SAFETY ANALYSIS REPORT

for the

UNION CARBIDE RESEARCH REACTOR

(UCNR)

Medical Products Division

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SECTION A

INTRODUCTION

This revision to the Final Hazards Summary Report (FHSR) of November 1960 has been prepared in compliance with Part 50, Title 10, Chapter 1, Code of Federal Regulations and is submitted as a part of an application for extension of the operating license of the Union Carbide Research Reactor beyond its current expiration date of June 30, 1980. Construction Permit No. CPRR-18 was issued by the A.E.C. on October 31, 1957.

The reactor is a pool-type research reactor located in Sterling Forest, near Tuxedo, New York. Operation is at thermal power levels up to 5 megawatts. MTR type fuel elements are used and are arranged on a grid supported by a bridge spanning a two-section pool. The forced circulation of light water moderates and cools the reactor. The reactor can operate in either of two sections of the pool. The stall provides beam tubes, pneumatic rabbits, and a thermal column; the open or pool end permits bulk irradiation.

The pool-type reactor is housed in a reinforced concrete structure designed for low air leakage. Adjacent laboratories provide supporting facilities, including hot laboratories, shops, and offices.

This report reviews the design and safety features of the reactor facility. These features are discussed under the following general headings:

Section	A.	Introduction
"	B.	Reactor Description
"	C.	Reactor Design
"	D.	Reactor Building
"	E.	Reactor Operation
"	F.	Site Geography
"	G.	Hazards

Subsequent to the publication of the original F.H.S.R. in November 1960, a number of Supplements were written to update the thermal-hydraulic analysis, the design basis accident, routine releases, and fuel element design, and were submitted to N.R.C. in 1977-78 to serve as detailed technical bases to the Technical Specifications that were approved in May 1979. These are now made a part of this Revised F.H.S.R., either by incorporation or as Appendices, which is now designated the "Safety Analysis Report (SAR)". The SAR also includes, or references, authorized changes made to the reactor facility in the past 20 years.

The succeeding analyses demonstrate that the reactor and facilities can continue to be operated without endangering the health and safety of persons inside or outside the facility. The replacement or inspection of safety-related components and systems assures safe operation for a license-renewal period comparable to the presently-expiring period. A discussion of those safety-related items that might be expected to suffer deterioration in the prior 20-year operating period, and the measures taken to assure their integrity during the renewal period is given in Appendix 7.

SECTION B

REACTOR DESCRIPTION

1. GENERAL

This 5-megawatt pool-type research reactor is a light-water moderated, heterogeneous, solid fuel reactor in which water is used for cooling and shielding. The reactor core is immersed in either section of a two-section concrete pool filled with water. One of the sections of the pool contains an experimental stall into which beam tubes and other experimental facilities converge. The other section is an open area permitting bulk irradiation. The reactor can be operated in either section.

Spanning the pool is a manually-operated bridge, from which is suspended an aluminum tower supporting the reactor core. Control of the reactor core is accomplished by the insertion or withdrawal of neutron-absorbing control rods suspended from control drives mounted on the reactor core bridge. Additional control is provided by the temperature coefficient of reactivity. The aluminum tower and movable bridge are shown in Figure 1.

Heat, created by the nuclear reaction, is dissipated by a forced circulation cooling system. Externally located pumps, storage tanks, water-to-water heat exchangers, a cooling tower, a demineralizer plant, and a filter complete the water handling systems for the reactor.

REACTOR SPECIFICATIONS

Fuel	93% enriched U-235. MTR type, Al clad fuel assemblies.
Power	Five megawatts (heat)
Cold Clean Critical Mass:	
Stall:	3.45 KG U-235
Pool:	3.83 KG U-235
Lattice	54 holes on 6x9 pattern
Flux Density	2.5×10^{13} n/cm^2/sec. (average, stall)
Moderator	H_2O
Reflector	H_2O, Graphite
Shielding	H_2O, Lead, Magnetite concrete, and regular concrete.

Cooling	Primary Loop - heat exchanger Secondary Loop - cooling tower
Water Purification	Demineralization of a portion of the primary flow
Control	5 Boron-Carbide or Ag/In/Cd control rods 1 stainless steel regulating rod
Irradiation Facilities	4 - 6" beam tubes 2 - 8" beam tubes 3 - pneumatic rabbits 1 - thermal column

2. COMPONENTS

The major components or systems may be classified as follows:

a. Pool (including embedments and accessories.)
b. Bridge and Support Structure
c. Core
d. Cooling System
e. Control and Instrumentation
f. Experiment Facilities

3. POOL

a. Concrete Pool

The reactor pool (Figures 2, 3, 4, and 5) is a reinforced concrete enclosure approximately 49 feet long, 23 feet wide and 32 feet high, with the open end embedded in the mountainside. The pool walls support the movable bridge from which is suspended the tower supporting the core. Adequate biological shielding is provided by the depth of water above the core and by the concrete pool walls.

The pool is divided into two sections separated by a four foot 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 permits bulk irradiations and provides storage space for irradiated fuel and experiments. A 12-foot deep canal connects the open pool with the hot cells to permit the transfer of irradiated material between the two facilities.

Shielding in the stall area consists of a 5.8 foot thick magnetite concrete wall to a height of 15 feet above the pool floor. The wall thickness is then reduced to three feet to the top of the stall. The lower four feet of the wall above the step is of magnetite concrete and the remainder of regular 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 foot step, and the sides of the pool behind the fuel storage racks.

b. Embedments

During the erection of the pool, certain equipment is positioned and permanently embedded in the concrete floor and walls. This equipment includes:

(1) The reactor cooling water inlet and outlet pipes in both the pool and stall operating positions.

(2) An overflow gutter, located on both sides of the pool near the top of the inner walls, provides a skimming action on the pool surface. The water flowing into the overflow gutters drains directly into the hold-up tank. Any air carried into the tank would be vented rather than be carried through the reactor primary water.

(3) Stainless steel anchor bolts and pads set in the floor for the core outlet assemblies in the pool and stall. Additional pads are provided in the floor and vertically along the stall rear for mounting the thermal column extension.

(4) Supplementary stainless steel anchor bolts and aluminum pads embedded in the stall area for mounting the thermal shield.

(5) Six beam tube liners cast integrally with the wall in the stall area. These inserts, at reactor core elevation, extend outward in a radial fashion from the inner wall surface.

(6) Additional embedments for mounting bridge rails, aluminum gate channels, etc.

(7) A 3/16 inch steel plate located approximately in the center of the concrete pour of the reactor pool and stall. All embedments penetrating the steel liner are welded to the liner. This is a design feature made to prevent leakage of water through the concrete pool walls, as has occurred with several other pool type reactors.

c. Watertight Gate

A tapered watertight gate, inserted in the dividing bulkhead opening, permits the draining of each pool section independently. A rubber seal, along two sides and the bottom, insures watertight sealing. The gate is guided into place by embedded channels in each side of the dividing bulkhead. Lifting lugs facilitate handling during insertion and removal by the over-head service crane.

d. Bridge Rails

The bridge rails are mounted atop the pool walls parallel to the longitudinal center-line of the pool. The rails are fastened to base plates that are positioned on anchor bolts embedded in the concrete. Stops, welded at the ends of the rails, limit bridge travel in either direction.

e. Storage Racks (Wall)

The wall storage racks are designed to afford adequate underwater shielding for the storage of fuel elements prior to re-insertion in the core. The racks are of aluminum construction and contain six short tubes, each the length of a fuel assembly. The design is such that a critical array may not be achieved with elements stored in the racks. Long hanger extensions suspend the storage tubes at core level along the sides of the pool.

4. BRIDGE AND SUPPORT STRUCTURE

a. Core Support Bridge

The core support bridge is a movable structural steel span across the width of the pool, as shown in Fig. 1. 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. A locking device prevents movement of the bridge by unauthorized personnel.

b. Core Support Tower

The core support tower is a structural aluminum frame suspended vertically in the pool from the core support bridge. An aluminum guide is mounted at the rear of the tower in a position close to the reactor core. This guide provides horizontal support for neutron detection instruments suspended from the core support bridge. The core support is an aluminum angle frame bolted to the lower end of the core support tower and serves as an extension of the tower.

5. CORE

The reactor core is composed of MTR-type fuel assemblies inserted in the grid plate, together with any additional partial fuel assemblies, the control-rod fuel assemblies with built in control-rod guides, graphite reflector elements (if any), and sample irradiation stringers. The elements may be arranged in a variety of lattice patterns depending on experimental requirements. Special handling tools are used for the underwater insertion or removal of any of the above assemblies from the grid plate.

a. Grid Plate

The grid plate is a five-inch thick aluminum plate bolted to the core support angles after alignment with the drive mechanism mounting plates on the core support bridge. A series of 54 holes capable of accommodating the end fittings of the fuel assemblies are arranged in a 9 by 6 pattern in the plate. Between the larger holes are 40 smaller holes to provide additional passages for cooling water flow past the side plates of the fuel assemblies during operation. After the core is loaded with the desired number of fuel elements, all unused holes are closed with hole plugs to confine coolant flow to the core elements and experiment positions.

Adjacent to each large hole is a locating dowel extending one-half inch above the top of the grid plate. The dowels are for proper orientation of the fuel assemblies upon insertion into the plate.

b. Fuel Elements

The three distinct types of fuel elements are supported by the grid plate.

(1) Standard Fuel Element Assembly

Each standard fuel element assembly is composed of three major components: the side plates, fuel plates, and lower end fitting.

The two side plates retain the fuel plates in an approximately 3-inch by 3-inch assembly. The fuel plates are composed of enriched uranium fuel "meat" sandwiched within high purity aluminum cladding. Each fuel plate is formed to a convex shape to minimize thermal stress and is fastened to the side plates by swaging. Attached to the lower end of the fuel plate assembly is an end fitting, machined to fit into the grid plate. A horizontal rod fastened between the side plates, near the upper end of the fuel assembly, serves as a handle for the insertion or withdrawal of the element from the grid plate.

(2) Partial Fuel Element Assembly

The partial fuel element assembly is of the same physical construction as the control rod fuel-element assembly, and contains only half the amount of enriched uranium of a standard element. These elements are used for fast-neutron sample irradiations.

(3) Control-Rod Fuel-Element Assembly

Six of the elements composing the reactor core are special control-rod fuel-element assemblies. The exterior dimensions of these assemblies are identical to those of the standard and partial assemblies. These special fuel assemblies contain a centrally-located slot into which the reactor control rods are inserted. Assembled to the top of each of these elements is a shock absorber seat which cushions the fall of the control rods when these are dropped.

GENERAL SPECIFICATIONS FOR FUEL ELEMENTS

Overall dimensions	2.996 in. x 3.010 in. x 34.4 in.
Length active portion	23-1/2 inches
Type of construction	MTR-type curved fuel plates mechanically attached to side plates.
No. plates in elements	16 fueled plates per standard element (18 total) 9 fueled plates per control element 9 fueled plates per partial element.
Plate sandwich dimensions	15 mils aluminum cladding, each side 20 mils uranium fuel.
Water gap	123 mils
Fuel content (nominal)	Uranium 93% enriched in U-235 isotope 196 grams U-235 per standard element 110 grams U-235 per control element 110 grams U-235 per partial element

c. Fuel Composition

Uranium-aluminum alloy (approx. 20 w/o uranium). Alternative compositions are described in Appendix 1 and in Refs. 8 and 9.

6. COOLING SYSTEM

a. Introduction

The cooling system for the reactor consists of three systems: the primary, secondary, and purification systems. These are shown schematically in the elementary flow diagram, Fig. 6.

The general (design-center) specifications for the cooling system are listed below:

Primary flow rate	2200 GPM
Secondary flow rate	2300 GPM
Pool temperature	100°F
Primary coolant temperatures	
Entering heat exchanger	116°F
Leaving heat exchanger	100°F
Secondary coolant temperatures	
Entering heat exchanger	87°F
Leaving heat exchanger	102°F

Flow through demineralizers	100-200 GPM
Hold-up Tank Volume	33,000 gallons
Storage tank	100,000 gallons
Pool Volume (pool + stall)	120,000 gallons

The following sections briefly describe each of the above systems with the requirements of the major components of each system.

b. Primary Cooling System

The primary cooling system transfers the heat of the reactor to a heat exchanger where it is subsequently removed to the cooling tower by the secondary cooling system. Water flows by gravity from either the stall or the pool end, depending upon the core position in use, downward through the reactor core grid plate and plenum, and then to the hold-up tank. Subsequently the water is drawn from the hold-up tank by the main circulating pump and is pumped through the shell of the heat exchanger and back into the pool.

Sufficient hydraulic head is provided in the design to overcome the fluid friction in the system and give the flow requirements of the reactor as specified above. Manually operated butterfly valves are provided in each of the drain lines from the reactor to the hold-up tank. One butterfly valve adjusts the flow rate through the core. The other butterfly valve will close off the coolant flow from the pool section not in use. The valve in the return from the stall, the usual reactor operating position, is motorized and operated from the control room.

Make-up water is supplied from the Research Center filtering and demineralizing plant.

The embedded portion of the reactor piping is constructed of stainless steel to eliminate the possibility of corrosive attack on the inaccessible piping.

(1) Flow Requirements

A thermal-hydraulic analysis that forms the basis of Safety Limits and Limiting Safety System settings is given in Appendix No. 2, Section A.

(2) Plenum

The plenum provides an enclosed passage for the cooling water flowing down through the fuel elements and grid plate to the core outlet pipe at either the pool or stall operating position. It is bolted to the lower mounting flange of the grid plate. A short guide tube containing a metal bellows is attached to, and extends from, the bottom of the plenum as shown in Figure 7. The bellows, which incorporates a special flange and sealing ring insert, may be displaced a small distance vertically. This displacement permits simple mechanical disengagement of the core support structures from the reactor core outlet assemblies at either operating position.

A hinged, counterbalanced safety flapper, is attached to the side of the plenum. When properly counter-balanced (with flapper ballast weights and adjusting weights) against the suction in the plenum, the safety flapper will immediately drop open to provide a path for thermal convection cooling of the core, should the cooling water flow fall below 700 gpm. The safety flapper is closed initially at start-up with an actuating rod manipulated from the bridge.

(3) Core Outlet Assembly at the Pool and Stall Operating Positions

A core outlet structure is mounted on the floor by anchor bolts at both core operating positions. The assembly consists of a vertical length of aluminum pipe and an independent aluminum support frame. The pipe serves as an extension between the floor drain pipe and the plenum bellows seal insert. The extension pipe, called the "spool piece", is rigidly flange-mounted to a floor drain pipe.

The cam guide support frame, mounted on the anchor bolts, supports the cam to permit easy mechanical disengagement of the plenum bellows seal when the core support structure is moved by the bridge.

(4) Hold-up Tank

The hold-up tank is an underground concrete enclosure adjacent to the pump room. It has a total capacity of 33,000 gallons and is suitably shielded to protect personnel from radiation. Operating volume is normally 11,000 gallons. This tank provides a ten-minute delay of the pool water in the primary system during normal operation to allow ample time for decay of N^{16} and other short-lived isotopes in the coolant before the water enters the pump room.

When the primary pump is shut off the water levels in the pool and hold-up tank will equalize at an elevation of approximately 21 feet above the reactor core. The hold-up tank is vented to the main exhaust stack, through a solenoid-operated isolation valve and is air-purged to remove radiolytic gases.

(5) Main Circulating Pump

A motor-driven, aluminum, centrifugal pump circulates the demineralized primary water from the hold-up tank through the shell of the heat exchanger and returns it to the pool. A level controller is installed in the hold-up tank, and controls the action of an air motor operated butterfly valve on the output side of the main circulating pump. By this means, the hold-up tank water level can be maintained within defined limits, creating a stable flow rate of water through the reactor core.

(6) Heat Exchanger

The pool water is cooled in two stainless-steel heat exchanger units located in the underground pump room. The heat exchangers are of the fixed tube sheet, two-pass type, and are in series. The shell fluid is mixed by baffles.

(7) Storage Tank

A 100,000 gallon aluminum storage tank is provided near the Reactor Building. This allows for storage of pool water while work is being done inside the pool. Water is transferred to the storage tank by means of a by-pass at the main circulating pump discharge. The bottom of the storage tank is about 8 feet above the pool surface so that the pool cannot be inadvertently drained by gravity.

c. Secondary System

The secondary cooling system transfers the heat from the heat exchanger to the cooling tower where the heat is dissipated to the atmosphere. The secondary system is maintained at a higher pressure than the primary system to prevent leakage of primary coolant to the secondary system.

(1) Flow Requirements

The flow for removing the reactor heat power of 5 MW from the heat exchanger with a temperature differential of 15°F. is 2300 GPM. Inlet temperature of secondary water in the exchanger from the cooling tower is assumed to be 87°F. and outlet from the exchanger to the cooling tower is 102°F.

(2) Cooling Tower

A two-bay forced-draft cooling tower, situated near the Reactor Building, receives heated secondary water from the heat exchanger. After cooling, the secondary water is pumped back to the heat exchanger through a closed loop of steel and cast iron pipe. Make-up water is supplied from the regular water-supply main of the facility. Each of the two fans has 3 speeds - OFF, LOW, and HIGH. Normal operation of the cooling tower is based on an atmospheric wet bulb temperature of 75°F. There is provision for fan reversal to remove ice in winter.

d. Water Purification System

The purification system provides for the recirculation of pool water through demineralization equipment to maintain the required water purity.

A bypass stream is bled from the discharge line from the hold-up tank by means of a 200 GPM capacity pump. This stream is passed through anion and cation exchange columns and a filter. Each column and filter is designed for 100 GPM flow. The system is arranged such that these columns and the filter can be arranged in series, parallel, series-parallel, or individual flow sequences. The system is also designed to receive the addition of a mixed bed resin ion exchange column if required to increase the water quality output of the system.

7. CONTROL AND INSTRUMENTATION

a. Neutron Flux and Rod Control

The reactor control system is typical of those used for pool-type research reactors. The reactor is controlled by means of five thermal neutron-absorbing control rods (B_4C or Ag/In/Cd) and one stainless steel regulating rod. The five control rods are operated manually for shimming the reactor. The regulating rod may be operated in both directions either manually or automatically. Control is obtained automatically in response to power level demand settings, or by manipulation of switches by the operator. The control rods provide coarse adjustment of the neutron flux level and the regulating rod provides fine adjustment. The instrumentation provides the operator with the necessary information for proper manipulation of the controls. The following instrument channels are provided:

Counting Rate or Start-up Channel
Log-N and Period Channel
Linear Power Level and Automatic Control Channel
Three Safety Channels

These channels provide neutron flux information from shutdown levels up through full reactor power. The general characteristics of each channel are as follows:

(1) Log Count Rate channel: range 1 milliwatt to 10 watts in its most sensitive position; up to 5 MW with repositioning. The detector is an in-core fission chamber that is movable through a 4-ft. vertical distance. For providing reliable neutron information (> 2 CPS) when the shutdown flux is very low, a nominal 50-curie antimony-beryllium neutron source is supplied for use in the core. For count rates below the range of the logarithmic amplifier, a counting scaler is provided.

(2) Log-N Channel: 5 watts through 5 MW. The detector is a boron-coated ion chamber that is gamma-compensated to allow reliable neutron information to be obtained even when fission-product gamma levels are high (e.g., shortly following a shutdown from full power). Compensation gives reliable neutron information below 50 watts, or .001% of full power. This level is designated "Log-N Confidence". To assist in counteracting the effect of the over-compensation that may arise in startups that closely follow shutdowns from full power, provision is made to supply a steady but adjustable current (1.5×10^{-10} A maximum) to the Log-N amplifier input. This is termed "Bucking Current". This current is reduced to zero when Log-N confidence is attained.

(3) Linear Power (or Linear-N) Channel: range about 2kW through 5 MW, in steps set by a Range Switch. This channel is used mainly for automatic power control in the power range (e.g., 1-100% of full power). To allow its use as a linear flux channel at the lower end of its range, it uses a gamma-compensated ion chamber as detector. Fine adjustment of power is through a Power Demand dial on the reactor console.

(4) Safety Channels: range about 1 decade from 15% to 150% of full power. Three separate and independent channels are provided, two of which give the desired redundancy required for the main purpose of the safety system, viz., to scram the reactor at excessive power, with the third channel as a backup to allow continued operation if one of the others should fail. Any one safety channel will scram all control rods. As these channels are used only in the power range, their detectors are uncompensated ion chambers.

All the neutron-sensing ion chambers are located external to the core and are adjustable over a limited distance to allow their respective channels (Log-N, Linear-N, Safeties) to be standardized to the reactor thermal power derived from primary flow-rate and core ΔT measurements.

Chart recorders are provided for the Log Count Rate, Log-N, and Linear-N neutron flux channels, and for the N-16 channel.

Block diagrams of the neutron channels are shown in Figs. 8, 9.

Controls are provided for operating up to 2 rods in gang. The speed of rod withdrawal is limited to insure an inherently safe rate of reactivity insertion. In addition to a scram system, the safety instrumentation includes a rod reverse and a rod inhibitor system to maintain the reactor in a safe operating range. The control rods are coupled magnetically to electromagnets located on the rod drive shafts. A logic diagram for the scram system is given in Fig. 10.

The regulating rod is designed to have a total worth of 0.3 to 0.6% (max.) $\Delta K/K$, depending on the reflector and location in the core. This is adequate for the regulating function and has the important advantage that mis-operation of the regulating rod could never result in prompt criticality.

Logic diagrams for the rod control system are given in Figs. 11 and 12.

b. Cooling System Instrumentation

Instrumentation for the primary and secondary cooling systems is provided to measure or indicate the following:

pool - level and temperature
primary - flow-rate, ΔT (differential through core), resistivity, plenum leakage, holdup tank level, core exit temp.
secondary - maximum or minimum flow,
heat exchanger - inlet and outlet temps. in primary and secondary, differential resistivity in primary.
cooling tower - basin water level.
storage tank - high and low levels.

The cooling system elementary control diagram (Fig. 13) shows the general location and relationships of the sensors for the above measuring or indicating systems. All readouts are in the control room, with the exception of those for resistivity which are in the pumproom (upper level).

Primary flow-rate is measured via an orifice plate in the core exit line and differential pressure transmitter in the pump room. The core ΔT is derived from resistance thermometers above the core and in the core exit line, with a digital readout in the control room. Pool water level is monitored with three float switches set at 6-in. above and below, and 12-in. below, gutter level. Pool water temperature is measured with a resistance thermometer located above the core. The temperature sensors at the heat exchanger inlets and outlets are also resistance thermometers. The purpose of the differential resistivity measurements across the primary circuit of the heat exchanger is to detect leakage from secondary to primary.

