SAFETY ANALYSIS REPORT

FOR

THE UNIVERSITY OF WISCONSIN NUCLEAR REACTOR

SUBMITTED TO

THE UNITED STATES ATOMIC ENERGY COMMISSION

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Chapter 1

SUMMARY AND PROPOSED TECHNICAL SPECIFICATIONS

1.1 INTRODUCTION

The University of Wisconsin Nuclear Reactor achieved initial criticality on 26 March 1961 as a 10 KW teaching and research reactor. The power level was increased to 250 KW on 7 December 1964, using the original flat-plate aluminum clad fuel.

The reactor was converted to a 1000 KW, TRIGA reactor with pulsing capability in 1967. Initial criticality with the TRIGA core occurred on 14 November 1967. Since that time the reactor has been operated for more than 3 million kilowatt hours. At the present time a partial refuelling is necessary to provide adequate reactivity for the operating program.

This and subsequent refuellings will use TRIGA FLIP fuel containing erbium oxide burnable poison and a higher uranium 235 enrichment which will provide extended fuel lifetime.

Since submission of the Safety Analysis Report for operation with TRIGA fuel in July of 1966, a number of changes in the facility have occurred. These changes were reported to the Commission in Annual Reports required under 20 CFR Part 50. In view of these changes and re-evaluation of safety analyses required by use of FLIP fuel, this document is submitted as a replacement for the entire Safety Analysis Report even though the present application is intended as a request for changes in Technical Specifications.

The remainder of this introduction section points out changes from the previous SAR. In addition to the changes indicated below, there were several places where references to 1.5 MW operation were omitted or where minor wording changes were made.

Chapter 1. Introduction Section and Proposed Technical Specifications--Extensively revised.

General Description and Summary of Reactor Data--Changes made to reflect use of FLIP fuel.

Proposed Technical Specifications are completely revised to conform to new format and FLIP fuel addition.

Chapter 2.

Section 2.1.3 Modified to include description of FLIP fuel.

Section 2.1.9 Changes to reflect restrictions on core arrangement with FLIP fuel.

Section 2.3.1 Changed to indicate use of three fuel storage baskets rather than previous one-piece storage rack.

Reference to pH control of pool water removed as previously reported.

Section 2.4.1 and Figure 19--Changed to show location of safety channel CIC's inside thermal column as previously reported.

Section 2.4.4 Changed to show use of CO_2 cover gas rather than helium.

Section 2.4.5 Changed to describe hydraulic sample irradiation devices previously reported.

Section 2.5.1 Changed to include scram on fuel temperature. This is a change in support of new Technical Specifications.

Section 2.5.5 Includes description of servo control system changes previously reported.

Section 2.5.7 Includes fuel temperature scram at LSSS.

Section 2.6.2 Changed to include installation of stairway from console area to to pool top as previously reported.

Section 2.7 Changed to make references to fuel element bow and elongation consistent with proposed Technical Specification Limits.

Chapter 3.

This chapter is extensively rewritten to reflect remodelling in the Reactor Laboratory and surrounding areas and to reflect construction of the Engineering Research Building east of the Mechanical Engineering Building as previously reported.

Chapter 4.

This chapter has been revised to show UWNR data on curves of prototype behavior, and to indicate behavior expected with FLIP and mixed cores.

Chapter 5.

Chapter 5 is extensively revised to show
(a) Procedures to be used in assembly of operational core using mixed FLIP and Standard fuel, and
(b) To indicate a change in review and audit functions as presented in the proposed Technical Specifications.

Chapter 6.

Chapter 6 has been extensively revised as a result of the higher power density due to peaking in mixed and FLIP cores. The sections on loss of coolant, pulsing while at full power, anf fission product release have been revised to reflect the power peaking and more recent values of fission product release fractions in TRIGA fuel.

1.2 GENERAL DESCRIPTION

Figure 1 is a pictorial view of the reactor. The reactor is a heterogeneous pool-type nuclear reactor fueled with TRIGA and TRIGA-FLIP fuel modified to adapt to 4-element bundle assemblies. The coolant is light water which circulates through the core by natural convection. The core is reflected by water and graphite. Maximum steady-state power level is 1000 KW.

A 7 by 9 grid, surrounded by a core box, positions fuel, reflector, and control elements. Three shim-safety blades, a transient control rod, and a regulating blade control core reactivity. The control blades move vertically in two shrouds extending the length of the core. The grid box and control element drive mechanisms are supported by a suspension frame from the reactor bridge.

Cold, clean core excess reactivity is about 4.9 per cent reactivity. The safety blades provide a shutdown margin of 6 to 14 per cent ΔK_{eff}.

The proposed technical specifications for the facility are listed below as Section 1.4.

1.3 SUMMARY OF REACTOR DATA

Responsible Organization	The University of Wisconsin
Location	Madison, Wisconsin
Purpose	Teaching and Research
Fuel	
Type	TRIGA and TRIGA-FLIP High Hydride in 4 element clusters
Number of elements in standard 1000 KW Core	99 (25 Fuel Bundles)
Control	
Safety elements	Three vertical blades
Regulating-servo element	One vertical blade
Transient control	One rod

Experimental Facilities

Thermal column	One, 40-inch square graphite, ϕ th = 2 x 10^8
Beam Ports	Four, 6-inch diameter ϕ th = 1 - 3 x 10^{10} nv at shield side of shutter; about 8 x 10^{11} at core end of port
Pneumatic Tube	One, 2-inch (sample size 1-1/4 inch diameter by 5-1/2 inches long), ϕ th = 10^{12} nv.

Thermal neutron fluxes for isotope production include the above, plus large irradiation spaces outside the core with thermal neutron fluxes of around 1.3 x 10^{13} nv.

Reactor Materials

Fuel-moderator element	8.5 wt-% Uranium, 89.9 wt-% Zirconium, 1.6 wt-% Hydrogen, 1.5 wt-% Erbium in FLIP 20% U^{235} (70% for FLIP)
Cladding	20 mil stainless steel
Reflector	Water and Graphite
Coolant	Light Water
Control	Boral and stainless steel; Borated graphite for transient rod
Structural material	Aluminum
Shield	Concrete and Water

Dimensions

Pool	8 x 12 x 27-1/2 ft. deep
Standard 1000 KW core	15 x 17 x 15 inches high
Grid box	9 x 7 array of 3-inch modules

Control blades	10-1/2 inches wide
Fuel Element	
Diameter	1.41 inches
Length	30 inches

Predicted nuclear characteristics

1 MW Steady state:	
Maximum thermal neutron flux	3.2×10^{13} nv
Maximum fast neutron flux	3.0×10^{13} nv
2000 MW Pulse	
Maximum thermal neutron flux	6.5×10^{16} nv
Maximum fast neutron flux	6.1×10^{16} nv
Prompt temperature coefficient of reactivity	-1.26×10^{-4} $\Delta K/^{o}C$
Void coefficient of reactivity	$-.2 \times 10^{-4}$ $\Delta K/\%$ void
Prompt neutron lifetime	42 μsec STD Fuel, 18 μsec FLIP
Effective delayed neutron fraction	0.007

~~1.4 PROPOSED TECHNICAL SPECIFICATIONS~~

~~Numerical values given in these specifications may differ from measured values due to normal construction and manufacturing tolerances or normal accuracy of instrumentation.~~

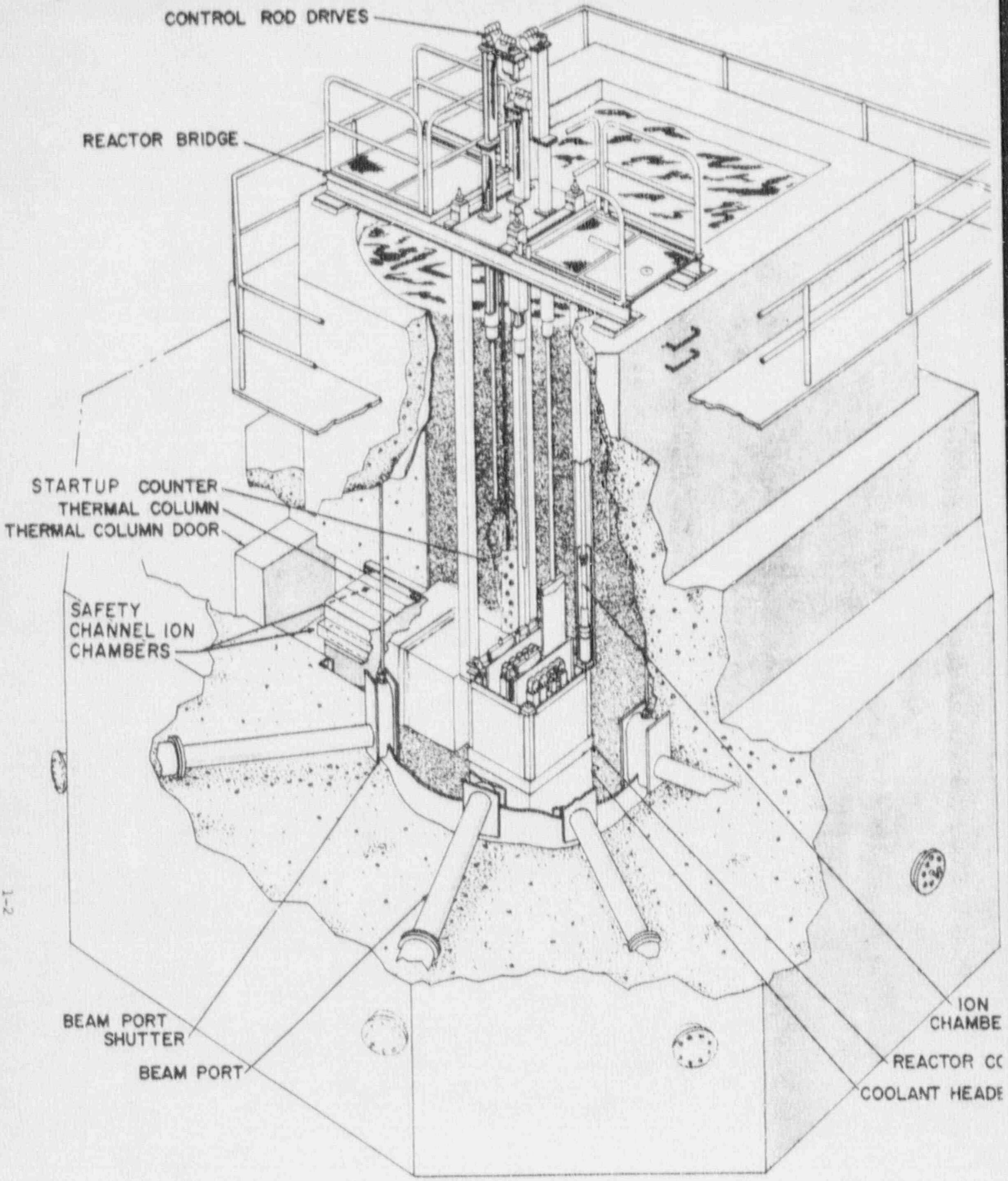


FIGURE 1

OPEN POOL REACTOR

Chapter 2

FACILITY DESCRIPTION

2.1 REACTOR CORE

2.1.1 Core Support

The core is suspended from an all-aluminum frame, Figure 1, which extends from the grid box to a height of about one foot above the pool surface. The hollow corner posts of the suspension frame serve as a guide~~s~~ for a gamma ~~nuclear instrumentation~~ detector~~s~~. 79-80 ann Report

The reactor bridge (mounted over the pool) supports the core suspension frame. The all-steel, prefabricated bridge was bolted together in the field and aligned with shims.

A locating plate, made of 1/2-inch steel, spans the upper end of the suspension frame. It is bolted to the bridge and aligns the four control blade drive mechanisms and the transient rod drive with the core. The five mechanisms work through individual clearance holes, each mechanism being secured to the locating plate. The plate and mechanisms are not removable as a unit to prevent accidental withdrawal of the control elements. The fission counter drive is mounted on a portion of the hand railing support structure.

An aluminum coolant header (not shown in Figure 1) mates with the bottom of the grid box and forms a transition to the coolant piping originally provided for future use with a forced convection cooling system. An opening in the side of the header, 24 inches wide by 12 inches high, allows cooling water flow for natural convection. A diffuser pump and jet above the core deflects the cooling water streaming from the core to reduce N^{16} activity above the core.

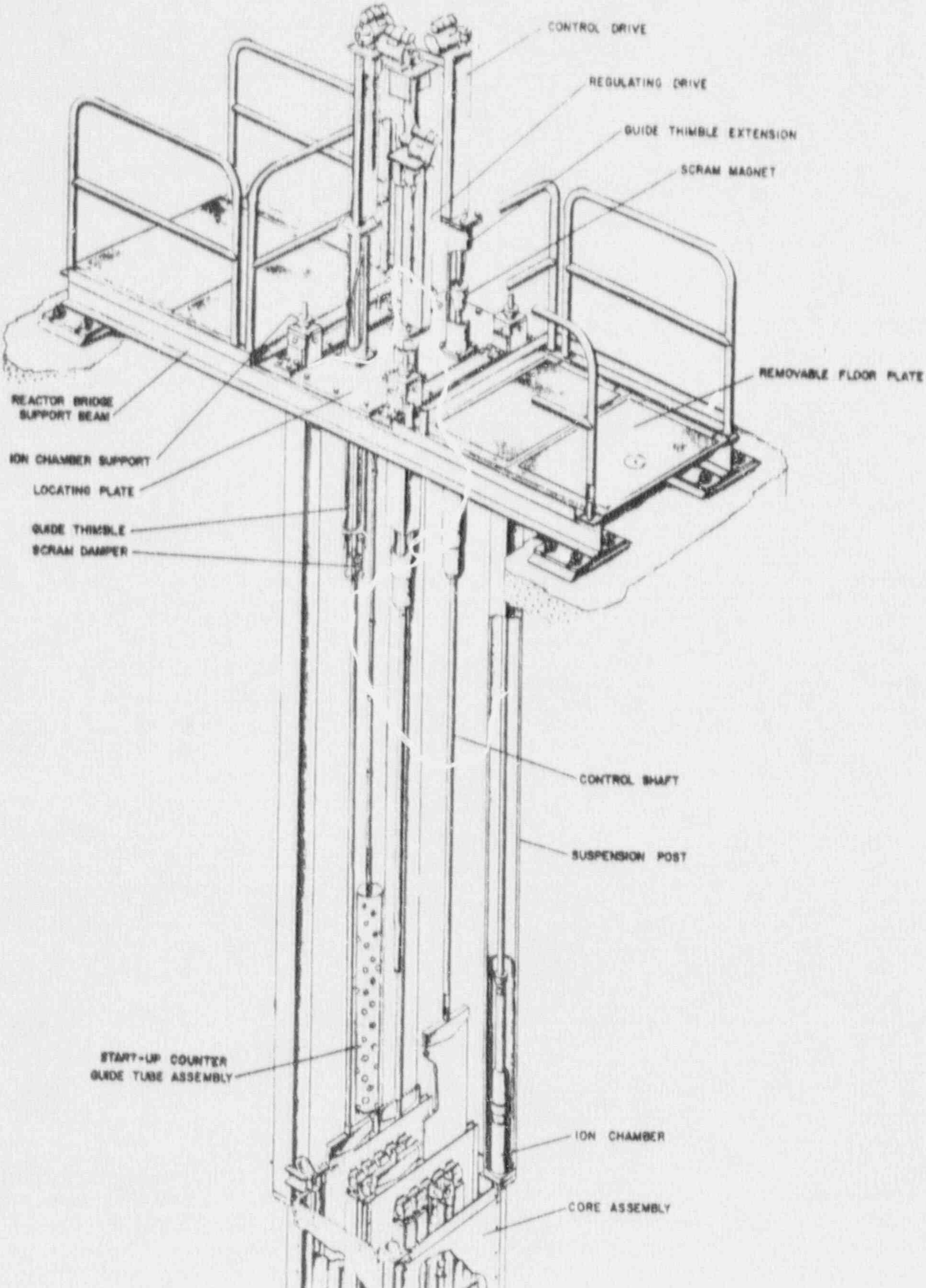


FIGURE 1

CORE SUSPENSION

2.1.2 Grid Box

The core elements are supported and enclosed on four sides by the grid box. The grid box is approximately 28 inches long, 24 inches wide, and 36 inches high. The bottom is an aluminum grid plate with a 9-by-7 array of square holes, spaced to conform with the basic 3-inch element module. The sides of the grid box direct the convection current of the cooling water through the core. Four corner posts attached to the lower end of the suspension frame support the grid box. All parts, except for mechanical fasteners, are made of aluminum.

2.1.3 Fuel

The fuel is of the TRIGA four-element bundle type developed to provide a simple means of converting reactors using flat-plate fuel to TRIGA reactors. Figure 2 shows a four-element bundle and a three-element bundle.

The four-element bundle consists of bottom adapter, top adapter, and four TRIGA elements.

The bottom adapter fits the existing grid plate as did the original fuel elements. The end fittings on individual TRIGA elements are threaded into the bottom adapter until a flange on the element seats firmly against the adapter, providing rigid cantilever-type support. (See Figure 3.)

The top adapter serves both as a handling fitting and as a spacer for the upper ends of the fuel elements. A sliding fit between this adapter and the fuel element end fittings allows for differential expansion of the elements. This top fitting can be removed with remote handling tools to disassemble the bundles for any required measurements or for shipping spent elements for reprocessing.

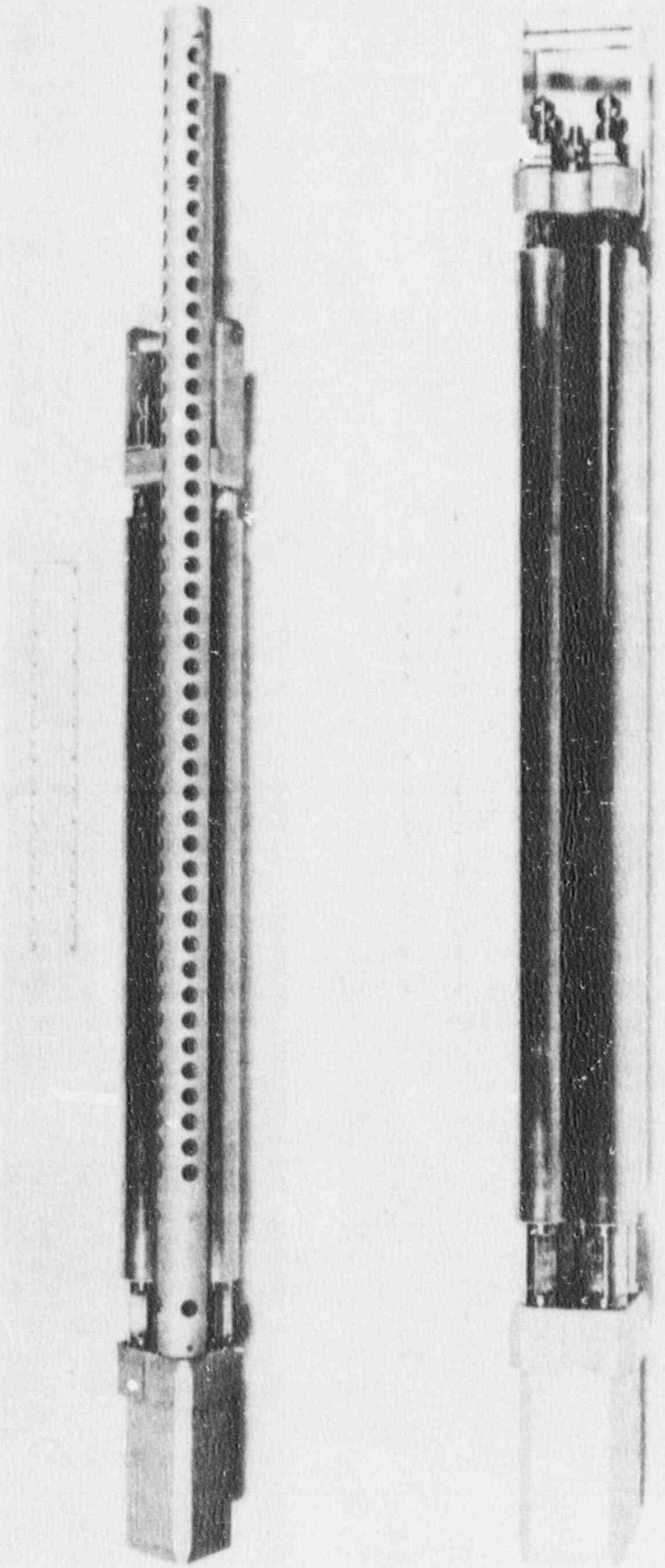


FIGURE 2

TRIGA fuel elements

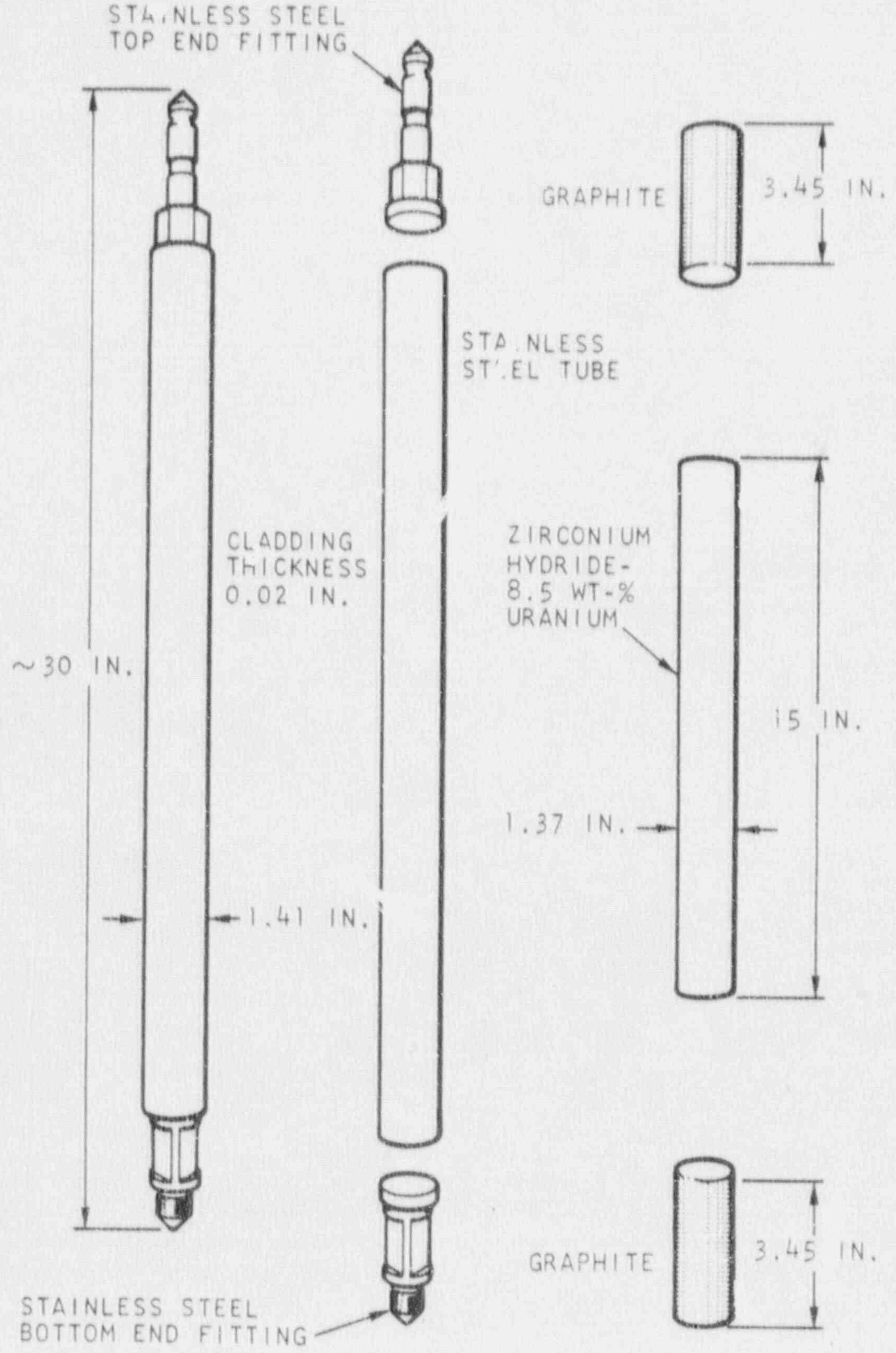


FIGURE 4

MIII-3C

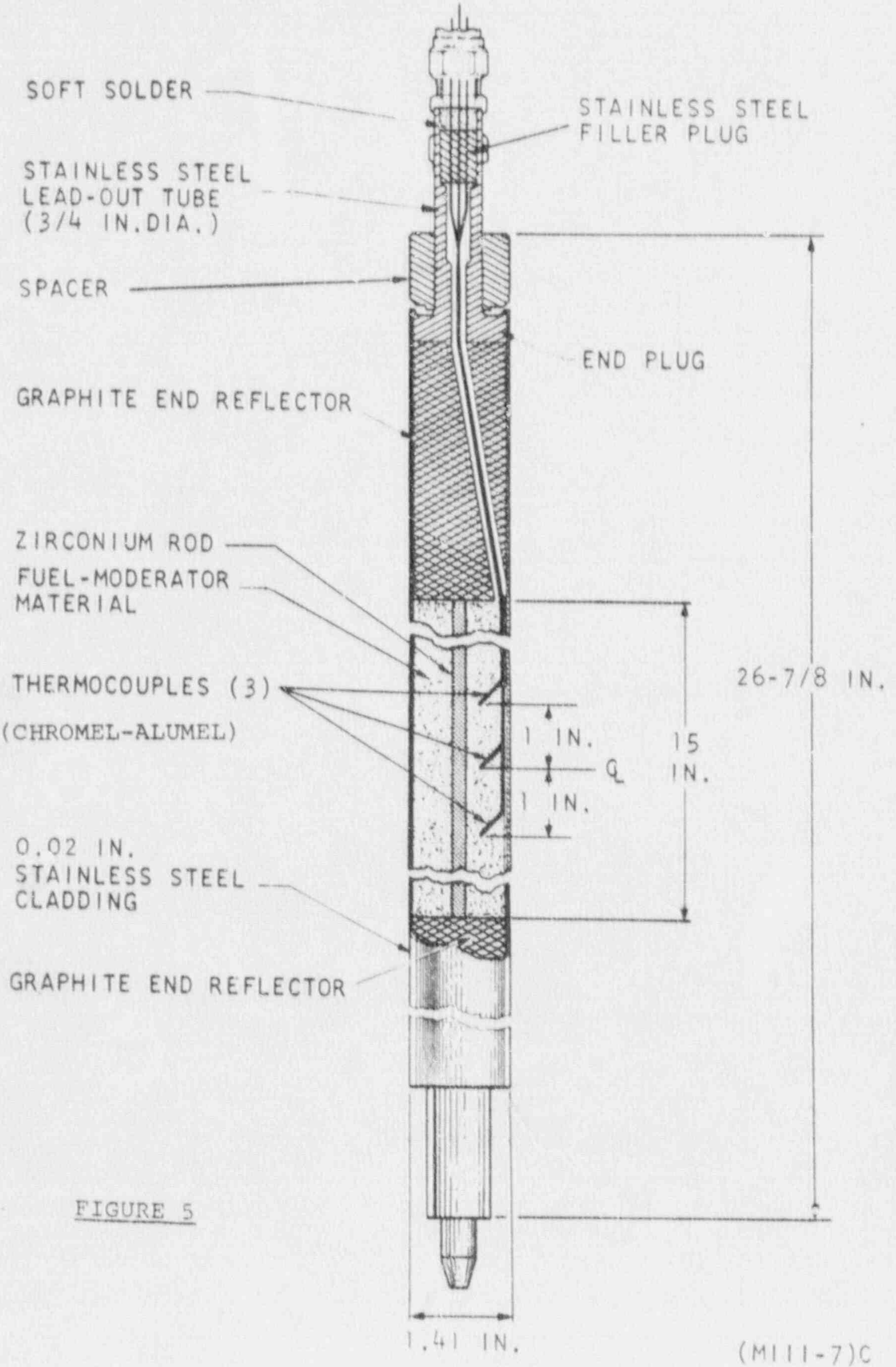


FIGURE 5

(MIII-7)C

2.1.4 Reflectors

The reflectors are standard General Electric Company reflectors as used in the Washington State University Reactor. The nominal 3-inch reflector elements are made of AGOT grade graphite clad with aluminum (see Figure 6). Reflector element lifting handles are diagonal to facilitate identification when viewing the core and storage racks.

2.1.5 Safety Blade

Reactor control for startup and shutdown is accomplished by three blade-type control elements, Figure 7, with a total shutdown worth between 6.9 and 11 per cent ΔK_{eff}. The poison section is boral sheet (boron carbide and aluminum sandwiched between aluminum side plates). It is 40.5 inches long, 15 inches providing active control of the core and the remaining 25.5 inches connecting the poison section to the drive tube.

Each safety blade is guided throughout its travel by a shroud shown in Figure 8. The shroud consists of two thin aluminum plates 38 inches high, separated by aluminum spacers to provide a 1/8-inch water annulus. The shrouds can be removed, if necessary, by use of a grapple hook. Small flow holes at the bottom of the shroud minimize the effect of viscous damping on the scram time.

2.1.6 Regulating Blade

The regulating blade, Figure 9, is a stainless-steel sheet. about 11 inches wide and 40 inches long, supported and guided in the same manner as the safety blades described in Section 2.1.5. It compensates for small changes of reactivity during normal reactor operation and may be actuated by a servo-control channel.

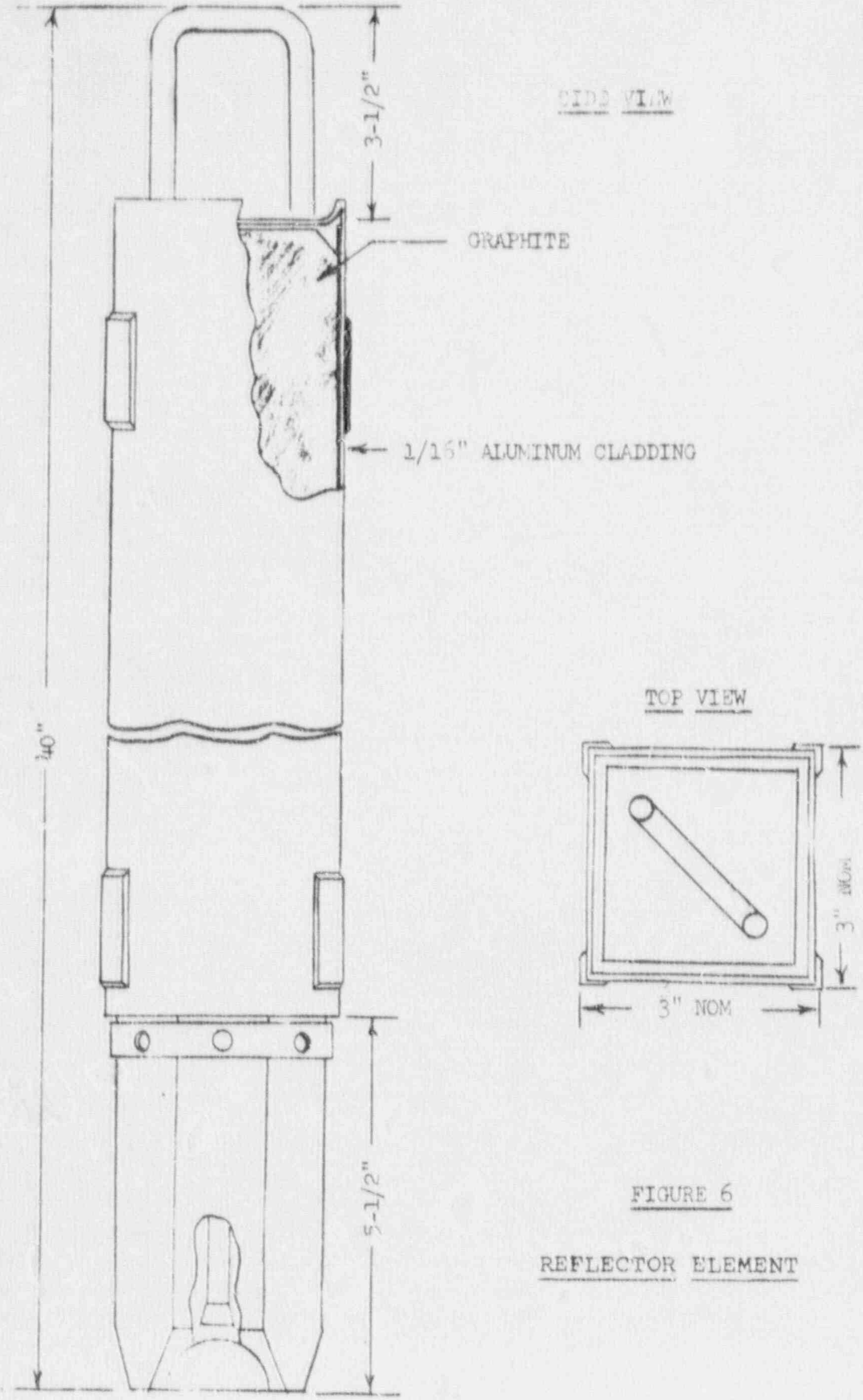
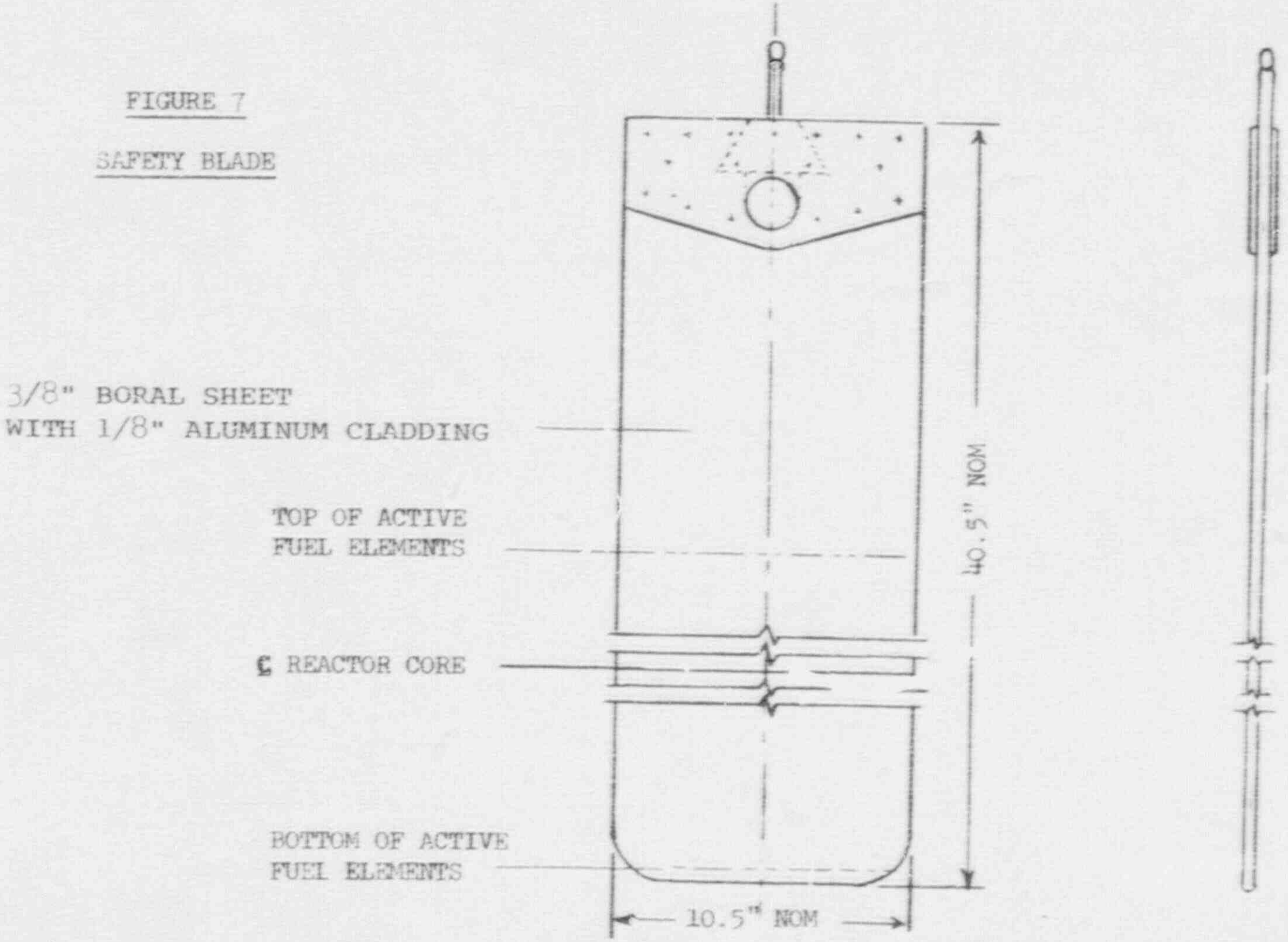


FIGURE 6

REFLECTOR ELEMENT

FIGURE 7

SAFETY BLADE



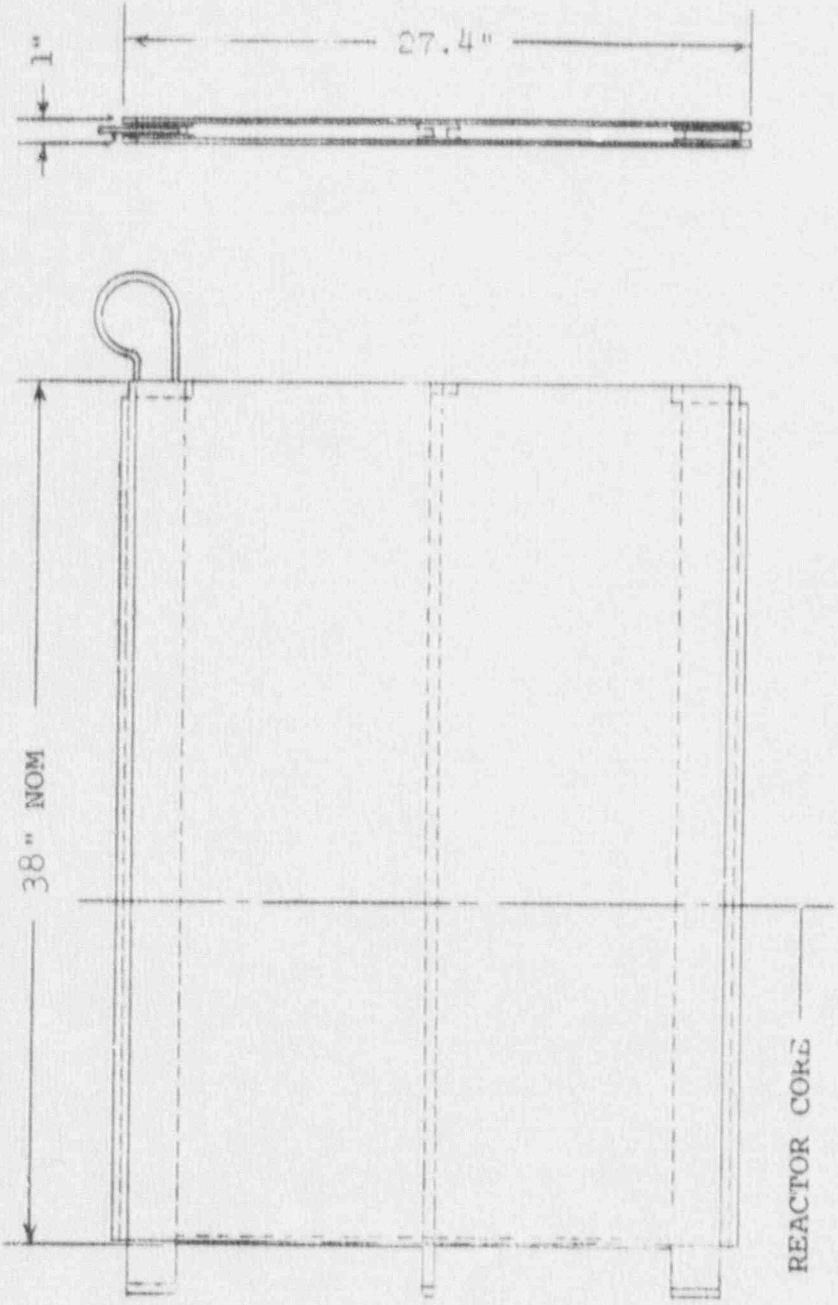


FIGURE 8
SHROUD ASSEMBLY

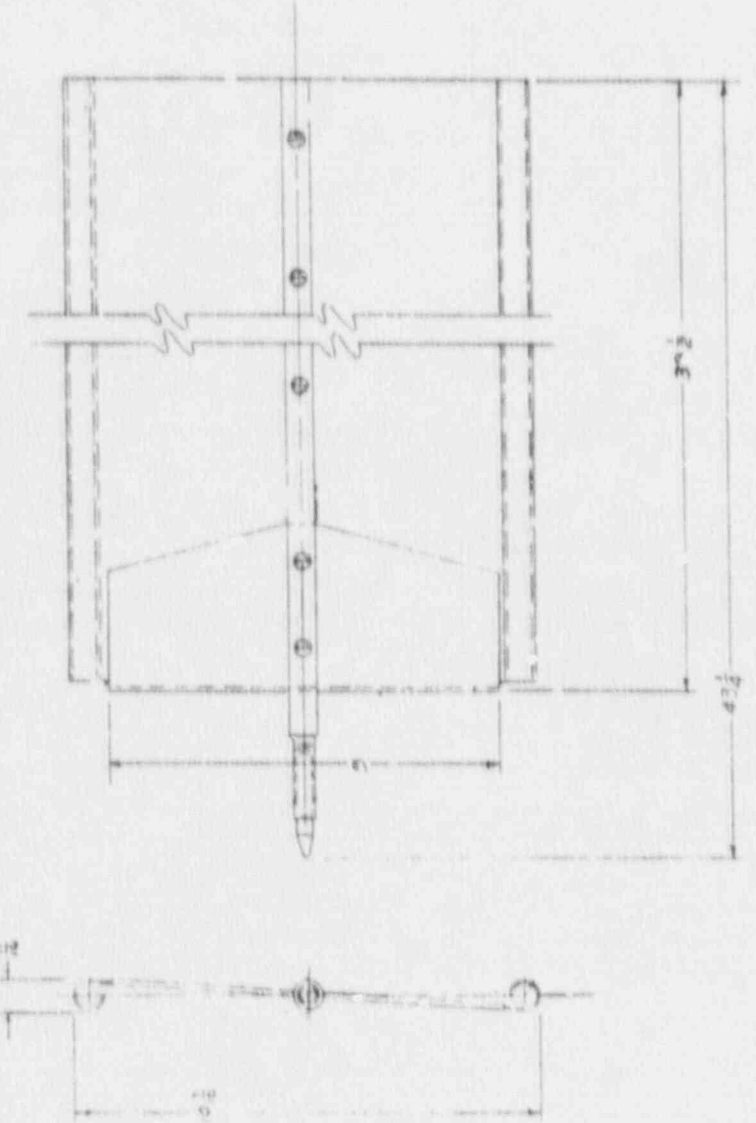


FIGURE 9

REGULATING BLADF

2.1.7 Transient Control Rod

The transient control rod is borated graphite contained in a 1-1/4" diameter stainless steel or aluminum tube (Figure 10). The poison section is 15 inches long. This rod is guided laterally by the aluminum guide tube in a special three-element fuel bundle. A hold-down tube extends from this guide tobe to the top of the reactor structure and holds the three-element bundle in place during transient rod movement.

2.1.8 Neutron Source

The neutron source is a 100 mg radium-beryllium source irradiated to give an output greater than 10^7 neutrons/second. It fits into a source holder which, in turn, fits into a radiation basket occupying one grid module adjacent to the active core. The source is usually left in for full power operation, and will, with the expected operating cycle, maintain its output of about 10^7 neutrons/second.

2.1.9 Core Arrangement

The use of the reactor as a training and research tool requires flexibility of core arrangement. These arrangements are subject, however, to the following criteria:

a. A mixed core must contain at least four FLIP elements

b. FLIP fuel must be located in a central contiguous region

c. The core must be a close packed array except for:

    (1) a maximum of two, 3-inch square lattice positions which must contain an in-core experiment or experimental facility;

    (2) single fuel element positions.

