot drum (55 gallon size) of concrete dust removed from the hot cells from which 10% of the contents are released due to breakage of the container.

Assumptions for analysis

- Surface of hot cell walls retain an average of 1 uCi/cm^2 each of the isotopes Cs-137 and Sr-90.

  This is derived from survey data of Cell #1 where a 55 mrem/hr gamma dose rate in the central cell region due to surface contamination of walls, ceiling and floor was observed. A computer model and manual calculation check verifies equivalence of a 1 uCi/cm^2 Cs-137 level evenly distributed on all cell #1 internal surfaces with the 55 mrem/hr dose rate measured. Of the long-lived fission products, Cs-137 and Sr-90 are the isotopes of concern. The assumption of an equal concentration of Sr-90 is independent of the gamma dose rate measurement since Sr-90 is not a gamma emitting isotope.

- Concrete dust volume is 40% greater than that of the original intact concrete.

- Worker inhales 10^{-6} of the radioactivity released.[1]

- Depth of scabbling is 0.25 inches = 0.64 centimeters.

- 0.75 ft^3 of concrete dust is released.

- Inhalation dose factor for an adult is 1.24×10^{-2} mrem/pCi Sr-90 inhaled for bone (Reg. Guide 1.109 Table E-7).

- Inhalation dose factor for an adult is 5.35×10^{-5} mrem/pCi Cs-137 inhaled for total body (Reg. Guide 1.109 Table E-7).

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[1] Brodsky, Alan, Resuspension Factors and Probabilities of Intake of Material in Process (or "Is 10^{-6} a Magic Number in Health Physics)" Health Physics Journal, December 1980, pp. 992-1000.

Calculation

$$1\ \mu Ci/cm^2\ \text{Sr-90} \times \frac{1}{0.64\ cm} \times \frac{1}{1.4} \times \frac{2.83 \times 10^4\ cm^3}{ft^3} \times 0.75\ ft^3 \times (10^{-6})$$

$$= 2.4 \times 10^{-2}\ \mu Ci\ \text{Sr-90 inhaled}$$

then,

$$2.4 \times 10^{-2}\ \mu Ci\ \text{Sr-90} \times \frac{10^6\ pCi}{1\ \mu Ci} \times \frac{1.24 \times 10^{-2}\ \text{mrem(bone)}}{pCi\ \text{inhaled}}$$

$$= 3.0 \times 10^2\ \text{mrem to bone from Sr-90}$$

The "effective dose equivalent" weighting factor for stochastic effects is 0.03 for bone surfaces (Ref. ICRP Publication 26, 1977, p. 21). Therefore, the equivalent total body dose is:

$$3.0 \times 10^2\ \text{mrem} \times 0.03 = 9.0\ \text{mrem} = 0.0090\ \text{rem}$$

and likewise, for Cs-137,

$$2.4 \times 10^{-2}\ \mu Ci\ \text{Cs-137} \times \frac{10^6\ pCi}{1\ \mu Ci} \times \frac{5.35 \times 10^{-5}\ \text{mrem (total body)}}{pCi\ \text{inhaled}}$$

$$= 1.3\ \text{mrem to total body from Cs-137} = 0.0013\ \text{rem}$$

The "total body effective dose equivalent" is estimated to be 0.010 rem:

$$0.0090\ \text{Rem} + .0013\ \text{Rem} = 0.010\ \text{Rem}$$

This value is 0.21 percent of the 5 Rem total body dose limit permitted for occupational workers.

Case II

Estimated dose to a teenager (the limiting case) at the property line due to a rupture of a 7.5 cubic foot drum (55 gallon size) of concrete dust removed from the hot cells from which 10% of the contents are released due to breakage of the container.

Assumptions for Analysis

- o As in Case 1 above, the radioactivity concentration is 1.1 uCi Sr-90/cm^3 and 1.1 uCi Cs-137/cm^3 in the concrete dust;
- o 0.75 ft^3 of concrete dust is released;
- o Teenager is located directly downwind at a distance of 600 feet (183 m) from the ruptured container at the site boundary;
- o Stability class is Pasquill-Gifford F;
- o Wind velocity = 1 meter/sec = u;
- o The release of radioactivity takes place over a period of 10 seconds and the teenager remains in the plume of the airborne radioactivity for the entire duration (also, 10 seconds);
- o The horizontal dispersion coefficient (σ_y) at 600 feet (183 m) is 7.1 m. (Slade, D. H. ed., Meteorology and Atomic Energy, July 1968, pp. 102 - 103);
- o The vertical dispersion coefficient (σ_z) at 600 feet (183 m) is 3.6 m. (Slade, D. H. ed., Meteorology and Atomic Energy, July 1968, pp. 102 - 103);
- o X is concentrations in air at 183 m downwind (Ci/m^3 or uCi/cm^3);
- o Q is uniform emission rate (Ci/sec);
- o 10% of radioactivity released is contained within respirable sized particles;
- o Breathing rate for teenager is 8000 m^3/yr (Reg. Guide 1.109 Table E-5);
- o Inhalation dose factor for a teenager is 1.35×10^{-2} mrem/pCi Sr-90 inhaled for bone (Reg. Guide 1.109 Table E-8);

- o Inhalation dose factor for a teenager is 3.89×10^{-5} mrem/pCi Cs-137 inhaled for total body (Reg. Guide 1.109 Table E-8).

Calculations

$$0.75\ ft^3 \times 2.83 \times 10^4\ cm^3/ft^3 \times \frac{1.1\ uCi\ Sr\text{-}90}{cm^3} \times 0.1 \times \frac{1}{10\ sec}$$

$$\times \frac{1\ Ci}{10^6\ uCi} = \frac{2.3 \times 10^{-4} Ci}{Sec}\ Sr\text{-}90$$

This is the release rate of "respirable" Sr-90.

The basic equation used to estimate dispersion in an airborne plume as it is blown downwind from a stack is the Gaussian plume equation of Pasquill (1961) as modified by Gifford (1961)[2].

$$X = \frac{Q}{2\pi\sigma_y\sigma_z u} e^{-1/2(\frac{y}{\sigma y})^2} \left(e^{-1/2(\frac{z-H}{\sigma_z})^2} + e^{-1/2(\frac{z+H}{\sigma_z})^2} \right)$$

Where y and z are the respective crosswind and vertical distances that the plume travels. H is the effective stack height (physical stack height plus the plume rise). Setting y = 0, H = 0, and z = 0, (worst case), one obtains the relationship used for the analysis.

$$X = \frac{Q}{\pi\sigma_y\sigma_z u} = \frac{2.3 \times 10^{-4}\ Ci/Sec}{(3.14)(7.1m)(3.6m)(1)} = 2.9 \times 10^{-6} Ci/m^3$$

(or uCi/cm3)

also,

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2 Gifford, F.A., Jr. 1961. Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion. Nucl. Saf. 2(4): 14-15 as quoted in RSIC Computer Code Collection-357 AIRDOS-EPA Estimation of Radiation Doses Caused by Airborne Radionuclides in Areas Surrounding Nuclear Facilities ORNL-5532 UC-41,11. June 1979.

$8000\ m^3/yr \times 1\ yr/3.16 \times 10^7\ sec \times 10^6\ cm^3/m^3 = 2.5 \times 10^2\ cm^3/sec$

which is the teenage breathing rate.

then,

$2.9 \times 10^{-6}\ uCi/cm^3 \times 2.5 \times 10^2\ cm^3/sec \times 10\ sec \times 10^6\ pCi/uCi \times$

$1.35 \times 10^{-2}\ mrem/pCi = 98\ mrem$ to bone

The "effective dose equivalent" weighting factor for stochastic effects is 0.03 for bone surfaces (Ref. ICRP Publication 26, 1977, p.21). Therefore, the equivalent total body dose is:

$98\ mrem \times 0.03 = 2.9\ mrem = 0.0029\ Rem$

Total body dose from Cs-137

$2.9 \times 10^{-6}\ uCi/cm^3 \times 2.5 \times 10^2\ cm^3/sec \times 10\ sec \times 10^6\ pCi/uCi \times 3.89 \times 10^{-5}\ mrem/pCi = 2.8 \times 10^{-1}$ mrem total body = 0.00028 Rem total body

The "total body effective dose equivalent" is estimated to be 0.003 Rem:

$0.0029\ Rem + 0.00028\ Rem = 0.0032\ Rem$

This accident value is 0.6 percent of the 0.5 Rem total body dose limit permitted for unrestricted areas.

APPENDIX G

Drinking Water Dose Analysis for Rain Water Runoff from Building 1 and 2 Excavation Site into Indian Kill Reservoir

Assumptions for Analysis

- o All potentially contaminated soil and its total radioactivity presumed remaining in buildings 1 and 2 after the buildings are razed is assumed to be washed into the Indian Kill Reservoir as a consequence of a severe rainstorm;

- o The volume of the Indian Kill Reservoir is 3.28×10^7 ft^3 or 245 million gallons. (Ref. G. Faller, Sterling Forest Water Company);

- o The radioactivity is conservatively assumed to be completely soluble and is diluted to only one-half the volume of the reservoir before it is collected as drinking water;

- o The soil contamination, based on Cintichem measurements of a T-1 room core, is pessimistically assumed to extend around the perimeter of the Hold Up Tank (32' x 15'), along the entire east wall (134') of the Hot Laboratory, and entire north wall of the Hot Lab building (68') below the T-1 room;

- o The contaminated soil remaining at the time buildings are razed is assumed to exist in a 4.5 ft^2 cross section beneath the above mentioned walls. Thereby, the total volume of soil that can be washed into the Indian Kill Reservoir is 4.5 ft^3 per linear foot of wall or 1332 ft^3 of soil;

- o No radioactive decay credit is assumed between the time of core sampling and this postulated accident;

- o Soil density is 106 pounds per ft^3 for a total of 141,192 pounds or 6.41×10^7 grams of soil;

- o The isotope concentrations in pCi/gm are based on measurements of the core sample with the highest radioactivity taken in the T-1 room. Total Curies in the soil are:

Isotope	pCi/gm	Total Ci
Ce-144	596.16	3.82×10^{-2}
Cs-137	224.64	1.44×10^{-2}
Sr-90	224.64	1.44×10^{-2}
Co-60	8.64	5.54×10^{-4}
Nb-95	6.91	4.43×10^{-4}
Ru-103	6.91	4.43×10^{-4}
Sb-125	6.91	4.43×10^{-4}
Cs-134	6.91	4.43×10^{-4}
Ir-192	6.91	4.43×10^{-4}

Based on the above total activities and assuming a dilution of only half the reservoir volume, the total concentration of all isotopes in the water is 1.5×10^{-7} uCi/ml. The combined MPC for the above isotopes, from 10CFR20 table II Col. 2, is 1.32×10^{-6} uCi/cm^3.

Therefore, this concentration is calculated to be 11% MPC.

