UNIVERSITY OF MISSOURI, COLUMBIA
MISSOURI UNIVERSITY RESEARCH REACTOR
LICENSE NO. R-103
DOCKET NO. 50-186

HAZARDS SUMMARY REPORT
IN SUPPORT OF AN APPLICATION
FOR A RESEARCH REACTOR

JULY 1, 1965

REDACTED VERSION*

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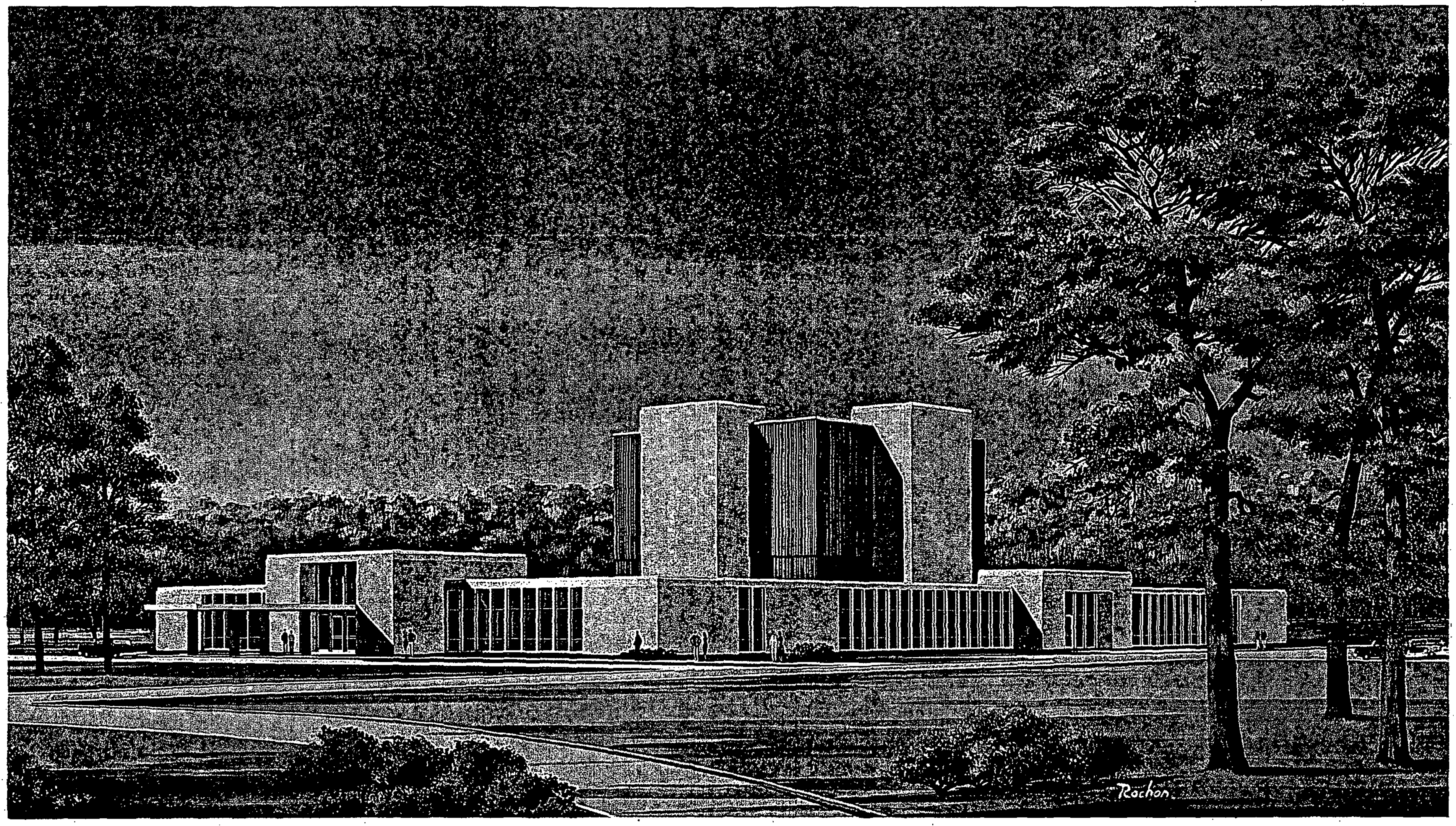

RESEARCH REACTOR FACILITIES UNIVERSITY OF MISSOURI CORNELIUS L. T. GABLER, AIA ARCHITECT

UNIVERSITY OF MISSOURI
RESEARCH REACTOR FACILITY
HAZARDS SUMMARY REPORT

In Support of An Application to
The United States Atomic Energy Commission
For a Class 104 Utilization Facility License

Submitted by
THE UNIVERSITY OF MISSOURI
Columbia, Missouri

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July 1, 1965

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1981-1982

1.0 INTRODUCTION AND SUMMARY

1.1 Description of the Facility

The Research Reactor Facility of the University of Missouri is the first of several large scale projects which have been undertaken by the State of Missouri to establish pre-eminence in research capability at the University.

1.1.1 Location

The Research Reactor Facility, a totally new construction, is located on a 7.5 acre lot within a University developed Research Park. The 85 acre Research Park is about one mile southwest of the main campus.

1.1.2 Laboratory Facility

The laboratory associated with the reactor facility is a one level building of poured concrete, block and brick construction which completely surrounds and is an integral part of the reactor building. The laboratory contains shops, conference rooms, library, offices, spaces for individual research projects, ~~animal quarters~~ DELETE and the mechanical equipment for the entire facility. 1981-1982

1.1.3 Reactor

The reactor is housed in a five level poured concrete containment building which extends one level below grade. The reactor containment building has a gross volume of approximately 280,000 cubic feet. Within the containment building are: the reactor and its biological shield, ~~a laboratory for reactor operations,~~ DELETE 1981-1982

1981-1982
1973-1974
2001

the control room, office spaces, auxiliary mechanical equipment, a 15 ton capacity rectilinear crane, and a passenger elevator that services all "four" levels. Access to containment is available through a personnel air lock at grade level and a freight door at the below grade level. 1981-1982

The reactor cooling and pressurization equipment and the coolant water treatment equipment is housed in shielded below grade spaces external to the containment building.

The below grade area also includes ~~a Co-60 gamma irradiation well,~~ a 15 ton capacity hydraulic freight elevator, and the blower unit for a pneumatic tube reactor irradiation system. All of these facilities or components are outside containment. 2001

The nuclear reactor is fueled with uranium enriched to greater than [redacted] per cent in the isotope U-235. It is water cooled and moderated and is reflected with beryllium metal and graphite. ~~The fuel region~~ 1973-1974

"The primary coolant system is pressurized to approximately 75 psia pressurizer pressure."

The reactor core consists of 8 fuel elements each occupying a 45° segment of a cyclindrical annulus which is nominally 12 inches O.D. and 9 inches I.D. Each element contains 24 concentric fuel plates of equal weight per cent loading. The fuel region is 24 inches long and each element has a total length of 32.5 inches. The volume in the center of the reactor fuel annulus is unpressurized ~~and consists of two regions. The innermost region is a 1.5 inch~~ 1981-1982

1994
1981 ·
1982

~~I.D. "flux trap" which is accessible from pool surface for insertion and removal of experiments. A water filled region, which is not accessible, separates the flux trap from the fuel region.~~ DELETE 1981 - 1982

The reactor is controlled by varying neutron reflection. A cylindrical shroud of neutron absorbing material is inserted between the fuel region and the beryllium reflector. This shroud is in five independently movable sections which, with the associated actuators, are totally outside the reactor pressure vessel. Four sections of boral constitute the shim rods, the fifth is the regulating rod made of stainless steel.

The center of the reactor core is located 5 feet 3 inches above the bottom of a 10 foot diameter, 30 foot deep pool. The pool is lined with aluminum. A heavy aggregate concrete biological shield surrounds the pool.

The reactor services the following experimental facilities:

(1) Center test hole, or "flux trap."
(2) 3 - 6 inch I.D. beam tubes.
(3) 3 - 4 inch I.D. beam tubes.
(4) **Pneumatic tube irradiation positions.** 1994
(5) ~~8~~ reflector region irradiation positions. 1981-1982
(6) A 4x4 foot thermal column.

1981-
1982

~~In addition to these "built in" facilities, a sector comprising approximately 30 per cent of the pool is free of impediments and available for the irradiation of large items.~~ DELETE 1981-1982

1.2 Contractors

The design, construction, construction supervision and testing of the University of Missouri Research Reactor Facility has required major contributions from six contractors; two for design and construction supervision and four for construction and testing.

1.2.1 Reactor Design

The reactor preliminary design study, final design and specifications were accomplished by the Internuclear Company of St. Louis, Missouri. Internuclear provided the preliminary design study which was the basis for the AEC Construction Permit. Subsequently, Internuclear completed all plans and specifications for the reactor core configuration, fuel, pressure vessel, control elements, in-pool equipment and piping, cooling equipment, water treatment equipment, instrumentation and controls, experimental facilities, and the shield design.

During reactor final design and construction Internuclear has reviewed all reactor shop drawings to assure compliance with specifications. Internuclear also has conducted in-plant or site inspection of fabricated items.

1.2.2 Facility Design

Design of the laboratory and reactor building has been accomplished by C. L. T. Gabler & Associates, Detroit, Michigan. This architectural firm furnished complete plans and specifications for the total facility including all structural, heating, cooling and electrical. The reactor containment building and the reactor biological shield structural design were furnished by this contractor.

C. L. T. Gabler & Associates is the University's representative for overall construction supervision and acts as arbitrator in matters of dispute.

1.2.3 Reactor System

Final engineering design, fabrication and installation of the reactor system is being accomplished by the General Electric Company, Atomic Power Equipment Department, San Jose, California.

1.2.4 General Construction

Building construction and installation of equipment is being accomplished by the B. D. Simon Construction Company, Columbia, Missouri. The mechanical work is being accomplished by the Natkin Company, Kansas City, Missouri who are subcontractors to the B. D. Simon Company. The electrical work is being accomplished by C. J. Hervey Company, St. Louis, Missouri who are also subcontractors to the B. D. Simon Company.

1.2.5 Reactor System Installation

Completing the contractual arrangements, the B. D. Simon Company is a subcontractor to General Electric for installation of the reactor system.

1.2.6 Operation

The University of Missouri Research Reactor Facility will be staffed and operated by the University. Initial startup of the reactor will be accomplished by the University with the assistance and supervision of the reactor system supplier, General Electric.

1.3 Characteristics of the Reactor

In the following three pages a summary table of the pertinent characteristics of the University of Missouri Research Reactor is presented. This table (Table 1.1) presents the most pertinent characteristics of the reactor.

1974-
1975

TABLE 1.1
SUMMARY OF PERTINENT REACTOR DATA

POWER

Initial power capability, MW	5
~~Eventual~~ DELETE power with modification, MW	10
Average power density at 5 MW, KW/liter	151
Average power density at 10 MW, KW/liter	303

1974-
1975

1974-
1975

AVERAGED THERMAL NEUTRON FLUX

In flux trap at 5 MW	3.1×10^{14}
In flux trap at 10 MW	6.2×10^{14}
In core at 5 MW	2.3×10^{13}
In core at 10 MW	4.6×10^{13}

REACTOR CORE

Geometry	Annular
Inner fuel radius, inches	2.66
Outer fuel radius, inches	5.90
Active fuel height, inches	24.0
Active Volume of Core, liters	33.0

Fuel loading for 10 MW operation
Kg U-235 6.2
AND

Fuel loading for 5 MW operation, Kg U-235	5.2
Cladding of elements	Aluminum
Number of elements	Eight
Fuel plates per element	24
Gap thickness, inches	0.080
Fuel plate thickness, inches	0.050
Meat thickness, inches	0.020
Clad thickness, inches	0.015

1981-1982
1974-1975
1973-1974
1972-1973
1969-1970

CORE COOLANT

Total flow rate @ 5 MW operation, gpm	1800	
Coolant pressure, psia	75	1973-1974
Inlet temperature, °F	140	
Outlet temperature, °F	157	
Demineralizer flow, gpm	50	

POOL COOLANT

Flow rate @ 5 MW operation, gpm	~~40~~ 650 ~~600~~	1973-1974
Mixed pool temperature, °F	100	
Pool inlet temperature, °F	99	
Pool outlet temperature, °F	105	
Demineralizer flow, gpm	50	

ASSORTED CHARACTERISTICS

Measured Reactivity requirements ($\Delta k/k$) @ 5 MW	0.0555	1969-1970
Measured Rod worth (Δk)	0.1655	1972-1973
Core lifetime, MW days	1200	1974-1975
U-235 Consumed, Kg~~m~~	1.27	1974-1975

EXPERIMENTAL FACILITIES

1 ~~1.064 inch~~ Deleted flux trap hole 1981-1982
3 6.0 inch I.D. beamports
3 4.0 inch I.D. beamports
"up to 4"→ ~~4~~ 1.5 inch pneumatic tubes
~~8~~ Reflector irradiation positions 1981-1982
1 Graphite thermal column
1 Fission Product Gamma Irradiation Facility

1981-
1982

REFLECTOR

Inner reflector	Beryllium
Inner reflector thickness, inches	2.71
Outer reflector	Graphite
Outer reflector thickness, inches	8.89
Height, inches	36
Coolant	Pool water

CONTROL RODS

Location	Outside vessel
Type	Curved plate
Material	Boral
Clad	Aluminum
Overall thickness, inches	0.250
Number	
Shim	4
Regulating	1

FLUX TRAP

Inner diameter, inches	3-5/16"	1981-1982
Coolant	Pool water	

WATER ISLAND

Thickness, inches	1"5"	1981-1982
Coolant	Pool water	

1.4 Comments Pertaining to the Construction Permit (CPRR-68)

1.4.1 Introduction

This section is an effort to provide a readily accessible enumeration of design changes which have occurred since issuance of the Construction Permit. These design changes have been made either to increase safety, reduce costs, or to comply with Commission recommendations.

The major headings which follow make reference to a specific heading in the Construction Permit document. The figures in parenthesis preceding each paragraph are the page number and paragraph of the Construction Permit. For example (2,2) is a reference to the second page, second paragraph of the Construction Permit document.

1.4.2 Site

No changes or alterations have occurred.

1.4.3 Reactor

(2,4) In the Construction Permit it states that the core and island are cooled by an external pressurized water loop. In its final form only the core is cooled by the pressurized water loop.

(2,4) In the Construction Permit reference is made to an invert loop in the pool wall. In final form the invert loop is located in the pool proper; not in the shield wall. Further, in the same paragraph

1994
1969-
1970

it is said that a vacuum breaker valve connects to the top of this loop to provide a siphon break. ~~In final form two parallel electrically controlled,~~ "In final form two parallel electrically controlled, air operated valves have replaced the vacuum breaker valve." ~~breaker valve.~~ 1994

(2,5) In the same place it is said that a spring-loaded check valve controls flow from the core to the fin-tube heat exchanger in the pool. In final form this spring-loaded check valve has been replaced with an electrically-controlled, nitrogen-operated, valve.

(2,6) In the same document it is said that cooling of the reflector, control rods and test hole is accomplished by drawing pool water through these regions. This sentence should read; cooling of the reflector, control rods, test hole and flux trap island is accomplished by drawing pool water through these regions.

(2,7) In the same place it is said that "the total worth of the rods is calculated as 19.2% delta k/k, which provides a cold clean shutdown margin of 8.5% delta k/k. The control rods are driven by units located on a traveling bridge structure above the pool water." These two sentences would read, as the device is built, as follows: ~~The total worth of the~~ The total clean core worth of the 5 blades is measured as 14.8% $\Delta k/k$ which provides a cold clean shutdown margin of 8.9% $\Delta k/k$. 1965-1970

1969-
1970

The control rods are driven by units located on a fixed bridge structure above the pool water.

(3,2) In the same document there are mentioned a number of signals which initiate scram. Those enumerated, plus a number of others, initiate scram. This is dealt with in detail in Section 9.0 and Figure 9.4 of this report.

(3,3) The Construction Permit states that seven nuclear instrumentation channels are provided. In final form a total of six nuclear instrumentation channels are provided. One of the original two start-up channels has been eliminated.

(3,4) The Construction Permit makes reference to a requirement for a lesser reactivity excess in the 5 MW core. ~~The excess reactivity has been decreased~~ 1969-1970

The excess reactivity has been decreased from 10.7% $\Delta k/k$ to the latest measured value of 5.6% $\Delta k/k$.

1.4.4 Experimental Facilities

(3,6) The original design as well as the final installation includes six (6) beamports. However, in place of the four 6 inch diameter and two 4 inch diameter ports originally specified, the final installation is of three 6 inch diameter and three 4 inch diameter beamports.

1.4.5 Building

(4,4) It is stated that "Cooling system water lines, heating steam lines, gas, air, and water are brought in through two 4.6 foot deep water legs which will function as seals." In actual form the reactor cooling system water lines enter from beneath the pool. These lines are sealed with a packing gland and are also welded to the aluminum pool liner. The heating steam lines, gas, air and water are brought through one 4.6 foot deep water leg which functions as a building seal.

(4,5) It is stated that both doors (for pedestrian entry) close against inflatable gaskets and may be latched shut with locking dogs. In fact the design is such that no locking dogs are required. When the gaskets inflate the door is forced against stops which effectively lock it in the closed position.

(5,2) This paragraph of the Construction Permit discusses containment integrity. The applicant will provide leak-rate data as a supplement to this Final Hazard Report. This will include a description of testing procedures.

1.4.6 Waste Handling

No changes or alterations have occurred which pertain to this section.

1.4.7 Accident Analysis

(6,1) This paragraph of the Construction Permit makes reference to the existence of a positive reactivity change which occurs with a reduction of water density in the island. The original design concept was to include the island in the core coolant flow stream so that any positive void effect would be offset by a corresponding negative void effect in the core. Subsequent design studies have determined that a safer design puts the island flow in parallel with the reflector, test hole and rod-gap flow (i.e. the pool coolant loop). The positive island temperature coefficient is larger in magnitude than the corresponding negative coefficient for the core. With the island removed from the pressurized core loop a power rise raises the temperature of the core loop but does not produce a similar increase in the pool loop temperature. With this design change the pool loop return temperature will still rise with the power rise, but it will do so with a very long time constant. The reason, of course, is because the volume of water in the pool loop is more than ten times that in the core loop.

1.4.8 Operating Plans

There have been no changes which are pertinent to this section.

1.4.9 Technical Qualifications

No change or alteration has been made.

2.0 SITE

A detailed description of the site and adjacent areas was presented to the AEC in the Preliminary Hazard Report dated March, 1961 and including Amendment No. 1 dated July 28th, 1961 (Docket No. 50-186). That information is not repeated here since the original data provided is pertinent and applicable to the site at Columbia, Missouri.

Figure 2.1 and 2.2 are air photographs of the site taken in May, 1965. Figure 2.1 was taken from an altitude such that the area immediately surrounding the site (i.e. about 2000 ft. radius) is displayed. Areas or facilities visible include the University Golf Course to the west, the baseball diamond to the north, Hinkson Creek to the south and State Route K to the east.

Figure 2.2 is an air photo taken from a much lower altitude and looking north. Figure 2.3 is a sketch of the site.

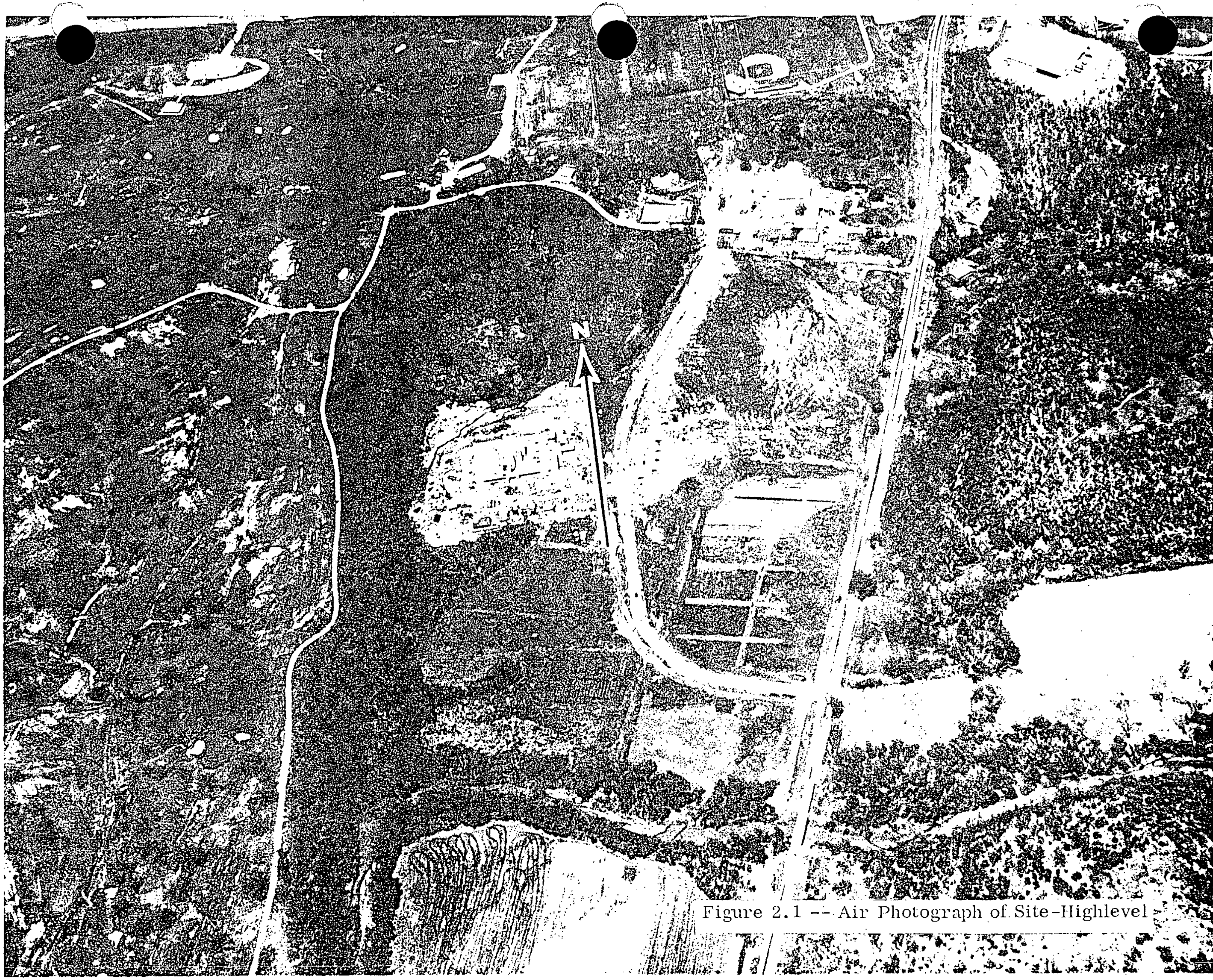


Figure 2.1 -- Air Photograph of Site-Highlevel

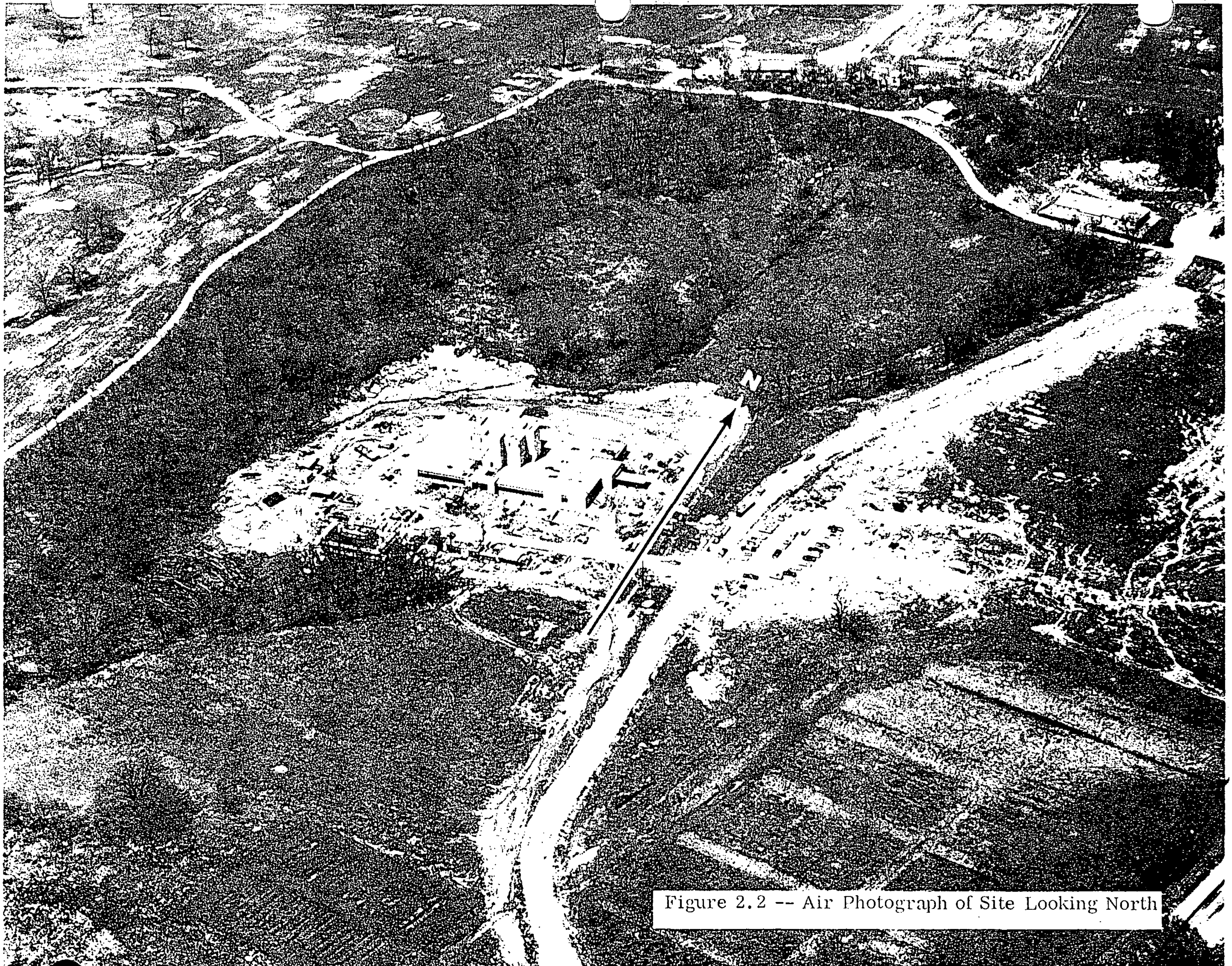


Figure 2.2 -- Air Photograph of Site Looking North

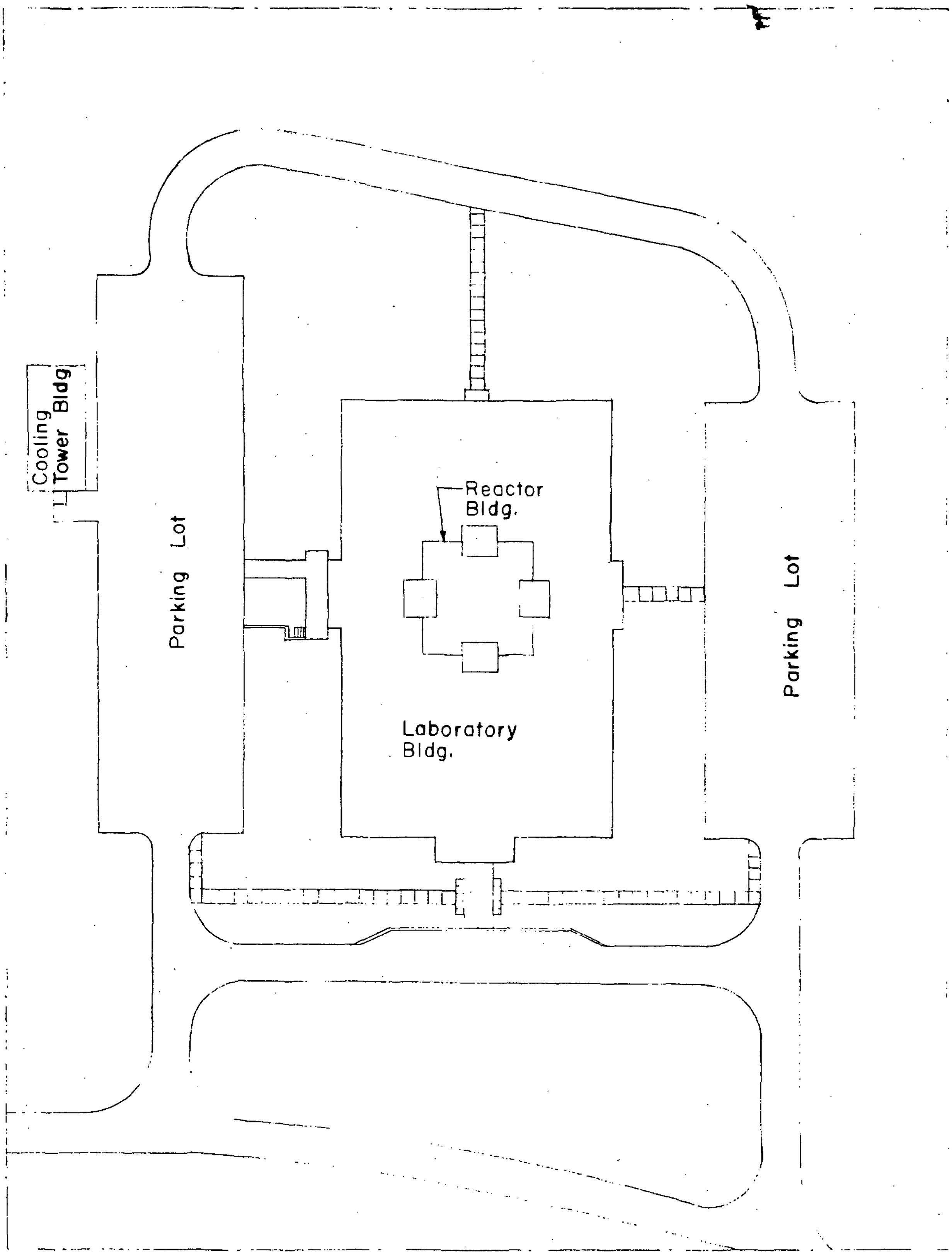


Figure 2.3 Site Plan

3.0 CONTAINMENT SYSTEM

3.1 Introduction

The reactor building is approximately a 60 foot cube with exterior walls of poured concrete. The exterior walls of this cubic structure are of 12 inch thickness. Each horizontal and vertical seam in the pouring is sealed with a rubber water stop. The exterior of the poured concrete walls is finished with removable aluminum sheet siding 16 feet long and 4 foot wide. This sheet siding is attached to angle standoffs on the exterior wall. Centered on each of the exterior walls are two pilaster support columns closed at the back to form a tower. Within each of these towers there is located mechanical equipment for the operation of the reactor facility. The pilaster support columns, comprising the side walls of the towers, are built into the structure to achieve the requisite structural strength. Below grade, and to the north, there is extended a room of 15 feet height by 37 feet in depth and 40 feet in width which is within the containment structure (Figure 3.1). The walls and ceiling of this room are also of poured concrete with the pour joints sealed with the rubber stop.

1973-1974
1967-1968

3.2 Penetrations and Closure

Penetrations through the concrete structure are limited to sealed electrical inlets, pedestrian and truck entry doors, inlet and exhaust air ducts, and a water seal containing various piped utilities.

3.2.1 Utility Entry Water Seal

The water, steam, gas and air pipes are brought in through a 4.6 foot deep water-filled trap. This water trap forms a seal between the containment building and the adjacent laboratory structure. This water trap functions as a seal to a 2.0 psi overpressure within the building. In the event of overpressure exceeding 2.0 psi the water is elevated out of the water seal partially flooding the floor in the adjacent room. ~~and then flowing back into the water seal to reclose the building after the release of the overpressure.~~ DELETE 1967-1968

The water in this trap then serves as a pressure release valve in either direction. It may also serve the purpose of acting as an absorbent of any radioactive particulate or offgas in the event of a major accident.

~~The water level in the trap is automatically~~
"The water level in the trap is maintained by manual addition as required to retain the required level."
~~periodic water additions as required to maintain constant level.~~ 1973-1974 There is also an annunciator signal to the control room in the event of low water level in the trap.

1996
1981-
1982

The services which enter (or leave) the containment building through the water trap are the following:

1. 6 inch fire protection water line (dry).
2. "6" inch emergency pool fill line. 1981-1982
3. 1 1/2 inch gas line.
4. 2 inch compressed air line.
5. 2 inch domestic cold water line.
6. 2 inch domestic hot water line.
7. 3/4 inch domestic hot water return line.
8. 3 inch pumped hot sewage line.
9. 1/2 inch demineralized water line.

→ ADD TO LIST

10. 3/8" copper drain to sewer for Beamport B
11. 1-1/4" vacuum line
12. 3/4" pvc Film irradiator helium supply
13. 3/4" pvc Alternate air supply line to exhaust plenum backup doors from emergency air compressor.
14. 3/4" pvc (blanked)

3.2.2 Electrical Entry

All electrical connections from components external to the reactor containment building and to the control room of the reactor, as well as the electrical power supply, ~~pass through a steel plate of 4x4~~ ". . . pass through two penetration plates located in the containment structure wall." Each electrical line is brought through a sealed connector to minimize air leakage from the containment structure. There are no through-the-containment-wall electrical conduits. 1973-1974

3.2.3 Pedestrian Entry

The pedestrian entry at second level of the reactor building (Figure 3.2) consists of a set of electric power driven doors and an intervening vestibule. The vestibule is a portion of the containment system.

"Each of the two doors is electrically driven closed and when in the closed position contacts **in a cam actuated microswitch activate** an air valve to inflate a gasket which seals against the door." 1996

~~which seals against the door.~~ Each door is driven open or closed by means of an electric motor operating through a gear reducer box. A chain drive connected from the door over a sprocket on the gear reducer box actually moves the door. The gear reducer box is equipped with a manually operated throw-out clutch so the drive motor may be disengaged and the door operated by hand (at zero differential pressure). The doors are of steel. They are suspended by adjustable trolleys from an overhead rail. They are designed to withstand a 2 psi overpressure.

The control system for the pedestrian entry is designed and interlocked to assure that one door is always sealed.

3.2.4 Equipment Entry Door

Another penetration through the containment vessel consists of a heavy equipment entry door located below grade (Figure 3.1) and serviced by the 15 ton equipment elevator in the laboratory structure. This heavy equipment entry is closed by a steel door which rides in a track which guides the door up against an inflatable gasket. The door is electrically driven into the closed or open position. When in the closed position the gasket is inflated and maintained under a low air pressure to insure a continuous seal around the door. This door is maintained in a closed position at all times during the reactor operation.

1995
1981-
1982

3.2.5 Supply and Exhaust Air Doors

On the fifth level of the reactor building (Figure 3.3) there are two doors, one for the incoming heated or conditioned air and the other for exit air. Each of these openings is approximately 4x4 feet. Each opening is closed by an electrically driven steel door riding in a track and against which an inflated gasket seals when the door is in the closed position. These doors are open except upon a closure signal from either the radiation detection monitoring system or the operator's console.

3.2.6 Pneumatic Tube System

"The pneumatic tube system components entering the reactor containment consist of eight lines, ..." ~~ering the~~ 1995

four 1 1/2 inch sample carrier tubes, AND four 1 1/2 inch air-vacuum driver lines. ~~and one [illegible] exhaust line from the system blower.~~ 1981-1982 The eight 1 1/2 inch lines, making up the sample carrier system, come through a steel plate located in the containment wall. The pneumatic tube carrier system is a closed system extending from the laboratory in the adjacent structure through the containment shell and then through the reactor shield down and adjacent to the core. There is no leakage path into the pneumatic tubes once they have penetrated through the containment shell so the system is really an extension of the laboratory into the containment. The tubes will withstand a 2 psi overpressure with zero leakage.

1996
1995
1973-
1974
1972-
1973

~~In addition to the eight pneumatic tubes running~~

"In addition to the eight pneumatic tubes running through this steel plate there are a number of pipes welded to the plate to permit **pressurization** of the building . . . ."

~~pipes welded to the plate to permit compression of the building~~ to 2 psi overpressure and to permit the attachment of the necessary instruments needed to measure internal pressure and leakage of contained air. 1996

~~The 4 inch exhaust line runs from the pneumatic system blower, located external to the containment,~~

"The four inch exhaust line runs from the pneumatic system blower, located external to containment, to the MURR off-gas stack."

~~to the steel collar. This line has a quick-closing valve immediately inside of the containment and runs from the valve to the hot exhaust line from the beam ports and the thermal column. Closure of this 4 inch exhaust line is described in the next section.~~ 1973-1974

3.2.7 Hot Exhaust Lines

~~There are two penetrations of the containment~~

"To exhaust potentially contaminated gases from the building, there is a 16 inch off-gas duct located at the fifth level of the reactor building and passing through the containment wall." 1973-1974

~~building and passing through the containment wall structure.~~ ~~This~~ **"This 16 inch pipe has a 12 inch long collar** ~~collar~~ 1995

surrounding it which is cast into the containment building wall. Adjacent to the wall, and within the containment building, there is located a 16 inch valve of the quick-closing positive-closure type provided by Henry Pratt Company. The valve has a closure time of "3.0" seconds. This valve is 1972-1973

1995
1973-
1974

equipped with its own air compressor-accumulator device and an electric motor so that in the event of power failure the accumulator contains the necessary compressed air for closing the valve.