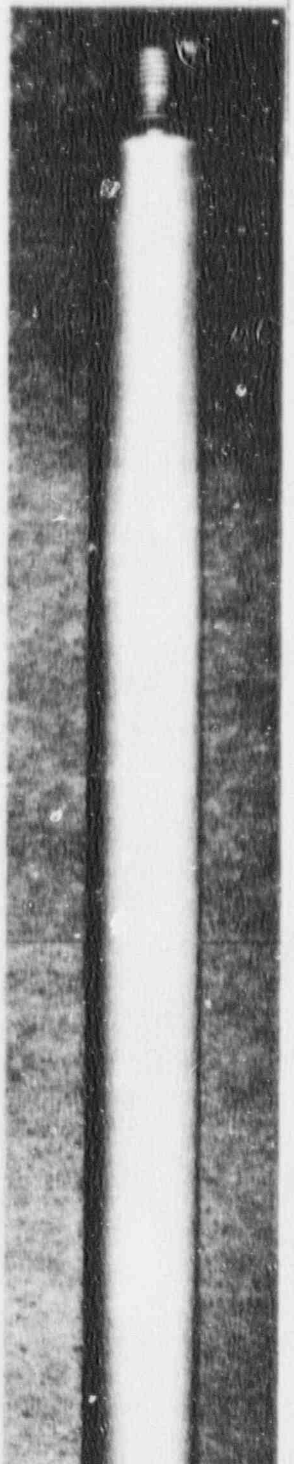


FIGURE 10

Transient control rod

d. The reactor may not be operated with a core lattice position vacant except for positions on the core periphery.

e. Calculations indicate that operation will be within technical specification limits on power generation per element and fuel temperature.

2.2. DRIVE MECHANISMS

2.2.1 Safety Blade Drive

The drive mechanism for the safety blades, Figure 12, right picture, includes a reversible electric motor with an integral worm-gear assembly to reduce speed and prevent drift of the control element. A mechanical slip clutch on the output shaft limits the force on the control blade to approximately 75 pounds. A ball-bearing screw and nut are used to raise and lower the control element.

Each blade is coupled to its mechanism through an electromagnet that provides gravity scram when de-energized. In order to minimize friction and possible binding, no more guide bearings than necessary for proper alignment are used, and large clearances are provided on guide bearings. All lubricants are sealed to prevent leakage into the reactor pool. The working parts are enclosed in a housing, and limit switches are protected with suitable covers.

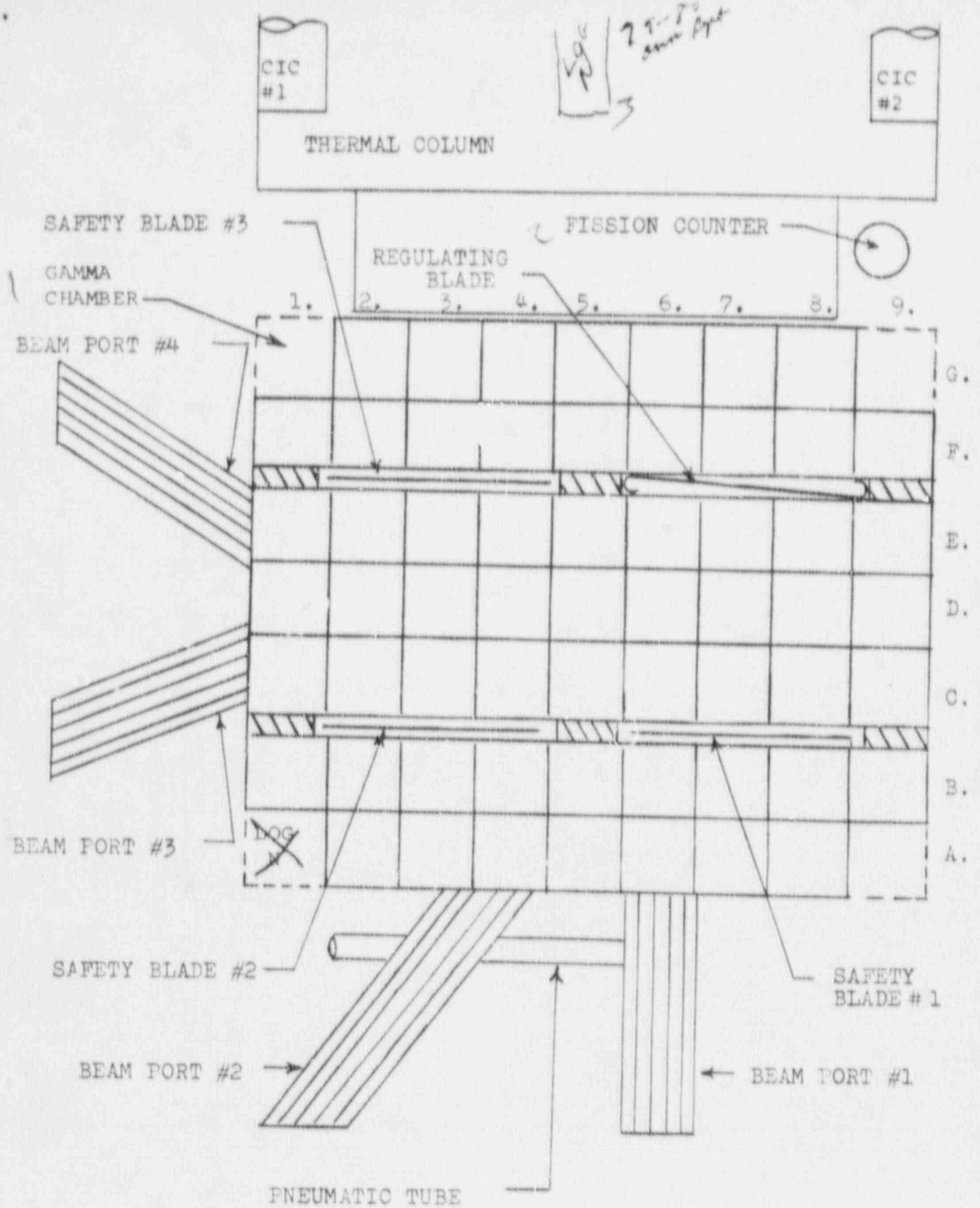
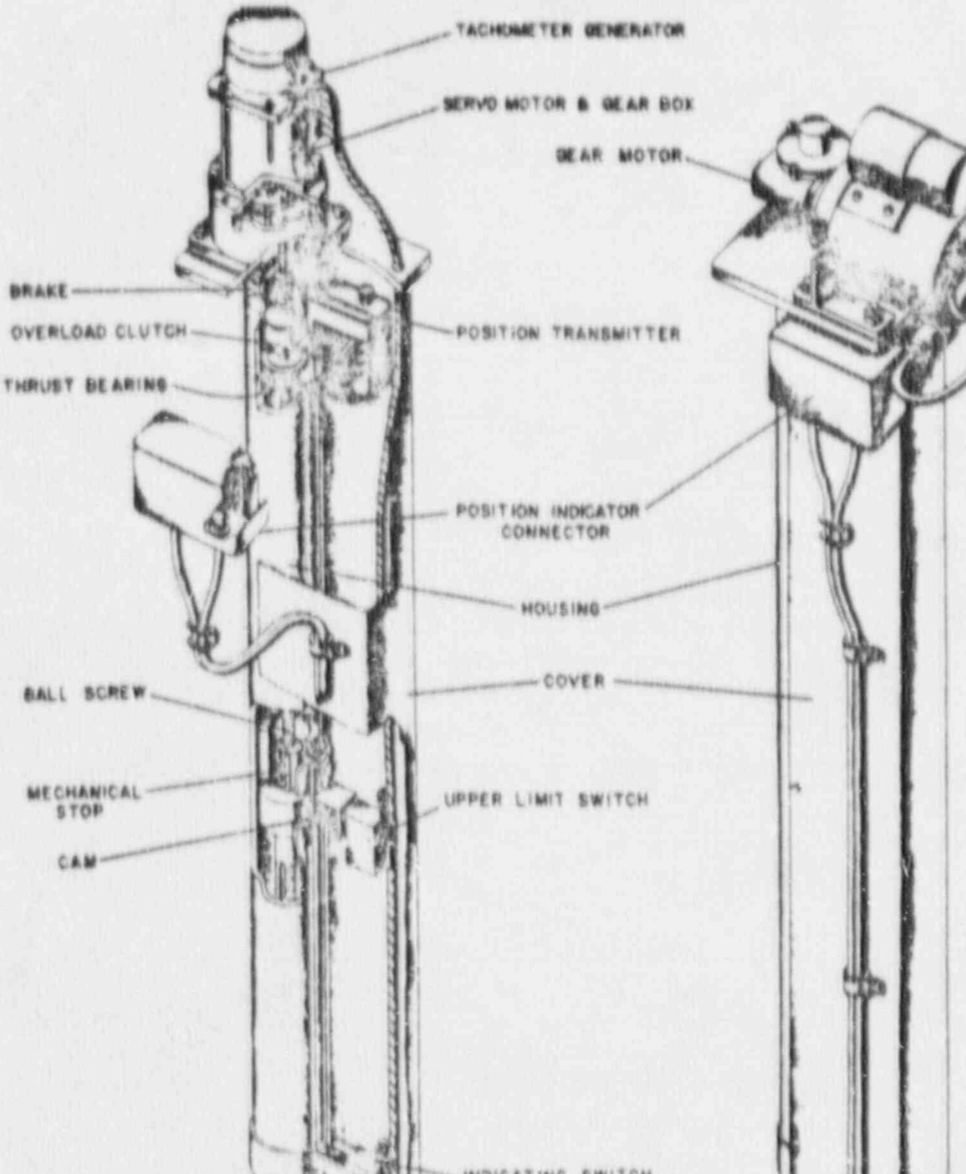


FIGURE 11

GRID ARRANGEMENT



SERVO DRIVE

CONTROL DRIVE

Figure 12

DRIVE MECHANISMS

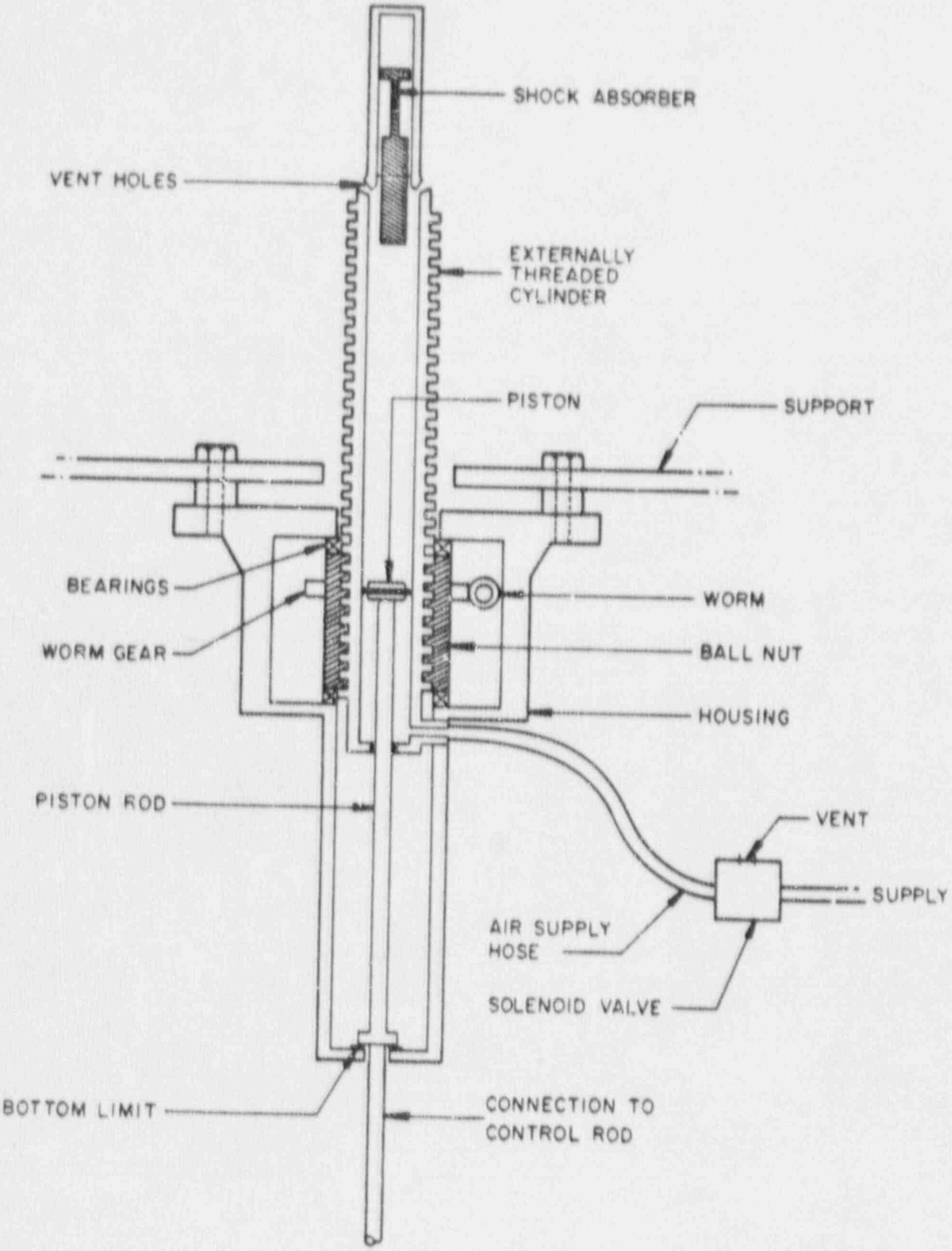


Fig. 13 -Transient-rod drive assembly

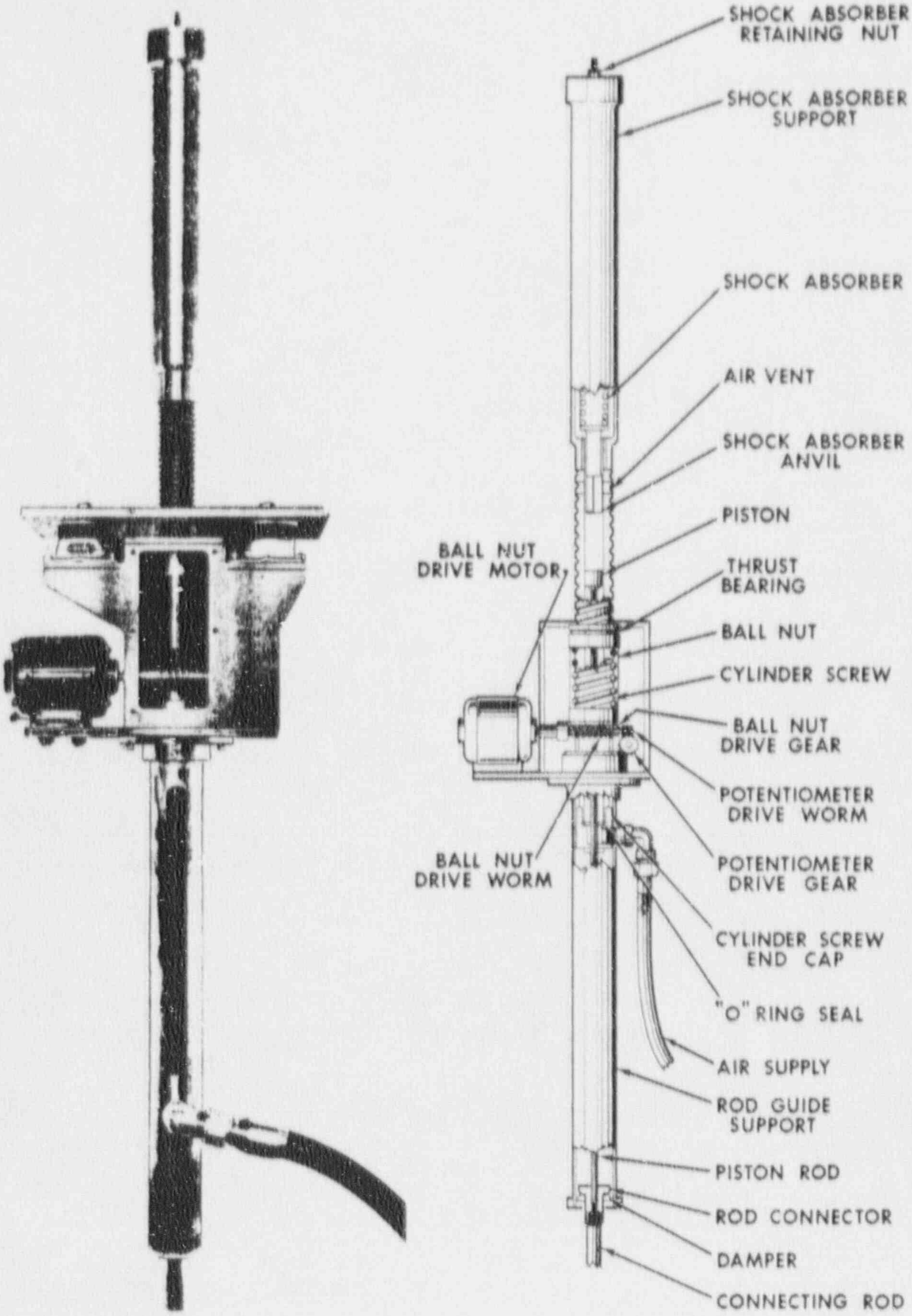


Fig. 14. -Pneumatic-electromechanical transient-rod drive

The safety blade drive mechanism operates through a stroke of 16 inches at a normal speed of approximately 14.5 inches per minute in either direction. Coasting of the mechanism is limited to less than 0.05 inch. Instruments give a continuous indication of position, accurate to 0.02 inch, and limit switches at the ends of the stroke shut off the drive motor and operate indicating lights. In addition, a limit switch within the scram magnet gives remote indication when the magnet engages the control element.

The control shaft joining each drive with its blade is made up of an armature rod, connecting shaft, piston, and lower tube. The piston and armature rod are screwed to the top end of the connecting shaft. The bottom end of the shaft is pinned into the lower tube. A polyethylene sleeve-bearing aligns the control shaft and limits radial play to 1/32 inch. The bearing guide tube is attached to the suspension frame.

A steel armature disc is welded to the top of the armature rod. During reactor operation, the disc is held by the scram magnet on the drive mechanism. When the reactor is scrammed, the magnet de-energizes and releases the control shaft and blade. The release completely separates the blade from the drive mechanism. The element is free to fall within 60 milliseconds of presence of a scram condition, and it then drops into the core under the force of gravity. A dashpot above the guide tube bearing receives the piston and decelerates the shaft over the last five inches of its 16 inch travel. In case of power failure, scram is automatic. To recover the control element after scram, the mechanism is run down and the magnet picks up the element.

2.2.2 Regulating Blade Drive

The regulating blade servo-controlled drive (Figure 12, left picture) is driven by a servo motor through a spur-gear train and components otherwise similar to those in the safety blade drives. Remote position indication is again provided. A solid coupling replaces the holding magnet; therefore the regulating blade cannot scram. The maximum speed of travel is 17 ± 0.5 inches per minute with a total stroke of 16 inches.

A control shaft joins the regulating blade to its drive mechanism. The shaft is aligned by a polyethylene sleeve bearing similar to that used for the safety blade shafts, but without a dashpot, with radial clearance between shaft and bearings of 1/32 inch. The guide tube is attached to the suspension frame, Figure 1.

2.2.3 Fission Counter Drive

The fission counter is positioned by a Morse chain-and-sprocket drive giving a total motion of 60 inches. The drive has adjustable stops and covers the range from source level to full power with its four positions. Position is indicated at the console, and the control element drives cannot be withdrawn when the fission counter is in motion.

2.2.4 Transient-Rod Drive

To allow transient operation, use is made of a pneumatic-electromechanical drive system to eject a predetermined amount of the transient rod from the core. This drive system, located on a special mount attached to the locating plate, is shown in Figures 13 and 14.

The pneumatic portion of the pneumatic-electromechanical drive, referred to herein as the "transient" rod drive, is basically a single-acting pneumatic cylinder. A piston within the cylinder is attached to the transient rod by means of a connecting rod. The piston rod passes through an air seal at the lower end of the cylinder. Compressed air is admitted at the lower end of the cylinder to drive the piston upward. As the piston rises, the air being compressed above the piston is forced out through vents at the upper end of the cylinder. At the end of its stroke, the piston strikes the anvil of a shock absorber. This piston is thus decelerated at a controlled rate during its final inch of travel. This action minimizes rod vibration when the piston reaches its upper-limit stop.

An accumulator tank mounted beneath the removable floor plate of the bridge (Figure 1) stores the compressed air that operates the pneumatic portion of the transient rod drive. A three-way solenoid valve, located in the piping between the accumulator tank and the cylinder, controls the air supplied to the pneumatic cylinder. De-energizing the solenoid valve interrupts the air supply and relieves the pressure in the cylinder so that the piston drops to its lower limit by gravity. With this operating feature, the transient rod is inserted in the core except when air is supplied to the cylinder.

The electromechanical portion of the transient rod drive consists of an electric motor, a ball-nut drive assembly, and the externally threaded air cylinder. During electromechanical operation of the transient rod, the threaded section of the air cylinder acts as a screw in the ball-nut drive assembly.

These threads engage a series of balls contained in a ball-nut assembly in the drive housing. The ball-nut assembly is in turn connected through a worm-gear drive to an electric motor. The cylinder may be raised or lowered independently of the piston and control rod by means of the electric drive. Adjustment of the position of the cylinder controls the upper limit of piston travel, and hence controls the amount of reactivity inserted for a pulse.

A system of limit switches is used to indicate the position of the air cylinder and the transient rod. Two of these switches, the Drive Up and Drive Down switches, are actuated by a small bar attached to the bottom of the air cylinder. A third limit switch, the Air switch, is actuated when the piston is held in the cylinder by air pressure.

2.3 COOLING AND LIQUID WASTE SYSTEMS

2.3.1 Pool

The aluminum-lined concrete pool, Figure 15, is 8 feet wide, 12 feet long, and 27-1/2 feet deep. It is penetrated by experimental ports. A pit, 6 feet by 4 feet by 6 feet deep, with shielded cover, is provided for storage of fuel elements awaiting shipment to a reprocessing facility and for storage of fuel in a shielded location, should maintenance of the in-pool portions of the reactor be necessary. Three aluminum fuel storage baskets (with aluminum clad cadmium poison plates) fit inside the pit and hold up to 60 fuel bundles for storage. The multiplication constant (K_{eff}) for the three storage baskets with full complement of fuel elements is less than 0.8. Three additional fuel racks, each holding up to 9 four-element bundles, are attached to the pool wall at about the level of the grid box. The multiplication constant for these racks, when fully loaded, is less than 0.8.

The pool water is kept within the following limits:

Temperature (at Core cooling
water inlet) . . . . . $<130^{\circ}$F

Resistivity . . . . . . $>2 \times 10^{5}$ ohm - cm

Activity . . . . $<5 \times 10^{-5}$μc/ml

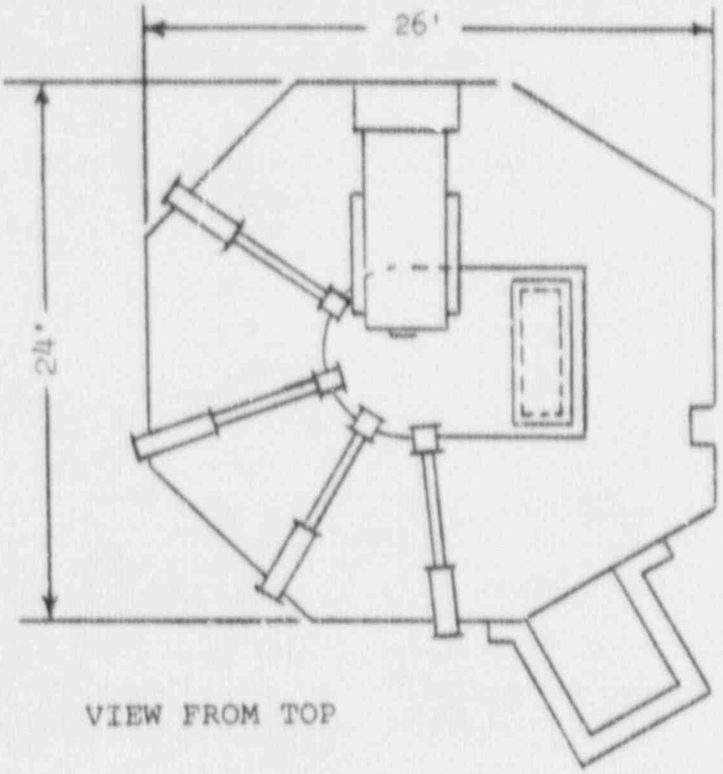
(See Section 3.1.4 for further information on pool activity.)

The reactor core is cooled by natural convection of pool water through the core. The 130°F temperature limit is imposed by demineralizer resin tolerance and by humidity control considerations.

The resistivity limit is set to reduce corrosion effects, extending the expected lifetime of the fuel elements and controlling water radioactivity. Routine checks of resistivity are made to determine the necessity of regenerating the demineralizer.

The radioactivity of the pool water is continuously monitored by an area monitor station located near the demineralizer. Should the pool water reach the activity limit above, the reading on this area monitor will increase during periods when the reactor is not operating (N^{16} activity may mask this activity during operation). In addition, water samples are routinely analyzed for activity by other methods which give a more exact identification of quantity and type of activity present.

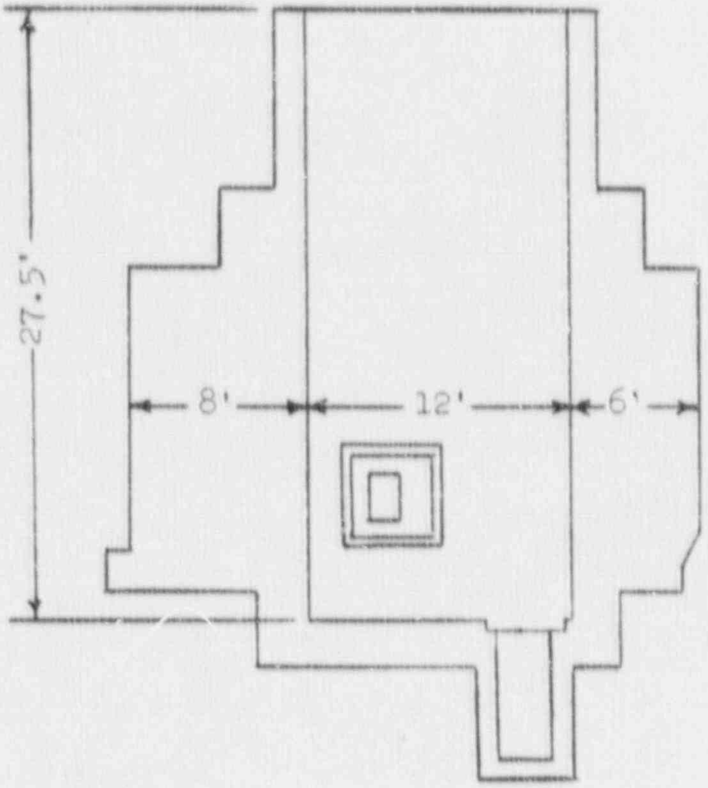
No problem is anticipated in maintaining pool water radioactivity below the indicated limit.



VIEW FROM TOP

FIGURE 15

REACTOR POOL



VIEW LOOKING SOUTH

2.3.2 Cooling System

The pool water is cooled by the system shown schematically in Figure 16. The secondary system is designed to operate year round, and is similar to other installations that must operate at subzero ambient temperatures. The secondary system has a combination of manual (seasonal) and automatic (day to day operation) controls which maintain the temperatures about as indicated for 1.0 MW operation.

The intake and outlet diffusers are constructed to preclude draining more than 1 foot of water even in the case of a pipe rupture. This will maintain at least 19 feet of water above the active core.

The secondary side of the heat exchanger is operated at a higher pressure than the primary side both when the pumps are running and under static conditions. A leak in the heat exchanger would result in secondary water entering the primary system. Such leakage could be detected in two ways. First, the secondary water influx will increase conductivity in the pool water. Second, the pool level float switch will be actuated by high water level should as much as 150 gallons of secondary water enter the pool water system.

The cooling system will dissipate 1.0 MW year round with primary temperatures approximately as indicated in Figure 16. The system components have enough capacity to enable dissipation of at least 2 MW for wet bulb temperatures less than 75°F.

The primary system continuously circulates pool water through the exchanger. Pump Number one circulates secondary water from the pumps through either the tower or the bypass line. The automatic diverting valve is controlled by sump water temperature and operates so there is either full flow through the tower or full flow through the bypass. Pump Number two circulates water from the sump through the heat exchanger. The automatic diverting valve in this loop is controlled by primary coolant temperature at the heat exchanger exit.

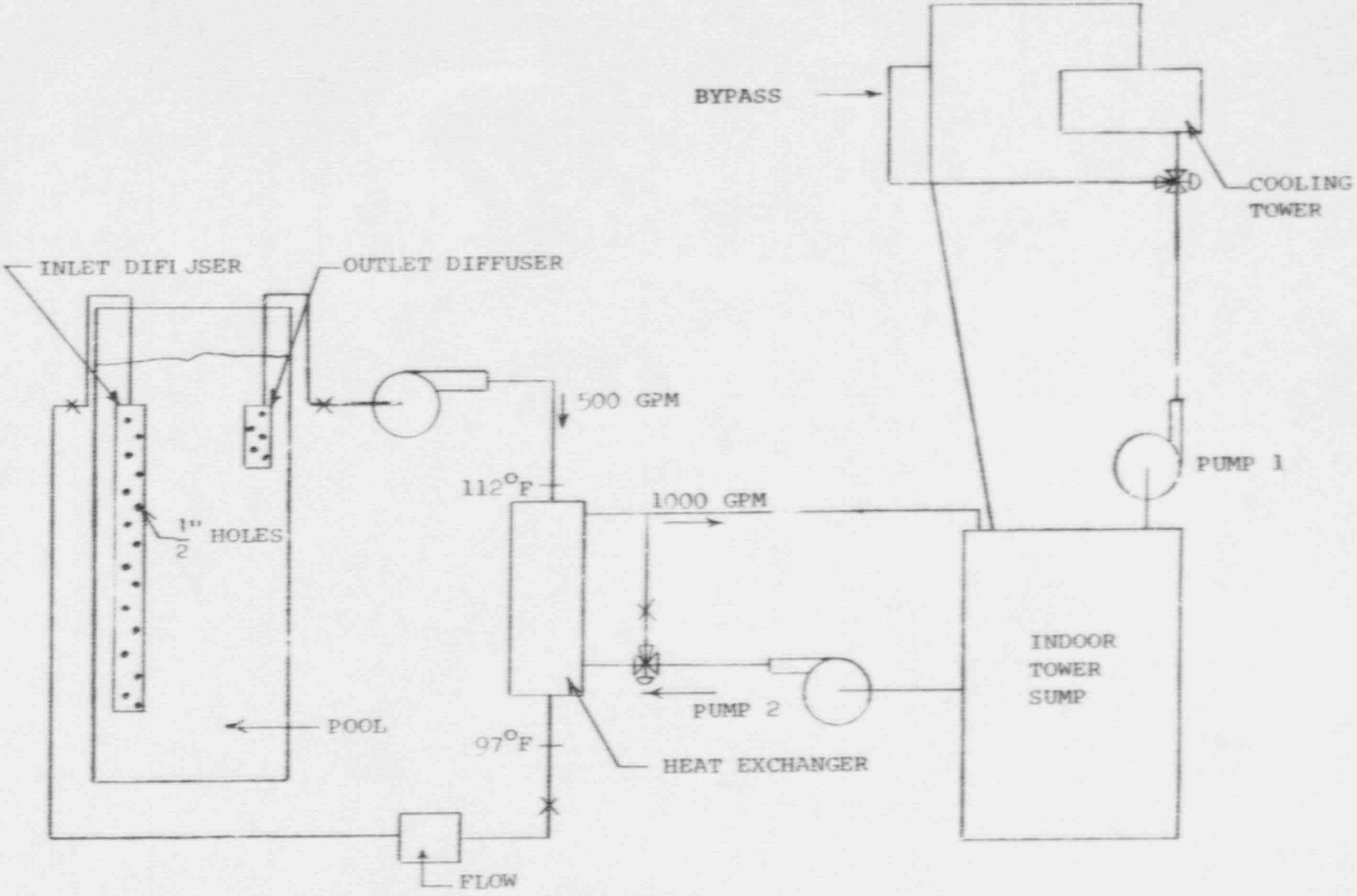


FIGURE 16

COOLING SYSTEM - 1 MW OPERATION

The valve operates to allow the proper amount of secondary flow through the heat exchanger to maintain primary coolant temperature at the desired point. Cooling tower fan speeds are manually controlled for seasonal variations.

Nitrogen 16 suppression is accomplished by a jet-type diffuser system. The system pumps about 80 gallons of water per minute from near the pool surface through a single nozzle having a 0.75 inch wide, 6.5 inches long opening. The nozzle is located 5.5 feet above and 0.5 feet east of the core, with the diffusing stream directed downward at a 45 degree angle toward the west end of the pool.

The pump for the N^{16} suppression system is located on the outside east face of the reactor's concrete containment structure, about 8 feet below the pool surface. The system is constructed so as to preclude draining more than one foot of water from the pool in the event of a pipe rupture.

2.3.3 Pool Make-up and Clean-up System

The pool make-up and clean-up system is shown schematically in Figure 17. Water is circulated from the pool surface, through the pump, through the demineralizer, and then into the pool under the core box and coolant header. The pump maintains about 10 gallons/minute flow through the demineralizer. The demineralizer is a mixed-bed type with provisions for regeneration of resins or discharge of spent resin and loading with new resin. A water softener supplies softened water for regeneration of the demineralizer.

Normally, make-up water is supplied by the still. The still delivers water to a storage tank from which it is pumped (by the pool recirculating pump) into the pool to maintain pool water level. An alternate and more rapid method allows water to be fed through the demineralizer into the pool. In either case, impurities in make-up water are reduced to less than 1 ppm before going into the pool.

Wastes from regeneration of the demineralizer are discussed in the next section.

Flow from the demineralizer to the pool is through valve 11, check valve 709 which prevents back flow, and valve 719 into the 8 inch pipe loop and into the bottom of the grid box. The 8 inch line is equipped with a siphon breaker at the top of the pool so that rupture of the line at the demineralizer outlet or of the 8 inch line outside the shield cannot drain the pool to a level that will uncover the core. A second 8 inch line is flanged off on both ends. The 8 inch lines were originally installed to allow a forced-convection cooling mode, but the lines are used only as indicated above.

A two inch line whose rupture could have caused loss of pool water has been permanently plugged inside the concrete shield and is presently sealed off outside the shield. A pool drain line and valve have been eliminated.

Should valve numbers 5 (shown in both figures 17 and 18) be left open upon placing the system in its normal operating condition, as much as 400 gallons of pool water could be pumped to the holdup tank. No further loss of water would then occur, since check valve 709 will prevent reverse flow from the 8 inch pipe loop to the demineralizer.

All operations involving the make-up and clean-up systems are performed by written checklist-type procedures designed to prevent draining of the pool.

Change 11-80

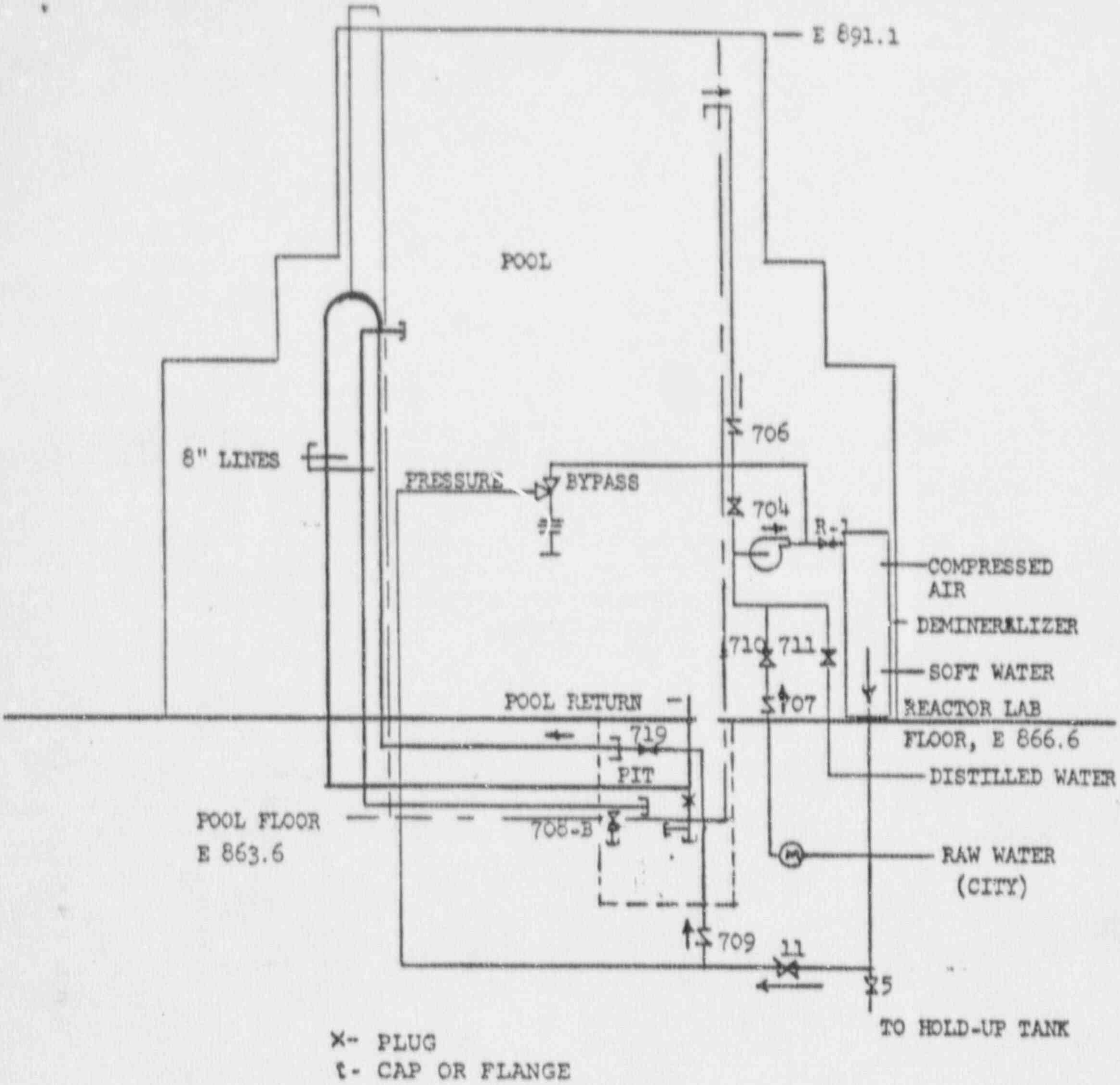


FIGURE 17

POOL MAKEUP AND CLEANUP SYSTEM

Change 11-80

2.3.4 Waste Disposal System

Figure 18 shows schematically the waste disposal system The hold-tank is a 2,000 gallon stainless steel tank buried beneath the Reactor Laboratory floor. Waste water from regeneration of the demineralizer flows through valve number 5 (normally closed) into the tank during regeneration. Valve number 713 is normally open, allowing pool overflow or pool curb drains to empty directly into the hold-tank. Valve 712 is normally closed, but it may be opened to allow draining flooded beamports. Once waste water is in the tank, it must be pumped out into the sanitary sewer system for ultimate disposal. The contents of the tank are not pumped out until it is determined that the concentrations and total quantities are below the applicable limits in 10 CFR 20. A senior operator's authorization is required before initiation of waste disposal into the sewer system. Since a valve must be opened and a pump turned on to effect discharge, accidental discharge is not to be expected.

Tank level is indicated in the demineralizer area, and an alarm sounds should the tank be filled to its maximum capacity. Provision is made for obtaining samples of the liquid in the tank for analysis.

The tank is vented to the roof of the Reactor Laboratory. No air intakes are located nearby, so personnel will not be exposed to any fumes that might be given off. The Reactor Laboratory floor drain also feeds into the hold-tank.

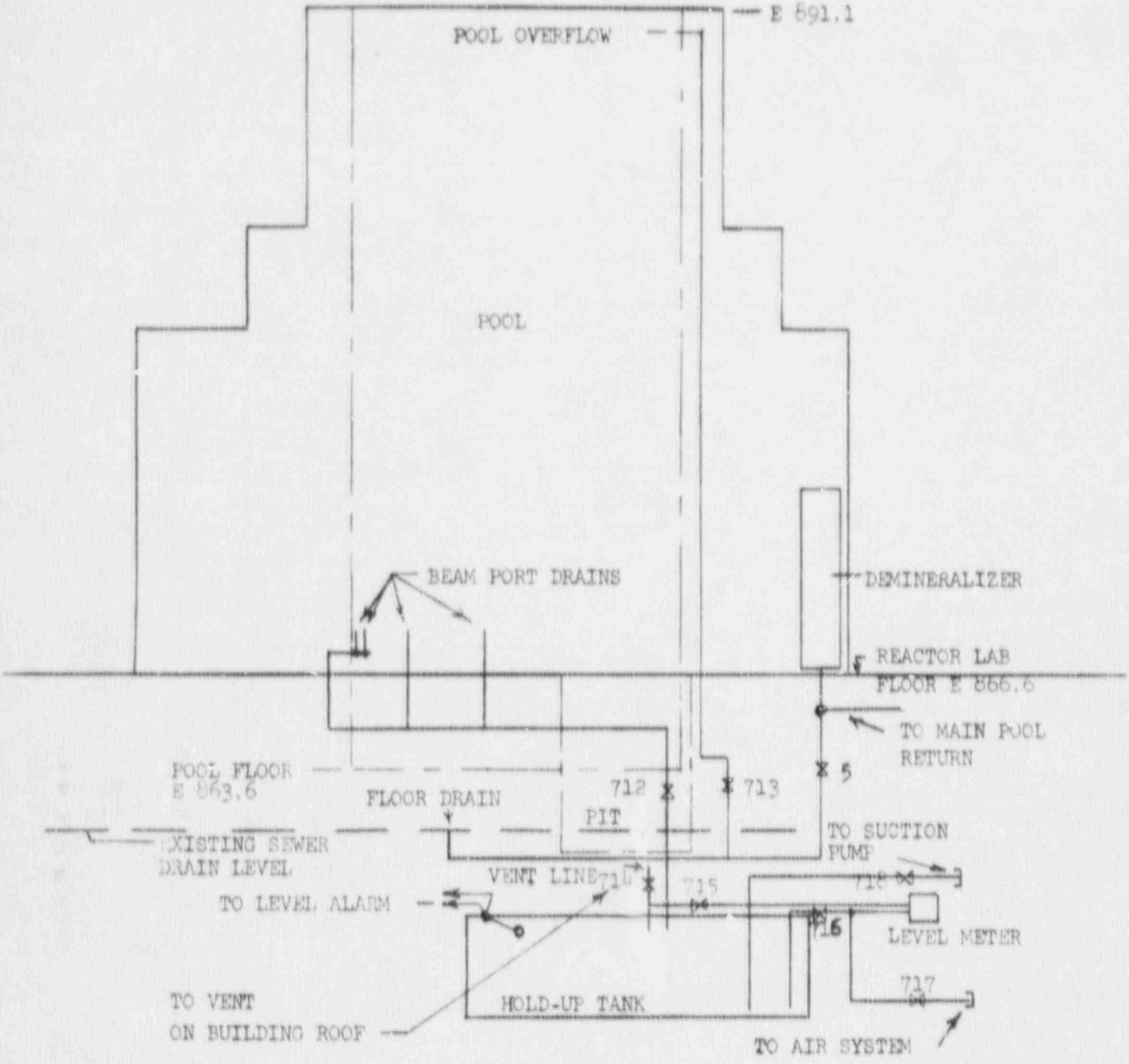


FIGURE 18

WASTE DISPOSAL SYSTEM

2.4 EXPERIMENTAL FACILITIES

Facilities are provided to permit use of radiation from the reactor in experimental work without endangering personnel. Facilities provided with this reactor include four beam ports, a thermal column, and pneumatic and hydraulic irradiation systems.

2.4.1 Thermal Column

The thermal column, Figure 19, is a graphite-filled, horizontal penetration through the biological shield which provides neutrons in the thermal energy range (about 0.025 ev) for irradiation experiments. The column, which is about 8 feet long, is filled with about 6 feet of graphite. A small experimental air chamber between the face of the graphite and the thermal column door has conduits for service connections (air, water, electricity) to the biological shield face. Detectors for the safety channels are located within the thermal column.

Personnel in the building are protected against gamma radiation from the column by a dense concrete door which closes the column at the biological shield. The door moves on tracks set into the concrete floor perpendicular to the shield face.

A ventilation system maintains a low pressure within the thermal column so that air flow is into the column when the door is open. The door is gasketed so that air flow is very small when the door is closed. When the door is opened, however, an air velocity of about 40 feet per minute into the column prevents the A^{41} activity from diffusing into the Reactor Laboratory. Section 2.4.3 contains further information on the ventilation system for the thermal column and beam ports.