APPENDIX H

ENVIRONMENTAL REPORT

ENVIRONMENTAL REPORT

for

DECOMMISSIONING

CINTICHEM, INC.
RESEARCH REACTOR

AUGUST 1990

NRC LICENSE NO. R-81
DOCKET 50-54

1.0 EXECUTIVE SUMMARY

This Environmental Report is written to allow the NRC to assess the environmental impacts associated with the decommissioning and dismantling of the Cintichem research reactor and hot laboratory.

It is concluded that no appreciable adverse environmental affects will be associated with this project if the safety aspects as discussed in the Decommissioning Plan are followed. There will be no significant exposure (approximately zero) of the general public to radioactive effluents released during the dismantling activities.

Low level radioactive waste requiring disposal or intermediate processing at a licensed facility is estimated to be 138,786 cubic feet. Such waste will be packaged, shipped and disposed of in accordance with applicable DOT and NRC regulations. All of the reactor components and supporting systems/equipment are to be dismantled, decontaminated (as necessary), packaged and shipped to an intermediate processor or a licensed disposal site.

The collective dose equivalent to workers during the dismantlement activities and waste shipment is estimated to be 366 person-rem.

The facility will be decontaminated to the release criteria based on Regulatory Guide 1.86 and a maximum dose rate of 5 uR/hr above background at one meter from the surfaces previously activated of the remaining structures and soil, or 10 mRem/yr above background, considering reasonable proximity and occupancy. The irreversible and irretrievable commitments of resources include the financial, human, and energy resources necessary to implement the plans and procedures for decommissioning.

2.0 DESCRIPTION OF DECOMMISSIONING ACTIVITIES

2.1 Introduction

The reactor site is located in Sterling Forest 3 1/4 miles north-northwest of Tuxedo Park, Orange County, New York. The reactor is a pool type research reactor which is light-water moderated, heterogeneous, and utilizes solid fuel with water for cooling and shielding.

Reactor operation began in 1961 and terminated in 1990 after running approximately 906,000 megawatt hours.

Cintichem, Inc. plans to decommission the reactor prior to December 31, 1992 and to release to DECON specifications as delineated in the Decommissioning Plan (DP) for unrestricted use.

2.2 Activities Required for Decon

The decommissioning scope will include the following items:

REACTOR BUILDING

- Reactor core structure and components (Activated);
- Biological shield (activated concrete and steel embedments);
- Activated experimental facilities (i.e. beam tubes, pneumatic rabbits, thermal column);
- Contaminated holdup tank and surrounding soil/backfill;
- Reactor pool/stall contaminated surfaces;
- Contaminated canal/gamma pit structures and underlying soil/backfill;
- Contaminated reactor building structural surfaces;
- Contaminated reactor building systems:
  - HVAC exhaust;
  - Primary coolant;
  - Coolant purification;
  - Floor drain and pool gutter collection;
- Primary water storage tank, piping and surrounding soil;
- Storage tubes (embedded in reactor building South wall);
- Storage tubes (embedded in reactor building West wall).

HOT LABORATORY BUILDING

o Hot cells (five);

o Contaminated structural surfaces;

o Class "B" waste storage facility;

o Liquid waste evaporator;

o Buried hot cell exhaust ducting, filter room and surrounding contaminated soil;

o Hot lab HVAC exhaust ducts;

o T-1 and evaporator rooms contaminated surfaces and surrounding soil;

o Hot lab floor drain and waste collection systems.

OUTSIDE AREAS

o Two 5,000 gallon waste water holding tanks (mall tanks);

o Class A storage building surfaces;

o Exterior exhaust ducts, blowers and discharge stack;

o Miscellaneous buried piping, manholes and collection basins and surrounding soil.

Descriptions and analyses of the above tasks are covered in the DP.

3.0 ENVIRONMENTAL IMPLICATIONS OF DECONTAMINATION

3.1 Implications of Decon

A comprehensive Safety Evaluation Report (SER) for the reactor was completed in June 1984 (published as NUREG 1059). This report addressed environmental implications of routine and accident conditions related to the operation of the reactor. A similar report entitled "Environmental Impact Appraisal" addresses environmental issues related to the license renewal for special nuclear materials (May 1984). The potential for environmental impacts will be greatly reduced because the reactor fuel and all special nuclear material other than residual contamination, will have been removed and normal operations will have been terminated for approximately one year prior to the start of DECON activities.

In consideration of decommissioning as it applies to the Cintichem facility, the U.S. Nuclear Regulatory Commission concludes the following (NUREG-0586):

- Based on the nearly completed data base results and on NRC staff considerations, taking account of the concerns of the State and public, and of the regulatory role the NRC must provide in protecting the public health and safety, the following conclusions appear evident;
- The technology for decommissioning nuclear facilities is well in hand. Decommissioning at the present time can be performed safely and at a reasonable cost;
- Decommissioning of a nuclear facility generally has a positive environmental impact.

3.2 Commitment of Resources

The major impact of decommissioning is the commitment of small amounts of land at the off-site disposal facility for waste burial. DECON activities will result in the generation of low-level radioactive waste as shown in Table 3.1.

TABLE 3.1

ESTIMATED SOLID RADIOACTIVE WASTE INVENTORY AND CHARACTERISTICS

Waste Type	(a) Disposal Volume ft^3	(b) Average Concentration Ci/ft^3	(c) Total Curies	Highest 10CFR61 Classification
Activated Concrete	1,490	0.0577	86	A
Activated Components	369	9.58	3,534	C
Activated Lead	3.5(c)	(c) 29.45	103.06	B
Contaminated Concrete	28,635	0.00745	213.2	A
Contaminated Equipment and Components	37,367	0.0012	44.8	A
Contaminated Lead	149	Nil	Nil	A
Contaminated Soil	51,271	0.000197	10.1	A
Dry Active Waste	8,247	Nil	Nil	A
Solidified Evaporator Concentrates	60	3.4	204	A
TOTALS:	127,592		4,195.06	

(a) Disposal volume including packaged void and package volumes

(b) Average concentration is derived from packaged disposal volume and total estimated activity

(c) Actual volume not packaged

4.0 HEALTH AND SAFETY IMPLICATIONS

Occupational, public, and transportation safety impacts from DECON activities are summarized in this section.

4.1 Occupational Radiation Doses

Estimates of occupational radiation doses are based on the estimated radiation dose rates in various areas of the Reactor and Hot Lab Buildings and environs and on the estimated staff labor required to complete the decommissioning activities. Summaries of radiation doses are contained in Table 4.1. Also presented are estimates of worker injuries and fatalities resulting from decommissioning activities. These industrial accident estimates presented in Table 4.2 are based on nuclear industry experience.

TABLE 4.1

CINTICHEM DECOMMISSIONING PROJECT COLLECTIVE-DOSE ESTIMATE
(person-rem)

TASK #	PERSON-REM DOSE	MAN HOURS
1	3.2×10^{1}	6400
2	8.82×10^{-1}	802
3	1.11×10^{2}	2220
4	8.4×10^{0}	2800
20	4.99×10^{1}	998
11	8.63×10^{0}	863
12	8.5×10^{0}	2833
13	6.28×10^{0}	126
19	3.23×10^{0}	650
16	9.6×10^{-2}	87
17	3.58×10^{0}	1790
18	4.7×10^{0}	392
22	2.28×10^{-1}	440
24	6.61×10^{-1}	661
5	1.57×10^{1}	3140
7	2.25×10^{0}	450
21	3.41×10^{1}	3410
23	2.08×10^{1}	2080
25	4.34×10^{0}	2170
26	1.28×10^{0}	1280
27	6.38×10^{0}	3190
28	1.49×10^{0}	745
6 (canal)	2.23×10^{1}	11150
8	8.49×10^{-1}	500
9	3.2×10^{0}	3200
10	4.0×10^{-2}	40
14	4.47×10^{0}	2235
15	1.26×10^{0}	1260
30	8.02×10^{-2}	802
31	1.34×10^{0}	6700
32	3.27×10^{-3}	327
33	2.20×10^{0}	5500
36	6.2×10^{0}	6208
37	3.2×10^{-4}	32
38	6.4×10^{-4}	64
TOTAL	3.66×10^{2}	75545

TABLE 4.2

ESTIMATED OCCUPATIONAL LOST-TIME INJURIES AND FATALITIES[(a)]

Category of Effort	Frequency (Accidents/10^6 man-hours)[(b,c)] Lost-Time Injuries	Fatalities	Decommissioning Activities Manhours	Lost-Time Injuries	Fatalities
Heavy Construction[(d)]	10	4.2×10^{-2}	3.2×10^5	3.2	1.3×10^{-2}
Light Construction	5.4	3.0×10^{-2}	8.1×10^4	4.4×10^{-1}	2.4×10^{-3}
Operational Support	2.1	2.3×10^{-2}	2.4×10^5	5.0×10^{-1}	5.5×10^{-3}

(a) NUREG/CR 1756, Table 12.2-12.

(b) Estimates of man-hours, injuries, and fatalities are rounded to two significant figures.

(c) Lost-time injuries and fatality frequencies are from Operation Accidents and Radiation Exposure Experiences Within the U.S., AEC 1943-1970, WASH-1192, 1971 as quoted in Technology, Safety, and Costs of Decommissioning Reference Nuclear Research and Test Reactors, NUREG/CR-1756, Vol. 1, p. 12-23

(d) Heavy construction involves demolition tasks such as removal of piping, equipment, and concrete.

4.2 Estimated Public Radiation Doses from Airborne Effluent

The estimated consequences of atmospheric releases of radioactivity from routine decommissioning tasks are presented in Table 4.3. They are adjusted to the Tuxedo Park area population estimates. The radiation exposure pathway considered is that from inhalation of particulate radionuclides. Details of this estimate are provided in Attachment A.

The dose listed is the committed total body effective dose equivalent which is small when compared to natural background. The dose to the maximally exposed individual is 1.5×10^{-6} rem which is 5.0×10^{-4}% of the average yearly background effective dose equivalent of 0.3 rem.

The dose to the general population within a five mile radius of the site is also estimated to be very small and is included in Table 4.3 on the following page.

TABLE 4.3

ESTIMATED RADIATION DOSES FROM
ATMOSPHERIC RELEASES DUE TO SCABBLING OF HOT CELLS

	Total Body Effective Dose Equivalent (Rem)	Collective Dose (Person-Rem)
Maximally Exposed Individual	1.5×10^{-6} (b)	N/A
General Population[a]	3.7×10^{-8} (average)	3.2×10^{-4} (b)

(a) Doses are estimated for a total population of 8600 people residing within a five mile radius (1980 approximate census).
(b) See Attachment A

4.3 Estimated Public Radiation Dose from Direct External Radiation Originating on Site

The estimated radiation dose at the nearest property (Laurel Ridge) is estimated to be 0.0013 Rem per year. This is primarily due to waste shipping operations on site during decommissioning (see Attachment B). The estimated annual dose rate at the property fence line (near the front entrance to the site) is 0.020 Rem/yr for continuous occupancy (see Attachment B). An onlooker or jogger spending five minutes each day of the year at this location would receive an annual dose of 0.00012 Rem. These estimated radiation exposures are based upon containers having the maximum DOT allowable radiation dose rates. In actuality, shipping containers will have lower dose rates due to lower radioactive material concentrations that will be present and truck payload weight and/or volume limitations.