Chart recorders are provided in the control room for primary flow and core ΔT.

Secondary flow-rate is modulated by an automatic control valve controlled by the temperature of the primary water leaving the heat exchanger. This control assists the operator in reducing variation in the temperature of primary water returned to the pool. Otherwise the only control is that afforded by the cooling tower fans which are under operator control in coarse steps (i.e., OFF, LOW, or HIGH speed).

The occurrence of a leak in the plenum below the core, which would bypass some of the core flow, is monitored by the plenum leak detector. In this, a differential pressure switch senses pressure in the core exit line upstream of the orifice plate and is set so that an increase in this pressure will activate an alarm in the control room.

An annuciator panel in the control room with lamps and alarms indicates conditions in the various portions of the cooling system, viz., cooling-tower fan speeds, primary pump, holdup tank level, storage tank level, secondary flow-rate, secondary pump, tower basin level, demineralizer pump.

The system includes temperature sensing elements that give an alarm in case of high water temperature. A digital switched meter in the Control Room shows water temperature in the pool, at the reactor outlet, and at the heat exchanger inlet and outlet for both primary and secondary flow.

c. Radiation Monitoring

Within the reactor building at various points, monitrons are provided to detect local increases in radiation level and to give alarms. The alarms and the levels are indicated in the control room. Duplicate monitrons are located at the bridge and serve to initiate the "evacuation sequence" for personnel in the building (see Sect. D). All monitrons are non-overloading and respond to gamma levels over a wide range.

In addition, two constant air monitors (CAMS) continuously sample the building air for radioactive particulates and print the results on a chart recorder. Each CAM gives an alarm at a preset level.

d. Drive Mechanisms and Accessories

The drive mecnanism is used for remote positioning, from the control console, of control rods, regulating rod, or fission chamber with respect to the reactor core.

The control-rod drives are mounted near the center of the core support bridge, on two aluminum plates each drilled with 88 mounting holes to permit versatility of lattice pattern. Essentially, the drive mechanism consists of a low-inertia, two phase, motor driving a pinion and its rack through a worm and two spur gear reductions.

The drive mechanism is equipped with a long drive shaft and arm which permits the mechanism to be mounted above the edge of the reactor core, with the rack extending directly over the rod being controlled.

(1) Regulating Rod and Drive Mechanism

The regulating rod drive mechanism provides the fine control of the reactor. The unit has a 24-inch stroke and a drive speed of 24 in/min. Regulating rod position is servo-controlled to maintain constant reactor power.

The regulating rod assembly consists of a stainless steel rod rigidly fastened to a long extension attached to the drive mechanism. Fine and coarse position indicators indicate the regulating rod position.

(2) Control-Rods and Drive Mechanisms

The control-rod drive mechanism provides coarse control and safety for the reactor. Control rods consist either of hollow aluminum and cadmium shells hermetically sealed and filled with boron carbide or silver-indium-cadmium (Ag/In/Cd) alloy in the form of a hollow shell. The latter type is now routinely used (Ref. 6). These rods serve as safety rods by providing a means for quick insertion. Quick insertion is attained by disrupting the magnetic hold between the drive mechanism shaft and the control rod.

The control-rod drive mechanism has a 24-inch stroke, rack speed of five inches per minute and two potentiometers as position transducers. Insertion or withdrawal of the control rods is manually controlled, or automatically controlled in the insertion direction.

The magnet assembly consists of an electromagnet attached to a long tube. The upper end of the tube attaches to the drive mechanism. Scramming or quick insertion is accomplished by de-energizing the electromagnet. The force of gravity separates the control rod from the magnet and the rod falls into the core. Upon separation the drive will automatically insert until it again touches the control rod. This feature may be bypassed if needed.

The control rods and the regulating rod are restrained and guided in slots in special fuel assemblies and in the shock absorber which is bolted to the upper end of the fuel assembly. The slots also guide the shim-safety rod during its fall when a scram or quick insertion occurs.

The control rod has a hydraulic shock absorber attached to its upper end. This shock absorber uses pool water for its fluid and decelerates the rod smoothly over the last 1-3/4 inch of its insertion.

(3) Fission Chamber Drive Mechanism

The fission chamber drive mechanism removes the fission chamber from the region of high flux. It has a 47-inch stroke. The fission chamber is housed in a waterproof can assembly which is positioned by the drive mechanism.

e. Safety System

Actual pick-up or release of the rods (by energization or de-energization of the supporting magnets) is the function of a group of instruments and associated circuitry comprising the safety system. The system is designed to shut down the reactor by immediate dropping (scram) of the control rods if any of the following conditions occur:

High power
Short period
Low flow
Unlocking of the core support bridge
Pool water level low
Guide tube lifted

In addition to the above, the safety system allows the reactor to be shut down quickly by the operator, or other personnel, by use of the manual scram stations. The scram logic diagram, Fig. 10, shows the conditions that result in dropping of all control rods.

(1) Safety Amplifier

The safety amplifier provides the amplification, control, monitoring and chamber power supply functions for the safety system. Four independent safety amplifiers are provided, in two chassis. Each safety amplifier can accept the signal current from one uncompensated ion chamber. The safety amplifiers also control, via electronic switches, the current for the control-rod electromagnets. Each magnet has its own switch. Each amplifier supplies current to a pair of high-speed relays, whose normally-open contacts are in series with those from all the other amplifiers. Opening of any one of these contacts causes all the switches to "open" thus cutting off current to all the rod electromagnets and releasing all the control rods. This action is termed a "fast" scram. This scram is reserved exclusively for the rapid reactor shutdown required by excessively high power or short period. Additional contacts in the high-speed relays are arranged to de-energize another set of relays that interrupt the A.C. supply to the magnet current power supply. The magnet current accordingly decays and results in a backup scram, called the "slow" scram. This scram mode is employed for protective actions where very rapid shutdown is not necessary - see (2) below.

Each safety amplifier chassis has an annunciator panel for indicating fault and scram location, test buttons for checking scram operation, and panel meters for amplifier output.

(2) Magnet Current Interlocks

Operating power for the safety amplifier is received from two sources through two separate power plugs. One source, the 115 volt, 1-phase regulated power line, supplies power for all circuits except the magnet current electronic switches. Power for the latter is obtained from an integral supply which is energized by 115 volt, 1-phase regulated A.C. voltage through a separate plug on the amplifier.

A.C. voltage to this magnet power supply is obtained through interlock contacts in series. The contacts represent, and are actuated by, conditions important to safe reactor operation. If a condition is unsafe, its associated contact will be open and the reactor cannot be started. Conversely, if during operation a contact should open, the reactor will be shut down immediately due to loss of magnet current. The magnet power supply interlock contacts are opened under the following conditions:

1. Manual scram buttons depressed.
2. Main cooling pump off.
3. Safety flapper open.
4. Bridge clamp unlocked.
5. Low pool water level.
6. Low flow.
7. Guide tubes lifted.

Conditions 2, 3, and 6 are bypassed at low reactor power by a backset switch on the Log-N recorder.

The magnet power supply interlock circuit is also used to obtain a rapid manual shutdown of the reactor. Contacts of a console-mounted push button are in series with all the other contacts. The circuit is also carried off the panel through several external scram stations before returning to the panel and thence to the magnet power plug. Opening the circuit with any one of these switches also removes power from the magnets.

The console-mounted annunciator provides a means of by-passing manually any or all of the following magnet power supply interlocks:

1. Main cooling pump off.
2. Safety flapper open.
3. Bridge clamp unlocked.
4. Low pool water level.
5. Low flow.
6. Guide tubes lifted.

These bypass provisions make it possible to check the interlocks during normal start-up.

(3) Fast Scrams

Under certain conditions shut down action must be initiated in a few milliseconds after an unsafe condition occurs. The two conditions requiring this action are:

High neutron density in the reactor.

Excessively large rate of increase of neutron flux.

The first condition is detected by the safety amplifier, using ionization chambers as the primary measuring element. When the neutron density exceeds a preset point, sensitive relays are de-energized in the safety amplifier and magnet current is rapidly reduced below the holding point. Contacts of the sensitive relays accomplish this function by opening the magnet current electronic switches, as described in (1) above.

The second condition is detected by the Log-N amplifier. This amplifier differentiates the signal from its ionization chamber. When the differentiation voltage exceeds a preset point, a sensitive relay is de-energized. A normally-open contact of this relay is wired into the safety amplifier in such a way that magnet current is reduced by the same mechanism as described in the preceding paragraph, namely by opening the electronic switches for all the magnets.

Total time from initiation of control action to breakaway of the rods is normally between 5 and 20 milliseconds. Operating magnet current is normally set to 2-3 times the drop out current, which is typically ?0 mA.

f. Annunciation

To provide the operator with a constant check on all the critical variables affecting reactor operation, a console-mounted annunciator is provided. The annunciator is energized continuously through the main disconnect switch. There are two lights for each annunciated condition, one red, and one green. All conditions are annunciated by means of relay or switch contacts. When the contact wired to a given point is opened, the corresponding red light is lit. When the contact closes, the red light is turned off and the green light is lit. These lights are affected only by the actual conditions of the external contacts.

The points on the annunciator are divided into three groups, according to the type of annunciation as follows:

1. Alarm Buzzer, Light. (SCRAM)
2. Alarm Horn, Light.
3. Light only.

(1) Alarm Buzzer, Light (SCRAM)

For the contacts associated with conditions causing interruption of magnet current, the scram buzzer sounds. A "SILENCE" button, located on the annunciator, is pressed momentarily to silence the buzzer. The individual light in question remains red until the condition is corrected. The lights for conditions 1 and 2 are located on the Safety Amplifier and Log-N amplifier respectively, and remain on till reset. The conditions causing annunciation are:

1. High neutron flux level, safety amplifier.
2. Short Period, Log-N amplifier.
3. Manual scram, console or external.
4. Main cooling pump off.
5. Safety flapper open.
6. Core support bridge unlocked.
7. Low pool water level.
8. Low flow.
9. Guide tubes lifted.

The alarm contacts for conditions 4, 5, 6, 7, 8, and 9, are duplicated in the magnet current interlock circuit. When bypassing a particular condition, both the condition itself and its annunciator are bypassed. The method of bypassing utilizes a phone plug and jack. Each bypass jack is located under the individual alarm lights of its respective point. To bypass, it is necessary to insert a phone plug in the jack. Conditions 1, 2, and the console manual scram cannot be bypassed in this manner.

(2) Alarm Horn, Light

The following conditions operate the alarm horn. The individual lamps are actuated as described previously.

1. Period Reverse.
2. High flux reverse.
3. Shimming required.
4. Controller power "OFF".
5. Pool water level abnormal.
6. Ion Chamber Low Voltage.
7. High radiation.
8. High core ΔT.

Contacts for condition 5 are high and low water level float switches sensing changes in water level of plus or minus 6 inches from gutter lip.

The relays for conditions 1 and 2 provide contacts to prevent rod withdrawal and to insert all shim-safety rods by a REVERSE when the period is less than 10 sec. or flux is 125% of design power.

(3) <u>Light Only</u>

The following conditions provide lights only.

1. High pool temperature.
2. Period inhibition.
3. Count-rate recorder off-scale inhibition.
4. Reverse.
5. Plenum leak.
6. Off-magnet bypass.

The alarm contacts for 1 are in a bimetallic thermostatic switch located in the pool above the core. They open when pool temperature exceeds the set point.

Contacts of point 2 are normally-closed backset switches on the period recorders. The relay function inhibits withdrawal of all rods.

The relay for point 3 provides contacts to inhibit withdrawal of all rods. This is operated when the count rate is below 2 cps or above 9800 cps, recorder power is off, or channel is switched to calibrate position. This inhibit is bypassed at .001% power on the Log-N recorder.

g. <u>Warning Lights</u>

1. A red light is provided for each magnet which lights when a rod is not in contact with its magnet.

2. A green seat light is provided for each control rod which lights when the rod is fully inserted.

3. A shim range light is provided which remains on until rod drives are withdrawn to shim range point.

4. On-Off lights are provided for all pumps operated from the console.

5. Lights are provided to show when a rod drive is at its lower limit (green) or its upper limit (orange).

8. EXPERIMENT FACILITIES

The experiment facilities listed below furnish a means for the irradiation of materials while affording protection to personnel through proper shielding.

Embedded in the concrete walls of the stall area at core level, are six beam tubes and a 4 $foot^2$ x 9 foot thermal column. Locations are shown in Figure 2. At the outer wall surface, shielding doors and plugs provide access to the experiment units. In addition, three pneumatic rabbit tubes are provided.

a. Beam Tubes

Two 8-inch diameter and four 6-inch diameter beam tubes radiate in horizontal planes outward from the reactor core. The basic tube assembly consists of an embedded stainless steel sleeve, retractable aluminum liner, and a set of interior shielding plugs of canned magnetite concrete and lead. Some features of this design are:

1. The beam tubes can be filled with demineralized water to reduce the number of shielding plugs required and to eliminate voids at the core face. Because of difficulties experienced at similar reactor installations, the design has been changed to allow water circulation through the liner. About 5 gpm of water are pumped from the pool through each liner and returned to the pool. This is called the Beam Tube Circulating System. Circulation prevents stagnant water from causing corrosion of the aluminum liner and also reduces the radiation level at the outside face of the beam tubes. Without the circulation the water becomes unduly radioactive due to a build-up of Na^{24} and Mg^{27} formed by irradiation of the aluminum liner at the core face. Rotameters at the shield face measure the flow rate to each beam tube liner. When not water-filled, beam tubes are continuously vented via a filter to the exhaust duct to prevent buildup of argon-41 (Fig. 16).

2. The annulus formed by the aluminum liner and the embedded stainless steel sleeve is open at the stall side and, therefore, filled with water. A circulating pump is used to pump about 1/2 gpm through each beam tube annulus to prevent stagnation in the area and the possibility of corrosion of the liner. This is called the Beam Tube Flushing System. Rotameters on the pump header indicate flow rates.

3. A water-tight gasket seal at the outer face of the shield eliminates hazardous and time-consuming replacement and maintainance procedures. With the gasket away from destructive radiation, conventional rubber-base materials can be used.

4. Retractable facilities afford greater working area, and lower activity in the stall area during experimental set-up or maintenance.

5. The two 6-inch beam tubes at the west end of the stall are shortened so that they terminate in a recess in the outer face of the biological shield. This "cave" facility is provided to reduce the amount of external shielding required when installing experiments such as a crystal spectrometer.

b. Pneumatic Rabbits

Two 3/4-in and one 1-1/2-in pneumatic tubes are provided to deliver sample containers or "rabbits" into the high flux region: two at the North core face, the third just above the South beamhole liners. These rabbits can be inserted or removed while the reactor is in operation by a constant exhaust system that is vented via a filter to the exhaust duct. Both tubes have automatic timing controls and shielded containers for receiving the irradiated specimens. The north rabbit tubes terminate in lead shields inside the counting rooms on the first and second floors of the Reactor Building and the south rabbit tube terminates in a lead shield on the side of the reactor. The rabbit is made of plastic.

c. Thermal Column

The Thermal Column (Figure 14) is a stacked graphite and lead assembly that functions as a neutron reflector, and slows down fast neutrons for irradiation experiments.

A steel and aluminum chamber is cast integrally with the stall wall and magnetite concrete shield at core level. This chamber is square and extends horizontally from the inside wall of the stall to the outer surface of the concrete shield. Forming a part of the embedment is a circular vertical access chamber extending from the top of the shield downward to the horizontal chamber.

The inner surfaces of the chamber are lined with boral sheet which acts as an absorber for a portion of the reflected neutrons. Stacked within the boral liner for the length of the horizontal chamber is a closely packed arrangement of graphite blocks.

Fastened over the outer face of the graphite stacking is a boral plate backed up by a lead block shield. A square opening in the shield is provided for the insertion of a lead plug. A 14-inch, 5'-6" square magnetite concrete door covers the entire exposed area of the horizontal column at the face of the concrete shield. Access to the thermal column vertical face is accomplished by using the overhead crane to open the 14-inch thick concrete door. As a safety precaution, the door is pinned against the vertical face by a safety lock bar at the top of the door. A central square opening allows for the insertion of a shielding plug. The smaller plug in the door and in the inner lead shield permits insertion of small specimens for irradiation experiments into the thermal column without the necessity of opening the full door.

The vertical portion of the thermal column is an air chamber with the opening at the top of the magnetite shield closed by a lead plug and a concrete access cover. Both the cover and plug have lifting lugs to facilitate removal by the overhead service crane.

A lead and graphite assembly forms a portion of the thermal column between the rear face of the reactor core and the inner wall of the stall. This assembly consists of an aluminum support frame bolted to the stall floor mounting pads. A lead shield is bolted to the front of the frame immediately adjacent to the reactor core, and a graphite and lead can assembly is fastened directly behind the lead shield. The thermal column is connected to the same vent system as the beam tubes.

SECTION C

REACTOR DESIGN

1. CORE PHYSICS

a. Introduction

The UCNR reactor is a 5 MW (thermal) pool-type reactor employing MTR type fuel elements, as described in Section B.5. The water reflected core will contain up to 38 fuel elements, depending on the degree of burn-up. The coolant is demineralized light water. The reactor has two operating positions; one in the open water of the pool, and the other in the stall position into which a thermal column and six beam tubes converge. The stall position is the one normally used.

In the following sections are given typical thermal and nuclear characteristics of the reactor core.

b. Table of Thermal Characteristics for Water Reflected Core

Total Power (thermal)	5,000 KW.
Specific Power (average)	960 KW/Kg U-235
Average Water Temperature:	
Cold	68°F
Design	108° F
Thermal Neutron Flux (Fuel Average)	2.5×10^{13} n/cm² sec.

c. Core Reactivity Effects

TABLE I

NUCLEAR CHARACTERISTICS

Item	Stall ΔK/K	Pool ΔK/K
Temperature effect	.0053	.0053
Control	.0025	.0025
Experimental Facilities (in core)	.0200	.0200
Beam Tubes (6)	.0080	
Burn-up + Low cross section fission products (7 days)	.0070	.0070
Equilibrium Xenon 135	.035	.035
Samarium 149	.0095	.0095
Max. excess reactivity required for 7-days operation	.0875	.0795

		$\Delta K/K$
Equilibrium Samarium 149		.0095
Temperature coefficient		$-8. \times 10^{-5}$ per ^{o}C.
Power Coefficient		$-7. \times 10^{-4}$ per MW
Control rod worths:		
Average worth/rod		.019
Total worth (5 rods)		.094
Regulating rod worth		.005
Thermal column worth		.008
Average fuel worth		1.6×10^{-4} per g U-235
Neutron lifetime		5.1×10^{-5} seconds

	Water Reflected	Graphite Reflected (3")
Cold Clean Critical Mass:		
Stall	3.45 Kg U-235	2.12 Kg U-235
Pool	3.83 Kg U-235	2.35 Kg U-235
Operating Critical Mass:		
Stall (typ.)	5.2 Kg U-235	

(1) Temperature Coefficient of Reactivity

The moderator temperature coefficient of reactivity was measured to be $-8. \times 10^{-5}$ $\Delta K/K$ per degree C for the temperature range of 20°C to 42°C. The power coefficient of reactivity in the 0-5 MW range has been measured to be $-7. \times 10^{-4}$ $\Delta K/K/MW$.

For a temperature rise of 22°C in the UCNR reactor (from cold to design water temperature), the required excess reactivity at 5 MW to compensate for temperature is calculated to be 0.53% $\Delta K/K$.

(2) Neutron Lifetime and Reactor Period

The effective neutron lifetime for the UCNR core has been calculated to be 5.1×10^{-5} seconds. A plot of the stable reactor period for step changes in reactivity is given in Appendix No. 2, Sect. B.

(3) Control and Experimental Reactivity Allowance

For reasons of safety, the total excess reactivity allowance for experiments in the core has been limited to 2.0% $\Delta K/K$.

To allow for the best control by the regulating rod, it is desired to have this rod in its most effective position during reactor operation. With this regulating rod inserted approximately half way, an excess reactivity of .25% $\Delta K/K$ would be required.

(4) Beam Tube Requirements

The UCNR reactor has four 6-inch and two 8-inch diameter beam tubes. The beam tubes converge on the core in the stall position. These beam tubes can be either flooded with light water or filled with air. The excess reactivity requirement of the four 6-inch beam tubes when air-filled is 0.35% $\Delta K/K$, and for the two 8-inch beam tubes is 0.41%.

There are in addition to the beam tubes two 3/4-in and one 1-1/2-in diameter rabbits. The rabbits run tangent to the side faces of the core.

The total excess reactivity requirement for all beam tubes and rabbit tubes is 0.80% $\Delta K/K$.

(5) Uranium-Burn-Up and Low Cross Section Fission Products

For seven days operation at a heat output of 5000 KW, an excess reactivity of 0.7% $\Delta K/K$ is required to compensate for the loss of uranium 235 atoms by burn-up, and the accumulation of low cross section fission products.

(6) Xenon 135 Poisoning Allowance

At a heat output of 5000 KW the average thermal flux in the core will be about 2.5×10^{13} n/cm^2 sec. At this flux, the excess reactivity to compensate for equilibrium xenon is calculated to be 3.5% $\Delta K/K$. It is estimated that it would require two days, with no initial xenon present, to reach 99% of the equilibrium level.

After three days of continuous operation at 5000 KW, the maximum xenon over-ride level is calculated to be approximately 7.3% $\Delta K/K$ occurring at about nine hours after shut-down.

(7) Samarium 149 Poison Allowance

Samarium 149 is a stable isotope formed during the fission process; its equilibrium level is independent of the core neutron flux. The excess reactivity required for equilibrium samarium 149 during operation is 0.95% $\Delta K/K$. With a new core this equilibrium value is closely reached in about 30 days.

(8) Thermal Column Effect

In the stall position, the presence of the graphite thermal column reduces the cold clean critical mass below that required for a totally light water reflected core. The thermal column has the effect of increasing the core reactivity by about 0.8% $\Delta K/K$.

The effect on the reactivity of the core by motion of the tower with respect to the thermal column is eliminated by placing the core support tower under slight stress against the thermal column when the core is placed in the stall operating position, and locking it in place.

(9) Critical Mass and Core Loading

The cold clean critical mass of a light water reflected core containing 10.89 grams of uranium 235 per plate, is calculated to be 3830 grams of uranium 235. Actual minimum critical mass in the pool position was determined to be 3234 grams of U-235 at initial startup.