An annunciator is activated whenever the thermal column door is not fully closed. In addition, an area monitor beside the thermal column door will give an alarm should the reactor be operated at a substantial power with the door open.

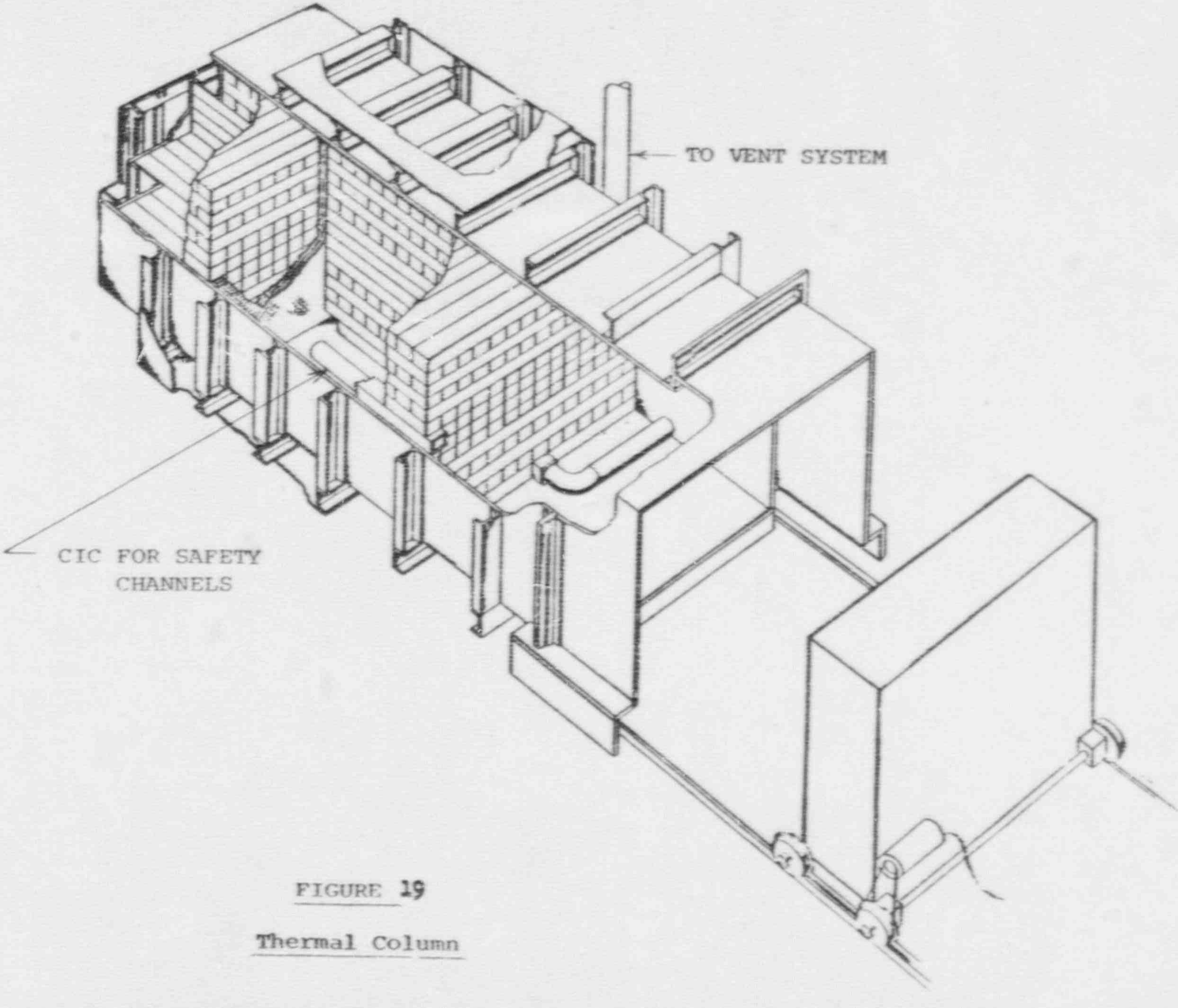


FIGURE 19

Thermal Column

2.4.2 Beam Ports

Four 6-inch beam ports penetrate the shield and provide fluxes of both fast and thermal neutrons for experimental use. Figure 20 shows the construction of the beam ports, while Figures 11 and 15 indicate the positions of the beam ports with respect to the grid box and shield. The ports are air-filled tubes, welded shut at the core ends and provided with water-tight covers on the outer ends. The portions of the ports within the pool are made of aluminum, while the portions within the shield are steel.

A shutter assembly, made of lead encased in aluminum, is opened for irradiations by a lifting device. When closed, the shutter shields against gamma rays from the shut-down core, allowing experiments to be loaded and unloaded without excessive radiation exposure to personnel.

Shielding plugs are installed in the outer end of each port. The plugs, made of dense concrete in aluminum casings, have spiral conduits for passage of instrument leads.

Since extremely high radiation levels could exist should the reactor be operated at substantial power levels with the shielding plugs removed, a beam port monitoring system is provided. The system consists of detectors mounted on the walls in line with each beam port and a read-out device at the console which gives an audible and visual alarm should a preset radiation level be exceeded. The system is set to alarm at a radiation level equivalent to a dose rate of about 60 mrem/hour at the beam port openings.

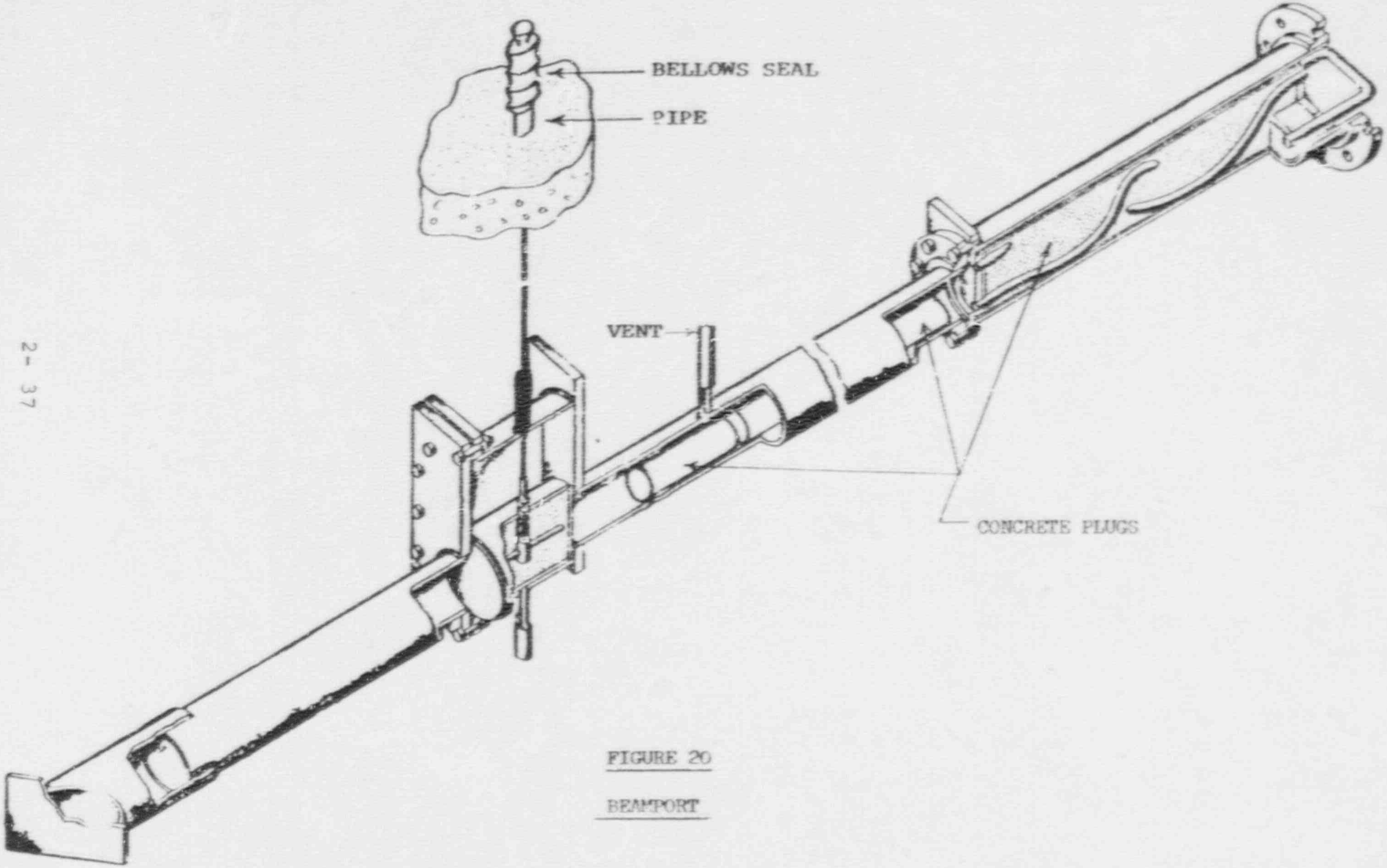


FIGURE 20

BEAMPORT

The thermal column-beam port ventilation system exhausts the beam ports through the pipes shown in Figure 20. With the beam port open, a linear flow velocity of about 40 feet per minute is maintained into the port opening. This prevents A^{41} activity from diffusing into the Reactor Laboratory. With the beam ports sealed up (the normal situation for most beam port use) very little air is exhausted from the ports, since the beam port drain valve is normally closed. The thin-walled aluminum cans in the in-pool portion of the ports prevents the higher A^{41} levels in these areas from diffusing throughout the beam ports.

2.4.3 Thermal Column and Beam Port Ventilation System

Figure 21 shows schematically the ventilation system for the thermal column and beam ports. The blower is sized to maintain an air velocity of about 40 feet per minute into all beam ports and the thermal column, should all be opened simultaneously.

The effluent from the system is filtered and discharged into the Reactor Laboratory stack. The filter and blower are at roof level to prevent leakage of water should a beam port rupture and fill with water.

The system is designed to sweep out the A^{41} activity present in an experimental facility when the facility is opened. During ordinary operation the facilities are closed and there is an essentially zero rate of discharge. When the facility is opened there is a slug of activity discharged. The average concentration discharged will therefore be extremely low due to dilution by the other blowers and the fact that no activity is discharged most of the time. Section 6.2 discusses the levels of activity discharged.

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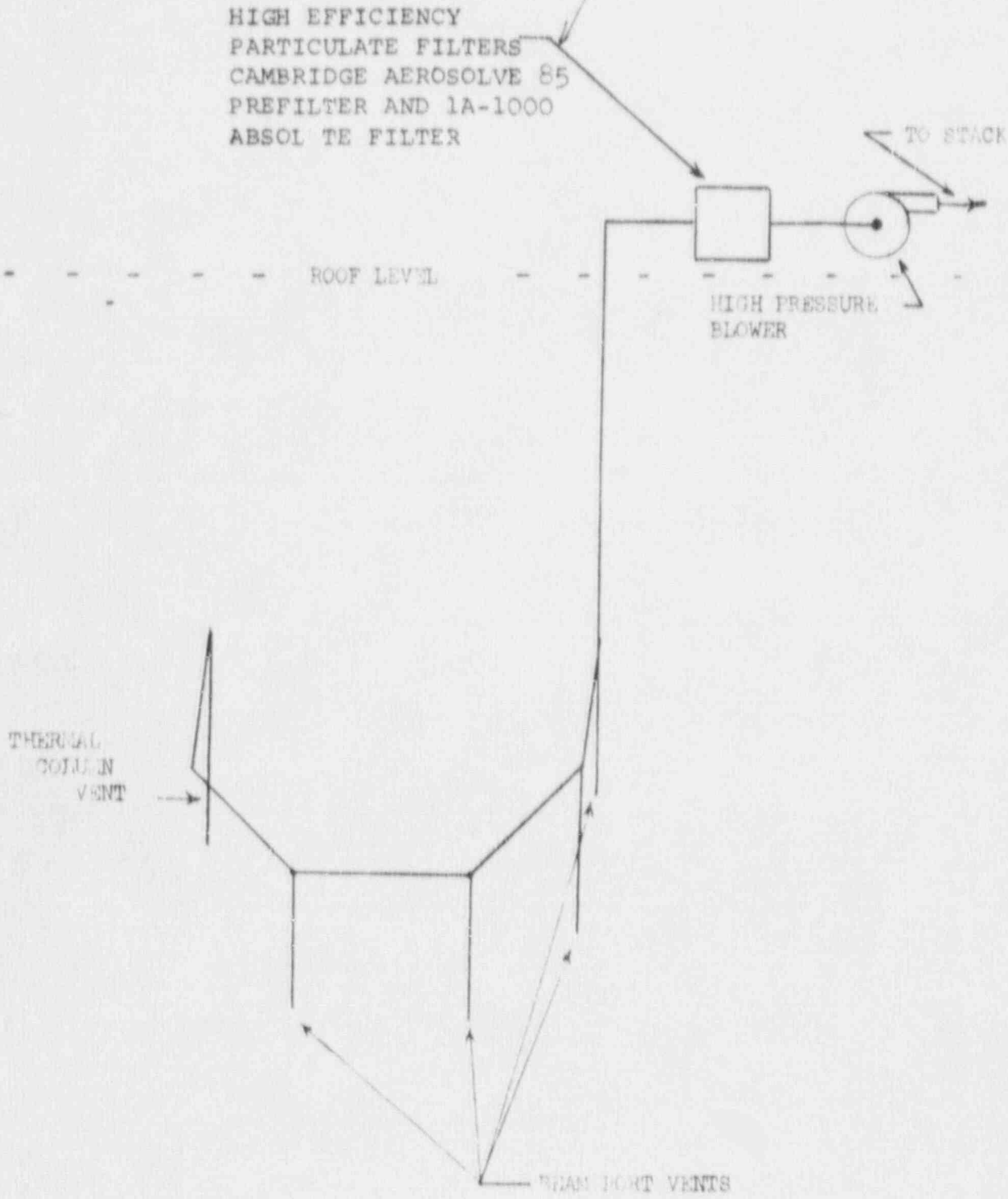

FIGURE 21

BEAM PORT AND THERMAL COLUMN VENTILATION SYSTEM

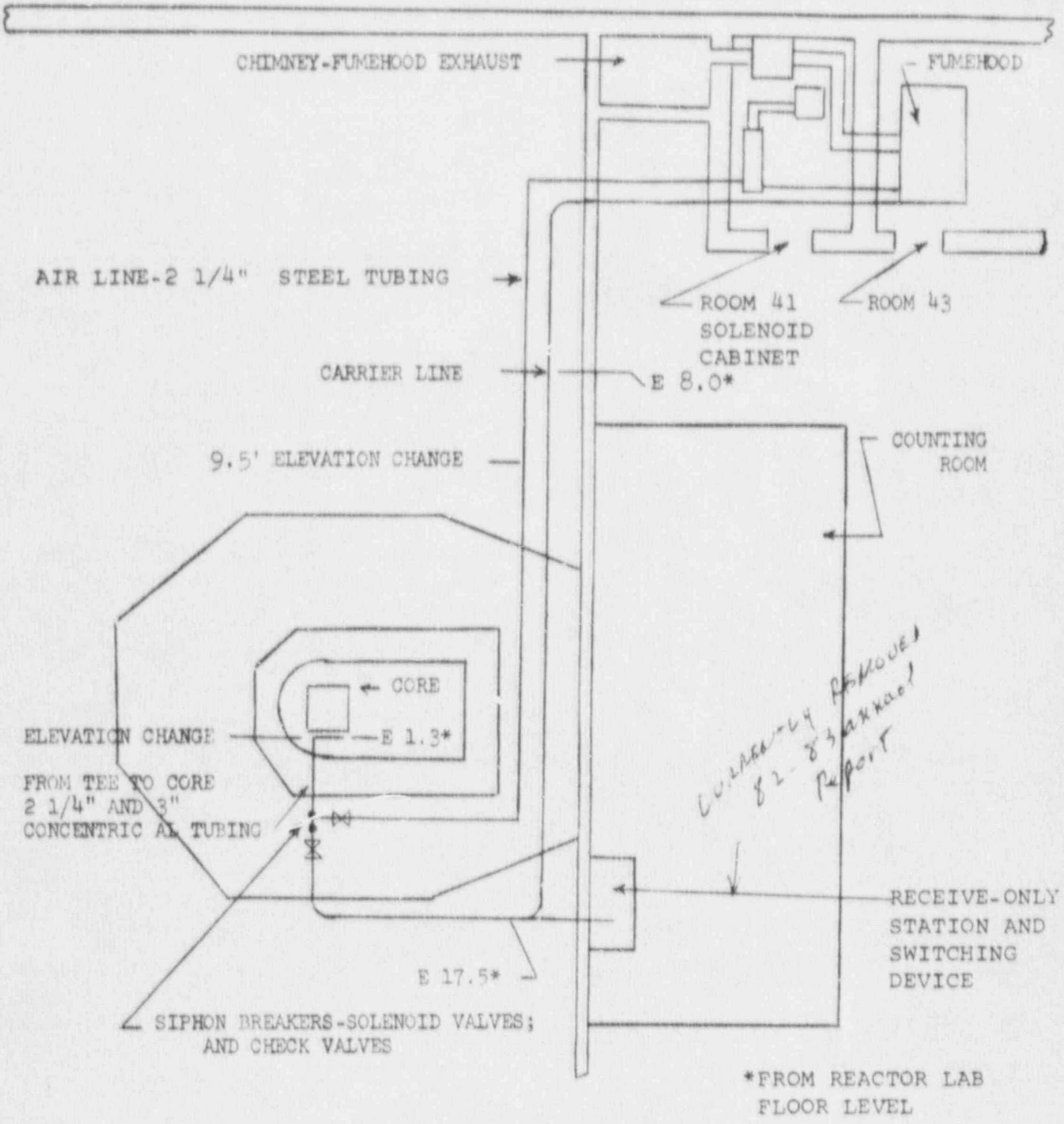

FIGURE 22

UWNR PNEUMATIC TUBE SYSTEM

2.4.4 Pneumatic Tube

A pneumatic tube system conveys samples from a basement room to an irradiation position beside the core (Figure 22). The "rabbits" used in the system will convey samples up to 1-1/4 inches diameter and 5-1/2 inches long. The system operates as a closed loop with CO_2 cover gas preventing generation of A^{41} activity.

The controls and send station are located in Room 41 (see Figure 5, Chapter 3). Indications of "Blower On" and "Rabbit in Reactor" are provided at the reactor console. An area monitor indicates radiation level and gives a visible and audible alarm should the radiation level exceed a preset level at each station. The preset level is selected according to the computed activity of the sample being irradiated, but is always less than 1 rem/hour.

The reactivity effect from a sample will be restricted to less than 0.2% ρ . Tests run with water and cadmium samples indicate that sample reactivity effects will normally be less than 0.01% ρ . Static reactivity measurements will be run for samples of fissionable material or particularly strong absorbers such as some of the rare earths.

Since the pneumatic tube penetrates the shield below water level, a leak in the tubing could drain the pool. To drain more than 8 feet of water, however, a siphon action would have to be set up. The siphon action is prevented by a solenoid valve controlled siphon breaker at the highest point in the system. The solenoid valves close when the blower motor is energized. When the blower motor is not energized, the solenoid valves open and check valves will then allow air to enter the system if a siphon action starts. Normally these check valves prevent loss of cover gas from the system.

The system will be operated using a written checklist type procedure to assure that the built-in safeguards remair. effective.

currently removed 82-83, report

A "receive only" ~~station is installed on the ground-level floor.~~ This station ~~is also installed~~ in a fume hood. Both stations are ventilated by 1200 cubic foot/ minute fans which draw air into the hood, past the sample outlet, and then exhaust it through absolute filters to a stack. Sample activity is limited to a level which, should the sample rupture upon discharge from the system, will result in keeping concentration exhausted below Part 20 limits for unrestricted areas when averaged over a 24 hour period.

2.4.5 Grid Box Irradiation Facilities

Most irradiations of more than twenty minutes duration are performed in irradiation facilities on the core periphery inside the grid box.

Radiation baskets are 3-inch square aluminum containers which fit into the grid plate and may contain one or more samples.

Two types of hydraulic transfer systems are used, however, for most of these irradiations. The smaller hydraulic irradiation device consists of an irradiation terminal which fits into a vacant grid position and a loading device and control valve assembly just beneath the pool surface. This system is powered by bypass flow in the N^{16} diffuser system.

In the large hydraulic tubes, sample movement is powered by a separate pump located beneath the north side of the reactor bridge. The motor for the pump is electrically parallelled to the diffuser pump. Each sample tube extends from about one foot below the pool surface to the grid box, where its bottom end fitting positions it rigidly. A clamping device at the top of the sample tube provides further support and prevents inadvertent movement. The pump takes its suction just below the pool surface and directs its flow to a jet pump near the bottom end of the tube, causing sufficient flow down the tube to move samples to the irradiation position and hold them in place. Samples which float return to the top of the tube where they are retained until removal by operating personnel. Non-floating samples can be removed with a retriever tool, or they may be installed with a retrieving string attached. Flow direction and "sample in" indicators and controls are located at the pool top and control console.

Rupture of piping connected to the facilities will not result in loss of pool water due to its location in and immediately above the pool. Reactivity effects of samples are much smaller than those associated with installation and removal of conventional irradiation baskets. The remarks regarding reactivity effects for samples in the pneumatic tube (Section 2.4.4) apply to these facilities.

2.5 CONTROL AND INSTRUMENTATION

The reactor will operate in three standard modes:

Mode 1	Steady state operation at power levels up to 1,000 KW.
Mode 2	Square wave operation (reactivity insertions to reach a desired steady state power level essentially instantaneously) at power levels between 300 and 1,000 KW.
Mode 3	Pulsed operation produced by rapid transient rod withdrawal that results in a step insertion of reactivity up to 2.1% $\Delta K/K$ (peak power of 2,000 MW).

Operation is from a console displaying all pertinent reactor operating conditions. A selector switch is provided for steady-state, pulsing, or square-wave modes of operation.

2.5.1 Steady-State Operation

For steady-state operation the control blades are slowly withdrawn to obtain the desired power level. At this level the reactor may continue to be operated manually or it may be switched to automatic control. The automatic control channel maintains power level by servo control of the regulating blade, transient rod, or #2 control blade. Figure 23 shows a block diagram of the control systen

Startup Channel

As shown in Figure 23, the sensing element for this channel is a fission counter. The counter has a range from 2 nv to 10^6 nv. Since the counter is movable, its effective range is thus from about 2 microwatts to 2 MW. The pulses from the startup counter are amplified and converted to a logarithmic count-rate displayed on a meter and recorded. The amplified pulses may also go to a scaler that is used for subcritical measurements.

Log N - Period Channel

This channel monitors the power level of the reactor over the range from 0.1 watt to full power. The Log N - period amplifier detects the signal from a compensated ionization chamber and amplifies the signal to provide a 7-decade logarithmic display proportional to power level. The amplifier also extracts period information. The Log N signal is recorded while the period signal is recorded, displayed on a meter on the console, and fed to the logic element.

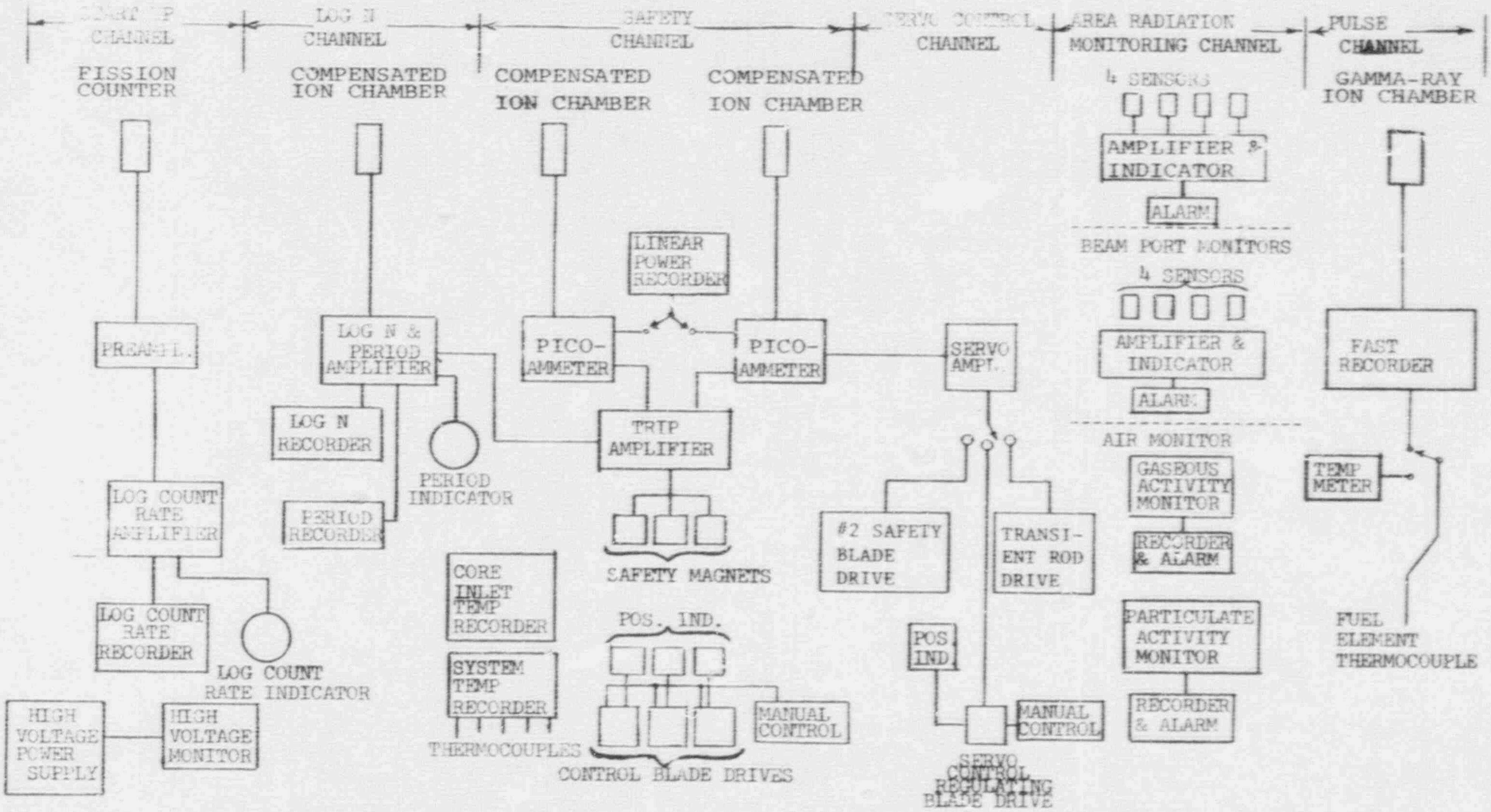

FIGURE 23

NUCLEAR INSTRUMENTATION

Safety Channels

Two safety channels monitor reactor power level from about 0.1 watt to full power. The signal from each channel originates in a compensated ionization chamber. The chamber signal is fed into a solid state picoammeter. The trip output signals from the picoammeters are fed to the logic element where they, along with the period signal from the Log N channel, determine whether power is supplied to the control blade magnets. Should any one scram signal or a combination of scram signals be present, the reactor shuts down. The power level scram trip point is set to 1.25 times the operating level.

Temperature Measurements

Fuel element internal temperature is indicated at the console. It causes an alarm and scram at the limiting safety system setting.

The temperature of the bulk pool water is measured at the core inlet by a resistance thermometer. This temperature is indicated on a recorder and causes an alarm and a scram on excessive temperature.

Primary and secondary cooling system inlet and outlet temperatures, and demineralizer inlet temperature are indicated on the system temperature recorder. An alarm on this recorder indicates excessive temperature at any of these points.

2.5.2 Square Wave Operation

This mode is provided for those applications which require that the power level be brought rapidly to some high level, held there for a period of time, and then reduced rapidly producing a square wave of power.

In the square wave mode the reactor is brought to a level of 1 to 1000 watts in the steady-state mode. The mode switch is then changed to the square wave position. A preadjusted step reactivity change is then made to bring the reactor to preset power levels between 300 and 1000 kW. The reactivity step change is made with the transient rod. Then the automatic control system inserts additional reactivity required to maintain the preset power level as the fuel heats up. The operator must manually augment the reactivity inserted by the servo. In this mode the period meter and scram are disconnected and the safety channel range switches must be set on their full power ranges. The linear power level scram is maintained at 1.25 P max. and an interlock prevents initiation of this mode if the range switch is not on the full power range setting.

2.5.3 Pulsing Operation

The reactor is brought to a power level of less than 1000 watts in steady state mode. The mode switch is then changed to pulsing mode. When the switch is in pulsing mode the normal neutron channels are disconnected and a high level pulsing chamber is connected to read out the peak power of the pulse on a fast recorder provided for that purpose. Changing of the mode switch to pulse removes an interlock that prevents application of air to the transient rod unless the transient rod is in the full "in" position. Only the transient rod is automatically reinserted after a preset time delay. Fuel temperature is recorded during pulsing operation. The pulse channels are also indicated on Figure 23.

2.5.4 Blade Control

The three safety blades are manually controlled by two switches: one selects the blade to be moved; the other, a pistol-grip switch with spring return to "off", has positions of "raise", "off" and "lower" and controls the selected blade. Only one blade may be raised at a time. A separate switch is available which will lower all blades at the same time. The position of each safety blade is indicated by a digital read-out, and the indicator lights on the console show when each blade drive is at its "in" or "out" limit and when the blade magnets are engaged with the armatures.

The safety blades will scram from any position during withdrawal and run-down. In the event of a scram, the manual controls are over-ridden and the blade drives run in to their "in" limits. The following conditions must be met before the safety blades can be withdrawn:

1. No scram conditions present and scram relays reset;
2. Count-rate on startup channel greater than 2 counts per second;
3. Fission Counter not in motion;
4. Console key switch set to "on" position.

The regulating blade has identical position indication and "in" and "out" limit indication. It is manually controlled by a separate pistol-grip switch and may be driven concurrently with one other control element. The blade drives may be tested by use of a "test" position on the key switch. The scram relay must be de-energized before the drives can be moved while the key switch is in the test position.

2.5.5 Automatic Control System

The servo amplifier controls reactor power level in "automatic" and "square-wave" modes. The servo amplifier output drives either the regulating blade, transient rod, or a safety blade as selected by a servo element selector switch on the console. The servo amplifier responds to a power level signal from one of the safety channel picoammeters and controls speed and direction of the servo element through a servo motor.

In "automatic" mode, the servo amplifier receives period information from the Log N - period channel and limits servo element withdrawal to maintain a period longer than a preselected level. In "square-wave" mode, the period amplifier is disabled and a "servo error" circuit is employed. This circuit allows servo operation only when the servo error is less than 5%. Servo error is indicated at the console in both "square-wave" and "automatic" modes.

Additional indicators on the reactor control cubicle are provided to indicate "automatic on" and scheduled power.

2.5.6 Transient Rod Control

Movement of the transient rod is controlled by the console-mounted, switch-light, pushbuttons which not only control movement, but also indicate in and out limits. Position indication is accurate to one per cent.

2.5.7 Scram Circuits

The scram circuits initiate either relay (slow) or electronic (fast) scrams. Relay scram is accomplished by de-energizing the scram relays under one of the following conditions:

1. Fast period (adjustable between 5 and 7 seconds and set to 5 seconds);
2. Manual scram;
3. High voltage failure in control console;
4. Temperature at core coolant entrance above $130^{\circ}F$;
5. Power level greater than 1.25 P max;
6. Loss of control power;
7. Log N - period amplifier calibrate switch not in input position;
8. Timed transient control rod scram in pulse mode;
9. Pool water level low;
10. Fuel temperature above LSSS.

The control blades are free to fall within 120 milliseconds of the initiation of a relay scram.

Electronic scram is accomplished by biasing transistors in the magnet power supply into non-conductance. The elimination of relays results in a shortening of the scram delay-time to less than 60 milliseconds. For the lowest range settings on the picoammeters, this delay is increased due to the time constant of the instrument at these low settings. Electronic scram is initiated by:

1. Power level exceeding 1.25 P_{max};
2. Period shorter than 3.5 seconds;
3. Loss of signal from picoammeters.

2.5.8 Alarm and Indicator System

When an abnormal condition develops, a horn sounds and a red light comes on. The operator may press the acknowledge button to silence the horn. When the condition is corrected, the green light goes on and the red light is extinguished.

The following conditions will actuate the alarm system:

1. Any scram;
2. Neutron flux exceeding 1.15 times the normal value;
3. Reactor period less than 10 seconds;
4. Safety blade disengaged from magnet;
5. Water level in pool two or more inches below normal (also gives an alarm at Protection and Security Headquarters);
6. Failure of high voltage power supply;
7. High area radiation level;
8. Beam port monitor actuated by high radiation level;
9. Air particulate or gaseous activity above normal level;
10. Fuel element temperature above **LSSS.**
11. Count rate on startup counter approaching saturation level;
12. Core inlet temperature above 125°F.;
13. Pneumatic tube blower on;

14. Hold tank full;

15. Chain switch across stair actuated (See Section 2.6.2);

16. Thermal Column door open.

To provide operating information for the reactor operator, the following indicating lights are provided:

1. Scram;
2. Scram reset;
3. Safety blade magnet engaged;
4. Power on;
5. Control elements in (distinct light for each);
6. Control elements out (distinct light for each);
7. Regulating blade on automatic control;
8. "Rabbit" in reactor.

2.5.9 Radiation Monitors

The radiation monitors are arranged into three systems; the primary area monitors, beam port monitors, and air activity monitor.

The primary area monitors are located as follows:

1. Demineralizer area;

2. On the reactor bridge about one foot above the water surface;

3. Beside the thermal column door;

4. At the pneumatic tube send-receive station;

~~5. At the pneumatic tube receive only station.~~ Removed when station was removed 83-84

Units 1 through 3 have ranges from 0.1 to 100 mr/hr, while units 4 and 5 have a range from 1 to 1000 mr/hr.

Unit 1 supplies information on radiation level from the demineralizer. It will be set at the beginning of a reactor run to alarm at a radiation level just above that reached in a normal run. Unit 2 will be set to alarm at a radiation level just above that reached at full power operation. Unit 3 is located beside the thermal column. It too is set to alarm just above normal operating level. This unit will give an alarm if the thermal column door is left open when the reactor is operated at any substantial power.

Unit~~s~~ 4 ~~and 5~~ are set to alarm at 10 mr/hr unless the calculated activity of a sample results in a level above 10 mr/hr. In that case the alarm will be set at the calculated radiation level. This unit has audible and visual signals and radiation level indicators at the pneumatic tube stations as well as at the console.

Units 1 through 3 are connected to the Reactor Laboratory evacuation alarm. An alarm from one of these units will sound the evacuation alarm if it is not acknowledged by the operator within 30 seconds (see Section 6.8). VANR 150

The beam port monitor is an area radiation monitor installed to preclude the possibility of unknowingly generating high radiation levels by operating the reactor at high power levels with the beam ports open. The sensors for this system are installed on the walls of the Reactor Laboratory in direct line with the beam ports. The system gives visual and audible alarms at the console if the radiation level exceeds a preset value. The monitors are normally set to alarm at a radiation level equivalent to a dose rate of 50-100 mr/hr at the beam port flange. The setting varies from beam port to beam port due to different distances between the walls and the beam port openings.

The air monitor measures both particulate and gaseous activity of the air discharged from the stack. Particulate activity is collected on a filter tape and counted with a thin end-window GM tube and count-rate meter. Gaseous activity is measured with a large Kanne ionization chamber. The system therefore operates by detecting β activity. Both particulate and gaseous activity levels are recorded, and provide annunciation should preset levels be exceeded.

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The sensitivity of the particulate activity monitor allows detection of concentrations of about 10^{-10} $\mu c/ml$ of a material with a single β particle emitted per disintegration. The efficiency is higher if more than one β particle is emitted per disintegration. The sensitivity of the gaseous activity monitor is such that a concentration of about 1×10^{-8} $\mu c/ml$ of A^{41} at the stack discharge can be detected by the instrument. The efficiency varies with the energy of the β particle associated with the isotope being detected. For β energies higher than that of A^{41} the efficiency varies extremely slowly; for β energies lower than A^{41} the efficiency varies more rapidly. The primary activity expected to be present in the stack discharge is A^{41} and the instrument is calibrated in terms of A^{41} activity.

2.6 SHIELDING AND EXPECTED RADIATION LEVELS

2.6.1 Basic Reactor Shield

The reactor is shielded by concrete and water. The core is covered by 20 feet of water. The shield at core level consists of about 3 feet of water plus 8 feet of ordinary concrete. Denser concrete is used in the thermal column door and beam port plugs. Calculations and measurements indicate radiation levels to be expected for 1000 KW operation are (excepting N^{16} activity which is discussed below):

Surface of shield, excepting beam port and thermal column openings - less than 1.5 mrem/hr.

Pool surface (leakage radiation - No N^{16} less than 15 mrem/hr.

"Hot spots" - measurements have shown that higher radiation levels exist around the beam ports and thermal column. Extrapolation from these measurements indicates the maximum radiation levels at these "hot spots" at

1000 KW will be about 10 mrem/hr around the beam ports and 40 mrem/hr at the hottest spot around the thermal column door. The dose one foot away from the hot spot will be about 5 mrem/hr.

2.6.2 Pool Surface Radiation Levels - N^{16} Activity

The expected radiation level due to N^{16} activity at the pool surface directly above the core when operating at 1000 KW is 120 mrem/hr. The diffuser jet system will normally be used, and the radiation level would normally be considerably less than the level indicated above. These radiation levels will be low enough that no hazard will exist to personnel outside the Reactor Laboratory or in normally occupied levels within the Reactor Laboratory. Radiation levels on the walkway surrounding the pool are expected to be around 20 mrem/hr while the reactor is operating at 1000 KW without the diffuser operating.

The entire Reactor Laboratory is posted as a radiation area. A chain and switch arrangement is positioned on the north stairway to the pool surface so that an alarm will be sounded should entry to that area be made while the reactor is operating, thus assuring that personnel will not enter the area without knowledge of the reactor operator.

The south stairway, leading from the console area to the pool surface does not have a chain and switch arrangement, as does the north stairway. Access to these stairs is gained only through the console area and is thus well monitored. No difficulty is expected in maintaining radiation doses to individuals below those doses permitted in 10 CFR 20.

2.6.3 Demineralizer

The dose rate in the demineralizer area due to activity removed from the pool water by the demineralizer may be as high as 100 mrem/hr for short periods of time at points close to the demineralizer. The location of the demineralizer is such that shielding can be easily provided should it prove necessary. Since it is expected that the radiation level will normally be much lower than the value quoted above, the area around the demineralizer will initially be roped off and shielding will be provided only if it appears that radiation levels will routinely approach the value quoted above.

2.6.4 Heating Effects in Shield and Thermal Column

Heating effects caused by absorption of gamma radiation and fast neutrons are within allowable limits. For all calculations, it was assumed that the pool water was at the 130°F temperature limit, and the reactor was operated continuously at 1.5 MW.

The heating in the concrete shield is approximately 20% of the maximum suggested by Rockwell*.

Analysis of the heating rate in the lead shield for the thermal column indicates that the maximum temperature of the lead will be less than 217°F. Calculation of the graphite temperature in the thermal column indicates a maximum of 244°F.

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* Rockwell, Theodore, Reactor Shielding Design Manual, McGraw Hill, 1956.

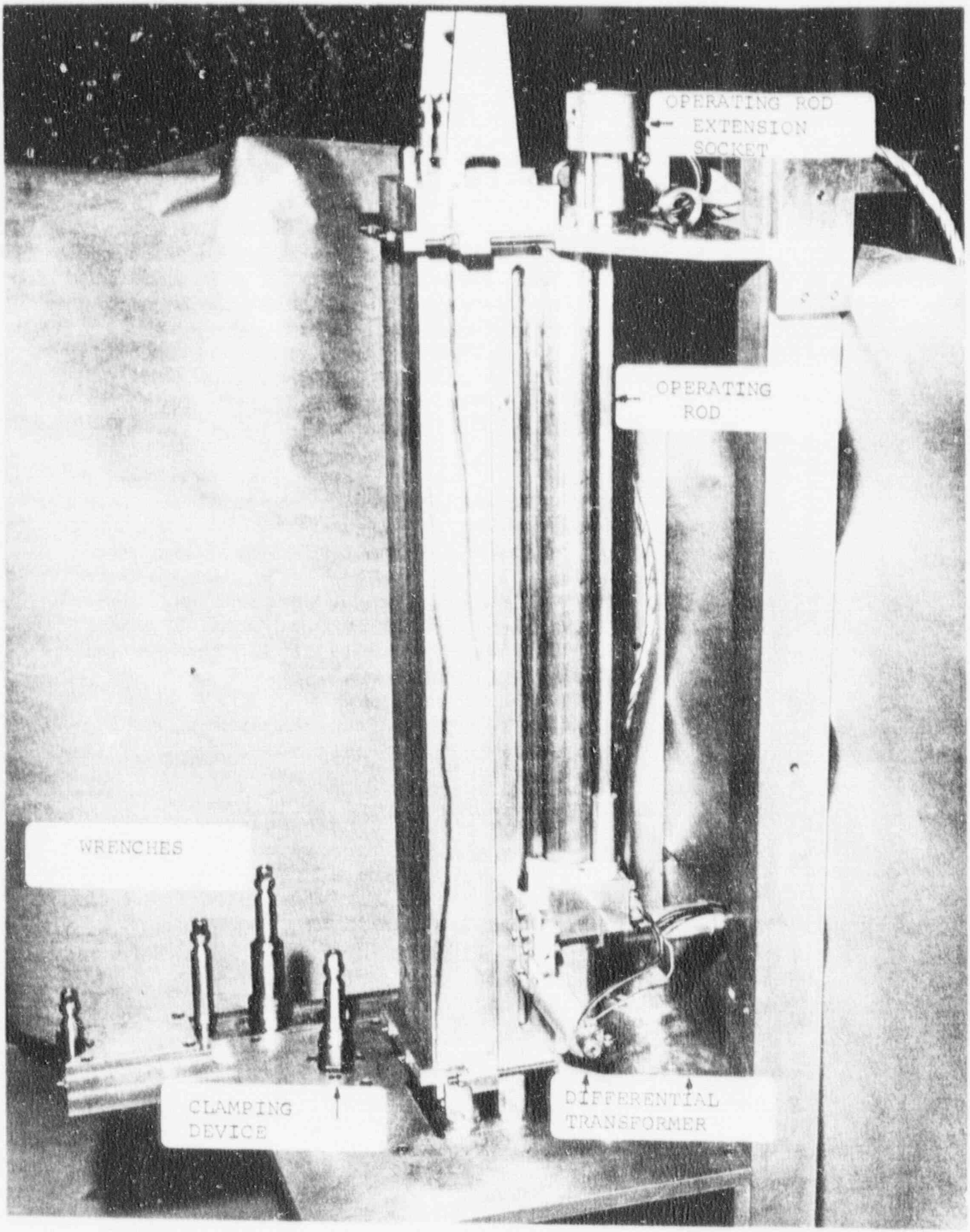


Figure 24 FUEL MEASUREMENT AND MAINTENANCE TOOL

2.7 FUEL ELEMENT MEASUREMENTS

Required measurements of fuel element bow and elongation are made with the measurement and maintenance tool (shown in Figure 24 with a dummy element installed). This tool is used for disassembly and assembly of fuel bundles as well as for fuel measurements. It is used under about 18 feet of water where it hangs from a fuel storage rack in the reactor pool. A fuel-element handling tool also operates the socket and crowsfoot wrenches. An air-operated clamping device reproducibly positions the bottom end box of the fuel bundle.

For disassembly or assembly of fuel bundles, the maintenance and measuring tool holds the bundle, provides a reference plate for the crowsfoot wrench, provides storage space for four individual fuel elements, and restrains the individual elements after they have been screwed out of the bottom end box. The handling-tool then may be used to remove individual elements and place them in the storage positions.

While it is possible to disassemble and reassemble the fuel bundles, it is tedious. The crowsfoot wrench must be used for the initial loosening of each element and for torqueing each element upon reassembly. While the elements are loose enough to turn freely, a socket wrench can be used on a hexagonal portion of the top fitting. Disassembly or assembly of an element takes about 30 minutes.

Because of the excessive time and added handling required to disassemble the bundle and measure each element in a separate measuring tool, the tool was designed to make the measurements without disassembly.

Figure 25 shows the three sensors employed (a portion of the housing is removed in this view). Each uses a differential transformer as transducer to give a remote electrical output proportional to displacement of the sensors.