4.4 Estimated Radiation Doses from Routine Transportation

Transportation of radioactive materials results in external radiation doses to the transportation workers and to the public along the transportation route. Table 4.4, adapted from NUREG/CR-1756, gives the estimated radiation doses from routine radioactive waste shipments. These estimates are also based upon shipping containers having the maximum allowable DOT radiation dose rates. Therefore actual radiation doses will be lower.

TABLE 4.4

ESTIMATED ANNUAL RADIATION DOSES FROM ROUTINE RADIOACTIVE WASTE TRANSPORT[a]

Alternative/Group	Radiation Dose per shipment (Person-Rem)[a]	Number of Shipments[b]	Total Population Dose per Group (Person-Rem)[c]
Truck Drivers	6.7×10^{-2}	100	6.7
Garage Men	3.3×10^{-3}	100	0.3
Total Worker Dose -			7.0
Onlookers	5.0×10^{-3}	100	0.5
General Public	1.8×10^{-3}	100	0.2
Total Public Dose -			0.7

(a) Based on NUREG/CR 1756 Table 12.4-1 and one-way trips of 800 km.

(b) Based on two shipments per week.

(c) All doses are rounded to the nearest one-tenth perso rem.

5.0 DESCRIPTION AND STATUS OF FACILITY

5.1 General Locale

Cintichem, Inc. is located in Sterling Forest in Orange County, New York. The plant is constructed along Long Meadow Road on the eastern slope of Hogback Mountain at an elevation of approximately 800 feet. Sterling Forest is an area of approximately 27 square miles which has been set aside by the owner for technological development. The reactor and hot lab site, itself, consists of 100 acres of land owned by Cintichem, Inc.. The immediate environs is generally wooded, hilly, and interspersed with lakes. The Town of Tuxedo is 3 1/4 miles to the South-Southeast. The Indian Kill stream runs nearby the site and discharges into the Ramapo River approximately two miles East of the site.

5.2 Facility Description

Reactor Building

The reactor building is a rectangular reinforced-concrete structure set into an excavation in the side of a rock mountain. Shielding and containment are provided on three sides of the building by solid rock against the west wall, and a combination of rock and fill on the north and south sides. The exposed portions of the walls and roof are reinforced concrete.

The building measures about 70 feet wide, 92 feet long, and 57 feet high from the reactor building base floor. The walls have a minimum thickness of 12 inches and the roof is a minimum of 8 inches thick. The volume of the reactor building is about 285,000 cubic feet. The building is designed to withstand an internal pressure of 3/4 psig.

The experimental area around the reactor is serviced by a 10-ton bridge crane traveling the length of the building. The reactor control room, several offices, and laboratories for low activity work are provided inside of the reactor building. Junior hot cells for medium activity work and for opening sample cans are also provided. All personnel entrances to the building are of the double airlock type. Large equipment can be brought into the reactor building via a motor-operated, air-tight sliding door.

Hot Laboratory

The hot laboratory, located adjacent to the reactor building, is designed to permit disassembly, inspection, testing and analysis of highly radioactive material. The building is a concrete structure, 139 ft. by 57 ft., completely enclosing five hot cells.

Irradiated samples are transferred from the reactor building to the hot laboratory by way of the canal.

The hot laboratory is equipped with the necessary areas for charging, operating, decontaminating and disposal operations, as well as office, locker, and change rooms for employees.

5.3 Environmental Characteristics

5.3.1 Demography

The reactor is located in a sparsly populated area. It is within a 18,000 acre woodland area, called Sterling Forest, which is owned by a private company, the Ambase Corporation, a subsidiary of City Investing Corporation. Sterling Forest contains three residential areas, several small research centers and a conference center. The remainder of the land is undeveloped. Adjoining Sterling Forest to the east is another large undeveloped area which is a part of the Palisades Interstate Park System. This 75,000 acre woodland contains approximately 31 summer camps but essentially no year-round residency.

The nearest public access to the site is a secondary road 490 feet from the Reactor Building. The closest off-site occupied area is the Laurel Ridge housing development, which contains 132 houses at a minimum distance of 1100 feet from the Reactor Building. A second development, consisting of 27 houses and called Clinton Woods, is located at a distance of 3200 feet. There are no other housing developments within 1.5 miles. Development of the Sterling Forest area is proceeding very slowly and the present low population density is not expected to change significantly in the foreseeable future.

Tables 5.1 and 5.2 show the population centers and population distribution in the vicinity of the site. The population figures are for permanent residents and in Table 5.2, 3.5 persons per single family dwelling is assumed. Accurate figures on any increase in population during the summer months is not available.

TABLE 5.1

Major Population Centers Within 10-Mile Radius of Cintichem Site

Town	1980 Population Estimate	Distance from Site
Greenwood Lake, NY	2,262	2.5 miles S.W.
Tuxedo Park, NY	3,458	3.25 miles S.S.E.
Sloatsburg, NY	3,134	6.0 miles S.E.
Warwick, NY	20,851	6.0 miles W.
Ringwood, NJ	10,393	6.5 miles S.
Monroe, NY	11,809	7.5 miles N.
Ramapo, NY	83,630	7.5 miles
Hilburn, NY	1,258	8.5 miles S.E.
Florida, NY	1,674	9.0 miles N.W.
Suffern, NY	8,695	9.5 miles S.E.
Chester, NY	6,208	10.0 miles N.
Mahwah, NJ	10,800	10.5 miles S.E.

TABLE 5.2

Population Distribution About Reactor Site

Distance	1980 Population Estimate
0.5 mile	210
1 mile	560
2 miles	1,500
5 miles	8,600
10 miles	162,595

5.3.2 Climate and Meteorology

The most reliable and longest-period surface weather observations for the area have been taken for more than a century at West Point Military Academy, approximately 18 miles east-northeast of the reactor site. The Weather Bureau has cooperative observing stations at Suffern and Warwick, New York, and Greenwood Lake and Ringwood, New Jersey, while the Air Force recently operated an upper air sounding station at Stewart Air Force Base about 20 miles northeast of the reactor site. While the climatology of a hilly region is strongly influenced by local topographic features so that observations from surrounding weather stations may not be entirely representative, still we can obtain from the dense network of observing stations and from field observations a fairly good idea of the climatic picture in the vicinity of the reactor site. In addition, the basic meteorological data has been confirmed by local observations of wind speed and direction at the reactor stack elevation for a 29 year period.

The climate of the Sterling Forest area is predominantly influenced by air mass movement and prevailing winds from an inland direction. The weather moves across the area from west to east at average velocities of 30 to 35 miles per hour in winter and somewhat more slowly in summer. This is a part of the normal cyclonic circulation in which the usual weather-producing low pressure systems follow paths toward the northeastern United States. About 40 percent of the low centers pass over or are close to south-eastern New York. Most of the others come close enough to exert an influence on the area's weather so that there is a regular change in weather patterns without any consistent periods of stagnation.

Centers of high pressure alternate more or less regularly with the lows. In the winter-time, their movement is variable, depending on the strength of cold air outthrusts from arctic areas to the northwest. This movement is slowest during summer and early fall so that, with the prevailing westerlies aloft reaching their most northerly movement at the same time, high pressure centers often become stationary for several days over the area during these seasons. The result is stable atmospheric conditions which encourage the formation of temperature inversions. Such conditions can occur in any month, but persistent stability of several days' duration occurs on the average only once in several years.

Cold air masses of the continental arctic or continental polar types dominate the area's weather in the fall, winter, and spring. These are very stable at their northern source, but by the time they have reached southeastern New York, having been heated from below as they moved across the land, their lower layers are generally unstable. During the summer, the continental outbreaks of cold air are weak and maritime tropical air masses migrate northward to exert an effect on the weather of the area.

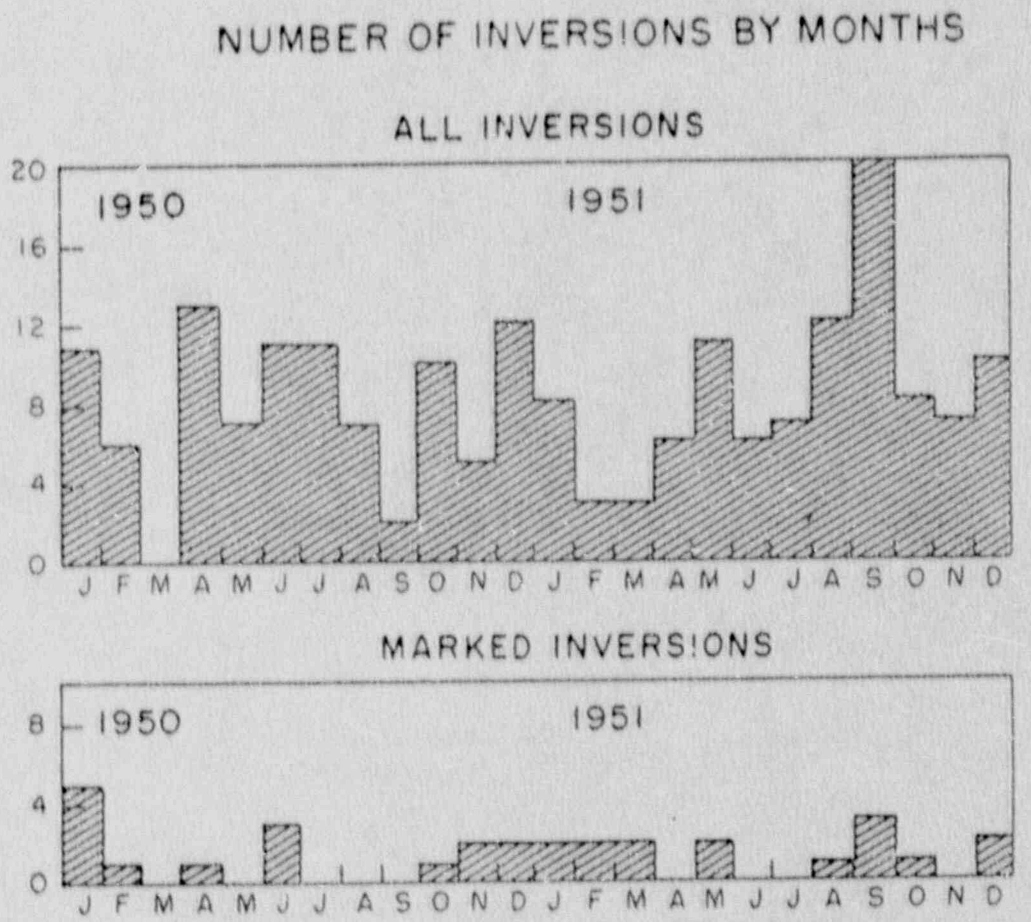
NUMBER OF INVERSIONS BY MONTHS
ALL INVERSIONS
20
16
12
8
4
0
1950
1951
J F M A M J J A S O N D J F M A M J J A S O N D
MARKED INVERSIONS
8
4
0
1950
1951
J F M A M J J A S O N D J F M A M J J A S O N D

FIGURE 5.1

At this time of year, nocturnal cooling results in frequent temperature inversions, but they are most often short-lived because of the heating which occurs during the day, resulting in turbulence and mixing of the atmosphere.