"This valve is backed up by an identical valve on the fifth level of the containment building. This valve is equipped with an air-to-open spring-close actuator. Both valves are actuated by the building isolation system. 1973-1974

~~In addition to this 16 inch off-gas duct there is a 4 inch line running from the blower which operates the pneumatic tube system. The off-gas line from this blower passes through the containment wall and is closed within the containment by another quick-closing positive-closure air operated valve equipped with electric driven air compressor and accumulator. This valve is also actuated by either the isolation alarm or by signal from the radiation detection monitoring system.~~ DELETE 1973-1974

3.2.8 Compressed Air and Inflated Gaskets

All inflatable gaskets are dependent upon a continuous supply of compressed air for their operation. There are two air compressors installed in the facility. One is positioned in the mechanical equipment space adjacent to the reactor building and operates on a continuous basis for providing compressed air to the laboratories as well as to the reactor. ~~Located within the containment building, ... compressor. This standby air compressor normally "rides" on the main air supply line until there is a drop of pressure in the main air line to less than 40 psi. At that time~~

"Located within the containment building on the fifth level in the mechanical equipment space (Figure 3.3) is an emergency air compressor. 1995

1968 REV STATED 85 PSI

1996
1995

~~the small standby compressor comes on and a check valve closes off all of the compressed~~

PARAGRAPH

"This emergency air compressor normally "rides" on the main air supply line until there is a drop of pressure in the main air line to less than **70** psi. At that time the small emergency air compressor comes on and a check valve closes off all of the compressed air system external to the containment building, **except for backup door supply**." 1996

~~building. The compressed air supplied by~~

This emergency air compressor, under these circumstances, carries the air requirements within the containment building. The compressed air supplied by one or the other of these two separate systems is maintained in accumulator tanks located on the compressor and goes from these tanks by means of a piping system to a pressure reducing and regulation system to provide the motive force for the inflated gaskets.

~~to a pressure reducing and regulation system to provide the motive force for the inflated gaskets.~~

PARAGRAPH

"This system is equipped with check valves and automatic starters such that if there is a failure or loss of compressed air from the main unit, the emergency air system will come on and carry the inflatable gaskets and any other air controlled devices associated with reactor **isolation**." 1996

~~air from the main unit,~~

The emergency air compressor is also connected to the emergency diesel generator in the event of electrical power failure."

~~inflatable gaskets and any other air controlled devices associated with the reactor. The standby air compressor is also connected to the standby electrical generator in the event of electrical power failure.~~ *

3.3 Operational Characteristics

In normal operation of the reactor the pedestrian entry will be closed by one or the other of the doors enclosing the reactor vestibule; the equipment entry door will be closed; the supply and exhaust air doors will be open; the two offgas exhaust line valves will be open. The containment structure will be under a slight negative pressure with respect to the surrounding laboratory building.

* SECTION 3.2.8 IS DELETED BY 1995 REV AND THEN REVISED BY 1996 REV

1996
1995
1967-
1968

"The containment system integrity is compromised only when the reactor is ~~raised~~ secured and only then . . . ." to the the extent that the heavy equipment door may be opened in the event that it is necessary to transfer materials from within to without or from without to within the containment structure. At all other times the containment system will not be violated. 1995

"This heavy equipment door to the outside will only be opened when the reactor is secured." 1996

It is intended that the containment system be completely closed in the event of detection of high radiation levels within the monitoring detection network. ~~Specifically, a high radiation level~~

"Specifically, a high radiation level detected in the off-gas line to the stacks **or high radiation at the reactor bridge** will initiate a signal that will provide building closure." 1996

~~closure.~~ The reactor operator may also provide a signal for building closure by pressing the building isolation alarm located on the reactor control console.

3.4 Pressure Containment

The reactor containment building has been designed to hold a peak internal overpressure of two pounds per square inch. The building has been designed to provide a leakage rate not to exceed 10% of the contained mass of air when the building is pressurized to two pounds per square inch. 1967-1968

Prior to the initial loading of fuel into the reactor the leakage rate of the containment building will be measured to establish a leakage base line

1967-1968

reference. The results of the leak rate measurements will be forwarded prior to start-up. Building leakage will be rechecked on approximately an annual basis thereafter to assure maintenance of building integrity.

~~For the initial leak rate test two measurement techniques will be used and the results compared.~~

The initial leak rate test used the air reference system technique. Technical difficulties prevented the use of the resistance wire measurement.

~~employs a long resistance wire to measure average absolute temperature.~~

1967-1968

Figure 3.1
Basement Level

[redacted]

0 5 10 40 FT.

north:	university of missouri – columbia	scale: N.T.S.
	building: RESEARCH REACTOR	MURR No. 2269
		revised: 12-19-95
	level: BASEMENT LEVEL	sheet: 1 OF 5

1996
1996

[redacted]

an

1996

0 5 10 20 40 FT.

north:	university of missouri – columbia	scale: N.T.S.
	building: RESEARCH REACTOR	MURR drawing no.: 2269
		date: 12-19-95
	level: GRADE LEVEL	sheet: 2 OF 5

1996

Figure 3.3.a.
Third Floor Plan

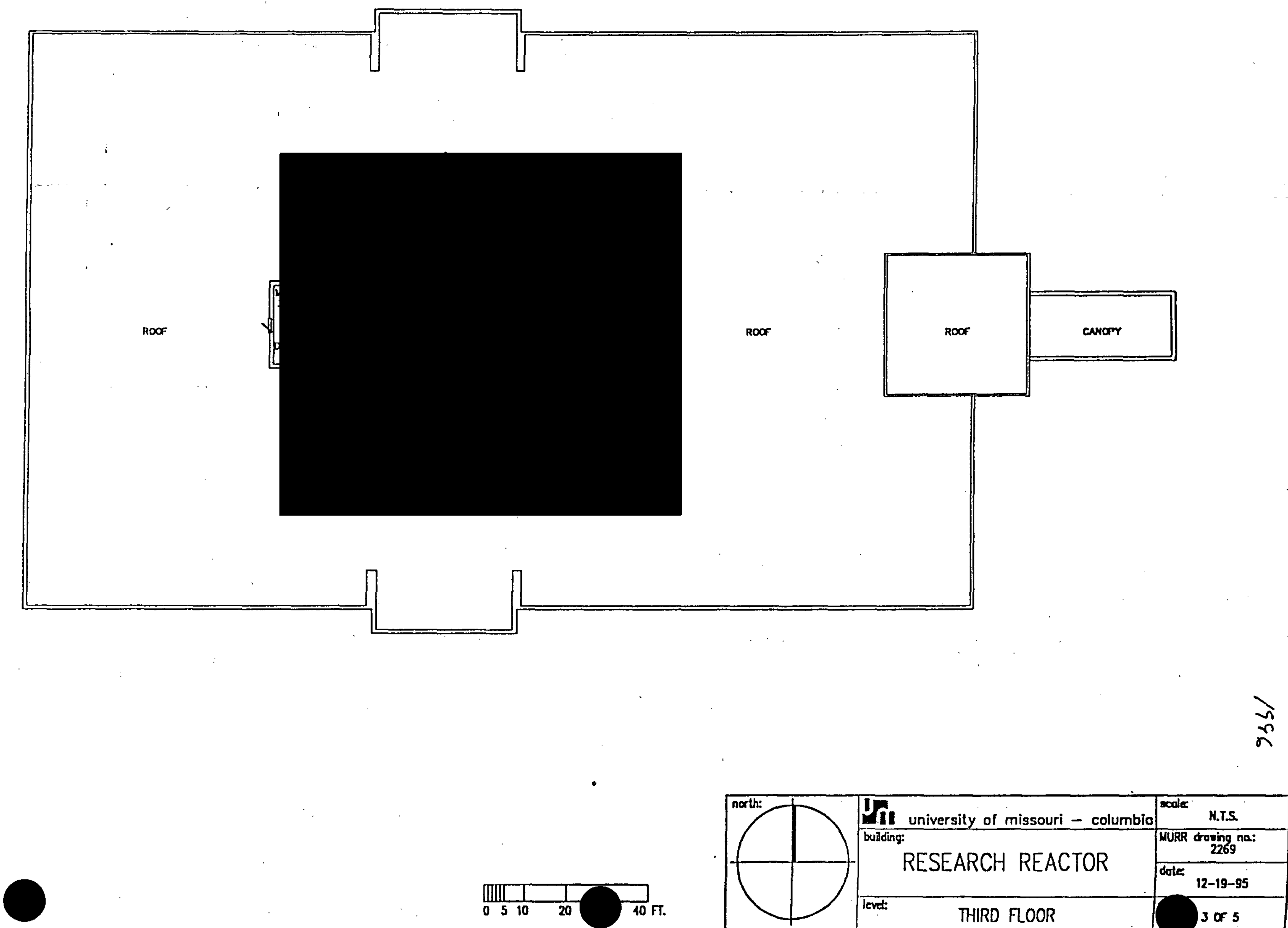
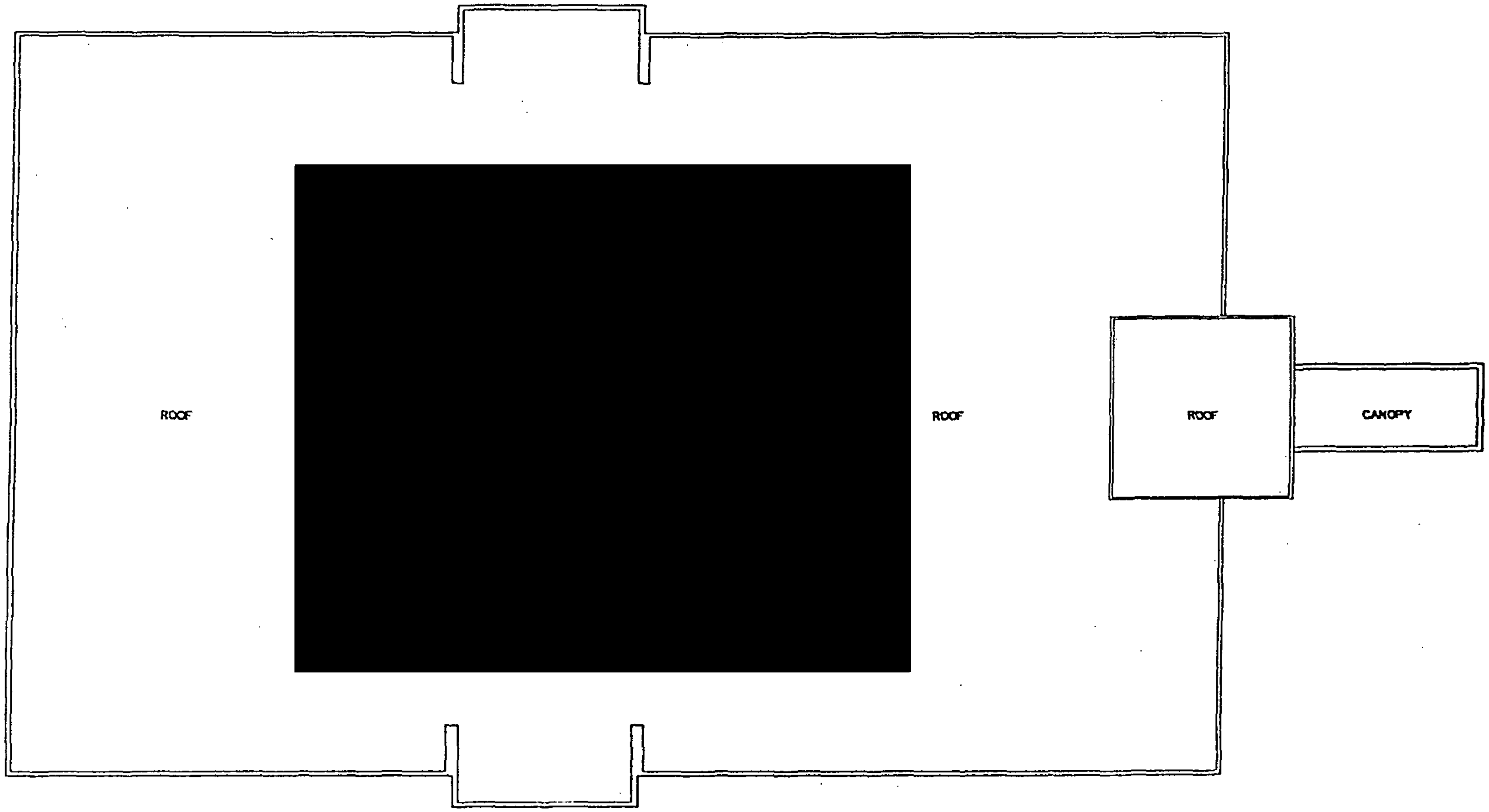


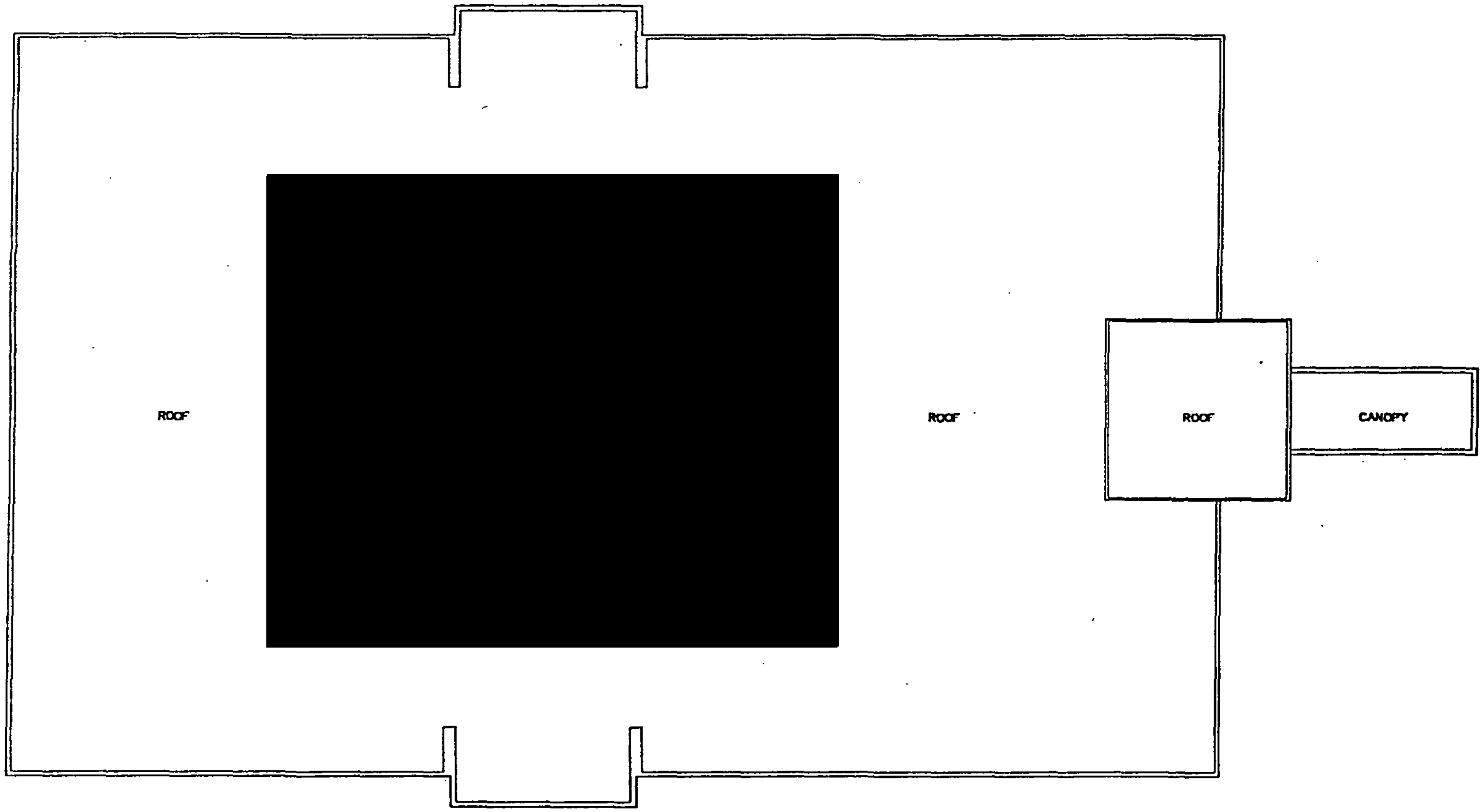
Figure 3.3.b
Fourth Floor Plan



north:	university of missouri – columbia	scale: N.T.S.
	building: RESEARCH REACTOR	MURR drawing no.: 2269
		date: 12-19-95
	level: FOURTH FLOOR	sheet: 4 OF 5

1996

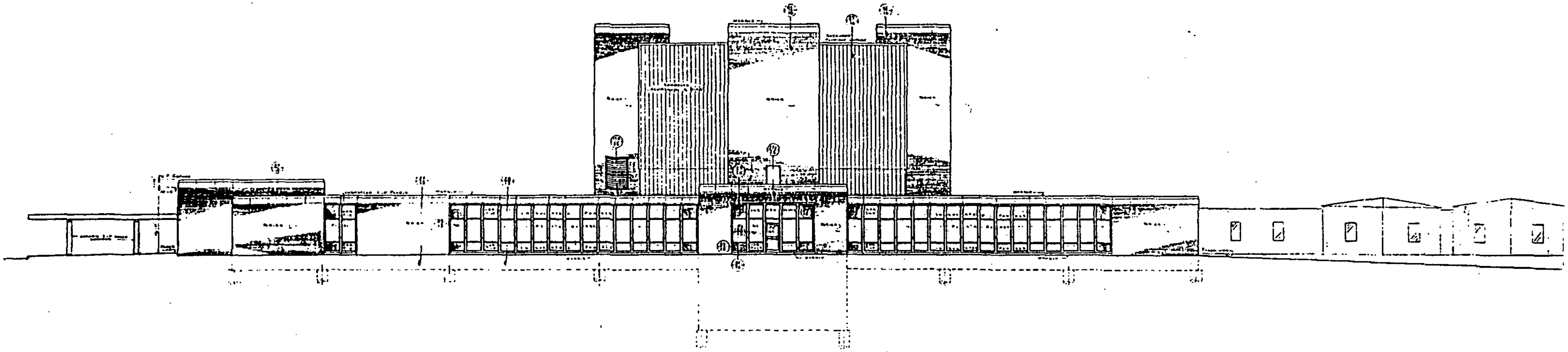
Figure 3.3.c
Fifth Floor Plan



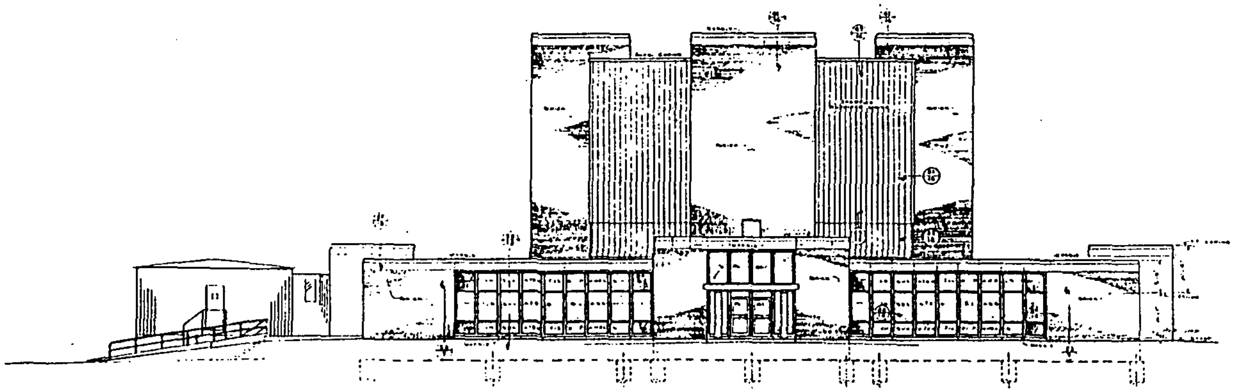
0 5 10 40 FT.

north:	university of missouri – columbia	scale: N.T.S.
	building: RESEARCH REACTOR	MURR drawing no.: 2269
		date: 12-19-95
	level: FIFTH FLOOR	5 OF 5

1996

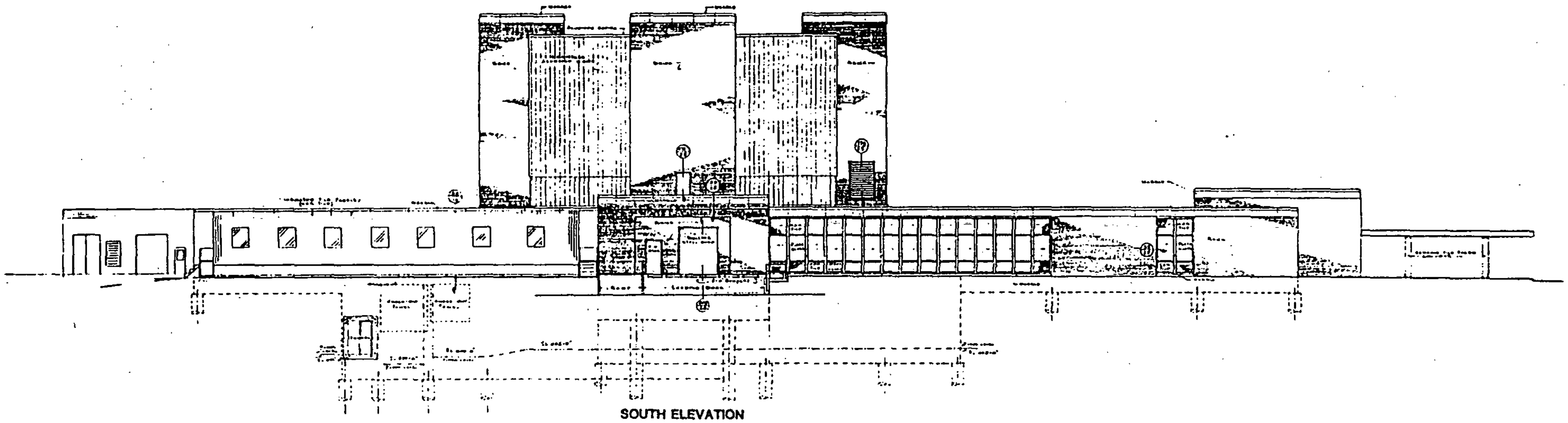


NORTH ELEVATION

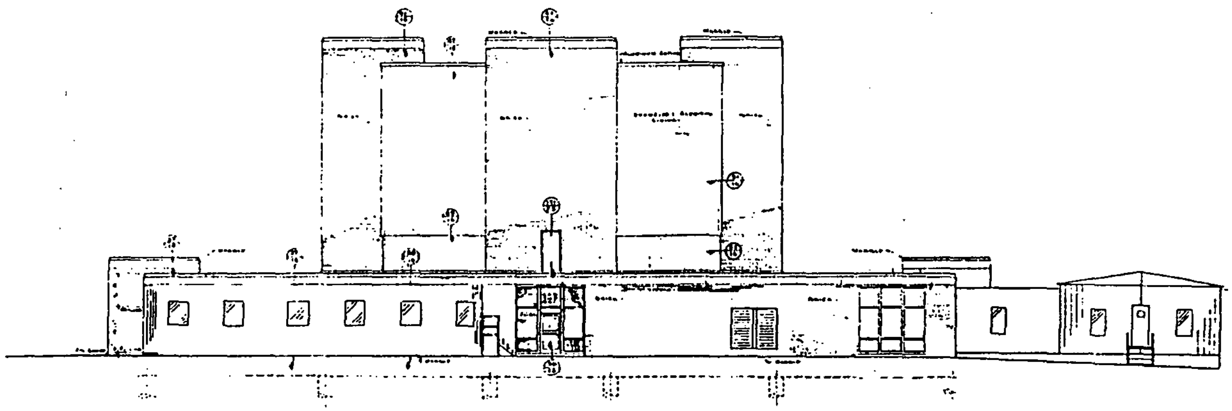


EAST ELEVATION

Figure 3.4. North and East Elevations
(Rev. 1/3/95)



SOUTH ELEVATION



WEST ELEVATION

Figure 3.5. South and West Elevations
(Rev. 1/3/95)

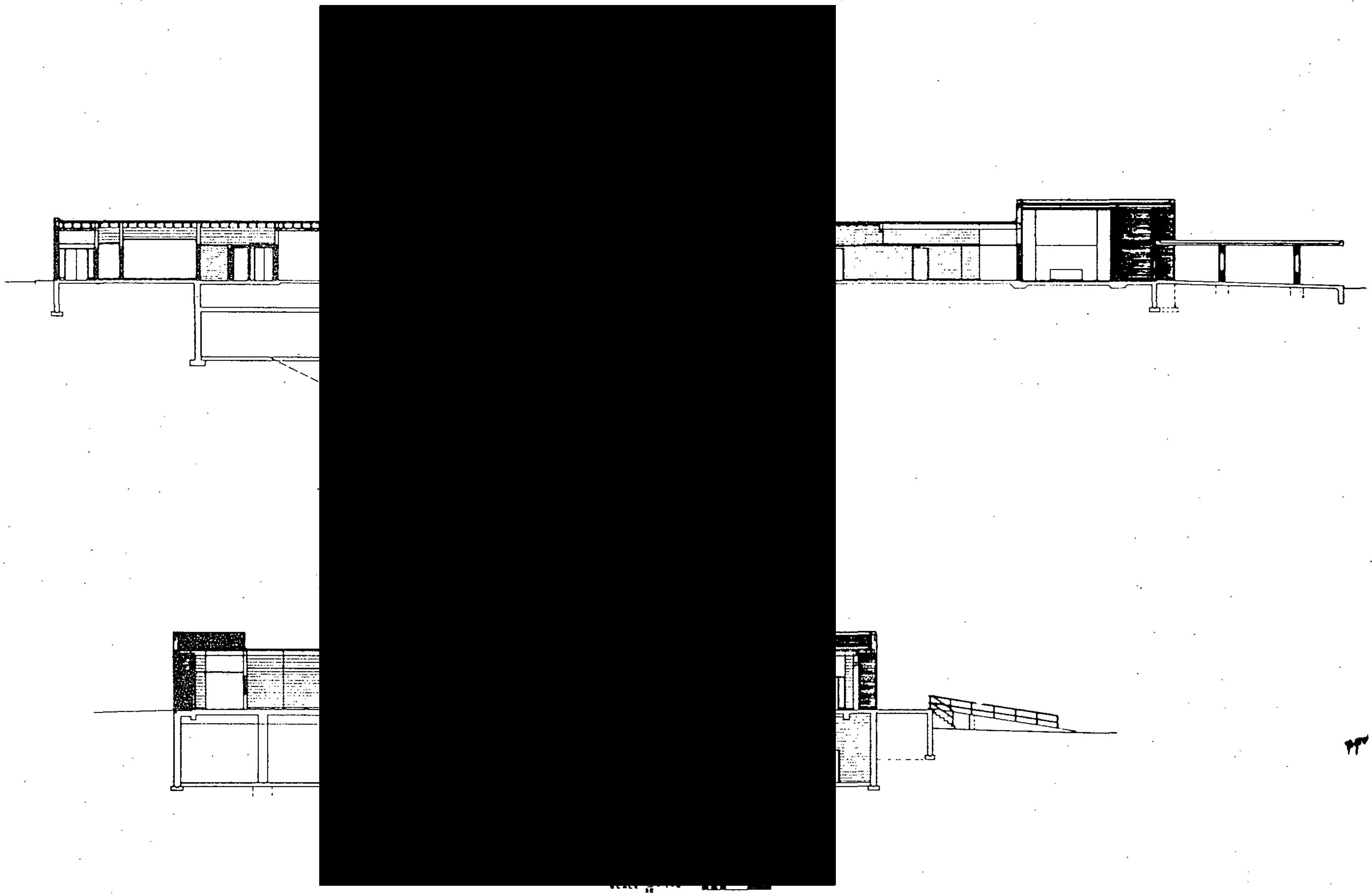


Figure 3.6 Building Sections A-A and B-B

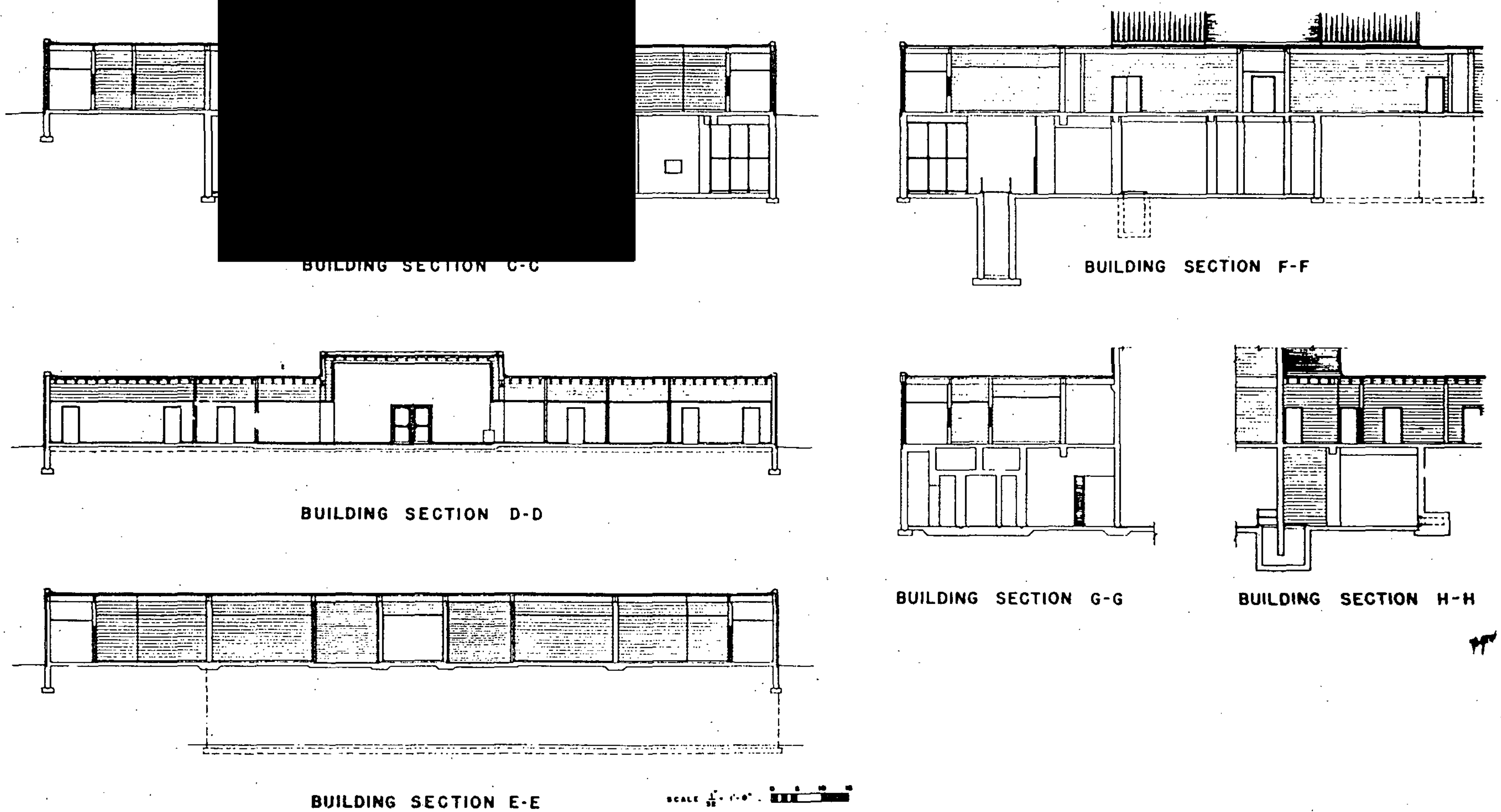


Figure 3.7 Building Sections C-C, D-D, E-E, F-F, G-G and H-H

1996

1996

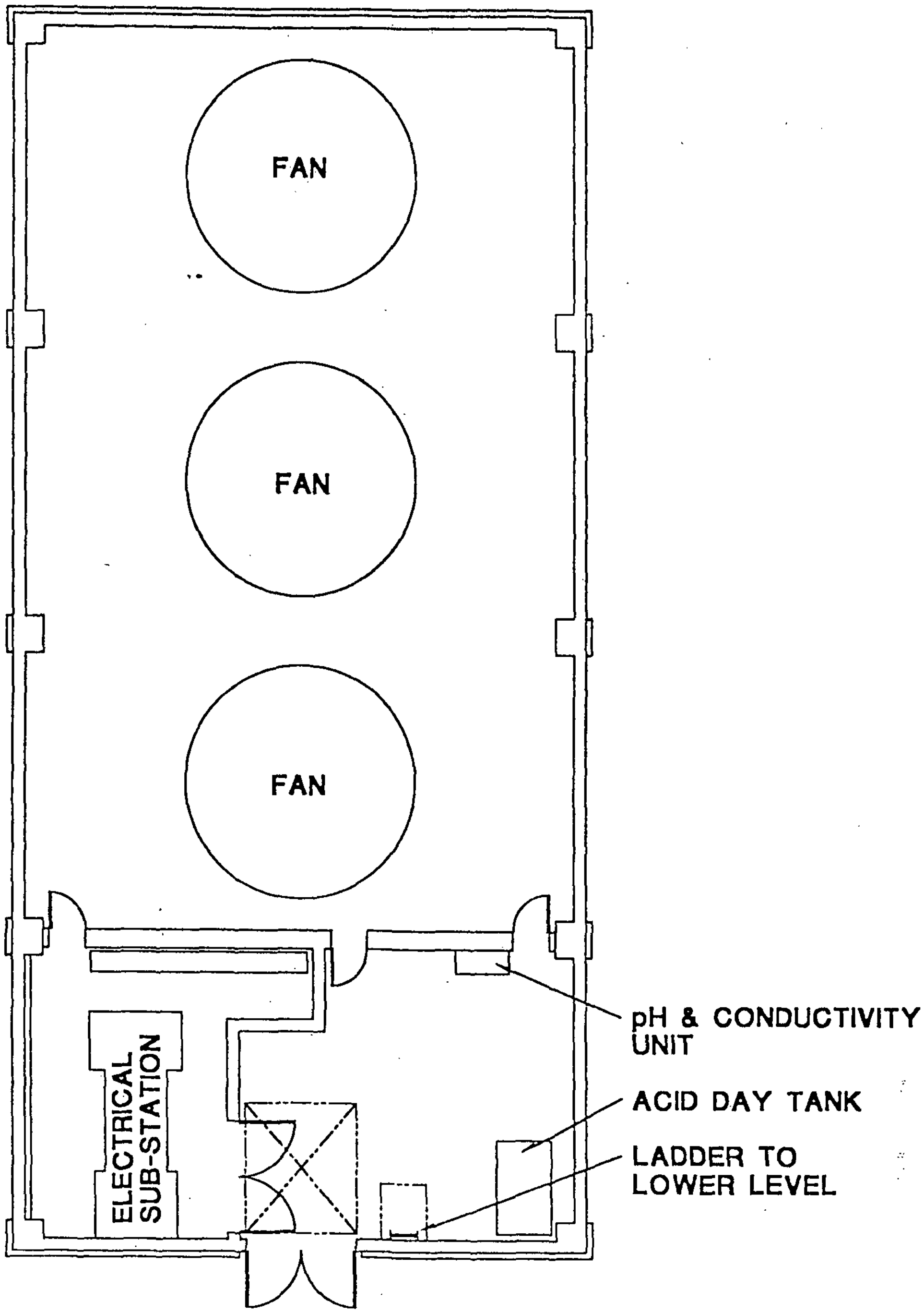


Figure 3.8

COOLING TOWER AT GRADE LEVEL - PLAN VIEW

(Rev. 5/20/96)

1989-
1990

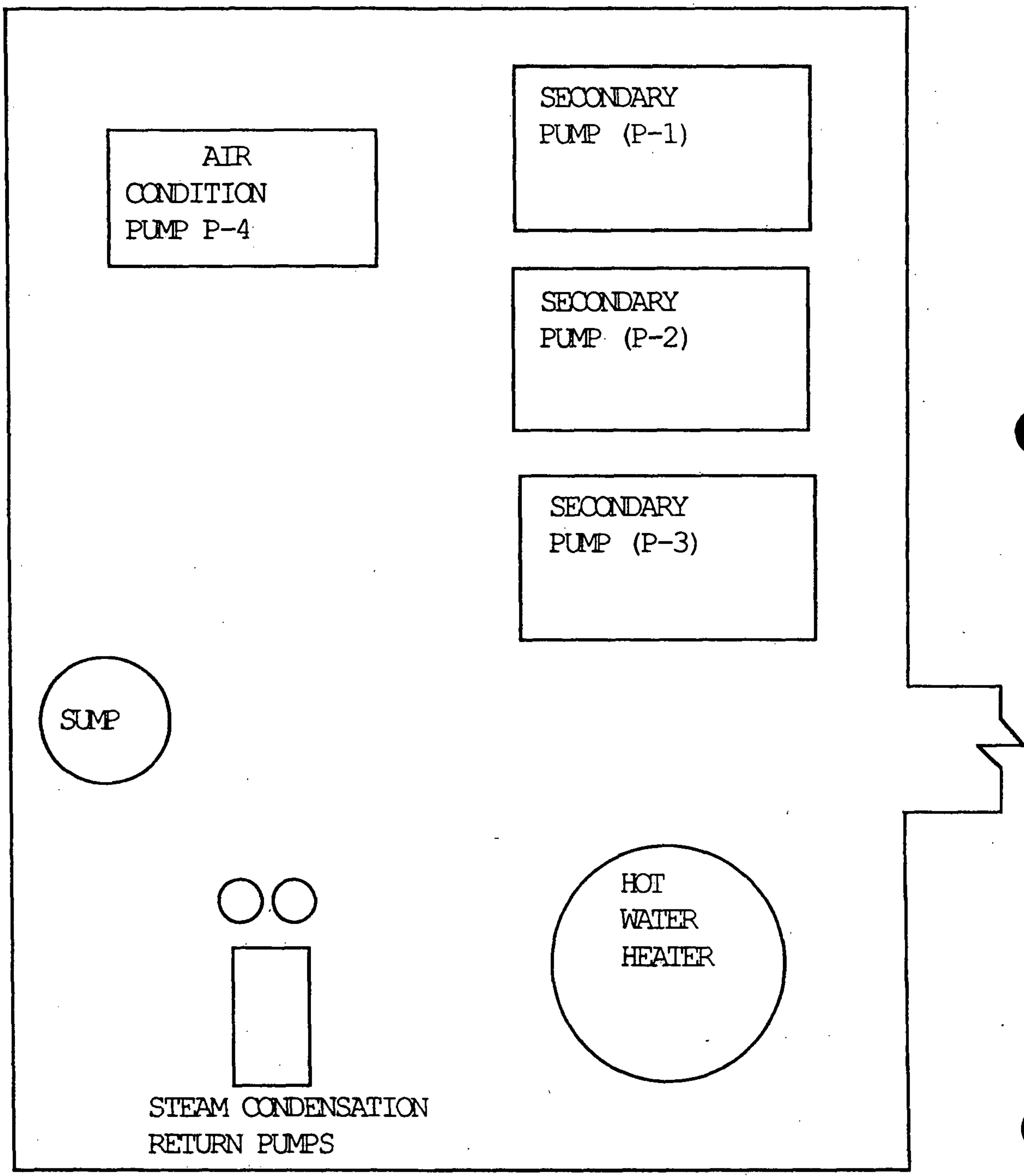


Figure 3.9 Cooling Tower Below Grade Plan

76

1995

4.0 REACTOR CORE

4.1 General Description

The major regions of the reactor core assembly are:

1. The fueled region
2. The control region
3. The reflector region

"The fueled region is an annulus of a right circular cylinder with a nominal inner radius of 2.67 inches and an outer radius of 5.88 inches." 1995

~~and an outer radius of 5.81 inches.~~ The fueled region has a fixed geometry consisting of 8 identical assemblies placed vertically around the annulus. Each assembly contains 24 circumferential plates and each plate contains uranium enriched to 93.5% in the isotope U-235 as the fuel material. Fuel assembly cladding and support material is aluminum.