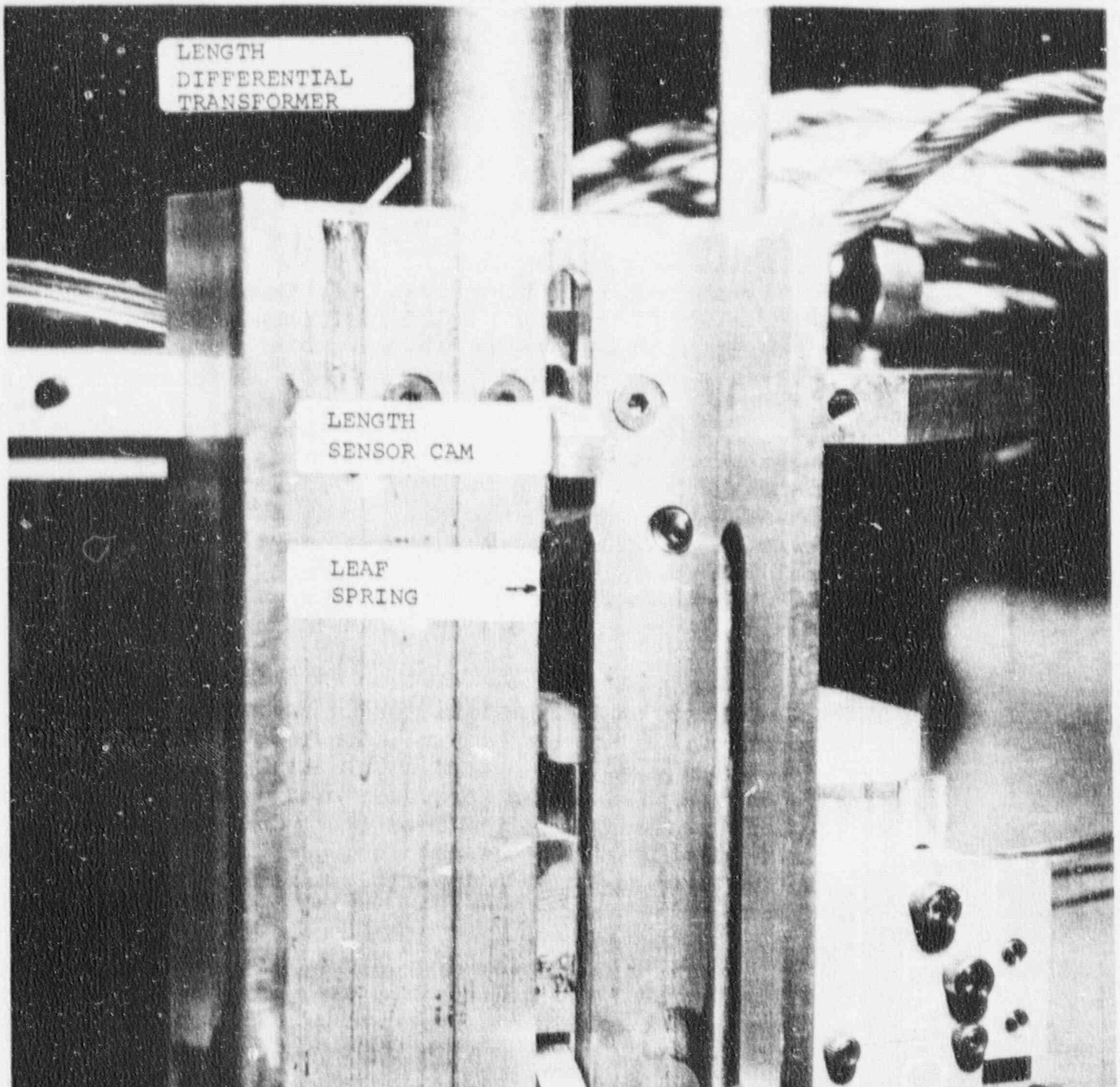


Figure 25 BOW AND ELONGATION SENSORS

The X and Y sensors employ spring-loaded aluminum wheels attached to the differential transformer cores. When the bundle is lowered into the tool the wheels are forced back and they then ride on the fuel element clad surface. These sensors are adjusted to give a zero signal for a standard fuel element dummy.

The length sensor differential transformer is actuated by one lobe of a cam. A second lobe of this cam is rotated into contact with the top edge of the fuel element cladding by a leaf spring attached to the operating rod. The cam pivots out into the measuring position only when the operating rod is fully withdrawn. The length sensor is also adjusted to zero output for the standard fuel element dummy.

A readout box located at the pool surface allows the operator to select X, Y, or length readout. Differential transformer core position is indicated by a center-zero meter. A recorder output is provided to the horizontal axis of an X-Y recorder. Polarity is set so an increase in element length or a bow away from the center-line of the bundle gives a positive meter indication or recorder readout.

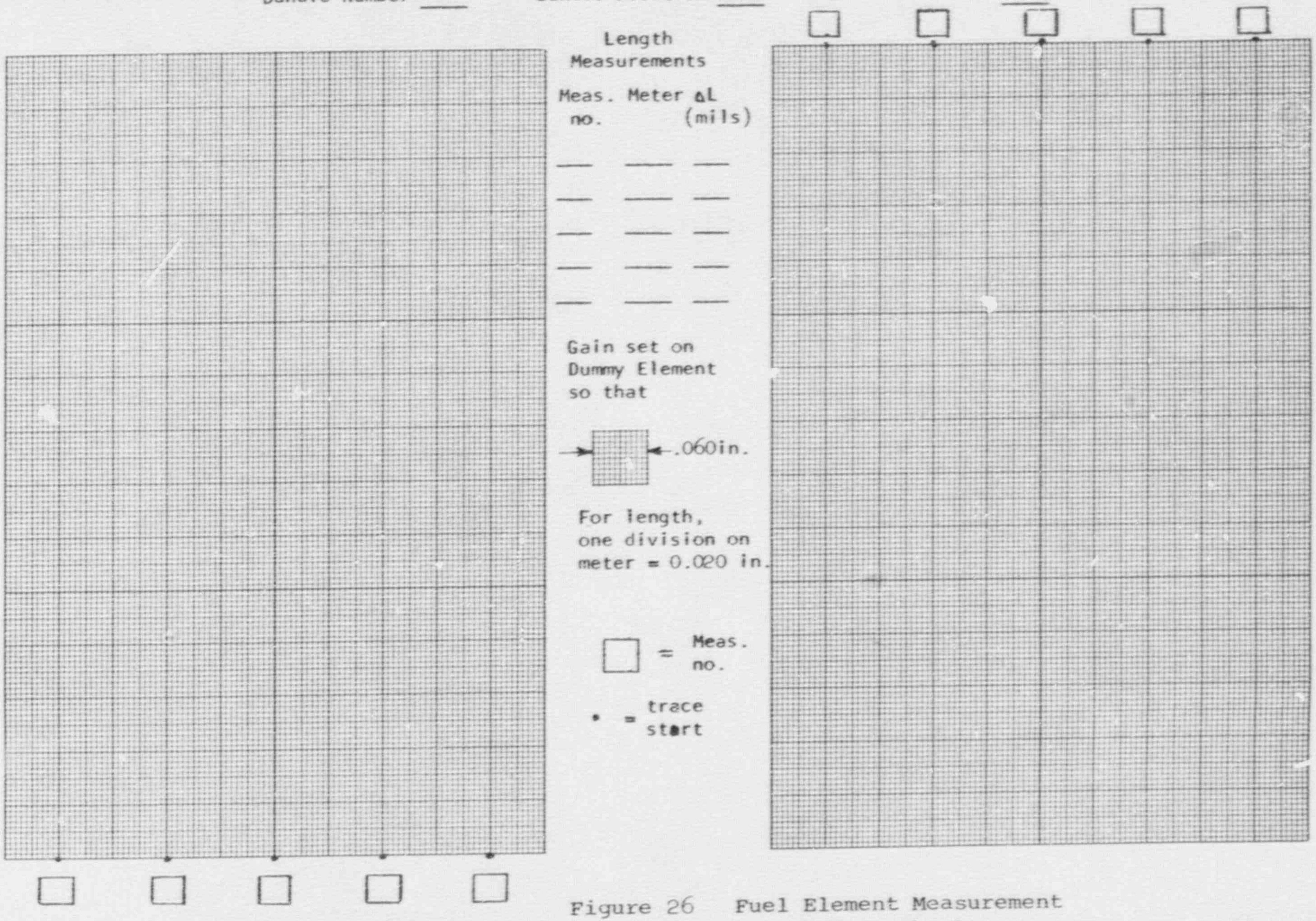
A guide at the pool surface is positioned directly over the measurement tool and clamped in place. An extension connected to the operating rod fits through a bushing in this guide and engages a gear-driven vertical position readout device. The vertical position readout signal is supplied to the vertical axis of the X-Y recorder.

The dummy fuel bundle has one dummy element exactly 0.100 inches longer than the other three elements. This element also has a section in which the radius has been reduced by 0.060 inch. By using this element the attenuation and zero controls on the readout box may be adjusted to give a calibrated readout of bow and length.

After calibration of the tool, measurement can be made on standard fuel elements. Figure 26 is a standard data sheet used to record measurements. The standard setup used gives a 1 cm horizontal displacement for 0.060 inch transverse bend (bow) and a meter reading of 5 units for a length increase of 0.100 inch. A trace is drawn for both the X and Y sensors while the length measurement meter reading is recorded in the center section of the form. A complete set of measurements for all four elements in a bundle can be completed in about twenty minutes.

Since the X and Y readouts are 90° apart, the maximum possible bow will be the square root of the sum of the squares of the bows indicated by direct measurement. As long as neither measured bow exceeds 0.088 inch, no calculations or other measurements are necessary. If either bow measurement exceeds 0.088 inch, then the square root of the sum of the squares of the measured bows must be calculated to determine whether or not this resultant is less than 1/8 inch. If the calculated number is less than 1/8 inch, the element is within technical specifications. Should the calculated bow exceed 1/8 inch, the crowsfoot wrench may be used to rotate the element being measured so that the reading of one bow sensor is maximized and the true bow may be determined directly to see whether it exceeds technical specification limits.

Bundle Number ___ Bundle Position ___ Element Number ___



Length Measurements

Meas. no.	Meter	ΔL (mils)
___	___	___
___	___	___
___	___	___
___	___	___
___	___	___

Gain set on Dummy Element so that

For length, one division on meter = 0.020 in.

☐ = Meas. no.

• = trace start

Figure 26 Fuel Element Measurement Data Sheet

Chapter 3

LOCATION AND BUILDING

3.1 LOCATION

3.1.1 Site Description

The reactor is housed in a building termed the Reactor Laboratory, on the campus of the University of Wisconsin in Madison, Wisconsin. The location of this building is shown in Figures 1, 2, 3 and 4. The building adjacent to and surrounding the Reactor Laboratory on the east, north and west sides is the Mechanical Engineering Building. The Engineering Research Building is adjacent to the Mechanical Engineering Building on the east. Drawings of the Reactor Laboratory, the Mechanical Engineering Building and the Engineering Research Building are contained in Figures 5-10.

The nearest residence is over 500 feet from the reactor location, and the population distribution surrounding the reactor is as shown in Figure 11. The nearest classroom is at least 60 feet from the reactor location with several walls of various materials separating the two.

Immediately to the south of the Reactor Laboratory is a service road and a parking lot and immediately to the north is a courtyard. To the west are Nuclear Engineering Laboratories which flank the Reactor Laboratory on both the basement and first floor levels. To the east is a Nuclear Engineering Laboratory and a Heat Power Laboratory flanking the Reactor Laboratory on the basement and first floor levels respectively.

The Heat Power Laboratory is a single room, with 10 foot partitions separating different experimental areas, and covers the entire first and second floors on the east

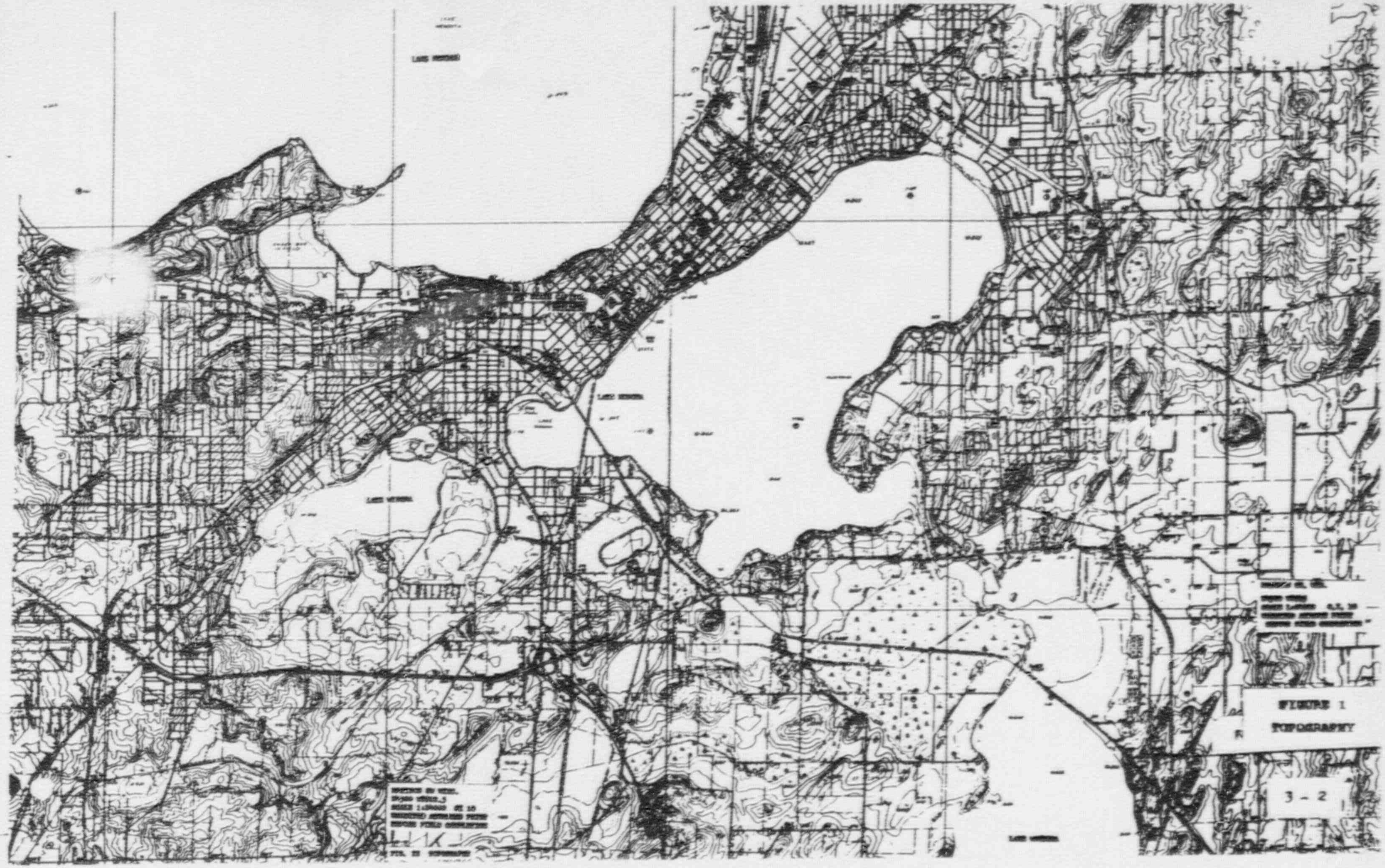
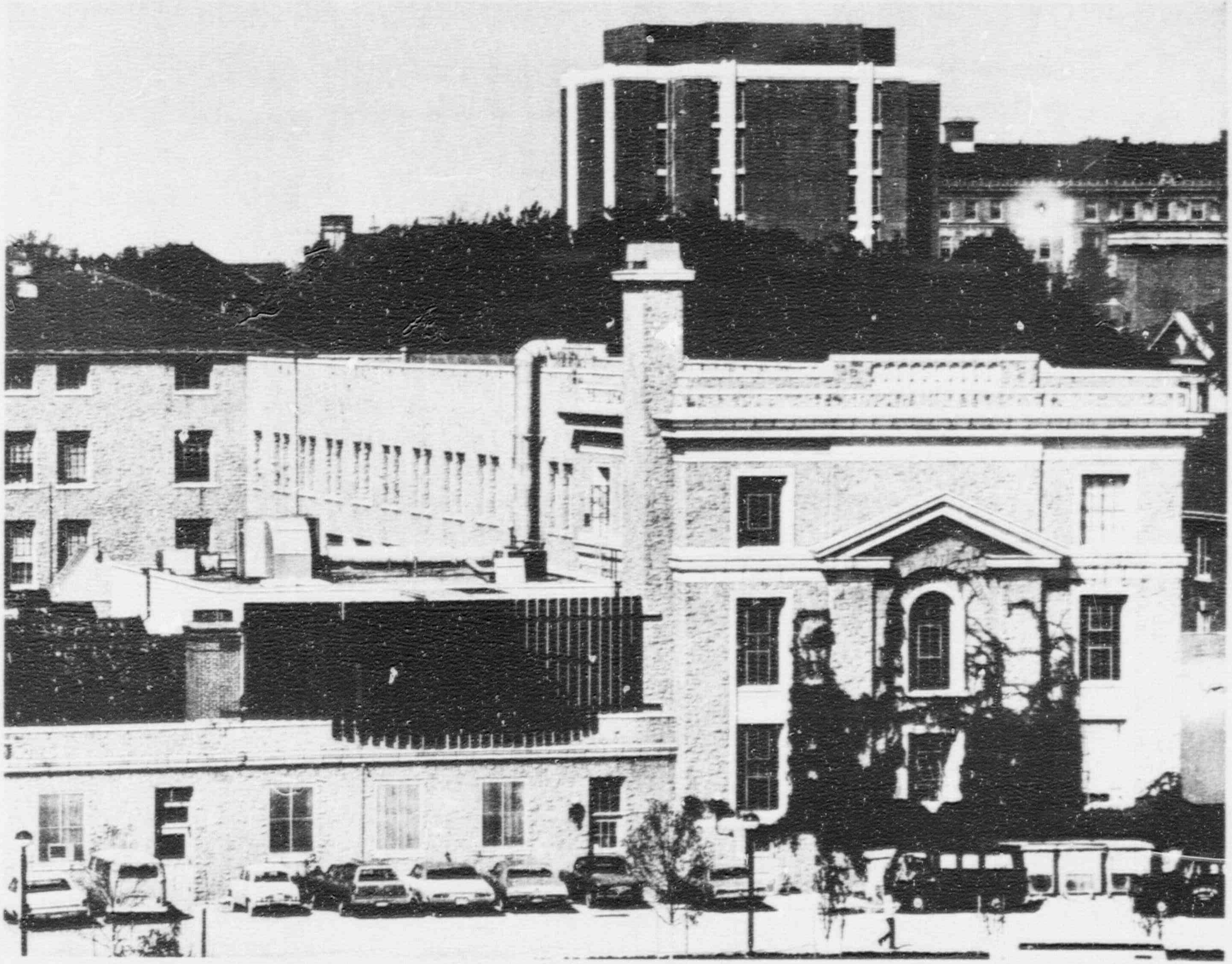


FIGURE 1

TOPOGRAPHY

FIGURE 2
CITY OF MADISON


U. OF W. CAMPUS
CAMPUS BOUNDARIES
REACTOR
500 FT.
1000 FT.
2000 FT.
CAMPUS BOUNDARIES



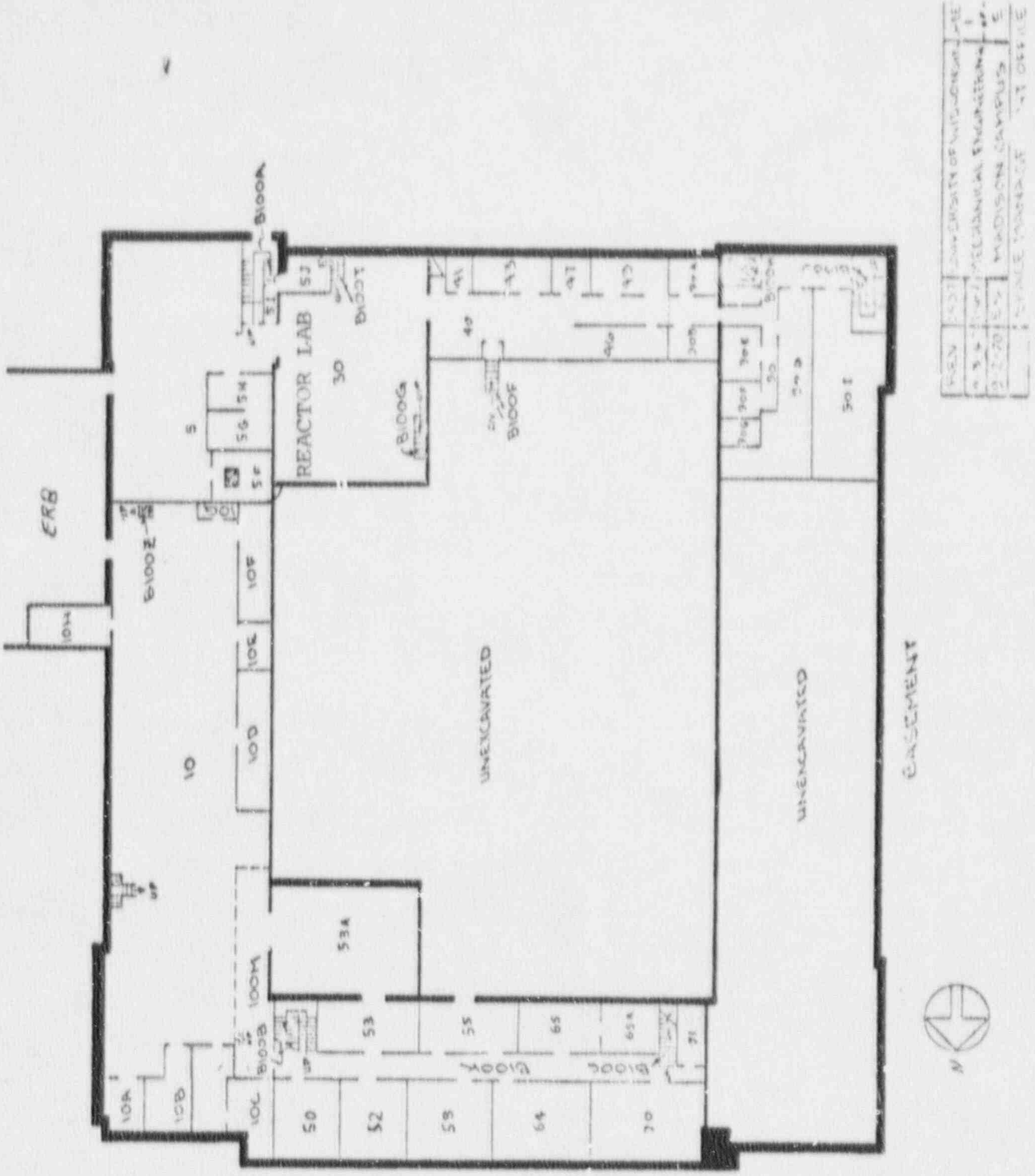


FIGURE 5

REACTOR LABORATORY AND BASEMENT
MECHANICAL ENGINEERING BUILDING

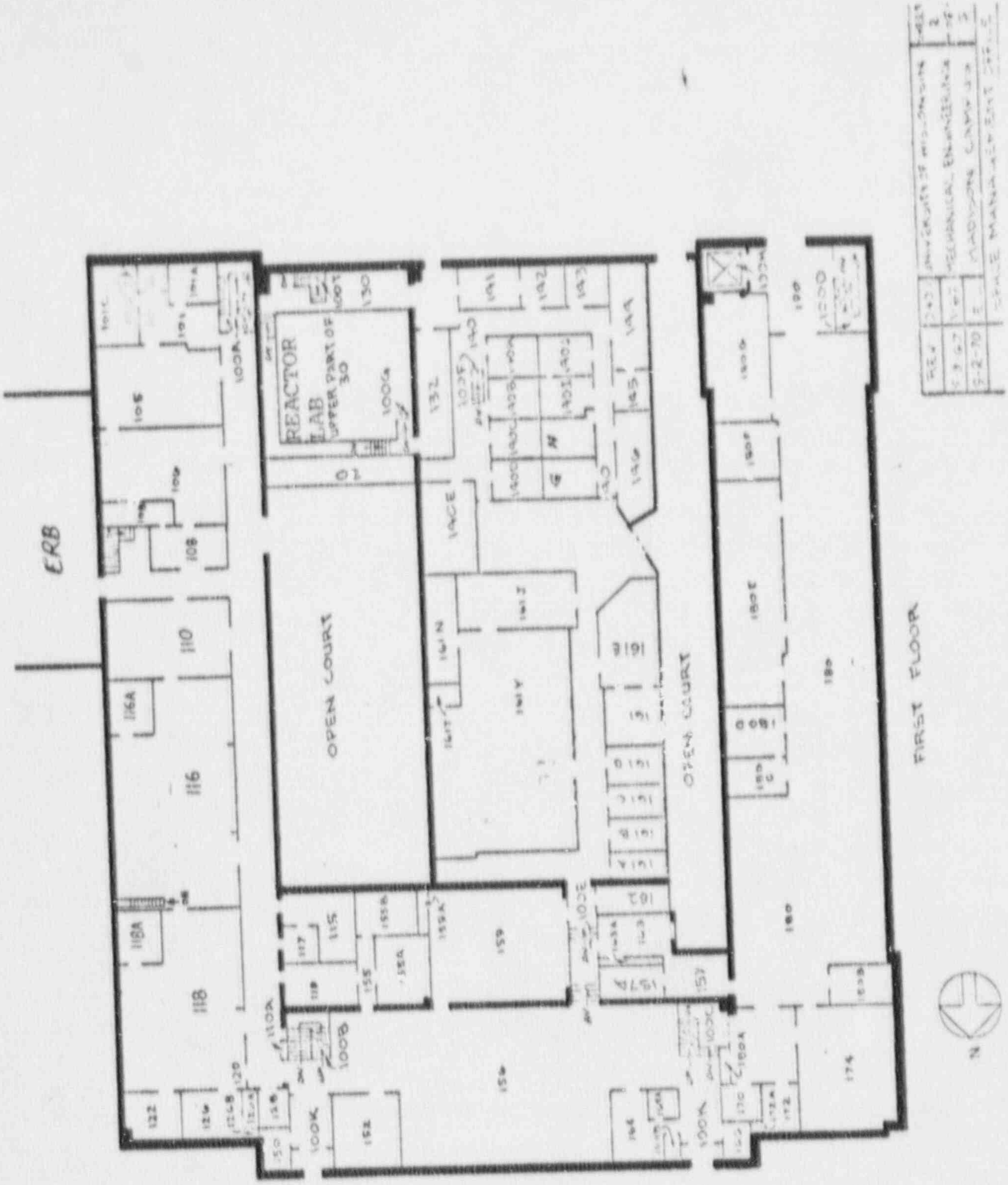


FIGURE 6

REACTOR LABORATORY AND FIRST FLOOR
MECHANICAL ENGINEERING BUILDING

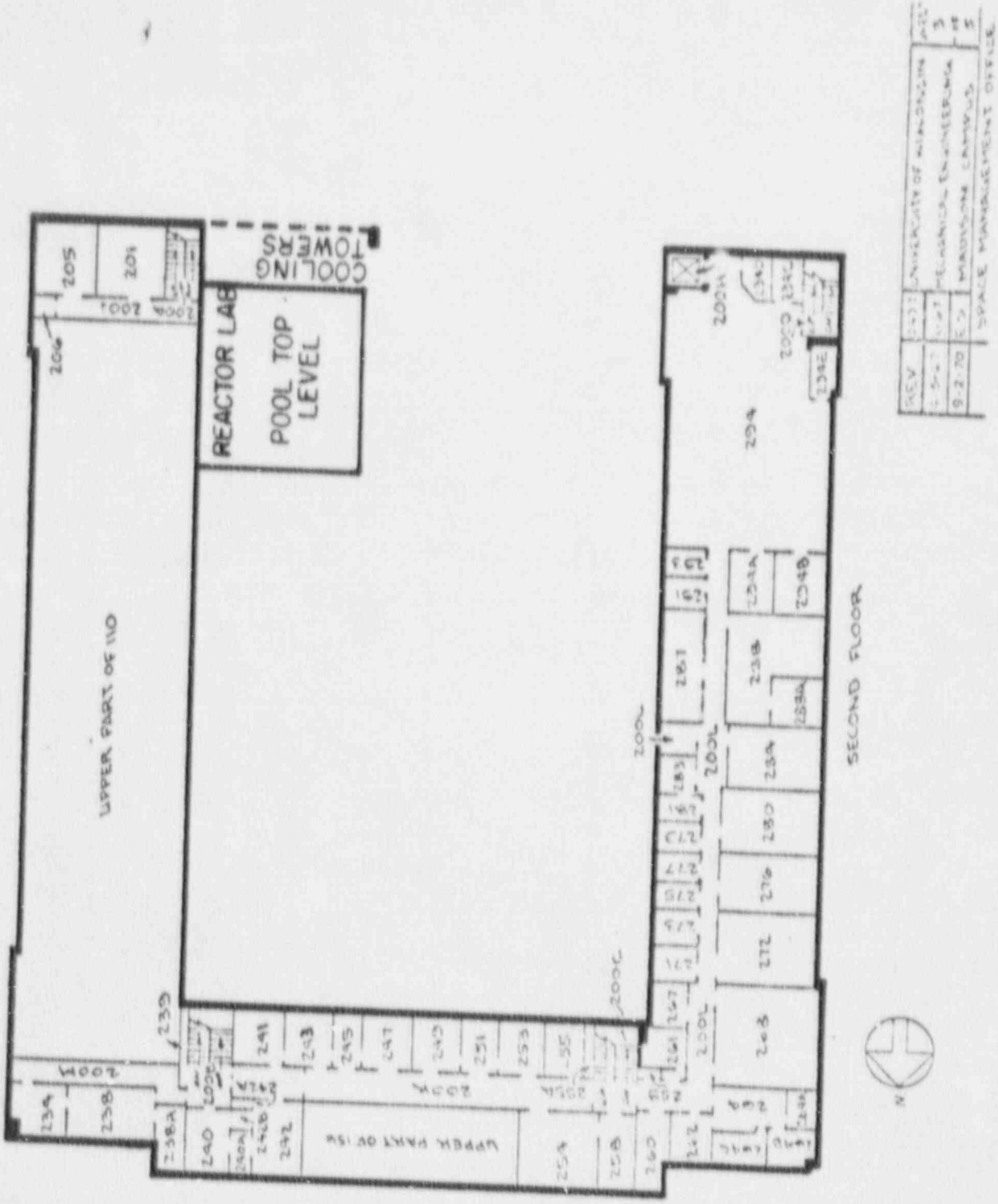


FIGURE 7

SECOND FLOOR
MECHANICAL ENGINEERING BUILDING

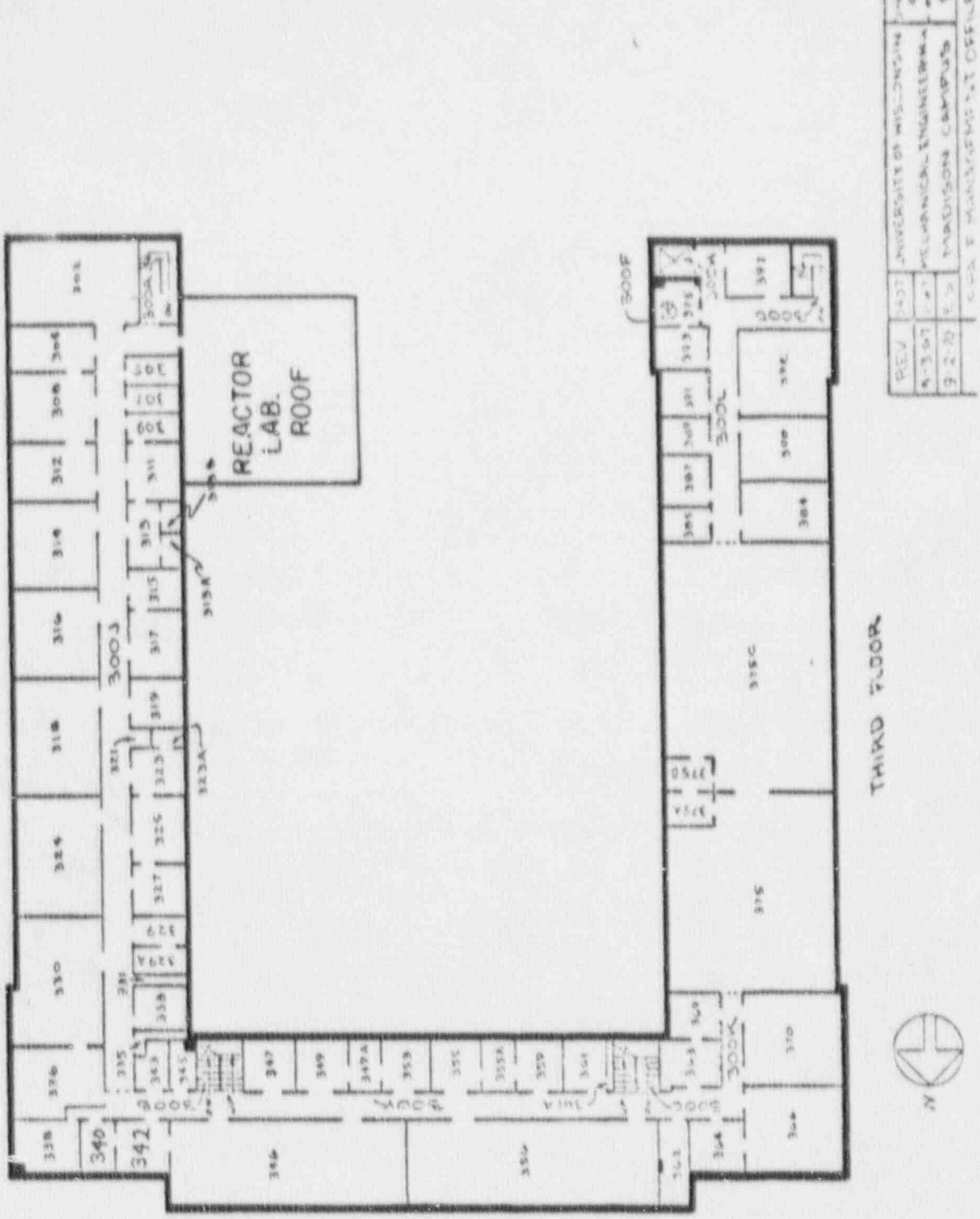


FIGURE 8

THIRD FLOOR
MECHANICAL ENGINEERING BUILDING

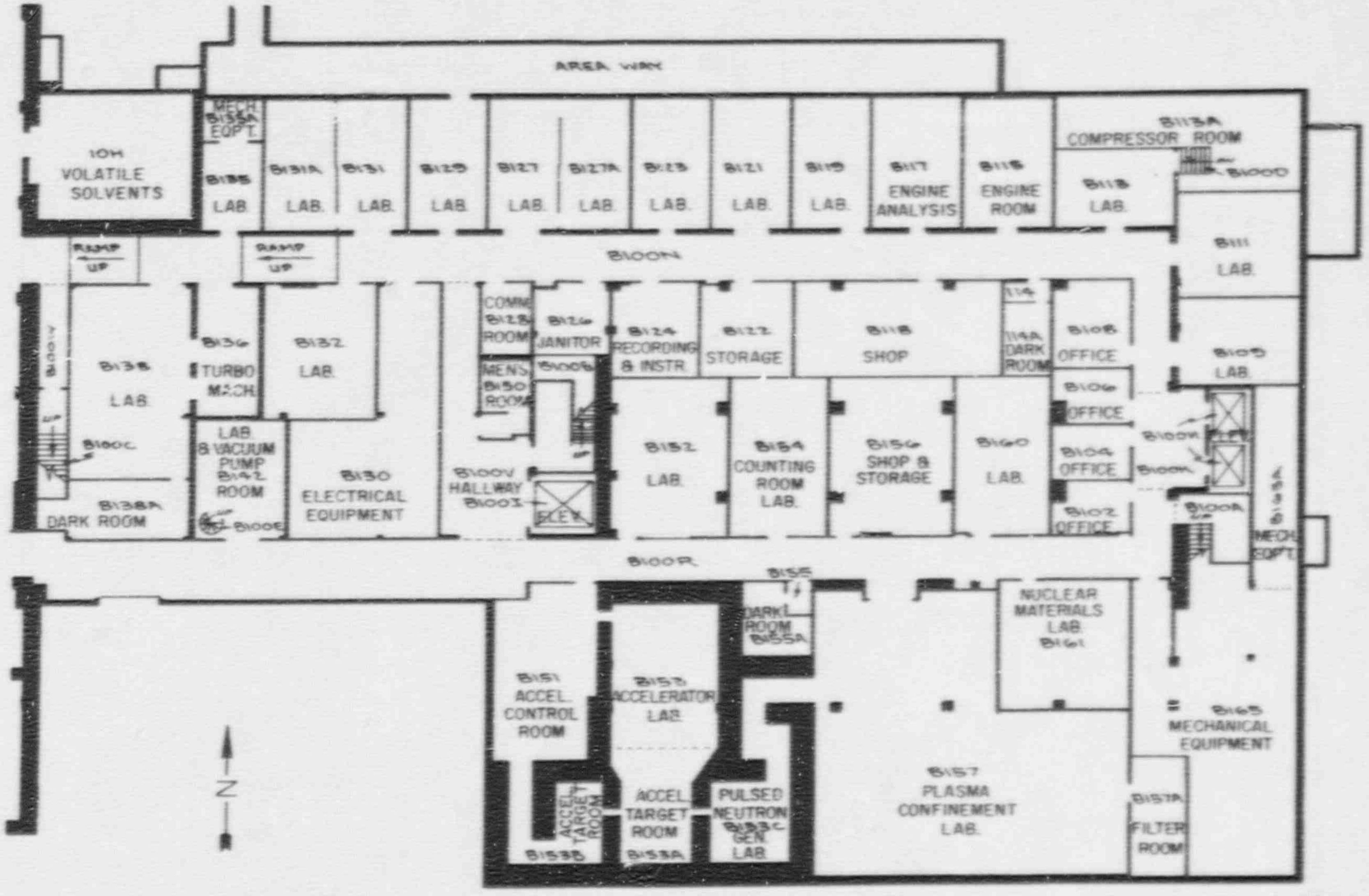


FIGURE 9

BASEMENT ENGINEERING RESEARCH BUILDING

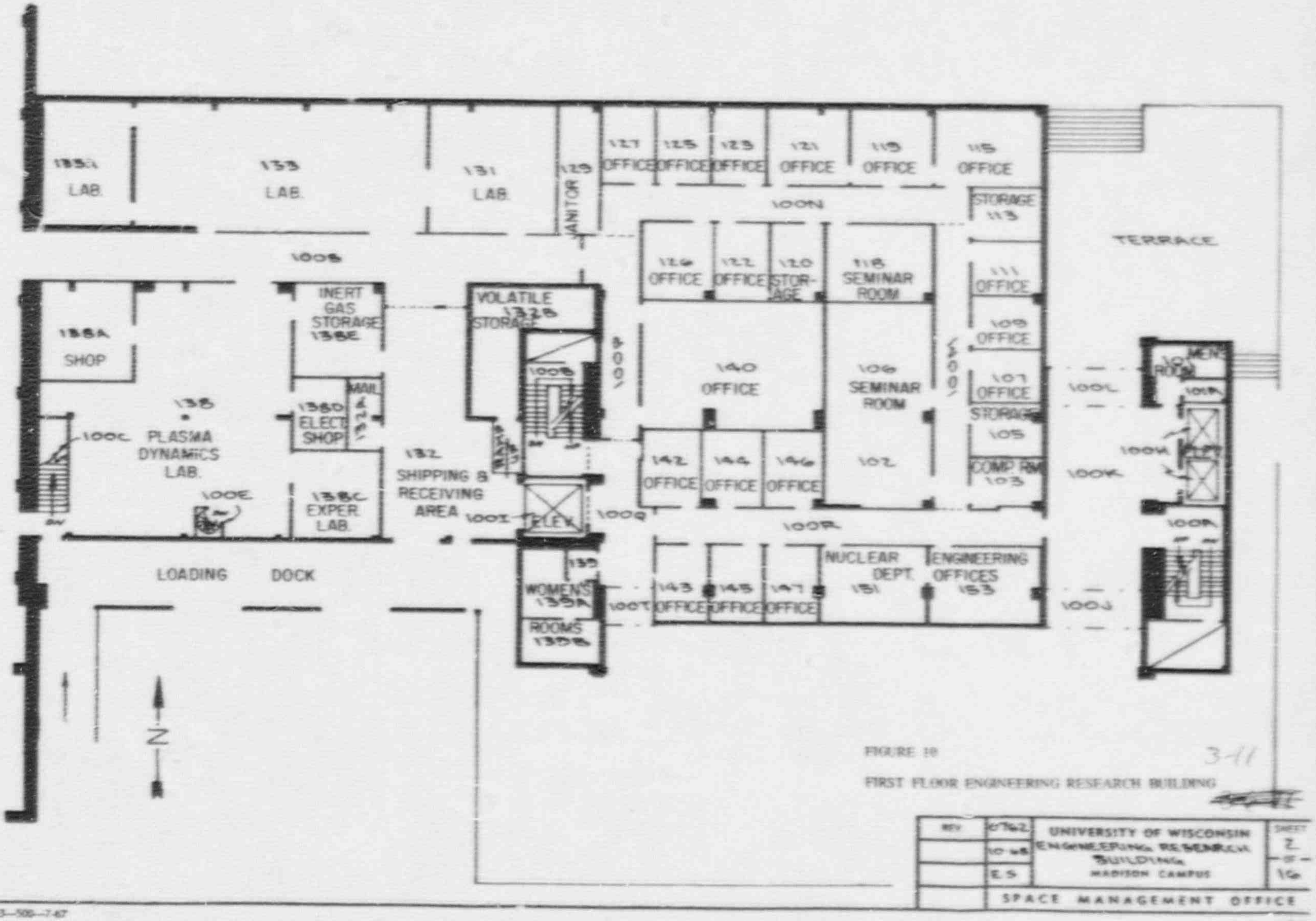


FIGURE 10

FIRST FLOOR ENGINEERING RESEARCH BUILDING

3-11

KAE—39-1153—500—7-67

Appendix

CALCULATION METHODS FOR ATMOSPHERIC RELEASE OF RADIOACTIVITY

References:

(1) Meteorology and Atomic Energy, U. S. Dept. of Commerce Weather Bureau, Govt. Printing Office, Washington, D. C (July 1955).

(2) F. A. Gifford, Jr., Atmospheric Dispersion Calculations Using the Generalized Gaussian Plume Model, Nuclear Safety, December 1960.

(3) Calculation of Distance Factors for Power and Test Reactors, (TID-14844), USAEC, Mar. 23, 1962.

A. Models Used for Calculations

For Sutton's diffusion model, the maximum concentration (X_{max}) at any point downwind is given as:

(1) $X_{max} = \frac{2Q}{e\pi\bar{\mu}\ {}^{2}}$, where (Ref. 1)

$\bar{\mu}$ = mean wind speed
Q = release rate, Ci/sec
h = stack height

For the generalized Gaussian Plume Model, the maximum concentration is given by the same equation (Ref. 2, eq. 8).

For calculations in this report, the following values are used:

$\bar{\mu}$ = mean wind speed = lowest monthly average = 3.54 meters/sec.

h = stack height above ground = 17.1 meters
and (2) $X_{max} = 2.26 \times 10^{-4}$ Q μc/ml.

Reference 2 presents a method applicable to release from buildings with zero stack height to approximate release from buildings with zero stack height to approximate release from leaks in a containment structure. This relation is given by equation 4 of the reference as

$$(3) \qquad X = \frac{Q}{(\pi\sigma_y\sigma_z + CA)\,\mu}$$

where X, Q, and μ are defined as above, and C is an empirical constant with a range from 1/2 to 2, and A is the minimum building cross section.

For calculations in this report we will neglect the atmospheric dispersion term in the equation, giving

$$(4) \qquad X = \frac{Q}{CA\mu} \quad .$$

Inserting values for this facility, and using a value of 1 for C,

$$(5) \quad X = \frac{Q}{(1)(30\text{ ft.})(44\text{ ft.})(9.29 \times 10^{-2}\text{m}^2/\text{ft}^2)(3.54\text{m/sec.})}$$

$$(6) \quad X = 2.3 \times 10^{-3}\ Q\,\mu\text{Ci/ml.}$$

B. Sample Calculations Supporting Section 6.2

The maximum release rate for A^{41} activity is 13.3 μCi/sec. and the resulting concentration is calculated to be, from equation (2)

$$(7) \quad X_{max} = (13.3 \times 10^{-6})(2.26 \times 10^{-4})$$
$$= 3.01 \times 10^{-9}\ \mu\text{Ci/ml.}$$

Calculation by equation (6) gives the value

$$(8) \quad X = (13.3 \times 10^{-6})(2.3 \times 10^{-3})\ \mu\text{Ci/ml.}$$
$$= 3.06 \times 10^{-8}\ \mu\text{Ci/ml.}$$

Although the two methods used to calculate the above values cannot both be applicable, the actual value will likely be between the two values above. The more conservative value is used in the text.

are derived as shown below:

$$(10) \qquad X_{Br^{83}} = \frac{(.0083\ Ci)(2.26 \times 10^{-4})}{4238}$$

$$= 4.43 \times 10^{-10}\ \mu Ci/ml.$$

The remaining values are calculated in the same manner.