Low-level inversions, as defined for this analysis, are those which occur between the ground surface and the 950-millibar pressure level, a height difference of approximately 1700 feet. The average diurnal temperature range at the surface is consistently close to 12oC over the entire year. Under standard conditions, a temperature rise of this amount at the surface would just destroy an inversion of 8oC between the ground and 950-millibar levels. Therefore, it is reasonable to consider as "marked" a temperature inversion strong enough to persist despite a surface temperature increase of as much as two-thirds of the normal range, namely: 2/3 x 12oC = 8oC. The pseudo-adiabatic diagram shows that such an inversion amounts to a rise of 4oC from ground surface to the 950-millibar pressure level.

Figure 5.1 shows the frequencies of occurrence of all inversions and "marked" inversions for the years 1950 and 1951 by month. These data, obtained from the upper air soundings taken at Stewart Air Force Base, 20 miles from the reactor site, are summarized by percentages in order to indicate the relative distribution of inversions throughout the year.

There were 95 inversions during 1950 and 101 in 1951. Most of the inversions (85 of the total of 95 in 1950) commenced during the night and few of them persisted through the day. In fact, for the two-year period cited above, only 21 inversions held for more than 12 hours and only 6 lasted more than 24 hours. Inversions were well distributed through the year with no apparent seasonal preference.

The pattern of prevailing surface winds throughout the year is shown in Table 5.3. Monthly wind roses are included in Figure 5.2. These wind roses are derived from monthly mean values of observations taken at Stewart Air Force Base from September, 1942, to December, 1952. Only eight points of the compass are employed for ease of interpretation and to emphasize the wind direction values. Four categories of wind speed, other than calm, are considered; they are: 1-3, 4-12, 13-24, and greater than 24 miles per hour. Numbers of observations and average percentage frequencies by month and by wind direction are recorded in Table 5.4. These data show that the predominant wind direction are southwest and west and that combined with periods of calm, they make up 56.1 percent of the observations. Winds from the north and northwest comprised only 14.4 percent of the yearly total. Wind speeds from the south and southwest averaged 10 - 11 miles per hour; from the north and northwest 11 - 15 miles per hour. In March, the north and northwest winds are more frequent than at any other time. During the Summer and Fall, the former directions constitute as high as 45 percent of all winds; in the Summer period the predominant wind speeds tend to be lowest, i.e., in the 4 - 12 miles per hour range.

TABLE 5.3

Prevailing Surface Winds, Stewart AFB, New York

1943 - 1952 Percent Frequency

Wind Velocity Mi/Hr	Wind Direction 0°	45°	90°	135°	180°	225°	270°	315°	Calm	Total Annual
Calm									12.4	12.4
1-3	0.4	0.5	0.4	0.3	0.3	0.5	0.4	0.3		3.1
4-12	4.2	6.7	4.3	3.8	5.6	16.1	7.7	3.5		51.9
13-24	2.3	3.1	0.8	1.3	2.1	8.3	8.3	2.7		28.9
> 24	0.3	0.1	0	0	0	0.6	1.9	0.8		3.7
Annual Total	7.2	10.4	5.5	5.4	8.0	25.5	18.3	7.3	12.4	100.0

The wind rose based on this data is shown on the following page as Figure 5.3.

SURFACE WINDS

TEN YEAR PERIOD, 1943-1952

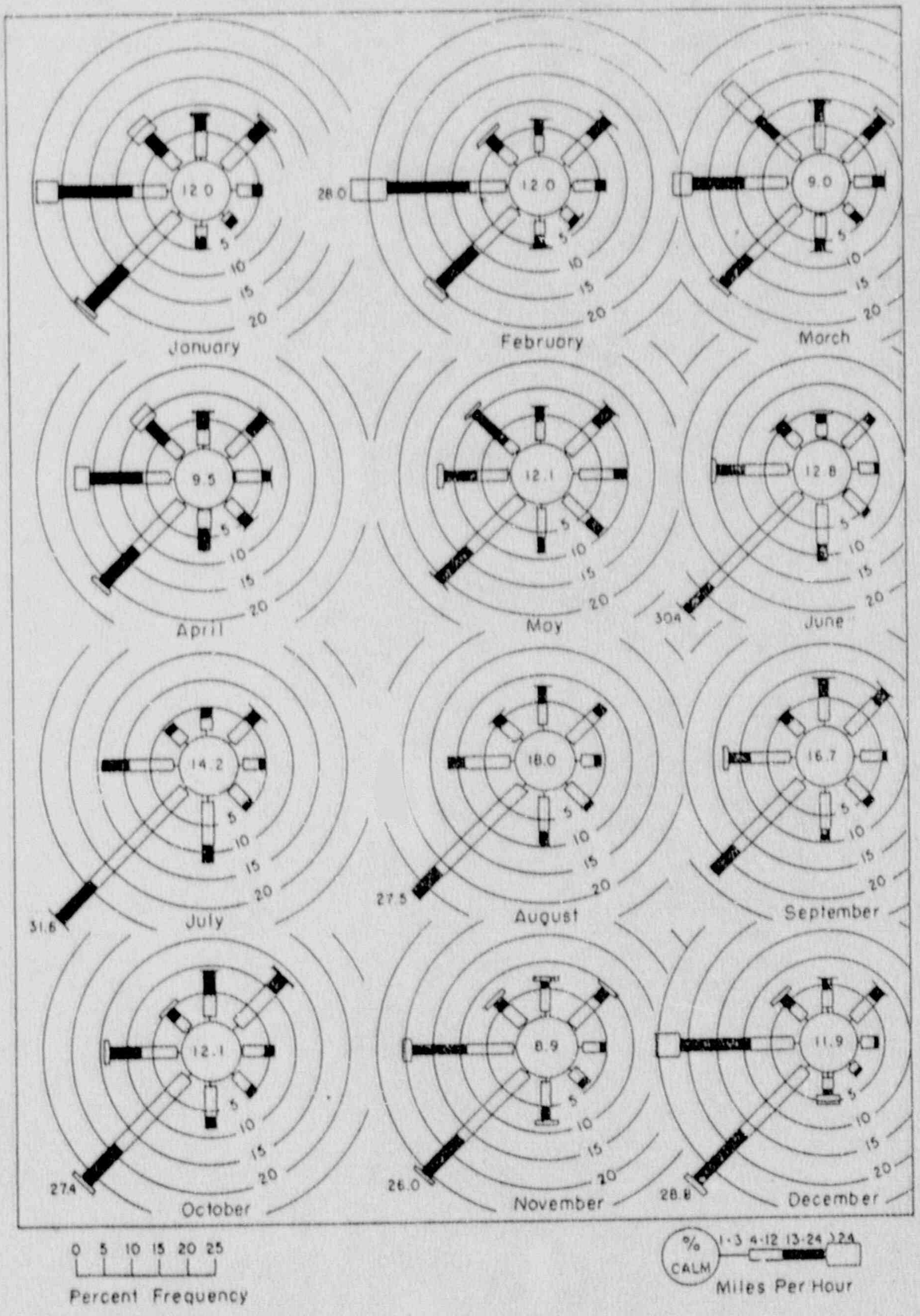

FIGURE 5.2

Average Surface Wind Direction
Ten Year Period (1943–1952)