The active height of the fueled region is 24.0 inches and the overall height is 32.5 inches. The fueled region is pressurized and is designed for operation at a power level of 10 megawatts.

The control region is an annular gap outside the pressure vessel at a radius of 6.275 inches. Five control elements operate vertically between the pressure vessel and the beryllium reflector, filling 85% of the gap circumference.

1995
1967-
1968

The reflector region consists of two concentric right circular annuli surrounding the control region. The inner reflector annulus is a solid ring of beryllium metal 2.71 inches thick. The outer reflector annulus is made up of vertical elements of graphite canned in aluminum and has a total thickness of 8.89 inches. The overall height of the reflector region is 36 1967-1968 inches, and it is centered vertically with the fueled region.

The entire core assembly, including the pressure vessel, the fuel element support matrix and the reflector support assembly are supported from the bottom of the pool in circular sections which also serve as part of the coolant piping. Cross sectional views of the reactor core assembly are shown in Figures 4.1 and 4.2.

4.2 Fuel Assemblies

Only one type of fuel assembly is used. A drawing of this assembly is shown in Figure 4.3.

The assemblies are longitudinally symmetrical and are 32.5 inches long, overall. ~~The overall radial dimension of an assembly is 3.28 inches.~~ " The overall radial dimension of an assembly is 3.21 inches (as determined from radial length from inside of inner roller to outside of outer roller)." Each assembly spans 45° and one core includes eight assemblies. 1995

Each assembly contains 24 fuel bearing plates. The plates are segments of concentric circles separated by a distance of 0.080 inches and they are 0.050 inches thick. ~~The radius to the innermost plate is~~ 2 "The nominal radius to the innermost plate is 2.77 inches and to the outermost plate is 5.76 inches." 1995

The fuel plates are supported along their vertical edge by appropriately slotted flat side plates. The dimensions of the side plates are 31.75 x 3.16 x 0.15 inches. A handling and guide fixture at each end holds the assembly together.

Both ends of an assembly are identical so the total core may be turned end for end thereby producing more uniform fuel burnup and longer core life.

"The fuel material is uranium-aluminum alloy or intermetallic UAl$_x$." ~~aluminum.~~ The uranium is fully enriched in the isotope U-235 ([redacted] The total impurities in the fissionable material does not exceed 1500 parts per million. The maximum elemental impurities, or their equivalent, is less than the following: 1971-1972

Element	ppm
Aluminum	100.0
Boron	1.0
Cadmium	1.0
Calcium	20.0
Lithium	2.0
Silicon	100.0

Impurities not listed above are less than the thermal neutron cross section equivalent of 4 parts per million of boron.

The aluminum for the fuel core alloy is Alcoa 99.8% pure alloy melt stock. The reactivity effect of all impurities in this alloy is less than the equivalent of 15 parts per million of boron.

The aluminum for the fuel plate cladding is ASTM B209, Alloy 1060 or, depending on hot rolling strength requirements, ASTM B209, Alloy 6061-T6. The reactivity effect of all impurities in this alloy is limited to the equivalent of 15 parts per million of boron.

The aluminum end fittings are of ASTM B108, Alloy SG70A-T6 castings or are machined from ASTM B209, Alloy 6061-T6 aluminum alloy solid bar stock. For permanent mold castings, the aluminum used is ASTM B108, Alloy SG70A-T71. Sand castings, if used, shall be ASTM B26, Alloy SG70A.

The side plate material is ASTM B209, Alloy 6061-T6 aluminum alloy. The reactivity effect of all impurities in this alloy are less than the equivalent of 15 parts per million of boron.

Each fuel plate has a thickness of 0.050 inches. The fuel material thickness is 0.020 inches and the cladding thickness is 0.015 inches.

The cooling gap between fuel plates is 0.080 inches and the inner and outer cooling gaps are 0.095 and 0.075 inches, respectively.

When installed, the fuel assemblies stand in a fuel support matrix and are held down by gravity and coolant flow. Lateral support is provided by roller assemblies which run on the inner surfaces of the pressure vessel. The fuel assembly side plates engage slots in the inner pressure vessel wall providing vertical support of the individual assemblies.

1996
1971-
1972

The type of fuel assembly being used in the University of Missouri Research Reactor, that is,

". . . curved plates of a uranium-aluminum alloy or intermetallic UAl_x." 1971-1972

~~years in training, research and test reactors.~~

No new concepts or innovations have been used in the fuel assemblies described here.

Fuel plates and assemblies are identified by individual marks. Each plate of each assembly is identified by numerals showing the specific alloy heats from which they were made and each plate is identified with the fuel assembly into which it is assembled. Each complete fuel assembly has an identifying number engraved on both side plates using block letters and numbers two inches high. The engraving is from 0.004 to 0.006 inches in depth.

Fuel plates are fastened into the side plates using a mechanical binding procedure that provides a tensile strength of greater than "150 pounds per linear inch per plate". 1971-1972

4.3 Control Elements

There are a total of five control blades used in the reactor. Four of these, the shim blades, each occupy approximately 72° of a circular arc around the pressure vessel. The fifth, the regulating blade, occupies approximately 18° of circular arc.

~~The shim blades are constructed of formed boral plate. The neutron absorbing material is 50% boron carbide and 50% aluminum, by weight. The~~ 1996

~~boron carbide-aluminum mixture is clad with .030~~

"The shim blades are constructed of formed boral plate. The neutron absorbing material is **nominally** 50% boron carbide and 50% aluminum, by weight. The boron carbide-aluminum mixture is **nominally** clad with **0.0375** inches of aluminum. The nominal blade thickness is **0.175** inches.

The sides and bottom of the blades have a **0.25** inch aluminum frame where the blade thickness is 0.25 inches."

~~aluminum frame where the blade thickness is 0.25 inches.~~

1996

The active length of neutron absorbing material is 34 inches, and the overall blade length is approximately 40 inches. The top 6 inches of the blade is a 0.25 inch thick mounting plate that is curved to the shape of the blade.

The regulating blade is formed from stainless steel.

The control rod drive mechanisms are mounted on the bridge structure over the pool surface. They are of a standard design as has been supplied by the General Electric Company for their open pool reactors. The mechanism for supporting and guiding the control blades is shown in Figure 4.4 and Figure 4.5.

4.4 Reflector

An annular ring of beryllium metal surrounds the reactor core and forms the outer wall of the control element gap. The beryllium reflector ring is shown in Figure 4.6.

Additional reflection is provided by a ring of graphite elements canned in aluminum and located as shown in Figure 4.6. The beam ports penetrate the graphite portion of the reflector.

4.5 Core and Reflector Support

The core fuel assemblies are supported within the pressure vessel on a matrix plate arranged as shown in Figure 4.1.

The reflector assembly is supported on a machined matrix plate which is in turn supported in an assembly that forms the reflector coolant water outlet plenum. The reflector supporting assembly is shown in Figure 4.1.

4.6 Core Nuclear Characteristics

The design of the University of Missouri Research Reactor core and its resultant nuclear characteristics arose from two major design objectives.

1. To provide enough cold-clean reactivity to enable 400 MWD of core life with either continuous operation at design power or intermittent operation on an eight-hour on, sixteen-hour off, and weekend shutdown operating cycle. Additionally, there must be a reasonable amount of reactivity available for experiments (approximately 2.5% $\Delta k/k$) at the end of core life.
2. To provide enough shim rod worth to enable shutdown with one "stuck rod" at any time and to provide reasonable assurance that the rods may be withdrawn far enough to allow continuous operation of all beam tubes without significant local flux depression by the rods.

The nuclear characteristics of the core design that satisfies the above two objectives are given in Table 4.1. Data are given for both the core design to deliver 400 MWD at a maximum power of 5 megawatts and similarly for 10 megawatts.

The calculated core average neutron fluxes for operation at 10 megawatts are given for four energy groups in Table 4.2.

1995
1981-
1982
1974-
1975
1971-
1972
1972-
1973

TABLE 4.1

REACTOR AND MAJOR NEUTRON PHYSICS DATA SUMMARY

Reactor Type	Flux trap pool type, fully enriched uranium, light water moderated and cooled, reflected by beryllium, graphite and light water.
Nominal Power	5 megawatts max. (initial core). 10 megawatts max. (later cores).
Active Core Volume	33.0 liters
Average Power Density	0.151 MW/liter at 5 MW 0.303 MW/liter at 10 MW 1981-1982
Average Specific Power	**0.962 MW/kg (235) at 5 MW** **1.613 MW/kg (235) at 10 MW** 1995
Initial Operating Cycle	8.0 hours on, 16.0 hours off, weekend shutdown.
~~Core lifetime, UAl_x~~	1200 MWD 1974-1975
Initial Fuel Loadings	**5.2 kg (235) for 5 MW cores** **6.2 kg (235) for 10 MW cores**
Uniform Critical Mass	**App. 2.6 kg (235)** 1995
Cold clean K_{eff}; 6.2 kg (235)/core	1.0795 1972-1973
Power Density; max/ave	
a) Uniform loading 3.306	~~3.44~~
b) Nonuniform loading 3.676	~~3.68~~ 1972-1973
Control Rod Characteristics	
Total Worth of Four Shim Rods	0.1655 Δk
Total Worth of Regulating Rod	0.0023 Δk
Total Worth With One Stuck Rod	0.10923 Δk
Max. Worth Per Rod (One Rod In)	0.0425 Δk 1971-1972
Max k_{eff} With One Stuck Rod, 6.2 kg (235) Core	**0.938 Δk** ~~0.918 Δk~~ 1995

1995
1981-1982
1974-1975

TABLE 4.1 (cont'd)

Xenon Worth, 5.2 kg (235) Core (1995)	
Equilibrium	-0.0297 Δk/k
After 16 Hour Shutdown	-0.0329 Δk/k
Xenon Worth, 6.2 kg (235) Core (1995)	
Equilibrium	-0.02730 ΔK/K" (1981-1982)
After 16 Hour Shutdown	-0.0607 Δk/k
Peak to equilibrium xenon ratio (1972-1973)	2.22 (1981-1982)
Samarium Worth	
Equilibrium	-0.008275 ΔK/K" (1981-1982)
Weekday Maximum	
a) 5 MW op.	-0.0079 Δk/k
b) 10 MW op.	-0.0099 Δk/k
Fuel Burnup; 1200 MWD operation (1974-1975)	1.27 Kg U-235 (1974-1975)
Worth of Burned Fuel and Fission Products	-0.0324 Δk/k
Maximum Worth of Experiments	-0.025 Δk/k
Total Cold to Hot Operating Temperature Reactivity Change	-0.0048 Δk/k
Prompt Neutron Lifetime	5.7×10^{-5} sec.
Effective Delayed Neutron Fraction (Assumes Number Fraction = 0.006544)	0.00738
Thermal Fission Fraction	0.874
Mass Coefficient of Reactivity: (Δk/k) / Δm/m) (1995)	
5.2 kg (235)	**0.1014**
6.2 kg (235)	**0.0736**
Temperature Coefficients (1972-1973)	
Core: 85°F to 145°F	-7.0×10^{-5} ΔK/°F (1972-1973)
Pool: 85°F to 115°F	$+1.34 \times 10^{-4}$ ΔK/°F
Void Coefficients	
Core	-2.51×10^{-3} ΔK/% void (1972-1973)
Flux Trap (1972-1973)	$+.865 \times 10^{-5}$ ΔK/cc void
Reactivity Increase Upon Flooding One Beam Tube	~0.001 (Δk/k)/tube

TABLE 4.2

CALCULATE CORE AVERAGE AND CENTER TEST HOLE NEUTRON FLUX AT 10 MEGAWATTS OPERATION

"Energy Group Structure for Four Groups" 1981-1982

Energy Range	Neutron Flux Core Average	Neutron Flux Center Test Hole Max.
Thermal	4.6×10^{13} nv	6.2×10^{14} nv
0.625 to 5530.0 ev	9.3×10^{13} nv	1.2×10^{14} nv
5.53 to 821 Kev	1.1×10^{14} nv	1.1×10^{14} nv
0.821 to 10.0 Mev	1.1×10^{14} nv	1.1×10^{14} nv

Fission Spectrum	Group
--	4
--	3
.25	2
.75	1

1981-1982

ADD TO LEFT SIDE OF TABLE

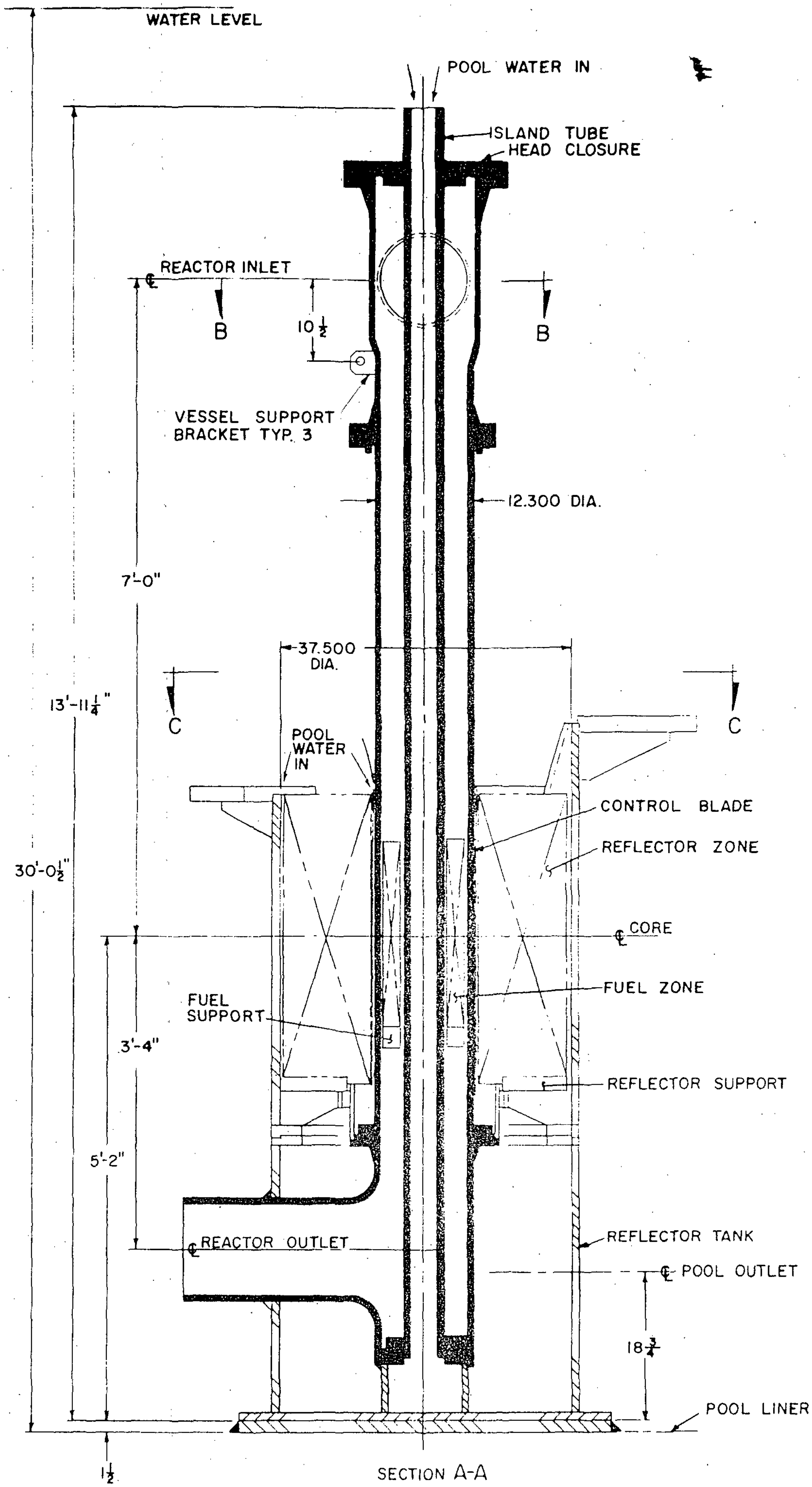


Figure 4.1 Reactor Assembly Section in Elevation View

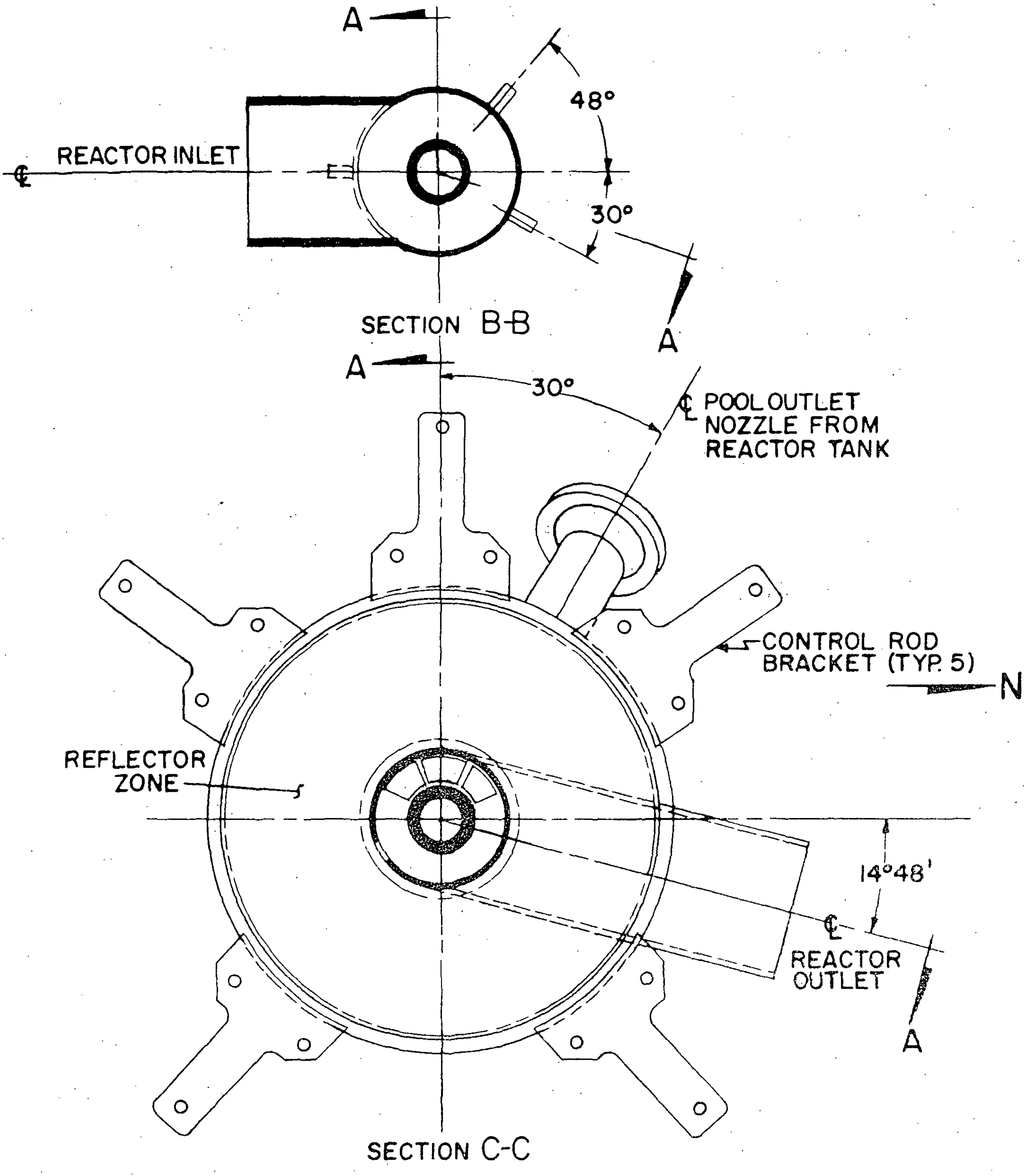


Figure 4.2 Reactor Assembly Sections in Plan View

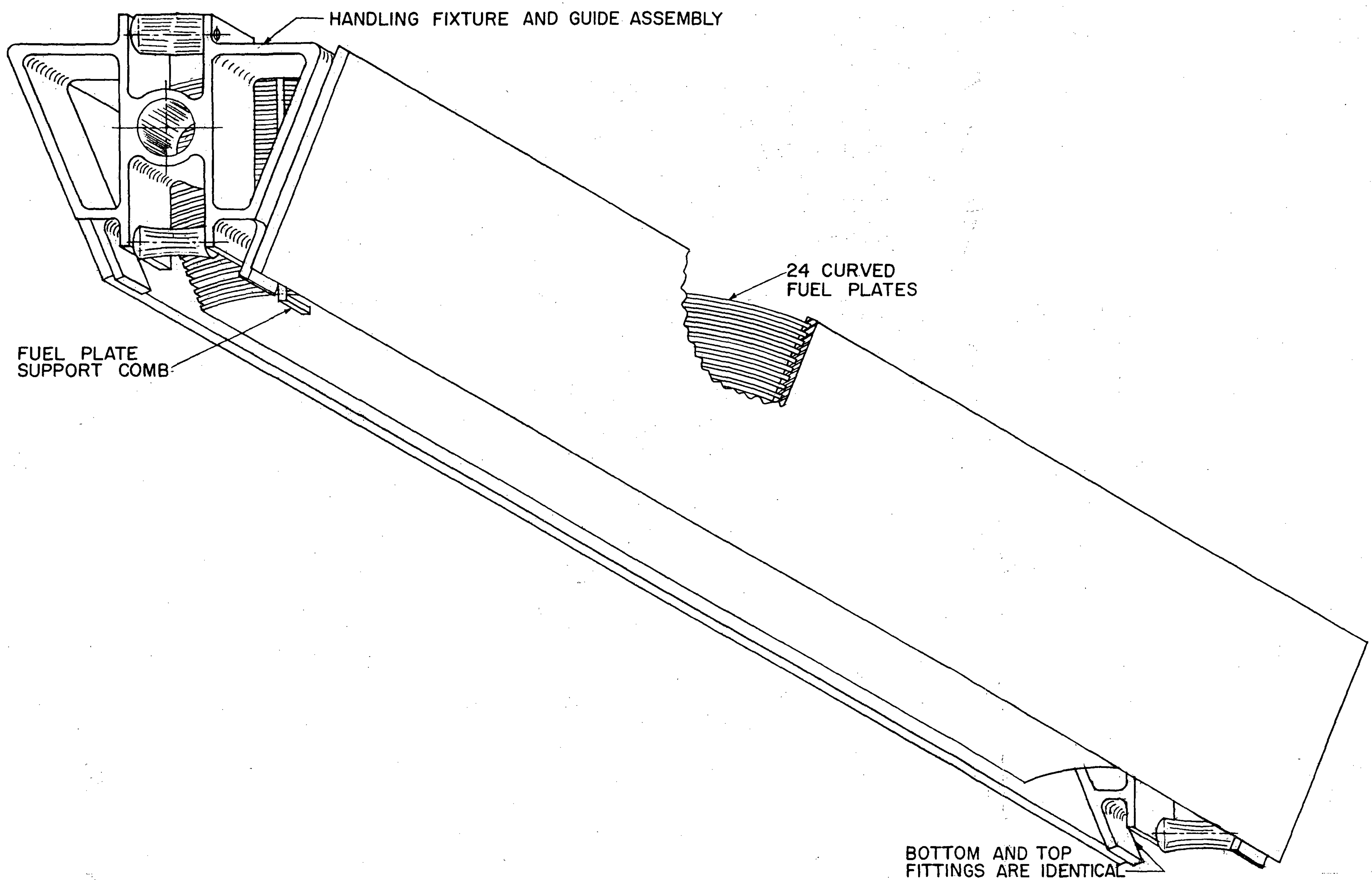
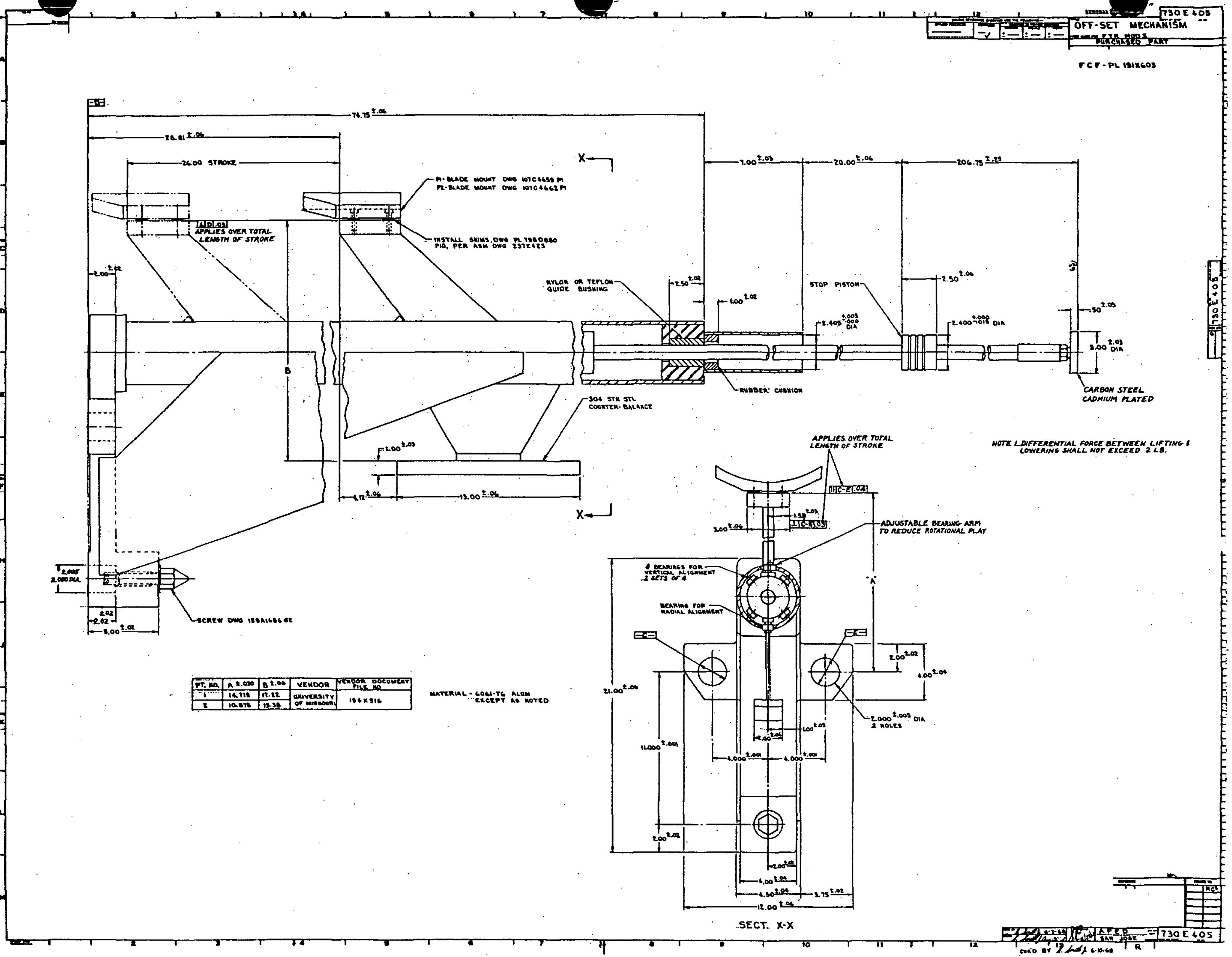


FIGURE 4.3 FUEL ASSEMBLY-PICTORIAL VIEW



PT. NO.	A ±.030	B ±.06	VENDOR	VENDOR DOCUMENT FILE NO
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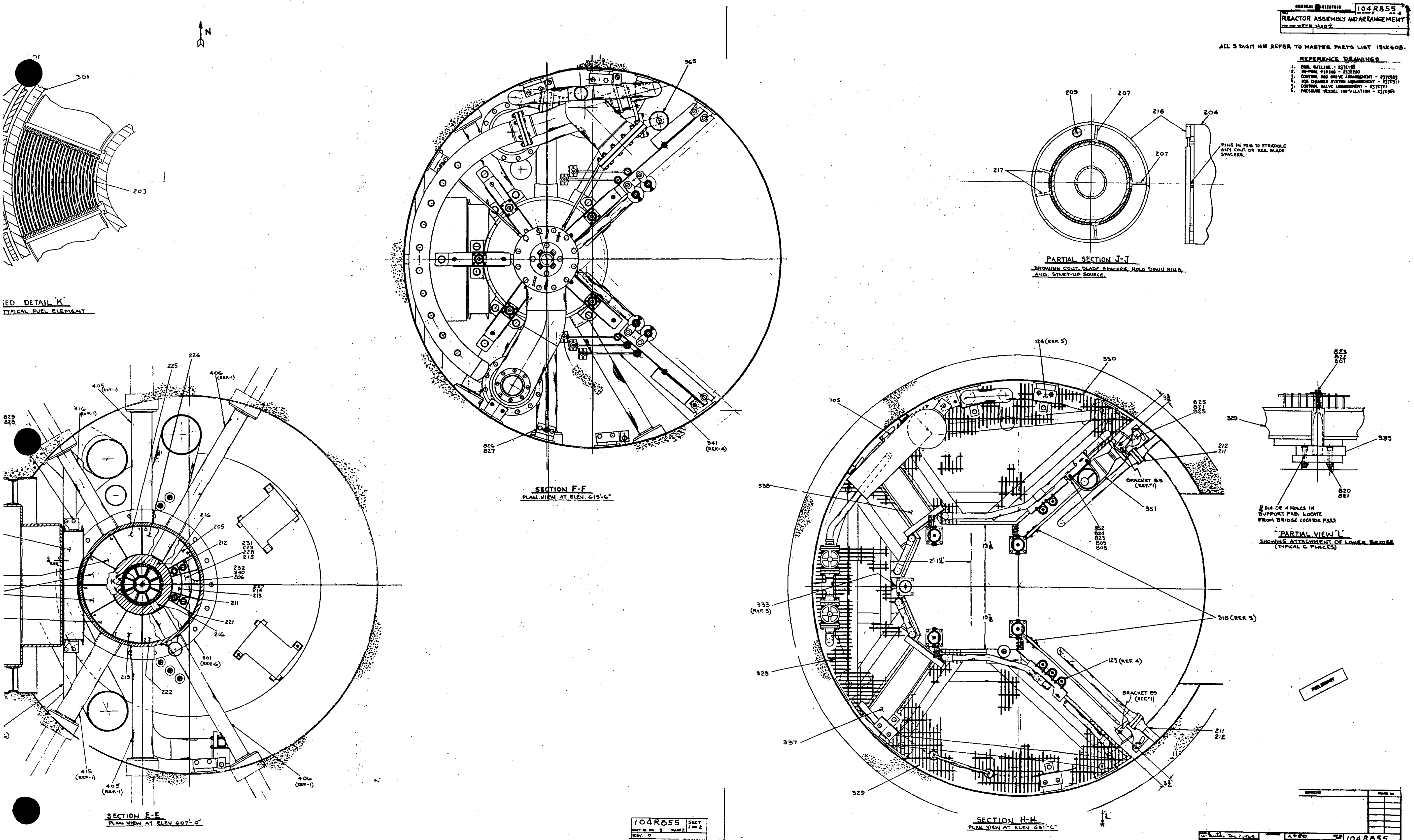
2	10.876	13.38		

C-1

1967-1968



104R855 Sheet 3 Figure 4.6.

1973-
1974

5.0 COOLING SYSTEMS

5.1 Introduction

The University of Missouri Research Reactor is designed for three modes of operation:

(1) Open pool type operation with vessel head off and flanged port open and the pool level at the elevation of either bridge. This mode of operation will be used in conjunction with core flux calibrations following the loading of a new core or a fuel rearrangement.

(2) At power levels of up to 5 MW with the reactor loop pressurized and at a flow rate of 1800 gpm and a pool loop flow rate of 550 to 750 gpm. This mode of operation will be used when only half the design heat exchange and pumping capacity is installed as will be the case at the time of initial startup. 1973-1974

(3) At power levels of up to 10 MW with the reactor loop pressurized and at a flow rate of 3600 gpm and a pool loop flow rate of 1200 to 1400 gpm. This is design maximum power operation and will be used when all heat exchange and pump capacity has been installed. 1973-1974

The reactor plant cooling systems and instrumentation which accomodates these three operating modes is shown schematically in Figure 5.1.

5.2 Reactor Loop Cooling System

The reactor coolant loop consists of the reactor pressure vessel, an in-pool invert loop and siphon break system, a closed in-pool convective cooling system, reactor isolation valves, main circulating

1994
1973-
1974

pumps and heat exchangers, pressurizer, and a bypass loop for water clean-up. The water clean-up bypass loop is described in Section 7.1.9.

5.2.1 General Operating Conditions

When the reactor is operated in the open pool mode the maximum power is limited to "50 kW" and core cooling is by natural convection to the pool. In this operating mode bulk pool temperature is controlled by operating the pool loop cooling system. 1994

The initially installed cooling equipment supplies water at 1800 gpm to the reactor with inlet conditions of 65 psia pressure and 140°F temperature when the reactor is operated at the maximum thermal power of 5.0 MW.

~~When the cooling system installation is completed the~~

"With the 10 MW cooling system installed, the reactor loop will supply water at 3600 gpm with a pressurizer pressure of 75 psia and 140°F inlet temperature when the reactor is operating at a power level of 10 MW." 1973-1974

~~level of 10 MW.~~

5.2.2 The Pressurizer System

~~The pressurizer system maintains sufficient pressure~~

"The reactor pressurizer pressure is maintained at 75 psia for both 1800 gpm and 3600 gpm operation. Pressure is maintained in the pressurizer by nitrogen gas which is automatically admitted or released in response to system instrumentation signals." 1973-1974

~~Pressure is maintained in the pressurizer by nitrogen gas which is automatically admitted or released in response to system instrumentation signals.~~

Pressurizer water level is maintained by a ~~continuously operating~~ [illegible] positive displacement pump which is valved 1973-1974

2006 - 1994 - 1973 - 1974

into the pressurizer tank on demand for more water.