The reactor will operate in either the open pool or stall position with different reactivity requirements, however, the effect of the thermal column in the stall position makes the operating mass in the two positions approximately the same, (assuming beam tubes are vented, i.e., air-filled).

The operational fuel requirement will depend on the type and number of experiments and the operating cycle. For a 3-week operating cycle with a typical load of in-core experiments and 2 beam tubes in operation, the total fuel requirement is about 5.2 Kg U-235.

(10) Control Rod Reactivity Effectiveness

The UCNR core has five control rods and one stainless steel regulating rod. The effectiveness of these rods, determined for a typical core and rod arrangement as shown in Fig. 15, is given below for reactor operation in the stall position.

Rod	Position	Worth % ΔK/K
Control #1	B8	1.3
" #2	B6	2.6
" #3	B4	2.2
" #4	E8	0.8
" #5	E6	2.4
Regulating rod	E4	0.5
	Total:	9.8

These worths vary somewhat, depending on core configuration and fuel distribution, in the range 9-12% ΔK/K for the total worth.

(11) Fuel Requirements

The UCNR reactor was designed on the assumption that it would operate continuously for ten-day periods between shut-downs. During these shut-down periods, additional fuel would be added to the core to compensate for the uranium 235 burn up, and for accumulation of low cross section fission products. The actual fueling cycle will vary as a function of the type and number of experiments to be performed.

The current reactor operating cycle varies between weekly and fortnightly operating periods with a duty cycle of approximately 95%. The burnup of a spent element is about 30%. At the operating power of 5 MW the long-term fuel consumption (196 gram elements) is accordingly the equivalent of 37 elements per year. As some of these would have to be control elements, the total number would be close to 40 per year of all types.

2. SHIELDING

Based on measurements at a power level of 5 MW the radiation levels at the UCNR reactor in areas normally occupied by personnel are well within limits prescribed in Part 20 of Title 10 of the Code of Federal Regulations. The shielding and the radiation levels are as follows:

a. Main Concrete Shield

Figures 2, 3, 4, and 5 show a plan view and cross sections of the main concrete shield. The main biological shielding consists of water, magnetite concrete, and ordinary concrete in varying proportions. The maximum magnetite thickness is 5.8 ft. Radiation levels from the core are reduced to below 2 mr/hr at all points outside the shield.

Lead thermal shielding is utilized in the stall to reduce gamma flux incident on the concrete, thereby relieving thermal stresses caused by radiation heating.

b. Pool Surface

The surface of the pool is 24 feet above the top of the core. This depth of water will reduce direct radiation at the surface of the pool to below 1 mr/hr.

The gamma level at the pool surface, directly over the core, is less than 15 mr/hr. The water-dispersed activityaccounts for nearly all of the gamma radiation above the core.

c. Beam Tubes and Thermal Column

The beam tubes are shielded with concrete plugs followed by a lead door at the outer end. Radiation levels at the ends of the beam tubes are less than 2 mr/hr. The concrete plugs are clad in aluminum, with vent holes to allow escape of radiolytic gases.

The thermal column shield consists of a lead block between the thermal column and the core, and the lead shield followed by a magnetite door at the outer end of the column. Radiation levels outside of the column are below 5 mr/hr.

The thermal column and any air-filled beam tubes are purged with a continuous air flow to prevent buildup of ^{41}A. This air purge is vented to the exhaust duct via an absolute filter.

d. Cooling System Components

The cooling-water core exit lines and the holdup tank are shielded to reduce radiation to tolerance levels. Shielding is installed around the cation demineralizer to provide maintenance access to the pump room during reactor operation. The butterfly valve room, as a high radiation zone, is locked shut to prevent personnel access during reactor operation.

e. Spent Fuel Elements

Spent fuel elements are transferred underwater and stored in the pool and in the canal. After an appropriate cooling time has elapsed (90 days min.), the spent elements have their end boxes cut off and are loaded under water into a cask and transported for reprocessing.

f. Handling of Radioactive Materials

The reactor pool is connected to the Hot Laboratories by a water-filled canal. This canal is used for the transfer of radioactive materials. Appropriate shields are used for movement of materials outside the water shield.

3. SUMMARY OF RADIATION LEVELS

a. During Normal Operation

Outside main biological shield	- below 2 mr/hr.
Pool surface, directly above core	- below 15 mr/hr.
Outside of beam tubes	- below 2 mr/hr.
Vicinity of thermal column	- below 5 mr/hr.

b. After Shut-down

During operation, reactor components exposed to a high neutron flux and the pool concrete become activated. In the stall, the major sources of induced radiation are the pool floor and walls, the beam tubes, the coolant components, and the thermal column. In the open end, only the neutron exposed coolant components and the pool floor and walls are expected to become activated, if the reactor should be used in this position.

After the pool has been drained, the radiation in the stall and open end can be sufficiently reduced to permit maintenance for controlled periods of time.

(1) Reactor Stall

By removing the beam tubes and graphite from the thermal column and by adding temporary shielding in front of the thermal column cover plate and over the exposed coolant components, the radiation level in the stall can be reduced to below 10 mr/hr.

(2) Open End

If a shield is placed over the coolant pipe flange and cam in the open end, the dose rate due to induced radiation is reduced to below 1 mr/hr.

SECTION D

BUILDINGS AND AUXILIARY SYSTEMS

1. REACTOR AND HOT LABORATORY

a. 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 Reactor Building is shown in Figures 19 through 25.

The building measures about 70 feet wide, 92 feet long, and 57 feet high from the beamhole 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. The controls for this door can be locked during periods of reactor operation.

b. 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.

2. HEATING AND VENTILATING

The Reactor Building ventilation system is designed to provide winter heating and summer cooling. Elementary diagrams of the ventilation system flow paths and controls are given in Figs. 16 and 17, respectively.

a. Normal Operation

The supply air fans, coils, and equipment for the reactor chamber are located in a fan room between the Reactor and Hot Laboratory Buildings. The heating boilers, compressors, and refrigeration equipment are located in the central Boiler House which serves the other buildings of the Research Center. The exhaust fans and filters for the system are located in an exhaust fan room in the northwest corner of the Hot Laboratory Building. The system is designed to maintain an inside temperature of 75°F in Winter when the outside temperature is 0°F. In Summer the system will cool the chamber to 80°F with a maximum relative humidity of 60%. A duct system distributes air throughout the chamber. Fresh air is drawn through intakes in the fan house, heated or cooled, filtered, and circulated through the duct system. No air is recirculated. The entire chamber experiences about 4 air changes per hour.

The reactor exhaust fan gathers air from the reactor building via a main exhaust duct and exhausts it via a 4-foot diameter duct to the exhaust stack located on the ridge at a high elevation northwest of the building. This stack, shown in Figure 19, discharges the air above the tree tops at an elevation of 188 feet above main floor level. There is a special air duct manifold which attaches beneath the reactor bridge and sweeps air across the reactor pool water surface above the reactor core. This 5000 cfm air sweep enters the main exhaust duct and discharges out of the stack with the building exhaust, for a total exhaust flow of 19,000 cfm.

The Reactor Building tunnel is ventilated by a jet-type nozzle which discharges heated or cooled air into the tunnel.

b. Emergency Conditions

In the event of a release of radioactive gas or particulate matter into the reactor building's atmosphere, the emergency ventilation system is put into operation. The effect of this system on the over-all ventilation system is as follows:

Immediately

1. The Reactor Building supply fan sh. :s down.
2. The holdup tank isolation valve and air purge valves close.
3. The supply duct dampers close.
4. The beam tube ventilation fan shuts down.
5. The damper in the 5000 cfm pool sweep closes.

After 7 Seconds (or when building pressure reaches 1" w.g. negative)

1. The main exhaust duct damper closes.
2. The main exhaust fan shuts down.
3. The emergency ventilation fan starts.
4. The damper in the emergency exhaust line opens.

The emergency exhaust duct is an auxiliary duct connecting the Reactor Building to the charcoal filter and the roughing and absolute filters in the fan room. The fan, at the discharge side of the filters, vents into the duct leading to the stack. The building and ventilation system is so designed that under emergency conditions building air leakage will be inward.

c. Stack Monitor

The effluent in the 4-ft duct is continuously sampled to provide indication of abnormal levels of airborne radioactive material. This is accomplished by withdrawing a 5 cfm side stream from the duct, passing this through particulate, iodine and gaseous radioactivity detectors and returning it to the suction side of the exhaust fans. The outputs of these detectors are indicated on chart recorders equipped with alarm set-points. The outputs are also repeated on chart recorders in the reactor control room. The particulate detector is a moving paper filter passing in front of an anthracene scintillation crystal. The iodine detector is a charcoal filter monitored with a scintillation detector. The gas detector is a shielded vessel containing a sodium iodide scintillation detector.

3. REACTOR BUILDING WASTE COLLECTION SYSTEMS

Reactor Building waste collection is performed by three systems:

a. Ground Water Removal System
b. Sanitary Waste System
c. Process Waste System

a. Ground Water Removal System

Due to water seepage from the fractured rock against which the Reactor Building is constructed, it is necessary to collect the water in a system of perforated drains located around the building and pool structures. These drains collect into a single drain that releases effluent to the Research Center storm water system.

Since the drain system also has lines around the Reactor Building hold-up tank, a french well was installed at the corner of the hold-up tank to allow sampling of this low point of the system. With this arrangement and a routine sampling procedure, any leakage of the hold-up tank can be detected.

b. Sanitary Waste System

Sanitary wastes collect into a main sanitary drain and pass through a deep trap before draining from the Reactor Building. The deep trap is designed to maintain air containment of the building even when the sanitary facilities are drained. The Reactor Building sanitary waste system drains to the Research Center sanitary waste system.

c. Process Waste System

The process waste system discharges effluent to either the Research Center non-radioactive and chemical waste system or the radioactive waste system, depending on the activity level of the waste. The system works as follows:

All Reactor Building liquid wastes other than ground water and sanitary wastes drain to two 400-gallon sumps located on the lower level of the Reactor Building pump room. From these sumps liquid waste is pumped to the radioactive waste treatment facility storage tank located in the Hot Lab.

4. WASTE DISPOSAL

The following waste disposal systems are provided for the Research Center:

a. Storm Water System
b. Sanitary Waste System
c. Non-radioactive and Chemical Waste System
d. Radioactive Waste System

A schematic diagram of these systems is shown in Figure 26.

a. Storm Water System

The storm water system is a gravity drainage system discharging to Indian Kill Pond. Maximum use is made of existing natural drainage for ground run-off. A storm sewer main collects storm water from building roof drains and paved area collection scuppers.

b. Sanitary Waste System

The sanitary waste system is a gravity drainage system discharging directly to the Sterling Forest sanitary sewage disposal system. This system handles all wastes from the sanitary plumbing in the various buildings on the site.

c. Non-radioactive and Chemical Waste System

The non-radioactive and chemical waste system is a gravity drainage system discharging to Indian Kill Creek down-stream from Indian Kill Pond. This system handles chemical waste from laboratory sinks, cup sinks, benches, and process equipment in the Engineering, Research and Administration, and Service and Boiler Buildings. All wastes flow through a sample pit.

Non-radioactive waste, i.e., cooling water, steam condensate, and decontaminated process waste, from the Reactor and Hot Laboratory Buildings discharges into either of two 5000 gallon capacity tanks located 300 feet from the Nuclear Laboratory. These tanks are operated on a collect-hold-sample philosophy. Such a system provides a positive method of preventing accidental discharge to off-site stream because the liquid collected in the tanks is sampled and analyzed prior to discharge to the non-radioactive and chemical waste system. If the activity level is higher than can be tolerated, the liquid is pumped to the 7200 gallon radioactive waste storage tank.

d. Radioactive Liquid Waste Treatment System

The Radioactive Liquid Waste System is designed to prevent the off-site disposal of wastes containing radioactivity or radioactive isotopes higher than the levels prescribed by Title 10, Part 20 of the Code of Federal Regulations or in Technical Specifications. The handling and treatment facilities combine storage, evaporation, ion exchange, and recycle (if required), to accomplish this objective. Concentrated liquid wastes containing the radioactivity produced are placed in a solid form by using such wastes to prepare concrete. This solid material is packaged in approved containers for shipment to an approved burial ground.

All radioactive liquid wastes resulting from Reactor or Hot Laboratory operations or from radioactive operations in other buildings are collected in a 7200 gallon stainless steel tank located in a separate cell under the main floor of the Hot Laboratory Building. All hot drains in the Hot Lab and drains from vent and off-gas headers also tie into this tank. There is access to this cell via a shielded 3' x 3' hatch. Waste from this collection tank is evaporated, with the distillate being sent after sampling and analysis to the chemical waste system.

Radioactive wastes produced as a result of research and development with nuclear fuel are processed and packaged within a particular hot cell. Such wastes are shipped, in approved containers, to an approved burial ground.

5. EMERGENCY POWER AND MOTOR CONTROL CENTER

a. Emergency Power

A gasoline-motor-driven electrical generator of 50 kw capacity provides emergency power automatically in the event of a utility electrical power failure.

The following Reactor Building equipment receives electrical power from this generator:

(1) Portions of the Reactor Building lighting
(2) Reactor control console
(3) Reactor Building exhaust fan (at half speed)
(4) Reactor Building supply fan (at half speed)
(5) Reactor Building motor-operated doors
(6) Emergency ventilation system
(7) Beam tube ventilation fan
(8) Beam tube flushing pump
(9) Swing-type airlock door controls
(10) Gasoline pump from storage tank to emergency generator
(11) Hot Lab exhaust fan at 1/2 speed or stand-by Hot Lab fan at 1/2 speed.
(12) Certain electrical receptacles for mobile equipment (radiation monitors, etc.)

The purpose of this generator is to supply emergency power for minimal operation of critical systems in the Reactor Building. The Reactor will never be operated, even at low power levels, on emergency electrical power. Sufficient gasoline storage has been installed to guarantee a six day supply for the emergency generator. The storage tank is 2000 gal capacity.

b. Motor Control Center

The main motor control center, in addition to controlling the numerous functions of equipment under normal operation, controls the operations and sequences necessary for the use of the emergency ventilation system and the transfer of power loads from the normal supply to the emergency generator supply. The functions of the motor control center that are of interest in this section are those required to control the latter two emergency operations. Operation of the emergency ventilation system was previously described in sub-section 2b of Section D.

The situation described here are those in which a power failure occurs during the following operating conditions:

(1) Simultaneously with the start-up of the emergency ventilation system.

(2) After the initial timing sequence of the emergency ventilation system has been completed.

(3) Restoration of normal line power after the emergency ventilation system has started.

Condition 1

If a power failure occurs in the same instant that the emergency ventilation system is put into operation, either automatically or manually, the loss of electrical power for the few seconds required to start the generator (approximately 5) will completely close the dampers in the air ducts of the entire ventilation system. These dampers have all been designed to be fail-safe, air-tight dampers so that on loss of either electrical power or pneumatic power, they will close. The closing time for these dampers is less than 3 seconds. After the emergency electrical generator has come up to speed and the automatic transfer switch in the motor control center has transferred power from the normal bus to the emergency bus, the emergency ventilation system will then be energized. The signals which initiate the emergency ventilation system are of a type that will not reset automatically after recovery of electrical power. There is no possibility that a power failure, coinciding with a need for the use of the emergency ventilation system, could return the ventilation system to normal, and thus exhaust large quantities of untreated air into the general atmosphere.

Condition 2

If electrical power from the local power company should fail after the initial 7 second timing sequence for the emergency ventilation system has been completed, it might be possible for the system to reset itself when the emergency electrical generators assume the load, and thus repeat the 7 second cycle. To eliminate this possibility, a manually reset relay is connected to the timer which controls the initial operation of the main exhaust fan for the 7 second period. In this way, the additional operation of the main exhaust fan is prohibited once the timer has cycled through its planned sequence. Loss of normal power and its subsequent replacement by emergency power would have the following effect:

(1) The entire system including the dampers would stay shut down for the period of time necessary for the emergency generator to assume the load.

(2) Once the load is assumed by the emergency generator, the system would continue to function as it had immediately prior to the power failure, i.e., the main dampers stay shut.

Condition 3

The emergency electrical generator and its associated automatic transfer switch are interconnected in such a way that upon resumption of normal power from the local power company, the electrical load is switched back to the normal bus automatically. The possibility of the emergency ventilation system being programmed automatically through the initial 7-second phase described in Sub-section 2b of Section D, with the resultant exhaustion of large quantities of untreated air, is prevented by the need to reset the emergency ventilation system manually.

6. FUEL STORAGE

New and "cold" fuel elements are stored in a vault-type room in critically-safe storage racks. This vault is of concrete block construction, with a locked hollow metal door providing the only access. The vault is ventilated, and is located in the rear corner of the maintenance shop between the Reactor and Hot Laboratory Buildings.

The storage vault measures approximately 12 ft long x 9 ft wide x 13 ft high. Fuel elements are stored upright in two separate metal racks located on opposite walls of the vault. The racks are fixed in place and separated by a minimum distance of 6 ft. Each rack will hold 50 elements in two rows, with 25 elements in each row. The rows are separated by a minimum distance of 6 inches. Rack separators maintain a minimum spacing of 2 inches between elements.

The vault is so located that flooding by water is extremely remote. If water should enter the vault, it would drain out through the vault ventilators which are located only a few inches above floor level. However, in the unlikely event that complete flooding of the vault did occur, the fuel storage spacing and geometry described above will guarantee that criticality could not be achieved. This spacing and geometry is based on ORNL data given below.

ORNL measurements (Ref. 1) have shown that two-row slab shaped arrays of ORR and BSF fuel elements remain subcritical under conditions of complete moderation and reflection by water. During these tests, the individual fuel elements and the rows were spaced at an optimum distance of 0.2 inches. Step additions of elements were made to the two-row array and increases in the source neutron multiplication were observed for all additions up to 17 elements per row; however, further additions to this 34-element array had no appreciable effect. The final array had twenty-four 200-gram elements in the center and fourteen 168-gram elements on each end. Other ORNL measurements have shown that a completely water moderated and reflected square array of ORR elements at a spacing of 1.25 inches between elements, requires twenty-eight 168-gram elements with six 200-gram elements in the center of the array and twenty-seven 140 gram elements on the outside edge of the array to become critical.

Thus, the fixed fuel element spacing maintained by the vault storage racks is, from a criticality standpoint, conservative and will be safe under all credible conditions. The possibility of interaction between elements stored in the two racks is ruled out because of the large (6 feet) distance between the racks.

b. Irradiated Fuel

Irradiated and spent fuel is stored, prior to shipment for reprocessing, in underwater racks located in the floor of the pool. Each rack is rectangular with spaces for 16 elements. The 6-in center-to-center spacing of element receptacles guarantees sub-criticality. Holes in the bottom of the receptacle allow natural circulation of water through the elements.

SECTION E

REACTOR OPERATION

1. FACILITY OPERATION AND STAFF

a. General

The Union Carbide Nuclear Reactor facility provides irradiation and research facilities for the Medical Products Division of the Corporation.

b. Site Management

The administrative control and operating responsibility for the Site rests with the Vice President and General Manager, Nuclear Products. These responsibilities include:

(1) Personnel health and safety.
(2) Safe and efficient operation of the reactor and other facilities at the Site.
(3) Maintenance of all Site facilities.

To assist the General Manager in these functions as they pertain to the reactor, a permanent staff is provided as shown in the organization chart, Fig. 18.

The functions of the various units of this organization as they relate to reactor operation and safety are outlined below.

c. Reactor Operations Group

This group includes licensed reactor operators and trainees, and the Reactor Supervisor. In direct charge of the operations of the reactor is the Reactor Supervisor. This position requires a graduate engineer or physicist with several years of previous supervisory and operating experience in reactor operations. A federal Senior Reactor Operator license is required for assumption of full duties and responsibilities in this position.

The duties and responsibilities of the Reactor Supervisor include:

(1) The safety of all members of the Reactor Operations Group, experimenters using the reactor, and visitors to the facility.

(2) The preparation of the operating procedures, emergency procedures, operating rules, reactor building evacuation procedures and their inclusion in a larger site emergency and evacuation plan.

(3) The operation of the reactor in accordance with all applicable Federal and State codes, including the reactor license and its Technical Specifications, concerning such matters as radiation protection, the discharge of radioactive waste, and the safeguarding of special nuclear materials.

(4) The final authority for the insertion of approved experiments into the reactor, after considering the handling problems during insertion and removal.

(5) The maintenance of the reactor with special regard to the control system and other features of the reactor concerned with the safety of operation.

(6) The investigation and reporting of unusual or unexpectex incidents and the recommendations of measures to prevent their recurrence.

(7) The supervision, training, selection, and requalification of the Reactor Operators.

The Reactor Supervisor is assisted by a Chief Operator, who will also be licensed, for the day-to-day scheduling of reactor operations and maintenance activities.

The day-to-day operation of the reactor will be performed by the reactor operators, licensex in accordance with Federal Regulations. The minimum qualification of these operators is a high school diploma. The minimum complement of operators is two per shift plus the additional needed for additional daily operations and for vacation relief.

d. Health, Safety and Environmental Affairs Group

The radiation safety for the site, including the research reactor, is the responsibility of the Manager of this Group. This position requires a graduate engineer (or physicist) qualified in health physics. His duties and responsibilities include:

(1) The instruction of all persons working with radiation equipment or radioactive materials in the hazards associated with them, and the safe methods of handling, operating, and using such equipment and materials.

(2) The proper instruction of all persons working with radiation equipment or radioactive materials in the purpose and use of any protective and monitoring equipment provided.

(3) The insuring by means of routine and special surveys that all hazardous conditions are known. This will include surveys to verify the adequacy of decontamination

and to establish permissible working times in the event of accidents, spills, and when necessary, for maintenance operations.

(4) The insuring by means of radiation surveys that any space normally occupied by persons not primarily engaged in radiation or associated work is not subject to excessive radiation levels.

(5) Monitoring and controlling the discharge of radioactive wastes and effluents so as to be in accordance with Federal and State limits and as low as reasonably achievable below these limits. Preparing required reports of such efforts.

(6) The maintenance and calibration of all fixed and portable health physics instruments, including stack and off-site monitoring equipment.

(7) The maintenance of records of accumulated radiation exposure of all personnel assigned permanently or temporarily to a controlled area.