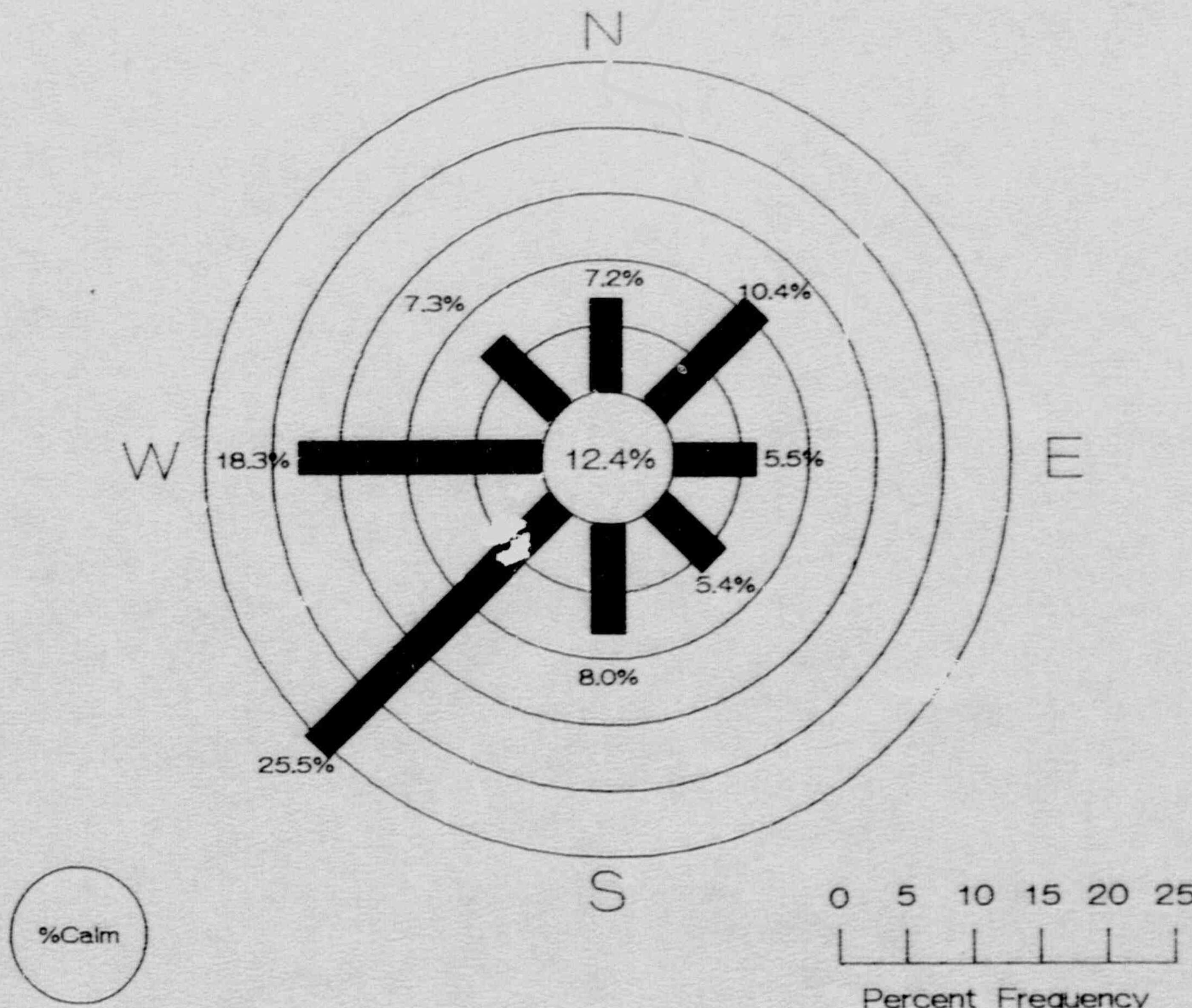

FIGURE 5.3

TABLE 5.4

AVERAGE FREQUENCY OF PREVAILING SURFACE WIND DIRECTION,

STEWART AFB, NEW YORK, 1943-1952

MONTH	Surface Wind Direction 0°	45°	90°	135°	180°	225°	270°	315°	Calm	Total No. of Obs.
January										
No. of Obs.	612	903	362	191	347	1818	1788	523	892	7436
Percent	8.3	12.2	4.8	2.6	4.7	24.4	24.0	7.0	12.0	100.0
February										
No. of Obs.	408	789	340	246	371	1382	1895	545	814	6790
Percent	6.0	11.6	5.0	3.6	5.5	20.4	27.9	8.0	12.0	100.0
March										
No. of Obs.	642	878	510	379	486	1598	1571	688	672	7434
Percent	8.7	11.8	7.0	5.1	6.5	21.5	21.1	9.3	9.0	100.0
April										
No. of Obs.	553	736	462	447	583	1625	1328	773	686	7193
Percent	7.7	10.3	6.4	6.2	8.1	22.6	18.5	10.7	9.5	100.0
May										
No. of Obs.	496	879	645	694	675	1567	989	592	897	7436
Percent	6.7	11.8	8.7	9.3	9.1	21.0	13.3	8.0	12.1	100.0
June										
No. of Obs.	379	587	354	485	814	2184	1013	454	922	7192
Percent	5.3	8.2	4.9	6.8	11.3	30.4	14.0	6.3	12.8	100.0

MONTH	0°	45°	90°	Surface Wind Direction 135°	180°	225°	270°	315°	Calm	Total No. of Obs.
July										
No. of Obs.	340	576	306	551	882	2355	1070	303	1054	7437
Percent	4.6	7.8	4.1	7.4	11.8	31.6	14.4	4.1	14.2	100.0
August										
No. of Obs.	555	694	342	442	700	2047	870	450	1338	7438
Percent	7.5	9.3	4.6	5.9	9.3	27.6	11.7	6.1	18.0	100.0
September										
No. of Obs.	614	906	465	547	776	1787	1046	445	1320	7906
Percent	7.8	11.4	5.9	7.0	9.8	22.6	13.2	5.6	16.7	100.0
October										
No. of Obs.	774	962	456	414	700	2236	1185	465	991	8183
Percent	9.4	11.8	5.6	5.0	8.6	27.3	14.5	5.7	12.1	100.0
November										
No. of Obs.	580	783	424	342	642	2059	1686	692	706	7914
Percent	7.3	9.9	5.3	4.3	8.1	26.1	21.4	8.7	8.9	100.0
December										
No. of Obs.	583	714	325	201	371	2353	2107	550	974	8178
Percent	7.3	8.7	4.0	2.4	4.6	28.8	25.7	6.7	11.9	100.0
Total No. of Obs.	6538	9407	5001	4939	7347	23011	16548	6480	11266	90537
Percent	7.3	10.4	5.6	5.4	8.1	25.4	18.3	7.1	12.4	100.0

Calms occur slightly more than 12 percent of the time on an annual basis and vary from a low figure of 9 percent in March and November to 17 - 18 percent in August and September. The influence of down-valley drift and the frequent formation of temperature inversions in valleys due to nocturnal cooling would tend to localize the distribution of air contaminants to the general vicinity of the reactor at those times.

Tables 5.5 and 5.6 give the direction of the surface winds during all inversions and during marked inversions for the years 1950 and 1951. Again, the winds are indicated to eight points of the compass for convenience. It can be seen that the wind directions found during the periods of inversions were quite similar to the prevailing wind directions at other times of the year. While inversions occurred, surface wind direction was from the southwest 31 percent of the time during both 1950 and 1951. Calms existed with more than 20 percent of the inversions. It is important to recognize that winds from a southerly or easterly quadrant would bear any postulated contaminants away from the principal urban centers of the New York City area and into the thinly populated ridge and valley terrain or toward the open spaces of the Palisades Interstate Park.

5.3.3 Topography and Geology of the Area Immediately Surrounding the Reactor Site

The reactor site is in Sterling Forest, 3-1/4 miles north-northwest of Tuxedo Park, Orange County, NY, some 1500 feet southwest of Indian Kill, a small stream flowing southeast for a mile and a half to the Ramapo River. The plant borders Long Meadow Road at an elevation of approximately 800 feet.

There is a very low north-south topographic divide between Indian Kill drainage and drainage of Warwick Brook to the south which also flows east to Ramapo River. These two small streams, Indian Kill and Warwick Brook, since they drain into the Ramapo River from the vicinity of the site dominate the drainage pattern insofar as it concerns the flow of surface or underground water away from the vicinity. The Ramapo River, cutting athwart the mountainous highlands and rising to the north near the town of Monroe, flows in a mountain-walled valley to Suffern, and thence to Passaic River at Mountain View and so on to the sea. The flow of the river varies with the season[1]. During the "water year", October 1952 to September 1953, the mean daily flow for the month of March was 611 cubic feet per second. During the dry month of September the flow fell to 16.3 cfs. An average of the monthly means for this year was 269 cfs; for 31 years this average was 218 cfs.. During a record of 31 years, occasional

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1 U. S. Geological Survey, Surface Water Supply of the United States, page 115, Water Supply Paper No. 1272, 1953

TABLE 5.5

SURFACE WINDS DURING INVERSIONS,

STEWART AFB, NEW YORK, 1950

All Inversions

Month	Surface Wind Direction 0°	45°	90°	135°	180°	225°	270°	315°	Calm	Total	Percent of Total
Jan.					4	4			3	11	12
Feb.			1			3	1		1	6	6
March										0	0
April				2	2	3		2	4	13	14
May		1		2	1	1	1		1	7	7
June				2	2	5	2			11	12
July				1	1	2	4	1	2	11	12
August						3			4	7	7
Sept.						1			1	2	2
Oct.		3		1	3	2			1	10	10
Nov.					1	3	1			5	5
Dec.	1	4	2		1	2		1		12	13
Total	1	8	3	8	15	29	9	4	18	95	
Percent of Total	1	8	3	8	16	31	10	4	19		100

Marked Inversions

Month	Surface Wind Direction 0°	45°	90°	135°	180°	225°	270°	315°	Calm	Total	Percent of Total
Jan.					1	3			1	5	33
Feb.						1				1	7
March										0	0
April					1					1	7
May										0	0
June						2	1			3	20
July										0	0
August										0	0
Sept.										0	0
Oct.				1						1	7
Nov.						1	1			2	13
Dec.		1			1					2	13
Total	0	1	0	1	3	7	2	0	1	15	
Percent of Total	0	7	0	7	20	46	13	0	7		100

TABLE 5.6

SURFACE WINDS DURING IN[illegible]IONS,

STEWART AFB, NEW YORK, 1951

All Inversions

Month	Surface Wind Direction 0°	45°	90°	135°	180°	225°	270°	315°	Calm	Total	Percent of Total
Jan.					1	5	1		1	8	8
Feb.					1	1		1		3	2
March		2		1						3	2
April				2	1	2	1			6	6
May	2	1		2		4		1	1	11	11
June	1				1	1	1		2	6	6
July				1		3	1	1	1	7	7
August		1				2	1	2	6	12	12
Sept.	1	2		1	4	6			6	20	20
Oct.		1			1	1	1	1	3	8	8
Nov.		1				3	1		2	7	7
Dec.						3	2	2	3	10	10
Total	4	8	0	7	9	31	9	8	25	101	
Percent of Total	3	8	0	7	9	31	9	8	25		100

Marked Inversions

Month	Surface Wind Direction 0°	45°	90°	135°	180°	225°	270°	315°	Calm	Total	Percent of Total
Jan.							1		1	2	13
Feb.					1	1				2	13
March		1		1						2	13
April										0	0
May				1		1				2	13
June										0	0
July										0	0
August		1								1	7
Sept.		1				1			1	3	20
Oct.							1			1	7
Nov.										0	0
Dec.								1	1	2	13
Total	0	3	0	2	1	3	2	1	3	15	
Percent of Total	0	20	0	13	7	20	13	7	20		100

floods attained a rate of 12,400 cfs for a period of a day or so. During very dry spells the rate fell as low as 7 cfs.

Though the "Highlands" are rugged, and the hillsides steep, relief is not great, only a matter of some 400 to 700 feet from the valley floors to ridge tops. A striking feature of the area, a feature resulting from a past era of glaciation, is the clearly evident clogged drainage system. Swamps and ponds abound along stream channels, as do a multiplicity of lakes, large and small, all bespeaking the fact that present streams have not, under prevailing gradients and climatic conditions cleared their over-burdened channels of glacial debris: fill, clay, sand, gravel and boulders of every size. These latter, especially, strew the hillsides. The reactor building is placed in a north trending spur of Hogback Mountain, a spur which slopes northward from something over 1500 feet of elevation to the level of Indian Kill Lake at 700 feet elevation. It is at the eastern foot of this spur, along Long Meadow Road, that the plant is located.

The detailed geology of this area has been discussed in a report entitled "Magnetite Deposits of the Sterling, NY, Ringwood, NJ Area" by Preston E. Hotz of the United States Geological Survey[2]. The area is underlain by closely folded sedimentary and igneous gneisses. The folds are in general overturned to the northwest, resulting locally in relatively steep isoclinal dips toward the southeast. The middle section of the eastern spur referred to above is underlain by quartz-oligioclase gneiss. This rock is regarded as a highly metamorphosed sediment, parts of which have taken on the aspects of an igneous rock. The "outcrop" width of this rock approximates 500 feet. From the data collected from core drillings covering the construction area, it was found that a drill placed at the center of this outcrop would not penetrate the western foot wall of the gneiss until a depth of nearly 400 feet was reached. Actually, the hole was completed at a depth of 200 feet, the approximately calculated elevation of the floor of the reactor building. To the east of the site folded metamorphosed sediments (quartzites) occur, and beyond these, granitic gneisses. These may or may not be derived from ancient sediments. Similar sequences of gneisses extend eastward to Ramapo River and beyond.