~~W~~ "Water is automatically discharged into the drain collection tank when the ~~e~~ pressurizer level becomes too high." ~~s~~ 1994

5.2.3 Heat Exchanger

~~The heat exchangers are water to water shell and tube type with removable tube bundles. The tubes, and all materials in contact with reactor coolant water, are of~~ ~~on the tub~~ ~~in exc~~ ~~ble of~~ ~~gpm~~

"The heat exchangers are plate-type with all surfaces in contact with the reactor coolant constructed of stainless steel. One heat exchanger is capable of removing 16.9 x 10^6 BTU/hr of heat from 1800 gpm of coolant and returning it at approximately 140 °F, using a maximum of 1600 gpm of secondary coolant flow at 87 °F. Two such heat exchangers are installed for design power operation."

~~of coolant water and returning it at 140°F, using a maximum of 1600 gpm of secondary water flow with its inlet water temperature at 87°F. Two such heat exchangers are installed for design power operation.~~ 2006

5.2.4 Pumps

The reactor coolant circulation pumps are horizontal, centrifugal, single stage type, and are direct-connected to the driving unit through flexible couplings. The pump and driving unit are mounted on a common base. One installed pump will furnish 1800 gpm and two installed pumps will furnish 3600 gpm.

5.2.5 Siphon Break System

In the event of a rupture in the reactor loop piping external to the pool the siphon break system will operate to prevent the loss of water from the core.

"When the siphon break system operates, pressurized air ~~ic~~ is admitted . . ~~d~~ to the reactor loop at the highest point in the invert loop, thus preventing the core 1973-1974

1973-
1974

from being drained by a siphoning action.

~~The siphon break system connects into the reactor piping system 20 inches above the centerline of the top of the invert loop and extends to just above the pool surface. The siphon break ventilation line is a 4 inch I.D. schedule 40 pipe. It contains two siphon break valves arranged in parallel for re-~~

"The modified antisiphon system is described in Appendix B of Addendum 4 to the Hazard Summary Report." 1973-1974

~~During operation of the reactor with the system pressurized the siphon break ventilation line is empty. The siphon break valves cannot be closed or the reactor operated until the ventilation line has been drained to below the siphon break valve elevation. Once closed, leakage of water past the siphon break valves causes an alarm.~~

5.2.6 Isolation Valves

Quick acting automatic isolation valves are located in the inlet and outlet reactor coolant lines as close as practicable to the biological shield in the pipe trench beneath the reactor building operating floor.

"If reactor pressure decreases to a critical level ~~ecrease to a critical level~~ the isolation valves close, isolating the in-pool portions of the reactor coolant system from the balance of the system. The same signals that actuate the isolation valves also de-energize the primary coolant pumps and activate the reactor in-pool convective loop. 1973-1974

1995
1973-
1974

5.2.7 Reactor Convective Cooling Loop

"The reactor convective loop consists of two, parallel, redundant valves, the in-pool heat exchanger, ~~ol heat exchanger~~ and the necessary piping and headers. 1973-1974

The convective loop serves to remove core decay heat following an emergency shut-down accompanied by reactor loop isolation or in the event of loss of normal coolant flow. It is designed for a duty using the assumption that prior to scram the reactor had been operating continuously for 30 days at 10 MW and with an inlet temperature of 140°F and the pool temperature at 100°F. Under these conditions, and with loss of reactor loop pressure, the convective loop will transfer the decay heat from the core to the pool with no net formation of steam in the loop.

The in-pool heat exchanger consists of 10 vertical finned tubes and associated headers and piping. Each finned tube has an internal diameter of approximately 1.71 inches and has 14 internal and 28 external fins.

The convective cooling system is shown on Figures 4.4 and 4.5.

5.3 Pool Loop Cooling System

The pool loop cooling system includes: The reflector plenum natural convection valve, an automatic isolation valve, a hold-up tank, ~~main circulating~~ 1995

". . . main circulating pumps, a plate type heat exchanger, a pool return diffuser . . . ." and associated piping and valves.

1995
1973-
1974

~~The basic design conditions for the pool loop cooling system are that it will transfer 2 x 10^6 Btu/hr from a flow rate of 600 gpm when the secondary water is delivered to the heat exchanger at a temperature of 87°F and at a flow rate of 400 gpm.~~

"**The basic design conditions for the pool loop cooling system are that it will transfer 3.6 x 10^6 BTU/hr and the flow rate is 1200 gpm. This allows for a maximum reactor thermal power of 10 MW. One heat exchanger has been installed for design power operation.**" 1995

~~When the complete system is installed the heat transfer rate will be approximately 3.7 x 10^6 Btu/hr. and the flow rate will be 1200 gpm. This provides for a maximum thermal power of 10 MW. Two heat exchangers are installed for design power operation.~~

Pool loop coolant flow rates are adjustable to provide optimum cooling rates for various possible experimental arrangements in the reflector.

5.3.1 Plenum Natural Convection Valve

~~The reflector plenum natural convection valve~~

"The reflector plenum natural convection valve, under the original design, opened to permit natural circulation through the reflector and flux trap regions upon loss of forced pool loop flow. Operational tests have shown that this valve can be left open while increasing normal pool flow to provide sufficient reflector cooling." 1973-1974

~~regions.~~

5.3.2 Pool Loop Isolation Valves

A check valve in the pool inlet line and an automatic isolation valve in the pool outlet line prevent loss of pool water in the event of a break in the pool coolant lines.

1995
1973-
1974

5.3.3 Pool Water Hold-up Tank

A pool water hold-up tank provides sufficient time for oxygen and nitrogen activity to have decayed before coolant water is returned to the pool. The tank is sized to provide a five minute delay when coolant is flowing at a rate of 1200 gpm.

5.3.4 Pool Coolant Circulation Pumps 1973-1974

The pool coolant is circulated by horizontal, centrifugal, single-stage pumps, direct connected to the driving unit through flexible couplings.

~~The pumps are sized to overcome system pressure~~

"The pumps are sized to overcome system pressure drops at variable flow rates to a maximum flow of 1075 gpm when one pump is used and 1325 gpm when two pumps are used."

~~when two pumps are used.~~ 1995

5.3.5 ~~Heat Exchangers~~

~~One heat exchanger is required to service the pool coolant system when the reactor is operated at a~~

"<u>Heat Exchanger</u>

One heat exchanger is provided to service the pool coolant system when the reactor is operated at a maximum thermal power level of 10 megawatts operation.

The heat exchanger is a water to water plate type with all surfaces in contact with pool water being constructed of stainless steel. The heat exchanger is capable of removing 3.6 x 10^6 BTU/hr of heat at a flow rate of 1200 gpm of pool water, returning at a temperature of approximately 99°F when secondary water is entering at 87°F and 500 gpm."

~~The tubes and all surfaces in contact with pool water are stainless steel. The demineralized pool water flows on the tube side of the exchanger and secondary coolant water flows on the shell side. The exchanger is capable of removing 2 x 10^6 Btu/hr of heat at a maximum flow of 600 gpm of pool water, returning at a temperature of approximately 99°F when secondary water is entering at 87°F and 400 gpm.~~ 1995

1996
1994
1973-
2006 1974

5.3.6 Pool Return Diffuser

Cooled pool water is returned to the pool through a diffuser spool which serves to provide good mixing and prevents flow directly to the pool surface.

5.4 Secondary Cooling System

5.4.1 Introduction

The heat from the primary and pool reactor system is transferred to the secondary cooling system by means of the heat exchangers, the heat is then dissipated to the atmosphere through the cooling tower. When in operation the secondary system also provides a heat sink for the laboratory air conditioning system.

~~Components comprising the secondary cooling system~~

"Components comprising the secondary cooling system include: the pool heat exchanger, the primary heat exchanger and temperature instrumentation, radiation monitoring instrumentation, chemical addition equipment, primary and pool heat exchanger automatic control valves, ..."

~~instrumentation, chemical addition equipment, primary and pool automatic control valves,~~ and other associated piping and valves. 1994

5.4.2 Heat Exchangers

~~The sec~~ ~~"The secondary w~~ ~~and on the oppo~~ ~~exchanger"~~ ~~5.3.5.~~

"The secondary coolant flows on the opposite plate side of the primary coolant in the primary coolant heat exchangers and on the opposite plate side of the pool coolant in the pool coolant heat exchanger. In both the primary and pool coolant heat exchangers, the secondary water flows in a cross-flow configuration to the process water being cooled." 1996 2006

5.4.3 Cooling Tower

The cooling tower is wood framed and is of the induced draft, cross flow type. It is capable of cooling 4632 gallons of water per minute from a hot water temperature of 104°F to a cold water 1973-1974

1994
1996
1973-
1974
2006 1970-
1971

temperature of 87°F at a maximum wet bulb temperature of 77°F. ~~The tower consists of two~~

c The tower consists of three cells each with a fan assembly. ~~lls provide~~ 1973-1974

"Vibration cut-out switches are installed in each fan motor circuit to prevent damage to the cooling tower in case of inbalance." 1970-1971

~~megawatts of thermal power. Each cell fan is~~

"The fans will be configured to run as necessary to provide sufficient cooling for reactor 10 MW operation." 1996

~~the temperature of the water returning to the heat exchangers.~~ *

Insulation and electrical heating tape on the exposed tower piping provide protection from freezing during periods of inoperation in the winter. The basin water is protected from freezing by a series of steam jets ~~on the perimeter of the basin wall.~~ DELETE 1994

5.4.4 Circulating Pumps

A 150 horsepower, variable speed, centrifugal ~~al~~ type circulating pump provides a flow capacity of 2200 gallons per minute to the secondary system for five megawatt reactor operation. Two such pumps operate in parallel to provide secondary flow for operation at ten megawatts. ~~One standby pump in addition~~ 2006

"One standby pump in addition to the number required for normal cooling is installed." 1994

5.4.5 Instrumentation

"The secondary water inlet and outlet temperature to the reactor heat exchangers and the total flow in the secondary system are displayed and recorded in the reactor control room." 1970-1971

1973-1974
* PORTION OF PARAGRAPH 1 OF 5.4.5 DELETED W/ EXCEPTION OF 1970-1971 REV. ADDITION

1994
1973-
1974
1972-
1973

~~A complete system to measure gross gamma activity~~
"**A complete system to measure gross gamma activity of the secondary system is installed with the output displayed in the reactor control room.**" 1994
~~displayed and recorded in the reactor control room.~~
For additional details on this system refer to Section 9.7.

5.4.6 Water Treatment

The pH of the secondary system will be maintained by the addition of concentrated sulphuric acid. ~~A~~
"A gravity feed, solenoid valve controlled acid addition line has been provided to serve this purpose." ~~...ve this purpose.~~ 1972-1973

Other chemicals, as necessary to control hardness and microbiological growth, will be added by the use of an automatic secondary water control system installed in the reactor cooling tower building. 1967-1968

5.4.7 Automatic Control Valves

"The amount of water passing through the secondary side of the parallel pool and parallel reactor heat exchangers is controlled by the use of two bypass lines around each parallel heat exchanger system. In one of the bypass lines to each parallel exchange system . . ." 1973-1974
~~to each exchanger~~ are manual valves providing adjustment from the main equipment room during non-nuclear operation. The other bypass line contains an automatic control valve which adjusts the flow in the bypass in response to the respective pool or reactor water temperature from the heat exchangers.

5.5 Cooling System Thermal and Hydraulic Analysis

5.5.1 Heat Transfer Data Summary

Table 5.1 lists and summarizes the reactor system heat transfer data for maximum power operation at both 5 and 10 megawatts.

TABLE 5.1 SUMMARY OF HEAT TRANSFER DATA

	5 megawatt operation		10 megawatt operation
Ave. Pwr. Density in Core	151 Kw/liter		303 Kw/liter
Ave. Specific Power	806.5 kw/kg U-235 1972-1973		1613 Kw/kg U-235
Core Coolant			
Total Flow Rate	1800 gpm		3600 gpm
Design Inlet Temp.	140°F		140°F
Mixed Outlet Temp.	158°F		158°F
Est. Press. Drop Across Fuel at 150°F Ave.	3.7 psi		13.9 psi
Est. Press. Drop Across Vessel at 150° F Ave.	5.3 psi		20.0 psi
Min. Press. at Vessel Inlet Nozzle	63.7 psia		63.8 psia
Min. Press. at Pressurizer	75.0 psia 1973-1974		75.0 psia 1973-1974
Heat Transfer Area		184.3 ft^2	
Flow Area-Fuel Elements		.3231 ft^2	
Total Flow Area		.3505 ft^2	
Heat Fraction Released in Core		.87 1972-1973	
Coolant Velocity in Core	11.55 ft./sec.		23.10 ft./sec.
Average Heat Flux	0.86×10^5 Btu/ft^2-hr		1.72×10^5 Btu/ft^2-hr
Fuel Assembly Ave. Outlet Temp.	159.3°F		159.3°F
Ave. Power Density in Hot Channel / Ave. Core Power Density	3.225 (Worst Case, Non uniform Loading) 1969-1970		3.225(Worst Case, Non uniform Loading) 1969-1970
Max. Power Density in Hot Channel / Ave. Power Density in Hot Channel	1.443		1.443

1969-1970
1973
1972-1973
1973-1974

	5 megawatt operation	10 megawatt operation
Hot Spot Position on Fuel, Rods Half Out	18 inches	18 inches
Fractional Bulk Rise at Hot Spot	.701	.701
F_b-Bulk Temp. Factor	1.263	1.263
F_θ-Film Factor	1.332	1.332
Maximum Wall Temp. at Nominal Power	301.1°F	291.3°F
Maximum Wall Temp. at 110% Nominal Power	315.4°F	304.6°F
Saturation Temp. at Hot Spot-Design Press.	295.4°F	284.6°F
Power Level at Which Local Boiling Occurs	5.5 MW	11 MW
Pool Coolant		
Total Flow Rate	$\boxed{550 \text{ to } 750 \text{ gpm}}$ 1973-1974	$\boxed{1200 \text{ to } 1400 \text{ gpm}}$ 1973-1974
Design Inlet Temp.	99°F	99°F
Estimated Heat Input	600 KW	1100 KW
Max. Pool Inlet Velocity	2.5 ft./sec.	5.0 ft./sec.
Nominal Press. Drop Across Reflector Region	$\boxed{1.25 \text{ psi}}$ 1972-1973	$\boxed{5.0 \text{ psi}}$ 1972-1973
Design Velocities:		
Island Region	2 ft./sec.	4 ft./sec.
Unobstructed Test Hole	4.7 ft./sec.	9.4 ft./sec.
Control Rod Gaps-Below Rods	2.2 to 2.9 ft./sec.	4.5 to 5.9 ft./sec.
Reflectors	variable	variable
Design Temp. Rise in Island	5.8°F	5°F

1973
1974

~~Figure 5.5 shows the pressure required to suppress boiling at 1 MW with the most adverse fuel loading.~~

SECTION 5.5.2 REMOVED
REV 1973-1974

~~Figure 5.6 presents the effect of the multiplier R, and is used to aid the interpretation of various peaking assumptions.~~

~~Figure 5.7 shows the total pressure drop across the core as a function of flow rate.~~

1973-1974

5.5.3 Analysis of Natural Convective Cooling of the Core

The reactor core will be cooled by natural convection during initial low power operation. To accomplish this a flanged opening is provided in the invert loop. By removing this flange and the reactor vessel head an open path is provided between the pool and the core allowing natural circulation to take place. The object of this analysis is to determine the natural convection flow rate and the corresponding maximum fuel plate temperature for the initial low power operation.

5.5.3.1 Natural Convection Flow Rate

During natural convection operation, the reactor core is cooled by pool water flowing in through the open flange, down through the 12 inch pipe, up through the pressure vessel and core, and out again into the pool. The flow rate is determined by equating the total system pressure loss to the driving head resulting from the heating of the water in the core.

The total pressure loss consists of turbulent friction loss in the piping and pressure vessel, laminar friction loss in the core, and expansion and contraction losses at changes in cross section throughout the loop. For turbulent flow the frictional pressure loss is given by,

$$\Delta p = 2f \frac{L}{D} \frac{w^2}{g\rho A^2} \text{ psf} \tag{2}$$

where:

f = Fanning friction factor
L = length - ft.
D_e = equivalent diameter - ft.
w = flow rate - lb./sec.
g = 32.2 ft./sec.^{2}
ρ = density of water - lb./ft.^{3}
A = flow area - ft.^{2}

To permit an analytical solution of the equations, the friction factor was evaluated from

$$f = 0.046\ (Re)^{-0.20} \tag{3}$$

This gives a conservative value for the friction factor for smooth pipe over a range of Reynold's Numbers (Re) from 5000 to 200,000 and was obtained from Bonilla[1*].

In the core the pressure drop was taken as that for laminar flow between broad parallel plates given by [3]

$$\Delta p = \frac{12\ \mu\ w\ L}{z^2\ g\rho\ A} \text{ psf} \tag{4}$$

---

*References for Section 5.0 are listed at the end of the section.

where:

μ = water viscosity - lb./ft. sec.

z = plate spacing - ft.

Expansion and contraction losses were determined from

$$\Delta p = \frac{K}{A^2} \frac{w^2}{2g\rho} \text{ psf} \qquad (5)$$

where K is the expansion or contraction loss coefficient based on the area, A. The loss coefficients were taken from Bonilla[2]. From equations (2) through (5) the total pressure loss may be determined as a function of only the flow rate (w) if constant temperatures are assumed. The loss will be of the form

$$\Delta p_L = C_1 w^2 + C_2 w^{1.8} + C_3 w \text{ psf} \qquad (6)$$

$C_1 w^2$ is the term representing the expansion and contraction losses. Table 2.1 lists the value of C_1 associated with each section of the loop. For simplicity the values are based on a single average loop temperature of 100°F, the pool temperature.

TABLE 5.5

Expansion, Contraction & Transition Losses

Location	$C_1 \times 10^3$
Pool to 12" flange	.175
Tee to invert loop	.624
90° Ell to pressure vessel	.220
Entry to pressure vessel	1.020
Core	2.130
Pressure vessel to pool	.625
Total	4.794

The second term in equation (5) is the turbulent flow frictional pressure drop as found from equations (1) and (2). Table 2.2 lists the length of pipe and the value of C_2 for each component, again assuming 100°F water temperature.

TABLE 5.6

Turbulent Flow Frictional Pressure Loss

Component	Length - ft.		$C_2 \times 10^4$
Pressure vessel	8.25		4.22
12" pipe	17.01		2.71
		Total	6.93

The third term in equation (5) is the laminar flow pressure drop in the core as determined from equation (3). At 100° F C_3 has a value of 3.14×10^{-2}.

The natural convection driving head, or pressure gain, is in general

$$\Delta p_g = L(\rho_i - \rho_o) \tag{7}$$

Assuming a linear variation of temperature in the core, L is the distance between the centerline of the core and top of the pressure vessel. ρ_i and ρ_o are the core inlet and outlet water densities respectively.

Expressing the density difference as a function of coefficient of volumetric expansion and temperature difference, and the temperature difference as a function of heat generation rate, flow rate and specific heat, equation (7) becomes

$$\Delta p_g = \frac{L \rho \beta q}{w C_p} \text{ psf} \qquad (8)$$

where:

β = coefficient of volumetric expansion - $^oF^{-1}$
q = heat generation rate - Btu/sec.
C_p = specific heat - Btu/lb.^oF

At an average temperature of 100^oF

$$\Delta p_g = .102 \ (q/w) \text{ psf}$$

for

L = 8.08 ft.
β = 2.0149×10^{-4} $^oF^{-1}$
ρ = 61.99 lb./ft.^{3}
C_p = .997 Btu/lb.^oF

Assuming that 93% of the reactor heat is released in the core, the heat generation rate (q), for a reactor power of 150 KW, is 132.2 Btu/sec. The pressure gain for this heat rate is

$$\Delta p_g = 13.4/w \text{ psf}$$

Equating the pressure loss to the pressure gain, the flow rate is found to be 11.96 lb./sec. For this flow rate the Reynold's Number in the core is 993, indicating the validity of the assumption of laminar flow in the core for determining the core pressure loss. The Reynold's Number in the balance of the system exceeds 17,400.

5.5.3.2 Maximum Wall Temperature

Calculation of the maximum core wall temperature is based on the procedures and equations presented in TM-WRP-62-10[4]. The maximum wall temperature (t_w) exceeds the core inlet temperature (t_i) by

$$t_w - t_i = F_b f(z)\,\overline{\Delta T} + F_\theta P_a \frac{\overline{\Phi}}{h} P_r R_2 R_3 \tag{9}$$

where:

F_b = bulk temperature factor = 1.263

F_θ = film factor = 1.332

$P_r = \dfrac{\text{average power density in hot channel}}{\text{average core power density}}$

= 2.263 for reference case of 5 Kg uniform loading.

$P_a = \dfrac{\text{maximum power density in hot channel}}{\text{average power density in hot channel}} = 1.443$

$f(z)$ = fraction of heat delivered at design point in hot channel = 1.0

ΔT = average bulk temperature rise = 11.0°F

$\overline{\Phi}$ = average heat flux = 2583 Btu/hr. - ft.^{2}

h = heat transfer coefficient = 218 Btu/hr. - ft.^{2} $^\circ$F

R_2 = design safety margin in power level = 1.1

R_3 = ratio of power peaking relative to that for a 5 KW uniform loading = 1.425

The values of peaking factors are those appearing in Section 5.5.2. The bulk temperature rise is from the power and flow rate results of the previous section. The average heat flux is for a heat transfer area of 184.3 ft.^{2} and a core heat generation rate of 132.2 Btu/sec. The core heat transfer coefficient is for laminar or steamline flow in flat channels. For these conditions the Nusselt Number has a constant value of about 8.0 [5]. At a water temperature of 100°F this yields a heat transfer coefficient of 218 Btu/hr. - ft.^{2}-$^\circ$F.

From equation (9) at 150 KW reactor power the temperature difference, $t_w - t_i$, is 130.2°F, which for a pool temperature of 100°F gives 230.2 as the maximum fuel plate surface temperature.

5.5.3.3 Pressure Required to Suppress Local Boiling

Local or nucleate boiling will occur when the maximum wall temperature is greater than the local saturation temperature. The necessary temperature difference to cause boiling may be found from the Jens-Lottes correlation [6]

$$\Delta T_{sat} = \frac{60\ (\phi/10^6)^{1/4}}{e^{p/900}} \quad {}^\circ F \tag{10}$$

assuming it to apply at these low pressures. The heat flux ϕ is given by

$$\phi = \overline{\phi}\ P_r P_a R_2 R_3 \tag{11}$$

Using the values of the factors presented previously, the maximum heat flux is 13,222 Btu/hr.-ft.^{2}-$^\circ$F. This conservatively neglects the effects of local fuel content and meat thickness, and width variations.

For a pressure (p) of 20 psia, equation (10) gives 19.9°F as the allowable wall superheat. The required local saturation temperature is

$$T_{sat} = t_w - \Delta T_{sat} = 247.1 - 19.9 = 227.2^\circ F$$

From the steam tables a pressure of 19.72 psia or 5.0 psig is required to suppress local boiling. For a pool temperature of 100°F, 11.7 ft. of water above the top of the fuel plates will give the required pressure. This pool depth is less than that required for shielding.

5.5.3.4 Conclusions

Calculations indicate that a reactor power of 150 KW may be safely attained with natural convection cooling of the core. The maximum fuel surface temperature is 230.2°F and the natural convection flow rate is 11.96 lb./sec. The 17 feet of water required for shielding at low powers is more than sufficient to prevent local boiling in the core.

The core temperature rise of 11.0°F is large compared to the 1-2°F rise which would occur coincidentally in the island. Consequently, no problems should be encountered with respect to the temperature coefficient of reactivity during cooling of both the core and the island by natural convection.

1973-1974

5.6 Decay Heat Removal

5.6.1 Introduction

A natural convection loop is incorporated into the design of the reactor to remove the decay heat generated in the core following the loss of primary coolant flow and reactor shutdown. "The loop consists of the reactor pressure vessel, the core, two parallel six inch automatic valves, 10 finned tubes in parallel (the in-pool heat exchanger) and necessary connection piping. All of the loop components are located in the reactor pool region and are shown in Figure 4.6. 1973-1974

The flow of coolant, normally downward through the core, will stop, then reverse, flowing upward through the core by natural circulation. Heat will be removed from the loop and transferred to the pool water at the finned tubes.

The ability of the natural convection loop to adequately cool the core is substantiated by the following calculations and results.

5.6.2 Calculations

Because of the complex transient nature of the initial period of decay heat removal in which the flow stops and reverses an exact treatment is not feasible. In general once the flow has reversed, the loop temperature rises and passes through a maximum because the heat rejection capability increases with loop temperature, and the decay heat diminishes with time. The maximum temperature and the time at which this occurs depend upon the initial loop temperature, the power history and the heat capacity and conductance of the convective loop.

5.6.3 Steady State Natural Convection

To examine the steady state conditions, all of the decay heat for some initial period of time is assumed to be absorbed by the water in the natural convection loop, increasing the average water temperature. The decay heat at the end of this period of time is assumed to be removed by steady state natural convection in the loop. Iterative calculation determines the conditions where the heat rejection capability equals or exceeds the heat production rate. The corresponding temperatures are then greater than the maximum loop temperatures which will actually occur. If the core outlet temperature under these conditions is less than 212°F the design is adequate.

Solution of the problem based on these assumptions involves iteration between the decay heat to be removed as determined by the time after shutdown and the amount of heat which can be removed by steady-state natural circulation. For the present analysis an allowable rise in average temperature was first assumed. Knowing the amount of water in the loop, the time required to raise the average temperature the assumed amount can be found from the integrated fission product decay heat release. The log mean temperature difference (LMTD) across the finned tubes necessary to remove this heat by steady-state natural convection may then be determined. The actual LMTD occurring may be found and compared with the required value. If the disagreement is large, the procedure is repeated for a different average temperature rise. Exact agreement between the required and actual LMTD is not necessary since if the actual LMTD is greater than the required LMTD the final temperatures will be less than those calculated.

5.6.4 Initial Transient Period

The initial transient period is that time from shutdown until steady-state natural convection occurs. During this period all of the fission product decay heat is assumed to be absorbed by the loop water, thereby raising the average loop temperature.

Table 5.7 lists the volumes of water in each segment of the natural convection loop. The total is subdivided according to the initial temperatures at the instant of scram. The normal core inlet temperature is 140°F and the normal core outlet temperature is 160°F.

TABLE 5.7

Natural Convection Loop Volumes

Segment	Volume at 140°F ft.^{3}	Volume at 160°F ft.^{3}	Total Volume ft.^{3}
Press. Vessel	4.3	2.015	6.32
Core	0.135	0.135	0.27
12" Pipe	2.33	12.0	14.33
6" Pipe	3.52	1.92	5.44
Finned Tubes	0	0.75	0.75
Total	10.285	16.825	27.11

Volume averaging the initial temperature gives 152.4°F as the initial average loop temperature. Based on a density of 61.1 lb./ft.^{3} at the average temperature there is 1658 pounds of water in the loop. Assuming a 20°F rise in average temperature, the total heat absorbed would be 33,160 Btu.

The integrated fission product decay heat was determined as a function of time after shutdown from Figure 12 of TM-DMS-59-1[1]. For 30 days operation at 10 MW the time integrated beta and gamma energy released after shutdown is

$$Q = .689\ t^{.868}\ \text{MW-sec.}$$

where t is in seconds after shutdown. Since a large portion of this total is in gamma rays which will not be absorbed in the core the actual decay energy in the reactor loop will be less than this total. It is estimated that 62.7% of the total energy appears in the core or

$$\begin{aligned} Q_c &= .432\ t^{.868}\ \text{MW-sec.} \\ &= 409.5\ t^{.868}\ \text{Btu} \end{aligned}$$

Based on this integrated heat release and the previously determined amount of water in the loop, and assuming no heat losses, it would take 156 seconds to raise the loop average temperature 20°F.

The fission product decay heat release rate may be found at any time after shutdown from Figure 11 of TM-DMS-59-1. At 156 seconds the decay heat rate (Q) is 167.9 Btu/second.

5.6.5 Steady-state Natural Convection

The steady-state natural convection flow rate is determined by equating the total loop pressure loss to the driving head obtained because of the different water densities in the hot and cold legs of the loop.

The total pressure loss consists of frictional losses in the components of the loop and the contraction and expansion losses at changes in cross sectional area. The friction losses may be either turbulent or laminar depending on the Reynold's Number in the particular section.

5.6.6 Pressure Losses

For turbulent flow the frictional pressure loss is given by

$$\Delta p = f \frac{2\ L}{D_e} \frac{w^2}{g\rho A^2} \quad \text{psf} \tag{12}$$

where:

f = Fanning friction factor
L = length - ft.
D_e = equivalent diameter - ft.
w = flow rate - lb./sec.
g = 32.2 ft./sec.^{2}
ρ = density of water - lb./ft.^{3}
A = flow area - ft.^{2}

To permit an analytical solution of the equations the friction factor was evaluated from

$$f = .046\ (R_e)^{-0.20} \tag{13}$$

This gives a conservative value for the friction factor over a range of Reynold's Numbers (R_e) from 5,000 to 200,000 and was obtained from Bonilla[1].

In the core the pressure drop was taken as that for laminar flow between broad parallel plates given by[3]

$$\Delta p = \frac{12 \; w \; L \; \mu}{z^2 \; g \; \rho \; A} \; \text{psf} \qquad (14)$$

where:

μ = water viscosity - lb./ft. sec.
z = plate spacing

Expansions, contraction and transition losses were determined from

$$\Delta p = \frac{K}{A^2} \; \frac{w^2}{2g\rho} \; \text{psf} \qquad (15)$$

where K is the expansion or contraction loss coefficient based on the area, A. The loss coefficients were taken from Bonilla[2].

From equations (12) through (15) the total pressure loss in the natural convection loop may be determined as a function of only the flow rate (w) if constant temperatures are assumed. The loss will be of the form

$$\Delta_{PL} = C_1 w^2 + D_2 w^{1.8} + C_3 w \; \text{psf} \qquad (16)$$

TABLE 5.8

Expansion, Contraction and Transition Losses

Location	$C_1 \times 10^2$
Core Fuel Element Support Assemblies	.793
Pressure Vessel to 12" Pipe	.065
12" to 6" Pipe	1.027
Automatic Valve	.102
Finned Tubes	2.757
6" to 12" Pipe	.475
90^o Ell	.026
12" Pipe to Pressure Vessel	.065
Total	5.310

The second term in equation (5) is the turbulent flow frictional pressure drop in the loop as found from equations (12) and (13). Table 5.9 lists the length of pipe and the value of C_2 for each component of the loop, again assuming 180^oF water temperature.

TABLE 5.9

Turbulent Flow Frictional Pressure Loss

Component	Length - ft.	$C_2 \times 10^2$
Pressure Vessel	9.33	.042
12" Pipe	16.41	.023
6" Pipe	27.00	.991
Finned Tubes	4.75	7.190
	Total	8.246

Table 5.10 lists the dimensions and properties of the finned tubes. These are for extruded aluminum tubes manufactured by Brown Fin Tube Company.

TABLE 5.10

Finned Tubes

Tube O.D.	1.90 inches
Tube I.D.	1.712 inches
External Fins	
number	28
thicknesses	0.05 inch
height	0.5 inch
Internal Fins	
number	14
thicknesses	0.05 inch
height	7 - 0.5 inch, 7 - 0.375 inch
Internal Net Flow Area	.01389 feet
Internal Equivalent Diameter	0.0377 feet
Total Outside Surface Area	2.831 $ft.^2/ft.$

The third term in equation (16) is the laminar flow pressure drop in the core as determined from equation (14). At 180°F C_3 has a value of 1.64×10^{-2}.

5.6.7 Pressure Gain

The natural convection driving head or pressure gain is in general,

$$\Delta p_g = L\ (\rho_c - \rho_h) \tag{17}$$

Assuming a linear variation of temperature in the core and in the finned tubes, L is the distance between the centerline of the core and the finned tubes.

1967-
1968

ρ_c and ρ_h are the water densities in the cold and hot legs respectively. Expressing the density difference as a function of coefficient of volumetric expansion and temperature difference, and the temperature difference as a function of decay heat rate, flow rate and specific heat, equation (17) becomes

$$\Delta p_g = \frac{L\ \rho\ \beta\ q}{w\ c_p} \quad \text{psf} \qquad (18)$$

where:

β = coefficient of volumetric expansion - $^{o}F^{-1}$
q = decay heat rate - Btu/sec.
c_p = specific heat = Btu/lb.-^{o}F

At an average temperature of 180^{o}F,

$$\Delta p_g = 44.4/w \text{ psf} \qquad (19)$$

This is for

L = 12 feet
β = 3.634 x 10^{-4} $^{o}F^{-1}$
q = 167.9 Btu/sec.

5.6.8 Flow Rate and Heat Transferred

Equating the pressure loss to the pressure gain results in a relation from which the natural convection flow rate may be determined. For the previously assumed conditions and from equations (17) and (18) the steady-state natural convection flow rate is ~~7.3~~ 6.75 lb/sec. For this flow the Reynold's Number in the core is 1212 indicating the assumption of laminar flow in determining the core pressure loss was valid. The Reynold's Numbers in the other portions of the loop exceed 8000. 1967-1968

For a flow of 7.390 lb./sec. and a decay heat rate of 167.9 Btu/sec. the temperature rise through the core is 20.2^{O}F. As indicated in Section 5.6.4 this heat rate occurs at the end of the initial transient period corresponding to the assumed 20^{O}F increase in average loop temperature. Since the initial average temperature was 152.4^{O}F the average temperature during natural circulation is 172.6^{O}F. This average temperature (t) is related to the core inlet (t_c) and outlet (t_h) temperatures by,

$$\bar{t} = .62\ t_c + .38\ t_h$$

For a temperature rise (t_h-t_c) of 20.2^{O}F and an average temperature of 172.6^{O}F, the core inlet and outlet temperatures are 160.1^{O}F and 180.3^{O}F respectively.

Since the outlet temperature is below the atmospheric boiling point of water, no net formation of steam will occur if the actual LMTD is greater than the required LMTD. Based on the inlet temperature of 100^{O}F, the actual LMTD is 70.2^{O}F.

The required LMTD depends on the heat transfer area and the film coefficients at the finned tubes. The film coefficient on the inside of the finned tubes was determined from the Dittus-Boelter equation

$$\frac{H_i D}{k} = .023 \left|\frac{DG}{\mu}\right|^{.8} \left|\frac{c_p \mu}{k}\right|^{.4} \qquad (20)$$

For the water physical properties evaluated at 180^{O}F and the finned tube properties taken from Table 5.10 the inside film coefficient is 396 Btu/hr.-ft.^{2}-^{O}F.

The film coefficient at the outside of the finned tubes is given by the relation for a long vertical surface[8].

$$h_o = .13\ k\ (a\Delta t)^{1/3} \tag{21}$$

where:

$a = g\beta\rho^2\ c_p/\mu k$

Δt = difference between the surface and the bulk temperatures

The local film temperature difference is a portion of the local total temperature difference, the fraction of the total being the ratio of the over-all heat transfer coefficient to the local heat transfer coefficient. Overall heat transfer coefficients (U_o) for finned tubes may be obtained from sets of curves giving the overall coefficient as a function of inside and outside coefficients for the particular finned tube being used[9].

Because of the relationship between the outside film temperature difference and the various film coefficients, repeated calculations must be made to establish the correct values. After several iterations the film coefficients on the outside of the finned tube were found to be 153 and 140 Btu/hr.-ft.^{2}-^{o}F for the hot and cold ends of the tube respectively. For an average value of 146.5 Btu/hr.-ft.^{2}-^{o}F assumed to apply over the entire length, the over-all heat transfer coefficient is 55 Btu/hr.-ft.^{2}-^{o}F referred to the outside surface area of the finned tubes.

The required LMTD may now be found from

$$q = U_o A_o \ (LMTD)$$

For ten tubes each with 4.5 feet of finned length the LMTD required to transfer 167.9 Btu/sec. to the pool is 70.7°F. This value varies only slightly from the actually occurring value of 70.2°F, therefore indicating the assumption of a 20°F rise in average loop temperature is reasonable. Since the maximum bulk water temperature in the loop for these conditions is 180.3°F, no net formation of steam would be expected.

5.6.9 Summary of Results

A summary of the characteristics of the natural convection loop and the results of the calculations are presented in Table 5.11 below:

TABLE 5.11

Summary

Finned Tubes	
Number	10
Size	1.9 inch O.D.
Finned Length	4.5 ft.
Fins	28 external - 14 internal
Loop Valve Size	6 inch
Header Size	6 inch Sch. 40
Initial Average Loop Temperature	152.4°F
Loop Volume	27.11 ft.^{3}
Time Required to Reach Steady-state	156 seconds
Increase in Average Loop Temperature	20°F
Decay Heat Rate	167.9 Btu/sec.
Steady-state Flow	7.93 lb./sec.

TABLE 5.11 (cont'd)

Hot Leg Temperature	180.3°F
Cold Leg Temperature	160.1°F
Pool Temperature	100°F
Finned Tube Over-all Heat Transfer Coefficient	55 Btu/hr.-ft.^{2}-°F
Internal Heat Transfer Coefficient	396 Btu/hr.-ft.^{2}-°F
External Heat Transfer Coefficient	146.5 Btu/hr.-ft.^{2}-°F
Required Log Mean Temperature Difference	70.2°F
Actual Log Mean Temperature Difference	70.7°F

5.6.10 Conclusions

Calculations based on conservative assumptions indicate that the natural convection emergency decay heat removal loop is adequate to prevent net formation of steam following loss of reactor coolant flow. After an initial period of about 2.5 minutes during which it was assumed that no heat is lost from the loop, calculations indicate that steady-state natural convection will transfer the decay heat to the pool with a maximum loop temperature of about 180°F. Any heat lost from the loop during the initial period will tend to reduce this maximum temperature.