The group includes as a minimum a Health Physics Supervisor, who is a graduate engineer qualified in health physics, and a number of survey technicians trained in health physics.

e. Manager, Nuclear Operations

The Manager, Nuclear Operations is responsible for the safe and effective operation of both the Reactor and the Hot Lab (or Radiochemical Processing) facilities. This is also the minimum level for management approval of procedures, new experiments, and modification following review by the Nuclear Safeguards Committee. The position requires a graduate engineering degree, or equivalent. A federal reactor operator license is desirable, but not required.

f. Nuclear Safeguards Committee

(1) Introduction

Reporting to the General Manager is a Nuclear Safeguards Committee whose duty is to provide independent review and audit of reactor facility operations.

(2) Composition and Qualifications

The Nuclear Safeguards Committee shall be composed of a minimum of 5 members. The members shall collectively provide a broad spectrum of expertise in the appropriate reactor technology. Members and alternates shall be appointed by and report to the General Manager. They may include individuals from within and/or outside the operating organization. Qualified and approved alternates may serve in the absence of regular members.

(3) Committee Rules

The committee shall function under the following operating rules:

a. Meetings shall be held not less than semi-annually or more frequently as circumstances warrant consistent with effective monitoring of facility activities.

b. A quorum shall consist of not less than one half the membership, where the operating staff does not constitute a majority.

c. Sub-groups may be appointed to review specific items.

d. Minutes shall be kept, and shall be disseminated to members and to Site management within one month after the meeting.

e. The Committee shall appoint one or more qualified individuals to perform the Audit Function.

(4) Review Function of the Committee

The following items shall be reviewed by the review group or a subgroup thereof:

a. Determinations that proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question.

b. All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

c. Tests and experiments in accordance with Section 2c below.

d. Proposed changes in technical specifications, license, or procedures.

e. Violations of technical specifications or license.

f. Violations of internal procedures or instructions having safety significance.

g. Operating abnormalities having safety significance.

h. Audit reports.

i. Reportable occurrences.

(5) Audit Function of the Committee

The audit function shall include selective (but comprehensive) examination of operating 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. The following items shall be audited:

a. The conformance of facility operations to the technical specifications and applicable license conditions, at least once per calendar year (interval not to exceed 18 months).

b. The retraining and requalification for the operating staff, at least once every other calendar year (interval not to exceed 30 months).

c. The results of actions taken to correct deficiencies occurring in reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval not to exceed 18 months).

d. The reactor facility Security Plan and implementing procedures at least once every other calendar year (interval not to exceed 30 months).

Deficiencies uncovered that affect reactor safety shall immediately be reported to the Manager, Nuclear Operations. A written report of the findings of the audit shall be submitted to Site Management and the Nuclear Safeguards Committee members within 90 days after the audit has been completed.

2. EXPERIMENTAL PROGRAM AND POLICY

a. Experimental Program

The experimental program (Ref. 7) at the Union Carbide Nuclear Reactor consists of work planned by the resident staff and by members of research and development groups in the Corporation.

The following list shows the general types of work which may be done:

(1) Production of Radioisotopes
(2) Neutron Activation Analysis
(3) Delayed Neutron Assay of Fissile Materials
(4) Neutron and Gamma Radiography
(5) Radiation Chemistry Studies
(6) Neutron Diffraction Studies
(7) Neutron Activation and Transmutation of Materials
(8) Pile Oscillator Measurements

b. Policy

The responsibility for operations in the Reactor Building, including the reactor and its auxiliary equipment rests with the Reactor Supervisor and Reactor Operators. This group also controls the use of much of the experimental equipment and is responsible for the insertion and removal of experiments into and from the reactor. They will assist experimenters in planning experiments and in obtaining the necessary prerequisite approvals for new types of experiments. The approval of the Reactor Supervisor, or his designated alternates, is required before previously-approved experiments or classes of experiments may be inserted into the reactor.

c. Experiment Review and Approval

(1) All new experiments or classes of experiments that could affect reactivity or result in release of radioactivity will be reviewed by the Nuclear Safeguards Committee prior to their insertion in the reactor. As guidance in evaluating the safety of experiments, ANS Standard 15.6 will be used. The review will assure that compliance with the license, technical specifications, and regulatory requirements is attained.

(2) The experimenter will prepare a written plan for the experiment, including all safety considerations related to its operation and subsequent disposal, and submit this plan for review to the Nuclear Safeguards Committee (1 copy for each member). One copy of the experiment approval sheet will accompany the Committee Secretary's copy.

(3) The Committee's review will be documented in Minutes, and any comments or limitations will be noted on the experiment approval sheet before this is sent to Manager, Nuclear Operations, for written approval.

(4) Substantive changes to approved experiments may be made only after the changes are reviewed by the Nuclear Safeguards Committee and approved as above. Minor changes, that do not affect the safety of the experiment, may be approved by the Reactor Supervisor.

3. HEALTH PHYSICS MONITORING

a. General

The standards for radiation exposure will be those contained in Code of Federal Regulations, Title 10, Part 20 (10CFR20), and in the New York State Industrial Code, Rule No. 38. In addition all absorbed doses will be maintained "as low as reasonably achievable" below these standards. To this end, surveys of radiation and contamination levels will be conducted on a regular basis. Warning signs, labels, personnel monitoring, and records will be in accordance with these regulations.

b. Personnel Monitoring

New employees are given an indoctrination in radiation protection covering federal and state regulations, monitoring devices, warning signs, female employee risks, exposure records, protective equipment, and radiation alarms.

Each radiation worker is assigned a film badge that is replaced periodically and is to be worn, together with a dosimeter, when in a radiation area.

Special dosimeters and film badges are vailable for situations where dose spatial patterns are unusual, e.g., wrist and finger badges or dosimeters.

c. Radiological Environmental Monitoring

An environmental monitoring program was started in 1957 and has continued with some changes in details to the present. The current program is summarized as follows:

(1) Airborne Activity and Direct Radiation

Two sampling stations for radioiodine (charcoal canister) and for particulates are operated continuously, both located downwind of the prevailing wind direction, one near the site boundary and the other at the nearest public habitation. Samples are collected every seven days for analysis. For direct radiation monitoring, the stations also have gamma dosimeters that are read at 1-3 month intervals.

(2) Ingestion

Water samples are taken monthly for gross-beta analysis from five separate locations, viz., Indian Kill inlet and outlet, Warwick Brook, Sterling Lake, and the Ramapo River.

These measurements are supplemented by the extensive environmental monitoring program conducted by the New York State Department of Health on water, milk, and on certain flora and fauna.

d. Evacuation Signal

A distinctive sounding evacuation horm for the reactor building can be activated manually, either from the inside or the outside of the reactor building, or automatically from a monitron located under the reactor bridge.

4. RADIOACTIVE WASTE DISPOSAL

a. Solid/Solidified Waste

No radioactive solid waste is disposed of at this Site. All such waste is transported to a federal or state-approved burial ground.

b. Liquid Waste

As described previously (Sect. D,4), all radioactive aqueous wastes are collected and then evaporated. The clean distillate is sampled before discharge to the non-radioactive chemical waste system, and then only if its activity concentration (including dilution) is below that specified in 10CFR20, Appendix B, Table II, Col. 2. This concentration is an average over no more than one month.

c. Airborne Effluents

Airborne effluents from routine reactor operation are, as described previously (Sect. D,2), discharged from the stack which is at an elevation of 995' above sea-level (or 188 ft above main floor level). An analysis of the concentration of the major radioactive constituent, argon-41, is given in Appendix 2, Sect. C,1. The dose resulting from this concentration is well below natural background dose levels.

Also as described before, the effluent is continuously sampled by the stack monitor which is set to detect abnormal increases in iodine, particulate, or gaseous activity. In the event that prescribed levels are exceeded, the reactor is shut down and the emergency ventilation system activated. In this event, the reactor building is placed under negative pressure with the result that all air leakage is inwards. The small flow of exhaust air from the emergency fan is sent through absolute (HEPA) and charcoal filters before discharge to the stack.

5. EMERGENCY PROCEDURES

An Emergency Plan is available as a separate document. This plan has been formulated following the guidelines of A.N.S. Standard 15.16 "Standard for Emergency Planning for Research Reactors" (December 1977 Draft) and N.R.C. Regulatory Guide 2.6 as they apply to this site. For procedures see Sect. 7 below.

6. OPERATOR TRAINING AND REQUALIFICATION PROGRAM

a. Selection

Criteria to be considered in the selection of operator trainees will include, but not be limited to, emotional stability, maturity, manual dexterity, ability to become licensed, and attitude towards safety. The minimum qualification will be a high school diploma or equivalent.

b. Training Program

The training program will consist of the following:

(1) Lectures on the fundamentals of nuclear physics and engineering. (Structure of the atom, radioactivity, fluid flow, etc.)

(2) Lectures on basic reactor concepts, such as, period, xenon buildup, temperature coefficient, reactivity, etc.

(3) Training in the use of auxiliary reactor equipment, such as, heat exchangers, demineralizer, cooling tower, etc.

(4) Lectures on the theory and operation of the safety and control system and components.

(5) Lectures on the basic concepts of health physics, the health physics equipment, and the procedures to be employed under emergency conditions.

(6) Familiarity with the experimental facilities in the reactor and the safe and proper operation of these facilities.

(7) Instruction in the importance and use of standard operating procedures for the reactor.

(8) Practice operation at the reactor controls under direct licensed operator supervision.

(9) Test questions and quizzes to gauge progress.

c. Operator Requalification Program

A requalification program in accordance with 10CFR Part 55, Appendix A, is available as a separate document.

7. OPERATING PROCEDURES

Written detailed procedures for the activities listed below are collected in a Procedures Manual that is readily available in the Control Room to all operators. Formal procedures are included for making and authorizing revisions to the contents of the Manual. All procedures are reviewed by the Nuclear Safeguards Committee.

a. Startup, operation, and shutdown of the reactor.

b. Fuel loading, unloading, and movement within the core.

c. Routine maintenance of major components of systems that could affect reactor safety (e.g., containment, emergency ventilation system, control rods, emergency generator, etc.).

d. Surveillance tests and calibrations related to reactor safety (e.g., control instrumentation channel tests, control rod worth, primary flow, physical security system checks, scram time, etc.).

e. Surveillance tests of radiation monitoring equipment (e.g., stack monitor, constant air monitors, area monitors, etc.).

f. Personnel radiation protection (indoctrination, training, protective equipment, surveys, etc.).

g. Conduct and review of experiments and irradiations.

h. Implementation of the Emergency Plan and the Physical Security Plan.

SECTION F

SITE GEOGRAPHY

This section is a short review of the salient geographic features of the reactor site and forms a basis for the next section which is a discussion of hazards and possible effects upon the surrounding population.

1. LOCATION

The reactor site is located in Sterling Forest, 3-1/4 miles north-northwest of Tuxedo Park, Orange County, New York, and is approximately 1500 feet southwest of Indian Kill, a small stream flowing southeast for a mile and a half to the Ramapo River. 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 site, itself, (Figs. 20, 21, 22) consists of 100 acres of land, owned by Union Carbide Corporation.

2. TOPOGRAPHY AND GEOLOGY

Topography and geology are described in the report, "Topography and Geology of the Site", Appendix 5. The terrain is rolling to rough and varies from lake and peat humus swamp to rocky outcrops. The site elevation varies from 700 to 1000 feet above sea level. Peaks or knolls adjoining the site extend upward to a height of 1200 feet. The area is largely covered with second growth oak and maple, and miscellaneous deciduous trees with undergrowth of laurel and rhododendron.

3. CLIMATE

Weather conditions at the UCNR site are described in "Climatology of the Tuxedo Park-Sterling Forest Area", Appendix 6. The climate of the Sterling Forest area is predominantly influencex by air mass movement and prevailing winds from an inland direction. 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 [illegible] to exert an effect on the weather of the area.

The average annual precipitation in the area is 44 inches.

Data taken at the reactor site show that the predominant wind directions are from the southwest and west, and that combined with periods of calm, they make up about 64% of the observations. Winds from the north and the northwest comprise only about 12% of the yearly total. Wind speeds are most frequently in the 1-3 or 4-12 mph ranges.

4. POPULATION

The reactor is located in a very thinly-populated area. It is within a 22,000 acre woodland area, called Sterling Forest, which is owned by a private company, the Sterling Forest Development Corporation, a subsidiary of City Investing Corporation. Sterling Forest contains three residential areas, several small research centers and a conference center. These developed areas make up a total of less than 1,500 acres. The remainder of the land is undeveloped. Adjoining Sterling Forest to the east is another large undeveloped area which is a parfx 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 III and IV show the population centers and population distribution in the vicinity of the site. The population figures are for permanent residents and in Table IV, 3.5 persons per single family dwelling is assumed. Accurate figures on any increase in population during the summer months is not available.

Table III

Major Population Centers Within 10-Mile Radius of UCNC Site

Town	1980 Population Estimate	Distance from Site
Suffern, N.Y.	8,695	9.5 miles S.E.
Warwick, N.Y.	20,851	6.0 miles W.
Sloatsburg, N.Y.	3,134	6.0 miles S.E.
Florida, N.Y.	1,674	9.0 miles N.W.
Monroe, N.Y.	11,809	7.5 miles N.
Chester, N.Y.	6,208	10.0 miles N.
Hilburn, N.Y.	1,258	8.5 miles S.E.
Ringwood, N.J.	10,393	6.5 miles S.
Tuxedo Park, N.Y.	3,458	3.25 miles S.S.E.
Greenwood Lake, N.Y.	2,262	2.5 miles S.W.
Mahwah, N.J.	10,800	10.5 miles S.E.
Ramapo, N.Y.	83,630	7.5 miles

Table IV

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

SECTION G

HAZARDS

1. GENERAL

The hazards which may be associated with the operation of the 5 MW UCNR pool-type research reactor are discussed below. A number of accidents and their consequent hazards are considered, and the results of the Design Basis Accident are treated in detail.

The accidents discussed are as follows:

1. Natural phenomena - the effects of windstorms, earthquakes, etc.
2. Minor accidents - the effects of faulty operation by personnel, or equipment malfunction.
3. Maximum start-up accident - the worst accident resulting from malfunction and misoperation at the time of start-up.
4. Credible serious accidents - severe accidents which can be postulated as occurring under foreseeable circumstances.
5. Design Basis Accident - an accident that exceeds in severity the most severe credible accident.

2. NATURAL HAZARDS

a. Windstorm

Since the reactor building is built into the side of a hill, and is ruggedly built, windstorm damage to it can be precluded. A strong windstorm could damage the cooling tower, but this would not introduce a radiation hazard.

b. Lightning

Lightning could interrupt the power to the site, or cause a fire in the surrounding forest. In case of power failure, the reactor will automatically shut down. An emergency power system will supply power to critical equipment.

c. Fire

The location of the building, and its concrete and steel construction should protect the reactor from damage due to a forest fire in the area. It is not likely that a radiation hazard would result from a forest fire.

d. Flood Hazard

The only probable type of flood in the area is a flash flood. The building is designed to withstand these without damage. However, even if dirty water should enter the pool, there is no immediate hazard. Of necessity, the pool will be cleaned out if this happens.

e. Earthquake

This area has a long record of freedom from violent earthquakes; no strong earthquakes have occurred since 1878. The reactor pool is placed in very firm hard rock. A violent shock would cause the reactor to fail safe from power failure or cause the rods to fall from their magnets. The worst hazard from an earthquake would be loss of pool water through a crack in the concrete and a simultaneous break in the pool water seal. Because of the structural strength of the pool wall, this occurrence is doubtful. The reinforced concrete pool walls should retain their integrity at least up through an intensity VII shock. The combination of earthquake rarity, rock foundations and structural design of the pool renders this hazard remote. Further discussion is given in Appendix 5.

A shock of lesser severity (e.g., VII) could conceivably cause a leak or break in one or more of the four primary cooling system pipes, leading to draining of the pool if there were no intervening valves. The shortest time to drain the pool entirely is 8 mins. (Appendix 3). After this time the decay power in the fuel would be only 1% of the operating power. Work done by Wett (Ref. 2; Ref. 4, p. 85) on temperatures of irradiated ORR fuel elements in both stagnant air and partially submerged in water indicates the maximum core temperature considering minimum drainage time would be about 950°F. The exposure of the core would not result in fuel melting. An added safety feature to reduce the temperature of the exposed core is the existence of two manually-operated spray nozzles located at the top of the pool at both reactor operating positions which will direct about 100 gpm of water on the core. The manually controlled valve for the nozzles can be operated from outside the Reactor Building.

If all primary cooling water were to be released from the pool via severed coolant lines, the depth of water, after filling the underground pump room and holdup tank, would be only about one foot on the reactor building ground floor and therefore would be expected to be retained within the building. As fuel melting is absent, this water would be only mildly contaminated with short half-life radioactive material.

3. MINOR ACCIDENTS

Accidents are considered minor if their results are less severe than those identified under "Credible Serious Accidents" in Sub-Section G-5.

a. Loss of Ventilation in Beam Tubes

Normally, the beam tubes and access hole in the thermal column are ventilated to prevent the buildup of air containing radioactive gases. If the ventilation system fails, radioactive gases may contaminate the room atmosphere.

The major contribution to air activity in the beam tubes is the A^{40} (n,γ) A^{41} reaction. The A^{41} activity reaches its saturation level after periods much longer than the A^{41} half life (110 minutes).

If the thermal neutron flux at the end of the beam tubes is 3×10^{13} neutrons/cm^2-sec and is assumed constant over the entire length of the tube, the activity of the air in the beam tubes due to A^{41} is calculated to be 1.26×10^{-4} curies/cc of air. The volume of the air in an eight inch beam tube, plugged except for 26 inches of its length, is 21,400 cc. This yields a saturation activity of 2.70 curies in the beam tube air. If the cover were removed from the beam tube and the activity were uniformly dispersed in the building atmosphere (volume = 7700 meters^3), an A^{41} activity of 3.5×10^{-4} uc/cc of air would be produced. This would be over the permissible limit of 2×10^{-6} uc/cc listed in Table I of Appendix B of 10 CFR 20.

To reduce the possibility of contamination of the building air, the reactor is equipped with a warning light that operates whenever the beam tube ventilation system fails. If such a failure occurs, the beam port will be kept closed until sufficient time has been allowed for the decay of the A^{41} to safe levels, after the reactor is shut down.

c. Loss of Ventilation over Pool

Failure of the special pool-top ventilating system should not be a health hazard. The two sources of airborne activity in this zone under normal conditions are water-dispersed activity released as the pool water evaporates, and radioactive gases dissolved in the pool which come out of solution.

d. Loss of Fuel Cladding

A defective or damaged fuel element could be loaded into the core, or cladding could be corroded or eroded from an element during reactor operation. This would cause an increase in water activity by fission products ejected by recoil from the bare fuel alloy during reactor operation, and a relatively minor and slow additional increase in activity by fission products released as the fuel alloy corrodes. This activity, while still low, will be detected by regularly scheduled water sampling. In addition, there is continuous monitoring of the building air and of the exhaust from the stack.

e. Loss of Pool Water

Pool water can be lost accidentally through several channels with varying results:

(1) Accidental Draining

The pool water can drain into the hold-up tank through a faulty valve, or through a valve accidentally left open. Since the core is normally cooled by water flowing into the hold-up tank by gravity, and the tank is normally one-third full of water, this will drop the water approximately 3.7 feet. This is not dangerous. If the reactor is operating, it will be scrammed automatically when the water level drops.

(2) Damaged Beam Tube

If an empty beam tube were sheared by a falling object, the water level could drop, possibly exposing the core, and resulting in high radiation levels inside the building. This accident can occur only with unplugged and unbolted cover plates, since a plugged and bolted tube can easily withstand pool water pressure, with some minor leakage occurring around the outer plug. To prevent this type of accident, the reactor will be moved to the open end of the pool whenever the beam tubes are unplugged and unbolted. The reactor will not be returned to the stall until the beam tubes are bolted.

If a beam tube should leak unnoticed a six-inch drop in pool water level will actuate the pool water low-level alarm. It takes less than ten minutes to move the reactor to the open end of the pool and lower the gate, during which time the water level will drop a negligible amount.

(3) Pumping

The water can be pumped from the hold-up tank, and therefore from the pool, into the storage tank. This cannot be done accidentally, since it requires that several valves and the pump be operated. An alarm will sound when the pool water level drops six inches, and the reactor will shut down automatically if the level drops one foot.

f. Coolant System Failure

Power failure to the coolant pump will scram the reactor. Should water returning to the pool be accidentally shut off, water will flow from the pool into the hold-up tank for between five and ten minutes. This will cause the water level in the pool to drop. When it drops one foot, the reactor will shut down automatically. After the hold-up tank is full, the safety flapper will automatically open, allowing convection cooling and protecting the core from thermal damage. Low flow in the water exit line to the hold-up tank will cause the reactor to scram. This safety feature is independent of others discussed above.

g. Minor Power Excursions

The main hazard associated with a minor power excursion is the high instantaneous radiation level outside the biological shield. With a BORAX reactivity insertion of 1.2% $\Delta K/K$ the energy release was 19.5 MW-seconds (Ref. 3). The above excursion corresponds to an integrated dose of approximately 0.0001 mr. at the outer surface of the stall shield which is negligible. The concrete stall shield is reinforced to withstand, conservatively, heating from continuous reactor operation at five times the maximum normal power level or 25 MW. Therefore the concrete should withstand short radiation bursts having much higher intensities than this without cracking.

h. Explosions

The reactor shield is an extremely efficient explosion barrier. It is not credible that an explosion external to the shield could damage the core.

4. MAXIMUM STARTUP ACCIDENT

In this accident it is postulated that, due to circuit malfunctions, all control rods are withdrawn simultaneously with the reactor initially shut down at a very low (source) power level. It is further assumed that no rod inhibits are operative, and that at criticalit· the rods are in their most effective region (50% withdrawn). The total rod bank worth is taken at its upper range (11.6% ΔK). These assumptions maximize the accident.

This accident was analyzed in the original Final Hazards Safety Analysis (Ref. 4) and has been updated with current values of reactor parameters (Ref. 5), The latter analysis is reproduced in Appendix 4. It is shown that even with a 200% scram trip level the total energy of the excursion is only 15 MW-sec. or some 2-1/2 times less than the "Borax threshold". An additional analysis is then made of the self-limited excursion that would result if no safety system were present. The results of SPERT-I and SPERT-IV tests are utilized in this analysis. It is found that the self-shutdown characteristics of the core are such as to limit further the power and energy generated in the excursion. In particular the all-important parameter, fuel-plate surface temperature, for the reactor period corresponding to a 200% scram level is shown to be several hundred degrees below the melting temperature.