The activity release was also evaluated through use of equation (6). (This calculation would be applicable particularly to leaks of the activity with the ventilation system not operating.)

Again using Br^{83} for an example, and assuming the same release time,

$$(11) \qquad X_{Br^{83}} = \frac{(.0083\ Ci)(2.3 \times 10^{-3})}{4238}\ \mu Ci/ml.$$

$$= 4.50 \times 10^{-9}\ \mu Ci/ml.$$

This value is a factor of 10.17 greater than that evaluated by the Gaussian Plume Model. All similar values in Table 1, columns H and I may be multiplied by this factor for a more conservative case.

C. Calculations Supporting Section 6.5.4

(a) Whole body exposure

The activity concentration of the insoluable volatiles in the reactor room air was determined by dividing released activity by room volume:

$$\frac{A}{V} = \frac{5.89 \times 10^{6}\ \mu Ci}{2 \times 10^{9}\ cm^{3}} = 2.95 \times 10^{-3}\ \mu Ci/cm^{3}$$

Since 3.7×10^{4} dpm = 1 μCi, $\frac{A}{V} = 109\ \gamma/sec\text{-}cm^{3}$.

The maximum dose rate is calculated by assuming the room is equivalent to a hemisphere with a radius of 782 cm. In addition, the average gamma energy is 0.7 MeV, the attenuation coefficient for the air is $3.5 \times 10^{-5}\ cm^{-1}$ and the flux to dose rate conversion factor is $4.2 \times 10^{4}\ \gamma/cm^{2}$ sec per mr/min.
Using the relation

$$DR = \frac{30S(1-\ {}^{-R\Sigma})}{C\ \Sigma}$$

where

DR = dose rate in mr/hr
S = volumetric source strength in $\gamma/sec\ cm^{3}$
R = outer radius of hemisphere
Σ = attenuation coefficient for air
C = flux to dose rate conversion factor

yields a dose rate of 60 mrem/hour.

(b) Dose to the Lungs

The dose to the lungs was calculated by first assuming uniform dispersal of the released volatiles in the laboratory volume, giving a concentration of

$$\frac{A}{V} = \frac{5.54 \times 10^{6}\ \mu Ci}{2 \times 10^{9}\ cm^{3}} = 0.00275\ \frac{\mu Ci}{cm^{3}}\ .$$

Since the "standard man" breathes 1.25 cubic meters of air per active hour, he would breathe approximately 0.210 m^3 in 10 minutes. If this number is increased to 0.300 m^3 to allow for excitement, then his lungs would be exposed to an activity of

$$\frac{\text{Activity}}{\text{volume}} \times \text{volume} = (0.00275 \frac{\mu\text{Ci}}{\text{cm}^3})(3 \times 10^5 \text{ cm}^3)$$

$$= 825 \ \mu\text{Ci} .$$

The dose to the lungs is then calculated from the following expression to be 1 rad.

$$\text{Dose (rad)} = \frac{ACR}{m} \sum_{i=1}^{8} \frac{FiEi}{\lambda i}(1 - e^{-\lambda it})$$

where: A = activity exposure (825 μCi)

C = conversions factor

$$\frac{3.7 \times 10^4 \ \beta/\text{sec} \ \ \mu\text{Ci})(1.6 \times 10^{-6}}{(100 \text{ erg/g-rad})} \ \frac{\text{erg}}{\text{MeV}}$$

R = lung retention factor (0.125 is customary)

m = mass of lungs (1000 grams)

Fi = fraction of total activity

Ei = energy of beta for nuclide i(MeV)

λi = radioactive decay constant + biological release constant (6.7×10^{-8} sec^{-1})

T = time of exposure (assumed infinite)

D. Sample Calculations Supporting Section 6.5.5

Release rate, Q, for an isotope is the total quantity released to air (column E of Table 1, Chapter 6) divided by the assumed release time. The release time used in further calculations is the time required for the exhaust fan to make a complete air change, i.e.

(9) $T \text{ release} = 2000 \text{ m}^3/0.472 \text{ m}^3/\text{sec} = 4238 \text{ sec.}$

Using the generalized Gaussian Plume Model (i.e., equation 2), and demonstrating with data for Br^{83}, concentrations released to unrestricted areas (Table 1)

TABLE OF POPULATION DISTRIBUTION*

FIGURE 11

Radius of Circle with Reactor at Center		Permanent Population**
500 ft.		0
1,000 ft.		686
1,500 ft.		1,687
2,000 ft.		4,497
5,000 ft.		26,259
5 miles	(approx.)	185,000

Total Population, City of Madison	173,258
Total Population of Suburbs	42,082
Total Urban Population	215,340**
Total University Student Population in Madison during academic year	32,806
Maximum number of Spectators contained by Camp Randall Football Stadium	78,280
Maximum number of Spectators contained by Field House	13,140
Estimated maximum number of Students attending classes at any one time in the East Wing of Mechanical Engineering Building during last year	300

NOTE: Of this 300, 150 were in classrooms on the third floor of the wing.

Estimated maximum number of Students attending classes or doing research at any one time in the Central Wing of the Mechanical Engineering Building during the last year	50

---

* Taken from 1970 census data

** Permanent population does not include students.

*** Includes the cities of Madison, Middleton and Monona, the villages of Cottage Grove, McFarland, Maple Bluff, Shorewood Hills, Verona and Waunakee; the <u>entire</u> towns of Blooming Grove, Cottage Grove, and Madison; and <u>parts</u> of the towns of Burke, Dunn, Fitchburg, Middleton, Pleasant Springs, Springfield, Verona and Westport.

wing of the Mechanical Engineering Building except for a few offices along the north and south walls. The south end of the Laboratory is being used for graduate student research and the north end is being used for undergraduate experiments. Eventually the entire Lab will be used by undergrads. The basement area north of the Nuclear Engineering Laboratory is being used for storage for the Heat Power Lab and for some undergraduate experiments.

3.1.2 Topography

The campus is set on a narrow neck of land between Lakes Mendota and Monona. Lake Mendota (15 square miles) lies northwest of Lake Monona (5 square miles) and the lakes are only 2/3 of a mile apart at one point. Drainage at Madison is southeast through the Yahara River into the Rock River which flows south into Illinois and then west to the Mississippi.

Madison is a glaciated area, and the topography tends to be irregular. In general, the terrain is hilly although the hills are not large.

3.1.3 Geology and Hydrology

In the vicinity of the reactor, a glacial deposit exists which consists mostly of sands and gravels but may be quite variable and contain clay and large boulders. Although this deposit may be as much as 100 feet thick, it is probably less than twenty. The reason for the variation in thickness is that the bed-rock sandstone which underlies the deposit is very uneven. The bed-rock consists of Cambrian sandstones which are 700 to 800 feet thick and which are permeable to water. No borings are available at the specific reactor site, but sandstone was found very close to the surface within about 1500 feet of the site. Below the sandstone is impermeable basement rock. A chart showing the geology at a well about 2,000 feet from the reactor site is shown in Figure 12.

The Cambrian sandstone layer constitutes a water aquifer, and the ground water flow from the reactor site is generally toward Lake Mendota-

Yahara River - Lake Monona system.
Thus, the general flow is toward the east and south. However, as discussed below, Madison obtains its water from wells drilled into the Cambrian sandstone. Consequently, local areas of depression of the water surface caused by pumping of the city wells could cause flow from the reactor site toward the wells.

3.1.4 Water Supply

Madison obtains its drinking water supply from several wells drilled into the Cambrian sandstone described above. The location of these wells is shown on Figure 2, and they supply the University as well as the city. All of these wells are cased from ground level into the sandstone so as to keep out water from the glacial deposit. The closest well to the reactor site is about 2,000 feet southeast.

An analysis was made of the possibility that loss of water from the reactor or from the hold tanks could affect the city water supply, and a negative result was obtained as indicated below. There are two methods by which such water may leave the reactor room: (1) by flow through the floor drains and (2) by loss through the floor and into the ground. In so far as the first method is concerned, it was ascertained that the flow through the floor drains would empty into a sanitary sewer main*. From there it would travel through mains via a pumping station to the main sewage plant, located south of and outside the corporate limits of the city. From there, the sewage travels through mains an additional five miles to the south before it empties into an open ditch. On the way, any water from the reactor would become considerably diluted since the minimum flow-rate into the ditch is 7,000 gpm whereas the

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*The Reactor Laboratory floor drain empties into the hold tank. Should the entire contents of the pool be let out into the room, however, some water could escape into the sewer system through a drain thimble into which waste water is pumped from the hold tank.

CITY UNIT WELL NO. 4, MADISON, WIS.
Randall and Regent Street
W. H. Cater, Contractor, 1930
Elevation 853.6

DRIFT	67	0-5	5	Soil, dark grey (fill)
		5-15	10	Sand, fine, gray, glacial, dolomitic
		15-67	52	Sand, medium, gray, glacial, dolomitic
FRANCONIA	73	67-75	8	gravel stones to 1/2 inch
		75-85	10	Sandstone, medium to fine, lt.gy.dol, glac.
				Ss. fine to med., white, part yel-gy, dol.
		85-115	30	Sandstone, medium to coarse, white
		115-120	5	Sandstone, coarse, lt yel-gy, part dol.
		120-125	5	Sandstone, fine to med., lt.gray, dol.
		125-140	15	Sandstone, fine to med, some yel, dol.
DRESBACH	70	140-210	70	Sandstone, medium, white
EAU CLAIRE	240	210-220	10	Sandstone, med. to fine, lt.gray, dol.
		220-2 5	5	Dolomite, lt. gray, sandy
		225-230	5	Shale, greenish gray, red, dol.
		230-335	105	Sandstone, fine to medium, lt.gray, dolomitic
		335-360	25	Sandstone, fine to med., lt gray, pink dolomitic layers
		360-375	15	Sandstone, fine to medium, white
		375-420	45	Sandstone, medium, white, some dolomitic layers
		20-430	10	Sandstone, medium to fine
		430-450	20	Sandstone, medl, white, pink dol. layers
MT SIMON	265	450-450	10	Sandstone, medium, white
		460-515	55	Sandstone, medium, white, a few dol. layers
		515-545	30	Sandstone, fine to coarse, white
		545-565	20	Sandstone, med. to fine, white, some dol.
		565-580	15	Sandstone, medium, white
		580-610	30	Sandstone, coarse to medium, white
		610-650	40	Sandstone, very coarse to med., lt. gray
		650-690	40	Sandstone, coarse to fine, light gray
		690-715	25	Sandstone, very coarse to medium, light
PC	22	715-737	22	Rhyolite, weathered, red

Formations: Drift; Franconia; Dresbach (Galesville); Eau Claire; Mt. Simon; pre-Cambrian

FIGURE 12 - GEOLOGY

expected to be low enough that no hazard exists.

3.1.5 Seismology

Reference to Figure 1 of "United States Earthquakes", U.S. Department of Commerce, Coast and Goedetic Survey--Washington, by R. A. Eppley, shows that there have been no recorded destructive or near destructive earthquakes in the State of Wisconsin through the year 1971. Further, a survey of "Preliminary Epicenters" Reports (daily to weekly publication by the U.S. Dept. of Commerce Environmental Research Laboratories) printed since 1971, indicates that there have been no such earthquakes since 1971. The records do indicate that three such earthquakes have occurred in northern Illinois within 125 miles of Madison; two occurring in 1804 and 1912 had intensities of VII to VIII; a third occurring in 1909 about 75 miles from Madison had an intensity of VIII to IX. The latest tremor felt in Madison occurred Sept. 14, 1972. Information obtained from Mr. Waverly Person of the Environmental Research Laboratories in Boulder, Colorado indicated the epicenter of the quake was located about five miles south of Davenport, Illinois (about 125 miles from Madison). The quakes magnitude was 4.5 on the Richter Scale; its intensity in Madison was estimated to be no greater than VI-VII on the Modified Mercalli Scale. Even considering the four quakes mentioned above, it is apparent that Wisconsin is not the center of strong earthquake activity.

3.1.6 Climatology

Madison has the typical continental climate of North America with a large annual temperature range, and with frequent short period temperature changes. The absolute temperature range is from 107°F to -30°F. Winter temperatures (December-February) average 20° and the summer average (June-August) is 70°F. Daily mean temperatures average below 32° for 108 days and above 42° for 210 days of the year.

Madison lies in the path of the frequent cyclones and anticyclones which move eastward over this area during the fall, winter, and spring. In the summer the cyclones have diminished intensity and tend to pass farther north. The most frequent air masses are of probable maximum rate of entry into the floor drain would not be more than 10 to 100 gpm. These points, coupled with the fact that stringent administrative precautions will be taken to ensure that water contaminated beyond established tolerance levels is not released to the drain, tend to preclude that the city water supply could be adversely affected by this method.

The possibility that the city water supply could be affected via the second method is also negligible. The base of the reactor is about 8 feet below ground level, and water cannot be dissipated via surface run-off. Since the walls of the building surrounding the reactor are made of concrete up to ground level, significant water loss through the floor could result only if the concrete was breached. In fact, it would appear that the only mechanism by which contaminated water could enter the soil would be the result of an earthquake sufficiently severe to rupture both the reactor tank and shield, as well as the floor of the building, at a time when the reactor pool water was radioactive beyond tolerance levels. Such a set of coincidental occurrences is considered extremely remote. Further, even if it did occur, there is no assurance that the water supply would be adversely affected. For example, the nearest city well is about 2,000 feet from the reactor site, and it has been estimated* that water would flow through the sandstone from the reactor to the well at not more than 0.1 foot per day. Thus, as long as 55 years might be required for the reactor water to reach the well.

Should the hold tanks rupture, a similar analysis indicates no adverse effect on the well. Furthermore, the quantities of water likely to be lost are small and activities are

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*Note: The basic geological and hydrological information in Sub-Sections 3.1.2, 3.1.3, and 3.1.4 has been supplied by Mr. C. Lee Holt, Jr., District Geologist, and Mr. Denzel R. Cline, Geologist, Ground Water Branch, U. S. Geological Survey.

3.1.7 Meteorology

Tables 1, 2, 3, and 4 present data on temperature, precipitation, and snow, as well as on miscellaneous factors. The data for the first three tables were obtained from "Local Climatological Data with Comparative Data, 1955, Madison, Wisconsin", U. S. Department of Commerce, Weather Bureau publication. It may be seen that Madison's weather is reasonably typical of the weather in this region of the country. Figures 13 and 14 present surface and upper air wind rose data.

TABLE 1

AVERAGE TEMPERATURE FOR MADISON*

	Average Daily Maximum Temperature	Average Daily Minimum Temperature	Average Monthly Temperature
January	26.7°F.	11.8°F.	19.3°F.
February	29.4	14.3	21.9
March	40.0	24.8	32.4
April	54.6	37.1	45.9
May	67.1	48.3	57.7
June	76.4	58.9	67.7
July	82.2	64.0	73.1
August	79.8	62.0	70.9
September	71.3	54.4	62.9
October	60.0	43.4	51.7
November	42.9	29.4	36.2
December	30.0	17.0	23.5
Year	55.0	38.8	46.9

*Normal values based on the period 1921-1950, and are means adjusted to represent observations taken at the present standard location (North Hall) on the campus.

polar origin. Occasional outbreaks of arctic air affect this area during the winter months. Although northward moving tropical air masses contribute considerable cloudiness and precipitation, the true Gulf air mass does not reach this area in winter and only occasionally at other seasons. Summers are pleasant with only occasional periods of extreme heat or high humidity. Noon relative humidity averages 52 percent in July and 54 percent in August. The average summer has only eight days with temperatures above 90°.

There are no dry and wet seasons, but 58 percent of the annual precipitation falls in the five months of May through September. Cold season precipitation is lighter but lasts longer. Soil moisture is usually adequate in the first part of the growing season. During July, August, and September, the crops are dependent on current rainfall which is mostly from thunderstorms and tends to be erratic and variable. Average occurrence of thunderstorms is on fifteen days during July and August.

March and November are the windiest months. Tornadoes are infrequent. The average occurrence for Dane County is about one tornado in every three to five years.

The ground is covered with an inch or more of snow about 60 percent of the time from December 10 to February 25 in an average winter. The soil is usually frozen from the first of December through most of March, with an average frost penetration of 25 to 30 inches. The growing season averages 175 days. The most probable period (50% of the years) for the last killing freeze in spring is April 17 to May 2. The first killing freeze in autumn is most probable from October 6 th through 25th. The latest recorded killing freeze was on May 25, 1925 and the earliest in fall was on September 16, 1916.

TABLE 4

MISCELLANEOUS WEATHER DATA FOR MADISON

Average number of days per year with precipitation of 0.01 inch or more . .	115
Average number of cloudy days per year . . .	145
Average number of partly cloudy days per year	120
Average number of clear days per year . . .	100
Average daily maximum summer (June-August) temperature (°F)	77.5
Average daily minimum winter (December-February) temperature (°F)	14.4
Record highest temperature, (July, 1936) . .	107
Record lowest temperature, (January, 1963). .	-30
Maximum monthly precipitation (September, 1915) (inches of water). .	10.69
Minimum monthly precipitation (October, 1889)	Trace
Maximum monthly snow, sleet, hail (January, 1929)	31.8

TABLE 2

NORMAL PRECIPITATION FOR MADISON*

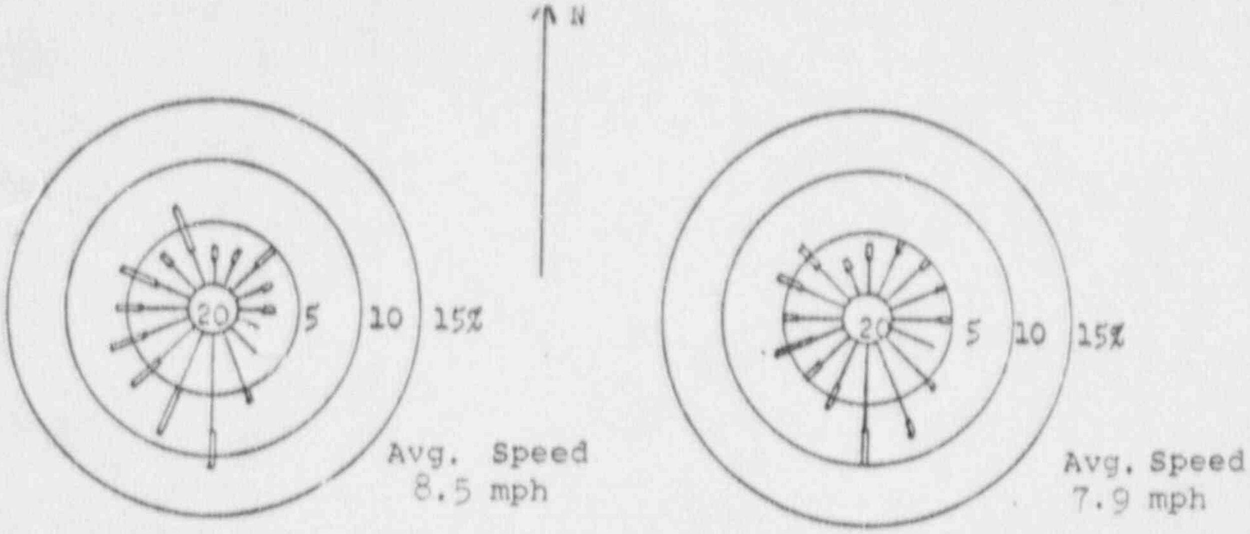
	Normal Total Precipitation
January	1.47 inches
February	1.27
March	2.03
April	2.49
May	3.21
June	4.02
July	3.40
August	3.07
September	4.11
October	2.00
November	2.20
December	1.44
Annual	30.71 inches

*Normal values based on the period 1921-1950, and are means adjusted to represent observations taken at the present standard location (North Hall) on the campus.

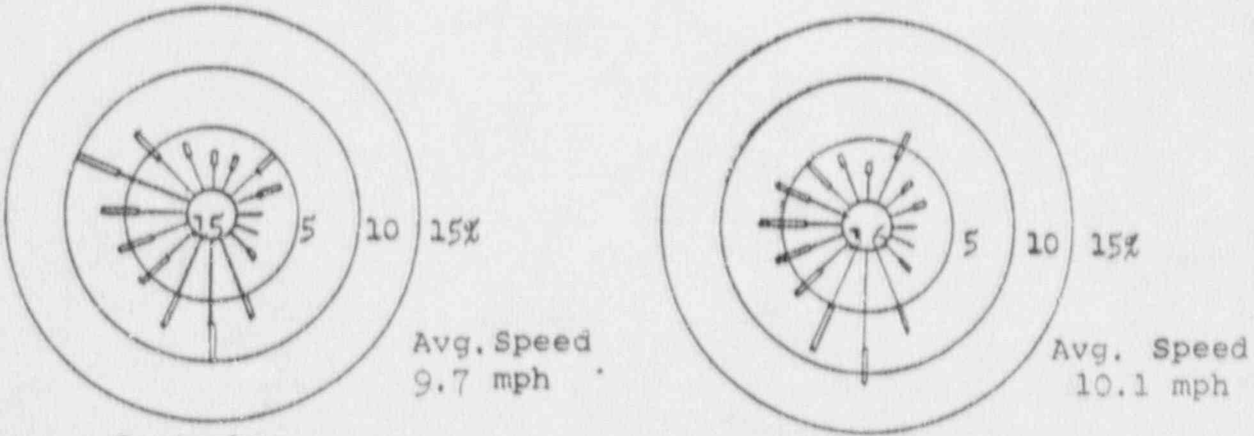
TABLE 3

SNOW, SLEET, AND HAIL FOR MADISON

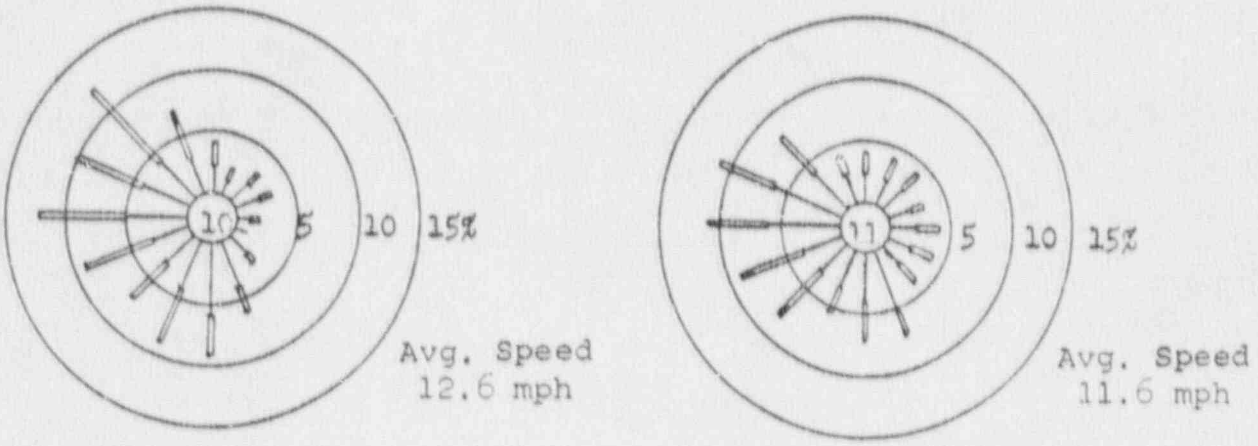
	Mean Total	Maximum Monthly	Year
January	9.6 inches	31.8 inches	1929
February	7.8	21.9	1898
March	7.9	28.4	1923
April	1.6	13.0	1921
May	0.1	5.0	1935
June	Trace	Trace	1954
July	0	0	
August	Trace	Trace	1949
September	Trace	Trace	1953
October	0.3	5.0	1917
November	3.1	14.9	1940
December	7.5	23.1	1909
Year	37.9 inches		



July August



September October



November December

FIGURE 13, CONTINUED

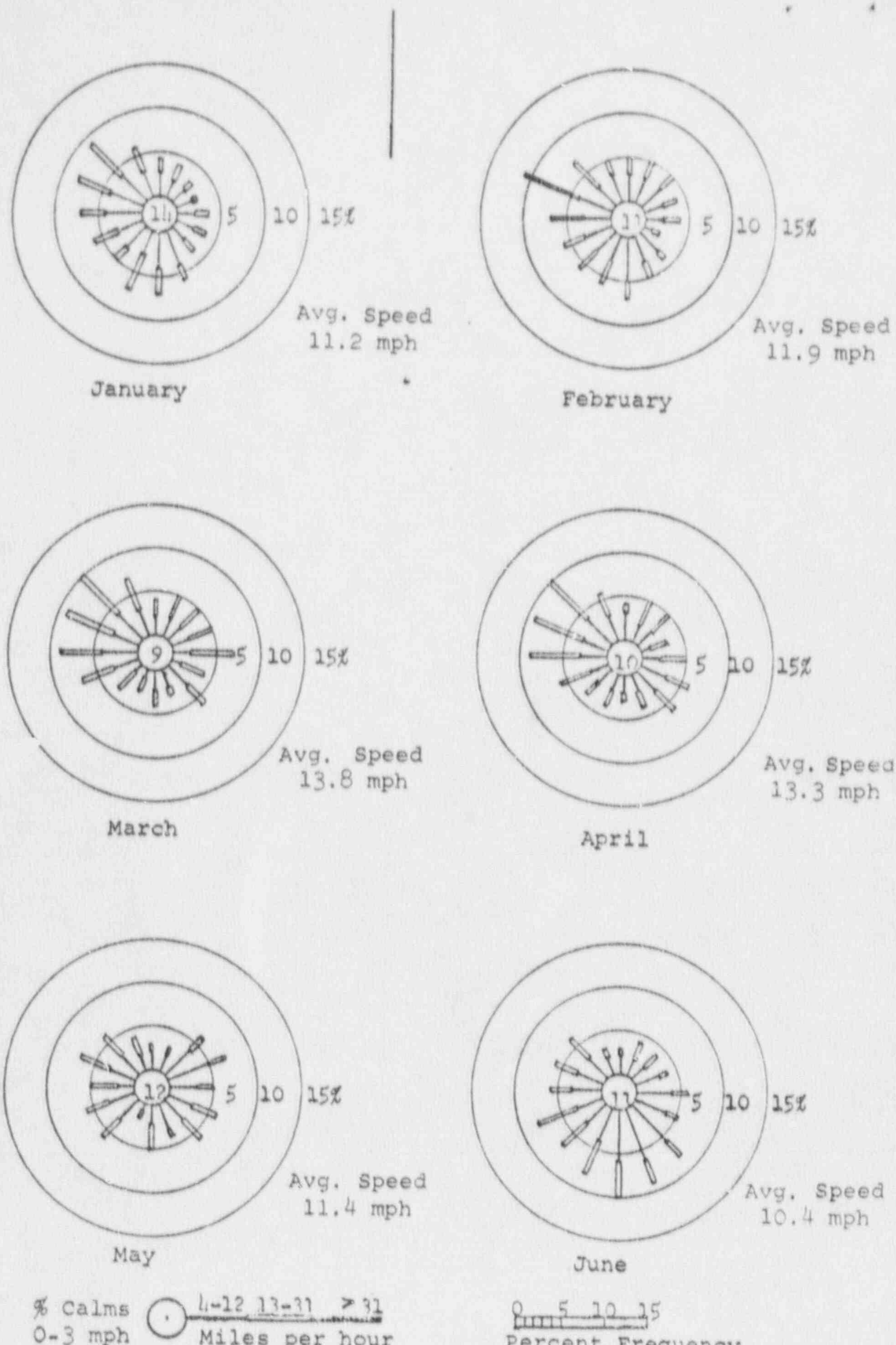
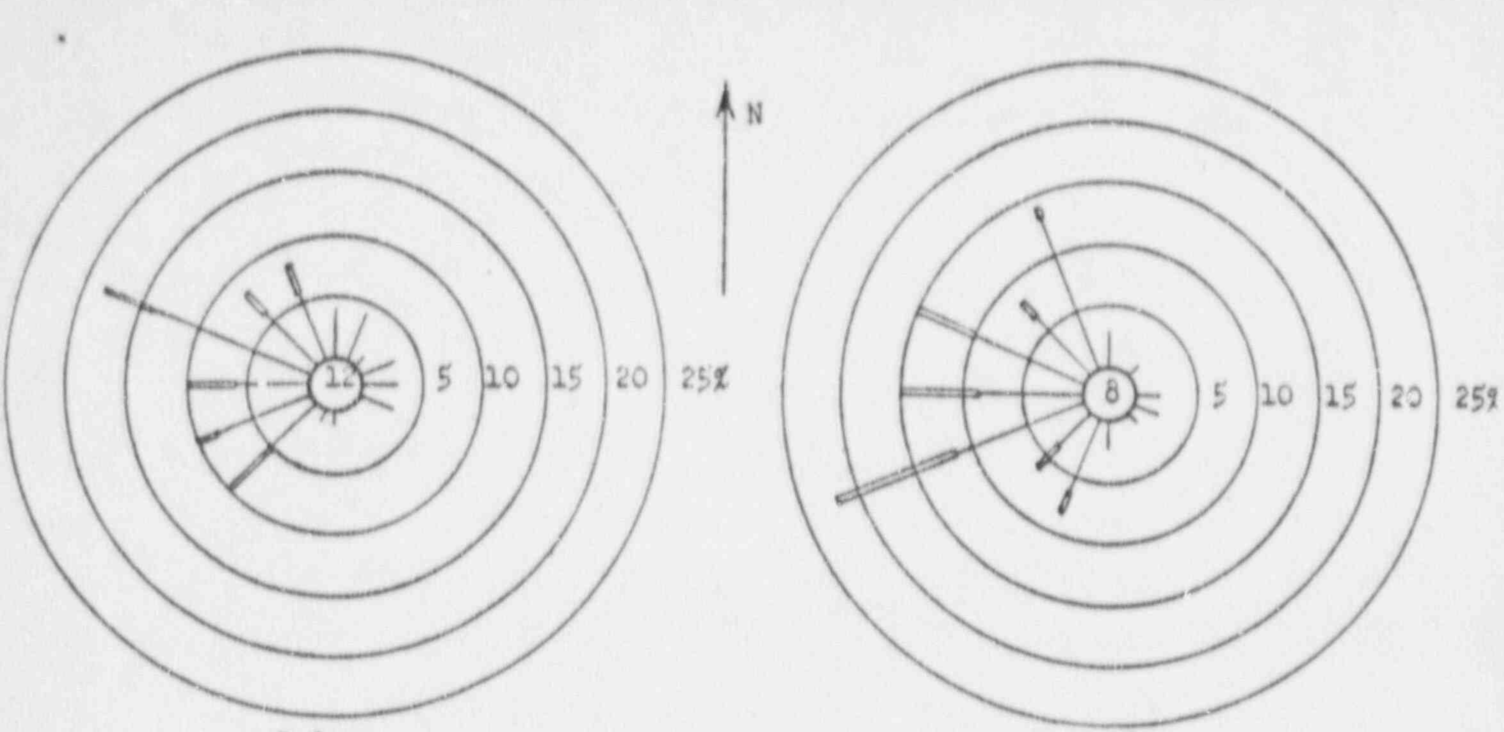


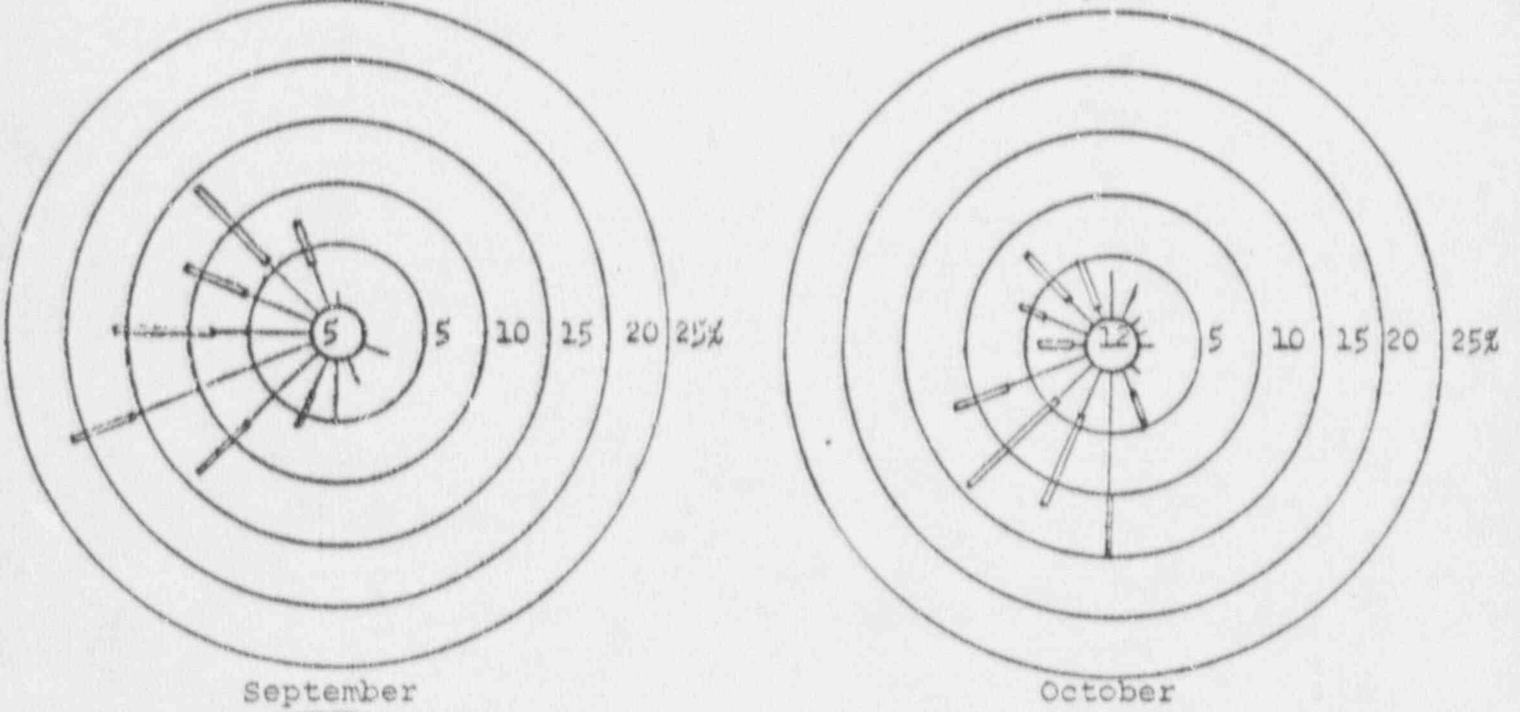
FIGURE 13

SURFACE WIND ROSES, 1949 - 1954, Madison, Wisconsin



July

August



September

October

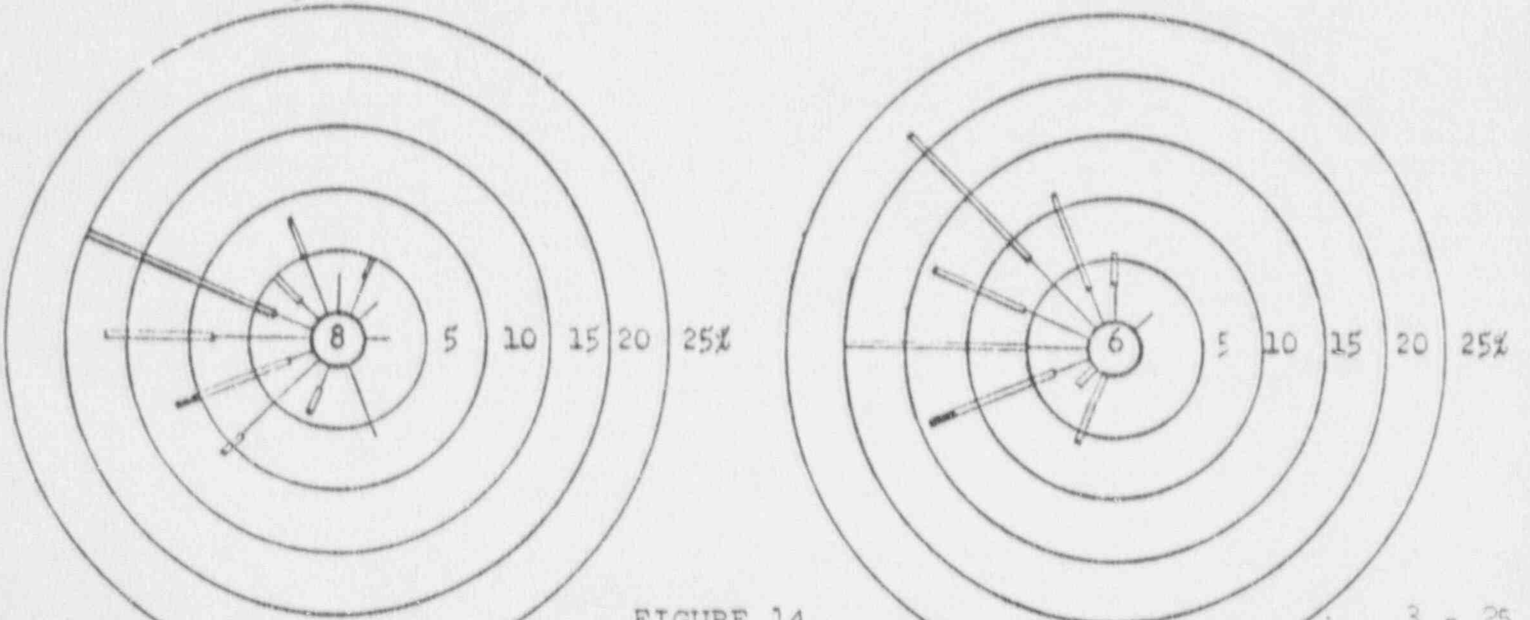
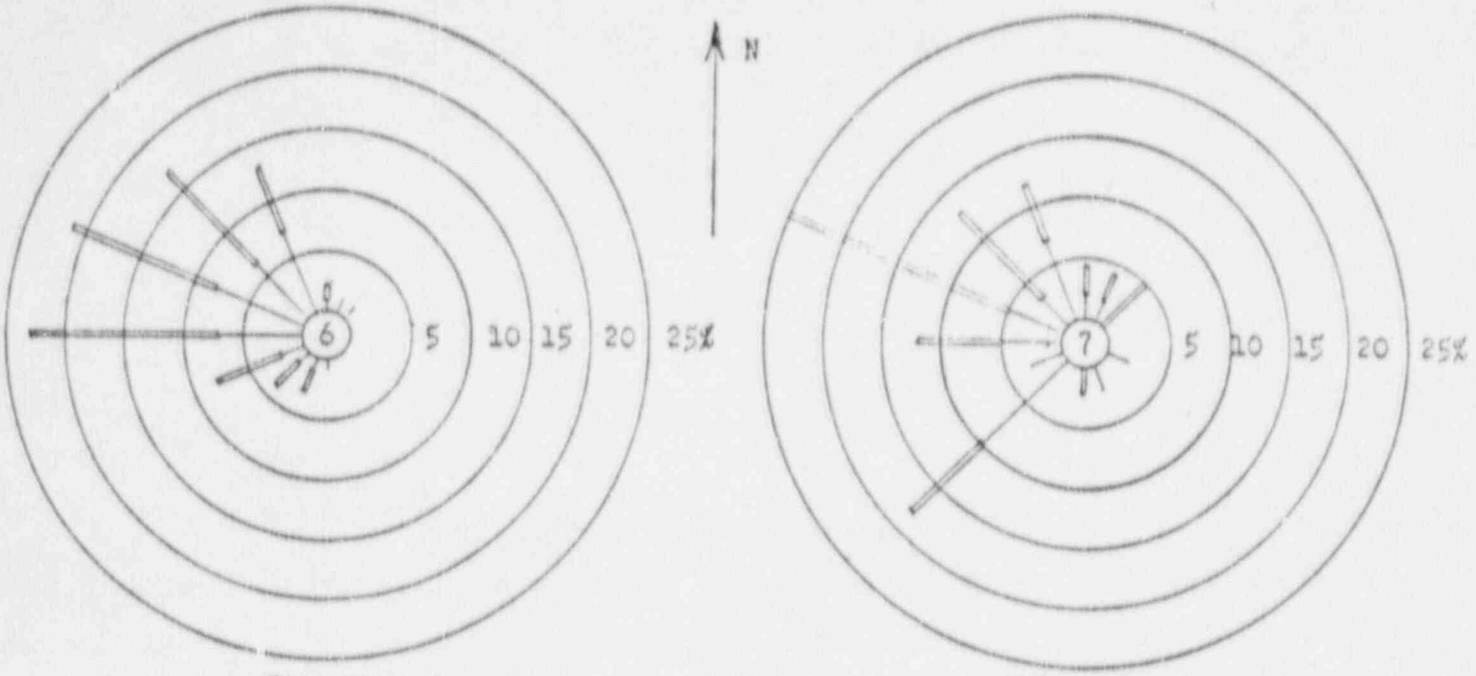


FIGURE 14

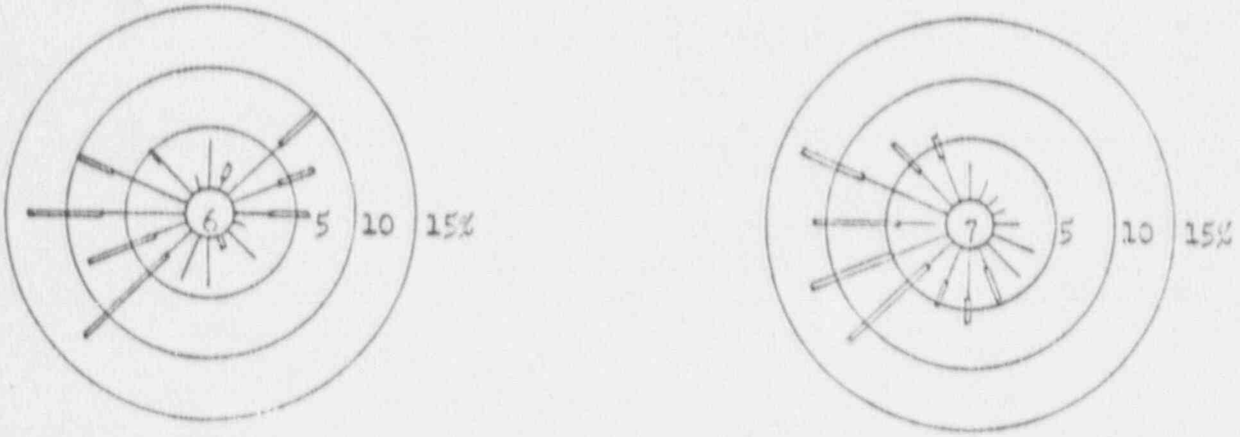


January

February

March

April



May

June

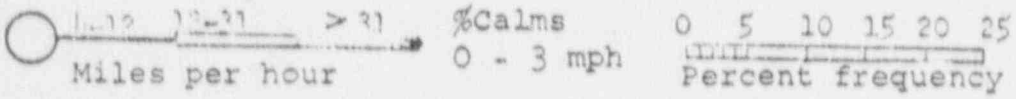


FIGURE 14

UPPER AIR WIND ROSES; ELEVATION 4000 FT., JULY 1956 - JUNE 1957

MADISON, WISCONSIN

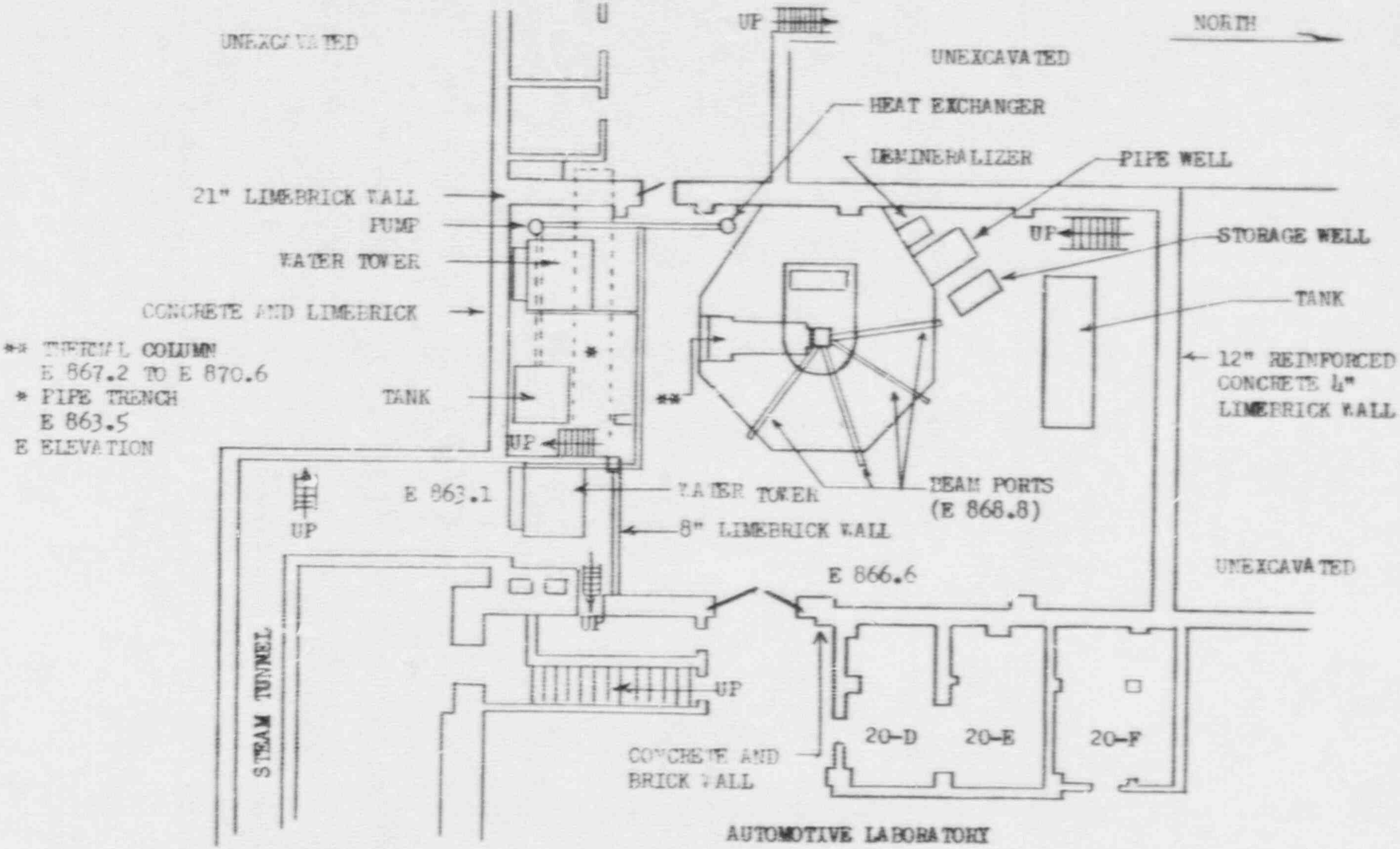


FIGURE 15

BASEMENT FLOOR LEVEL LAYOUT

3.2 BUILDING

3.2.1 General Description

The Reactor Laboratory is shown in Figures 15 through 20. The Laboratory is about 43 by 70 feet with a ceiling height of approximately 36 feet in most of the room. The portion of the ceiling above the console area is at a height of 22 feet.