5.7 References for Section 5.0

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(7) Shapiro, D. M., "Fission Product Decay Power," Internuclear TM-DMS-59-1, January, 1959.

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(9) Brown Fin Tube Company, Chart U0.584.64.

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ENGINEER: CBE
TONY S.

CODE: LPF 10-2-07

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FIGURE 5.1

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

PIPING & INSTRUMENT DIAGRAM

MURR NUMBER: 156

SHEET 1 OF 1

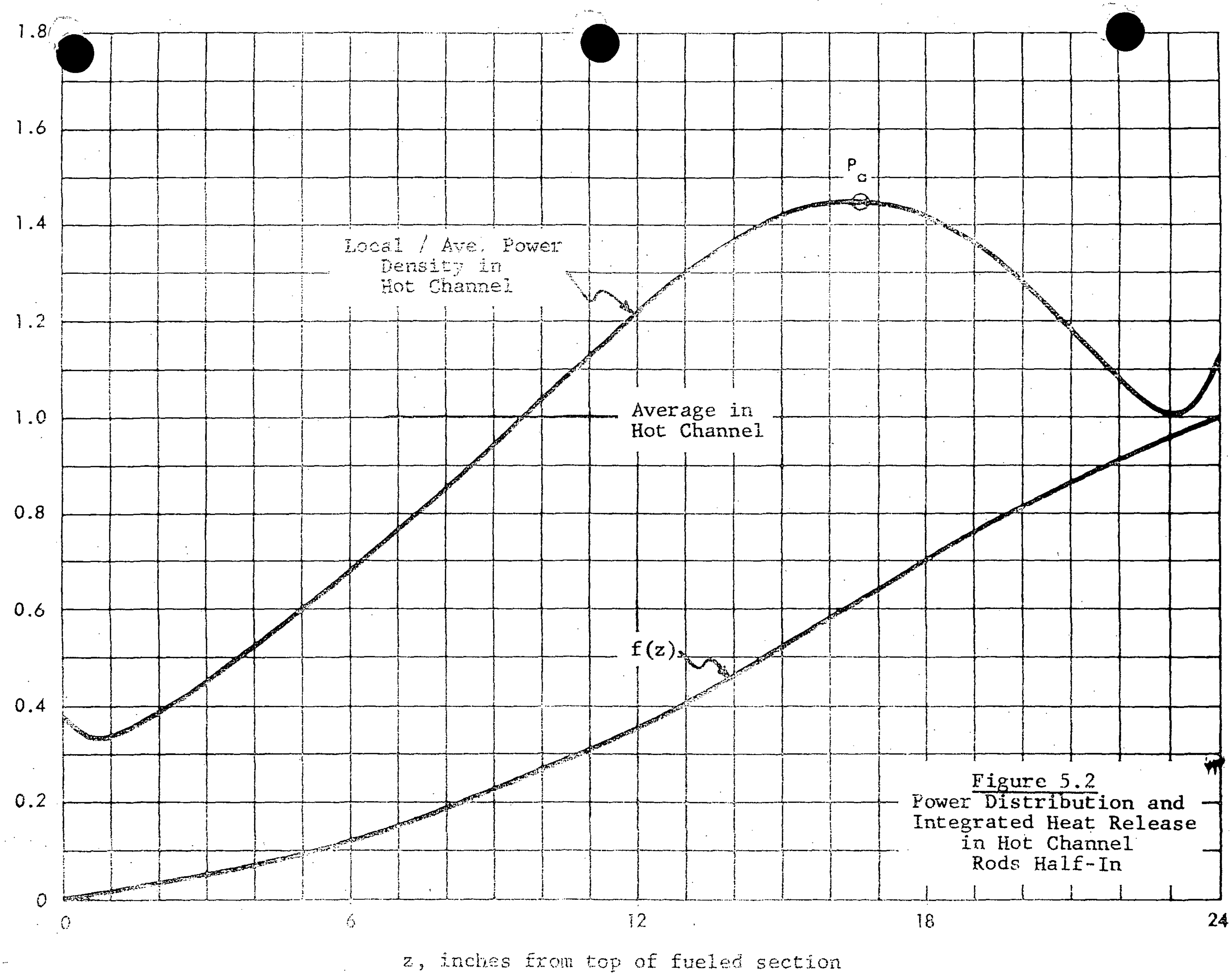


Figure 5.2
Power Distribution and Integrated Heat Release in Hot Channel
Rods Half-In

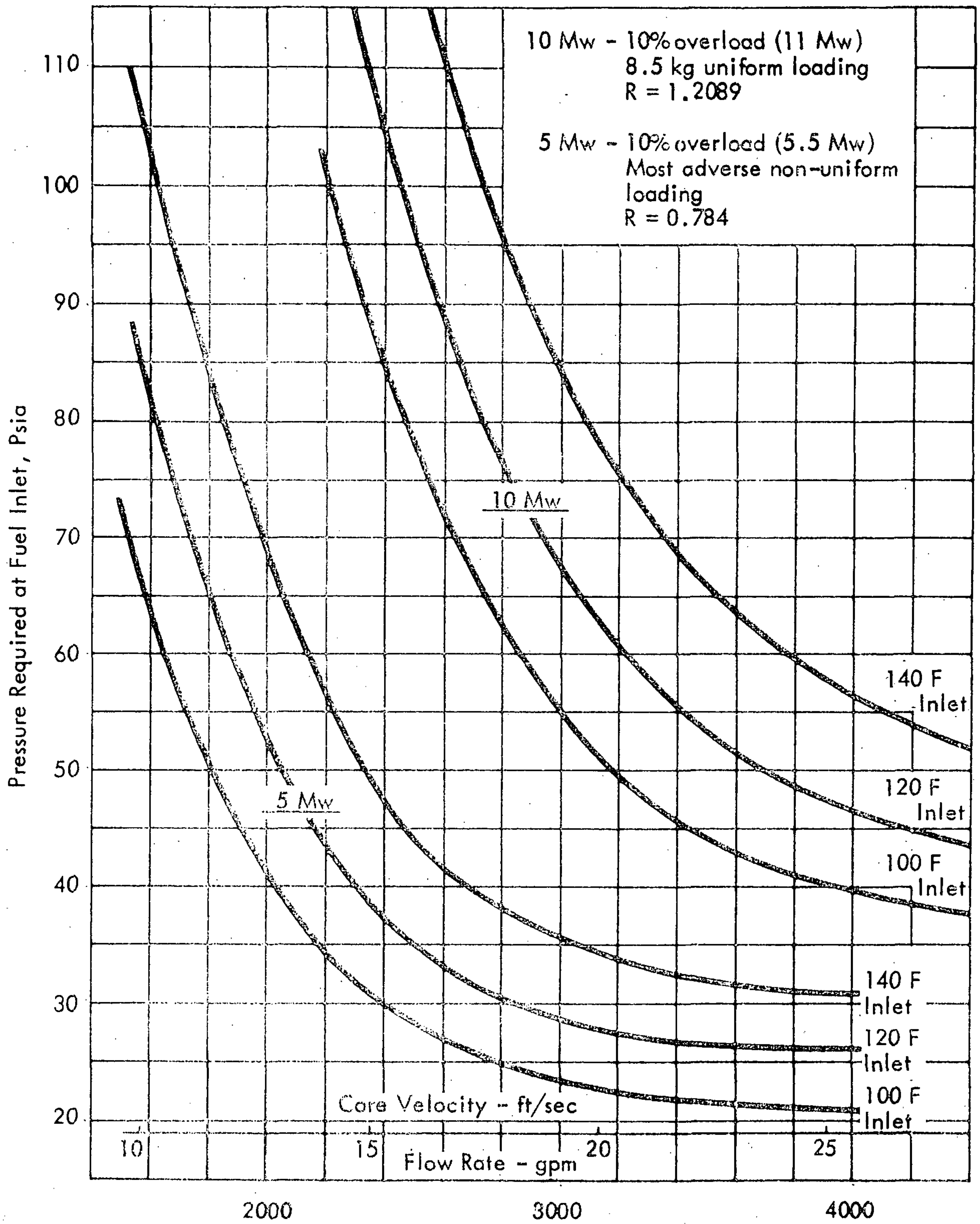


Figure 5.3 Coolant Conditions Required to Suppress Boiling

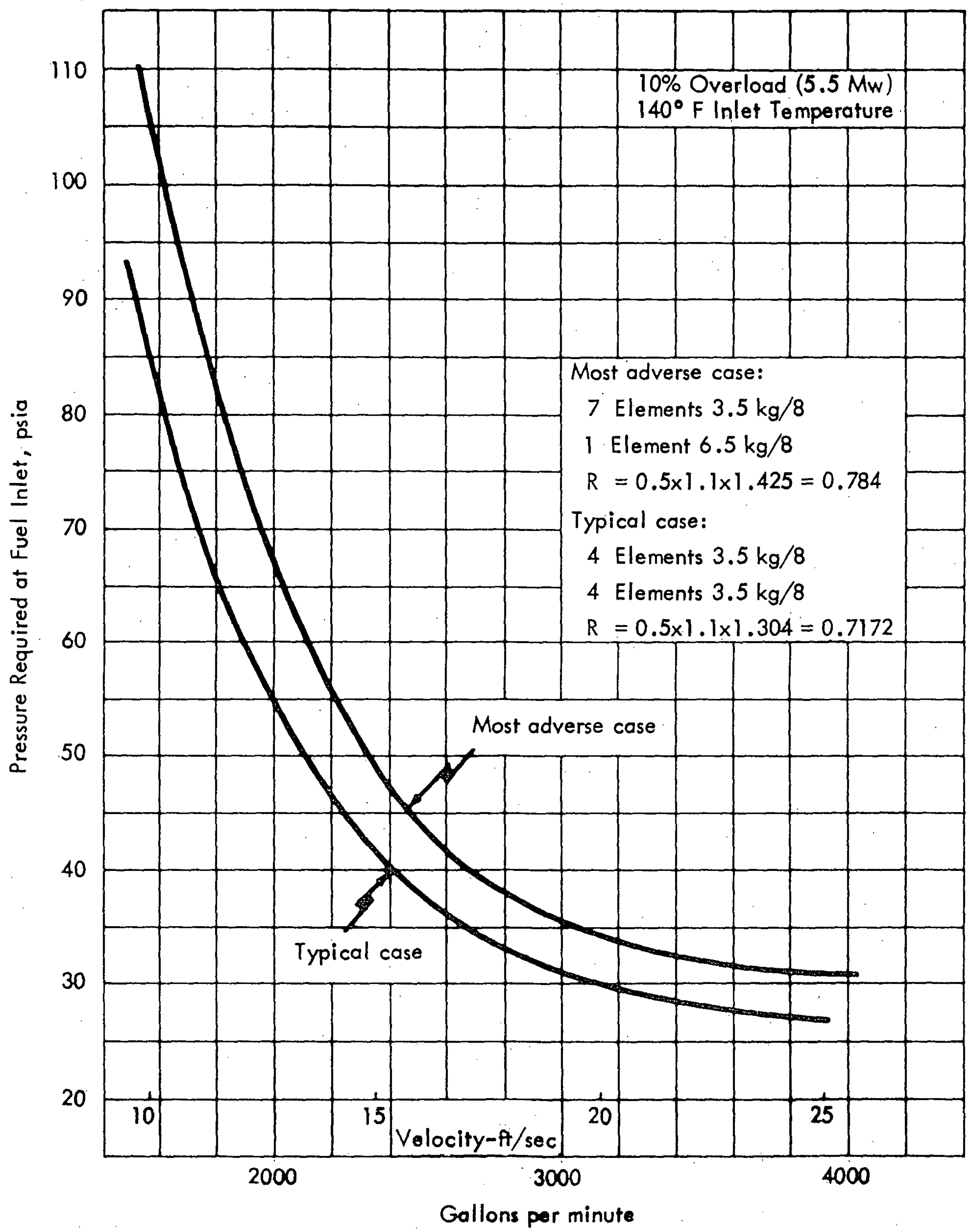


Figure 5.4 Effect of Non-Uniformity of Fuel Loading at 5 Megawatts

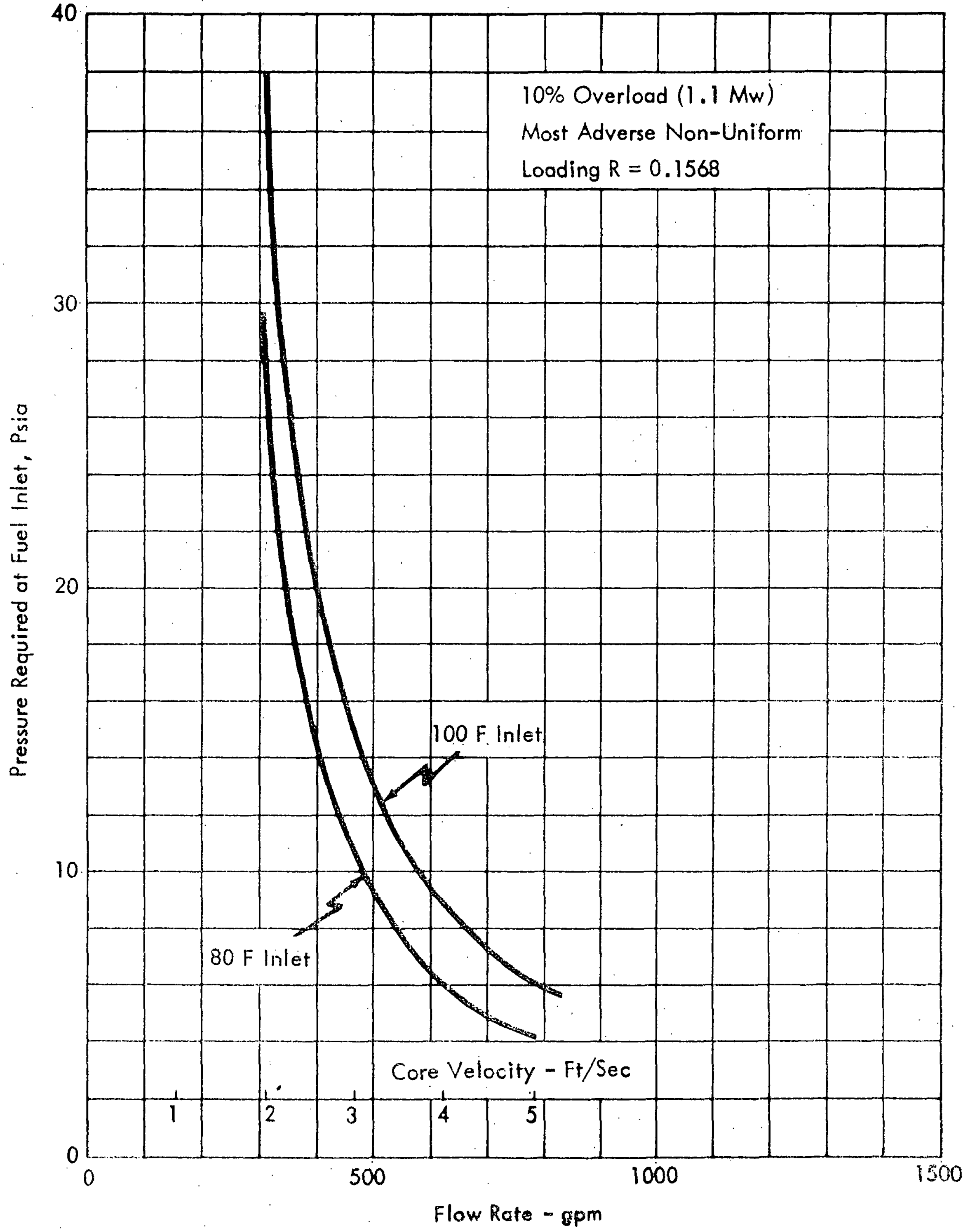


Figure 5.5 Coolant Conditions Required to Suppress Boiling at 1 Megawatt

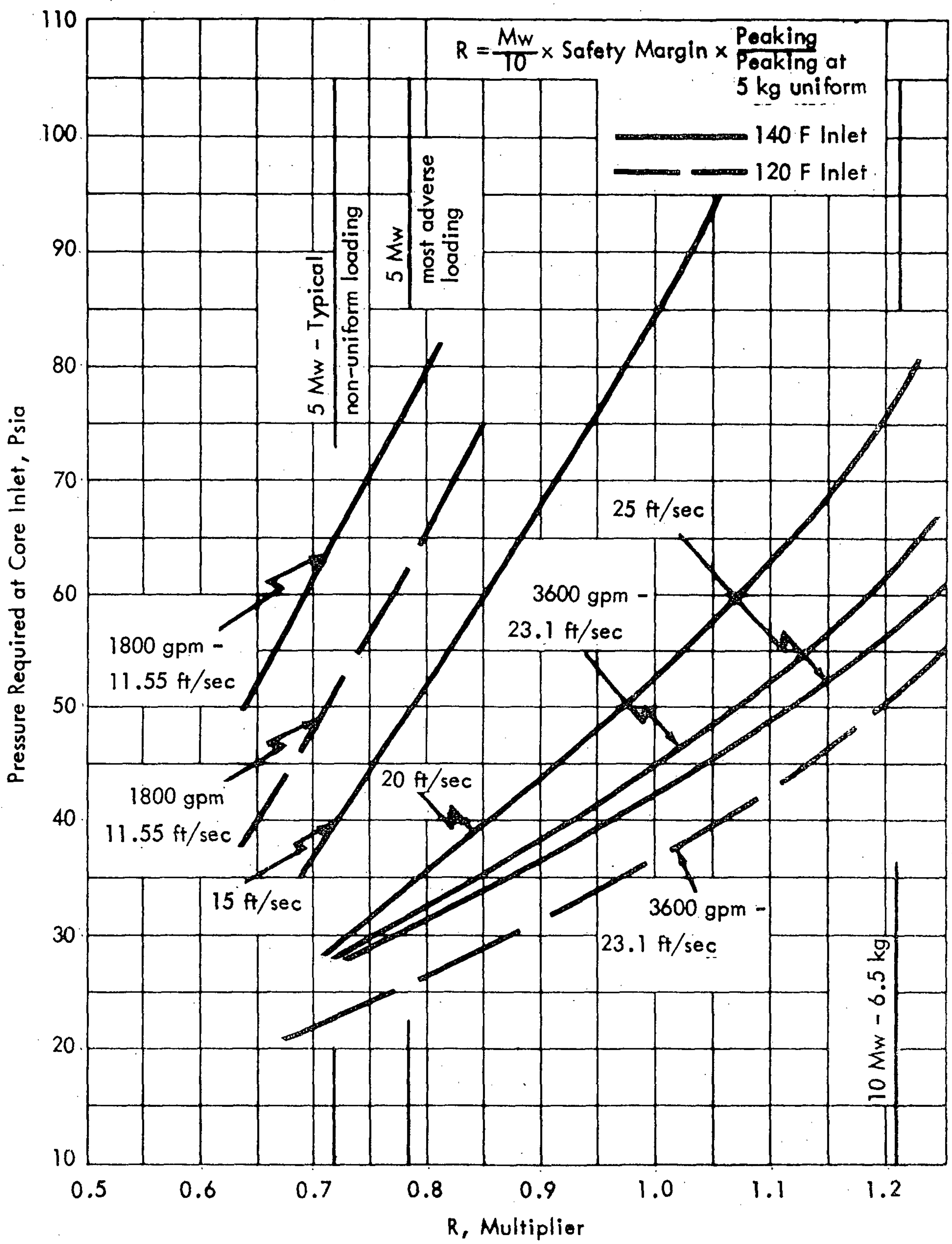


Figure 5.6 Effect of Multiplier R on Coolant Conditions Required to Suppress Boiling

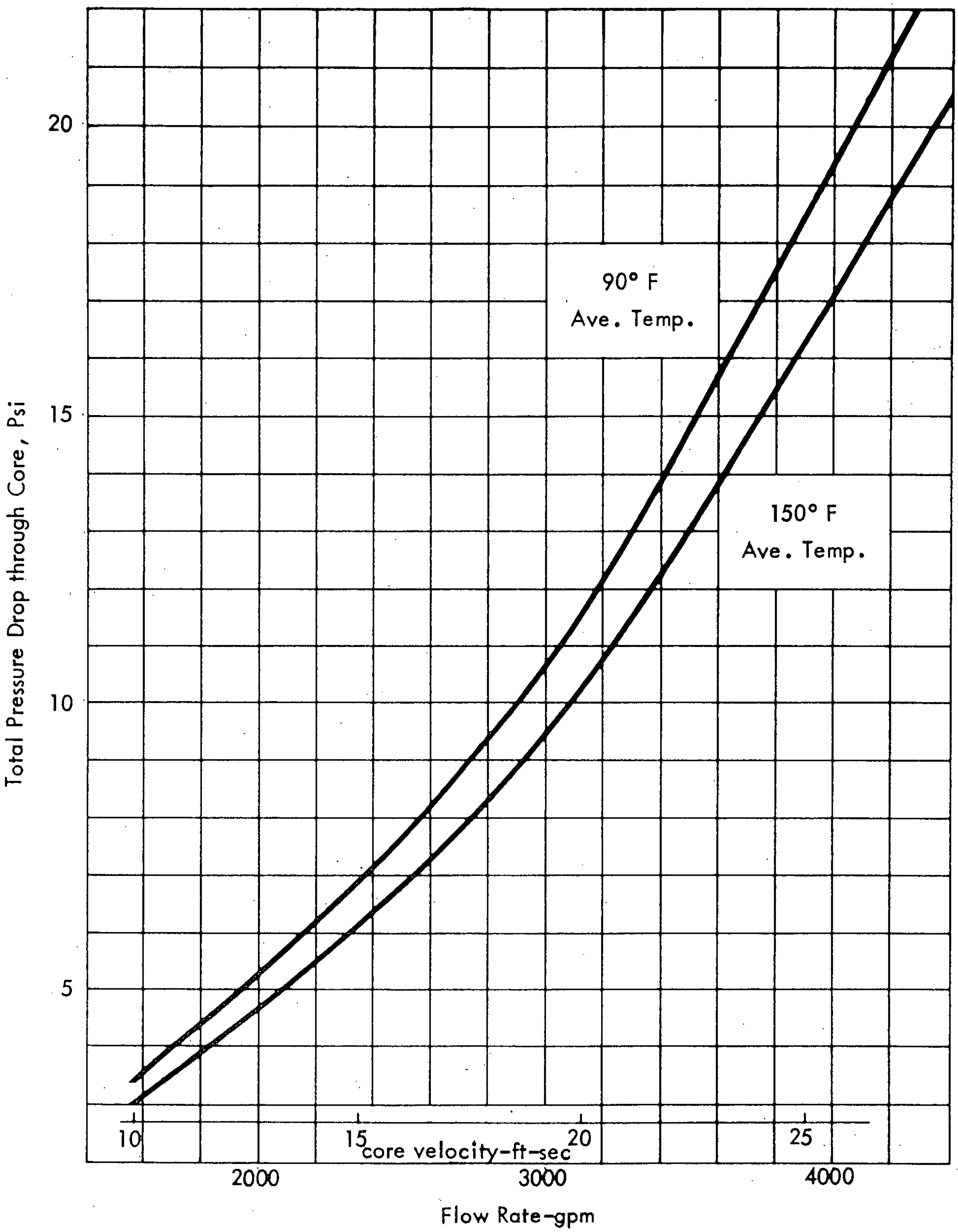


Figure 5.7 Total Pressure Drop Across Fuel Elements

6.0 SHIELDING

6.1 Introduction

The required biological shield thicknesses are presented for the primary reactor shield of the University of Missouri Research Reactor. Included are required magnetite concrete thicknesses for the bulk shield structure, water shielding requirements above the operating reactor core, and shielding requirements during spent fuel element transfer and storage as a function of the fission product decay.

All shielding thicknesses are based on an operating power level of 10 MW. Fuel storage and handling requirements are based on 40 day continuous operation at 10 MW prior to shutdown and removal of fuels. All shielding is designed to conservatively satisfy the permissible dose limitations to personnel as advocated in NBS Handbooks 59 and 63.

In the analysis and design of the primary reactor shielding the following dose rate schedule for 10 MW operation has been used.

(1) At one foot from the primary biological shield at core centerline midway between beam ports the radiation level shall not exceed 2.5 mr/hr.

(2) At one foot from the primary biological shield and three feet from any experimental facility opening in the shield the radiation level shall not exceed 2.5 mr/hr.

(3) At three feet from any experimental facility opening in the primary biological shield and on the centerline of the opening the dose rate shall not exceed 2.5 mr/hr.

(4) At one foot from the heat exchanger room walls and ceiling the radiation level shall not exceed 2.5 mr/hr.

(5) At one foot from the demineralizer room walls and ceiling the radiation level shall not exceed 2.5 mr/hr.

(6) At one foot above the reactor room floor but ten feet from the primary reactor shield the radiation level shall not exceed 2.0 mr/hr.

(7) At any location at the top surface of the pool the radiation level shall not exceed 20 mr/hr.

Several additional dose rate criteria with regard to fuel handling were selected for design purposes. The design conditions are 40 days continuous operation at 10 MW, followed by 10^5 seconds fission product decay time (1.16 days) prior to fuel handling and storage. Shield thicknesses for fuel storage are based on these conditions and the same dose rate criteria as for the bulk shield. The required water cover during fuel handling is based on a 100 mr per hour maximum dose rate from a fuel element at these reference conditions during fuel transfer.

6.2 Gamma Ray Attenuation-Biological Shield

The attenuation of the gamma rays originating in the reactor core from prompt fission, fission products, and radiative neutron capture gamma rays generated throughout the reactor complex were considered in the analysis of the biological shield requirements. An analytical model of the reactor core and surrounding regions was developed. External to this analytical model of the reactor system, the thickness of the water and magnetite concrete cylindrical annuli were varied to account for the eccentricity of location of the reactor core in the pool.

All calculations of required shield thicknesses were based on regional gamma ray source spectrums generated by Internuclear Company. The calculations of the gamma ray dose external to the shield was performed using a computer program. In this computer program the gamma ray buildup factors were included. In an effort to assure the validity of use of the selected buildup factors, a series of identical applicable problems were calculated using iron dose buildup factors and water dose buildup factors. These calculations of iron and water bracketed the value obtained for the concrete shielding and serve to substantiate the use of the constants selected.

The concrete thicknesses required to achieve the applicable dose rate criteria, as defined in the introduction, are shown as a function of pool water radius between the core and the biological shield in Figure 6.1.

Attenuation of the gamma rays in an axial direction was calculated to size the pool water depth for an operating power level of 10 MW. Calculations were based on the gamma ray sources as determined in the analytical model. The dose rate as a function of water depth over the active core is shown in Figure 6.2. It will be noted that to reach a tolerable dose rate from the direct penetration core gamma ray contribution in comparison to the pool water activity dose rate, a minimum water depth over the operating active fuel region of 23.6 feet is required at a power level of 10 MW.

6.3 Neutron Attenuation-Biological Shield

Neutron attenuation in the biological shielding was calculated to insure adequate neutron removal. A calculation was performed by Internuclear Company using a computer to integrate the point water attenuation kernel over the reactor core volume. Correcting this kernel by the exponential attenuation of the non-hydrogenous materials in the system, the dose rate from fast neutrons is then calculated assuming a dose rate conversion of 0.15 mrem per hour per unit neutron flux.

The required magnetite concrete thickness, as a function of pool radius, to achieve a neutron dose rate of 10 per cent of the dose rate criteria is shown in Figure 6.3. It can be seen that the neutron dose rate is of negligible importance in comparison to the gamma ray dose rate at the bulk shielding requirements needed to attenuate the gamma radiation.

6.4 Spent Fuel Transfer and Storage

Transfer of the spent fuel elements as well as storage in the reactor pool or in the spent fuel element storage pit were studied to limit dose rates during these operations to a reasonable level or, if applicable, to the dose rate criteria presented in the introduction to this section. During fuel element transfer, the philosophy used in design was to limit the dose rate during the transfer operation to less than 100 mr per hour at some reasonable time after shutdown. Shield requirements for storage of spent fuel elements in the pool or in the fuel element storage pool are calculated to meet the dose rate criteria of the bulk shielding.

The calculations were performed by Internuclear Company using a computer program. The cases of spent fuel elements, in arrays of single, four, and eight elements adjacent to the concrete wall and of a single element in a horizontal or vertical position shielded by water are calculated as a function of fission product decay time. All calculations were based on an average element with uniform axial burnup and 400 megawatt days of reactor operation at 10 MW.

Dose rates through magnetite concrete shielding from single, four and eight spent fuel element arrays adjacent to the shielding wall are shown in Figures 6.4, 6.5, and 6.6. In these figures the results are presented for fission product decay

1968-1969

times of 10^3, 10^5 and 10^6 seconds. Figures 6.7 and 6.8 show the dose rate through water shielding from a spent fuel element in a horizontal and vertical position for decay times of 10^3, 10^5 and 10^6 seconds.

Application of these results to the design shield thicknesses assumes that the following operational procedures will be followed:

(1) The pool level is lowered to the elevation of the lower bridge before removing fuel from the reactor vessel.

(2) ~~Fuel is transferred from the core to the temporary storage racks located at the bottom of~~ ~~time~~ ~~ther~~ ~~sper~~

Fuel is transferred from the core to the temporary storage racks located at the bottom of the pool. However, it is possible to transfer the fuel directly from the core or from the temporary racks located at the bottom of the main pool to the permanent racks located in the east pool while the pool level is at the refuel level.

~~transfer cask which will be placed on the ledge behind the weir.~~ 1968-1969

(3) ~~No fuel transfer over the weir or into a~~

No fuel transfer over the weir into a shipping cask is intended at the reduced pool level and no spent fuel storage is intended on ledge behind the weir.

~~intended on the ledge behind the weir.~~ 1968-1969

Functional shielding requirements can then be defined as follows:

(1) The minimum concrete thicknesses around stored fuel.

(2) The minimum elevation of the lowered pool to facilitate transfer from the vessel and into temporary storage.

1967-
1968

(3) The minimum required submergence of the weir and ledge to facilitate fuel transfer to the spent fuel storage pit or into a transfer cask.

(4) The minimum required depth of water above spent fuel stored in the spent fuel storage pool in the event that the pool proper is drained. This minimum depth is maintained by the weir separating the two pools.

For a fission product decay time of 10^5 seconds, the shielding requirements for the described conditions are as follows:

(1) Storage of Elements Adjacent to Primary Reactor Shield:

(a) Storage of four spent fuel elements adjacent to primary reactor shield or in the spent fuel storage pit with a dose rate criteria of 25 mrem per hour at one foot from the shield surface requires four foot four inches of magnetite concrete. 1967-1968

(b) Storage of eight spent fuel elements in the spent fuel element storage pool adjacent to the dividing wall between the pool and the storage pool to meet a dose rate criteria of 50 mrem per hour for a worker in the reactor pool requires that the separating wall between the spent fuel storage pit and the reactor pool have a magnetite concrete thickness of three foot eleven inches.

(2) Transfer Operations at Lowered Pool Water Depth:

(a) The minimum water shielding depth over the reactor vessel lip during transfer of a spent fuel element in a vertical position meeting the dose rate criteria of 100 mrem per hour at the water surface is eleven foot of water (this includes a 12 inch clearance of the two foot long fuel element over the vessel lip).

(b) The minimum water shielding depth over the reactor pool spent fuel element storage rack with seven elements in the rack and one element in transfer, or eight elements in the rack, to meet a dose rate criteria of 10 mr per hour from all elements during a single element transfer, and the dose rate with eight elements in the rack not to exceed 1 mr per hour, requires twelve foot six inches of water (this includes 12 inch clearance of the fuel element being transferred into the storage rack).

(3) Transfer Operations at Upper Pool Water Depth: The minimum shield water depth over the pool weir, or over the cask, during transfer of a spent fuel element from the temporary storage racks in the pool to the transfer cask, or to the spent fuel element storage pool, to meet a dose rate criteria of 1 mr per hour from a single fuel element requires fourteen feet of water over the weir or over the shipping

cask (this includes 12 inch clearance of the fuel element over the weir).

(4) Pool Water Level Lowered to the Pool Weir: The minimum water shielding depth over the spent fuel element storage rack with eight spent fuel elements in the rack and to meet a dose rate criteria of 50 mrem per hour at the water surface is ten feet of water over the racks.

6.5 Experimental Facilities Shielding

An analysis of the shielding of the beamports, thermal column, and the gamma irradiation facility has been made. The beamport analysis included a determination of supplementary shielding requirements for the beamport plug and vestibule as necessitated by the loss of concrete represented by the vestibule, by streaming around the beamport plug, and the induced activity of the beamport coolant. All calculations were based on an operating power of 10 megawatts.

The shielding requirements of the experimental facilities were based on radiation dose rate criteria as presented in Section 6.1. The radiation level criteria of 2.5 mr per hour at one foot from the surface of the primary reactor shield was used as a design basis in determining shielding for the beamports and the thermal column.

The shielding analysis of the thermal column is based on a column arrangement utilizing a five foot thickness of graphite along the axis of the column following the lead shielding nose piece.

In the analysis of shielding requirements for the beamports the basis was a six inch typical beamport. Since the primary reactor shield has been sized to compensate for the eccentricity of location of the reactor core in the reactor pool, the analysis of beamport system shielding was required on only one beamport.

6.5.1 Beamport Shielding

Supplementary shielding for the beamports is based on material replacement necessary to maintain the primary reactor shield integrity. Except when used with special experimental facilities, the beam tube volume in front of the shield plug is flooded with water to complete the primary reactor shield. In the calculations to establish the supplementary shielding requirements for the beamports the following areas were studied:

(1) Annuli (water or air filled) between the primary reactor shield casing, the beam tube, and the beamport shield plug.

(2) Annuli of materials (primary reactor shield casing, beam tube, and beamport shield plug casing) which have densities less than the primary reactor shield design density of 3.5 grams/cm^3.

(3) The beamport vestibule at the primary reactor shield surface.

(4) The coolant activation in the beam tube and/or experiment can and the subsequent passage to the outer portion of the reactor shield.

The desired radiation level criteria are satisfied in the beamport design by stepping the aluminum and water or air annuli of the beamport system and by including lead shielding.

Calculations were performed on the following annular geometries:

(1) An air or water annulus of varying thickness followed by lead and concrete shielding or vice versa.
(2) Homogenized annulus of aluminum and air or water of varying thicknesses followed by lead and concrete shielding or vice versa.
(3) An air or water annulus of varying thickness traversing the entire primary reactor shield.
(4) A homogenized annulus of aluminum and water or air traversing the entire primary reactor shield.

From the analysis of the various contributions from the combination of annuli which exist in the design, the amount of lead required at the step can be established and the required step can be sized.

It should be noted that, since a collimated beam of gamma rays is studied, no gamma ray buildup is included in the calculations to account for scattering.

For the beamport design as shown in Figure 8.1, thickness of aluminum and water or air does not exceed 7/8 inches and any single annular thickness of water or air does not exceed 1/16 inches, the lead shielding and step requirements are as follows:

(1) At the step, a minimum of 3 inches of lead in the "line of sight" is required for each annular channel.

(2) A minimum step of 2.0 inches is required with the inclusion of the above lead shielding. The step is defined as the distance from the inner radius of the smaller annulus of the step to the outer radius of the larger annulus or the radial increment from inner magnetite concrete radius to the outer concrete radius at the step.

The additional considerations of the vestibule and coolant activation were solved using standard techniques. The vestibule represents a deficiency of concrete which must be supplemented by additional lead. A minimum thickness of 3 inches of lead is provided based on the concrete deficiency and the relative material densities. This thickness of lead backs up the entire vestibule and overlaps the edges of the vestibule by a minimum of 1 inch. The beamport plug requires a similar minimum thickness of lead.

Coolant activation of the experiment coolant was based on a light water cooled experiment with a total flow rate of 2 gpm to the experimental can or to the flooded volume. Consideration was given to the transient times in the irradiation region and to mixing of the activity in the beam tube. Further consideration of shielding was given to drain lines from the beamport. For those lines carrying the activated coolant a minimum of 2 feet of magnetite concrete is provided between the lines and the surface of the shield.

6.5.2 Thermal Column Shielding

A drawing of the thermal column is shown in Figure 8.2. The thermal column is positioned vertically with the centerline of the column on the centerline of the reactor core. Positioned between the 9 inch outer graphite reflector ring and the thermal column case is a lead nose piece serving as a gamma shield. This shield is shaped on the inner surface to conform to the radius of the aluminum skirt surrounding the graphite reflectors. The minimum thickness of the gamma shield is 4 inches of lead. The edges of this shield are 6 inches of lead. The whole is encased in aluminum.