5.3.4 Seismology

The New Jersey-New York Highlands have a long record of freedom from violent earthquakes. There is no historical record of earthquakes of intensity VIII or greater occurring in this area.

The swimming-pool reactor is constructed in very firm hard rock. Only a very violent shock could affect the reactor in the chamber in which it is placed. Such a structure would vibrate as a unit even under violent shock, which is not expected.

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[2] P. E. Hotz, Magnetite Deposits of the Sterling, New York, Ringwood, New Jersey Area, U. S. Geological Survey, Bulletin 982-F, 1952.

In the past 20 years, the number of seismic recording stations in this general area has increased steadily to the point where detection of events adjacent to New York City of magnitudes > 1.8 is practically complete. Coupled with interest in the seismic safety of the power reactors located at Buchanan, NY, this has resulted in intense study of the seismic activity in the vicinity of the Ramapo Fault and estimates of the probability of occurrence of strong earthquakes at those reactor sites. In "Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey", a detailed discussion of these matters appears[3]. What follows is excerpted from this article.

The Cintichem reactor site is interior to a lithospheric plate. In such a case, potential earthquakes tend to occur along major pre-existing faults, with the larger shocks showing a greater tendency than the smaller ones to be located on the major throughgoing fault. A survey of the seismic events that have occurred in and around this region shows that most of the activity is in the Precambrian Hudson Highlands, that earthquakes in this area occur along pre-existing faults, with the large majority within 1-2 Km of the faults. About 50% of all events are almost collinear and lie along or close to the Ramapo Fault system.

Using data on small shocks obtained in the past 20 years or so from the seismic network, a relation for cumulative frequency of occurrence has been obtained. Extrapolation to larger magnitudes shows excellent agreement with the few larger historical events, of intensity VI and VII. This relation has been used by the authors[4] to predict the probability of occurrence of intensity VII and VIII shocks at the upper end of the Ramapo Fault. The results realistically are as follows:

	Recurrence Period (years)		Probability of Occurrence in 20-year period (%)	
	VII	VIII	VII	VIII
(a) Excluding events > 10 Km distance:	630	2870	3.2	0.7
(b) Including all events along Ramapo Fault:	340	1880	5.9	1.1

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[3] Aggarwal, Yash P., and Sykes, Lynn R., "Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey", Science 200, 425 (Apr. 1978).

[4] Aggarwal, Yash P., and Sykes, Lynn R., "Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey", Science 200, 425 (Apr. 1978).

The Cintichem site is about 12 Km northwest from the Ramapo Fault and therefore the predicted probabilities at this site should be less than those given above, viz. less than 3 - 6% for intensity VII, and less than 1% for intensity VIII, for a 20-year period in the future.

5.3.5 **Topographic and Geologic Relationships Affecting the Plant**

a. Excavation of the Reactor Building

The rock is quartz-oligioclase gneiss. In texture it varies from coarse to fine-grained, but these differences are interpreted to mean not so much differences in the grain size of the original sediment, as that metamorphic processes have obliterated, to varying degrees and in various places, the original bedding planes of the rocks. That is, parts of the rocks could hardly, if at all, now be distinguished from an igneous rock.

The rock is very dense and tough and relatively free from fractures, though all rocks near the surface are somewhat fractured. The first drill hole produced nearly 98% of core, the second hole drilled at an angle of 66° from the horizontal produced 99% of core.

A compression test conducted at the Brooklyn Polytechnical Institute revealed that a cylinder 4-1/4 inches long and 2-1/8 inches in diameter broke only when subjected to the great load of approximately 8 tons to the square inch, even though the section was cut with planes of schistosity dipping 55° from the axis of the cylinder.

All rocks near the surface are somewhat fractured, and rain water will enter these fractures at the surface and descend to points of escape or to points where fractures give out. Thus, even in fractured hard rocks a water table is built up. In such rocks water descends to depths where fractures die out, beneath which position such rocks may be essentially dry. In core hole No. 1 water appears to stand at 85 feet ± below the surface. Thus, there is 115 ± feet of water in the hole, which suggests that fracturing at depth is slight.

b. Drainage

It does not appear that contamination of Tuxedo Lake or any of the connecting smaller lakes from plant operations is possible for reasons noted below.

Surface drainage from the site is exclusively by way of Indian Kill. The Kill enters Ramapo River 1-1/2 miles east of the plant at elevation 463 feet. Tuxedo Lake stands at elevation 560 feet. Wee Wah, the adjoining lake to the north stands lower than Tuxedo Lake to which it is joined by a small stream of high gradient. Wee Wah Lake consists of two segments. The southern, higher segment is separated from the lower northern segment by a stream of steep gradient. This northern segment, in turn, discharges over an earth dam and masonry spillway to a small stream that discharges into Ramapo River. Thus, it may be seen that even if the Indian Kill were contaminated, it is not remotely possible to carry such contamination by surface flow to any of this chain of three lakes.

Indian Kill presents the only obvious path for contamination by underground flow, that is, through alluvial sand, silts and gravels that lie beneath the stream channel, resting on the gneissoid bedrock of the region. Water passes downstream easily but slowly through these alluvial deposits. Obviously such waters could not possibly ascend into the chain of Tuxedo Lakes. Water passing underground beneath the mountainous ridges, through the fractures in the hard rocks does not seem possible. The mountainous tract which is bounded by Indian Kill, Long Meadow Road, Warwick Brook and Ramapo River naturally contains some ground water within fractures in the rocks. But this water drains outward to the nearest and most accessible exit, namely either Indian Kill, Warwick Brook or Ramapo River. Water cannot pass against this outward flow, across this mountainous tract, and even assuming it could, it could not pass the boundary of Warwick Brook which flows east to Ramapo River.

6.0 STATUS OF COMPLIANCE

The decommissioning plan has been prepared in accordance with the applicable regulations and guidelines for reactor decommissioning and worker safety. Engineering controls and implementation of the occupational and radiological protection programs will lead to compliance with applicable environmental quality standards during decon activities.

7.0 <u>ADVERSE IMPACT INFORMATION FROM ROUTINE DECOMMISSIONING OPERATIONS</u>

Implementation of the decommissioning activities will not have a significant impact on the human or natural environmental resources. These activities will be carried out in controlled environments out of necessity due to the potential hazards involved. Specific impact evaluations that were performed are described below.

<u>Radiological Impact to the Public and to Workers</u>

The potential radiological adverse effects are (1) exposure to workers participating in the dismantling operation to the direct external radiation levels associated with the facility and to airborne concentrations of radioactive materials released to the work environment during dismantling and (2) exposure to the public to direct external radiation during the transport of packaged radioactive wastes and to airborne and waterborne radioactivity due to releases beyond the site boundaries during dismantling activities.

<u>General Public Exposure</u>

Limits on releases of radioactive liquids and airborne materials to the environs will be in effect during decommissioning activities. These limits will be commensurate with meeting ALARA policies and federal and state regulations. Prior to shutting down the reactor and related production operations, Cintichem was maintaining well developed Health Physics and Environmental programs with departments staffed by highly qualified and experienced employees. The caliber of programs and staff will be maintained throughout decommissioning activities and will be enhanced as necessary to meet the needs of this effort. Large scale medical radiochemical production has ceased and decommissioning activities will be performed under well controlled and monitored conditions. Releases beyond the site boundaries during decommissioning will be significantly below those experienced prior to cessation of radiochemical production. Total dose to the public from releases and direct radiation from the facility during decommissioning activities is not expected to be measurably above background levels. The frequency of radioactive waste shipments required for decommissioning activities will be greater than those required during facility operation. However, the potential population dose per shipment will be much less due to decreased quantities of radioactive material in each shipment.

<u>Exposure to Workers</u>

Total dose to workers during decommissioning activities will be maintained ALARA and well within regulatory guidelines. Decommissioning tasks will be reviewed in advance by the Cintichem Nuclear Safeguards and Radiation Safety committees and closely controlled and monitored during their implementation. ALARA principles will be applied for all decommissioning activities. Radiation work permits will be required for each decommissioning task. They will define the scope of the work activities, the protective measures and conditions which must be met to perform the job, etc.. For each task, health physics will establish and maintain radiation control areas, assure that radiological safety measures are implemented, and exposures are monitored.

Air Quality Impact - Non-radiological

Throughout the first phases of the decommissioning activities the non-radiological air quality impact will be negligible to non-existent. During the first phases the buildings will remain relatively intact and decommissioning tasks will be doubly enclosed by containment tents erected within the buildings. HEPA filters will be maintained at high efficiency to remove potential air contaminants. During the latter phases of decommissioning, after all radioactivity above unrestricted release levels have been safely removed, large equipment will be utilized to raze the buildings and approach conditions that existed with the original environment. Depending on the climatic conditions at the time and the particular task being accomplished, small amounts of non-radioactive nuisance dust may be introduced into the air during building demolition. Since the facility is located in a relatively remote area and efforts to minimize the generation of dust will be implemented, any impact to the public, although greater than that of earlier phases, will still be negligible.

Socioeconomic Impact

No adverse social or economic effects are expected from the Cintichem dismantling activities. The current number of employees at the facility is about 20 to 30 lower than the staffing that existed when the reactor and hot lab facilities were in operation. It is expected that the number of workers required for dismantlement of the facility, will be made up of existing employees, individuals hired to meet the special requirements of certain tasks, and contractors. The decommissioning workforce will be approximately the same as that during normal operation. This employment level will continue to provide income to the area and create commerce required to provide materials and services during the approximately two year dismantling period. Dismantling operations will not change population or growth patterns in the area significantly due to the relatively few people involved.

Impact on Disposal Site Operation

Radioactive waste will be disposed of at licensed commercial burial grounds. No adverse effects on disposal site operations will occur due to the one time decommissioning disposal volume.

Other Miscellaneous Impacts

Noise Impact

Noise levels at the facility during the dismantling operation may increase at times as a result of heavy equipment operation and increased waste shipments. Since the facility is relatively isolated from the nearest public residence, this impact is expected to be negligible.

Biota Impact

Decommissioning activities will primarily be focused within and immediately surrounding the foundations of the hot lab and reactor buildings and no impact is expected to the biota. A slight adverse impact may result to the flora in the vicinity of the main exhaust stack when it is dismantled and removed from the hillside at the rear of the hot lab facility, but this effect will be minimal. There are no known endangered species in the vicinity of the facility.

Topography Impact

Dismantling of the Cintichem facility will not effect the present water drainage system. No new major structures will be erected during decommissioning and excavated cavities will be backfilled to ground level.