5. CREDIBLE SERIOUS ACCIDENTS

Accidents which are credible and which could have serious implications include fuel element mishandling, improper fuel element loading, and experimental facility accidents.

a. Fuel Element Mishandling

Radiation hazards can result from faulty handling of irradiated fuel elements or experimental radioactive samples. The reactor fuel elements are the greatest potential hazard.

No danger of overexposure to personnel exists in the normal handling procedure, but it is conceivable that some malfunction of equipment would result in a fuel element being removed from all shielding, as in the following examples. The building crane will often be used to transport elements either across or along the length of the pool. This would be done by hanging the fuel element handling tool, with the element attached, from the lift hook of the crane. Operations involving the use of the building crane to move fuel elements under water will be performed by persons manipulating the crane controls while standing on the reactor bridge or at the side of the pools. These positions place the crane operators not more than 90 feet from the crane disconnect switch. A fault in the crane control circuit during such an operation might cause the crane hook to rise to its maximum height of 22 feet above the water surface.

Because of the depth of the reactor pool, any fuel element handling tool used for work in the core would have to be a minimum of 25 feet long. Therefore, an element could be raised to within three feet of the pool water surface. An element removed from the core immediately after extended operation of the reactor at power would, under the above conditions, produce a radiation field at the water surface of less than 20 R/hour. It would require about one minute for the element to be raised to this height. This would enable the crane operator to reach and throw the crane disconnect switch located on the wall by the west airlock chamber of the Reactor Building.

It is also conceivable that an element would be moved during a similar operation using a handling tool less than 25 feet long. In the event the crane controls malfunction, the building evacuation alarm would sound when exposed to a radiation field of 5 R/hour. This corresponds to an element about 4.5 feet from the pool water surface. This would warn other people in the Reactor Building and Hot Laboratory to evacuate while the operator went to disconnect electrical power to the crane. (Operator has 12 additional seconds between time alarm sounds and element is exposed). In the event that he failed to disconnect power to the crane and left the building instead, a freshly run element suspended in the center of the Reactor Building would create a radiation field of less than 160 R/hour outside the building wall. If operator fails to leave building before the element is exposed and he takes an additional minute to reach tolerance radiation level zone, he would have absorbed less than 3 rads.

With the building emptied, time would be available to establish an exclusion area and to carefully plan procedures to be used in correcting the situation.

There does not exist the possibility of a similar accident occurring with the pool bridge. Engagement of the core support bridge by the crane hook would not cause the sudden removal of part or all of the shim rods from the reactor core. The support bridge, the control rods, and the core are tied together as a unit and would move as such. Moreover, such a dislodging force would undoubtedly cause release of the shim rods. Lifting of the bridge would not expose the core unless the crane hook engaged a lower portion of the reactor core support tower. This potential hazard is eliminated by clamping the bridge structure to the bridge rails.

b. Fuel Element Loading Accidents

The possibilities of a loading accident are extremely remote. In order to evaluate the magnitude and probability of a loading accident, the technique employed during loading fuel must be considered, along with each of the four separate loading conditions.

The UCNR reactor has five shim safety rods with a total worth of approx. 10% $\Delta K/K$ and a regulating rod worth not more than 0.6% $\Delta K/K$. Whenever fuel is added to the core, it will be done with the shim rods fully inserted.

(1) The First Loading Condition to be considered is the approach to criticality on a new core configuration. During such a loading, elements are added at the outer faces of those elements already in place. This is done with the shim rods fully inserted. After the addition of a maximum of two elements, the rods are withdrawn to check for criticality. If the core does not go critical, the sub-critical count rate is recorded and plotted on a graph of reciprocal count rate versus total mass of fuel in core at time of count. This aids in estimating the mass at which the core will achieve criticality. At completion of the count, the rods are re-inserted and an additional one or two elements added. This process is repeated until criticality is attained. As the core size builds up, it is possible that two additional elements could be added to a core that is almost critical with rods fully withdrawn. The addition would, of course, be made with the rods fully inserted.

These two additional elements, when added to an outer face, would not add more than 5% $\Delta K/K$. It would be extremely improbable that a loading accident would occur during the build-up of a new core because this would require the insertion of a minimum of four additional elements without a criticality check, to just compensate for the negative reactivity value of the inserted rods.

(2) The Second Loading Condition to be considered is routine refueling at the end of a run without a change in core configuration. In reloading such a core the shim rods will be fully inserted and elements from the core will be replaced with elements which have a mass within ± 10% of the weight of the exchanged elements at the beginning of the previous cycle. Neither the core size nor the geometry would change for such a core. The largest loading error readily conceivable would involve a net increase of core mass by 10% or a maximum of 540 gms. U-235. If all mass deviations were +10% this could be equivalent to a reactivity increase of 8%, which is approaching the negative-reactivity effect of the inserted shim rods.

The total number of elements that can be changed without a criticality check is therefore limited to 15, or roughly half the core. This restricts the possible reactivity increase to a safe value of about 4%.

(3) The Third Loading Condition to be considered is the replacement of sufficient elements to allow a restart after the reactor has been "caught" by xenon. This, at its worst, would involve a loading error of less magnitude than that listed in (2) above since that case considers a change of all elements, a case which would not occur during a xenon reload. Again, restricting the number of elements to 15 (before a criticality check need be made), reduces the reactivity change to a safe value.

(4) The Fourth Loading Condition involves the placing of an element into a central core position. This operation would be required for removal of a central flux trap. The procedure for loading a central element into such a core is to remove a minimum of six outside elements for each central element to be inserted. After addition of the central element, a criticality check is made to determine how many of the removed elements are to be returned to the outer faces of the core. These elements are then reloaded as described for the first loading condition.

An operational misjudgment could allow the loading of the central element without first removing elements from the outer face of the core. The worth of a new fuel element when inserted into the center of a core could be as much as 6% $\Delta K/K$. As this is a large fraction of the total worth of the inserted control rods, the removal of enough outer elements to approximately total 6% $\Delta K/K$ is desirable. A total of six elements would be adequate.

c. Experimental Facility Accidents

Fast reactivity steps are considered to furnish the most credible serious accidents. For these reactivity steps to achieve such effects, they must take place in less than 50 milliseconds. Reactivity changes that require longer periods than the response time of the safety system require coincident failure of the safety system to achieve serious consequences.

Reactor power response to various reactivity transients is given in Appendix 2, Sect. B. The reactivity worth of experiments and experiment facilities (beam tubes) is limited, either by Technical Specifications (Ref. 9) or by design, to safe values.

6. DESIGN BASIS ACCIDENT

The assumed DBA, which is an accident with consequences exceeding the most credible accident, is one in which a partial fuel melt accompanied by a loss of all pool water occurs. No mechanism is postulated for this hypothetical accident.

The DBA and its results are covered in detail in Appendix 2, Sect. C2.

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2. J. F. Wett, Jr., Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air, ORNL-2892, March 23, 1960.

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5. Letter USAEC to UCC dated Apr. 11, 1966. (Change No. 3 to Tech. Specs.)