Minimum shielding thickness occurs on the centerline of the thermal column. A tabulation of materials "seen" in a traverse from the inner face of the gamma shield to the external face of the thermal column door include:

1/4 inches of Al	Gamma shield face plate
4 inches of Pb	Gamma shield
1/4 inches of Al	Gamma shield back plate
3/4 inches of Al	Thermal column face plate
60 inches of Graphite	Thermal column pack
1/4 inches of Boral	Door face plate
25 inches of Steel	Thermal column door

The thermal column case is made of two square boxes. The in-pool portion immediately behind the lead shield is a box 37 1/2 inches square extending back 12 1/4 inches where it is stepped to the next box. The second box is 49 1/2 inches square and 68 inches front to back. The larger (second) box is completely lined with 1/4 inches of boral plate. The analysis of the shielding requirements was made by Internuclear Company using the computer codes GH-4 and GRACE-I. Calculations performed with code GH-4 utilized an approximation of the thermal column geometry of successive cylindrical annular segments to describe the lead, graphite and shield door of the column. Calculations using GRACE-I, which is a multiregion, multi-group gamma ray attenuation program utilizing slab geometry with the option of truncated cone geometry, were performed by describing the regions as truncated cones.

The off-axis shielding of the column and streaming of radiation down the gaps around the door were also studied. The off-axis shielding and streaming shielding is provided by a door overlap of five inches on all four sides of the 4x4 foot thermal column face.

6.5.3 Spent Fuel Element Irradiation Facility

The capability of using an array of spent fuel elements as a gamma radiation source has been installed. A section of shielding in the wall of the spent fuel element storage pool is removable. It is possible to replace this segment of shielding with an irradiation unit fabricated of lead. Initially this shield cavity will be filled with blocks of magnetite concrete. Prior to completion of burn-up of the first core, a lead irradiation mechanism will be designed, fabricated and installed. A license amendment for installation and utilization of the gamma irradiator will be requested about one year after startup.

The spent fuel gamma irradiation facility consists of a cavity in the biological shield wall of the spent fuel storage pool. The cavity is fabricated of aluminum. It is essentially three boxes. The innermost box is 2x2 feet and 20 inches deep. This box steps to the second box which is 2 feet 4 inches square and 10 1/2 inches deep. The third (outer) box is 2 feet 8 inches square and 2 feet deep.

The cavity in the shield, formed by this spent fuel irradiation facility casing, is filled with cast blocks of magnetite concrete. The blocks are of two sizes 4x4x10 inches and 4x4x10 1/2 inches. The blocks are positioned in the cavity in staggered rows to minimize gap lengths. When the cavity is filled with blocks the effective shielding is 4 feet 6 1/2 inches of magnetite. The calculated dose rate external to the shield with six spent elements in the storage rack is 1.4 mr/hr for elements subjected to 400 MWD reactor operation and 10^5 seconds decay time.

6.6 Coolant System Shielding

6.6.1 Piping and Coolant Equipment Room

The shielding requirement for the coolant piping and the heat exchanger room are determined entirely by the N-16 activity in the two coolant loops. Required thicknesses of shield have been calculated on the basis of concrete densities of 2.2 gm per cm^3 for ordinary concrete and radiation level criteria as defined in the introduction to this section. The calculated equilibrium specific activities of N-16 are 6.6×10^6 and 1.54×10^7 mev per cm^2 second at the exit of the core activation regions of the reactor and pool loops respectively.

The shielding calculations are based on the locations of the in-pool invert loop and piping in the tunnel beneath the reactor as indicated on the drawings. Decay times, transient times, and source geometries are based on 12 inch reactor loop piping, six inch pool loop piping and the design flow conditions.

The calculations for the heat exchanger room assume that one inlet pipe is located at any point along the wall or ceiling. The shielding requirements for a heat exchanger room are generally determined by such a critical pipe location, since the radiation dose rate from this geometry exceeds that from the distributed components in the room. This assumption provides the most conservative estimate of shielding requirements.

The resulting shielding requirements are as follows:

(1) Invert Loop:
   (a) The invert loop is located adjacent to the primary reactor shield in the reactor pool.
   (b) Radiation level criteria is 2.5 mr per hour at one foot from the shield surface.
   (c) The shielding requirement is three and one half feet of magnetite concrete or its equivalent in all directions from the pipe section.

(2) Pipe Tunnel:
   (a) The pipe tunnel referred to is located beneath the beamport floor and runs from the reactor pool to the heat exchanger room. The area of concern consists of the beamport floor area immediately above the pipe tunnel.
   (b) Radiation level criteria is 2.0 mr per hour at one foot from the shield surface.
   (c) The shielding requirement is five feet of ordinary concrete or the equivalent.

(3) Heat Exchanger Room:
   (a) The heat exchanger room is located below mechanical equipment space and adjacent to the reactor containment building. The areas of concern are the walls and ceiling of the heat exchanger room. Occupied area is immediately above the heat exchanger room.

(b) Radiation level criteria is 2.5 mr per hour at one foot from the shield surface.

(c) The shielding requirement is five feet of ordinary concrete or the equivalent.

6.6.2 Demineralizer Shielding

The demineralizer shielding requirements are reduced by locating the pool loop bypass stream after the holdup tank and by providing a two minute holdup tank on the reactor loop bypass line, thus avoiding the need to shield high energy N-16 gamma radiation. The demineralizer shielding requirements are then determined by the specific activities deposited on the beds. The coolant lines serving the units however, still have appreciable 0-19 activity which influences the shielding requirements for these lines.

Separate demineralizer beds are used for the reactor loop and the pool loop, but these units, plus the spare unit, are interchangeable, hence the shielding requirements are identical for all units. Similarly the regeneration station for the spent resins has identical requirements because it must receive the activity from any one unit.

Calculations of the equilibrium activities present on the demineralizer beds during operations have been made. Based on the specific activities calculated and on a minimum ordinary concrete density of 2.2 gm per cm^3, the shielding requirements of the demineralizer system were found to be as follows:

(1) Demineralizer Unit Piping

(a) Location - This piping extends from coolant loops in the heat exchanger room to the demineralizer beds.

(b) Radiation level criteria is 2.5 mr per hour at one foot from the shield surface.

(c) The shielding requirement is one foot of ordinary concrete or its equivalent.

(2) Demineralizer Beds

(a) Location - The demineralizer unit cells are located in an area adjacent to the heat exchanger room. The calculation of shielding requirements are based on occupied space above and readily accessible areas to the front of the demineralizer cells.

(b) Radiation level criteria is 2.5 mr per hour at one foot from shield surfaces.

(c) Shielding requirement is three feet of ordinary concrete or its equivalent in those directions where access is provided during operation of the units. The assumption is made that sufficient delay time has been built into the system to permit N-16 and 0-19 activities to decay.

(3) Regeneration Lines

(a) Location of the regeneration lines is in the valve tunnel adjacent to and in front of the demineralizer unit cells. The regeneration lines are used to sluice the resin from the demineralizer units to the regeneration unit.

(b) Radiation level criteria is 2.5 mr per hour at one foot from the shield surface.

(c) Shielding requirement is two feet of ordinary concrete or its equivalent.

(4) Waste Storage Tanks

(a) Location of these tanks is immediately to the west of the demineralizer unit cells. The area of concern is located above the waste storage tanks where there are radioisotope laboratories.

(b) Radiation level criteria is 2.5 mr per hour at one foot from the shield surface.

(c) The shielding requirement above and to the sides of the waste storage tanks is two feet of ordinary concrete or its equivalent.

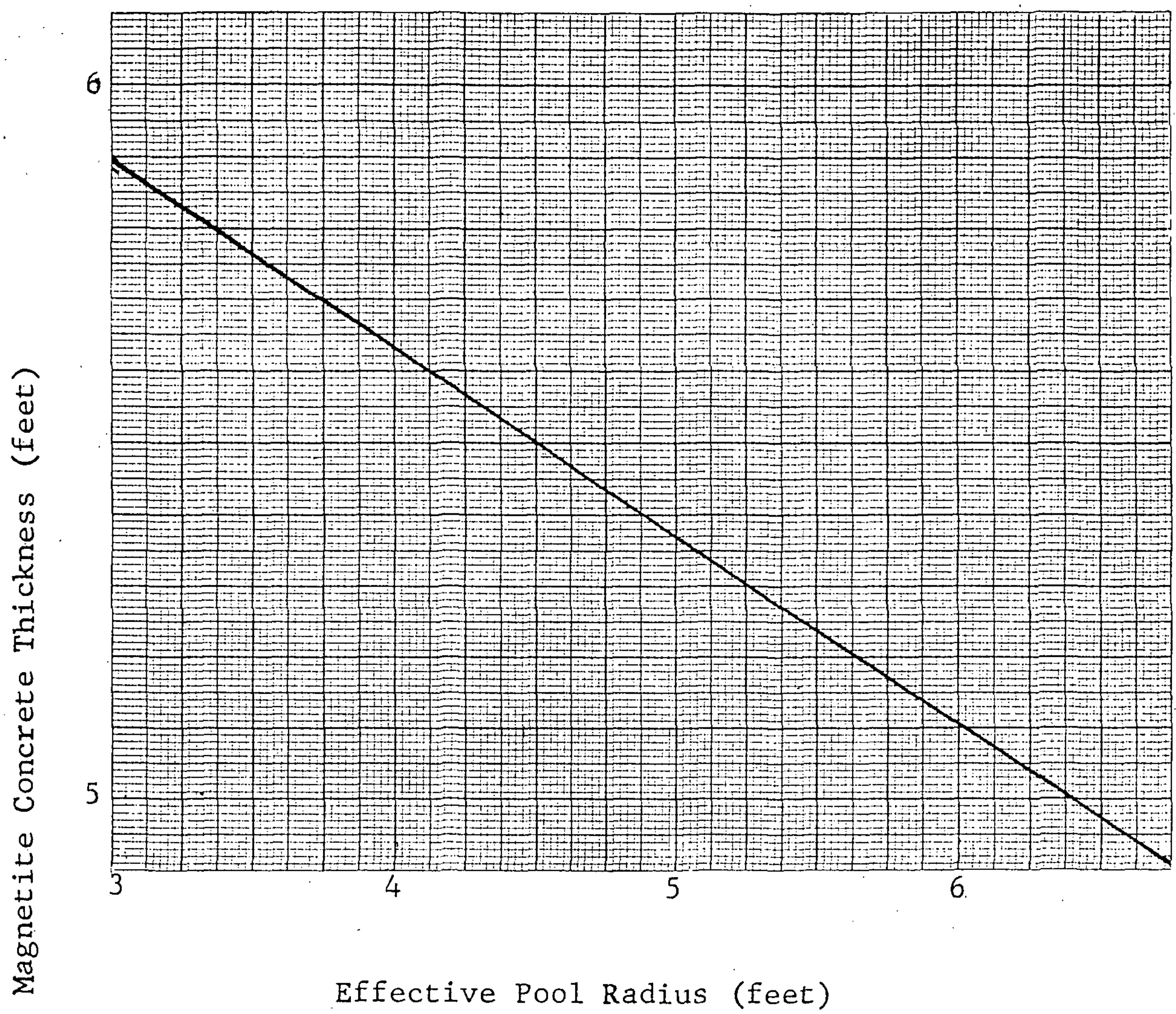


Figure 6.1 Biological Shield Magnetite Concrete Requirements versus Effective Pool Radius (Reactor Power 10 MW)

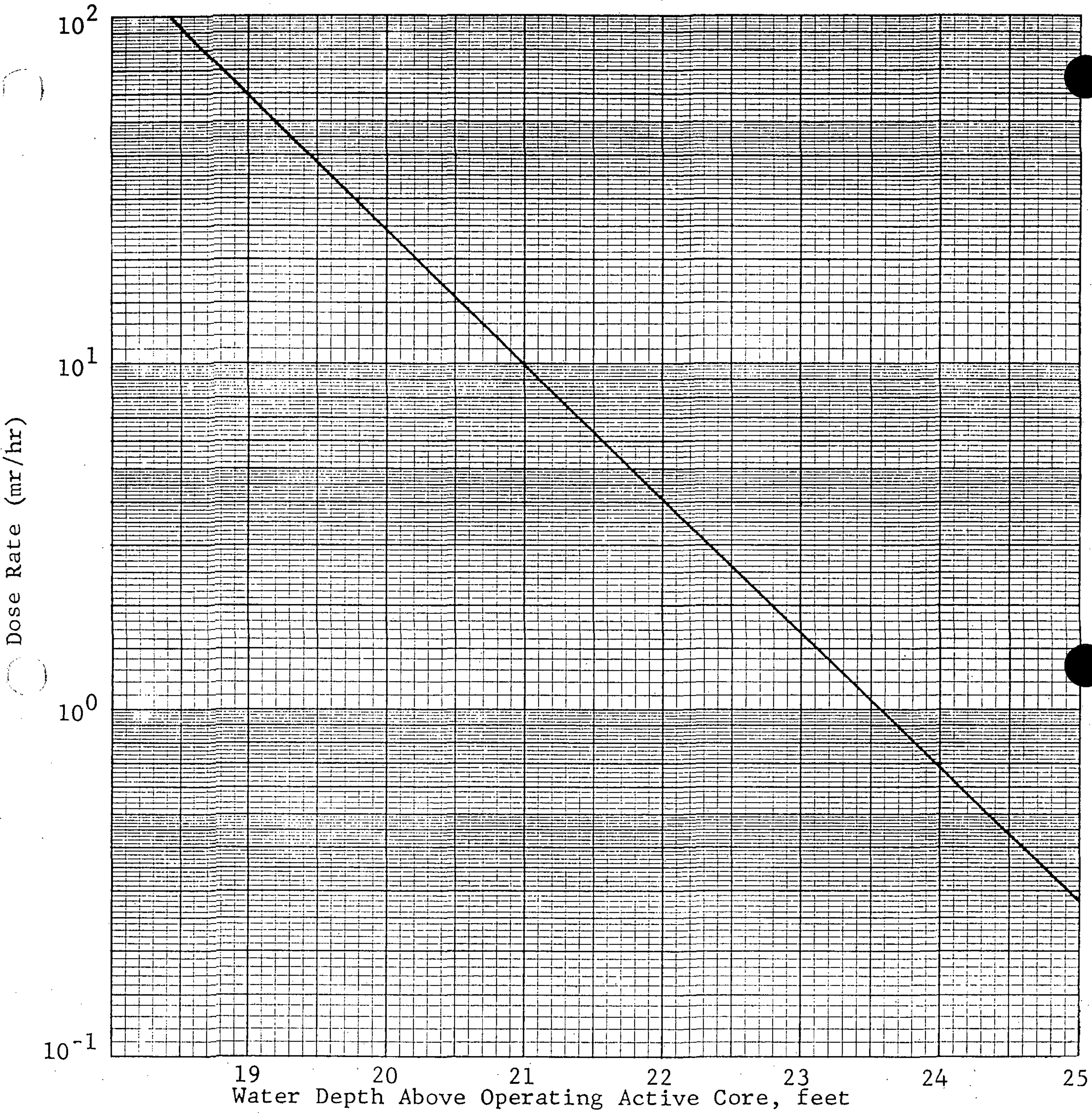


Figure 6.2 Direct Penetration Gamma Ray Dose Rates as a Function of Water Depth Above the Active Core Region (Reactor Power 10 MW)

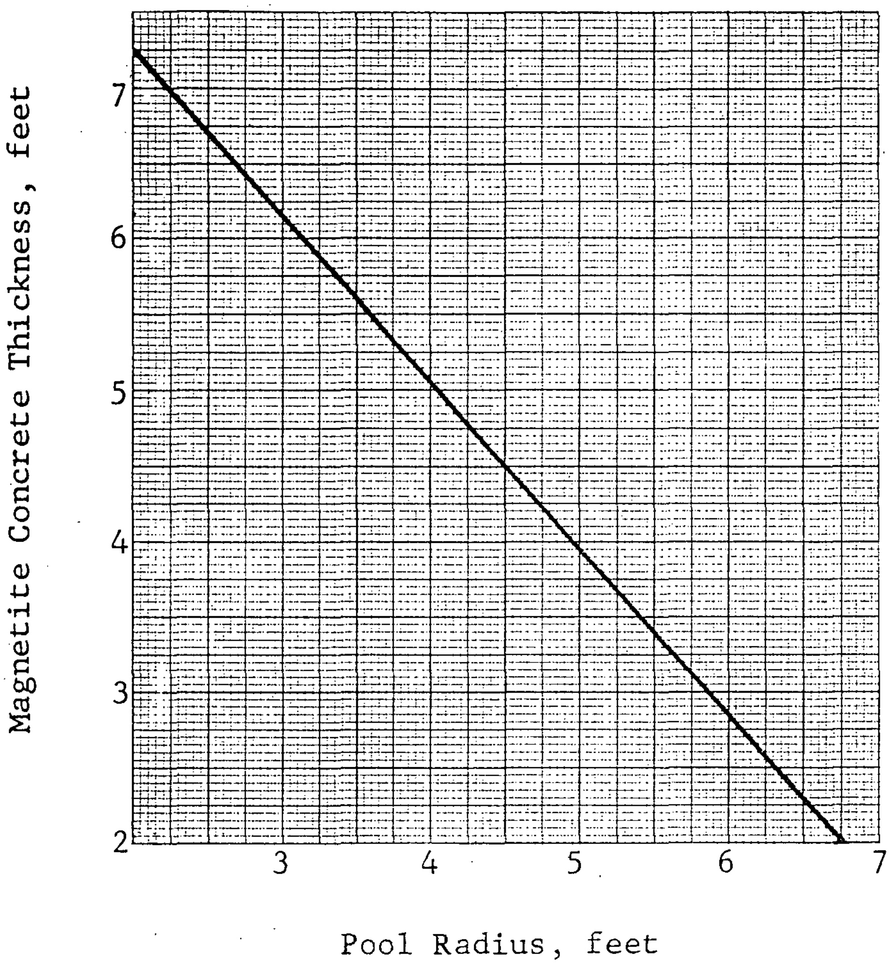


Figure 6.3 Magnetite Concrete Biological Shielding Requirements versus Effective Pool Radius to Achieve One-Tenth of Dose Rate Criteria from Fast Neutron Dose (Reactor Power 10 MW)

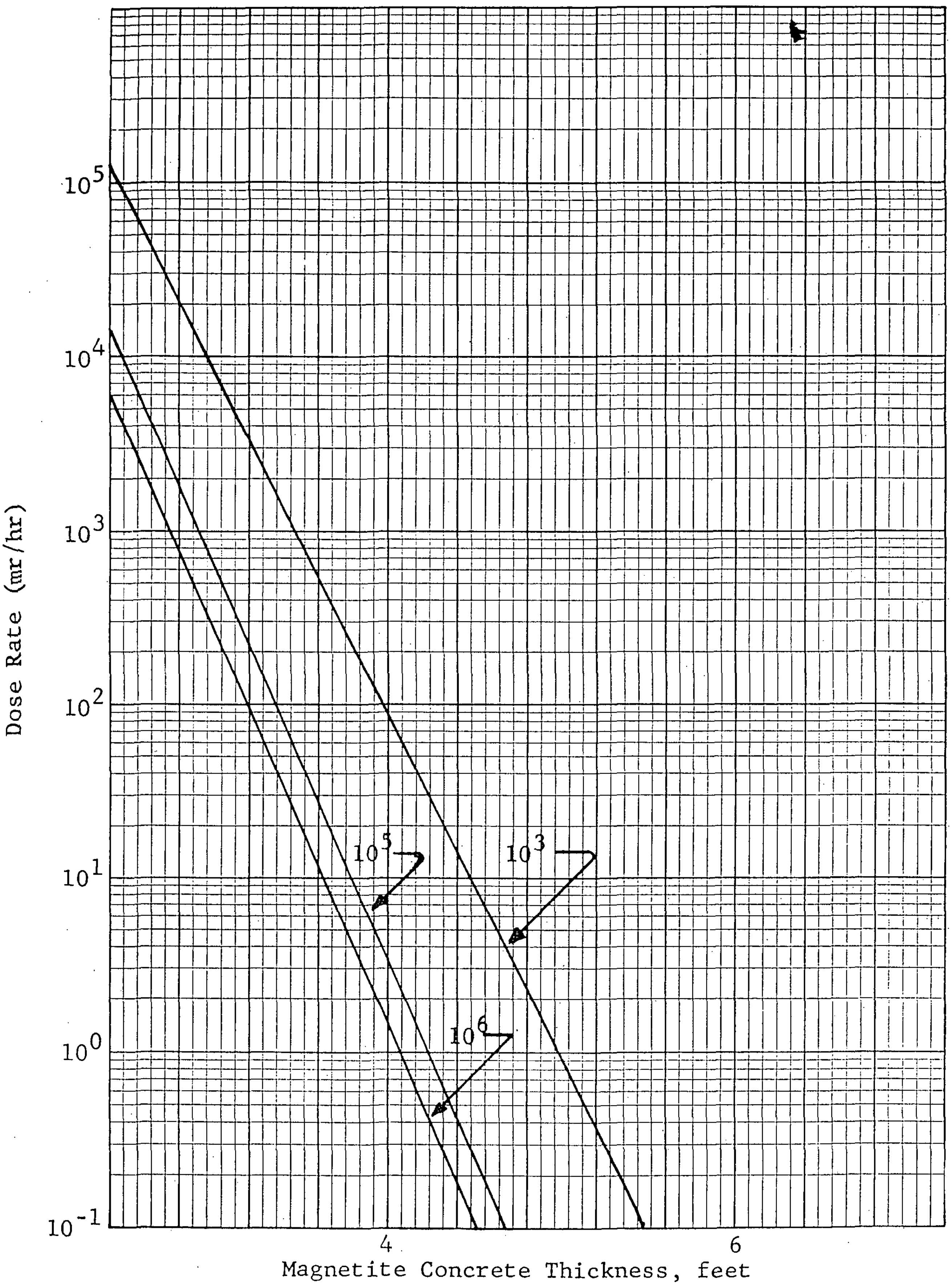


Figure 6.4 Spent Fuel Element Dose Rates Through Magnetite Shielding for a Single Element as a Function of Fission Product Decay Time (400 MWD Operation)

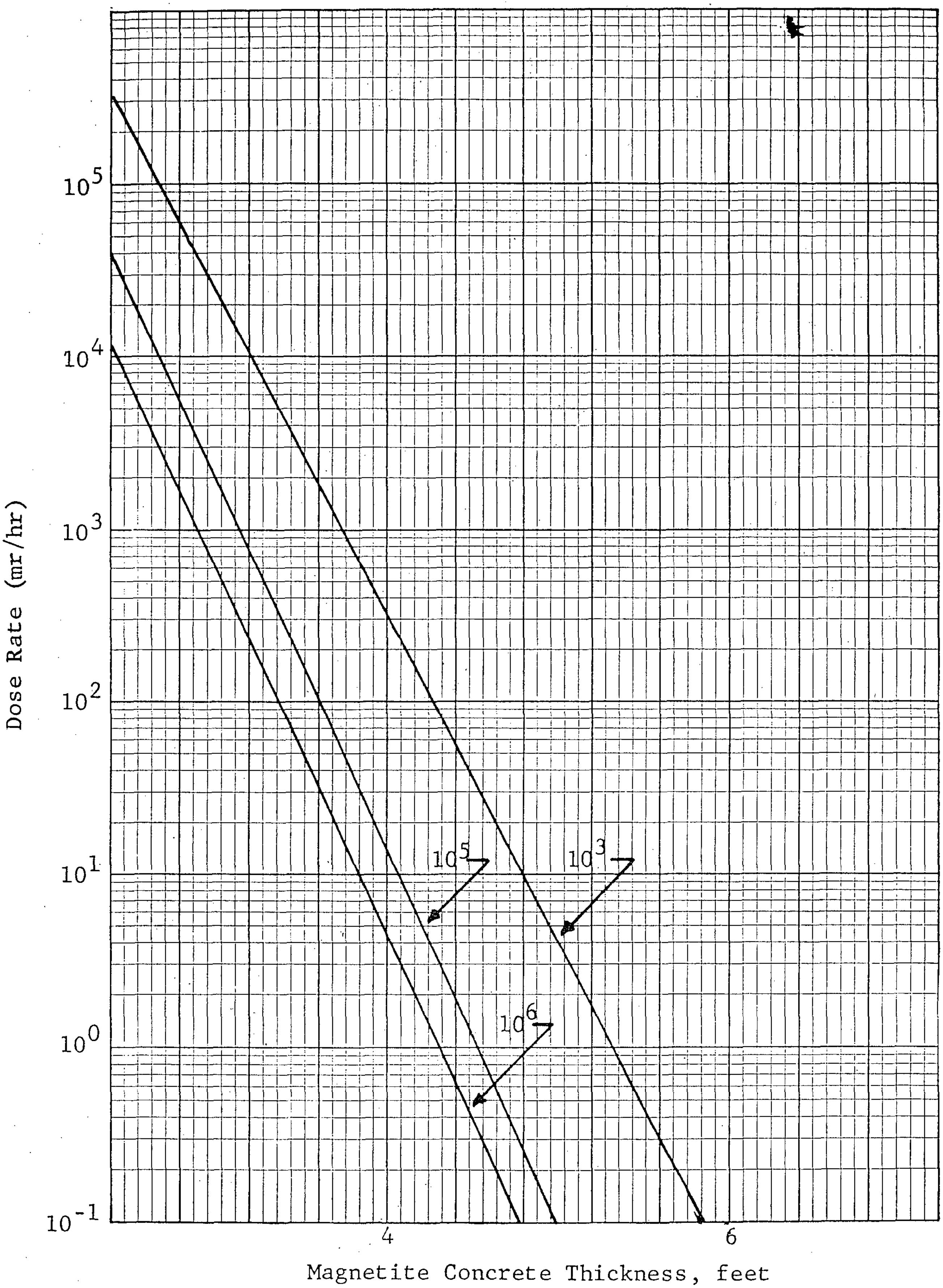


Figure 6.5 Spent Fuel Element Dose Rates Through Magnetite Concrete Shielding for Four Elements as a Function of Fission Product Decay Time (400 MWD Operation)

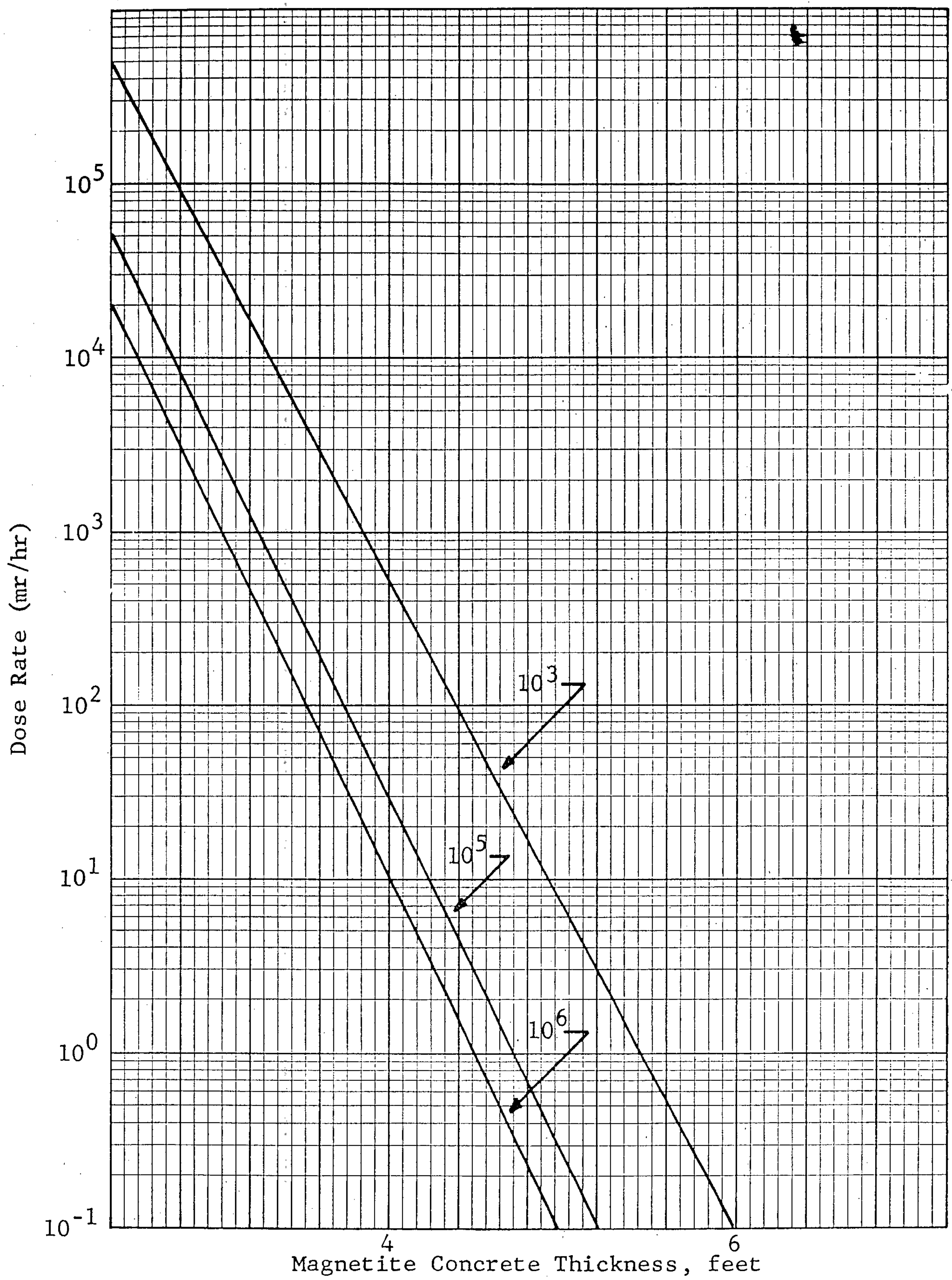


Figure 6.6 Spent Fuel Element Dose Rates Through Magnetite Concrete Shielding for Eight Elements as a Function of Fission Product Decay Time (400 MWD Operation)

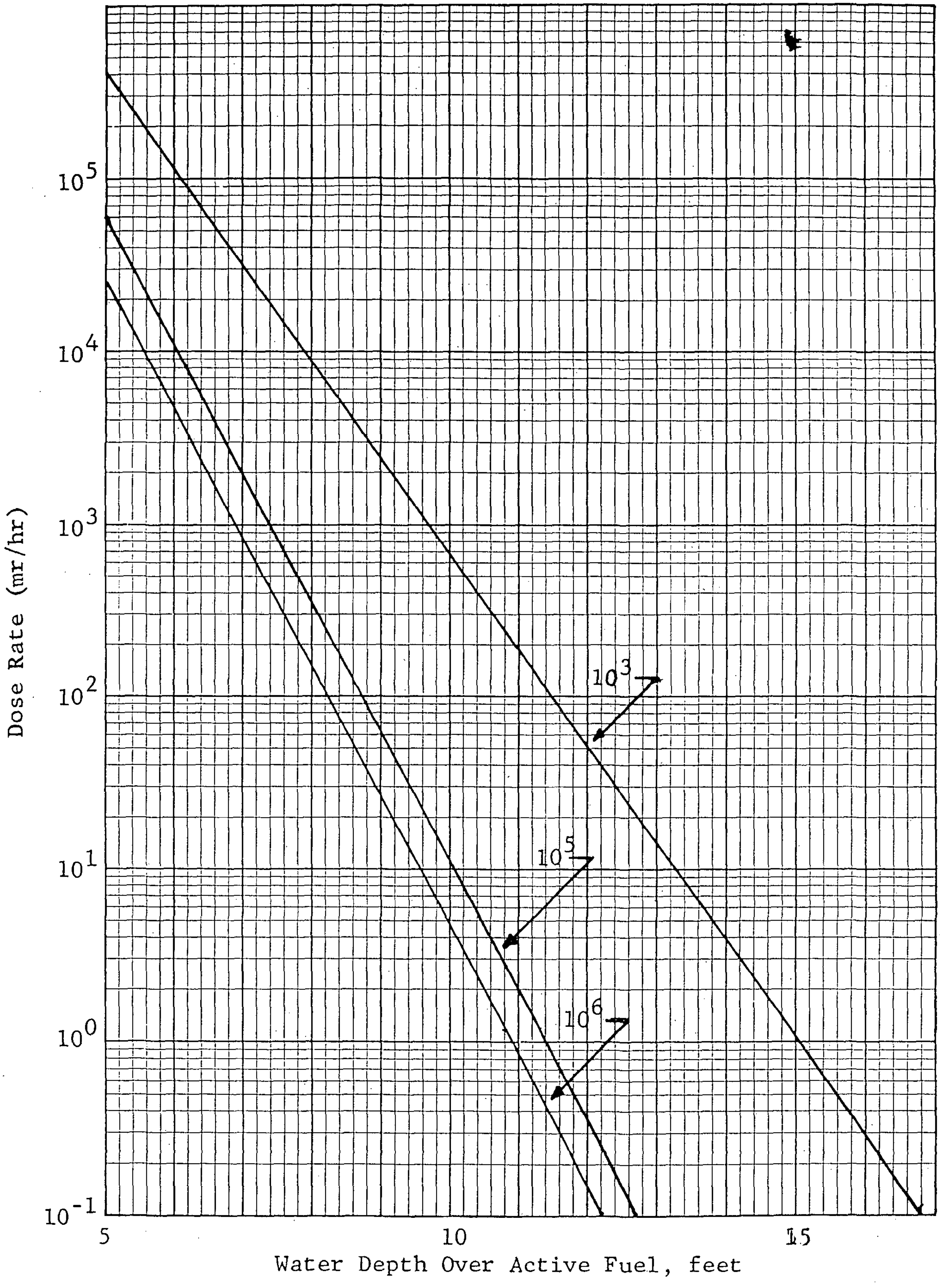


Figure 6.7 Spent Fuel Element Dose Rate Through Water Shielding as a Function of Fission Product Decay Time-Horizontal Element (400 MWD Operation)

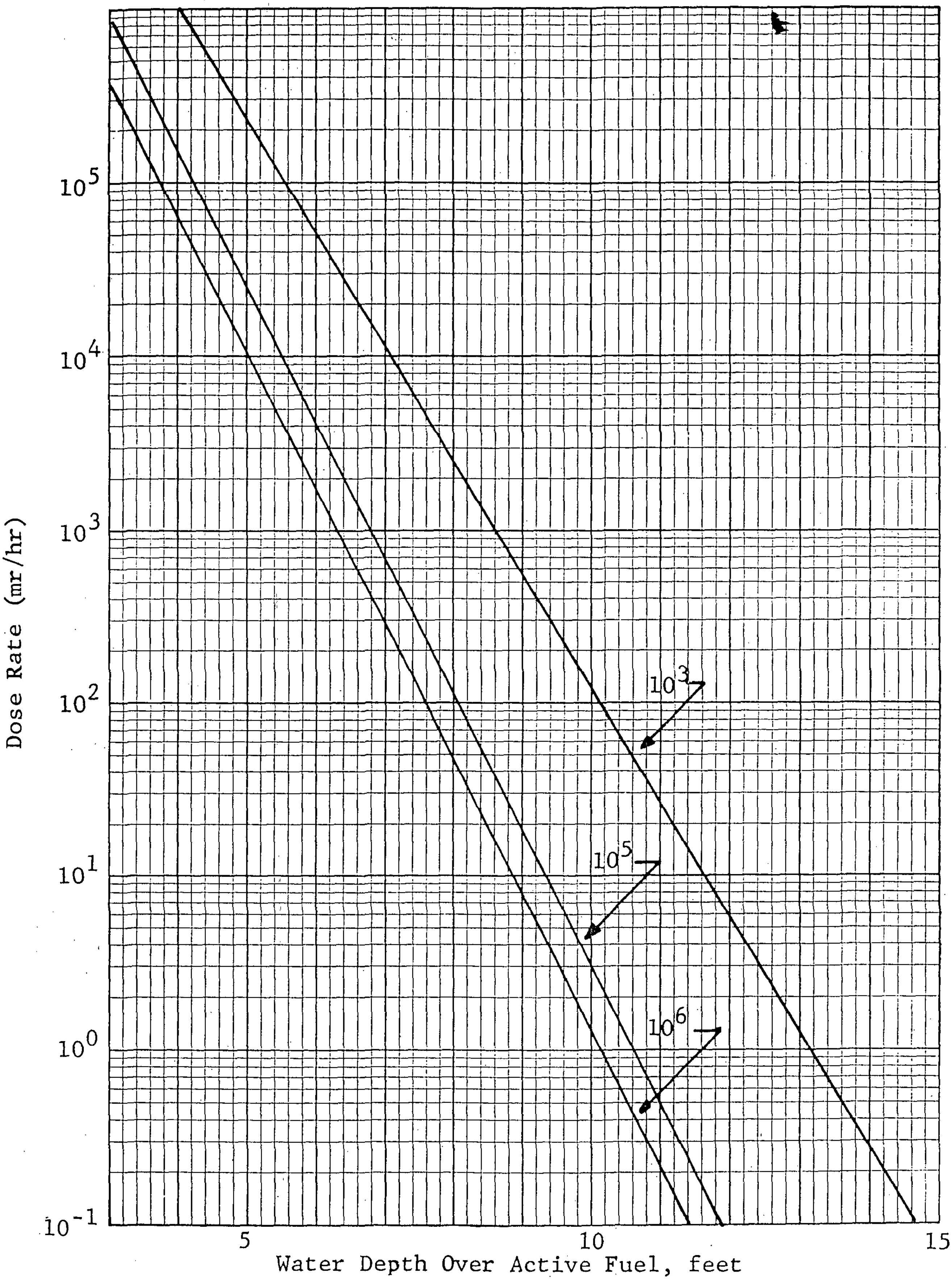


Figure 6.8 Spent Fuel Element Dose Rate Through Water Shielding as a Function of Fission Product Decay Time-Vertical Element (400 MWD Operation)

7.0 AUXILIARY SYSTEMS

7.1 Reactor Auxiliary Systems

7.1.1 Decay Heat Removal

The Missouri Research Reactor has an emergency decay heat removal system consisting of an in-pool heat exchanger and associated valving located within the reactor pool. The function of this system is to assure adequate decay heat removal from the core coolant system in the event of pump failure or isolation.