The floor of the room is concrete laid on the ground. The walls are concrete and brick. The roof is a 1-1/2 inch steel deck with 2 inches of rigid insulation and a 4-ply, built-up surface.

The console area is located in the southwest corner of the Reactor Laboratory. It is separated on the north and east sides from the laboratory proper by wire reinforced glass. This acts effectively to reduce noise originating from surrounding laboratories and equipment. Reactor Laboratory air is circulated through the console area for ventilation.

The Reactor Laboratory has three windows which face the parking lot and stadium. There are five single doors; two opening on the west wall into the Nuclear Engineering Laboratory at ground level, one opening on the west wall into the basement level of the Nuclear Engineering Laboratory, one opening into the parking area, and one opening into the Heat Power Laboratory. One double door opens into the Nuclear Engineering Laboratory to the east of the Reactor Laboratory. All doors have glass panes which are covered with expanded steel grating.

The Nuclear Engineering Laboratory to the east of the Reactor Lab includes Room 5 of the Mechanical Engineering Building and extends down the corridor of the Engineering Research Building (connected by double doors) and includes Rooms B-154, B-157 and B-160 of the Engineering Research Building. Room 5 is used primarily for teaching laboratory courses and contains a subcritical assembly, a reactor simulator, a hot lab, and various detection and counting instruments.

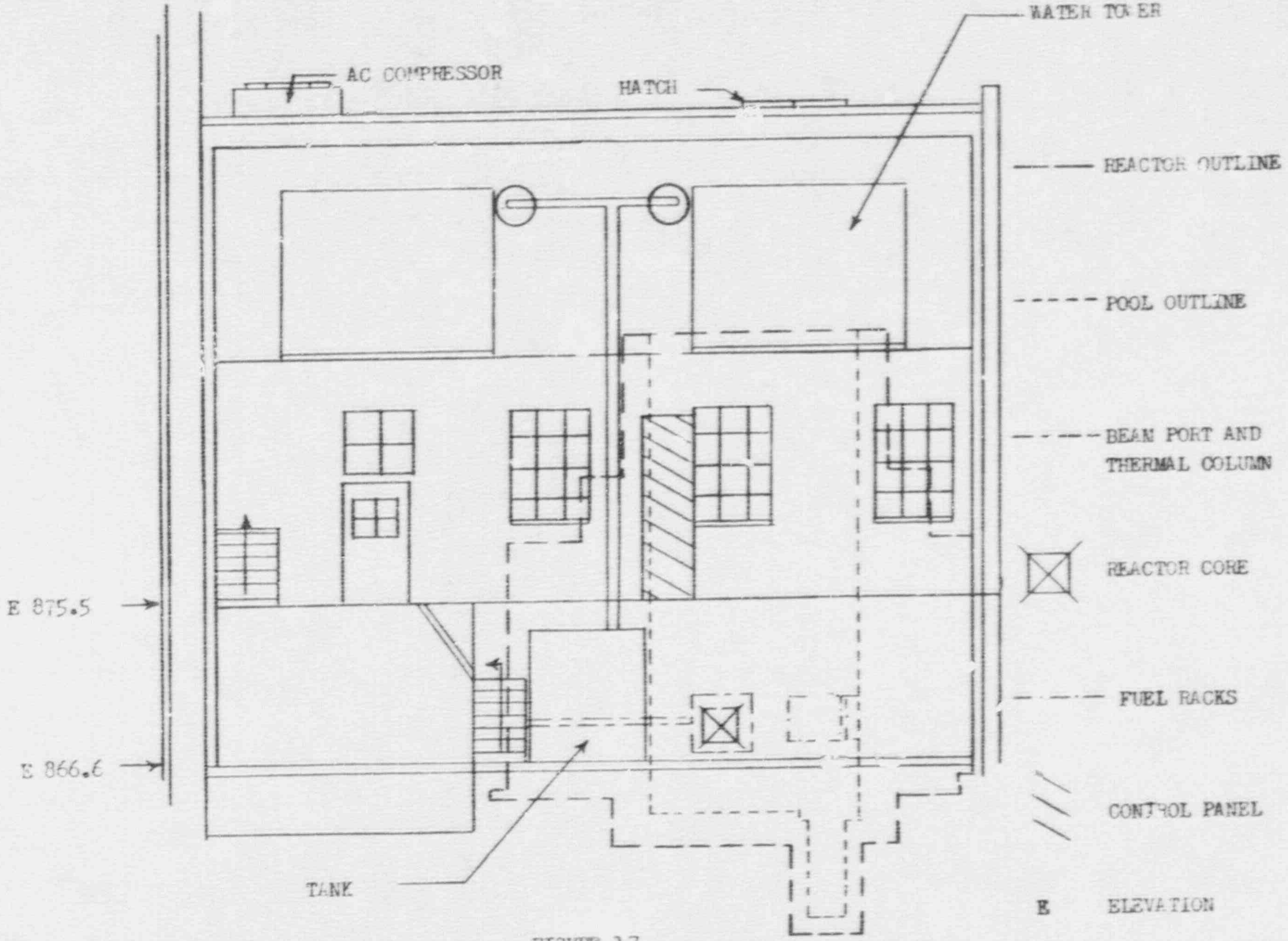


FIGURE 17

LAB LOOKING SOUTH FROM ℄ OF CORE

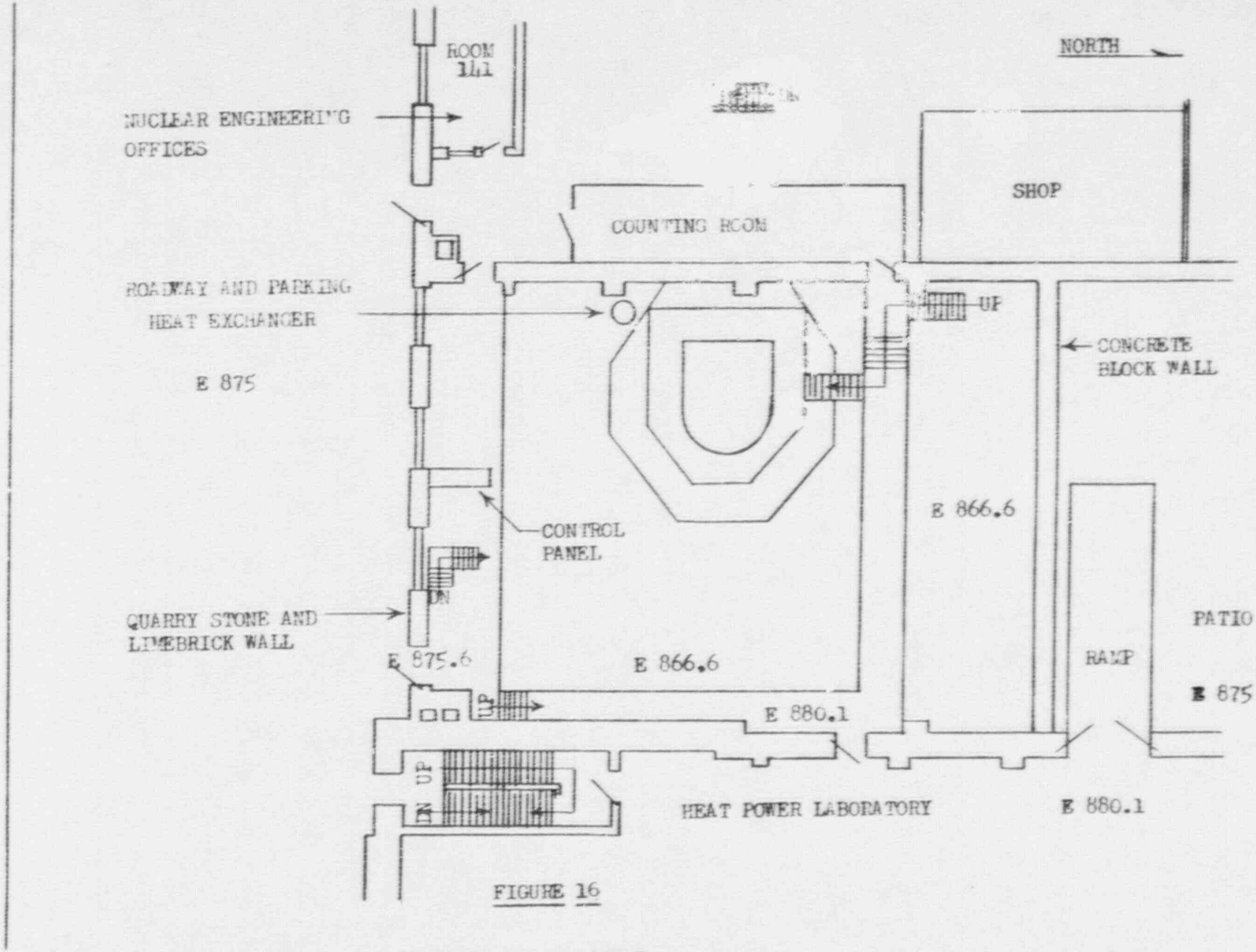


FIGURE 16

FIRST FLOOR LAYOUT

The basement level of the Nuclear Engineering Laboratory to the west contains small rooms having concrete walls on the order of 8 inches thick. The small rooms house the pneumatic tube equipment and dispatch station, a radiochemistry laboratory, and both general storage and radioactive storage areas. An instrumentation shop is located on the east end and a machine shop area for the Reactor Lab is located on the west end.

The ground-floor level of the Nuclear Reactor Laboratory on the west houses graduate student research and office areas and an office area for the Reactor Laboratory.

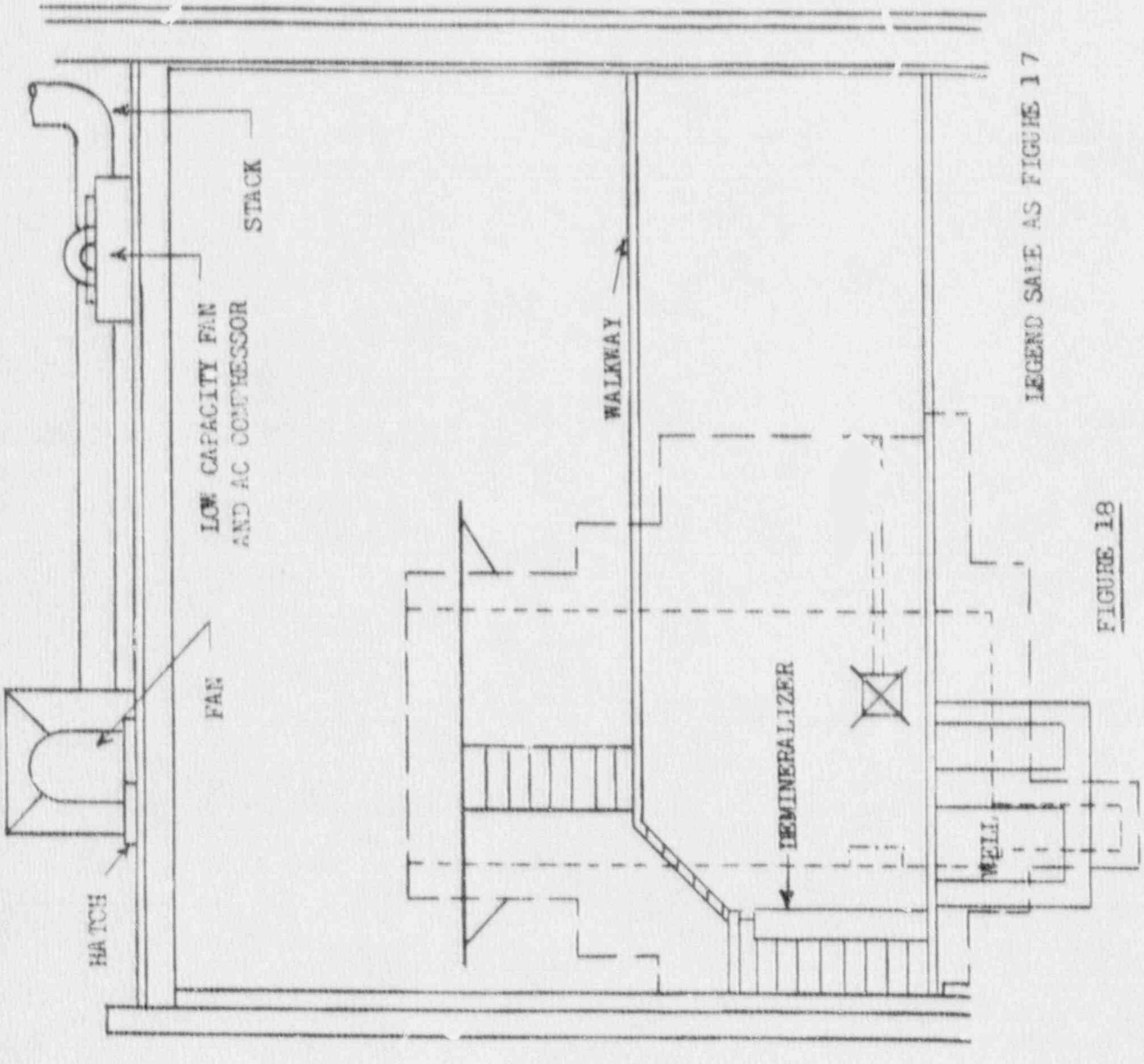
Plans exist for building a room inside the Reactor Lab which will include the upper level along the north wall. It would be used either for offices for the Reactor Laboratory or for an electronics shop.

The Reactor Laboratory is a restricted area. All doors are kept locked at all times except when authorized personnel are in the room. Keys are issued to a small number of authorized personnel.

3.2.2 Heating and Air Conditioning

The Reactor Laboratory is heated by the exposed steam pipes running along the east wall of the room and by steam heated convectors located in the laboratory. The convectors circulate the air in the room, and do not cause exchange of air with other areas.

Two air conditioning systems are provided. The first, of about 5-ton capacity, is controlled by a thermostat located in the control console and provides cooling air and humidity control for the



LEGEND SAME AS FIGURE 17

FIGURE 18

REACTOR LAB LOOKING NORTH

The ventilation system is designed to prevent the spread of airborne radioactive material into occupied areas. Accidents which might result in discharge of radioactive material from the stack are discussed elsewhere in this report, and remarks may be found there indicating the concentrations which might be expected. Justification of such deliberate discharge, especially from use of the large fan, is also indicated.

Figures 19 and 20 show the locations of the system components. Figure 21 is a schematic diagram of this system.

The high-efficiency filters used in this system and in the fumehoods in which pneumatic tube stations are located are all "absolute" filters manufactured by Cambridge or Flanders. These filters provide 97.97% efficiency for 0.3 micron particles. All but the emergency exhaust fan filters have prefilters.

instrumentation. Part of the cooled air from this unit is exhausted into the Reactor Lab. The system brings in no outside air, nor does it discharge any air from the laboratory. The second air conditioning unit, of about 20-ton capacity, cools and alleviates humidity problems in the remainder of the Reactor Laboratory. It consists of an air-cooled condenser and compressor located on the roof of the building, and evaporator-convectors located in the Reactor Laboratory. This unit does not cause air flow into our out of the Reactor Laboratory.

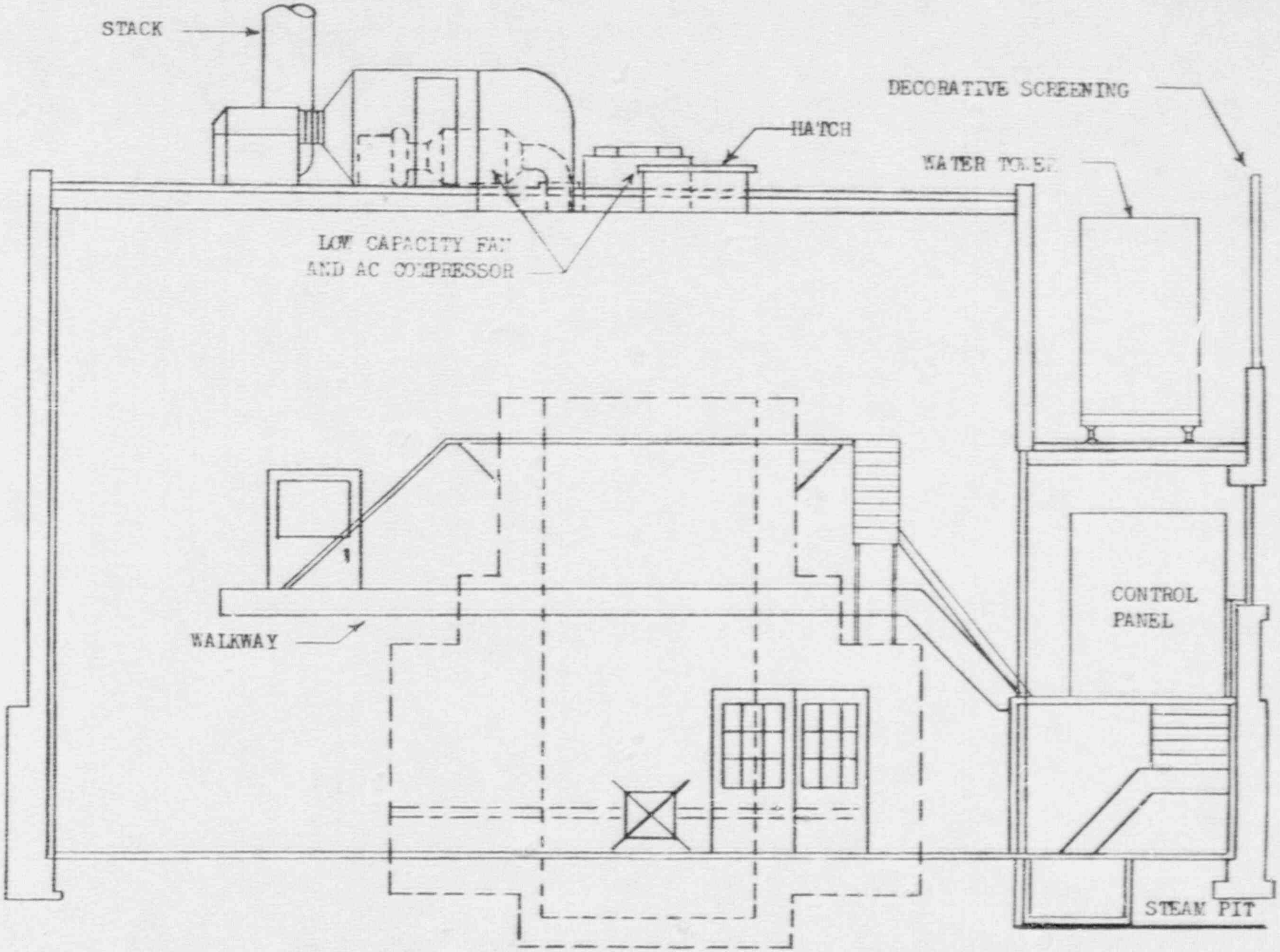
Condensate water from both air conditioning systems goes to the building sewers. Samples may be obtained for assay.

3.2.3 Ventilation

The ventilation system for the Reactor Laboratory consists of two fans with high-efficiency filters, ductwork, backdraft dampers, and a stack discharging above the roof of the east wing of the Mechanical Engineering Building.

The smaller fan (the room exhaust fan) has a free air capacity of about 1200 cfm. It is normally "on" to assure that any air flow is from adjacent areas into the Reactor Laboratory.

The larger fan (the emergency exhaust fan) is for use in those cases in which it is considered desirable to completely change the air in the Laboratory to prevent spread of contamination to adjacent occupied areas. It has a free air capacity of 12,000 cfm. This fan may also be operated to force large quantities of outside air through the stack to dilute air exhausted from the stack to permissible concentration of radioactive materials.



LEGEND SAME AS FIGURE 17

FIGURE 20 -- REACTOR LAB LOOKING EAST

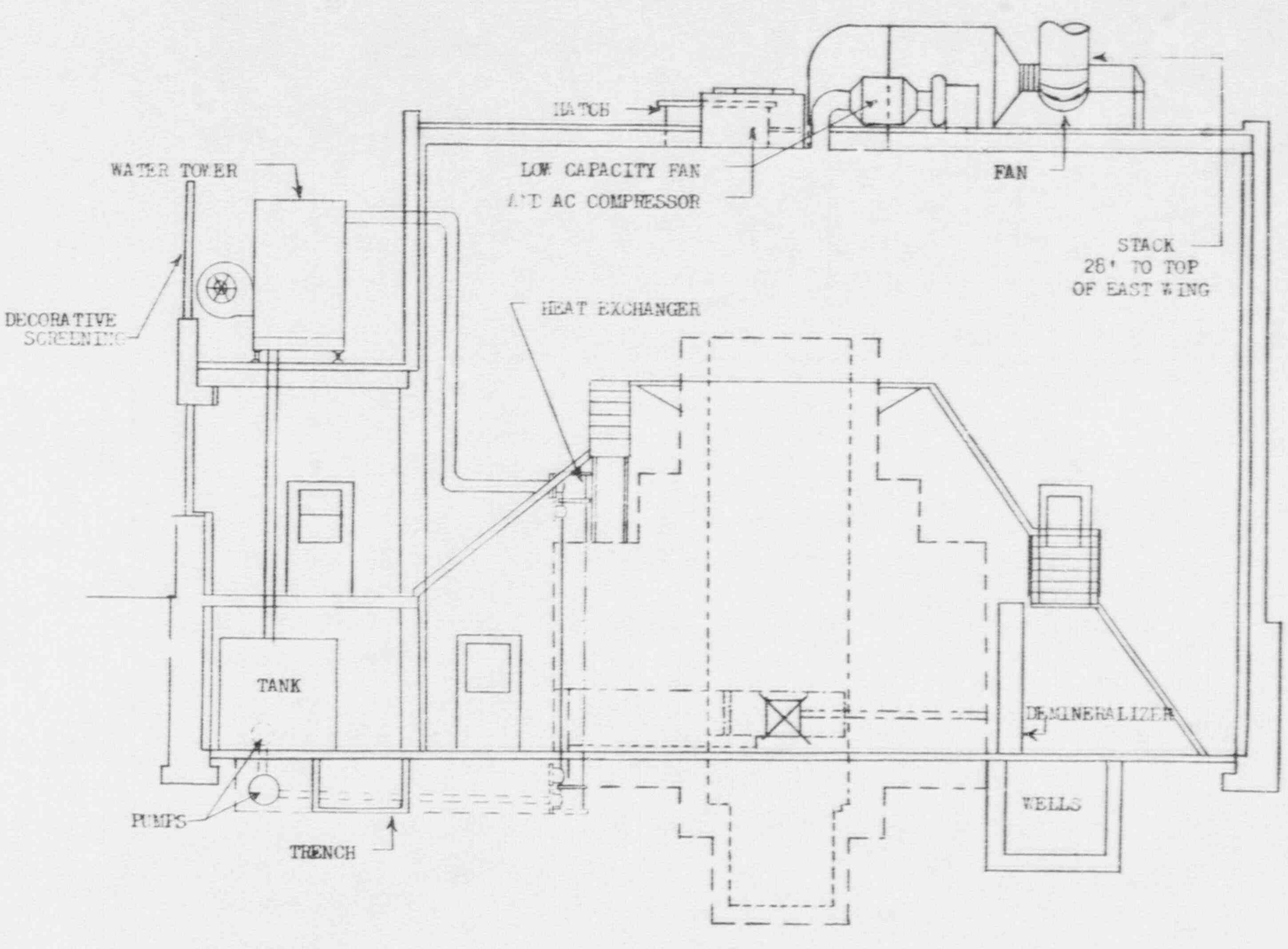


FIGURE 19 -- REACTOR LAB LOOKING WEST

LEGEND SAME AS FIGURE 17

Chapter 4

PROTOTYPE PERFORMANCE CHARACTERISTICS AND REACTOR PARAMETERS

Although the University of Wisconsin Nuclear Reactor will differ in appearance from the TRIGA Mark III prototype, its behavior will be quite similar. Among the differences will be a rectangular array of elements consisting of about 99 fuel-moderator elements instead of the 91 elements in the prototype and four blade type control elements instead of the rod type control elements of the prototype. The fuel element diameter is smaller in the UWNR core in order to maintain the proper metal-to-water ratio in the core.

4.1 INTRODUCTION TO PROTOTYPE TESTS

The TRIGA Mark III prototype, located at General Atomic's San Diego facilities, was first loaded to critical with stainless-steel-clad high-hydride fuel-moderator elements in December of 1961. It has been operated since that time at steady-state power levels up to 1.5 MW and pulses have been performed at peak power levels up to 6500 MW. The core has accumulated some 1500 MW-hr. of operation and more than 5000 transients. Specification for the fuel in the prototype are given in Table 1. The elements to be used in the UWNR facility have a higher U^{235} loading (8.5 weight percent) in the ZrH, as well as the slightly smaller diameter cited above. (See fuel description in Chapter 2.)

Filter Model #'s Changed
83-84 ann report

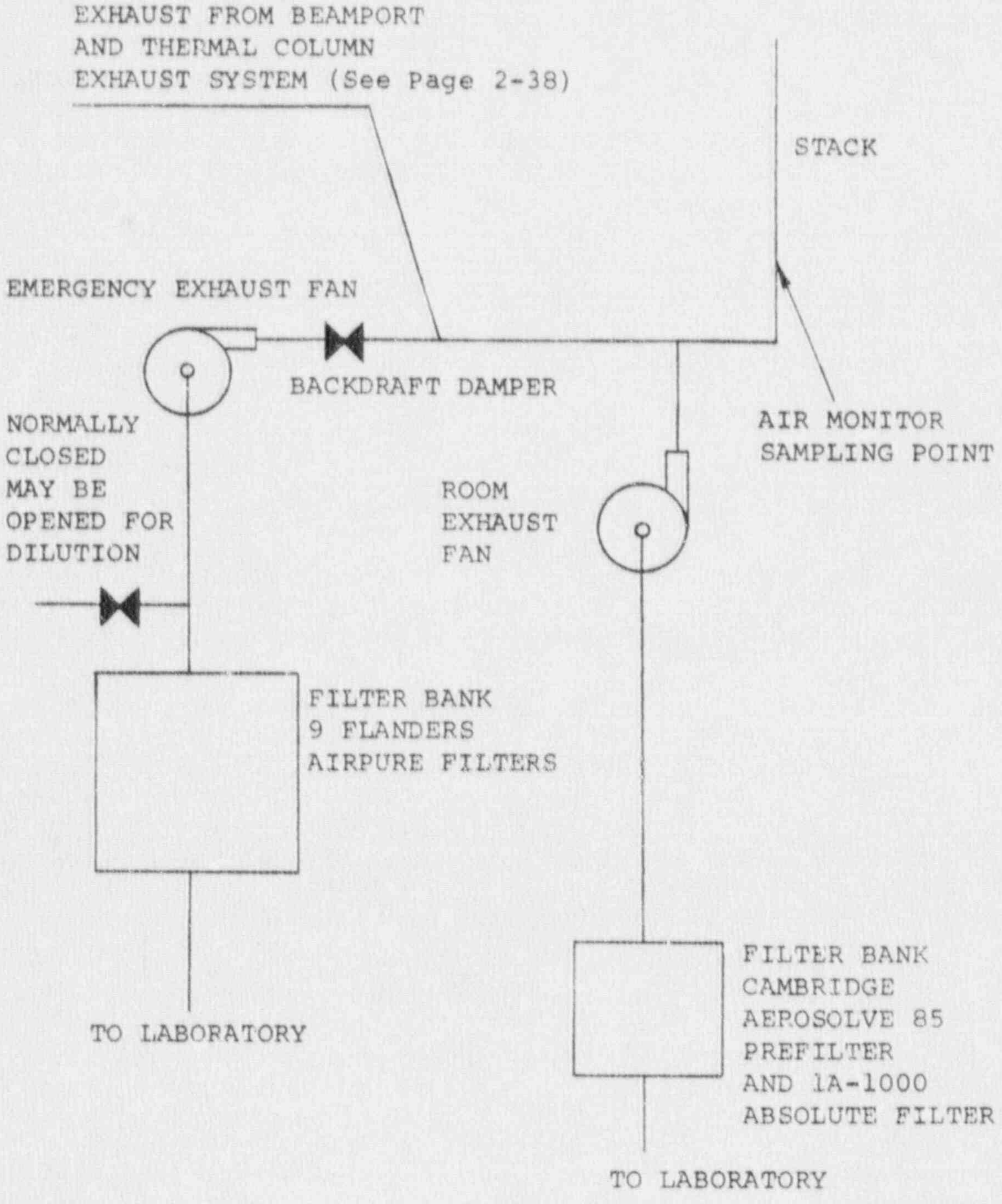


Figure 21 - VENTILATION SYSTEM

4.2 REACTIVITY CHANGES DUE TO REFLECTOR VARIATIONS

Since the University of Wisconsin uses the existing experimental facilities originally used for the U-Al flat-plate type core, the reactivity effects of reflector variations have been measured. Table 2 lists these reactivity worths.

TABLE 2

REACTIVITY CHANGES ASSOCIATED WITH REFLECTOR CHANGES

Condition	Result, % Reactivity
Flooding of all 4 beam ports	+ 0.0005 est.
Flooding of pneumatic tube	+ 0.0002
Pneumatic tube samples	
water	+ 0.0002
cadmium	- 0.0005
Adding 1 graphite reflector on center of one side of core	+ 0.230
Dropping fuel element on top of operating core	+ 0.5
Adding fuel element on side of core	+ 0.77

4.3 FUEL ELEMENT WORTHS

Because of the different core geometry (rectangular parallelepiped rather than cylindrical) and the fact that each bundle contains four fuel-moderator elements like those in the prototype TRIGA Mark III, the data from the prototype are not applicable to the UWNR core. However, based on measured data from the UWNR TRIGA core startup, fuel element worths are indicated in Figure 1.

TABLE 1.

TRIGA MARK III PROTOTYPE

REACTOR FUEL MODERATOR SPECIFICATIONS

Overall length, inches	28.37
Outside diameter. inches	1.47
Fuel outside diameter, inches	1.43
Fuel length, inches	15.0
Fuel composition	$U\_ZrH_{1.7}$
Weight of U^{235}, grams	36.5 (avg.)
Uranium-235 content, wt%	8
Uranium-235 enrichment, %	20
Hydrogen-to-zirconium ratio	1.68 (avg.)
Cladding Material	304 Stainless-steel
Cladding thickness, mils	20

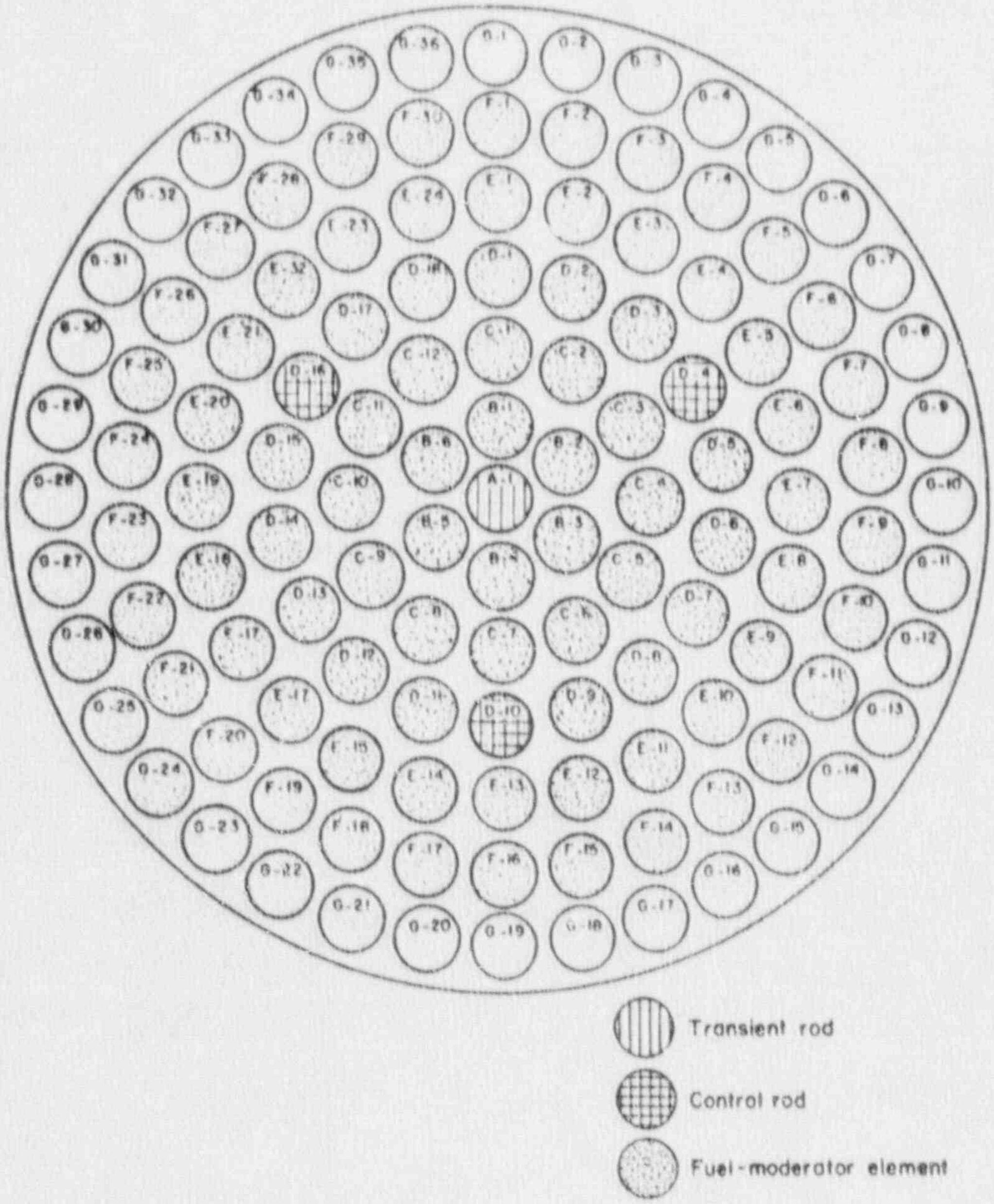
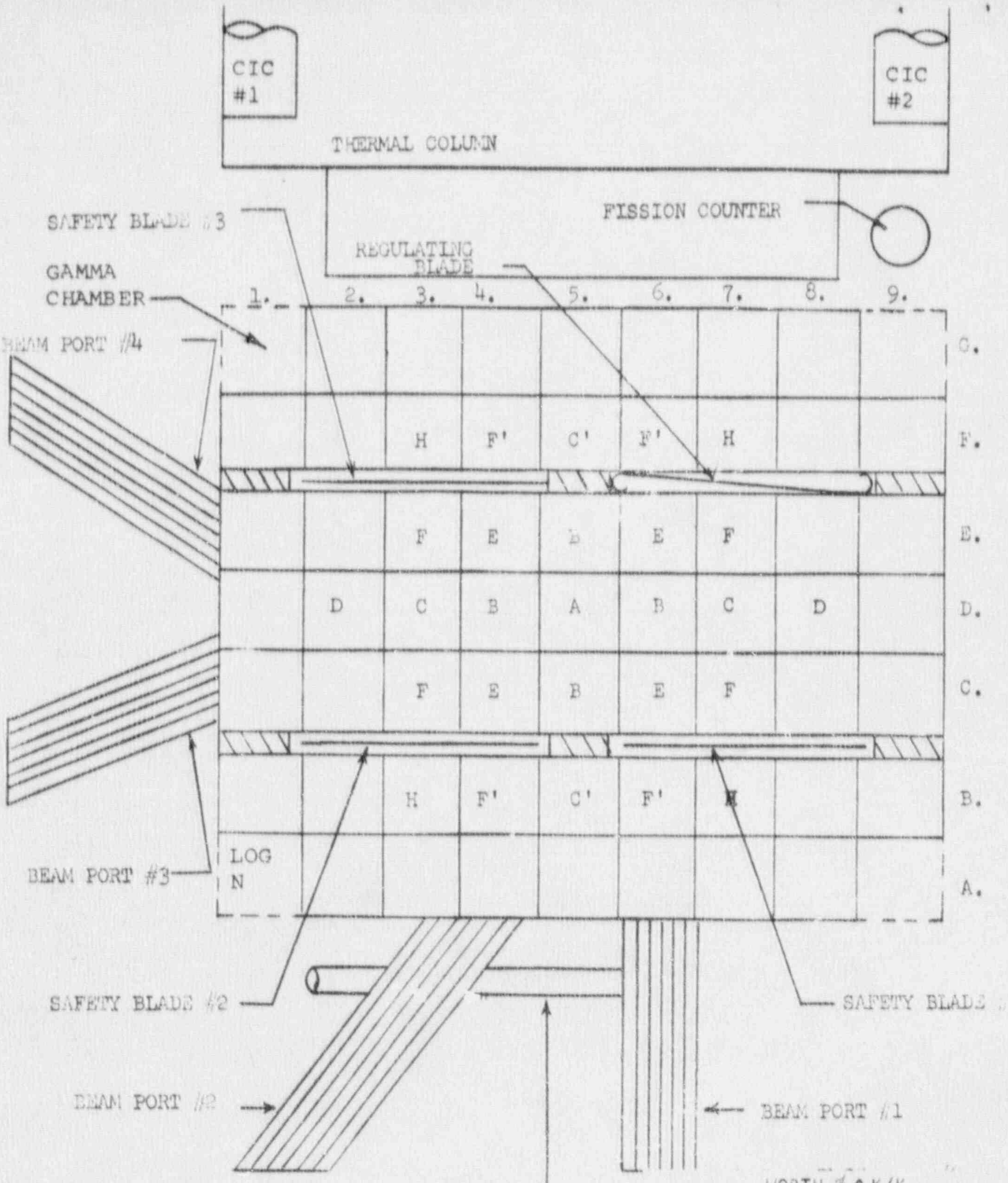


Fig. 2 --- Fuel loading diagram - prototype reactor



WORTH %ΔK/K

POSITION	H_2O REF.	GRAPHITE RE
A	2.60	4.0
B	1.95	3.46
C	1.22	2.18
C'	1.16	1.15
D	0.77	—
E	1.76	2.76
F	1.85	1.55
F'	1.01	1.08
H	0.47	0.63

FIGURE 1

FUEL ELEMENT BUNDLE WORTH

The loosened cladding reduces thermal conductivity and results in a higher fuel temperature. The change in reactivity loss at various powers is most rapid during the first few pulses. After approximately 100 pulses, the reactivity reached a value nearly as large as that value quoted for operation after 2400 pulses. Extrapolation of the reactivity loss at 1 MW versus number of pulses indicates that only after a very large number of pulses (>8,000) will the reactivity loss approach 2.8%.

Figure 3 shows the reactivity loss versus reactor power for the prototype Mark III core after 2,400 pulses.

The power coefficient will be slightly smaller in the UWNR due to the larger number of fuel elements.

4.4.3 Fuel Temperatures

Measurements of fuel temperature were made in various positions in the prototype core. The highest temperatures were measured in the "B ring" -- the central part of the core.

Figure 4 shows the "B ring" temperatures for power levels up to 1.5 MW. The maximum temperature expected at 1.5 MW is about 410°C (738°F) above ambient pool temperature. With the 130°F limit imposed on core inlet cooling water in UWNR, the maximum temperature of the fuel would be ~ 870°F. Since the UWNR core will be larger and will have essentially the same hot-spot factor, the temperature in the central element will probably be about 70°F lower at 1.5 MW. However, both these temperatures are well below the temperature limits for the fuel materials.

4.4 STEADY-STATE PARAMETERS

Measurements have been made of various core parameters in the 91 element prototype core shown in Figure 2. Important differences expected between the prototype core and UWNR core will be pointed out at the conclusion of each section.

4.4.1 Critical Loading

The critical loading for the Mark III prototype was 79 fuel moderator elements. This loading was one incorporating a poisoned follower section on the transient rod. The critical loading contained 2.86 Kg of U-235.

The UWNR core will require a larger loading for criticality due to the reactivity loss caused by the control blade shrouds. It is expected that, for a number of fuel elements greater than 60, the UWNR core will have about 1 per cent less reactivity even with the slightly higher loading of Uranium 235 in each element. It is expected that the critical core will contain about 3.1 Kg of U^{235}.

4.4.2 Power Coefficient

Measurements of the "power coefficient" (loss of reactivity at various power levels) have been made at several intervals during the pulse history of the core (2,500 pulses to date). Some change in values occurs with pulsing. The loss in reactivity at 1 MW was measured to be $\sim 1.96\% \Delta K/K$ for the new core with less than 10 pulses; this value increased somewhat to a value of $\sim 2.27\%$ at 1 MW after 2,400 pulses of 1800 MW amplitude. The changes observed are due to a slight loosening of the clad due to thermal expansion of the fuel meats in going from 20°C pre-pulse temperature to $\sim$ 400°C after-pulse temperature.