6. Letter USAEC to UCC dated Apr. 2, 1970. (Change No. 9 to Tech. Specs.)

7. Letter USNRC to UCC dated March 15, 1979. (Amendment No. 13 to License R-81).

8. Letter USNRC to UCC dated Sept. 21, 1979. (Amendment No. 12 to License R-81).

9. Letter USNRC to UCC dated May 17, 1979. (Safety Evaluation, Amendment No. 14 to License R-81).

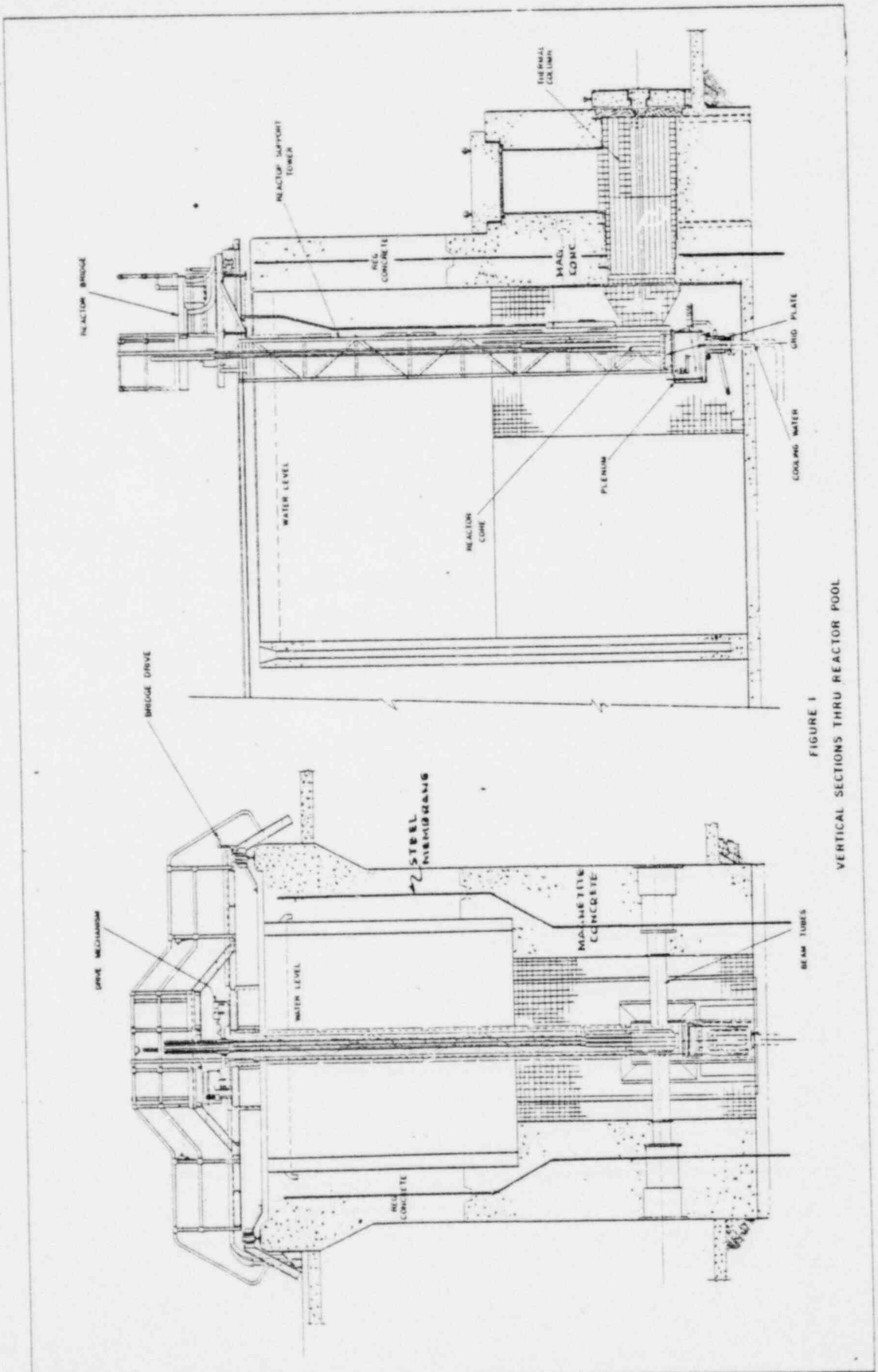


FIGURE 1
VERTICAL SECTIONS THRU REACTOR POOL

FIG. 1

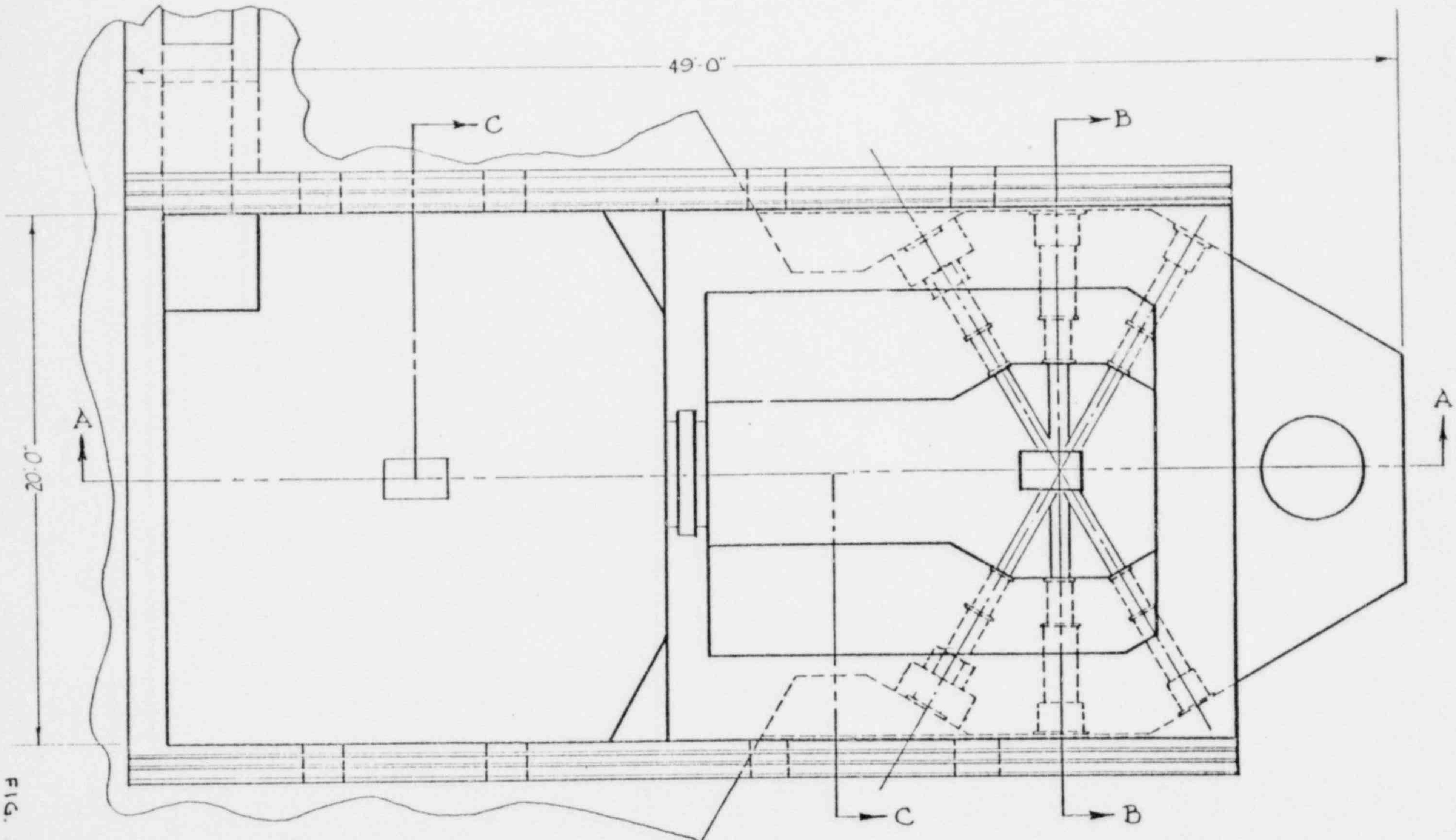


FIGURE 2

REACTOR POOL PLAN

FIG. 2

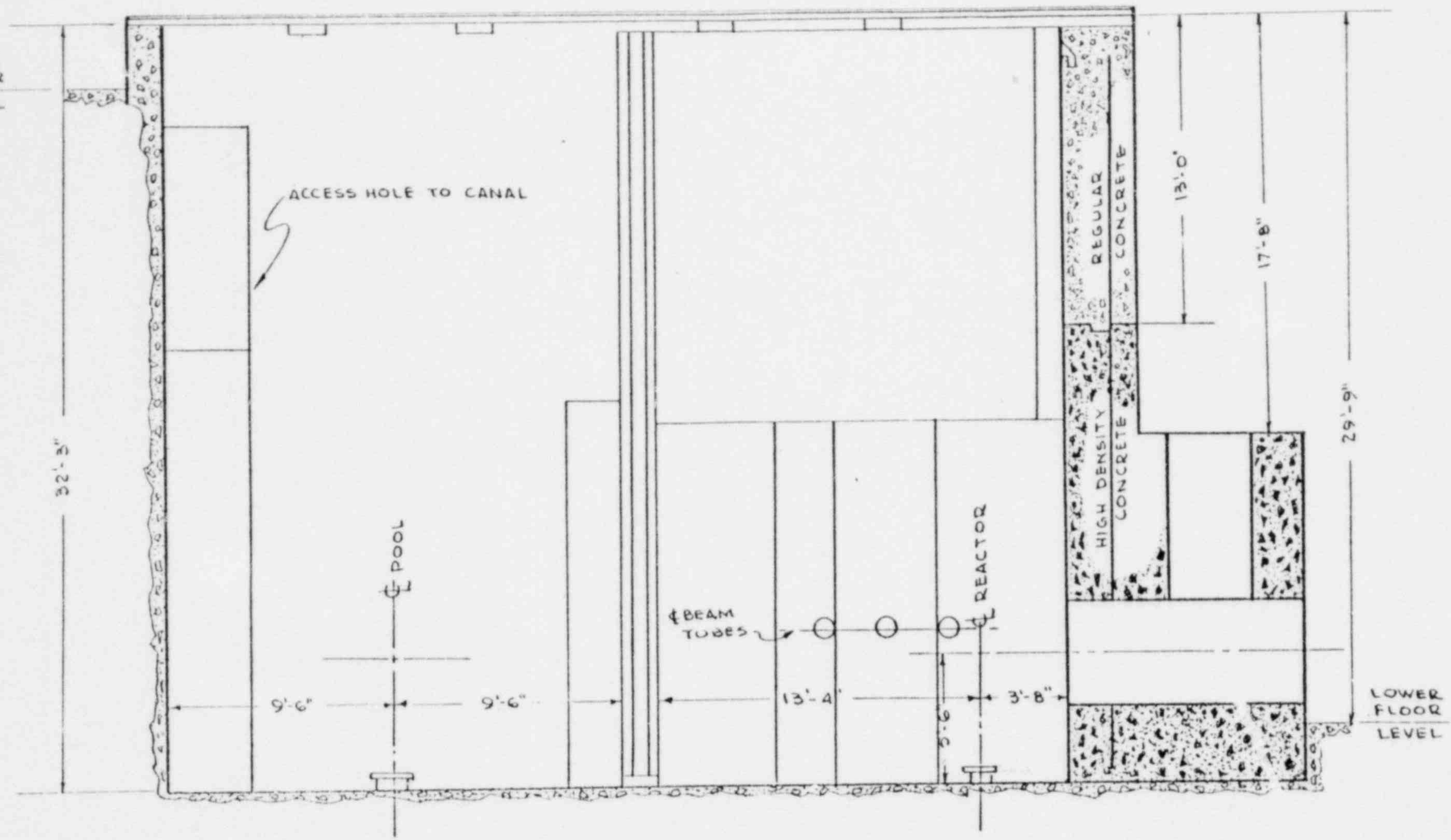


FIGURE 3

REACTOR POOL- SECTION A-A

FIG. 3

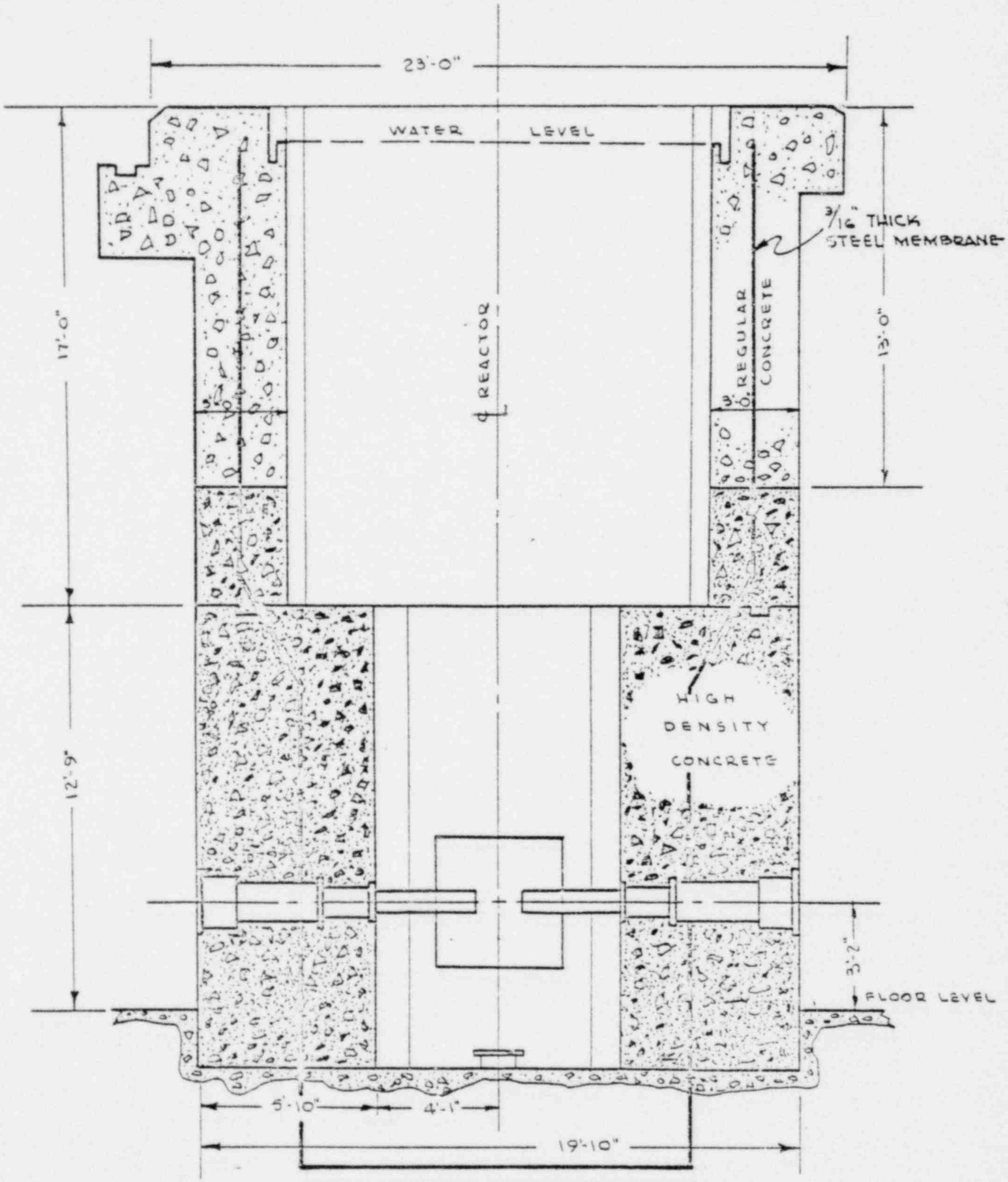


FIGURE 4

REACTOR POOL- SECTION B-B

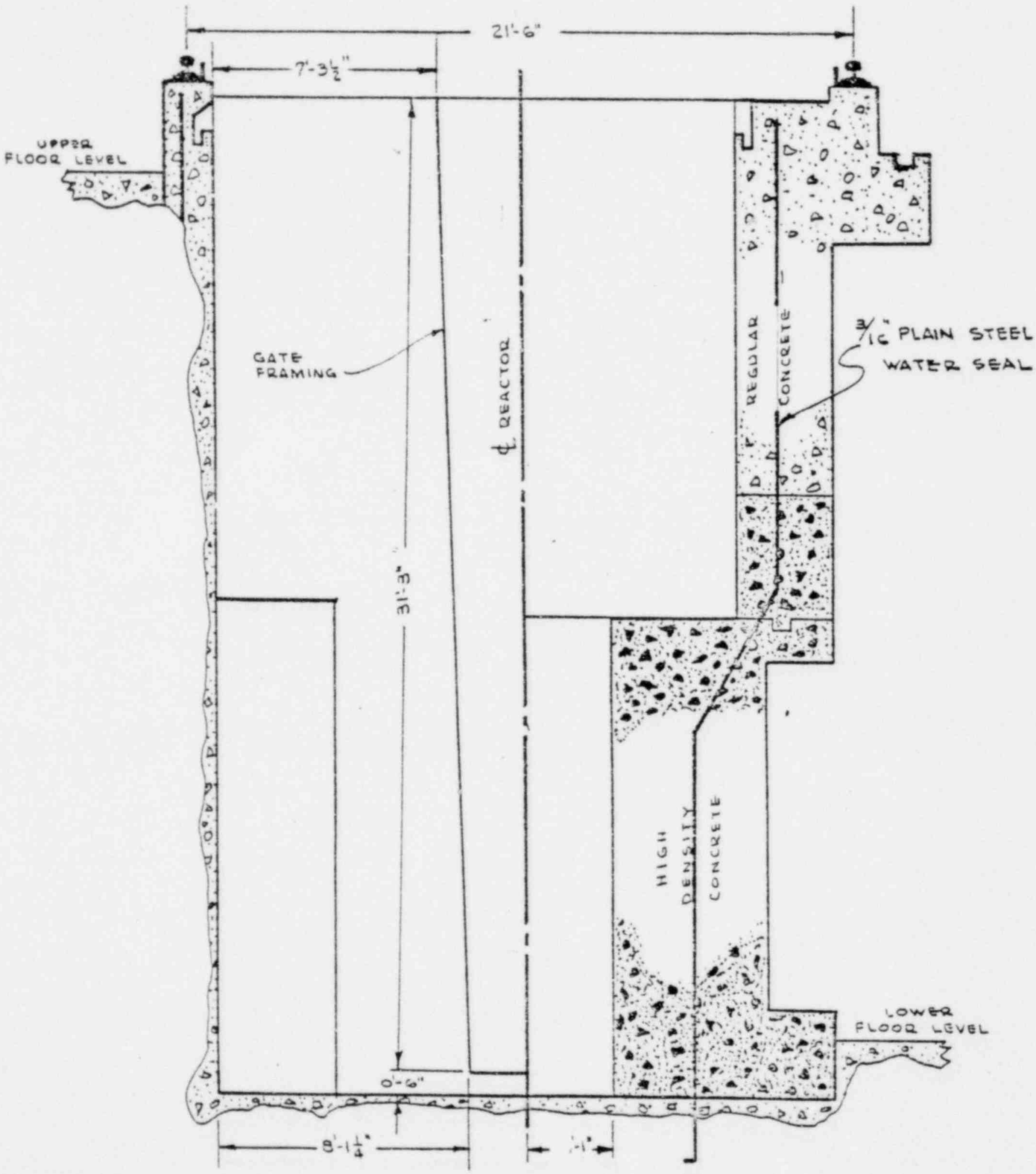
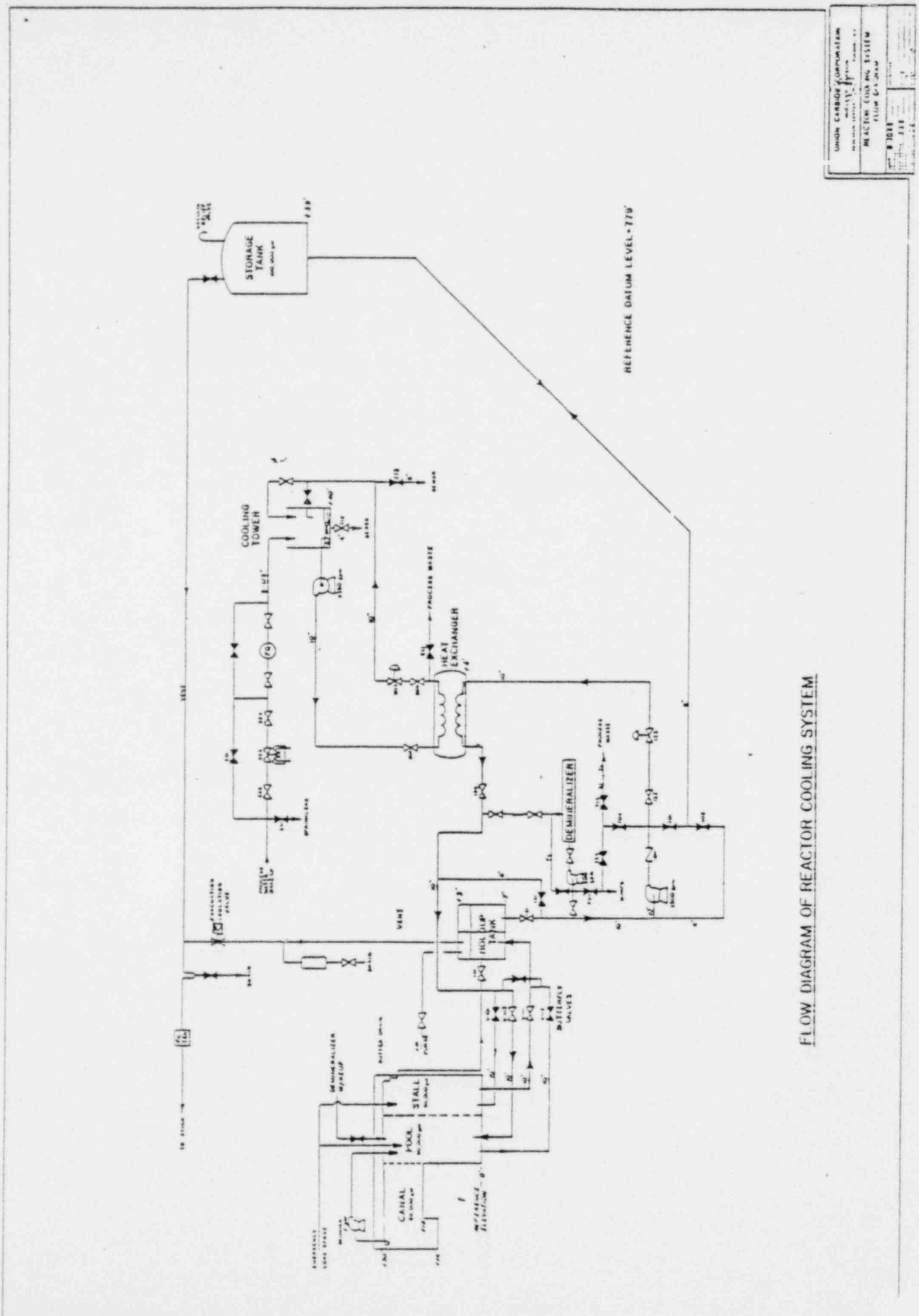


FIGURE 5

REACTOR POOL - SECTION C-C



FLOW DIAGRAM OF REACTOR COOLING SYSTEM

Fig. 6

FIG. 7

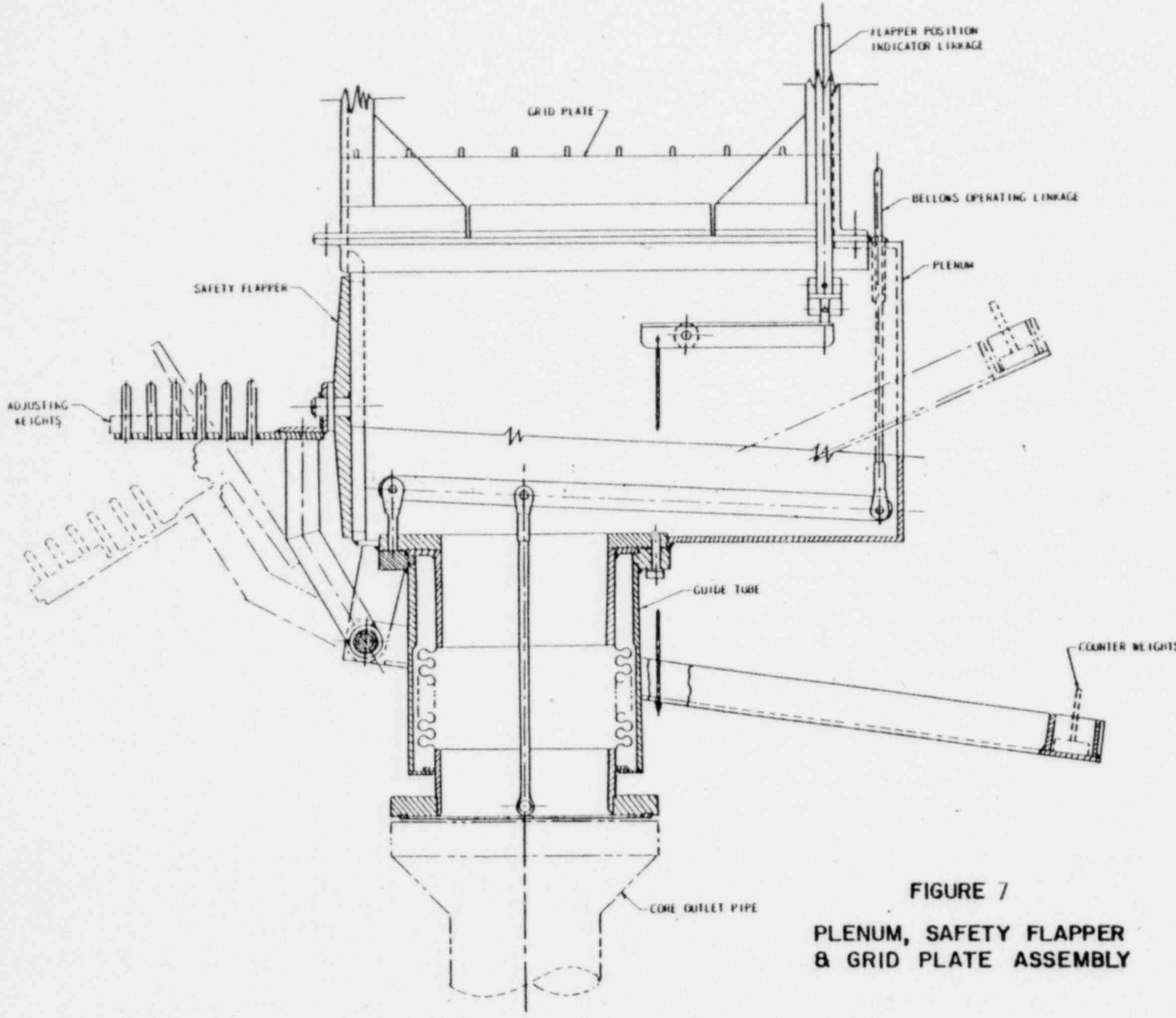
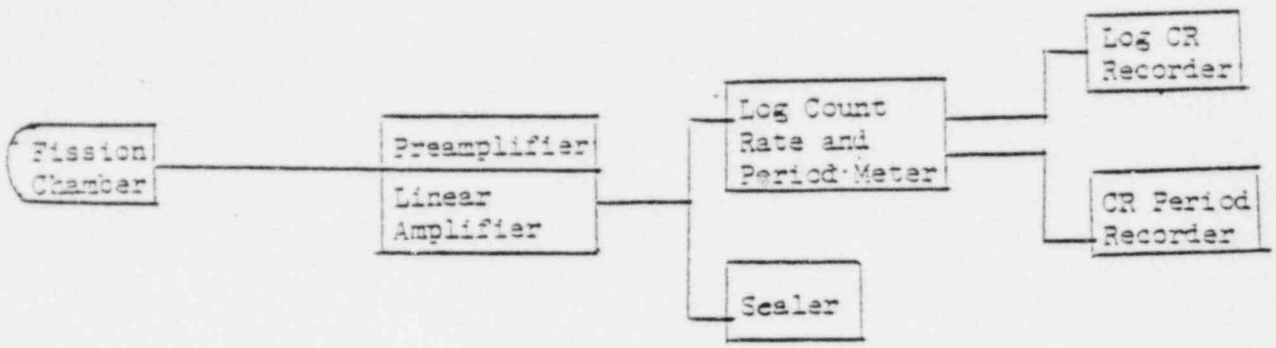
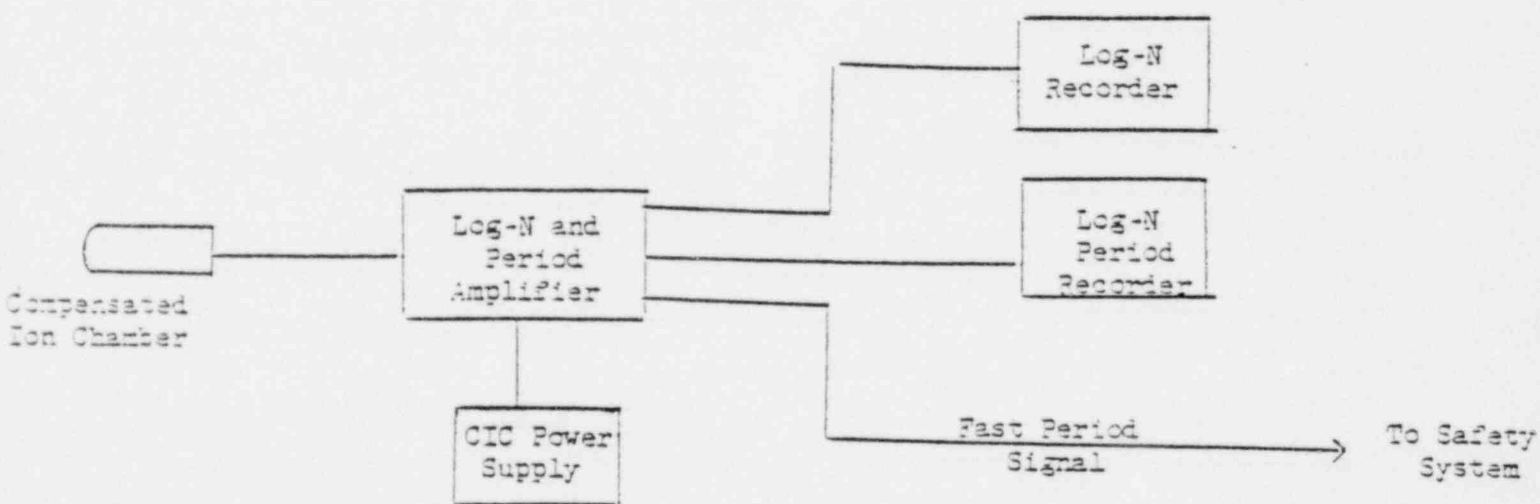