In the event of a loss of coolant flow by pump failure, isolation or electrical power failure the in-pool heat exchanger will dump the decay heat. The in-pool heat exchanger operates on a gas actuated valve system such that a loss of flow within the primary loop actuates valves which permit the coolant water within the pressure vessel to circulate through the in-pool heat exchanger. This system is described in Section 5.2.7 and analyzed in Section 5.6.

It is intended that the operation of the in-pool heat exchanger will be checked once each year during reactor operation to determine that it satisfactorily dumps the core decay heat to the reactor pool in the event of loss of flow. This condition will be simulated by power operation followed by shutting down of the main coolant pump.

1990-1991
1989-1990
1973-1974

7.1.2 Invert Loop (Siphon Break)

In an accident in which one does not have loss of flow but has breakage of the effluent or influent line to the reactor ~~core, the invert loop standing within the pool is immediately vented to the atmos-~~

". . . core, the invert loop acts as described in Appendix B of Addendum 4 to the HSR to insure that the core remains covered."

~~system is described in Section 5.2.5.~~

1973-1974

7.1.3 Fuel Element Failure Detection System

~~A controlled amount of reactor primary water in the clean-up system is bypassed around the demineralizers~~

A controlled amount of reactor primary water in the clean-up system is bypassed around the demineralizer through a Tracerlab MWP-1A Fission Products Water Monitor.

This system measures principally the fission product activity associated with I-135 and contributions from I-132, I-134, and I-136. The system is on line continuously when the reactor is in operation. When out of service, the primary coolant is sampled at least once every four hours for evidence of fuel failure.

The output of the scintillation probe detector is fed into one channel of the Eberline Radiation Monitoring system (RMS II). This signal is displayed on a analog meter that is equipped with an adjustable set point trip to alarm at a predetermined radiation level.

~~an adjustable set point trip to alarm at predetermined radiation levels.~~

1990-1991

7.1.4 Emergency Power System

Attached to the southwest corner of the reactor laboratory building is an addition that houses the emergency diesel generator and provides space for future addition of a 1250 KW substation. The emergency generator is a 275 KW diesel engine driven unit. Operation of the engine and generator is automatic. It starts one second following failure of normal power. After reaching rated voltage and frequency, the unit will automatically assume the

1989-1990

1989-
1990

emergency electrical load. Upon restoration of the normal electrical power source, the emergency electrical load will be automatically shifted after an adjustable delay time and the engine will be stopped after an additional adjustable time delay.

The emergency generator (EG) is powered by an 855 cu. in., 395 h.p., diesel unit with a direct injection fuel system. The diesel EG is sized to meet current and anticipated loads with an excess capacity approaching 50% for future load additions. The unit is designed to assume the emergency load within seven seconds of a cold start.

The generator is rated for an output of 344 KVA (275 KW at 0.8 PF), 277/480 volt, three phase, 60 cycles. The emergency power generator will provide for the electrical requirements of the following systems:

(1) Reactor Control Room Instrumentation

(2) Personnel Entry Doors and Controls

(3) Supply and Exhaust Air Doors and Controls

(4) Facility Exhaust Fans (EF-13 and EF-14)

(5) Emergency Air Compressor

(6) Evacuation/Isolation Alarm System

(7) Fan Failure Warning Light System

(8) Communication and Paging System

(9) Exit Signs

(10) Isolated Lights

(11) Stairway Lighting

1989-1990

1995
1989-
1990
2002
2004

(12) Diesel Generator Electrical Controls

(13) Offgas Stack Monitor

"(14) Nitrogen Station Controls" 2002

The system will be tested once per week for 30 minutes to assure operability.

Periodic maintenance will be performed according to manufacturer's recommendations.

(15) Fire Protection System 2004

1989-1990

7.1.5 Reactor Loop Vent System

The reactor loop vent system provides for the venting of any gases released within the in-pool piping.

A vent tank installed within the pool collects gas through 1/2 inch tubes which connect to the highest points on the invert loop and the in-pool heat exchanger. The vent tank is fitted with two liquid level controllers. ~~The higher of these vents the tank through a relief valve whenever the liquid level~~

"The higher of these vents the tank through a solenoid valve to a pressure regulator whenever the liquid level in the tank recedes to a preset elevation. If the vent tank liquid level continues to recede, the lower controller operates a second relief solenoid in parallel with the first. Operation of the lower controller also initiates alarm and rod run-in."

~~controller also initiates alarm and rod run-back.~~

1995

7.1.6 Emergency Air Compressor

~~In addition to the normal laboratory air compressor,~~

"In addition to the normal laboratory air compressor, the inflated gasket seals for the equipment entry, personnel entry, and the fifth level isolation doors are connected to a standby compressor located in the mechanical equipment room on the reactor fifth level."

~~mechanical equipment room on the reactor fifth level.~~

1995

1995
1981-
1982

This compressor has a capacity of 15.8 CFM at 100 psi. There is an 80 gallon air receiver tank connected to the system. In the event of failure of the laboratory compressor this compressor will maintain sufficient pressure for operation of the door seals. A check valve isolates this line from the balance of the normal air system.

~~Both of the reactor exhaust isolation valves are of~~
"One of the reactor exhaust isolation valves is of the air open-closed type. The air for its operation is backed-up by a smaller compressor and accumulator located on the fifth level."
~~accumulator unit located at each valve.~~ 1995

Complete operational checks of this system will be performed "prior" to startup. 1981-1982

7.1.7 Emergency Pool Fill

There is a "8 inch wet fire line extending from the fire protection water loop outside of the laboratory building directly into the reactor building. This line provides an emergency raw water supply for pool filling. The line enters the containment structure via the **seal trench."**. The line terminates in a goose-neck extending over the curb of the reactor pool and pointed down toward the core. The line terminates about 10 inches above the pool surface. There is no possibility of back-siphon action.

The flood valve controlling the flow from this emergency pool fill line is recessed in a box adjacent to the curb at the east end of the pool.

1995
1964-
1970

The valve moves from closed to full open with a quarter turn of the lever handle. This 4 inch line will provide an emergency water flow at a rate in excess of 1,000 gpm. This is an addition rate which is completely adequate to maintain more than 3 feet of water above a completely severed 6 inch beamport.

ADD

The water pressure is monitored by a pressure switch which will alarm on low pressure, sounding an audible and visual alarm in the control room indicating that the water pressure is insufficient to produce 1,000 gallons per minute flow.

1964-
1970

7.1.8 Fuel Handling System

Fuel handling will be entirely manual using specially designed remote tools.

PLACE AFTER FIRST SENTENCE

"All fuel handling will be done in accordance with SNM Control and Accounting Procedures and as outlined in the 10 MW Standard Operating Procedures." 1995

~~New fuel, when first received, will be placed in a storage vault provided with criticality detection~~

"New fuel, when first received, will be placed in a storage vault provided with security against unauthorized entry. Fuel will be stored in the vault in critically safe geometry such that the calculated k_{eff} is less than 0.9." 1995

~~entry. The fuel will be stored in the vault in the original shipping containers.~~ If it is necessary or desired to store the fuel in other than the shipping container, special racks will be provided assuring ~~stor~~ storage in a subcritical geometry." ~~nner.~~ 1995

During inspection of the new fuel the elements will be handled one at a time and each replaced to its proper storage rack prior to inspecting the next element.

As fuel is required for loading into the reactor each element will be transported individually from the storage vault to the reactor. Only one element will be allowed out of proper storage or its position in the reactor at one time.

1995

The special source materials custodian will assure that proper documents are available at receipt of the fuel and it shall be his responsibility for a complete and accurate fuel inventory.

Irradiated fuel does not leave the reactor pool until after it is loaded into suitable casks for shipment. During all fuel transfers sufficient water shielding is provided to maintain acceptable radiation dose levels to personnel.

~~The highest dose rate during the spent fuel transfer operation will be approximately 100 mr/hr during the short interval of time required to lift the element over the lip of the pressure vessel. This will not require over one minute to accomplish per element.~~

~~To remove fuel the reactor will be shut down and a decay period allowed. The pool level will be lowered to the lower bridge (approximately six feet) and, by~~

"The highest dose rate during spent fuel transfer is approximately 25 mr/hr during the short interval of time when an element is passed into the weir area for storage and when an element is raised to clear the top of the pressure vessel.

The reactor must be shut down and the pressure vessel head removed using special tools prior to removing fuel from the reactor. Reactor operators working from the upper bridge will move fuel in and out of the core to designated spent fuel areas one element at a time. The normal fuel handling tool is air operated, has buoyancy assist tanks and provides position indication of latching prior to movement of an element.

Following each refueling sequence a test to assure fuel elements are fully seated is performed and inventory management checks are completed prior to bolting the pressure vessel head in place."

1995

~~or all elements will be turned upside down and reloaded. With the water at the refueling level a minimum of eight feet of water will be between the operator and any single fuel element during the transfer.~~

1995

~~Following fuel loading the pressure vessel head will be replaced and the water level in the pool raised to normal operating height. Any spent fuel in the pool storage racks will then be transferred to the long-term fuel storage racks adjacent to the reactor gamma irradiation facility (i.e. in the spent fuel storage pool).~~

Transfer of elements to the shipping cask will be done manually with the cask underwater resting on the pool shelf between the reactor pool and the spent fuel storage pool. The cask will be decontaminated prior to release for shipment.

7.1.9 Core Coolant Clean-up System

There are two demineralizer loops associated with this reactor. One serves to provide clean-up of the reactor water, the second to provide clean-up of the pool water. Both are operated at an inlet flow of 50 gpm. Both systems utilize the same size demineralizer beds. As will be pointed out in the paragraphs which follow, the actual demineralizers are interchangeable from one system to the other by means of a valving arrangement. In the paragraphs which follow the systems will be described together under the general heading of core coolant clean-up system but the differences that exist between the core coolant and the pool coolant clean-up system will be pointed out for the reader.

A portion of the reactor water return stream from the heat exchanger flows to a holdup tank which provides a minimum holdup time of two minutes at a flow rate of 50 gpm. ~~The holdup tank is provided~~

1995
1972-
1973

~~with an off-gas vent which is manually operated at intervals to assure limited gas accumulation.~~ DELETE In the discharge line of the holdup tank is a centrifugal 50 gpm pump which serves to overcome the head losses in the ~~bypass~~ DELETE demineralizer system. 1995 1995

The coolant cleanup streams are filtered both before and after passing through the demineralizer beds. The filters are of the replacable cartridge type with a stainless steel shell, base and cover. They are sized to remove "1 micron" or larger particles. The filters are equipped with valving which permits isolation during exchange of cartridges. 1972-1973

The water purification system consists of three mixed bed demineralizers capable of demineralizing water from the reactor loop and the pool loop each at a flow of 50 gpm. Each demineralizer is sized to remove 1500 grains hardness per day from water of temperature not to exceed 140°F. ~~with 40 days between regenerations.~~ DELETE The effluent from any one of these demineralizers will contain not more than .05 ppm of sodium and .05 ppm of silicon. ~~These units~~ 1995

"These units are capable of maintaining the bulk reactor and pool water at conductivities of less than two micro-mho." 1995

~~micro mho.~~ Each of these three demineralizer beds is located in a separate shielded cell and all are connected to a common manifold distribution system of aluminum pipe. Any two of the three units are in service simultaneously, one connected to the pool loop and the other to the pressurized reactor loop. The third unit is held in standby. The common

1995
1972-1973

BOLD FACE IS 1995 REV

manifold system permits complete interchangeability with regard to function of the three units.

All valving has been installed with reach rods through a two foot shielding wall and is positioned such that it matches the various shielding penetrations. In addition to a two foot shield wall separating the valve handles from the valve bodies there is one additional foot of shielding behind the valve gallery such that the minimum shielding from demineralizer cell to the operator of this equipment is three foot of concrete. The cells are arranged in such a manner that the demineralizers are shielded by means of stub wall shadow shields from a limited access passageway behind the cells.

The cleanup system is also equipped with a rubber-lined resin storage tank which is piped into the resin sluice lines. The resin storage tank enables the use of an additional (fourth) bed of resin in the system and significantly increases the decay time from resin depletion to resin replacement. The longer decay time for the activity in the depleted bed greatly reduces the activity levels sent to the radioactive liquid waste system during resin replacement.

ADDED FROM 1972-1973 REV

In a similar manner, the regeneration station and the filters for the demineralized water are located in separate cells with provisions for remote valving in and out of the system. The regeneration station consists of a large tank for the holdup of resins which are sluiced from any one of the three demineralizer units by means of water carrier. ~~After the resin has been sluiced into the regenerant shell the bed is separated remotely by means of a~~

"After the resin has been sluiced into the regenerator column, R-200, the bed is dumped to a resin drying apparatus and a new, pre-regenerated, bed is loaded for service."

~~solution piped to the regenerant station through a plastic (PVC) piping system.~~

1995

~~The regeneration station has been designed to provide~~

"The regeneration station (R-200) has been designed to be a central transfer access point and piping and valves are provided to permit the resin transfer to and from any of the DI tanks, including the storage tank.

All effluent (wastes) from the R-200, including sluicing water, are directed into the radioactive liquid waste storage system."

~~station into a disposal container.~~

1995

1995

~~The caustic and acid mixing tanks for the regeneration station are located in the cooling tower building approximately 150 feet south of the reactor building. The regenerant solutions are provided to the regeneration station by means of pumps located at the mixing tanks through a PVC piping arrangement. All effluent (wastes) from the regeneration station, including the sluicing water are directed into the hot waste storage system.~~ 1995

The total water purification system for both the reactor and pool loop is contained behind 3 foot concrete shielding walls and is operated remotely. By means of reach rods stepped to exclude streaming, one can divert flow from the core or pool loop to any one of the three mixed bed demineralizers.

"When a particular unit is depleted and in need of replacement it is valved off and the standby demineralizer is valved into that loop." 1995

~~demineralizer is valved into that loop. The de-~~

The depleted resin bed in the storage tank is then sluiced, by means of a water carrier, from the resin storage tank to the regenerator where it is dumped. The depleted resin bed in the demineralizer is then sluiced to the resin storage tank and the pre-regenerated bed is sluiced to the empty demineralizer which is placed in standby. 1995

~~separated, recharged, mixed and sluiced back to the demineralizer shell.~~

Move sentence here

All of these operations are performed remotely by means of instrumentation and valving positioned external to the shield wall. The physical arrangement of these facilities is shown on Figure 3.1, the beam hole level plan.

~~It will be noted that access to the demineralizer~~

"It will be noted that access to the demineralizer cells is restricted by a number of wire gates each of which is locked and remotely alarmed in the control room." 1995

~~of which is locked with the keys in the possession of the reactor supervisor.~~

1995

7.1.10 Pool Coolant Clean-up System

The general operation of the pool and core coolant demineralizer systems was described in some detail in the preceding section. The only significant difference between the reactor core clean-up system and the pool clean-up system is the presence of the holdup tank in the core loop to the demineralizer. ~~The pool coolant clean-up line does not have a holdup tank.~~ DELETE 1995

"The pool coolant cleanup line does not have or require its own holdup tank, but rather has its suction at a point in the pool system before the heat exchanger and after the bulk pool water has passed through the pool water hold-up tank." 1995

This water, which is approximately 5 degrees warmer than the main coolant stream return, is returned to the pool about 2 feet below the surface. The purpose of this is to create a blanket of warm water at the pool surface to reduce mixing rates of the bulk pool water to the surface and hence reduce pool surface dose rate due to pool water activity.

7.1.11 Pool Skimmer System

"The pool skimmer system consists of an adjustable skimmer surface box" 1995 ~~skimmer tank~~ located within the reactor pool, a pump located in the mechanical equipment room for the circulation of the skimmed water, a filter for the removal of particulates skimmed from the surface of the pool and piping and valves to accomplish this skimming filtration operation. This skimmer system provides continuous clean-up of the reactor pool surface. The skimmer piping and pump are also used to drain the pool water from the portion of the reactor pool between the upper and lower bridges.

1995
1972-
1973

In the event it is desired to drain this water the pool skimmer is valved to a pump by means of two valves located approximately 4 feet below the surface of the pool. In addition, a valve located downstream from the skimmer pump is opened to allow the pumping of the pool water to the demineralized water storage tank located in the south tower.

The actual skimmer assembly has been designed to be adjustable over a nine inch span of pool surface level. The skimmer pump is a centrifugal unit sized to pump 50 gpm at a discharge head sufficient to raise the pool water to the demineralized water storage tank.

The skimmer filter is of the replacable cartridge type of flow capacity of 50 gpm. ~~The skimmer filter has~~ **"The skimmer filter has removable one micron filter cartridges."** 1995

The 50 gpm return flow from the skimmer system re-enters the pool via the six beamports. ~~and the spent fuel storage pool.~~ DELETE A 2 inch aluminum header line is in the shield circling the reactor pool. At each beamport a 1/2 inch line takes water from the 2 inch header, into the gap between the fixed port liner and the movable port liner. This water then flows into the pool volume. The maintenance of this flow should minimize corrosion problems in this area. 1972-1973

Two 1/2 inch lines from the skimmer return header enter the bottom of the spent fuel storage pool.
"These lines have been capped to prevent siphoning of the fuel storage pool when the pool level is lowered."
~~In the spent fuel storage pool.~~ 1972-1973

1973-
1974
2008

7.1.12 Closed Circuit Television System

A closed circuit television (CCTV) system is installed in the reactor facility which allows the control room operators to monitor selected areas of the reactor containment and laboratory buildings. 2008

complete with wide angle lens, one camera complete with normal lens. one camera complete with semi-

"The system is capable of providing a different view with any one of four cameras at the selection of the reactor operator. The cameras are of the stationary mount type. One camera is installed in the personnel airlock for surveillance of personnel entering and leaving the reactor building. A second camera can be positioned to view a particular area of the beamport floor. The third camera views either the mechanical equipment room or the cooling tower tunnel. The fourth camera is located within the fuel storage vault for surveillance of this area." 2008 1973-1974

cameras are of the stationary mount type. Each camera is positioned such as to provide a view of a particular area of the beam hole floor. The reactor operator can monitor experiments being carried out on the beam hole floor by means of the TV camera monitor system.

The video monitor for this system consists of a rack mounted 14 inch screen located in the reactor instrument panel. It is designed to provide a clear bright picture under continuous duty operation. A front access panel allows control of horizontal and vertical hold, vertical linearity, contrast, brightness, and focus. 2008

7.2 Facility Auxiliary System

7.2.1 Make-up Water Demineralizer

A general purpose demineralized water supply system is provided to meet the needs both of the laboratories and the coolant water systems of the reactor.

1995

~~This system for demineralized water supply includes a pre and post filter, a mixed bed demineralizer unit, common piping and regenerant tanks as well as the necessary connecting piping and isolation valves. Demineralized water from this system is pumped to two 7,000 gallon demineralized water storage tanks located in the south tower external to the reactor containment building. The position of these tanks with respect to the containment structure is evident on the fourth and fifth level plan, Figure 3.3. On this plan these tanks are denoted as storage tanks.~~

~~The make-up demineralizer provides water for the initial filling of the pool, for the reactor loop fill and make-up, for pool make-up additions, for~~

"This system consists of a water softening unit, a reverse osmosis (RO) unit, and a mixed bed demineralizer system. Demineralized water from the system is pumped either to a pair of 7,000 gallon storage tanks located in the south tower (denoted as storage tank in Figure 3.3), or the facility demineralized water storage tanks located in the north tower. The 7,000 gallon tanks are used for reactor and pool make-up and storage, and the facility tanks are used for clean water laboratory use.

The unit is capable of producing effluent water of less than 0.1 ppm total hardness.

In addition, the DI-300 mixed bed demineralizer is maintained as a backup. This system includes a pre- and post-filter and a mixed bed demineralizer unit as well as the necessary connecting piping and isolation valves."

1995

~~0.1 ppm total hardness.~~

~~The demineralizer is a mixed bed unit with a manual regeneration station attached to the shell. It is a standard unit with the exception that the mixing tanks for acid and alkali are not located at the regeneration station. These tanks are located in the cooling tower building south of the reactor facility. The alkali and acid for regeneration are piped via a PVC piping system from the pumps at the mixing tanks to the regeneration station.~~

1995
1981-1982
1969-1970
2008

~~The pre and post filters on this demineralizer are of the replaceable cartridge type capable of removing 25 micron or larger particles from a continuous flow of 30 gallons per minute. This demineralizer is capable of operation at pressures up to 125 psi.~~ 1995

7.2.2 Demineralized Water Storage

Two 7,000 gallon ~~lined~~ ~~st~~ stainless steel ~~tanks~~ ~~cs~~ for demineralized water are positioned in the south tower of the reactor building, external to the containment structure. ~~These tanks perform two services. They provide a 7,000 gallon source of supply for make-up water to the reactor and pool coolant loops as well as the supply of demineralized water to the various laboratories. The second tank~~ 1981-1982 2008

"They provide 14,000 gallon capacity to contain that water which is removed from the pool when the pool level is reduced to the lower fuel handling bridge or below. About 2,500 gallons inventory is kept for normal operations."

~~pool filling purposes and secondly it can be isolated and used to contain that water which is removed from the pool when the pool level is reduced to the lower fuel handling bridge.~~ 1995

~~Each of these demineralized water storage tanks is equipped with water level readout sensors with the~~

Each of these demineralizer water storage tanks is equipped with a pressure gauge which is connected to the bottom of the tank and is calibrated in gallons. These gauges are located in the south inner corridor. Connected with these gauges are pressure switches which sound an audible and visual alarm in the control room to keep the operator aware of the status of demineralized water storage.

~~equipment is located in the control room and is equipped with both high and low level alarm so the operator is aware of the status of demineralized water storage.~~ 1969-1970

1995
1989-
1990
2004

"The three facility DI water storage tanks are located in the north tower and are made of 304 stainless steel and have a capacity of 420 gallons each. The tanks are all cross connected resulting in a total capacity of 1260 gallons. This system is used only for supplying non-contaminated water to the facility laboratories. A high level alarm and make-up system trip is incorporated to prevent over filling and a low level alarm alerts the Operations staff of the need to refill." 1995

7.2.3 ~~General Water Softener~~

~~A pressure type fully automatic down flow sodium zeolite water softening system manufactured by Elgin Company is installed in the cooling tower building. All incoming water from the University main to the facility is softened.~~

~~The softening system has the following characteristics:~~

DELETE

~~Number of softener units~~	~~2~~
~~Working pressure, psig~~	~~100~~
~~Size of main piping, inch~~	~~3~~
~~Exchange value of zeolite in grains/cubic feet~~	~~20,000~~
~~Cubic feet of zeolite per unit~~	~~54~~
~~Continuous rate of flow of each unit, gpm~~	~~150~~
~~Maximum rate of flow of each unit, gpm~~	~~194~~
~~Capacity of each softener between regenerations, gallons~~	~~106,000~~

1989-1990

DELETE SECTION AND REPLACE WITH NEW

7.2.4 ~~Fire Control System~~

~~The~~ "The Research Reactor Facility has eleven fire hose cabinets ~~ose cabinets~~ ~~strategically located throughout the building. These hose cabinets are connected to a dry fire system which connects to three siamese fittings located outside the building, one each at the north, west and south entries. Fire hydrants~~ 1995

2004

7.2.4 Fire Protection System

The MURR fire protection system is designed to protect the facility and staff, and to mitigate any property loss in the event of a fire. The system provides two primary functions: (1) detection, which affords an early warning of an actual or potential fire condition by a combination of heat, smoke, and remote manual devices, and (2) suppression, which incorporates a normal sprinkler system with a pre-action system that is used in areas with sensitive electronic equipment, and a deluge, non-freezing system used in the cooling tower. It should be noted that fire protection is not required to accomplish a safe shut down of the reactor or to maintain a safe shutdown condition.

The fire detection system is a combination of thermal, photoelectronic, and ionization-type sensors, flow switches, and manual pull stations. A central control station, located in room 204, and a repeater station, located in the reactor control room, monitors each system component. These stations will annunciate an alarm if any component is not in its normal condition.

The fire suppression system is a combination of many types of systems: a deluge system used in the cooling tower; a pre-action system used in areas that contain highly sensitive electronic equipment; a dry fire main system used in the reactor containment building; a traditional sprinkler system used throughout the rest of the laboratory building; and a damper isolation system for the facility ventilation exhaust system. The containment building fire suppression system consists of three fire hose cabinets connected to a dry fire main. Cross-connecting it to the rest of the facility's wet system by a manual isolation valve located in the laboratory basement can flood this system. 2004

The fire protection system receives a virtually unlimited supply of water from the combination University fire and domestic cold water main. Four (4) Siamese or storz hose fittings are also connected to the MURR fire main. These fittings are located outside the facility and they facilitate connecting a pumper truck between the fittings and fire hydrants, which are located in the vicinity of the hose fittings, thus providing an additional water supply path to the fire main. In addition, fire extinguishers are strategically located throughout the facility.

The fire protection system is powered from the emergency electrical distribution system (ELP-2). The system also has a self-contained 24-hour battery backup.

1995
1973-
1974
2004

~~are located outside the building in a vicinity near the external siamese connections to facilitate connecting a pumper truck between the hydrants and the hose connections on the building.~~

DELETE ENTIRE SECTION AND REPLACE WITH NEW

2004

7.2.5 Battery Operated Emergency Lights

~~The~~ **"The reactor and laboratory has sixteen emergency battery lights** ~~ncy battery lights~~ strategically positioned throughout the building. Each light has a self-charging battery pack and a switching circuit to actuate the light upon electrical power failure.

1995

~~7.2.6 Local Weather Loop Teletype~~

~~A teletype receiver is to be installed in the facility lobby near the receptionist desk. This receiver will be tied into the local weather loop teletype system so that in the event of any emergency the facility staff will have complete weather information including barometric pressure, temperature, inversion conditions, wind direction and velocity and a prediction of any weather changes that may occur in the near future.~~

DELETE ENTIRE SECTION

1973-1974

7.2.7 Ventilation and Air Treatment Systems

The Research Reactor Facility building ~~complex~~ is totally air conditioned. The building air intakes are on the north and south faces of the

1995

Building

The Research Reactor Facility building complex is totally air conditioned. The building air intakes are located on the north and south faces of the east reactor containment building tower and include two roof top air handlers (RTAH) in the laboratory building roof [one mid way on each of the north and south corridors]. Building air is exhausted through a stack in the west tower.

INSERT AFTER THE WORD TOWER

THIS IS 1989-1990 REV THAT CORRECTS 1995 REV

1995

1981-1982
1989-1990

Air from the laboratory fume hoods is passed through a system of absolute filters prior to being mixed with reactor containment building exhaust air and passes out of the building through the stack in the west tower.

~~Reactor containment building air that is discharged~~

Reactor containment building air that is discharged to the atmosphere is thermal column cooling air, beamport ventilation air, air which is drawn from the surface of the pool and exhaust from the film irradiator shield box. Reactor containment building exhaust air is mixed with and diluted by the laboratory building exhaust air. The reactor cooling equipment room ventilation air and the pneumatic tube system exhaust air pass through filters and then also exhaust through the building stack.

~~exhausted directly through the building stack.~~

1989-1990 DELETED 1973-1974 REV

7.2.8 Laboratory, Shops and Auxiliary Support Systems

The Research Reactor Facility has an electronics shop and a machine shop each equipped and staffed to do most of the maintenance on all of the facilities systems and to fabricate special research equipment.

One laboratory ~~located within the reactor containment building~~ is equipped for use by the operators and staff to handle such problems as water analysis, foil counting, smear surveys, monitoring and similar routine laboratory functions. 1981-1982

Additionally, the Research Reactor Facility has library facilities, reproduction facilities, photographic darkroom facilities and catalogued storage of administrative and laboratory supplies.

1995
1981-1982
1974-1975
2001
2002
2007
2008

7.2.9 Intercommunication and Paging System

Add a syste incor to or

"The in stations as the paging Any sta the master

"The reactor facility utilizes two principal communication systems: a computerized telephone system and an intercommunication system that allows two-way communication between a master station and a staff station. A paging feature, which allows several different telephones to originate a page, is incorporated into the telephone system.

Master and staff station locations for the intercommunication system are shown in Table 7.1. The master station is in the reactor control room. Speakers for the paging system are located as shown in Table 7.2. Any staff station may be called from the master control station and any staff station may call the master control station. Voice paging may be accomplished from the master control station."

elephone m being able 1981-1982

eral staff . Speakers for ation may call ontrol station." 1995 2001

TABLE 7.1

INTERCOMMUNICATION SYSTEM MASTER AND STAFF STATION LOCATIONS

Master Stations

Room 205 DELETE Reception Desk 1995

"Master Stations	
Room 302	Reactor Control Console
Staff Stations	
Cooling Tower Basement	
Cooling Tower Entrance	
Laboratory Building Roof	
Room 101	Beamport Floor
Room 105	Hot Cell Work Area
Room 114 (2)	Mechanical Equipment Room
Room 115	Demineralizer Area
Room 216	Laboratory
Room 218	Laboratory
Room 227	Laboratory
Room 242	Laboratory
~~Room 271~~	~~Machine Shop~~ 2008
Room 278	Mechanical Equipment Room
Room 282	Electronic Technician Shop
Room 286	Airlock
Room 288	Health Physics Office
Room 307	West Tower Third Level
Room 501	Containment Building Fifth Level
Room 507	West Tower Fifth Level"
~~"Room 210~~	~~Laboratory Building Outer Corridor~~
Room 285	Containment Building Lobby"
Corridor C2002	Cyclotron Suite
Corridor C2000	Exit to Lobby
Room 2041	Laboratory Area
Room 287	Central Corridor 2008

* REARRANGE IN SEQUENTIAL ORDER

rol Console

oor
k Area
er Room
r Area 2001

oratory 1995

Machine Shops
th Physics Office"

Fuel Vault 1974-1975

2008
2002 2008
2007 2008

1995
2001
2007

~~Room 278 DELETE West Mechanical Equipment Room~~
~~Fifth Level Mechanical Equipment Area~~

TABLE 7.2

PAGING SYSTEM SPEAKER LOCATIONS

Containment Building

Room 101 (3)
Containment Building Third Level
Containment Building Grade Level (2)
Containment Building Fourth Level

Reactor Laboratory Building

Cooling Tower Grade Level	Room 241
Room 103	Room 242
Room 111	Room 244
Outside Room 114	Room 245
Outside Room 214A	Room 247
Outside Room 224	Room 251
Outside Room 228	Room 255
Outside Room 241	Room 257
Outside Room 244	Room 258
Outside Room 258	Room 259
Outside Room 264	Room 260
Outside Room 288	Room 262
Outside Room 293	Room 267A
Room 202	Room 271
Room 210, Lobby	Room 273
Room 212	Room 278 (2)
Room 215A	Room 280
Room 216	Room 281
Room 218	Room 288
Room 224	Room 299
Room 227	TOB-1
Room 232B	TOB-2
Room 238	TOB-3
Room 231	TOB-4
Room 231A	TOB-5
Room 231C	

North Office Addition

Corridor C2000	Room 2008C
Corridor C2002 (2)	Room 2009
Room 2005	Room 2009A
Room 2006	Room 2011
Room 2007	Room 2015 (17)
Room 2008	Room 2041
Room 2008A	

2001 2007 2001

7.3 Waste Disposal Systems

7.3.1 Gaseous Wastes Disposal

Gaseous wastes are disposed of through the building ventilation and air treatment system described in Section 7.2.7. Minimum concentrations of radioactive wastes are assured by continuously monitoring the gases leading to the exhaust stack, and by assuring maximum dilution of the potentially contaminated air with uncontaminated air.

1995
1976-
1977

7.3.2 Solid Waste Disposal

Solid waste will be deposited in polyethylene bags within containers placed in each of the laboratories. Solid waste deposited in these containers will be collected on a routine basis and stored in the below grade area of the laboratory building until such time that a sufficient volume has accumulated for the waste to be packaged in ~~se~~ **sealed containers."** ~~n- tainers.~~ ~~The co~~ **"The containers will then be processed** ~~ked up~~ by a member of the radiological safety office who will dispose of the solid waste in accordance with established procedures. 1995

7.3.3 Liquid Waste Disposal

All drains for potentially contaminated liquids are delivered to a liquid waste retention system in the below grade area of the reactor building. ~~The liquid~~

The liquid waste retention system consists of three tanks of approximately 5000 gallons capacity each, to which chemicals can be added for water treatment and from which samples can be taken for an assay of radioactive contamination.

~~... and from which samples can be taken for an assay of radioactive contamination.~~ Liquid waste will be held in these tanks until an assay shows that the waste may be dumped to the sanitary sewage system, or that the waste may be dumped after dilution. ~~or that the waste must be concentrated by a distillation unit. Since the concentrated liquid waste will represent a significantly high level of activity, these wastes will be prepared at the site for subsequent direct land burial.~~ 1976-1977

DELETE

1995

1996

8.0 EXPERIMENTAL FACILITIES

8.1 Introduction

~~The~~ "The **primary** purpose of a university ~~-sity~~ research reactor 1996
is to provide the maximum flux, or current, of neutrons to the maximum number of users. This then dictates certain design parameters. Additionally, and of paramount importance, these goals must be met within a framework of ultra safety. This latter requirement further dictates design criteria. ~~The result may suffer from certain shortcomings which may b~~ "**The research history of MURR shows it is possible** ~~ever it is possible~~ to produce a rather respectable machine 1996
for university research.

In the paragraphs which follow there are descriptions of the experimental facilities existent in the University of Missouri Research Reactor. It is pertinent to point out that all other systems of this facility are subordinate to the purpose of providing neutrons to these experimental facilities.

The reactor facility described herein has a "built-in" safety feature which is not immediately obvious. The 3 inch beryllium reflector surrounding the core very effectively decouples the core from experimental variations in the beamports, thermal column, pneumatic tubes and the irradiation baskets. The only experimental facility not decoupled by beryllium is the flux trap.

1996
1967-1968
2000

8.2 Flux Trap

The flux trap facility of the University of Missouri Research Reactor provides a rather large reactivity affect if used without proper restrictions and supervision. Consequently, use of this facility will be subject to a high degree of administrative control to minimize the possibility of ~~either inserting or removing a sample which would result in~~ "... inserting or removing a sample with high reactivity worth during reactor operation." ~~content sample during reactor operation.~~ 2000

~~To eliminate the possibility of either of these happenings, samples will be inserted or removed only during reactor shutdown. Means have been provided to eliminate sample movement during reactor operation.~~ DELETE 2000

The number and the volume of samples is limited mechanically by the design of the center hole canister which holds the samples to be irradiated. "This canister is made of hollow aluminum tubes and is inserted in the flux trap prior to startup. A latching device located at the top of the canister positively determines the canister position." 1996 2000

~~The assembly~~ ~~of the assembly.~~ The experimental volume in the assembly is at all times filled with either aluminum spacers or experimental capsules. These capsules are originally maintained in their positions by a locking rod **or rods** latched at the assembly top canister. A 1/16 inch cooling annulus is provided between the capsule experiments and the canister assembly to provide adequate cooling. ~~and allow adequate cooling. The canister has a~~ "The canister **top is designed** to preclude the possibility of dropped objects entering the test hole while the assembly is in place. 1967-1968 & 1996* 1996

"All sample insertions into the flux trap position are subject to review by the Reactor Manager." ~~ion are The~~ 1996

1996
1967-
1968
2000

~~reactor supervisor will supervise the installation~~

"A senior reactor operator will supervise the installation and removal of the canister during shutdown and see that either a canister is in place or that a strainer is provided over the test hole prior to startup."

~~cover is provided over the test hole previous to startup.~~

* 1996

0007

"Technical Specification Amendment 31, which allows movable and unsecured experiments in the flux trap, was approved by the NRC on September 20, 1999."

A form, similar to that utilized for pneumatic tube irradiations, will be used to describe the sample which the experimenter wishes to have inserted in the flux trap position. Applications will be carefully reviewed with regard to size and material to predict reactivity affects and with regard to size, cladding or containment to insure adequate cooling by pool water and to avoid contamination of the reactor pool. Standards for approval of radiation will be established on the basis of the initial reactivity studies. Any application submitted for a sample irradiation in this position which involves extraordinary conditions will be referred to the Reactor Advisory Committee for comment and consideration.

ADD TO END OF PARAGRAPH

~~It is felt that alternate means for usage of this facility should be provided only after operational experience has provided complete data on reactivity value. At some future date, after the acquiring of suitable information pertaining to the flux trap worth, a request will be made to the "NRC" for license amendment to permit insertion and removal of samples during reactor operation.~~

1996

~~The canister for insertion into the flux trap position consists of two parts. An outer can closed at~~

DELETE

1967-1968

* DELETES A 1967-REV. 1968

1967-
1968
1972-
1973
2000

~~the bottom with an aluminum plug and shaped like a funnel at the top. The top of this funnel is covered with~~

"The irradiation canister consists of aluminum tubes welded into a single unit as depicted in Figures 8.4 and 8.5. The tubes are clearly identified both physically on the canister and on all sample loading documentation so that sample position is positively controlled."