Figure 4 -- Measured Fuel Temperature above Ambient vs. Reactor Power - Prototype Reactor
UWNR Data Indicated

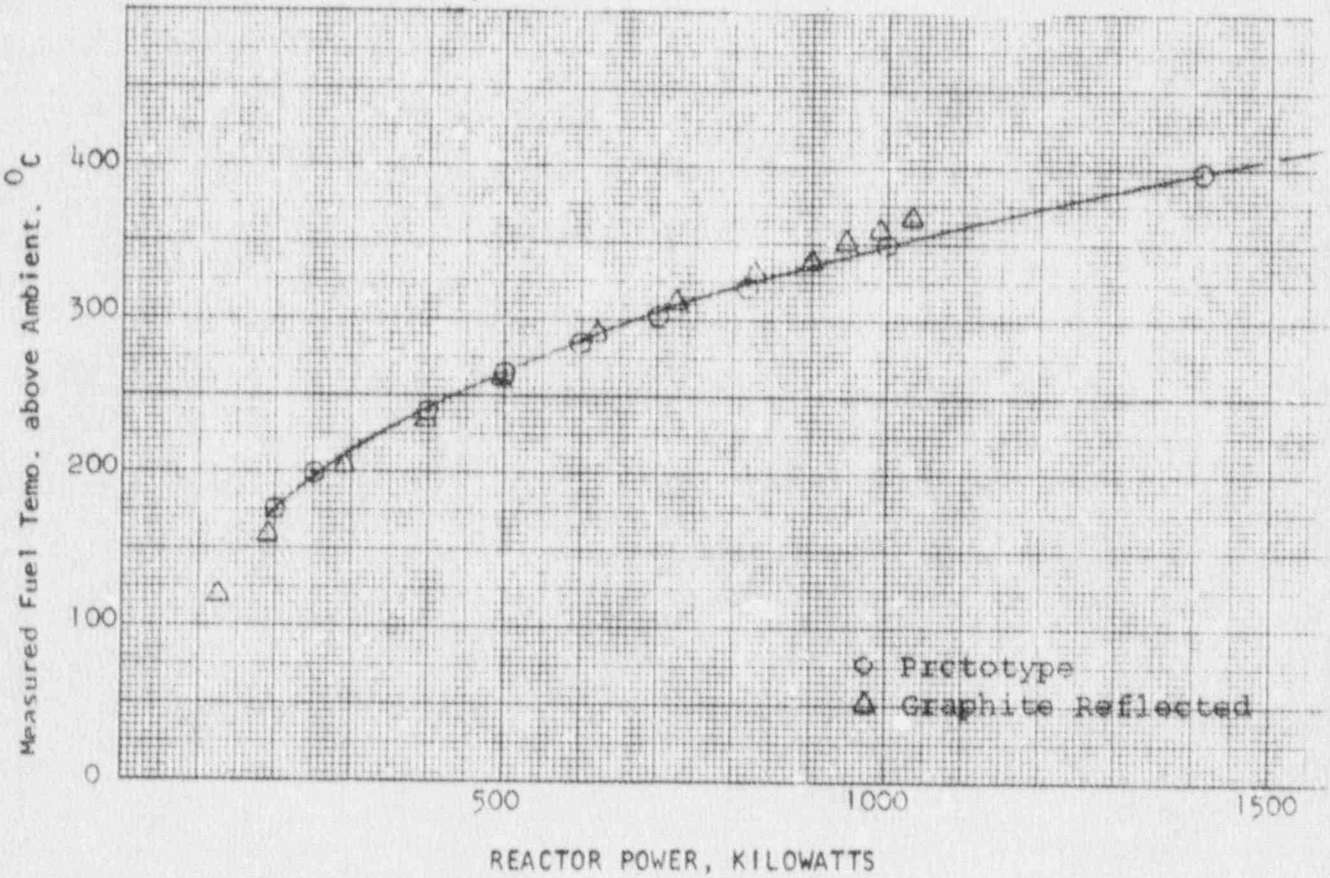


Figure 3 -- Reactivity Loss vs.
Reactor Power - Prototype Reactor
UWNR Data Indicated

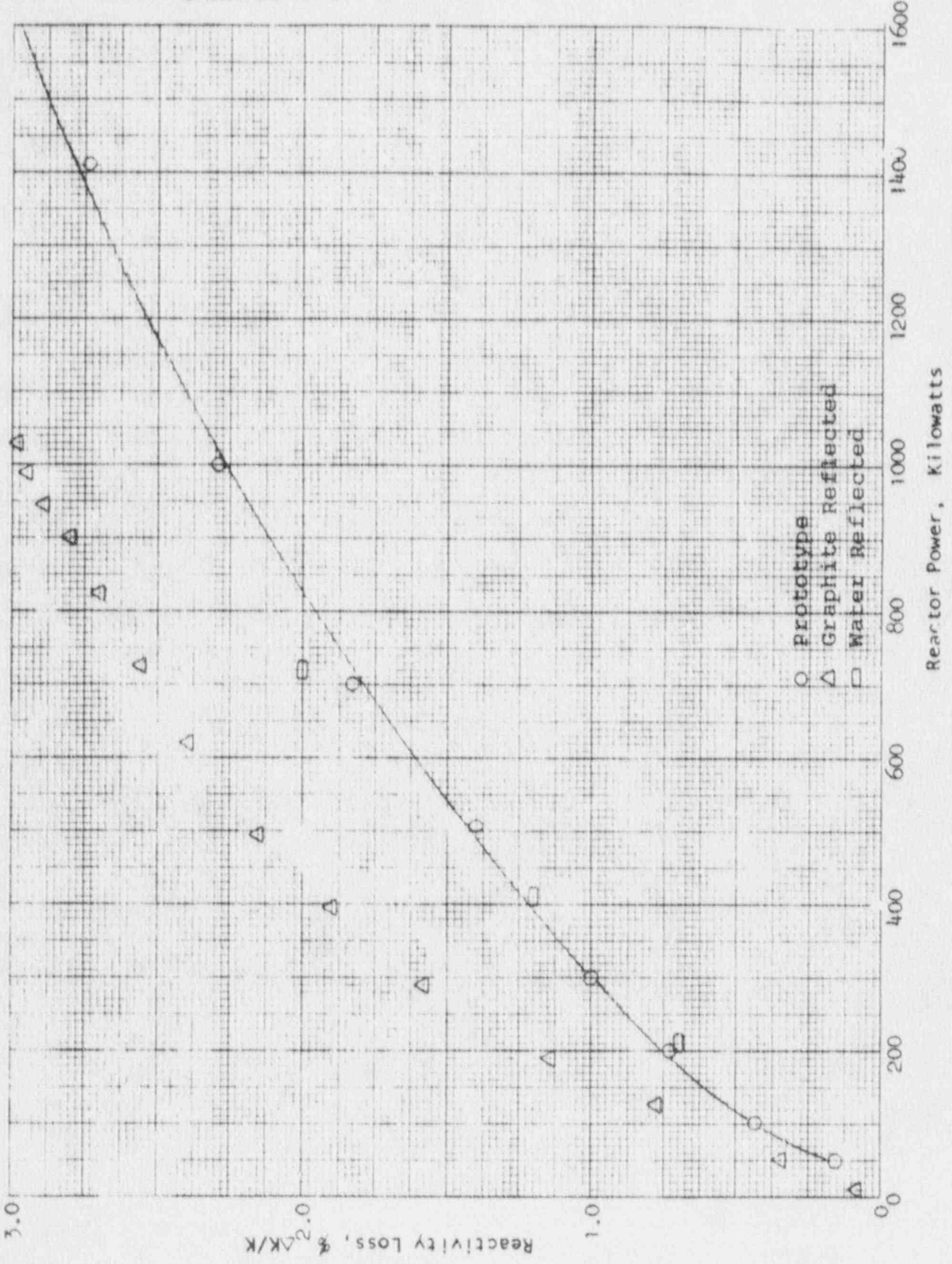
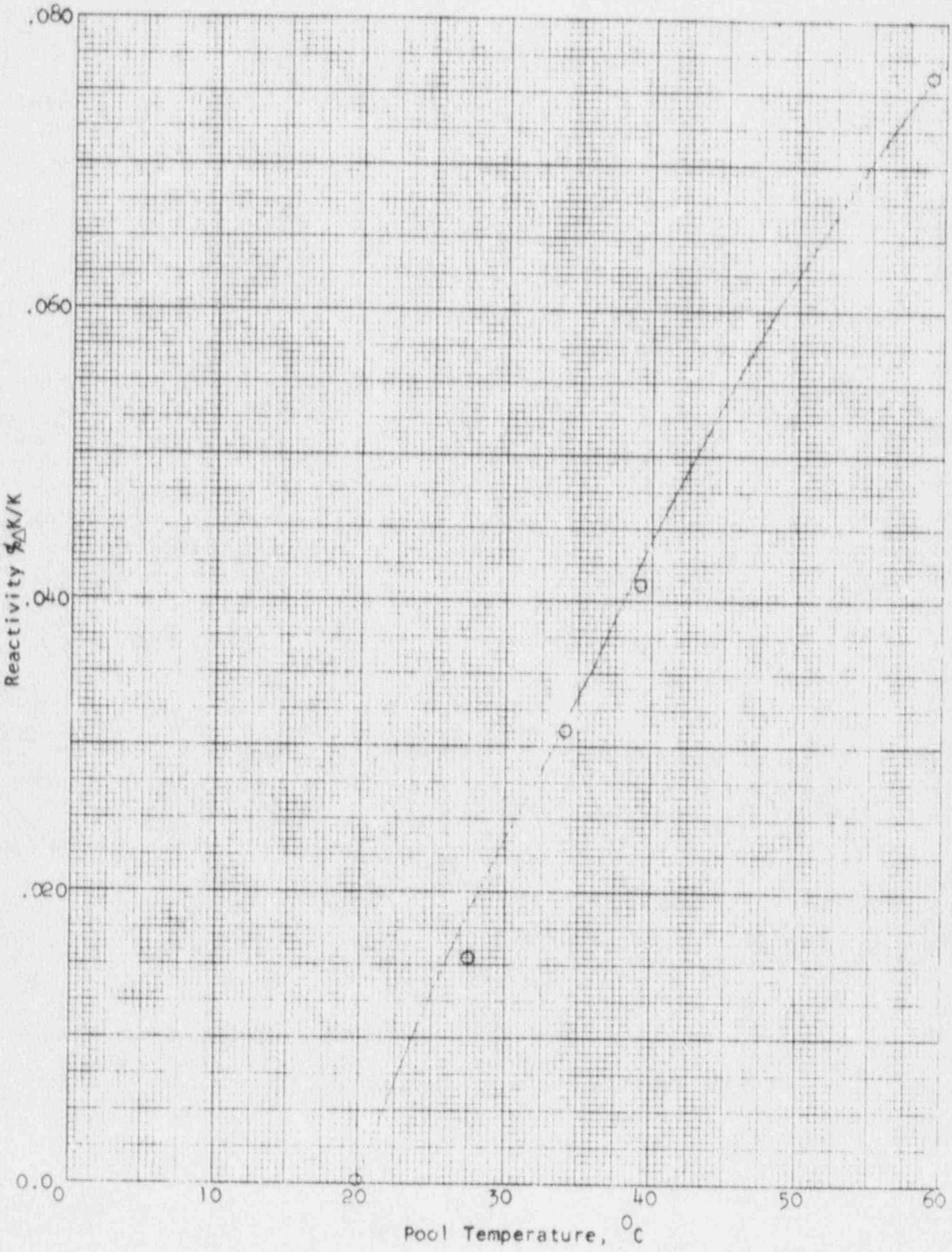


Figure 5 -- Reactivity vs. Pool Temperature - Prototype Reactor



4.4.4 Isothermal Temperature Coefficient (Bath Coefficient)

The coolant water temperature in the prototype was varied over wide ranges to measure the resulting reactivity change. Figure 5 is a plot of the change in available excess reactivity (relative to excess at 20°C) for bath temperatures from 20° to 60°C. The measurements were made at power levels of less than 10 watts. The coefficient is slightly positive with a net gain in available reactivity of 0.077%. The average coefficient, 0.0019%/°C, is small enough that it is essentially negligible for normal operating conditions.

The effect of the water gap left in the shrouds when the control blades are withdrawn is expected to increase the temperature coefficient by about 20% in the UWNR, giving a temperature coefficient estimated at 0.0024%/°C. This value is small enough to be considered negligible for normal operating conditions.

4.5 PULSE PARAMETERS

Measurements were made of the various parameters relating to pulsing operation of the prototype. The most important of these are given below for step insertions of reactivity up to 2.1% $\Delta K/K$.

4.5.1 Period

During pulsing operation the reactor is placed in a super-prompt-critical condition. The asymptotic period is inversely related to the prompt reactivity insertion. Figure 6 shows the results of plotting the reciprocal of the measured period versus the prompt reactivity insertion. Since the period data must be obtained from an oscillographic recording of the reactor power versus time at a portion of the pulse before fuel temperature limiting effects have begun, the accuracy of the measurements is not so good as for other parameters. The scatter of points about a straight line in Figure 6 is due entirely to this difficulty. As can be seen, the minimum period obtained for reactivity insertions of 2.1% $\Delta K/K$ is ~ 3 msec.

4.5.2 Pulse Width

The width of the power pulse is most conveniently described as the time interval between half-power points. Also shown in Figure 6 is a plot of the reciprocal of the measured width versus prompt reactivity insertion, and Figure 7 shows the linear relationship between peak power and $(1/\text{width})^2$.

4.5.3 Peak Power

Figures 8, 9, and 10 show the interrelationship between maximum transient power, pulse widths, and period. When considered together, these plots serve to describe the general features of the Mark III core performance in pulsing modes. For a given core configuration, the peak power, integral power in the prompt burst, and width of the pulse are determined by the reactivity insertion made. It can be seen from the plots that the peak power is controllable over a rather wide range since this parameter is very nearly proportional to $(\Delta K/K - 0.7\%)^2$. Pulse width and integral powers, on the other hand, are approximately linear functions of reactivity insertions above prompt critical so that their range is more limited.

Figure 6 -- Inverse Period and Inverse Width at Half Maximum Power vs. Prompt Reactivity Insertion - Protype Reactor UWNR Data Indicated

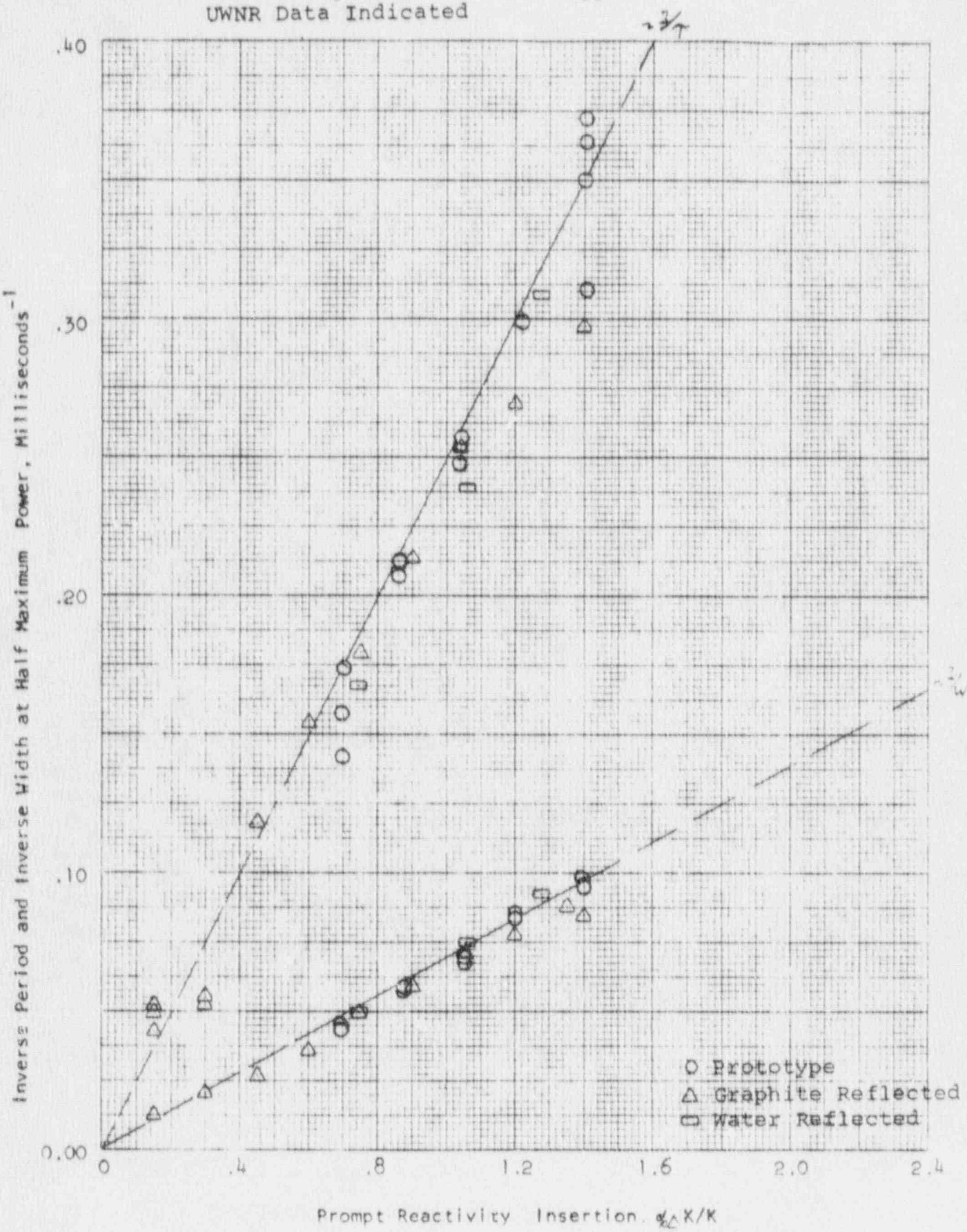


Figure 8 -- Full Width at Half
Peak Power vs. Period
Prototype Reactor
DWNR Data Indicated

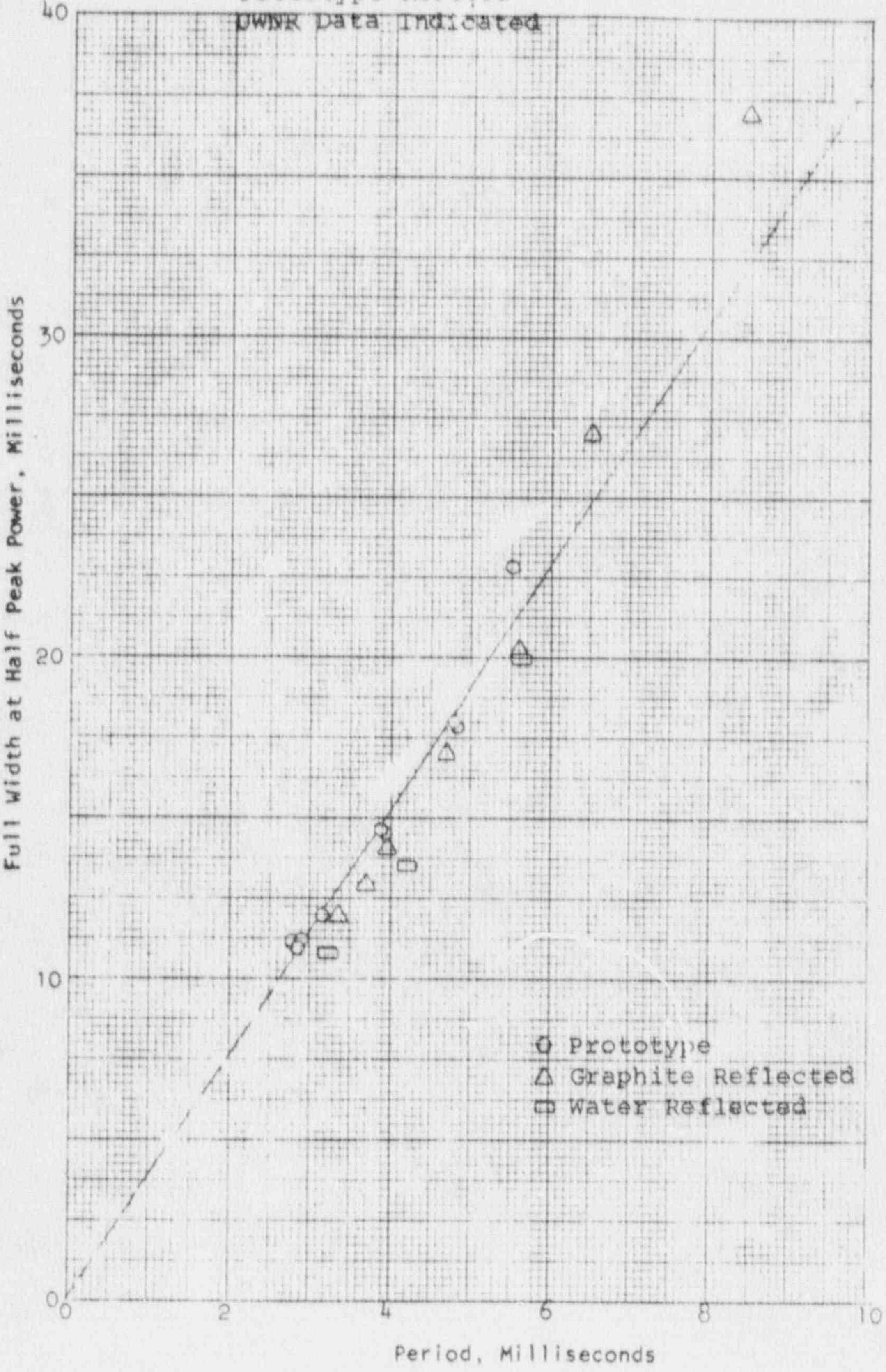


Figure 7 -- Peak Power vs. Inverse Half-Width Squared - Prototype Reactor UWNR Data Indicated

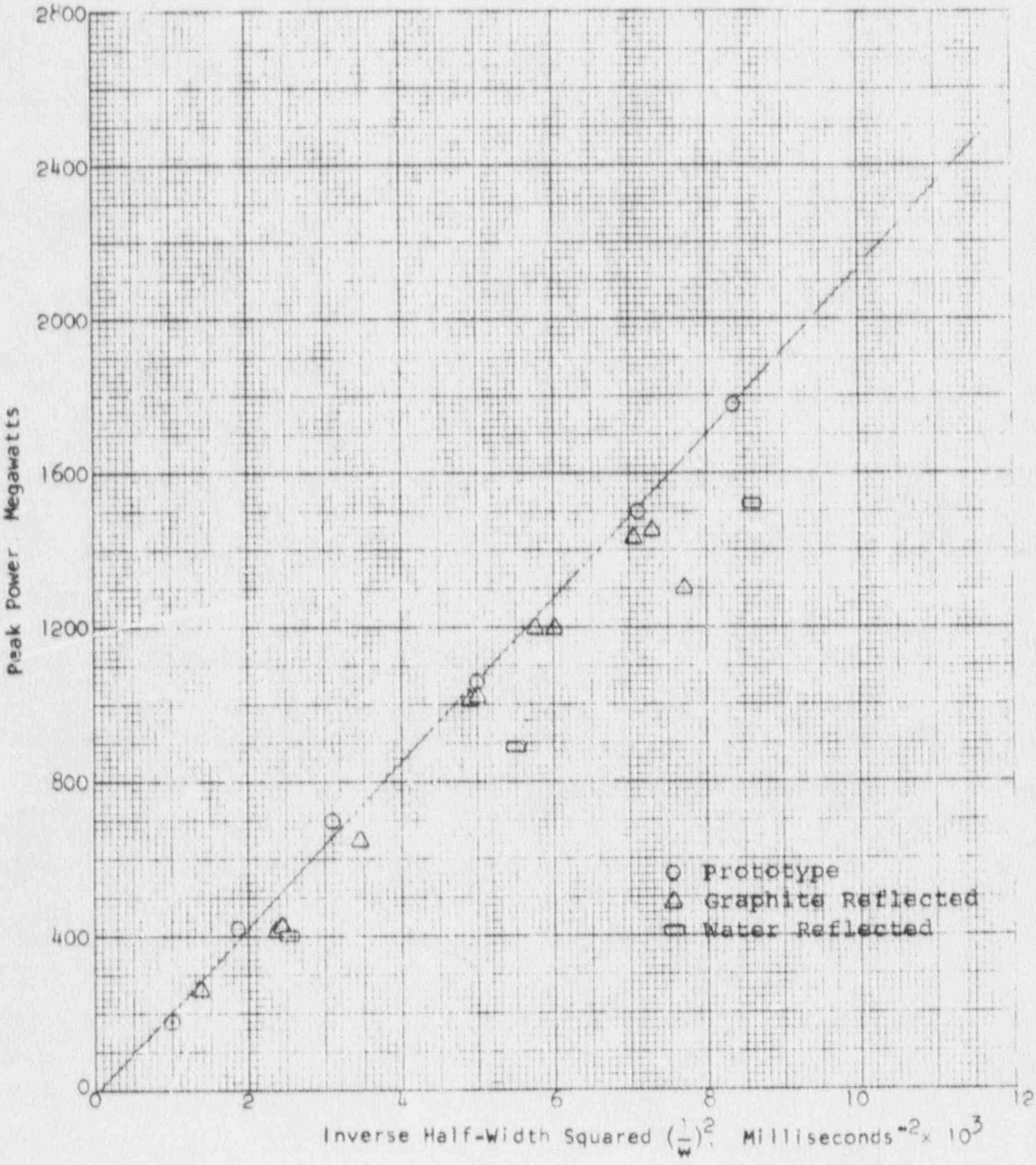


Figure 10 -- Peak Power vs.
Reactor Period -
Prototype Reactor
UWNR Data Indicated

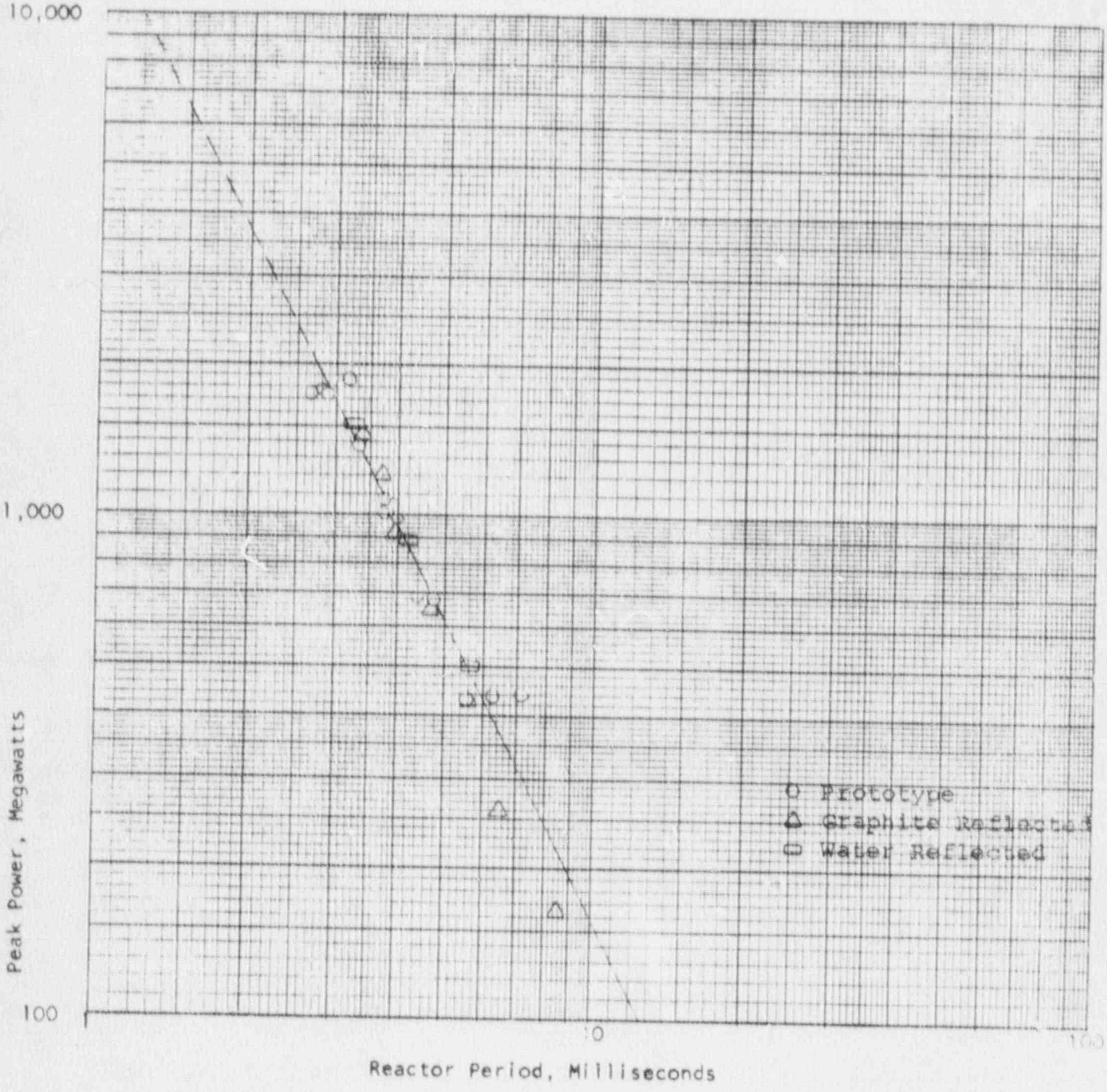
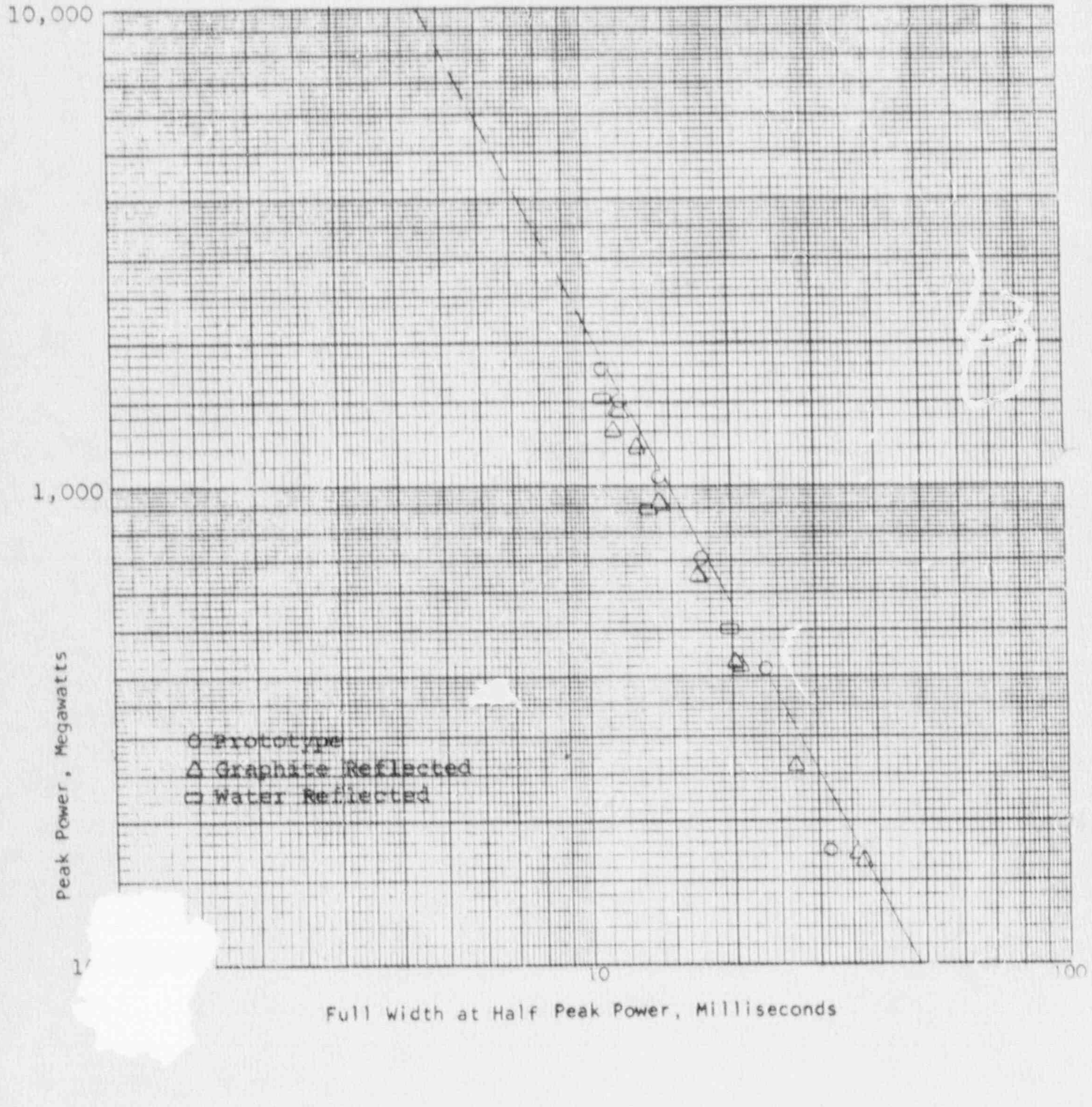


Figure 9-- Peak Power vs.
Full Width at Half Peak Power -
Prototype Reactor
UWNR Data Indicated



The calculational accuracy was checked by analysis of cores with known values of $^{\Delta}K_{eff}$ and power density. Results on calculation of mixed cores and FLIP cores were found to be consistent with similar calculations performed elsewhere*.

FLIP fuel elements will not be mixed with standard elements in the same fuel bundle at Wisconsin. Thus, the smallest increment of FLIP fuel addition possible will be three FLIP elements (in a bundle containing the transient rod guide tube). Placing such a bundle in the center of a 5 x 5 array of standard TRIGA fuel leads to the highest value of power peaking possible, with a resultant power generation of 31.2 KW in an element. Although no operation with a core of this type is anticipated or desired, other TRIGA reactors have operated with power generation rates at least as high as 32 KW per element.

Calculations were performed for cores with 1, 2, 5, 9, 15 and 25 FLIP bundles in central contiguous regions of the core. All calculations were for a 5 x 5 array of fuel bundles with the transient rod guide tubes in the fuel bundle at grid position D5.

Addition of less than five FLIP fuel bundles (24 FLIP elements) is not considered useful for a full power operating core, since it would not provide sufficient additional reactivity to compensate for burnup in the standard elements.

Table 3 indicates data for the 5, 9, 15, and 25 bundle cores, showing maximum power density, worth of replacing FLIP bundles with water, and resultant peaking of power in adjacent fuel if a three inch square water gap is introduced in the most important positions in the core.

Reference to Table 1 and Figure 1 points out that power generation in any individual element is well below 23 KW in all compact FLIP fuel arrangements. Further, the presence of a 3-inch square water gap in the FLIP fuel region will result in power generation rates below 23 KW/element in most of the cores.

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*GA-9064

4.6 FLIP FUEL PERFORMANCE CHARACTERISTICS*

The higher enrichment of FLIP fuel coupled with erbium poisoning causes changes in operating characteristics relative to standard fuel. The most marked changes are a reduction of prompt neutron cycle time to about 10×10^{-6} seconds at beginning of core life (20×10^{-6} at end of core life) and a temperature coefficient that is strongly temperature dependent. (Figure 3-16, page 3-26 of reference).

In addition, the harder spectrum in a FLIP core leads to power peaking in regions near water traps. This leads, in a compact core, to a peaking factor within a FLIP element of 1.43. If a large water-filled flux trap is located adjacent to an element, the peaking factor in the element can increase to 2.65 peak/average within the cell.

Thermal and hydraulic parameters of FLIP fuel remain the same as standard fuel.

4.7 MIXED FLIP-STANDARD FUEL CORE

The longer operating lifetime for FLIP fuel was the major reason for selecting this fuel type for refuelling the University of Wisconsin Nuclear Reactor. Although the intent is to eventually convert the entire core to FLIP fuel, economic considerations require replacing only a portion of the fuel at the present time.

Various combinations of FLIP and standard fuel were investigated in order to study power PEAKING ~~probing~~ and reactivity values for mixed cores. Calculations were performed with a two-dimensional diffusion theory code (Exterminator 2). Standard seven group cross sections obtained from Gulf-General Atomic were used in the calculations.

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*This information is extracted from GA-9064, Safety Analysis Report for the Torrey Pines Triga Mark III Reactor, Section 3.2.

TABLE 3

POWER PEAKING AND FUEL BUNDLE WORTH IN MIXED FLIP-STANDARD TRIGA CORES

Core Arrangement*	Power in Maximum Element-KW	Reactivity Worth of Removing Bundle in Indicated Position %ρ
5 FLIP + 20 Standard		
(FLIP in Positions A & B)	21.4	
Replace FLIP in E5 with H_2O	28.3	$\overline{2.83}$
9 FLIP + 16 Standard		
FLIP in Positions A, B, and E	18.1	
Replace FLIP in D5 with H_2O	20.0	$\overline{0.93}$
Replace FLIP in C5 or E5 with H_2O	25.9	1.69
Replace FLIP in D4 or D6 with H_2O	23.2	0.98
Replace FLIP in E4, C4, C6, or E6	22.3	1.49
15 FLIP + 10 Standard		
FLIP in Positions A, B, C, E & F	17.2	
Replace FLIP in D5 with H_2O	19.0	$\overline{0.87}$
Replace FLIP in C5 or E5 with H_2O	24.6	1.65
FULL FLIP - 25 Bundles	15.5	
Replace FLIP in D5 with H_2O	17.2	$\overline{0.79}$
Replace FLIP in C6 or D6 with H_2O	20.0	0.51
Replace FLIP in C5 or E5 with H_2O	22.1	1.42

*FLIP element locations keyed to Figure 1 Positions.

The initial operational mixed core will contain nine FLIP fuel bundles (35 elements), and the calculations indicate that flux traps cannot be permitted for full power operation in this arrangement for locations C5, E5, D4, and D6. The combination of proximity to control blade shrouds and the transient rod guide tube causes the greatest power peaking in any of these cores.

It is also apparent from the Table that the reactivity worth of an individual FLIP bundle is lower than that of a standard fuel bundle, even in mixed cores.

In order to retain as much flexibility as possible the proposed Technical Specifications on core arrangements will require calculation of power generation in each fuel element for mixed cores or full FLIP cores which contain water filled spaces larger than one fuel element diameter in order to insure that the limits of power density in fuel elements will not be exceeded.

4.8 MIXED CORE ARRANGEMENT FOR INITIAL OPERATION

In order to obtain interim Technical Specifications which will allow experimental verification of the calculated values presented above, a specific initial core loading will be used. This loading is the nine (9) FLIP plus sixteen (16) standard bundle core referenced above. It contains FLIP bundles in Figure 1 reference positions A, B, and E plus sixteen (16) standard TRIGA fuel bundles to fill out a 5 x 5 array. Since the initial Technical Specifications will not allow water gaps within the FLIP region, the maximum power density in any fuel element at 1,000 kW will be 18.1 kW. This is approximately 11% higher than in an all-standard fuel core. Table 4 gives further power density and fuel temperature information for 1,000 kW steady state operation of this core.

The fuel temperatures are calculated from experimental measurements of fuel temperature made at Wisconsin, as measured in an instrumented element.

Chapter 5

ORGANIZATION AND PROCEDURES

5.1 OPERATING ORGANIZATION

Figure 1 is a chart indicating operating organization. The position responsibilities are summarized as follows:

5.1.1 University Radiation Safety Committee

1. To exercise its prerogatives (as a campus-wide committee appointed by the Chancellor of the University of Wisconsin-Madison Campus to review all activities on campus which involve the use of radiation) in reviewing all activities related to the Reactor Laboratory.

2. To advise the Reactor Director of all studies and/or actions taken with regard to the Reactor Laboratory.

3. To overrule the Reactor Director where necessary in carrying out its function.

4. To supply health physics services to the University.

5.1.2 University Health Physics Office

1. To assist the University Radiation Safety Committee by conducting inspections, making recommendations, maintaining records, and establishing procedures for emergency operations, waste disposal, etc.

2. To provide similar inspections and service functions to the Reactor Safety Committee.

5.1.3 Reactor Director

1. To approve all policy decisions and all basic regulations, basic instructions, and basic procedures governing the use and operation of the reactor and related facilities.

2. To designate the Reactor Supervisor and other Senior Operators.

TABLE 4

POWER DENSITY AND FUEL TEMPERATURES
INITIAL MIXED CORE ARRANGEMENT - 35 FLIP ELEMENTS

Fuel Element Description	Power in Element	Fuel Temperature
Average in core	10.10 kW	282°C
Highest Power Density-FLIP	18.11 kW	372°C
Lowest Power Density-FLIP	13.95 kW	325°C
Highest Power Density Standard	9.34 kW	272°C
Lowest Power Density Standard	4.54 kW	201°C

3. To take cognizance of all recommendations and actions by the University Radiation Safety Committee (which relate to the reactor facility) and the Reactor Safety Committee.

4. To appoint qualified members to the Reactor Safety Committee as necessary.

5.1.4 Reactor Safety Committee

1. To evaluate and approve (or disapprove) all proposed operations and procedures involving the reactor. This shall include a review of all new operating procedures, new experiments or experimental plans, and changes to the reactor structural, electrical or component design.

2. To advise the Reactor Director of all studies and actions undertaken by the Committee.

3. To review operation and conduct inspections to assure proper adherence to approved procedures and practices.

4. Disapproval by any member of the Committee of a proposal shall kill the proposal unless the proposal is one establishing safety restrictions. In the latter case, a majority vote is necessary to kill the proposal.

5. The Committee will meet at least twice per year. Business requiring more frequent meetings is generally handled by (a) telephone polling of members by the Chairman for specific approval of minor changes and (b) Subcommittee action for approval of minor changes in written procedures and for inspections in behalf of the main committee.

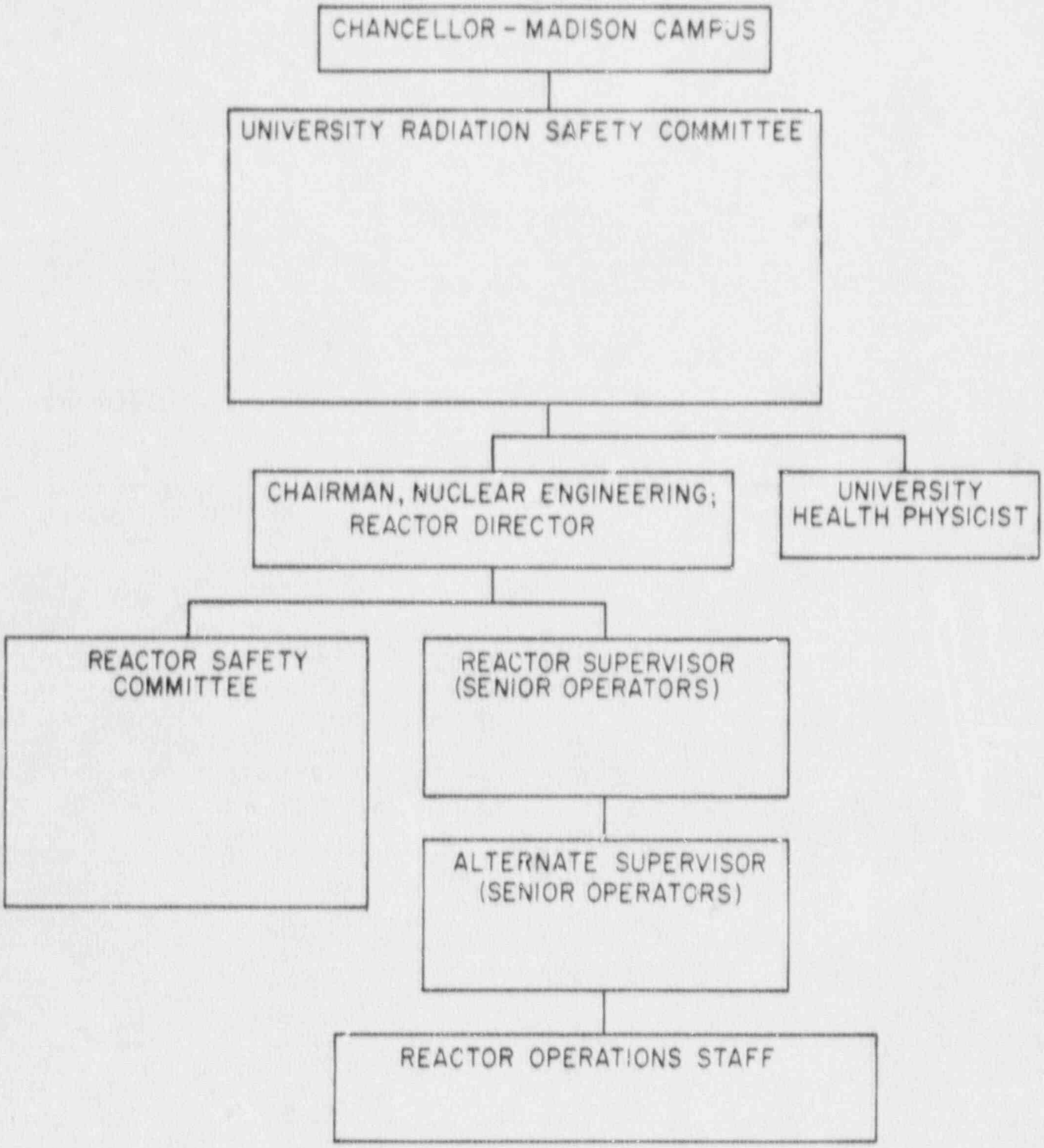
CHANCELLOR - MADISON CAMPUS
UNIVERSITY RADIATION SAFETY COMMITTEE
CHAIRMAN, NUCLEAR ENGINEERING; REACTOR DIRECTOR
UNIVERSITY HEALTH PHYSICIST
REACTOR SAFETY COMMITTEE
REACTOR SUPERVISOR (SENIOR OPERATORS)
ALTERNATE SUPERVISOR (SENIOR OPERATORS)
REACTOR OPERATIONS STAFF

FIGURE 1

9. To advise and prepare information for the committees concerned with the Reactor Laboratory, and to present such information to the committees.