FIGURE 7

PLENUM, SAFETY FLAPPER
& GRID PLATE ASSEMBLY



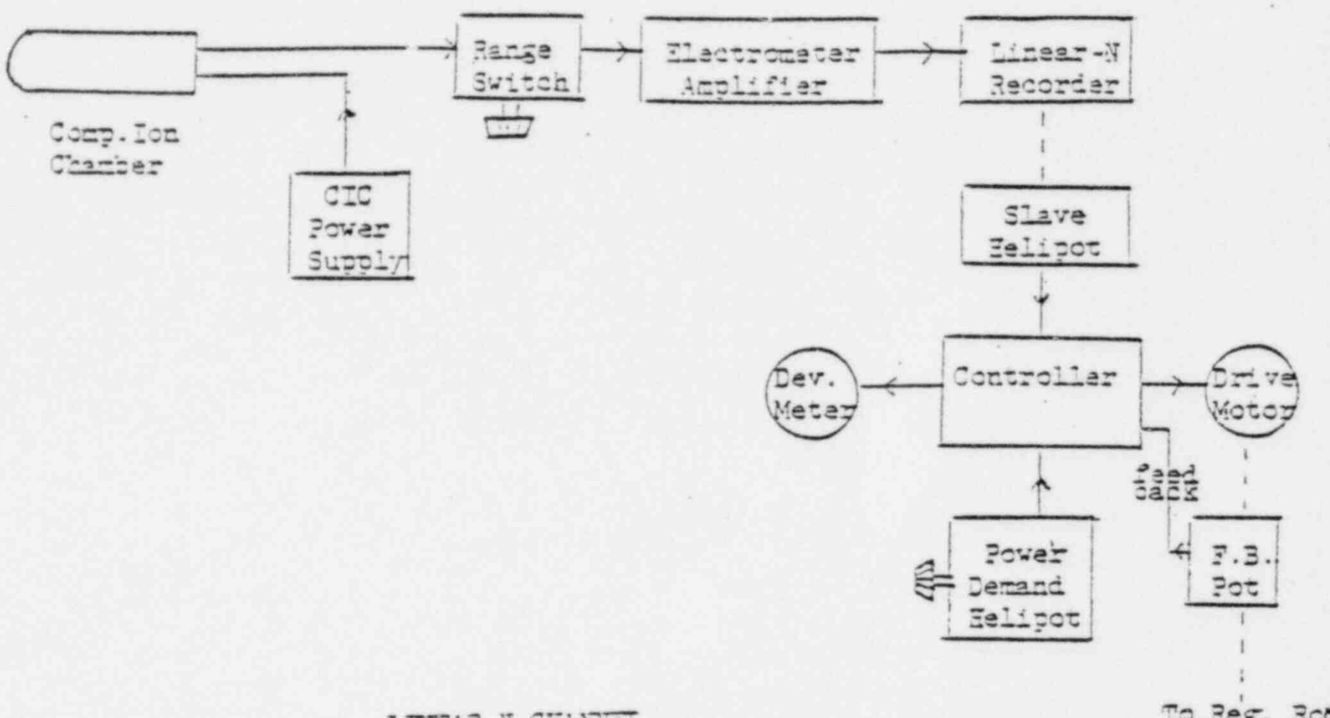
LOG COUNT RATE CHANNEL



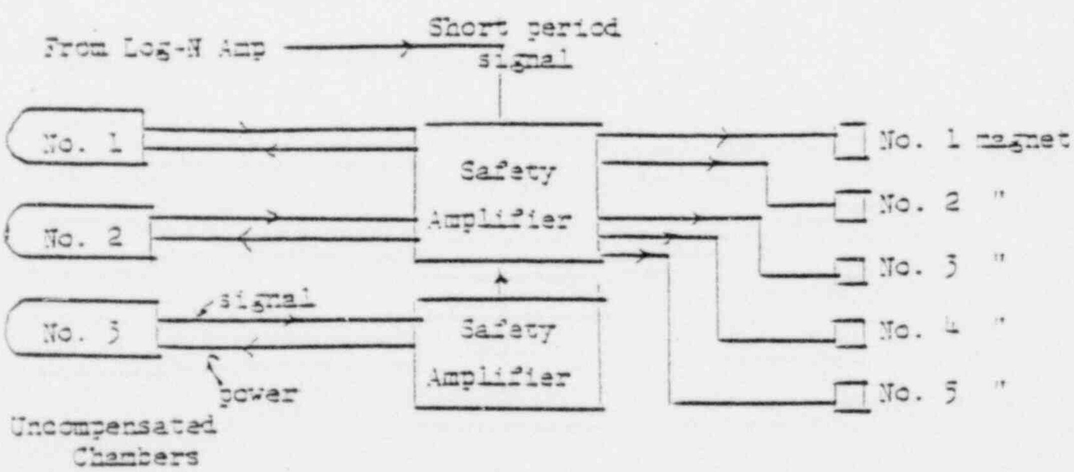
LOG-N CHANNEL

BLOCK DIAGRAMS

Fig. 8



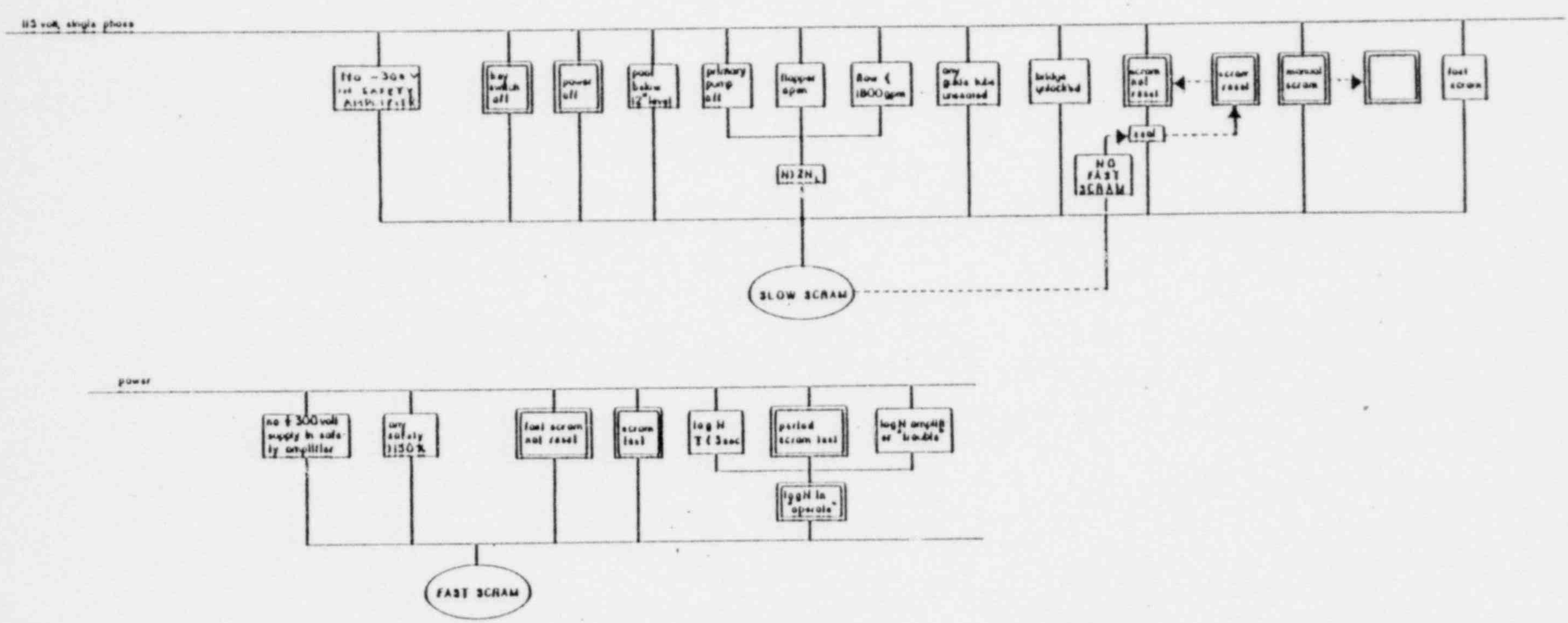
LINEAR-N CHANNEL



SAFETY SYSTEM

BLOCK DIAGRAMS

Fig. 9

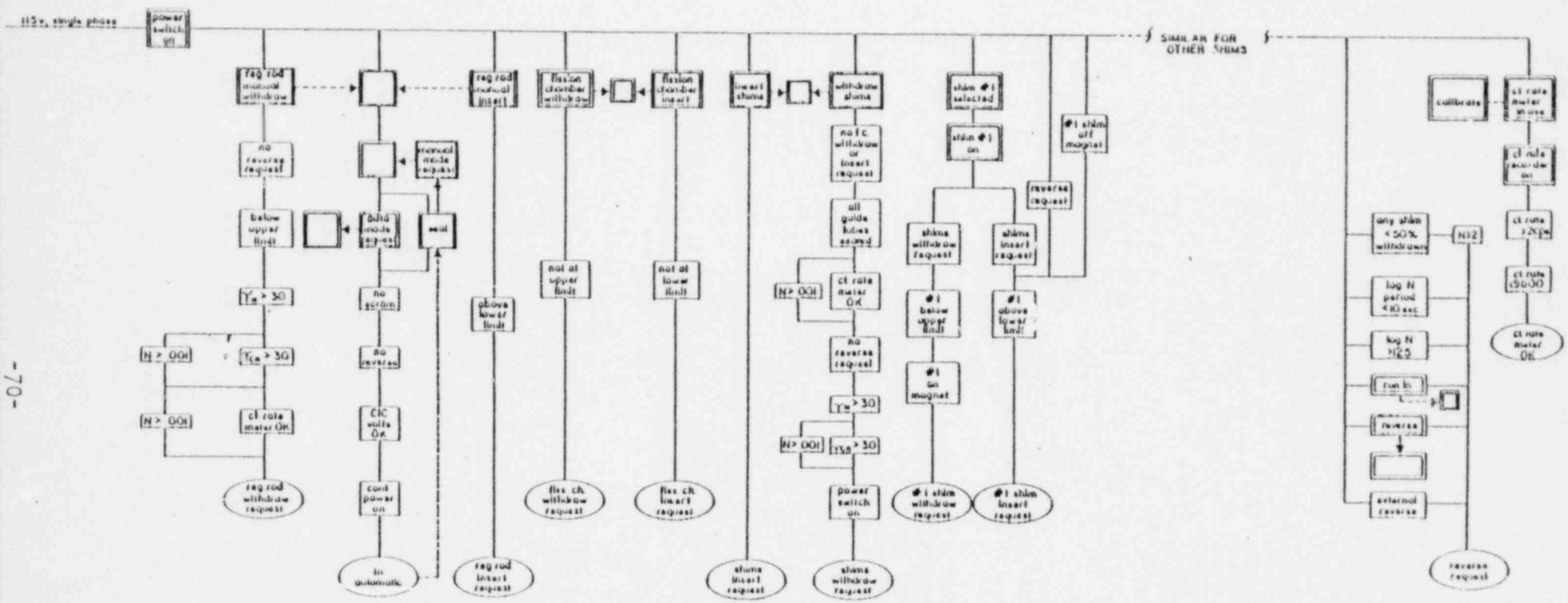


REACTOR SCRAM CIRCUITS, LOGIC DIAGRAM

UNION CARBIDE CORPORATION

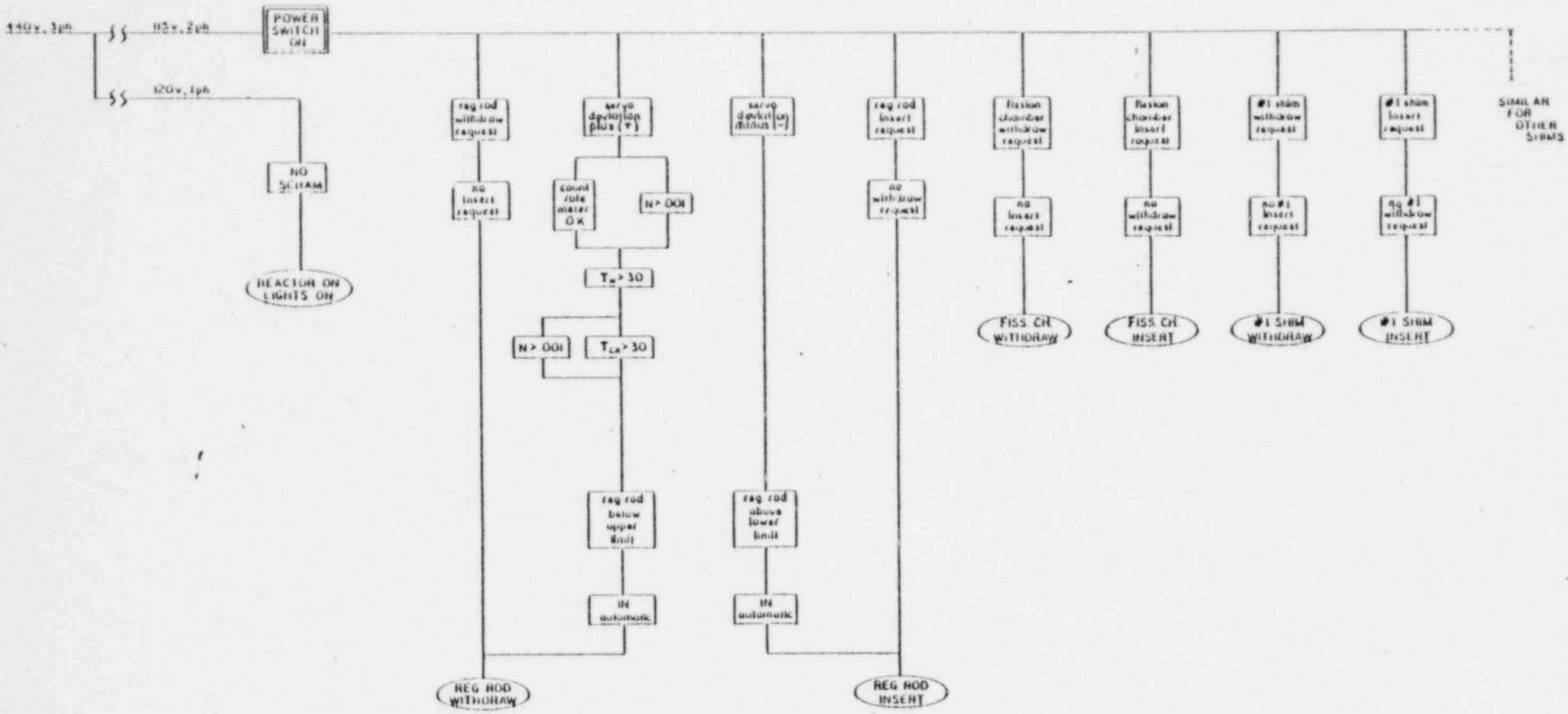
SCRAM LOGIC DIAGRAM

Fig. 10



REACTOR ROD CONTROL CIRCUITS, LOGIC DIAGRAM

Fig. 11

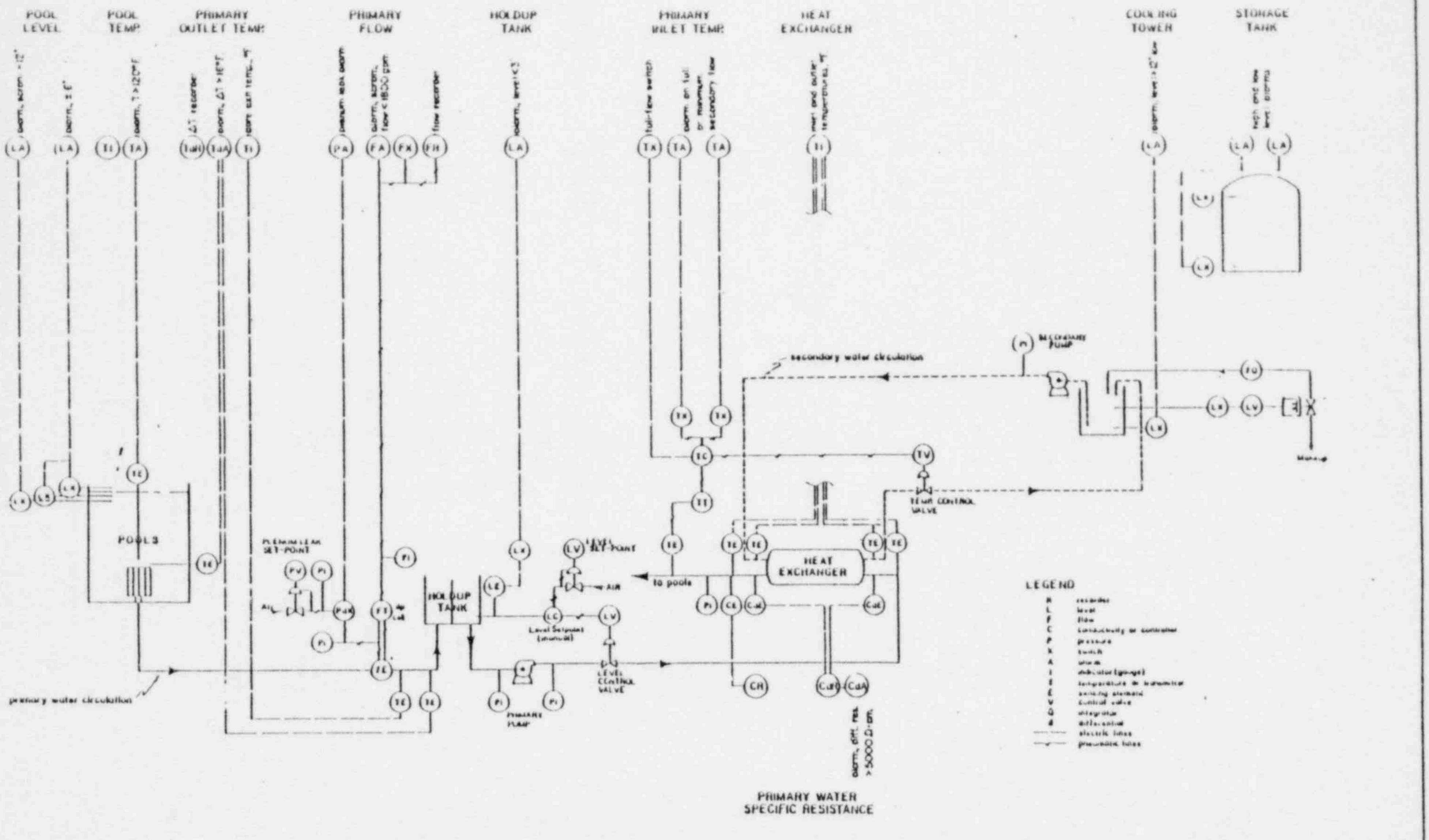


REACTOR ROD DRIVE CIRCUITS, LOGIC DIAGRAM

UNION CARBIDE CORPORATION

Reactor Rod Drive Logic

Fig. 12



REACTOR COOLING SYSTEM-ELEMENTARY FLOW CONTROL DIAGRAM

Fig. 13

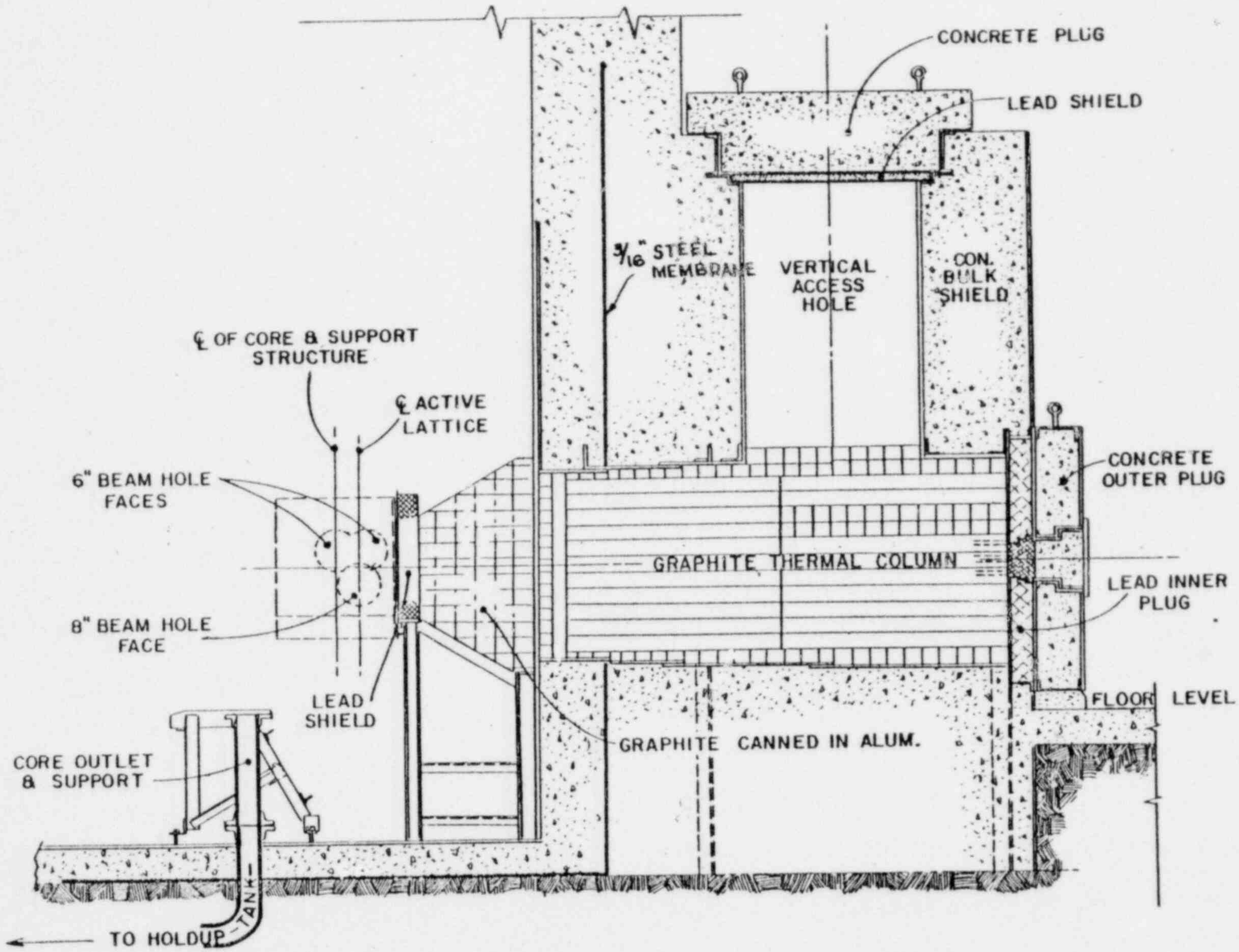
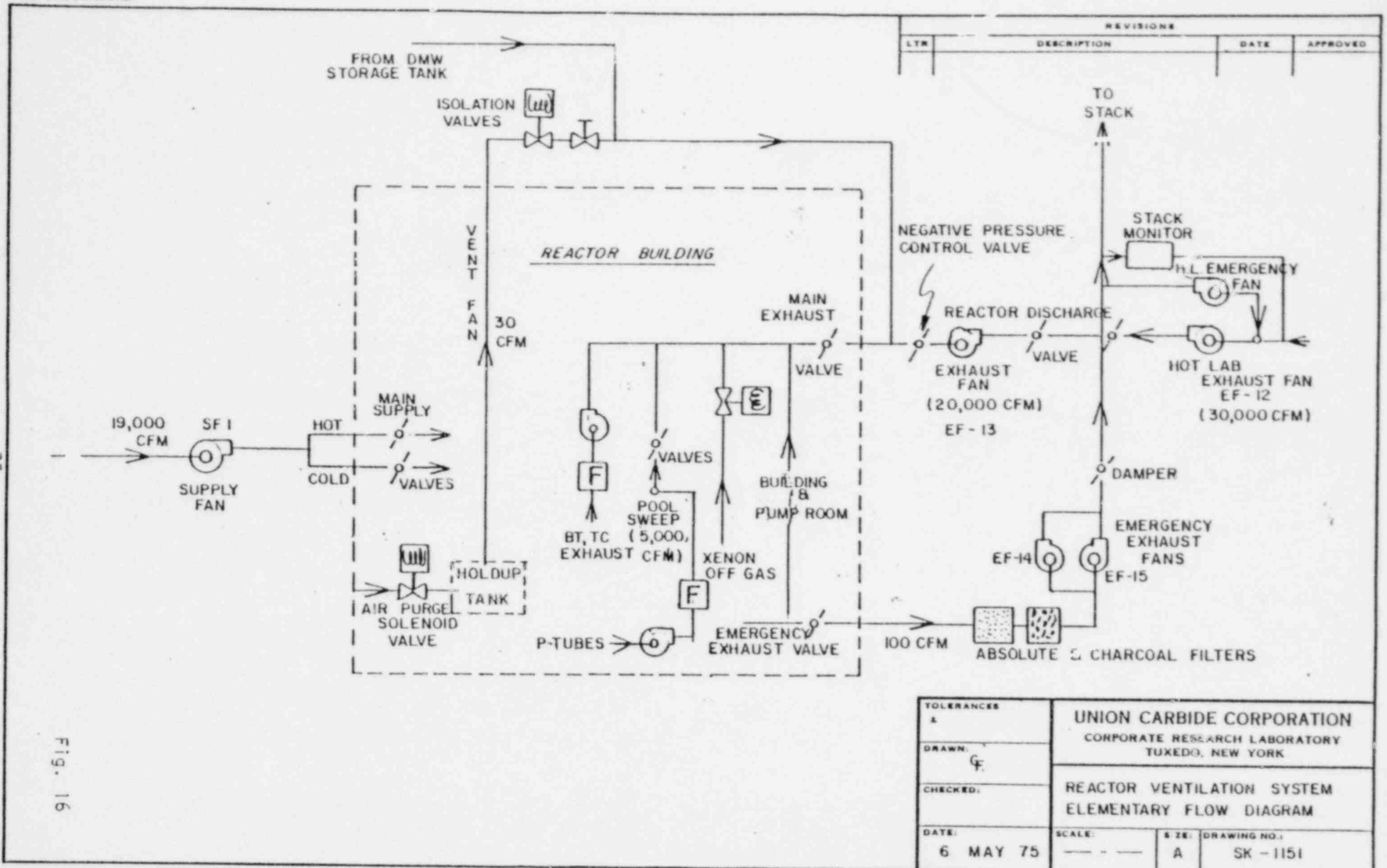


FIGURE 14
VERTICAL SECTION THRU THERMAL

A1 ++++++ 0.0 .000	B1 ++++++ 0.0 .000	C1 ++++++ 0.0 .000	D1 ++++++ 0.0 .000	E1 ++++++ 0.0 .000	F1 ++++++ 0.0 .000
A2 ++++++ 0.0 .000	B2 UC 345 117.8 .005	C2 F 7749 120.4 .008	D2 UC 356 124.1 .009	E2 F 7740 112.0 .008	F2 UC 347 119.6 .005
A3 F 7739 116.9 .007	B3 ++++++ 0.0 .000	C3 UC 378 137.3 .010	D3 UC 369 134.3 .013	E3 ++++++ 0.0 .008	F3 UC 355 126.7 .007
A4 F 7748 116.1 .009	B4 UC 55S 81.8 .016	C4 UC 363 153.3 .017	D4 UC 376 147.7 .016	E4 UC 61S 72.6 .018	F4 UC 354 129.4 .010
A5 ++++++ 0.0 .000	B5 UC 374 177.1 .015	C5 UC 382 174.6 .022	D5 UC 385 184.3 .022	E5 UC 372 177.3 .015	F5 ++++++ 0.0 .000
A6 UC 359 117.0 .010	B6 UC 60S 95.3 .020	C6 UC 368 188.4 .022	D6 UC 373 178.6 .023	E6 UC 58S 63.4 .020	F6 F 7747 117.0 .012
A7 ++++++ 0.0 .000	B7 UC 387 178.6 .018	C7 UC 366 173.9 .020	D7 ++++++ 0.0 .000	E7 UC 383 175.7 .017	F7 ++++++ 0.0 .000
A8 F 7746 113.5 .011	B8 UC 59S 73.8 .018	C8 UC 370 128.9 .016	D8 UC 360 134.3 .018	E8 UC 57S 72.5 .018	F8 UC 349 115.1 .009
A9 F 7744 114.0 .009	B9 UC 362 121.3 .010	C9 ++++++ 0.0 .011	D9 UC 358 119.7 .012	E9 F 7742 113.4 .011	F9 ++++++ 0.0 .000