Utilities and Services

There will be little or no impact to utilities and services as result of decommissioning activities. Domestic water will continue to be supplied through the existing domestic water system. Water usage for decommissioning activities will be small and strictly controlled. Water usage by employees for drinking, sanitary facilities, etc. will remain relatively unchanged. Because makeup water for the reactor pool and cooling tower is no longer required, overall water usage during decommissioning will be less than that of the previous operating facility. Discharge from the sanitary facility will continue to flow into the existing sanitary drain system and no increased discharge is anticipated. Electrical power consumption is expected to remain at the same level or below that of the operating facility. Many large pieces of equipment used for operation and production will no longer be required (welders, system pumps, reactor and other consoles, etc.). Electrical loads for dismantling equipment, in comparison, will be much lower. The solid waste will continue to be disposed of by contract in a county sanitary landfill. The amount of solid waste generated during decommissioning will be equal or less than that from the operating facility.

ATTACHMENT A

CASE 1

Estimated total dose to the population within a five mile radius of the Cintichem site from environmental release caused by scabbling of the hot cell walls.

Assumptions for Analysis

a. The internal surfaces of all five hot cells, conveyor station, and elevator shaft are at a level of 1 uCi/cm^2 Sr-90 and 1 uCi/cm^2 Cs-137.

This is derived from survey data of Cell #1 where a 55 mrem/hr gamma dose rate in the central cell region due to surface contamination of walls, ceiling and floor was observed. A computer model and manual calculation check verifies equivalence of a 1 uCi/cm^2 Cs-137 level evenly distributed on all cell #1 internal surfaces with the 55 mrem/hr dose rate measured. Of the long-lived fission products, Cs-137 and Sr-90 are the isotopes of concern. The assumption of an equal concentration of Sr-90 is independent of the gamma dose rate measurement since Sr-90 is not a gamma emitting isotope.

b. The total surface area of the structures listed in a. is 3657 ft^2 as follows (Ref. telephone conversation with J. Adler of TLG on September 12, 1990)

Hot Cell #1	932
Hot Cell #2	520
Hot Cell #3	520
Hot Cell #4	520
Hot Cell #5	888
Conveyor Station	94
Elevator	183
	3657 total ft^2

This total corresponds to 3.40×10^6 cm^2

$$3657 \text{ ft}^2 \times \frac{929.0 \text{ cm}^2}{\text{ft}^2} = 3.40 \times 10^6 \text{ cm}^2$$

c. Based on a. and b. above, the total radioactivity removed by scabbling is 3.4×10^6 uCi Sr-90 and 3.4×10^6 uCi Cs-137.

d. Ten percent of the radioactivity removed by scabbling becomes airborne within the hot cells and is available for filtration by HEPA filters in the ventilation system.

e. Two sets of HEPA filters are in series in the hot cell ventilation system. Each set of HEPA filters removes only 99 percent of respirable particles of Cs-137 and Sr-90 radioactivity. (The manufacturer will certify the HEPAs to be 99.97% efficient).

f. The hot lab ventilation flow rate is 14760 ft^3/min (Ref. Measurement of Flow Rate on November 9, 1988). This corresponds to 2.2×10^{14} cm^3/yr.

$$14760 \ ft^3/min \times 60 \ min/hr \times 24 \ hr/day \times 365 \ day/yr \times 2.83 \times 10^4 \ cm^3/ft^3 = 2.20 \times 10^{14} \ cm^3/yr$$

g. The scabbling process takes place within a one year period.

h. The estimated total dose to the population is based on inhalation dose only because the annual Land Use Survey shows that there are no gardens of leafy vegetables > 500 ft^2 within a 5 mile radius.

i. The population in the five mile radius is based on a total of 8600 (1980 data) with 3458 at Tuxedo Park, New York (3.25 miles SSE) and 2262 at Greenwood Lake, New York (2.5 miles SW). The 2880 remaining are considered to be evenly distributed among all other directions around the site at an average distance of 1.5 miles.

j. The most conservative wind stability class, Pasquill-Gifford F, is assumed at all times.

k. A conservative wind velocity of 1 meter/sec = u is assumed.

l. Dose calculation assumes that the entire population in each octant is conservatively located along the centerline of the plume (i.e. no vertical or horizontal displacement from the centerline).

m. Breathing rate for teenager is 8000 m^3/yr (Reg. Guide 1.109 Table E-5).

n. Inhalation dose factor for teenager is 1.35×10^{-2} mrem/pCi Sr-90 inhaled for bone (Reg. Guide 1.109 Table E-8).

o. Inhalation dose factor for teenager is 3.89×10^{-5} mrem/pCi Cs-137 inhaled for total body (Reg. Guide 1.109 Table E-8).

SUMMARY OF RESULTS

The estimated total person-rem to the population in a five mile radius is 3.2×10^{-4} person-rem (see Table A-1).

Sample Calculation (for SE direction)

3.25 mi x 5280 ft/mi x 1 m/3.28 ft = 5230 meters

Release rate for Sr-90 and Cs-137

$$\frac{3.4 \times 10^6 \text{ uCi}}{\text{yr}} \times \frac{1 \text{ Ci}}{10^6 \text{ uCi}} \times 0.1 \times 0.01 \times 0.01 = 3.4 \times 10^{-5} \text{ Ci/yr} = Q$$

The basic equation used to estimate dispersion in an airborne plume as it is blown downwind from a stack is the Gaussian plume equation of Pasquill (1961) as modifed by Gifford (1961).[1]

$$X = \frac{Q}{2\pi\sigma_y\sigma_z u} e^{-1/2\left(\frac{y}{\sigma_y}\right)^2}\left(e^{-1/2\left(\frac{z-H}{\sigma_z}\right)^2} + e^{-1/2\left(\frac{z+H}{\sigma_z}\right)^2}\right)$$

X = concentration in air at x meters downwind, y meters crosswind, and z meters above ground (Ci/m^3),

Q = uniform emission rate from the stack (Ci/sec),

u = mean wind speed (m/sec),

σ_y = horizontal dispersion coeffiient (m), Figure A-1

σ_z = vertical dispersion coefficient (m), Figure A-2

H = effective stack height (physical stack height plus the plume rise) (m)

y = crosswind distance (m),

z = vertical distance (m).

Setting y = 0, H = 0, and z = 0, (worst case), one obtains the relationship used for the analysis.

$$X = \frac{Q}{\pi\sigma_y\sigma_z u} = \frac{3.4 \times 10^{-5} \text{ Ci/yr}}{(3.14)(160\text{m})(35\text{m})(1\text{m/sec} \times 3.15 \times 10^7 \text{sec/yr})}$$

$$= 6.1 \times 10^{-17} \text{ Ci/m}^3 \text{ or uCi/cm}^3$$

[1] Gifford F.A., Jr. 1961. Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion. Nucl. Saf. 2(4): 14-15 as quoted in RSIC Computer Code Collection-357 AIRDOS-EPA Estimation of Radiation Doses Caused by Airborne Radionuclides in Areas Surrounding Nuclear Facilities ORNL-5532 UC-41,11. June 1979.

Since the wind is blowing into the SE octant during 7.3% of the time (see Section 5):

$0.073 \times 6.1 \times 10^{-17}$ Ci/m^3 = 4.5×10^{-18} Ci/m^3 average concentration

The dose to a teenager is the limiting case for Sr-90

8000 m^3/yr x 1.35×10^{-2} mrem/pCi x 4.5×10^{-18} Ci/m^3 x 10^{12} pCi/Ci = 4.9×10^{-4} mrem/yr

The "effective dose equivalent" weighting factor for stochastic effects is 0.03 for bone surfaces (Ref. ICRP Publication 26, 1977, p. 21). Therefore, the equivalent total body dose is:

4.9×10^{-4} mrem/yr x 0.03 = 1.5×10^{-5} mrem/yr

For Cs-137 the total body dose is:

8000 m^3/yr x 3.89×10^{-5} mrem/pCi x 4.5×10^{-18} Ci/m^3 x 10^{12} pCi/Ci = 1.4×10^{-6} mrem/yr

The estimated "total body effective dose equivalent" is 1.6×10^{-5} mrem/yr

1.5×10^{-5} mrem/yr + 1.4×10^{-6} mrem/yr = 1.6×10^{-5} mrem/yr

The estimated person-rem for the SE octant for the year during which scabbling takes place is 5.5×10^{-5} person-rem

1.6×10^{-5} mrem/yr x 1 rem/10^3 mrem x 3458 persons = 5.5×10^{-5} person-rem

CASE II

The estimated maximum dose to the closest residential individual (teenager) offsite due to environmental release caused by scabbling of the hot cell walls

ASSUMPTIONS FOR ANALYSIS

a. As in Case I of this Appendix, the following assumptions are made:

   1. Total radioactivity removed from the hot cells by scabbling is 3.4×10^6 uCi Sr-90 and 3.4×10^6 uCi Cs-137.

   2. Ten percent of the radioactivity becomes airborne within the hot cells.

   3. Two sets of HEPA filters in series at only 99 percent removal efficiency are in operation in the ventilation system. (The manufacturer will certify the HEPAs to be 99.97% efficient).

   4. The hot lab ventilation flow rate is 2.2×10^{14} cm^3/yr.

   5. The scabbling process takes place within a one year period.

   6. The most conservative wind stability class, Pasquill-Gifford F, is assumed at all times.

   7. A conservative wind velocity of 1 meter/sec = u is assumed.

   8. Inhalation dose factor for teenager is 1.35×10^{-2} mrem/pCi Sr-90 inhaled for bone (Reg. Guide 1.109 Table E-8).

   9. Inhalation dose factor for teenager is 3.89×10^{-5} mrem/pCi Cs-137 inhaled for total body (Reg. Guide 1.109 Table E-8).

b. The total dose to an individual (Laurel Ridge development) is based on inhalation of Cs-137 and Sr-90 released. Also,

   1. The housing development is located east at 109^o from the stack where the percentage of wind into this direction is 18.3.

   2. Individuals at this location are conservatively assumed to be at the same height as the stack plume centerline so that there is no vertical displacement to consider.

   3. The nearest residence is 450 m from the stack.

c. The estimated individual release rates for Sr-90 and Cs-137 are 3.4×10^{-5} Ci/yr (see sample calculation in Case I).

d. The vertical dispersion coefficient (σ_z) for Class F condition at 450 meters is 8.0 m. (Slade, D.H. ed., <u>Meteorology and Atomic Energy</u>, July 1968, pp. 102 - 103).

e. The horizontal dispersion coefficient (σ_y) for Class F condition at 450 meters is 18 m. (Slade, D.H. ed., <u>Meteorology and Atomic Energy</u>, July 1968, pp. 102 - 103).

f. Teenage breathing rate is 8000 m^3/yr.

g. The "effective dose equivalent" weighting factor for stochastic effects is 0.03 for bone surfaces (Ref. ICRP Publication 26, 1977, p. 21).