5.1.6 <u>Senior Operators</u> (alternate Supervisors)

1. To accept responsibility for safe and efficient operation of the Reactor Laboratory when designated by the Reactor Supervisor.

2. To maintain a Senior Operator's License.

5.1.7 <u>Reactor Operators</u>

1. To hold a Reactor Operator's License.

2. To conform to all rules and regulations for operation of the reactor.

3. A reactor operator will be present at the control console at all times when the reactor is in operation.

4. To monitor laboratory activities from a health-physics standpoint.

5.1.5 Reactor Supervisor

1. To initiate and enforce policies, administrative rules, regulations, and operating procedures relating to the Reactor Laboratory, subject to the appropriate approvals of the Reactor Safety Committee, the University Radiation Safety Committee, and the Reactor Director.

2. To ensure that all activities within the Reactor Laboratory are in accordance with prior approvals from the appropriate committees or from the Reactor Director.

3. The Reactor Supervisor shall have authority to authorize experiments and/or procedures which have been approved by the Reactor Safety Committee. He will prepare specific detailed procedures based on the general procedures approved by the Committee.

4. To see that all proper records are kept.

5. To maintain a Senior Operator's License.

6. To appoint Reactor Operators.

7. The Reactor Supervisor or another Senior Operator shall be in charge of the Reactor Laboratory at all times (although not necessarily physically present). The individual in charge, if physically present, shall be responsible for prompt execution of emergency procedures. The Reactor Supervisor or another Senior Operator will be present at the facility during fuel manipulation, reactor start-up and approach to power, and recovery from unscheduled scrams and shut-downs. He shall be available on call at other times during reactor operation.

8. To be responsible for safety in the Reactor Laboratory, including responsibility for health physics matters.

5.3 OPERATIONAL PROCEDURES

Step-by-step written procedures are used in all cases to which such procedures are applicable.

Check-list procedures are used extensively for pre-startup checks. Maintenance procedures (as included in an instruction manual provided by the reactor manufacturer) are used for routine maintenance. Emergency procedures are also in written form.

All standing written procedures must be approved by the Reactor Safety Committee. Revisions of existing standing procedures must also be approved by these committees.

The more important aspects of the standard procedures for this reactor are indicated below.

5.3.1 Initial Test Program

The addition of FLIP fuel to the University of Wisconsin Nuclear Reactor has been planned as a normal refuelling operation. For that reason Section 5.3.3 describes the procedure to be used. The FLIP fuel will comprise the central portion of the final core. It is expected that the mixed core will contain nine FLIP bundles (35 elements) and sixteen standard TRIGA bundles (64 elements).

A test and acceptance program will be performed to include the following:

(a) Verification that core excess reactivity is within proper limits; verification of control element worths and shutdown margins.

(b) General verification of the characteristics indicated in Chapter 4;

(c) Stepwise increases in power level to licensed power level with checks of operating temperatures. The initial step will be to 100 KW at which point a power level calibration will be performed. Subsequent steps will be

5.2 OPERATING STANDARDS

The basic premise of all proposed operating standards is the safety of the reactor, its operating personnel, and the immediate surroundings. The limitations described below will be imposed upon operation of the reactor.

5.2.1 Limitation of Experiment Reactivity Worth

The reactivity worth of any individual experiment shall not exceed 2.1% ΔK/K.

5.2.2 Operations Which Might Involve Changes in Core Reactivity Conducted When the Reactor is Shut Down

1. All such operations are conducted under direct and personal supervision of qualified Senior Operators.
2. Nuclear instrumentation will be used if reactivity might be increased.
3. Loading of fuel into other than previously verified configurations will be done only under conditions of cocked safety blades and after a check-out of instrumentation.
4. Such operations will be conducted in accordance with special written procedures if significant changes are made.

5.2.3 Shut-Down Margin

Operation will not be permitted with a reactor core that does not provide a shutdown margin greater than 0.1% ΔK/K (relative to cold clean condition) with the highest worth control element fully withdrawn from the core.

power level on a preset period and then maintains the power at the scheduled level.

For square wave operation the reactor is taken to a stable power level between 1 and 1,000 watts. The picoammeter range switches are changed to the full power range and the servo power schedule is set to the desired power level. The mode switch is set to the square wave position. Then a preadjusted step reactivity change is made. When the power reaches the scheduled power, the servo, with manual augmentation by the operator, maintains power level as the fuel heats up.

For pulsing operation the reactor is taken to a stable power level of less than 1,000 watts in the manual steady state mode (the automatic power level channel can be used to level the power level at the desired steady state point). The mode switch is then changed to the pulse mode, the recorder for the pulsing chamber readout and fuel temperature readout is started, and the pulse is initiated by the transient rod. The transient rod is automatically reinserted after a preset time delay.

5.3.3 Refueling

(a) Unloading Old Core

The fuel element grapple is used to move fuel elements into the storage racks. At no time during this operation will the fuel elements come closer to the pool surface than 15 feet.

If spent fuel is to be shipped to a reprocessing facility, the fuel will be transferred to a shipping cask. The shipping cask will be leased. Due to the limited overhead clearance in the Reactor Laboratory, a transfer cask is used to transfer fuel bundles, one at a time, from the pool to the shipping cask which will be placed on the floor of the Laboratory. This transfer cask provides sufficient shielding that the loading of shipping casks can be conducted without exceeding CFR Part 20 limits on radiation exposure. The fuel will then be to 200 KW, then 200 KW or less increases to licensed power level. Additional power level calibrations will be performed.

(d) Calibration of pulsing operation of the reactor. The pulse characteristics will be measured, and no pulse exceeding the limits in the technical specifications shall be programmed. At least two pulses of each reactivity input will be performed to assure repeatability of data.

The reactor will then be prepared for routine operation.

5.3.2 Routine Start-Up Operation

After completion of the above checks and measurements, the reactor will be started up for power operation with the aid of a check-sheet to insure proper setting and functioning of all facilities and instruments. If any component does not function properly, the reactor will not be started up until the condition is corrected. All applicable scram circuits are checked before each startup.

The power for operating controls is supplied through a key switch. This switch is normally locked "off" and the keys are in the possession of authorized operating personnel.

For manual steady state mode operation, the safety blades are withdrawn one at a time, and the reactor becomes critical. The startup counter is moved out as the count rate approaches the saturation level of the detector until it is in the full out position. Control blade withdrawal is halted while the detector is moving. The reactor is leveled off at the desired power level.

Automatic operation in the steady state mode differs only in that:

(a) The power schedule must be set for the desired power level;

(b) The mode switch is switched to automatic and the servo system attains the scheduled

Curves of inverse multiplication are drawn for each of these conditions. The "cocked" configuration is changed so that it can be determined that each blade has a reasonable reactivity worth. Data from the 1/M plot for all blades out are used to predict the critical loading. The loading increment is based on the predicted critical loading. If the data are inconclusive, only one bundle is loaded. If three data points fall on a straight line, one-half the number of bundles predicted to make the core critical may be loaded (or one bundle if less than one bundle is indicated). All subsequent loading steps are done with the cocked blade configuration. The data from the blades one-half out and full-in configurations are used to verify shutdown margin.

The loading is continued until the required core excess reactivity is reached and shutdown margin is verified.

shipped, subject to appropriate approvals, to the proper processing facility. The shipment will be conducted by an outside contractor experienced in shipping irradiated fuel elements.

(b) Loading New Core

Refueling is accomplished by a crew of three. One operator is on duty at the console. The loader does the actual fuel manipulations with a fuel grapple designed to lock on the fuel bundle until intentionally released. A Senior Operator supervises the operation and insures that proper records are kept. The supervisor selects the proper loading increments based on the criteria listed below.

A loading chart is prepared before the operation of loading begins. The loading plan indicates the order in which grid spaces will be filled. The loading sequence is selected to keep the core balanced within the grid and as well centered on the control blades as possible. The transient control rod will be installed in the central bundle as soon as it is loaded.

A pre-startup check is run on all instruments. The control element drives and scram circuits are checked. An additional fission counter or BF_3 is placed close to the active lattice positions but in a place that will not be filled with fuel at a full core loading. The source is placed in its holder in a position that will be across the core from the detectors.

Instrument readings are taken with the blades "in" and "out". Two of the safety blades are cocked at 8 inches and the other is run full in. Fuel is loaded with the blades in this "cocked" position. When fuel is loaded, instrument readings are taken at the cocked position, with all blades at 8 inches, with all blades full out, and with all blades full in.

FUELED EXPERIMENTS

Fueled experiments will be so controlled that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies and the maximum Sr^{90} inventory is not greater than 5 millicuries.

CONVERTER PLATES

~~The University has in its possession a removable-plate fuel element. This element has never been used in cores operated at power levels above a few watts and thus has essentially zero fission product inventory.~~

~~All of the element except the fueled plates has been discarded and the plates will be used for converters in thermal column or beam port experiments which require small neutron fluxes with approximately a fission spectrum.~~

~~Each of the ten flat fuel plates contains a nominal 13.5 grams of U^{235}(93 enrichment) in a 21.5 wt-% alloy with aluminum. The meat is 0.039 inch thick, and it is clad with 0.030 inch aluminum in the so-called "picture-frame" technique. The overall fuel plate dimensions are 0.099 inch thick, 2.79 inches wide, and 29 inches long.~~

~~Since these plates are standard General Electric flat-plate fuel element plates it is unlikely that leaks will occur. Exposure will be limited so that each plate remains within the criteria for fueled experiments.~~

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5.4 EXPERIMENTS

Present plans call for use of the reactor in performance of the following experiments:

1. Reactor Start-up and Operation;
2. Radiation Survey of the Reactor and Surroundings;
3. Control and Regulating Blade Calibration;
4. Measurement of Reactor Power and Calibration of Reactor Instruments;
5. Measurement of Shutdown Power Level;
6. Measurement of Reactor Period;
7. Measurement of Temperature Coefficient of Reactivity;
8. Measurement of Void Coefficient of Reactivity;
9. Experiments Involving the Danger Coefficient Method;
10. Experiments of Measure the Disadvantage Factor;
11. Studies of Reactor Buckling and $\Delta K/K$.
12. Critical Mass Experiments;
13. Measurement of Thermal-Neutron Cross Sections;
14. Delayed Neutron Emission;
15. Activation Analysis;
16. Experiments Utilizing Pile Oscillator Techniques;
17. Flux Distributions in Reactor and Effect of Absorbers on Flux Patterns;
18. Shielding Experiments;
19. Experiments on the Production of Radioisotopes;
20. Neutron Diffractometer Measurements.
21. Neutron Radiography

The above represents the experiments planned at present, but it is anticipated that further experiments (both for training and research) will be added.

Chapter 6

SAFEGUARDS EVALUATION

6.1 GENERAL

For initial operation of the reactor at 1 MW, the reactivity will be about 5% Δ K/K above clean cold critical. The reactivity is allocated approximately as indicated below:

Power Coefficient	1.75% Δ K/K
Xenon Poisoning	1.75% Δ K/K
Control & Flux Balancing	1.40% Δ K/K
	4.90% Δ K/K

In addition, the maximum reactivity for an experiment will be limited to 2.1% Δ K/K. All in-pool experiments will be constrained at least as well as the fuel bundles. In-core experiments will be designed so they are constrained by the grid or grid box structure, although part of their support may be from other pool structure.

Should an experiment having the maximum reactivity worth allowed for all experiments (2.1% Δ K/K) fail, the resulting step change in reactivity worth would be less than that deliberately inserted during pulsing operation.

Should the beam ports and pneumatic tube flood while the reactor is operating at full power, a step reactivity addition of 0.07% Δ K/K would result. This reactivity change is so small that it would not cause any disruption of normal operation.

If a gross departure from procedure were to be made and a fuel element bundle were added to the outside of the core while operating at full power, the maximum reactivity that would result would be about 0.7% Δ K/K. This is a reactivity smaller than that routinely inserted during pulsing operation.

Despite the built-in safeguards and inherent safety of the reactor and its fuel, great attention is paid to proper supervision of operation and adherence to procedures approved by competent authority. It is the policy of the

The actual activity discharged will average about 10% of the value indicated as maximum. Thermo-Luminescent area monitor dosimeters (TLD) are used to assure that the exposure is at these low levels in the non-restricted areas surrounding the laboratory.

The possibility that activity emitted through the stack might be drawn back into the laboratory and other parts of the building has been investigated. Based on this investigation and on measurements made in the facility, this condition does not exist. Further, the area monitor dosimeters used within the laboratory will give indications of β exposure if the A^{41} level in the area approaches MPC levels.

6.3 SPILLAGE OF RADIOACTIVE MATERIALS

A problem of importance in the analysis of a reactor location is the effect on surrounding unrestricted areas of the spillage of radioactive materials. This problem might arise, for example, if a highly-volatile liquid were irradiated in the reactor for the production of isotopes. If, while it was being transferred from the reactor to a cask, it were dropped and its container broken, the atmosphere within the Reactor Laboratory could become conceivably contaminated; further, this atmosphere could conceivably be released to the surroundings in such a fashion as to present a health hazard in unrestricted areas. This problem may be of importance when the material being irradiated is highly volatile, or is a solid in powdered form. For a typical solid or liquid no special problems exist other than the direct radiation from the sample and the problem of cleaning up contamination. Since the level of radiation will be known for each sample, adequate equipment for handling the sample will be available when the material is discharged from the reactor. Equipment adequate for cleanup of spills will be kept available so that spills can be dealt with immediately, lessening the possibility of spreading contamination to adjacent areas. The remainder of this section will deal with gases, highly volatile liquids, or powdered samples which might cause air-borne activity in the event of a spill.

University of Wisconsin that standard operating procedures are carefully prepared and reviewed, strictly followed, and kept current. Likewise, competent supervision assures that operation is kept within the limits set by licenses, technical specifications, existing procedures, and general good practice.

6.2 PRODUCTION AND RELEASE OF GASEOUS RADIOACTIVITY

Calculations were performed to determine production and release rates of the various activities that might be discharged due to normal operation. Sample calculations are given in the appendix.

Due to the operation of the beam port and thermal column ventilating system and the laboratory exhaust fan, the airborne activity levels in the laboratory are low. The concentration of gaseous activity (primarily A^{41}) in the laboratory is about one-twentieth of MPC for non-restricted areas. Therefore, further discussion will be concerned with the activity released to the atmosphere.

Argon-41 is the only activity expected to be released in significant quantities. The maximum release rate of A^{41} would occur with the reactor operating continuously at 1 MW and all four beam ports and the thermal column open. Such operation is not reasonable, but it does establish an upper limit to the activity that might be discharged. This maximum release rate is 13.3 μc/sec, giving an A^{41} concentration at the stack outlet of 2.4×10^{-5} μc/ml.

The maximum concentration to which personnel would be exposed (using Gifford's model as discussed in the appendix) in this case would be about 3×10^{-8} μc/ml.

As previously indicated, the above value is for a situation not likely to occur during operation. The usual procedure is to have the experimental facilities in a no-flow condition if possible. Under no-flow conditions the beam port and thermal column ventilation system keeps the pressure in the experimental facilities lower than room pressure, and the activity produced in the facilities remains there and decays.

Although special packaging requirements are enforced to prevent breakage of pneumatic tube samples, such breakage may occur. Therefore, a more restrictive limit is placed on the activity which may be produced in the pneumatic tube. Volatile sample size is limited to that level which, diluted by the flow through the hood during a 24-hour period, (4.9×10^{10} ml) will result in average concentrations less than 10 CFR Part 20 limits. The blowers operate automatically whenever the pneumatic tube system is used. As with the other samples, the maximum activities generated must have special approvals, and only quantities considerably smaller are routinely approved.

Finally, no such sample breakage has occurred during previous operations involving thousands of sample irradiations, and such breakage is considered quite unlikely to occur. Should 10 CFR Part 20 not specify maximum permissible concentrations for a particular isotope to be produced, limits will be established by consultation with the University Radiation Safety Committee.

6.4 REACTIVITY ACCIDENT--Ejection of the Transient Rod while at Maximum Steady-State Power.

Calculations performed by Gulf General Atomic[1] indicate that a peak temperature of 1150°C in FLIP fuel will not produce a stress in the fuel clad in excess of the ultimate strength. Further, TRIGA fuel with a H/Zr ratio of at least 1.65 has been pulsed to temperatures of about 1150°C without any damage to the clad[2]. In a mixed FLIP-Standard TRIGA core the peak temperatures in FLIP fuel are much higher than in standard fuel due to the peaking of the power distribution near water gaps. For this reason the subsequent analysis in this section is concerned with internal temperatures in FLIP fuel elements.

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[1] "Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor", GA-9064, Gulf General Atomic, Jan. 5, 1970.

[2] "Annular Core Pulse Reactor", General Dynamics, General Atomic Division Report GACD 6977, Supplement 2, 9/30/66.

This problem is handled at Wisconsin by a combination of administrative and operational procedures. For the normal situation, a concerted effort will be made to keep the concentration of contaminants in the atmosphere released from the Reactor Laboratory well below the limits as stated in Table II, Appendix B, 10 CFR Part 20, "Standards for Protection Against Radiation". Among the procedures which will be followed to achieve this goal will be the double-encapsulating of materials to be exposed in the reactor in aluminum containers (for long exposure) or sealed polyethylene containers for exposures of less than 4 x 10^{17} thermal neutrons/sq. cm. with accompanying gamma ray and fast neutron fluxes. Only members of the reactor staff (or selected people working under their supervision) will be permitted to handle these capsules within the Reactor Laboratory and the capsules will normally be opened only at appropriate locations outside the laborator. Further, a log book will be maintained of all material exposures. However, it is recognized that accidents can occur, and the amount of radioactivity which will be generated in any one sample of material will be limited. Specifically, this amount of radioactivity will be limited such that, should a container be broken and its contents disperse in the air within the Reactor Laboratory, the concentrations discharged through the stack when averaged over one week will be within the maximum concentrations of 10 CFR Part 20. Since the large fan has a capacity of 9,800 cfm through its filters, the weekly flow of dilution is 2.8 x 10^{12} ml. Normal approvals will be given for concentrations considerably smaller than these, however, and samples of such size as to approach these limits must have special approvals. These approvals will consider all other activity discharged, and will insure that the total stack discharge lies within permissible limits should the sample rupture.

Pneumatic tube stations are located outside the Reactor Laboratory and thus not subject to the laboratory ventilation system. Each station is installed within a fume hood having a high face velocity (to protect the system operator in case of sample breakage). The blowers for the fume hoods have capacities of 1200 ft^3/minute. The air discharged from each hood is passed through high efficiency filters and then exhausted to the atmosphere.

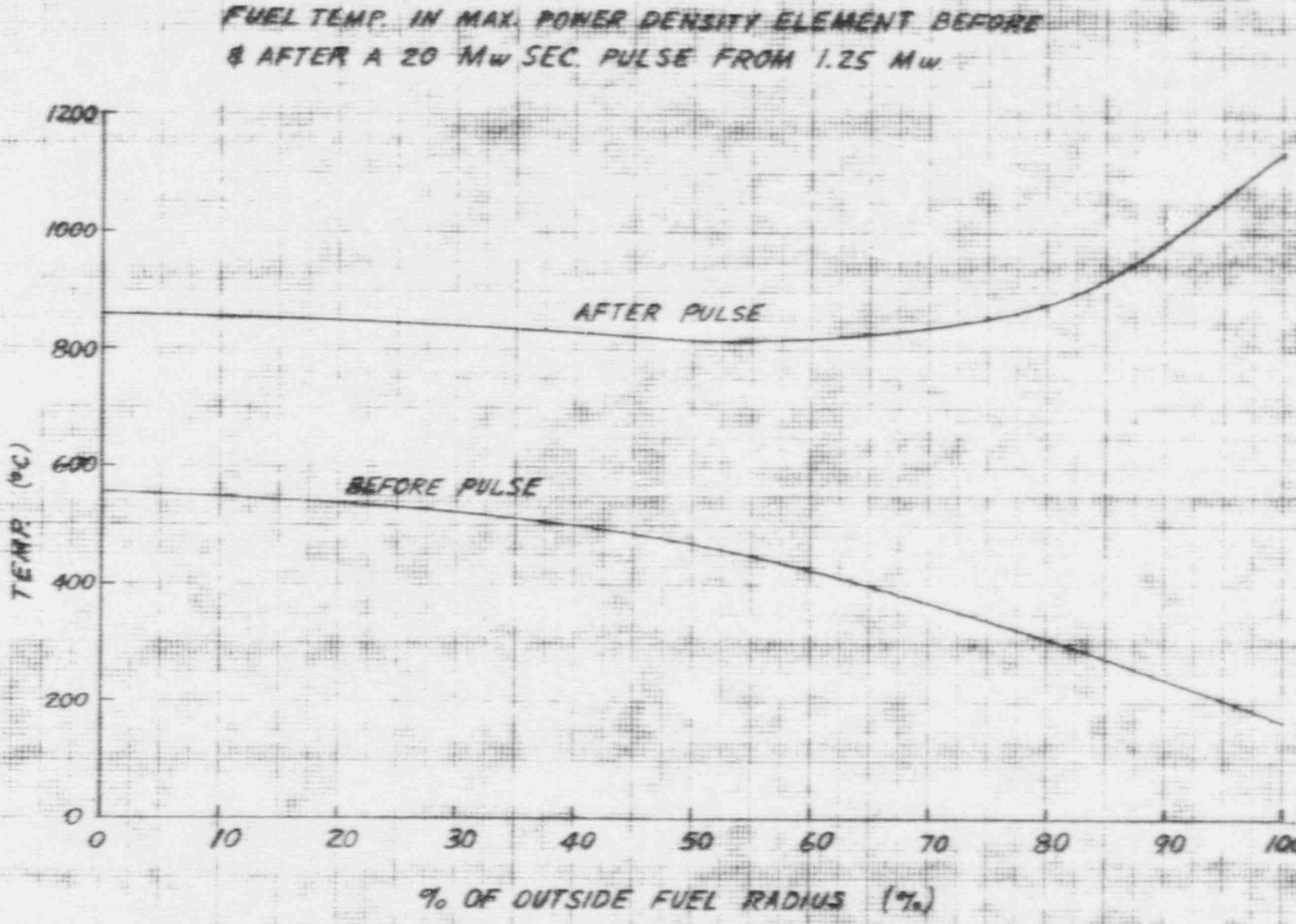
FUEL TEMP. IN MAX. POWER DENSITY ELEMENT BEFORE & AFTER A 20 Mw SEC. PULSE FROM 1.25 Mw.

FIGURE 1

A worst case core arrangement is considered, in which a FLIP element is located adjacent to a 3-inch square water gap. The power density in the FLIP element is at the maximum permissible value based on consideration of the loss of coolant accident (22 KW when the core is operating at 1 MW). The core is operating at the power level scram point of 1.25 MW, and the transient control rod is fired to initiate a pulse.

Pulses of 2.1% $\Delta K/K$ fired at this facility have had energy releases of less than 20 MW seconds. Although no radical change in energy release in pulses is expected for the core arrangements contemplated, Technical Specification limits will restrict pulsing reactivity insertions to the value which causes an energy release of 20 MW seconds in a pulse.

The limitation of experiment reactivity to 2.1% Δ K/K will insure that reactivity insertions from experiment removal or failure will insure that such an accident will result in consequences no worse than those considered here.

Firing the transient rod while at full power is prevented by interlocks and administrative requirements. Removal of an experiment while operating at full power would not result in a reactivity insertion rate as large as that resulting from firing the transient rod, and the most likely result of experiment removal under the conditions assumed would be a reactor scram from power level, period, and full temperature trips. Further, experiments having worths approaching 2.1% Δ K/K are fastened to prevent inadvertent removal, and administrative restrictions do not allow such manipulations while the reactor is in operation. The predicted conditions establish an upper limit for a reactivity accident.

6.4.1 Fuel Temperatures from Operation at the Scram Point

Calculations for the SAR of the Puerto Rico Reactor resulted in the information presented in the lower curve in Figure 1.[3] This curve shows the fuel temperature distribution at the axial centerline in a FLIP fuel element operating at Conditions of slightly higher power density

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[3] Safeguards Summary Report for the TRIGA-FLIP Reactor at Puerto Rico Nuclear Center, Report PRNC 123, Revision C, November 11, 1969.

6.5 FUEL ELEMENT CLADDING FAILURE

The liklihood of a major fuel element cladding failure is considered small. The elements must meet rigid quality control standards; pool water quality is carefully controlled; and much care is taken in handling fuel.

Such clad failures are, however, possible and the remainder of this section is concerned with the consequences of such a failure.

Calculations are performed for cladding failure in a fuel element exposed at the maximum power density of 23 KW for an infinite operating time.

The release of radioactivity by corrosion and leaching by the pool water has been measured at Gulf General Atomic. About 100 micrograms of U-ZrH per square centimeter of exposed fuel surface per day is released for shutdown conditions. This release is easily controlled by isolating the leaking element in a container provided for that purpose. The gaseous and highly volatile fission products that have collected in the space between fuel and cladding would be the activity contributing to personnel hazards.

6.5.1 Fission Product Inventory in Fuel Element

The quantity of these volatile and gaseous fission products was determined by the use of Perkins and King* data. Column B of Table 1 indicates the fission product activities in the fuel element exposed to the maximum power density.

6.5.2 Fission Product Release Fraction

The release of fission products from U-ZrH fuel elements has been extensively studied by Gulf General Atomic and others. The results of this work indicate that the release of fission product gases into the gap between fuel and cladding is given by the following relationship:

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*J. F. Perkins and R. W. King, "Energy Release from the Decay of Fission Products", Nuclear Science and Engineering, 3, 726 (1958) than that predicted here. (The Puerto Rico case is an element operating at a power density in the maximum element of 1.4 times the average of 22.3 KW/element. The axial peaking factor is 1.3. The UWNR case is an element operating at 23 KW times the ratio of 1.25 of scram setting/licensed power level, with the same axial peaking factor of 1.3.) The fuel centerline and average temperatures will be lower in the UWNR core, but the temperature at the outer surface of the fuel would be approximately the same in both cases,

6.4.2 Temperature after Pulse

Firing a pulse while at the scram point would cause the reactor to scram from period, power level and fuel temperature scrams. The entire pulse energy release is used, however, in the following analysis.

The temperature distribution in the fuel element immediately after a 20 MW second pulse is plotted as the top curve in Figure 1. The peaking factor within a FLIP element adjacent to a 3-inch square water-gap is 2.49, and an axial peaking factor of 1.3 is used as in the steady state conditions. The energy deposited in the element under consideration is calculated using the same peaking factor (power in maximum element/ power in average element in core) which resulted in the 23 KW steady state level.

The maximum adiabatic temperature reached in the element will occur at the outer surface of the fuel element adjacent to the water-gap. This maximum temperature would be 1133°C, slightly below the safety limit of 1150°C.

Although such an event is considered highly unlikely, it would not cause fuel damage or release of fission products from the reactor.

TABLE 1

FISSION PRODUCT YIELD AND RELEASE POTENTIAL

A	B	C	D	E	F	G	H	I	J
ISOTOPE	SATURATED INVENTORY (Ci)	RELEASED ACTIVITY (Ci)	AMOUNT IN WATER (Ci)	AMOUNT IN AIR (Ci)	LABORATORY CONCENTRATION (µ Ci/ml)	MPC	CONCENTRATION TO NONRESTRICTED AREA (µ Ci/ml)	MPC TO NON-RESTRICTED AREA (µ Ci/ml)	RATIO COL. H/J
Xe 131m	5	0.004		0.004	2.0 x 10^{-6}	2 x 10^{-5}	0.21 x 10^{-9}	4 x 10^{-7}	0.000525
133m	31	0.025		0.025	1.6 x 10^{-5}	1 x 10^{-5}	1.33 x 10^{-9}	3 x 10^{-7}	0.00443
133	1282	1.015		1.015	3.1 x 10^{-4}	1 x 10^{-5}	5.41 x 10^{-8}	3 x 10^{-7}	0.18040
135m	350	0.277		0.277	1.4 x 10^{-4}	8 x 10^{-6}	1.48 x 10^{-8}	2 x 10^{-7}	0.07385
135	1243	0.984		0.984	4.9 x 10^{-4}	4 x 10^{-6}	5.25 x 10^{-8}	1 x 10^{-7}	0.52470
137	1185	0.938		0.938	4.7 x 10^{-4}	1 x 10^{-6}	5.00 x 10^{-8}	[illegible] x 10^{-8})	1.66733
138	894	0.707		0.707	3.5 x 10^{-4}	2 x 10^{-6}	3.77 x 10^{-8}	4 x 10^{-8})	0.9425
TOTAL		3.950		3.950					

SELECTED RELEASE TOTALS

Halogen Gamma Emitters	5.20 Ci
Halogen Beta Emitters	5.87 Ci
Total Halogens	5.87 Ci
Insoluble Gamma Emitters	3.52 Ci
Insoluble Beta Emitters	5.50 Ci
Total Insoluble Volatiles	5.89 Ci

TABLE 1

FISSION PRODUCT YIELD AND RELEASE POTENTIAL

A	B	C	D	E	F	G	H	I	J
ISOTOPE	SATURATED INVENTORY (Ci)	RELEASED ACTIVITY (Ci)	AMOUNT IN WATER (Ci)	AMOUNT IN AIR (Ci)	LABORATORY CONCENTRATION (μ Ci/ml)	MPC	CONCENTRATION TO NONRESTRICTED AREA (μ Ci/ml)	MPC TO NON-RESTRICTED AREA (μ Ci/ml)	RATIO COL. H/J
Br 82	30	0.024	0.024	0.002	1.2×10^{-6}	1×10^{-6}	0.13×10^{-9}	4×10^{-8}	0.00325
83	105	0.083	0.083	0.008	4.2×10^{-6}	8×10^{-5}	0.44×10^{-9}	(2×10^{-6})	0.00022
84	194	0.153	0.154	0.015	7.7×10^{-6}	8×10^{-5}	0.82×10^{-9}	(2×10^{-6})	0.00041
85	253	0.200	0.200	0.020	1.0×10^{-5}	1×10^{-3}	1.07×10^{-9}	(3×10^{-5})	0.00004
87	600	0.473	0.475	0.047	2.4×10^{-5}	2×10^{-3}	2.53×10^{-9}	(4×10^{-5})	0.00006
	TOTAL	0.933		0.093					
I 130m	200	0.158	0.158	0.016	7.9×10^{-6}		0.84×10^{-9}		
131	563	0.446	0.446	0.045	2.2×10^{-5}	9×10^{-9}	2.38×10^{-9}	1×10^{-10}	20.40
132	855	0.677	0.677	0.068	3.4×10^{-5}	2×10^{-7}	3.61×10^{-9}	3×10^{-9}	1.20
133	1282	1.015	1.015	0.102	5.1×10^{-5}	3×10^{-8}	5.41×10^{-9}	4×10^{-10}	13.53
134	1554	1.230	1.230	0.123	6.2×10^{-5}	5×10^{-7}	6.56×10^{-9}	6×10^{-9}	1.09
135	1185	0.938	0.938	0.094	4.7×10^{-5}	1×10^{-7}	5.00×10^{-9}	1×10^{-9}	5.00
136	602	0.477	0.477	0.048	2.4×10^{-5}	(1×10^{-7})	2.54×10^{-9}	1×10^{-7}	0.03
	TOTAL	4.941		0.494					
Kr 83m	105	0.084		0.084	4.2×10^{-5}	(7×10^{-4})	0.44×10^{-9}	(2×10^{-6})	0.00022
85m	253	0.200		0.200	1.0×10^{-4}	6×10^{-6}	1.07×10^{-8}	1×10^{-7}	0.10660
85	51	0.040		0.040	2.0×10^{-5}	1×10^{-5}	2.13×10^{-9}	3×10^{-7}	0.00710
87	486	0.386		0.385	1.9×10^{-4}	1×10^{-6}	2.05×10^{-8}	2×10^{-8}	1.02400
88	699	0.556		0.555	2.8×10^{-4}	1×10^{-6}	2.95×10^{-8}	2×10^{-8}	1.47700
89	855	0.669		0.669	3.4×10^{-4}	(8×10^{-7})	3.61×10^{-8}	(2×10^{-8})	1.805
	TOTAL	1.935		1.935					

laboratory for ten minutes; and (c) the dose to the thyroid of an individual remaining in the room ten minutes. For the latter calculations, it is assumed that 10% of the iodine radioisotopes escape from the pool water.

(a) Whole body exposure due to gamma emitters

The amount of insoluble volatiles released to the room would be 5.89 Ci. If this activity is distributed uniformly in the laboratory volume, the resulting concentration would be 2.95×10^{-3} µCi/cm^3. (See Appendix)

The resulting maximum dose rate is calculated to be 60 mrem/hr. An individual remaining in the laboratory for 10 minutes after a release would receive a whole body dose of 10 mrem.

(b) Dose to the lungs

The lung is the critical organ when considering the effects of inhaling the insoluble volatiles from a ruptured fuel element. The beta emitting nuclides become more important than those emitting gamma rays since all the decay energy is absorbed in lung tissue.

The calculation outlined in the appendix indicates the lung exposure for an individual remaining in the laboratory for 10 minutes after a clad rupture to be 1.0 rad.

(c) Thyroid dose

The thyroid dose to a person in the reactor room was calculated assuming that he remained in the laboratory for 10 minutes after the fission product release. If the pool water is not lost and 10% of the halogens released escape into the atmosphere, the concentrations of the various iodine isotopes would be as presented in Table 1. In a ten minute period the lungs would be exposed to the iodine isotope activities shown in Table 2. As before, it was assumed that the "standard man" breathes 1.25 M^3/active-hour and his lungs hold 3 liters of air. A conservative calculation results in a dose to the thyroid of 18.9 rads.

Although all doses were calculated based on an individual remaining in the laboratory for ten minutes,

$$FR = 1.5 \times 10^{-5} + 3.6 \times 10^{3} e^{\frac{-1.34 \times 10^{4}}{T}} \qquad (1)$$

where T is the maximum fuel temperature (^{o}K) in the element during normal operation.

The maximum fuel temperature in a fuel element operated in the steady-state mode at 23 KW will be less than 440°C. Calculations of release fraction however, are based on 600°C in order to assure a conservative result.

The release fraction corresponding to 600°C is 7.9×10^{-4}. Applying this fraction to the total inventory of the fuel element as given in column B of Table 1 gives the released activity as shown in column C of the table.

For the purpose of further calculations, it is assumed that all gaseous fission products are released to the room air whether the pool is filled with water or not. For soluble volatiles, calculations assume all activity is absorbed in pool water for calculations of pool water activity (column D). For calculations of air activity, the assumption is made that 10% of the volatiles escape with the pool filled with water (columns E and F) and 100% escape with the pool empty.

6.5.3 Activity in Pool Water

If 100% of the soluble fission products are absorbed in the pool water, the resulting activity level will be 0.075 uCi/ml. Within 24 hours the level would be reduced by radioactive decay to about 0.012 μCi/ml. After 24 hours the activity decay rate would be chiefly determined by the I^{131} half life (8.05 days). The demineralizer will remove most of this activity, giving a radiation dose rate of of about 88 mrem/hr at one meter after the activity is deposited in the resins. The resins can be dumped to an underground storage pit where the activity will decay without hazard to personnel.

6.5.4 Fission Product Release to Air within the Reactor Laboratory

Calculations were performed to determine (a) the dose rate due to gamma emitters uniformly dispersed throughout the volume of the reactor lab; (b) the dose to the lungs from beta emitters for an individual remaining in the

The total of the ratios of individual concentrations to MPC was calculated to be 48.1, where MPC values are for nonrestricted areas, 168 hours per week. When averaged over a year's time, the resulting average concentration is 0.00715 of the maximum indicated by 10 CFR Part 20 for nonoccupational exposure in nonrestricted areas. Even with the effluent discharge from normal operation (see Section 6.2) the total concentration to which personnel might be exposed is not excessive.

A more conservative calculation which assumes zero stack height (see appendix) was performed. This analysis is applicable to a situation in which the laboratory ventilation system fails and the release takes place through building leaks. For purposes of comparison, it was assumed that the release occurred in the time required for the ventilation system to make an air change in the laboratory. The effect of this analysis is to multiply the values in columns H and I by a factor of 10.17, giving a resulting average concentration (yearly average) of 0.0727 times Part 20 limits. Finally, an additional calculation was performed assuming 100% release of Br and I and the more conservative calculation (zero stack height) of atmospheric dilution. The resulting ratio of concentrations to MPC in this case would be 419.4. Averaged over a year's time, the resulting concentration (yearly average) is 0.635 times MPC.

In terms of approximate exposure, these cases would result in the exposures tabulated below if persons were to remain in the area where the maximum concentration exists for the 4238 second assumed release time. Actual doses could be reduced by selected evacuation of the area.

Case	Maximum Exposures In Nonrestricted areas	
	Total Body Dose	Thyroid Dose
Fuel clad leak with normal operation of ventilating system; pool filled	0.006 rem	0.010 rad
Fuel clad leak with failure of ventilating system; pool filled	0.084 rem	0.102 rad
Fuel clad leak with concurrent loss of pool water and failure of ventilating system	0.153 rem	1.019 rad

All calculated exposures are within 10 CFR 20 limits averaged over a one year period even in the event of the worst condition postulated.

emergency procedures require immediate evacuation after scramming the reactor, and re-entry to the area is made using self-contained breathing apparatus. Actual doses in the event of the accident would be a factor of 10 less than calculated, considering reasonable evacuation times.

TABLE 2

IODINE-THYROID DOSE INFORMATION

Iodine Isotope	10 minute Exposure (μCi)	$D\infty/A_t$* (rads/Ci)	DOSE (rads)
130+	2.4	(2.5×10^4)	(0.06)
131	6.6	1.48×10^6	9.77
132	10.2	5.35×10^4	0.55
133	15.3	4.0×10^5	6.12
134	18.6	2.5×10^4	0.46
135	14.1	1.25×10^5	1.76
136+	7.2	(2.5×10^4)	(0.18)
			18.9

*From TID-14844, p. 25

+These isotopes have a very short half life. Corresponding values for $D\infty/A_t$ are conservative estimates.

6.5.5 Release of Fission Products to Unrestricted Areas

Columns H, I, and J of Table 1 are concerned with the exposure of personnel outside the restricted area. Calculations were performed as indicated in the appendix. The maximum concentrations which might be expected in nonrestricted areas were calculated under the assumption that venting took place in the time required for the ventillation system to make one complete change in the laboratory. Wind velocity was assumed to be the lowest average for any month.

The total dose to personnel in the nonrestricted area is independent of whether the large exhaust fan or the normal ventilating fan is used; the concentration would be considerably higher if the larger fan were used, but the period of exposure would be proportionally shorter. It is also emphasized that the total exposure figure is a maximum to be expected at any point other than within the areas evacuated in the event of an accidental release.

TABLE 3
CALCULATED RADIATION DOSE RATES

Time After Shutdown	Floor Level R/hr.	Console R/hr.	Pool Top	
			Direct Radiation R/hr.	Scattered Radiation R/hr.
10 seconds	1.6	2.0	1.0×10^4	2.6
1 day	0.17	0.24	1.2×10^3	0.30
1 week	0.08	0.12	5.4×10^2	0.14
1 month	0.03	0.04	1.4×10^2	0.04

These levels are not too high to allow emergency repairs to be made. Facility emergency procedures cover the situation of pool water loss. A copy of this procedure is appended to this chapter.

6.6.3 Fuel Temperature After Loss of Pool Water

Calculations performed at Texas A & M University have treated the loss of coolant accident in detail, based on reactor shutdown 15 minutes before the core is uncovered. At Wisconsin, the pool level scram would cause automatic shutdown much sooner, as the A & M calculation is based on pool drainage by rupture of a 10-inch line. Other parameters of the two facilities are identical. The calculations employed the Gulf computer code TAC for calculation of system temperatures.

The results of these calculations (Pages 25-31 of Texas A & M University Nuclear Science Center Amendment II to the Safety Analysis Report, November 1, 1972 submitted under Docket for License R-83) indicate that for a maximum power density of less than 21 kW/element for standard and 23 kW/element for FLIP fuel, loss of coolant water would not result in fuel clad failure and release of fission products.

6.7 RE-ANALYSIS FOR COMPACT NINE BUNDLE FLIP MIXED CORE

Sections6.1 through 6.3 of this chapter are independent of core arrangement. The other safety analyses are reevaluated below for the mixed core described in Section 4.8.

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6.6 LOSS OF POOL WATER

Although there is little liklihood of complete loss of water from the reactor pool, an analysis is made to demonstrate that such loss will not damage reactor fuel.

6.6.1 Possible Means of Water Loss

The pool is contained within the thick reinforced concrete reactor shield which will maintain its integrity under the most severe earthquake that would be expected in this area.

A sheared and open beam port could drain the water level to mid-core height in about 400 seconds, but water would still be in contact with the fuel and would prevent excessive temperatures.

The 8-inch stainless steel pipes built into the pool walls for possible future use in a forced convection cooling system are flange sealed on the outer ends. In addition, one of these pipes has a loop and a siphon breaker extending well above the core so that a rupture cannot lower pool level below the core. The other pipe is flange sealed inside the pool and penetrates the shield wall well above the core. Rupture of either of these lines will not uncover the core.

Rupture of the piping in the demineralizer could cause only slight water loss due to location of the outlet lines from the pool and a check valve at the demineralizer outlet.

6.6.2 Radiation Levels Due to Unshielded Core

Calculations of radiation levels at various points in the Reactor Laboratory were made assuming operations at 1000 kW for an infinite time. Doses from direct and scattered radiation were considered, with the scattered dose calculated for the case of a thick concrete ceiling nine (9) feet above the pool. Results of the calculations are given in Table 3.

Appendix

CALCULATION METHODS FOR ATMOSPHERIC RELEASE OF RADIOACTIVITY

References: (1) Meteorology and Atomic Energy, U. S. Dept. of Commerce Weather Bureau, Govt. Printing Office, Washington, D. C (July 1955).

(2) F. A. Gifford, Jr., Atmospheric Dispersion Calculations Using the Generalized Gaussian Plume Model, Nuclear Safety, December 1960.

(3) Calculation of Distance Factors for Power and Test Reactors, (TID-14844), USAEC, Mar. 23, 1962.

A. Models Used for Calculations

For Sutton's diffusion model, the maximum concentration (X_{max}) at any point downwind is given as:

(1) $X_{max} = \frac{2Q}{e\pi\bar{\mu}\;{}^{2}}$, where (Ref. 1)

$\bar{\mu}$ = mean wind speed
Q = release rate, Ci/sec
h = stack height

For the generalized Gaussian Plume Model, the maximum concentration is given by the same equation (Ref. 2, eq. 8).

For calculations in this report, the following values are used:

$\bar{\mu}$ = mean wind speed = lowest monthly average = 3.54 meters/sec.

h = stack height above ground = 17.1 meters
and (2) $X_{max} = 2.26 \times 10^{-4}\ Q\ \mu c/ml$.

Reference 2 presents a method applicable to release from buildings with zero stack height to approximate release from buildings with zero stack height to approximate release from leaks in a containment structure. This

The major changes resulting from use of the initial technical specifications will be a limitation of power peaking to the 18.11 kW in the maximum element and the limitation of reactivity insertion from firing the transient rod (or failure of an experiment with the maximum permitted reactivity worth) to 1.4% Δ K/K.

The analysis of Section 6.4 will differ in three respects. First, the temperature of the fuel in the maximum element will be lower due to the difference between the 23 kW/element used in the calculation and the expected value in the initial core of 18.11 kW. The temperature at the thermocouple will be about 70°C lower at the beginning of the pulse. Second, the use of a compact array of nine (9) FLIP bundles reduces the possible peaking factor within a FLIP element from the 2.49 value used in the original calculation to a value of 2.03 for a FLIP element beside the transient rod guide tube (this is the position with highest power density in the core.) Finally, reduction of allowable pulsed reactivity insertion from 2.1% Δ K/K to 1.4% Δ K/K will substantially reduce the energy generation in a pulse, while the limititation of experiment worth to 1.4% Δ K/K will provide similar safeguards for experiment failure or removal. Measurements performed on the Puerto Rico Nuclear Center TRIGA-FLIP reactor indicated that a pulse insertion of 1.4% Δ K/K resulted in a maximum fuel temperature rise of approximately 400°C.*

Consideration of all these differences indicates that the analysis in Section 6.4 shows a peak fuel temperature of about 450°C higher than is expected in the case considered in this section. It is therefore concluded that fuel damage would occur in neither case, but with a much larger safety margin in the more restrictive case considered here.

Sections 6.5 and 6.6 are constent and based on maximum values well above the 18.11 kW in the maximum element of the core considered here.

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*Docket 50-120, Change No. 11 to the Technical Specifications Facility License R-83, Texas A & M University, Section 3.2 Basis.