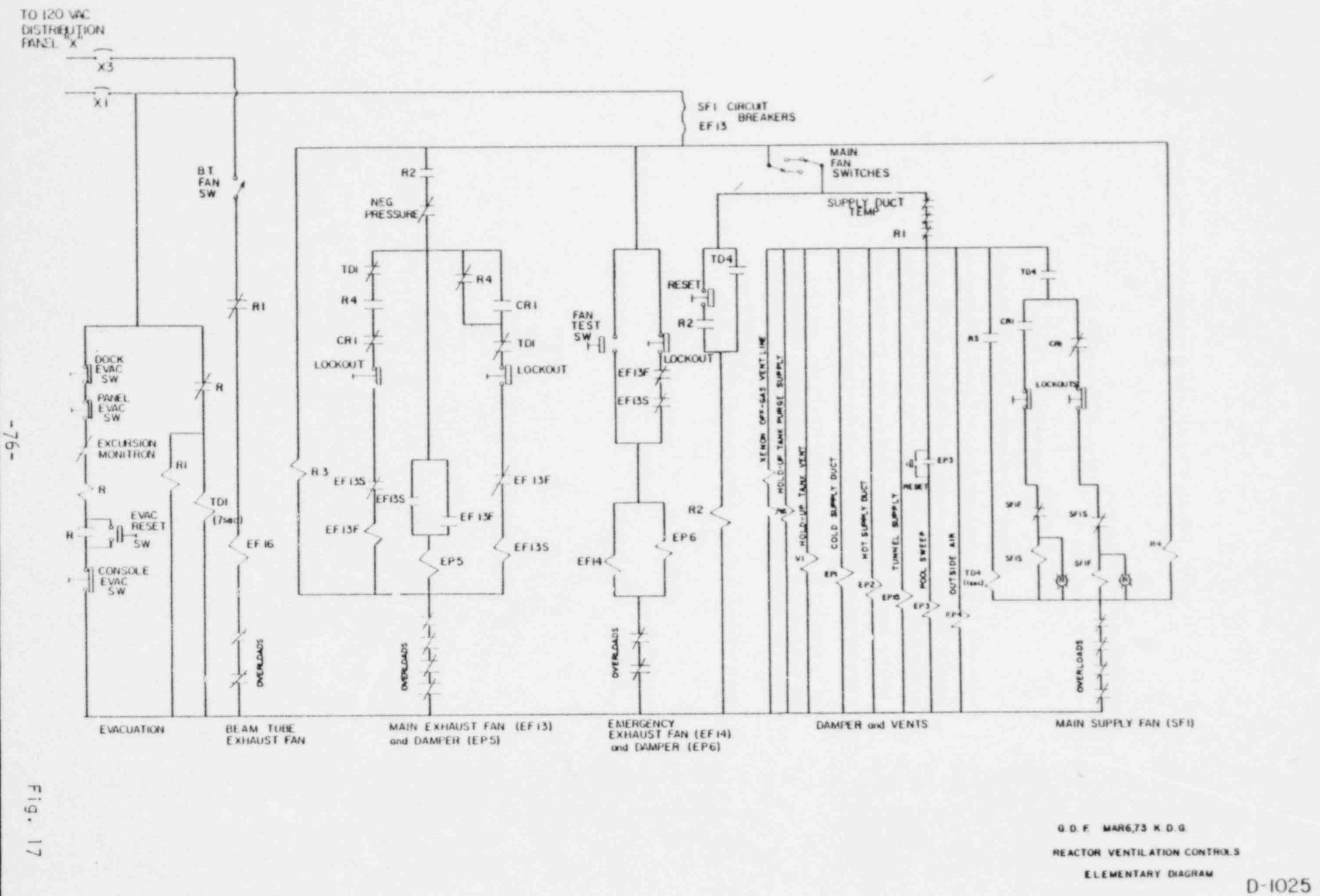
TYPICAL CORE ARRANGEMENT

Fig. 15



REVISIONS
LTR
DESCRIPTION
DATE
APPROVED
FROM DMW STORAGE TANK
ISOLATION VALVES
TO STACK
REACTOR BUILDING
VENT FAN
30 CFM
NEGATIVE PRESSURE CONTROL VALVE
STACK MONITOR
H.L. EMERGENCY FAN
MAIN EXHAUST VALVE
REACTOR DISCHARGE
EXHAUST FAN (20,000 CFM) EF - 13
VALVE
HOT LAB EXHAUST FAN EF - 12 (30,000 CFM)
19,000 CFM
SF 1
SUPPLY FAN
HOT
COLD
MAIN SUPPLY VALVES
VALVES
BT, TC EXHAUST
POOL SWEEP (5,000 CFM)
XENON OFF GAS
BUILDING & PUMP ROOM
DAMPER
EMERGENCY EXHAUST FANS
EF-14
EF-15
AIR PURGE SOLENOID VALVE
HOLDUP TANK
P-TUBES
EMERGENCY EXHAUST VALVE
100 CFM
ABSOLUTE & CHARCOAL FILTERS
TOLERANCES
DRAWN: CF
CHECKED:
DATE: 6 MAY 75
UNION CARBIDE CORPORATION
CORPORATE RESEARCH LABORATORY
TUXEDO, NEW YORK
REACTOR VENTILATION SYSTEM
ELEMENTARY FLOW DIAGRAM
SCALE: —— - ——
SIZE: A
DRAWING NO.: SK - 1151

Fig. 16



TO 120 VAC DISTRIBUTION PANEL "X"
X3
X1
SF1 CIRCUIT BREAKERS
EF13
MAIN FAN SWITCHES
SUPPLY DUCT TEMP
B.T. FAN SW
R1
R2
NEG PRESSURE
TD1
R4
CR1
LOCKOUT
R3
EF13S
EF13F
EP5
FAN TEST SW
EP6
EF14
TD4
RESET
DOCK EVAC SW
PANEL EVAC SW
EXCURSION MONITRON
R
EVAC RESET SW
CONSOLE EVAC SW
TD1 (7sec)
EF16
OVERLOADS
XENON OFF-GAS VENT LINE
HOLD-UP TANK PURGE SUPPLY
HOLD-UP TANK VENT
COLD SUPPLY DUCT
HOT SUPPLY DUCT
TUNNEL SUPPLY
POOL SWEEP
OUTSIDE AIR
V1
EP1
EP2
EP18
EP3
EP4
SF1F
SF1S
EVACUATION
BEAM TUBE EXHAUST FAN
MAIN EXHAUST FAN (EF13) and DAMPER (EP5)
EMERGENCY EXHAUST FAN (EF14) and DAMPER (EP6)
DAMPER and VENTS
MAIN SUPPLY FAN (SF1)
G.D.F. MAR6,73 K.D.G.
REACTOR VENTILATION CONTROLS
ELEMENTARY DIAGRAM
D-1025

Fig. 17

ORGANIZATIONAL STRUCTURE

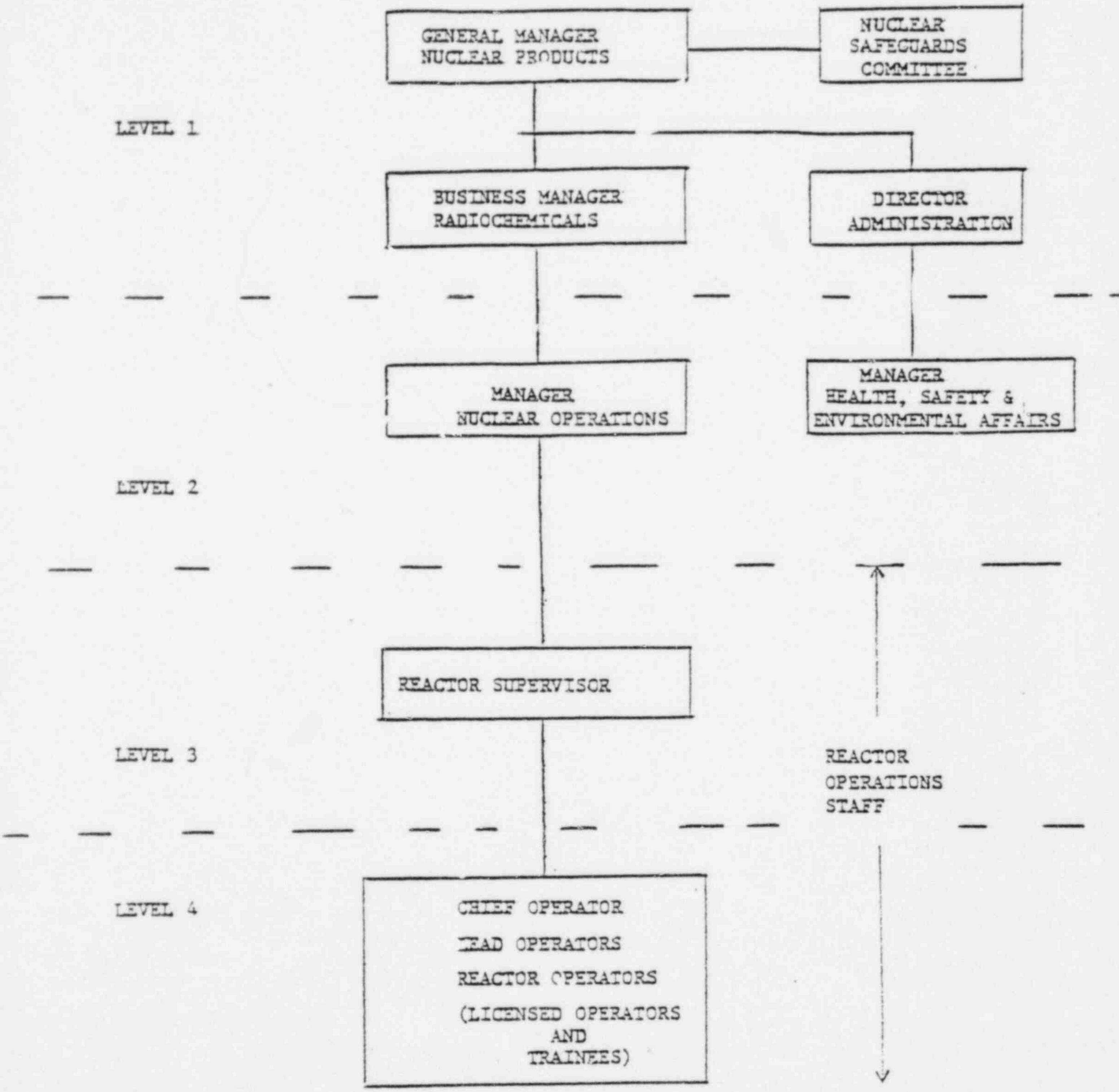


Fig. 18

GENERALIZED CONFIGURATION OF REACTOR POOL SYSTEM

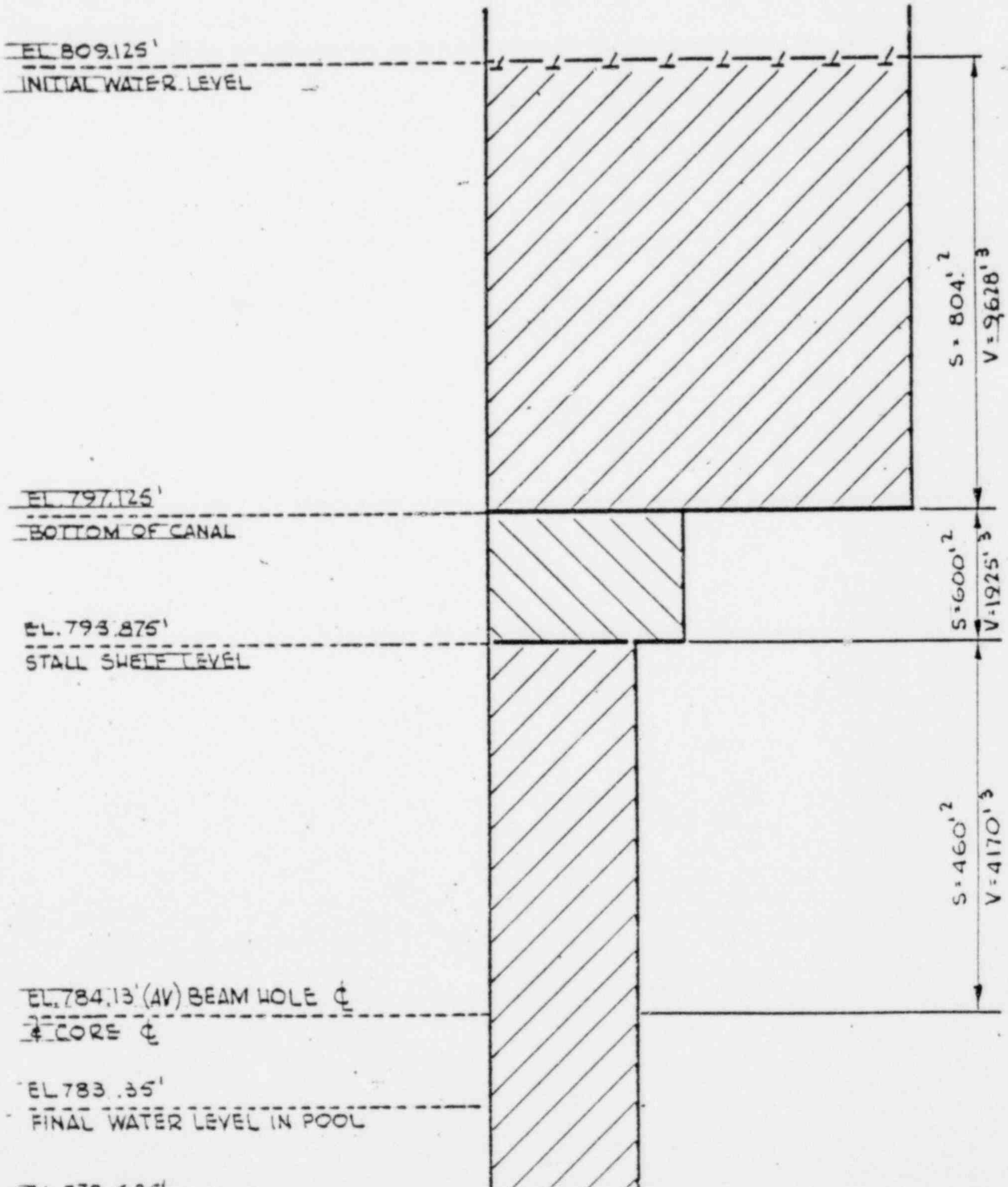


FIG. 19

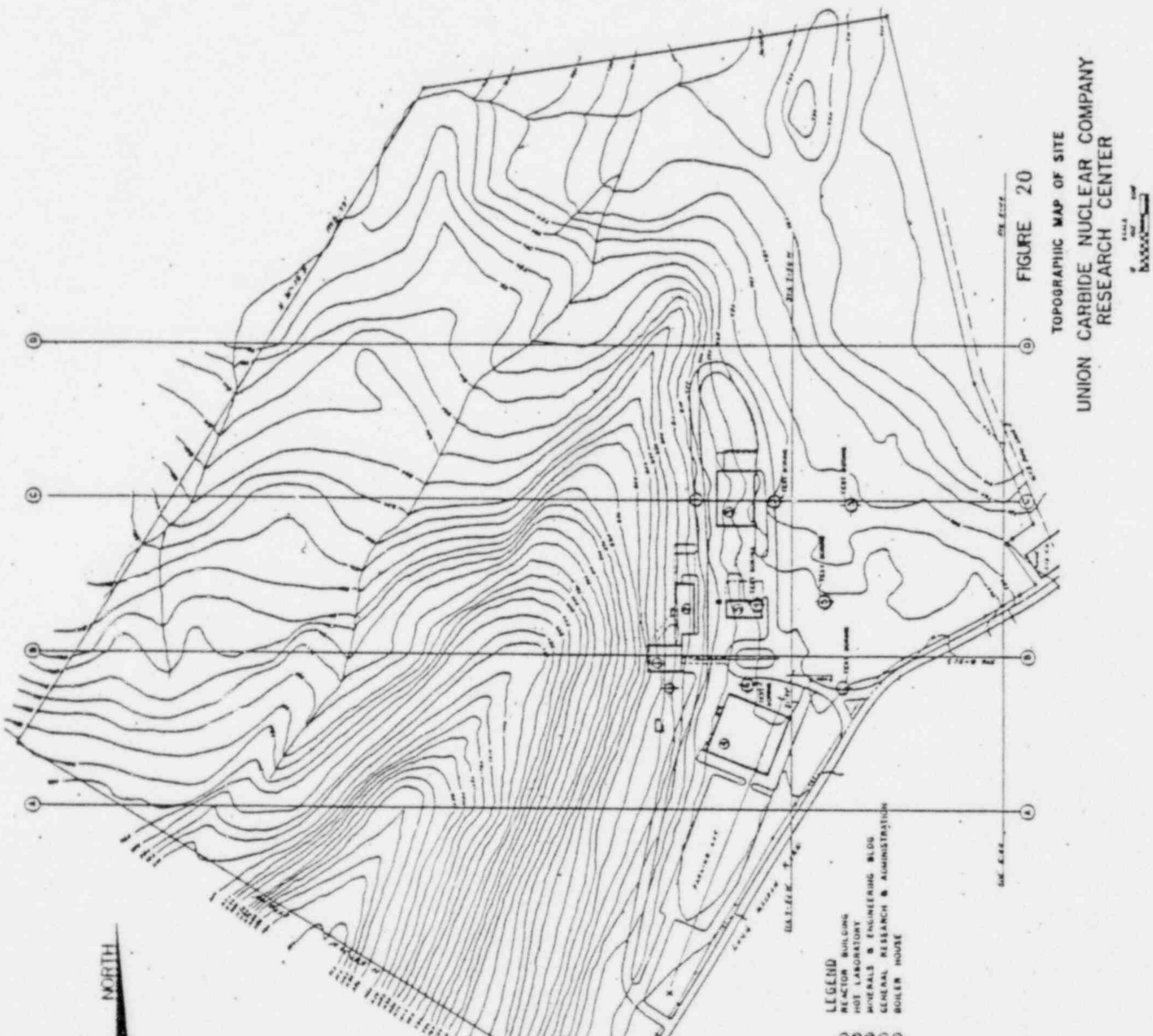
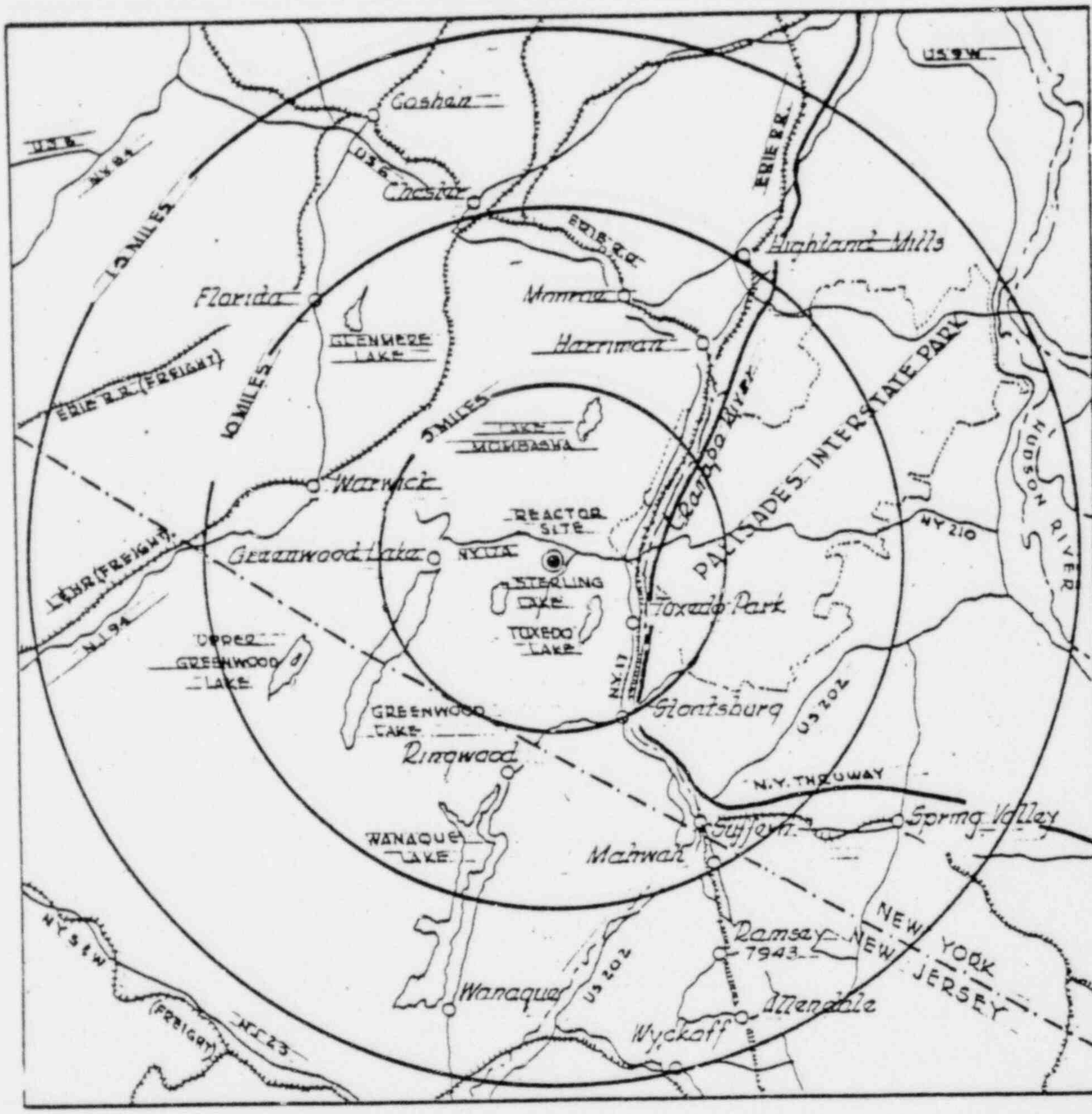


FIGURE 20
TOPOGRAPHIC MAP OF SITE
UNION CARBIDE NUCLEAR COMPANY
RESEARCH CENTER
NORTH
LEGEND
REACTOR BUILDING
HOT LABORATORY
MINERALS & ENGINEERING BLDG
GENERAL RESEARCH & ADMINISTRATION
BOILER HOUSE

FIG. 20


Cedar Mtn
Tiger Mtn
Flag Swamp
Bare Mtn
Hogback Mountain
Site
GREENWOOD
LAKE
Spruce Pond Camp (BSA)
Spruce Pond
Wildcat Mountain
Warwick Brook
Cairn Mtn
POTUCKET
STERLING
ROAD
We-Wah Lake
CONTINENTAL
Golf Course
Tuxedo Park Sch
Tower Hill
LAKE
Fox Hill
TUXEDO
Tuxedo Park
RAMAPO
PRIVATE
PARK

Fig. 21



Legend

- THRUWAY
- U.S. OR STATE HIGHWAY
- RAILROAD

0 2 4 6 8 10

Scale in miles

FIGURE 22

SETTLEMENTS & TRANSPORTATION LINES IN THE ENVIRONS OF THE REACTOR SITE