<u>CALCULATION</u>

$$X = \frac{Q}{\pi \sigma_y \sigma_z u} \; * = \frac{3.4 \times 10^{-5} \text{ Ci/yr}}{(3.14)(18\text{m})(8.0\text{m})(1\text{m/sec} \times 3.15 \times 10^{7} \text{ sec/yr})}$$

$$= 2.4 \times 10^{-15} \text{ Ci/m}^3 \text{ or uCi/cm}^3$$

* see Case I

The Sr-90 dose to a teenager at Laurel Ridge is based on the average concentration for Class F at 450 meters (calculated above), times the fraction of wind blowing in that direction.

$$2.4 \times 10^{-15} \text{ Ci/m}^3 \times 0.183 = 4.4 \times 10^{-16} \text{ Ci/m}^3$$

The dose from Sr-90 is:

$$8000 \text{ m}^3\text{/yr} \times 1.35 \times 10^{-2} \text{ mrem/pCi} \times 4.4 \times 10^{-16} \text{ Ci/m}^3 \times 10^{12} \text{ pCi/Ci} = 4.8 \times 10^{-2} \text{ mrem/yr}$$

After considering the weighting factor, the "total body effective dose equivalent" is 1.4×10^{-3} mrem/yr.

$$4.8 \times 10^{-2} \text{ mrem/yr} \times 0.03 = 1.4 \times 10^{-3} \text{ mrem/yr}$$

The dose from Cs-137 is:

$$8000 \text{ m}^3\text{/yr} \times 3.89 \times 10^{-5} \text{ mrem/pCi} \times 4.4 \times 10^{-16} \text{ Ci/m}^3 \times 10^{12} \text{ pCi/Ci} = 1.4 \times 10^{-4} \text{ mrem/yr}$$

The estimated "total body effective dose equivalent" is 1.5×10^{-3} mrem/yr

$$1.4 \times 10^{-3} \text{ mrem/yr} + 1.4 \times 10^{-4} \text{ mrem/yr} = 1.5 \times 10^{-3} \text{ mrem/yr}$$

This estimated dose of 1.5×10^{-3} mrem/yr is 3×10^{-4} percent of the 500 mrem whole body dose limit recommended for persons living in unrestricted areas.

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TABLE A-1

Direction From Stack	Fractional Wind	Population	Distance (m)	(m)	(m)	u (m/sec)	X (Ci/m^3)	(X)(Fractional Wind)	Sr-90 Teenager Total Body Effective Dose Equivalent (mrem/yr)	Cs-137 Teenager Total Body Effective Dose Equivalent (mrem/yr)	Sr-90 & Cs-137 Total Body Effective Dose Equivalent (mrem/yr)	Person-Rem
N	0.080	480	2415	89	24	1	1.8×10^{-16}	1.3×10^{-17}	4.2×10^{-5}	4.0×10^{-6}	4.6×10^{-5}	2.2×10^{-5}
NE	0.255	480	2415	89	24	1	1.6×10^{-16}	4.1×10^{-17}	1.3×10^{-4}	1.3×10^{-5}	1.4×10^{-4}	7.0×10^{-5}
E	0.183	480	2415	89	24	1	1.6×10^{-16}	2.9×10^{-17}	9.5×10^{-5}	9.0×10^{-6}	1.0×10^{-4}	5.0×10^{-5}
SE	0.073	3458	5230	160	35	1	6.1×10^{-17}	4.5×10^{-18}	1.4×10^{-5}	1.4×10^{-6}	1.6×10^{-5}	5.5×10^{-5}
S	0.072	480	2415	89	24	1	1.6×10^{-16}	1.2×10^{-17}	3.7×10^{-5}	3.6×10^{-6}	4.1×10^{-5}	2.0×10^{-5}
SW	0.104	2262	4025	125	31	1	8.9×10^{-17}	9.3×10^{-18}	3.0×10^{-5}	2.9×10^{-6}	3.3×10^{-5}	7.4×10^{-5}
W	0.055	480	2415	89	24	1	1.6×10^{-16}	8.8×10^{-18}	2.9×10^{-5}	2.7×10^{-6}	3.1×10^{-5}	1.5×10^{-5}
NW	0.054	480	2415	89	24	1	1.6×10^{-16}	8.6×10^{-18}	2.8×10^{-5}	2.7×10^{-6}	3.1×10^{-5}	1.5×10^{-5}
												3.2×10^{-4} person-rem

ATTACHMENT B

ESTIMATION OF DOSE AT THE PROPERTY LINE AND TO NEAREST RESIDENT RESULTING FROM SHIPMENTS OF RADIOACTIVE WASTE

Assumptions

1. Each tractor trailer contains one-hundred 7.5 cubic foot capacity drums of Class A waste.

2. Each drum contains approximately 5.3 mCi of radioisotopes with average energy 0.66 MeV. This quantity yields the maximum allowable Department of Transportation dose rate of 10 mRem/hr at 6 feet from the surface of the trailer at the center of its length.

3. The property line at Long Meadow Road is 600 feet (18290 cm) from the trailer.

4. The nearest resident at Laurel Ridge development is 1350 feet (41150 cm) from the trailer.

5. Trailers containing waste are present on site for four days each week during an entire year (5006 hr/yr).

6. The attenuation coefficient in air for 0.66 MeV gamma radiation is 1×10^{-4}/cm (Ref. Cember, H., Introduction to Health Physics, Pergamon Press, 1969, p. 136).

Calculation

If the entire inventory of the trailer is considered a point source, the dose rate at one foot is 2.1 Rem/hr. This is derived using 6 CEn = Rem/hr at one foot (Ref. Radiological Health Handbook, USHEW, 1970, p. 32) for gamma radiation.

C = total Curie inventory in trailer = 100 (0.0053 Ci) = 0.53 Ci

E = energy of the gamma radiation = 0.66 MeV

n = fraction of the decays that produce gammas = 1.0

6(0.53)(0.66)(1.0) = 2.1 Rem/hr at one foot

At the site boundary, the dose rate is 3.9×10^{-6} Rem/hr
For a point source:

$$(\text{dose rate})\ (\text{distance})^2 = (\text{dose rate})(\text{distance})^2$$
$$(2.1\ \text{Rem/hr})(1\ \text{ft})^2 = \text{dose rate}\ (600\ \text{ft})^2$$

5.8×10^{-6} Rem/hr = dose rate at 600 ft due to spreading with no attentuation.

With attentuation, the dose is

$(5.8 \times 10^{-6}$ Rem/hr$)(e^{-ud})B$

where:

u = attenuation coefficient in air = 1×10^{-4}/cm

d = distance through air that the gamma rays travel = 600 ft

B = buildup factor = 4.2 (for ud = 1.8)

$$(5.8 \times 10^{-6} \text{ Rem/hr})e^{-(1\times10^{-4}/\text{cm})(18290 \text{ cm})}(4.2) = 3.9 \times 10^{-6} \text{ Rem/hr}$$

For an estimated total annual dose of 0.020 Rem at the site boundary.

5006 hr/yr x 3.9×10^{-6} Rem/hr = 0.020 Rem

Similarly, at the closest Laurel Ridge property to the site, the dose rate is 2.3×10^{-7} Rem/hr

$$(1^2/1350^2)(2.1 \text{ Rem/hr})e^{-(1 \times 10^{-4}/\text{cm})(41150\text{cm})}(14) = 2.7 \times 10^{-7} \text{ Rem/hr}$$

Using B = 14 (for ud = 4.1)

For an estimated total annual dose of 1.3×10^{-3} Rem/yr at the nearest resident.

5006 hr/yr x 2.7×10^{-7} Rem/hr = 1.3×10^{-3} Rem/yr

The estimated annual dose of 1.3×10^{-3} Rem at the nearest resident corresponds to 0.26 percent of the annual limit of 0.5 Rem.

APPENDIX I

BACKGROUND DETERMINATION

Characterization data will be evaluated as levels above site background. Background will include both "instrument background" and background due to naturally occurring radioactive materials, including enhanced background radiations due to technology (e.g., nuclear weapons tests). Therefore, reliable material background data will be obtained for each type of measurement or determination. These will be for:

- Direct surface beta or alpha contamination;
- Removable beta and alpha contamination, as applicable (activity in smear disc only);
- Soil sampling;
- Gamma exposure rates at one meter from soil or construction materials.

Good background data will be obtained by using:

- Unbiased sampling;
- Proper number of samples;
- Selection of background sample areas least likely to be affected by the Cintichem reactor/hot lab facility;
- Use of calibrated and stable instruments;
- Quality assured results.

Of primary importance will be the selection of background sampling areas which closely resemble the materials to be sampled or measured but which have not been affected by Cintichem operation.

Removable surface contamination determinations do not typically have a clean material background other than the counting instrument's background. This will be determined by counting an unused smear.

A measurement of background for the total surface contamination determinations will be made for bare concrete, painted concrete, painted and bare cinder block, painted steel surfaces, asphalt paving and, wood and sheet rock surfaces. Background will be determined from data obtained with at least 30 measurements on each type of material. Surfaces used for the background determinations will be chosen to simulate, as closely as possible, the construction materials to be encountered during the characterization.

Materials used for the total surface background determinations will be chosen from site structures at locations that are not likely to have been affected by Cintichem operation. These locations should be upwind, upstream and up-elevation, as is feasible, to avoid any contamination due to reactor operations, no matter how small it might be.

Outdoor background gamma exposure rates will be taken at an off-site location which will simulate a rock cliff type terrain (such as a highway cut-out).

Background measurements may vary considerably from point to point. However, for each type of measurement, a background level, "B", will be determined if detectable results are obtained. Above this level an individual characterization measurement will be interpreted as reflecting contamination. The definition of a "background level" is based on the assumption that the distribution of background data is either lognormal (i.e., their logarithms fit a normal [Gaussian] distribution) or normal.

"B" will be determined so that the probability that x (the random variable for the given radiological determination) is less than or equal to B is 95%, or, symbolically, probability $(x \leq B) = 0.95$. Some measurements less than B could be due to slight contamination, but there will be background measurements at the same levels. Measurements that are above B will have a small likelihood of being background measurements or, equivalently, a large likelihood of reflecting contamination.

Once the sample background measurements are made, the natural logarithms of each will be determined and the sample mean (ln x) and sample standard deviation "s" will be computed:

$$\overline{(\ln x)} = \frac{(\sum_k \ln x_k)}{n} \qquad s = \sqrt{\frac{\sum_t [\overline{(\ln x)} - \ln x_i]^2}{n-1}}$$

The "maximum likelihood" estimate of ln B is then:

$$\ln B = \overline{(\ln x} + 1.645 \sqrt{\frac{(n-1)}{n}} \; s)$$

so that B can be estimated from the formula:

$$B = \exp \overline{(\ln x} + 1.645 \sqrt{\frac{(n-1)}{n}} \; s)$$

The preceding equation will be used to obtain an estimate of the background level B for each radiological determination to be made. Where a normal distribution is found, 'he above equations are adjusted accordingly.