1972 - 1973
2000

~~is a~~ ~~the~~ ~~The~~ ~~shelves. Extending from the top of this array of shelves there is a lift rod terminating at the very top with a lifting ring. In use the shelves are loaded with samples and the assembly is positioned in the external tube of 1 1/2 inch OD by 1/8 inch wall thickness of 6061 aluminum and the whole assembly is lowered into the flux trap position.~~

DELETE

~~In summary, the flux trap canister assembly consists of two pieces; the sample retaining center rod with its sample positioning shelves surrounded completely by the aluminum tube. The total assembly, consisting of tube and rod, is ten foot nine and 1/4 inches in length. The top of the tube has a snap-on cover which minimizes the probability that any samples can be withdrawn during reactor operation. Further the cover prevents the inadvertant dropping of any sample into the flux trap position in the event that the center rod is removed from the tube.~~

1967-1968

8.3 Beamports

Six beamports are provided in this reactor. Four of these ports are radial and two are "tangential." The tangential ports actually approach the core straight on, however they terminate below the fuel level of the core and therefore do not directly view the radiation from the reactor core.

1996
1987-1988

~~Th~~ "The **primary** use of the beamports ~~orts~~ is expected to be in neutron and solid state physics experiments. It is planned that neutron beams will be brought out of the reactor through filters and collimators installed in the beamports. 1996

When these beamports are used experimentally, radiation surveys will be made of each installation to determine that the beam catchers and shielding barricades are adequate to control radiation hazards. In addition to the routine surveying which will be employed, there are permanently installed radiation monitors located on the beam hole floor for the purpose of detecting any radiation leakage from beamport experiments. ~~These installed radiation~~ 1987-1988

These installed radiation monitors report to an analog readout in the control room.

~~The tangential beamports have been installed to minimize the fast neutron and gamma ray background. The beamport plug assemblies, which provide shielding for the tangential ports, are identical to those used in the radial ports.~~

DELETED

~~In addition to the expected utilization of the beamports for neutron diffraction or neutron chopper work it is contemplated that a port may occasionally be used in a "loop" type experiment. For this purpose there are available experimental cans mounted as extensions of a modified shield plug. These modified shield plugs have helical conduits through them to admit electrical leads and utility lines to the~~ 1996

1996
1967-
1968

~~experimental can. A water line is put through all shield plugs whether the plug is designed for experimental use or shielding. Utilizing this water line it is possible to flood the port when it is not in use or to provide cooling water to an experiment if this is required.~~ DELETED 1996

~~It is pertinent to point out that all changes in~~
It is pertinent to point out that all changes in beamport experiments will be performed only when the reactor is shutdown. 1967-1968
~~when the reactor is shut down.~~ All beamport experiments are semi-permanent in nature, and in no instance could the contained experimental apparatus be rapidly withdrawn.

In any instance where an experiment is proposed which would involve the insertion of a loop in one of the beamports, the Reactor Advisory Committee will review such an experimental proposal and make recommendations regarding the critical parameters which must be monitored and recorded, such as temperature, pressure or coolant flow. ~~Further,~~
"Further, the Reactor Manager will determine if any of the parameters should be used to activate an annunciator and/or the reactor scram system." 1996
~~and/or the reactor scram system. Spare annuciator positions are available for this purpose.~~ All design and operation procedures for loop experiments must be reviewed by the Reactor Advisory Committee.

Located on the beam hole floor, and in close proximity to the beamports, are a number of storage holes for contaminated beamport apparatus. These storage holes

1996

are shielded with concrete and lead plugs to minimize radiation leakage. The storage ports are also vented through a duct system which carries any off-gas to the reactor stack.

Figure 8.1 provides construction details on the beamports. This figure is applicable to all six ports. The six beamports are spaced three on each side of the reactor core and terminate at the face of the biological shield in a vestibule recess.

There are available three 4 inch I.D. ports and three 6 inch I.D. ports. All ports, whether 6 or 4 inch, step to an 8 inch size approximately 40 inches in from the biological shield face. ~~All ports are~~ 1996
"All ports can be closed with a 3-inch lead d vestibule door when not in use. The vestibule door is opened by means of the overhead crane and pinned in the open position when it is desired to extract a neutron beam through a collimator assembly. ~~When the port is not in use~~
"When the port is not in use it may be filled or drained without opening the door by means of an external valving system when the reactor is shutdown." 1996
~~immediately below each beamport.~~

The port assembly consists of three major pieces. Working from the concrete of the biological shield in toward the center of the port one finds first the fixed liner which is in contact with the concrete and is integrally welded to the reactor pool liner. This outer shell provides the form work around which is poured the biological shield. A number of welds connect the port shell with the port vestibule and the lead door guide.

1996

The next major component of each beamport is known as the movable port liner. This long aluminum assembly extends from the vestibule inward, penetrating the graphite reflector and terminating adjacent to the beryllium reflector. Between this movable liner and the outer shell there is circulated demineralized water returning from the pool skimmer system. This demineralized water flows in between the fixed shell and the movable liner and proceeds on into the pool. This flow of water minimizes the possibility of corrosion resulting from stagnant water. The movable liner is attached to the external shell by means of a bolt ring located in the vestibule. There is a large flange which draws down against a gasket and seals the gap between the fixed liner and the movable port liner.

~~Within the movable port liner is located the beamport plug. The plug is also closed with a bolt~~

"Within the movable port liner is located the collimator liner. The collimator liner is also closed with a bolt ring and gasket such that the region in front of the collimator liner and the end of the movable port liner can be flooded with water."

~~and, if necessary at higher power, can be water cooled.~~

1996

The vestibule of each beamport is provided with a 2 inch off-gas vent, a 1 inch drain, two 2 inch conduit sleeves from the vestibule to the stepback in the biological shield, which is approximately 13 feet above the beamport floor. Other services are available, at each beamport location, from the service plenum located on top of the step of the biological shield. The services of each port are demineralized water, cold water, vacuum and 110 volt power supply.

At the reactor end of each beamport the ports penetrate through the graphite portion of the reflector and terminate adjacent to the beryllium reflector. Each of these beam tubes within the reflector region is cooled by water flowing through grooves between the graphite reflector pieces and around the beamport extension into the graphite. The water flows downward through the graphite reflector elements, around the nose pieces of the beam tubes, down through the support plate for the graphite reflector elements and into the plenum for outflow to the pool heat exchanger.

8.4 Irradiation Baskets

~~There are a total of twelve modified reflector elements containing sample holes for irradiation purposes. These removable reflector elements are positioned in the graphite reflector region outside of the permanent beryllium reflector. They~~

"The graphite reflector region outside the permanent beryllium reflector is made up of removable reflector elements which can be reconfigured to provide sample irradiation positions. These irradiation positions are used to introduce samples of greater size, and for a longer time, into a relatively high thermal neutron flux than would be normal for the pneumatic tube system.

Each of the removable reflector element locations is capable of supporting several types of irradiation samples. Changes in sample irradiation configurations are analyzed for reactor safety, approved by the Reactor Manager, and reviewed by the Safety Subcommittee."

~~flux for longer periods of time than would be normal for the pneumatic tube system. In addition to using a number of these baskets for long term irradiations, four of the graphite sector pieces under discussion are utilized as mounting positions for the pneumatic tubes.~~

1996

1996
2001

~~All of these irradiation baskets are removable in the event that it is desired to irradiate a relatively large sample. If this is desirable, and the reflector elements are removed, flow restricting plugs are inserted in the positions normally occupied by these baskets to prevent flow bypass.~~

DELETE

~~All samples for insertion into the irradiation baskets are turned over to the reactor supervisor who assigns an irradiation position and denotes this irradiation position in the reactor log book. A form is filled out by the experimenter and approved by the reactor supervisor prior to putting the sample into the reactor. This form is similar to that used for pneumatic tube irradiations.~~

~~All entry and removal of the irradiation baskets will be performed during shutdown of the reactor. In no instance will these irradiation baskets be moved during reactor operation.~~ 1996

8.5 Pneumatic Tubes

The University of Missouri Research Reactor is equipped with a pneumatic irradiation system provided by Airmatics Corporation. This is their standard 1 1/2 inch diameter, vacuum operated, system. ~~The system consists of four reactor ter-~~

"The system consists of up to four reactor terminals, up to six sending-receiving stations, two deflector switches, one solenoid cabinet, **two turbo-compressors** together with the necessary piping, tubing, electrical conduit, and mounting hardware. 1996

~~stations ar~~ ~~ing~~

"Currently, only two reactor terminals and three sending-receiving stations are in use." 2001

1996
1981-
1982
2001

Replace
"up to 4"

This pneumatic transfer system is capable of transporting ~~four~~ rabbits simultaneously at a velocity of from 30 to 45 feet per second either into or away from the reactor. The system is completely automatic after the initiation of the rabbit start. Automatic timing equipment has been provided and is located at the sending stations. This timing equipment controls the irradiation time and return of the rabbits. ~~Automatic timing is~~ 1981-1982

"Automatic timing is available for periods varying from 2 seconds to 120 minutes." 1996

~~20 minutes.~~

The system differs from "ordinary" only in two respects. ~~First~~ "First, each of the pneumatic tubes ~~tubes~~ are ~~equipped with double send-receive stations such that each tube services two laboratories adjacent one ... irradiation.~~ 1996

"First, each of the pneumatic tubes may be equipped with double send-receive stations such that each tube can service two laboratories adjacent one to the other. The control panels of the sending-receiving stations for these laboratories are electrically interlocked such that when a rabbit is dispatched from one station it is impossible to utilize the remaining station for an irradiation." 2001

Replace
"up to 4"

The second way in which the system differs from ordinary is that the in-pool terminals for the ~~four~~ 1981-1982 pneumatic tubes consist of concentric tubes rather than a separate sample and air tube as is conventional. That portion of the system which penetrates the reactor pool biological shield and terminates within the reflector elements near the reactor core is of prefabricated concentric tubing. The concentric portion of each rabbit carrier tube begins outside the reactor wall and penetrates through a sleeve

1776
1969-
1970
1981-
1982

imbedded in the pool wall and terminates in the reflector element. The outer concentric tube is 2 1/4 inch O.D. and the inner concentric tube is 1 1/2 inch O.D. Immediately within the pool wall, an aluminum concentric flange connection is used to permit removal and installation of the terminal tubes.

Control of the turbo-compressor is by means of a stop-start switch located in the reactor control room. The turbo-compressor is located immediately adjacent, and outside of, the building containment wall. The 4 inch exhaust line from the turbo-compresor p exhaust to the facility stack plenum. wall. Immediately inside the containment wall this 4 inch exhaust line has a quick closing valve which is electrically interlocked into the containment safety system. The valve is closed in the event high activity is detected within the exhaust off-gas or in the event that the reactor operator initiates the building evacuation alarm. This 4 inch exhaust valve operates at the same time as the off-gas 16 inch valve. 1981-1982

The pneumatic tubes terminate in the graphite reflector pieces adjacent to the beryllium reflector. 1969-1970
The terminal end of each pneumatic tube is of a webbed geometry between the concentric tubes to prevent excessive heat generation in the sample or the aluminum. The webbed zone is 24 inches in length extending from the bottom of the inner concentric tube up. There are four webs each of 0.22 inch 1996

"There are four webs each of 0.22 inch thickness with a vertical web cross section of 0.875 square inches per inch of length."

1996
1981-
1982

~~inch of length, of 0.875 square inches.~~ The separating webs are equally spaced and permanently bonded to both tube walls. The outside of the outer concentric tube within the reflectors has spacers welded to it at intervals to assure coolant water flow past the outside of the concentric tubing. 1996

8.6 Thermal Column

The thermal column consists of a 60 inch thick graphite pack contained within a water-jacketed aluminum casing (Figure 8.2). The column has a lead gamma shield positioned between the reflector ring around the core and the inner end of the case. The protrusion of the thermal column into the pool is a portion of the thermal column casing. This section is 3 feet 1 1/2 inches square and 12 1/4 inches deep. This section is then stepped into a box 4 feet 2 inches square and 5 feet 8 inches deep. The larger box is completely lined with 1/4 inch boral.

~~The thermal column is provided with a steel shield door 25 inches thick with a face plate of 1/4 inches~~

"The graphite stack incorporates a bismuth filter, neutron radiographic variables aperture, and a slot for the irradiator case for the film irradiator experiment. The original thermal column door has been replaced with a new door that incorporates a film irradiator experiment. Figure 8.3 is a cross sectional view of the thermal column door, and the film irradiator experiment. The thermal column door moves on two floor level tracks and is driven by an electric motor through a gear reducer box. An electrical interlock assures that the control rods cannot be withdrawn unless the door is fully closed." 1981-1982

~~Nine 4x4 inch stringers are removable from the graphite stack. Access to this one foot square portion of the face of the thermal column graphite~~

1981-
1982

stack is by means of a stepped door in the thermal column door. The hole size at the inner face is 13x13 inches from which it steps to 15x15 inches, to 17x17 inches, and to 19x19 inches at the outer face of the door. This door is secured by a key lock with the key in the possession of the reactor supervisor. The thermal column door is shown in Figure 8.3. 1981-1982

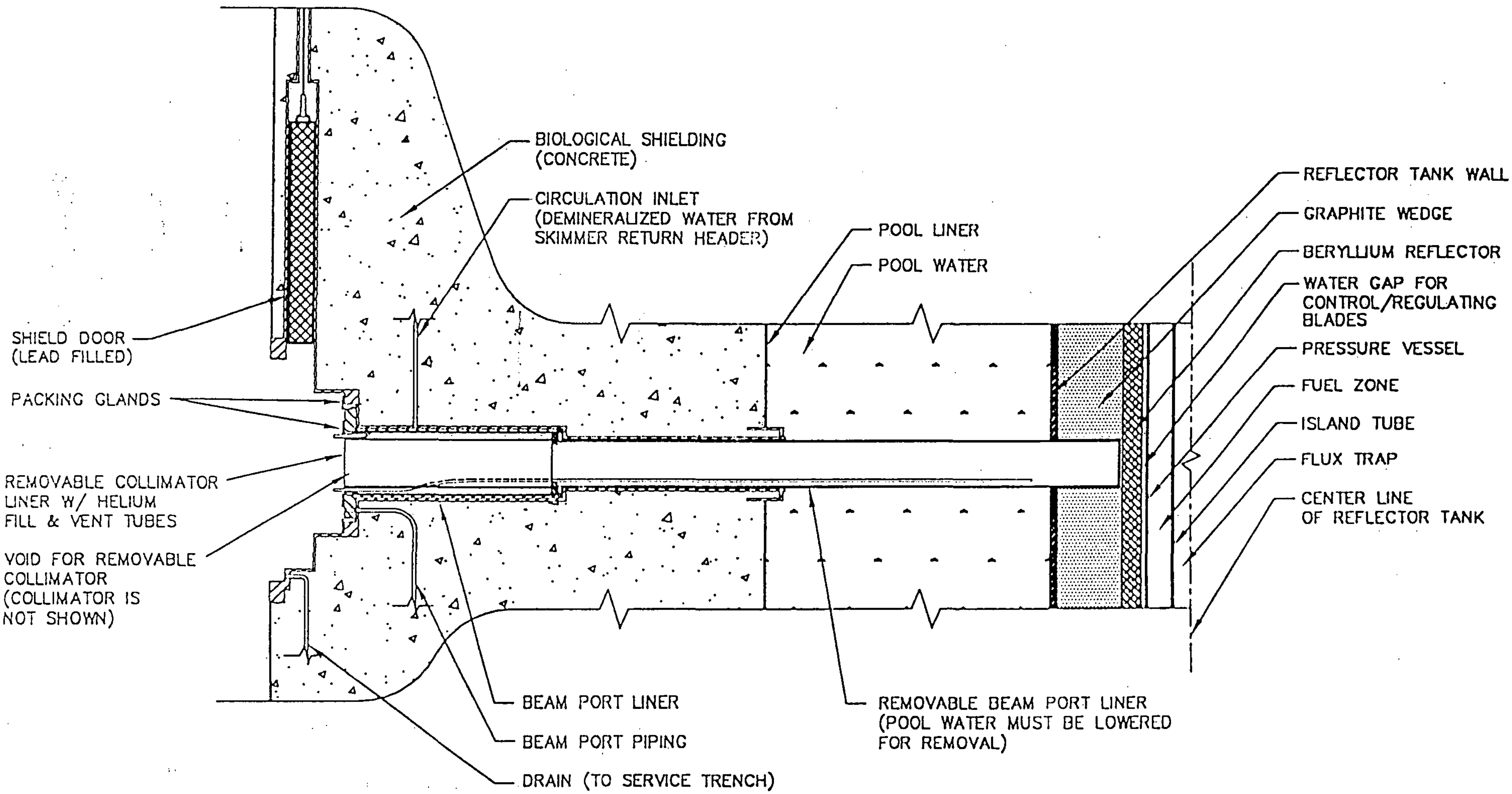
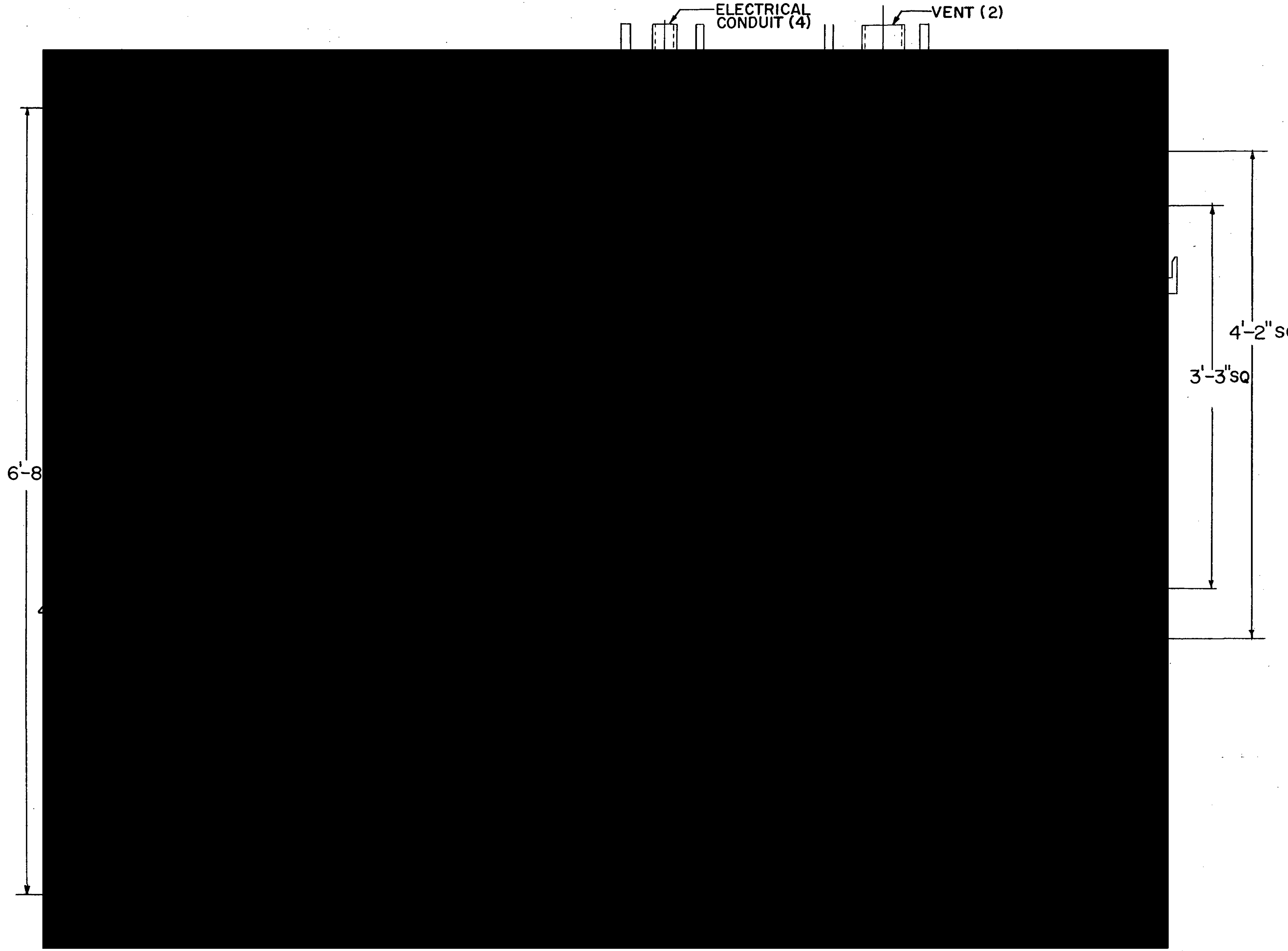


Figure 8.1

SECTION THRU TYPICAL BEAM PORT

(Da 11/15/96)

1996



[redacted]

Figure 8.2 Thermal Column Assembly

798D453

GENERAL ELECTRIC 798D453

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:—

APPLIED PRACTICES	SURFACES	TOLERANCES ON MACHINED DIMENSIONS		
		FRACTIONS	DECIMALS	ANGLES

REV. NO. 61B34

798D453

TITLE SPEC. CONTROL

THERMAL COLUMN DOOR

[redacted]

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JAN 64

Figure 8.3 Thermal Column Door

FIGURE 8.4

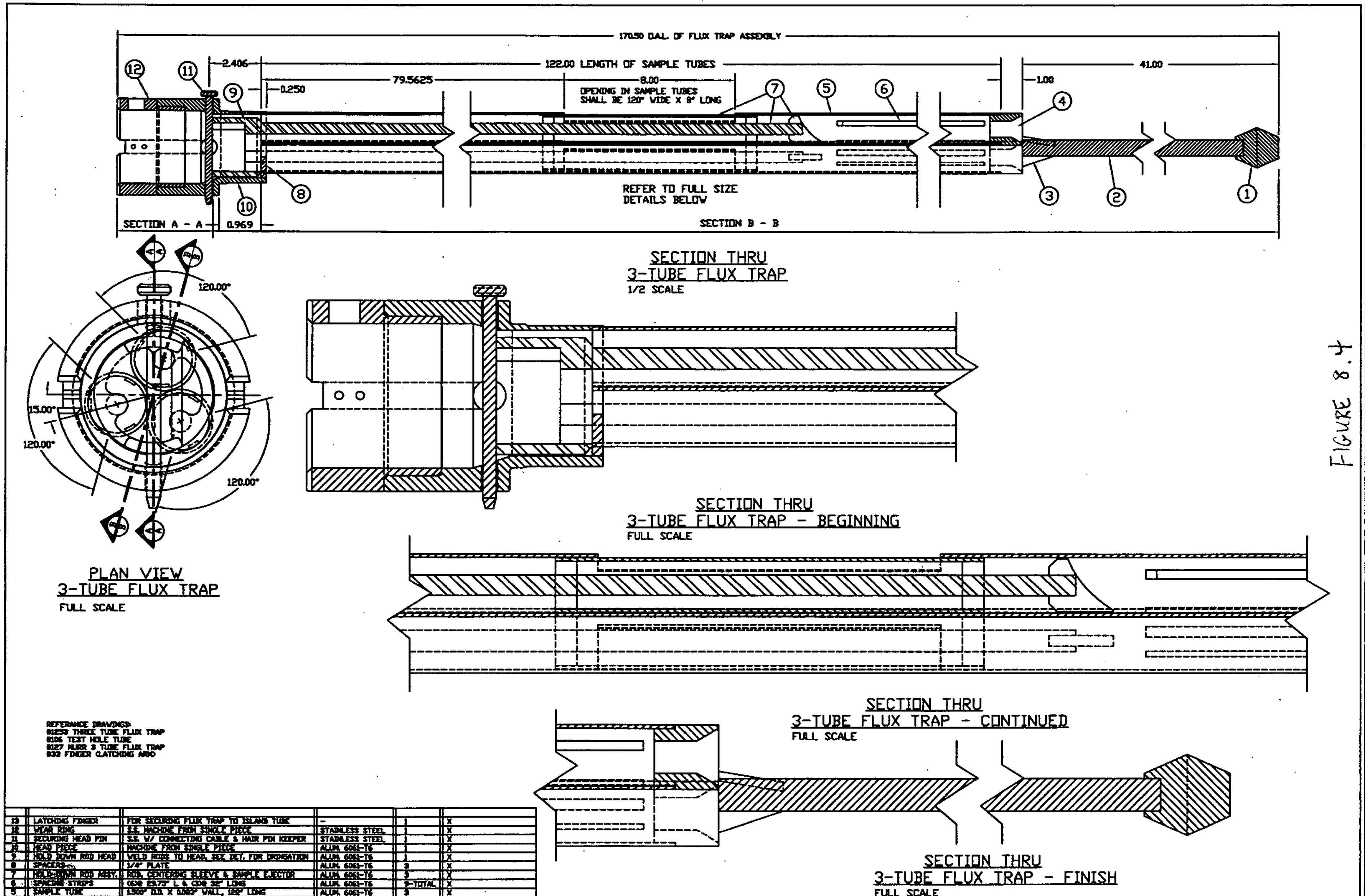


FIGURE 8.4

REFERANCE DRAWINGS:
#1253 THREE TUBE FLUX TRAP
#106 TEST HOLE TUBE
#127 MURR 3 TUBE FLUX TRAP
#33 FINGER CLATCHING ARO

NO.	ITEM	DESCRIPTION	MATERIAL	NO. REQ'D.	SEE DETAIL ON SHEET
13	LATCHING FINGER	FOR SECURING FLUX TRAP TO ISLAND TUBE	-	1	X
12	WEAR RING	S.S. MACHINE FROM SINGLE PIECE	STAINLESS STEEL	1	X
11	SECURING HEAD PIN	S.S. W/ CONNECTING CABLE & HAIR PIN KEEPER	STAINLESS STEEL	1	X
10	HEAD PIECE	MACHINE FROM SINGLE PIECE	ALUM. 6061-T6	1	X
9	HOLD DOWN ROD HEAD	WELD RODS TO HEAD, SEE DET. FOR ORIENTATION	ALUM. 6061-T6	1	X
8	SPACERS	1/4" PLATE	ALUM. 6061-T6	3	X
7	HOLD-DOWN ROD ASSY.	ROD, CENTERING SLEEVE & SAMPLE EJECTOR	ALUM. 6061-T6	3	X
6	SPACING STRIPS	(6)@ 23.75" L & (3)@ 32" LONG	ALUM. 6061-T6	9-TOTAL	X
5	SAMPLE TUBE	1.500" O.D. X 0.083" WALL, 122" LONG	ALUM. 6061-T6	3	X
4	END PLUG	MACHINE FROM 2" DIA. STOCK, 1 1/2" LONG	ALUM. 6061-T6	3	X
3	GUSSET	1/4" PLATE	ALUM. 6061-T6	3	X
2	SUPPORT ROD	3/4" ROD X 39 1/4" LONG	ALUM. 6061-T6	1	X
1	BASE PIECE	MACHINE FROM 2" DIA. STOCK, 2" LONG	ALUM. 6061-T6	1	X

NO.	REVISION DESCRIPTION	DATE	INIT.
1	REMOVED 2ND HOLE IN WEAR RING	4/17/07	DGN

2007

DATED 2/4/98
DRAWN BY: DGN
CHECKED BY: ENGINEER
CODE
REVISION NUMBER
REVISION DATE

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

FLUX TRAP: 3-BARREL ASSEMBLY VIEW

MURR NUMBER
2505

SHEET
1 of 3

2007

FIGURE 8.5

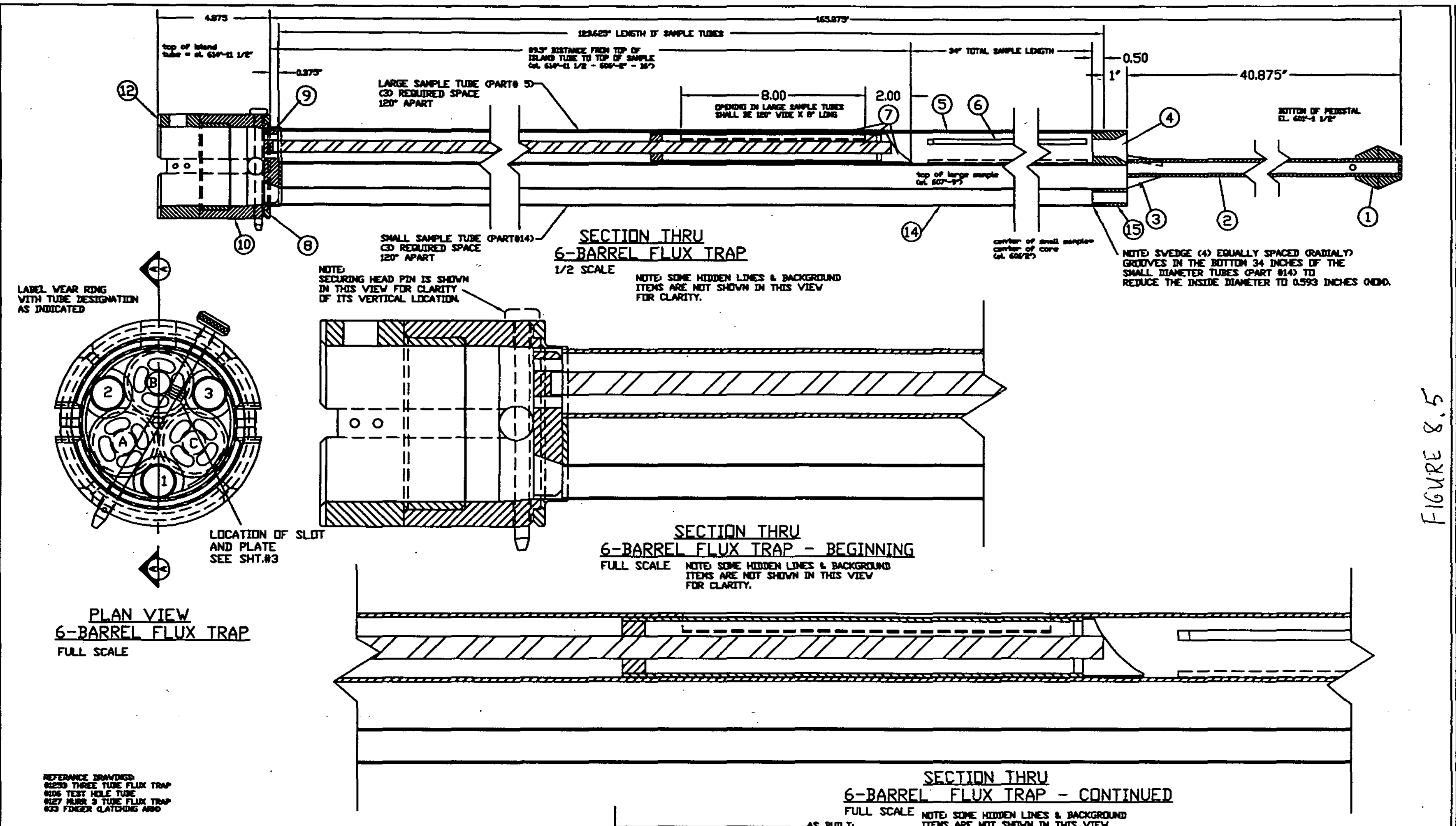
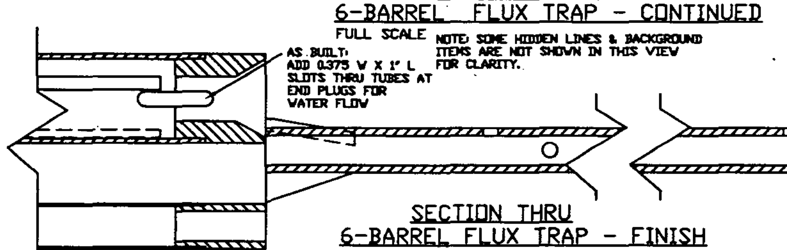


FIGURE 8.5

REFERANCE DRAWINGS:
#1253 THREE TUBE FLUX TRAP
#106 TEST HOLE TUBE
#127 MURR 3 TUBE FLUX TRAP
#33 FINGER (LATCHING ARM)



NO.	ITEM	DESCRIPTION	MATERIAL	NO. REQ'D.	SEE DETAIL ON SHEET
16	WELDING TABS	0.250 THK.	ALUM. 6061-T6	3 SETS	2
15	SMALL END PLUG	0.750" O.D. X 1 1/2" LONG	ALUM. 6061-T6	1	3
14	SMALL SAMPLE TUBE	0.750" O.D. X 0.035 WALL X 123.625" LONG	ALUM. 6061-T6	1	X
13	LATCHING FINGER	FOR SECURING FLUX TRAP TO ISLAND TUBE	-	1	3
12	WEAR RING	S.S. MACHINE FROM SINGLE PIECE	STAINLESS STEEL	1	2
11	SECURING HEAD PIN	S.S. W/ CONNECTING CABLE & HAIR PIN KEEPER	STAINLESS STEEL	1	3
10	HEAD PIECE	MACHINE FROM SINGLE PIECE	ALUM. 6061-T6	1	2
9	HOLD DOWN ROD HEAD	WELD RODS TO HEAD, SEE DET. FOR ORINGATION	ALUM. 6061-T6	1	2
8	HEAD PIECE BASE CUP	MACHINE FROM SINGLE PIECE	ALUM. 6061-T6	1	2
7	HOLD-DOWN ROD ASSY.	ROD, CENTERING SLEEVE & SAMPLE EJECTOR	ALUM. 6061-T6	3	3
6	SPACING STRIPS	(6)@ 33.75" L & (3)@ 36" LONG	ALUM. 6061-T6	9-TOTAL	3
5	LARGE SAMPLE TUBE	1.500" O.D. X 0.083" WALL, 123.625" LONG	ALUM. 6061-T6	3	X
4	LARGE END PLUG	MACHINE FROM 2" DIA. STOCK, 1 1/2" LONG	ALUM. 6061-T6	3	3
3	GUSSET	1/4" PLATE	ALUM. 6061-T6	3	2
2	SUPPORT TUBE	3/4" O.D. TUBE X 0.125 WALL X 40 7/8" LONG	ALUM. 6061-T6	1	3
1	BASE PIECE	MACHINE FROM 2" DIA STOCK, 2" LONG	ALUM. 6061-T6	1	3

NO.	REVISION DESCRIPTION	DATE	INIT.
2	REMOVED 2ND HOLE IN WEAR RING	4/17/07	DGN

2007

DATE 7/23/99
DRAWN BY: DGN
CHECKED BY: ENGINEER
CODE
REVISION NUMBER: D. AS BUILT
REVISION DATE 9/21/99

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

FLUX TRAP: 6-BARREL
ASSEMBLY VIEW (as built)

MURR NUMBER 2528
SHEET 1 OF 3

9.0 INSTRUMENTATION AND CONTROL

9.1 Reactor Control Scheme

All reactor control is achieved by varying the amount of neutron absorber interposed between the reactor core and the reflector. No other reactor parameter, such as temperature or pressure, is used to change or modify reactor power other than through inherent reactivity feedback effects.

Since the reactor is designed to provide neutrons for experimental use an essential feature of the core design is high neutron leakage. Startup and control of the reactor is achieved by surrounding the core with a two region (beryllium metal followed by graphite) reflector to increase the fraction of leakage neutrons that return to the core region and by interposing a movable shroud of a material opaque to thermal neutrons (boral) between the core region and the reflector.

All experimental facilities, except the center test hole, are outside the beryllium metal reflector region to minimize the reactivity effects of experiments. The control scheme used has the advantage (in addition to mechanical advantages mentioned elsewhere) that when the reactor is shutdown all experiment positions, except the center test hole are neutronically decoupled from the core.

1995

The control scheme used has been designed such that cold clean criticality will be reached with the control rods 50 per cent withdrawn. This criticality position of the rods is based on three criteria.

(1) At least 50 per cent of rod worth available at all times for shutdown margin.

(2) At least half the core be exposed to the experiments.

(3) Provide sufficient excess reactivity in the core for a fuel loading lifetime of at least 400 megawatt-days.

~~Since the purpose of the UMRR is to provide neutrons~~

"Since the purpose of the MURR is to provide neutrons for experimental users it is intended that the reactor be operated continuously, with a regular schedule of shutdowns for maintenance and refueling."

~~at some fixed power level with a minimum of unscheduled shutdowns throughout the work day.~~ Startup of the reactor will be completely manual. Once the desired power level has been reached the reactor will be put on automatic control. The controlling parameter is neutron flux as sensed by the detector for one of three linear level channels. The output of the channel being used for control is compared with the power demand setting in a servoamplifier unit. The servoamplifier adjusts the position of the regulating rod, stepwise, in a direction to minimize the power error.

1995

An additional automatic control feature has been included to further reduce operational complexity. After the reactor has been operated at full power and then re-started after a 16 hour (nominal) shutdown period considerable negative reactivity will have built up in the core due to Xenon. As Xenon

1981-
1982
1967-
1968

is "burned out" the regulating rod will travel its full operating range several times before the Xenon affect reaches a minimum and begins to buildup again. In order to enhance operating ease a control feature has been incorporated to automatically re-shim the reactor each time the regulating rod completes a stroke. If the reactor is on atuomatic control and the auto-shim circuit has been activated, when the regulating rod reaches a preset point (nominally 20 per cent withdrawn) the shim rods will "jog in" until the regulating rod has returned to the 60 per cent withdrawn position. This cycle is repeated each time the regulating rod has inserted to the 20 per cent level. The auto-shim feature operates only in a direction to decrease power. At no time will the shim rods be automatically withdrawn. As Xenon concentration builds up (negative reactivity is being added in the core) and it exceeds an amount that the regulating rod can automatically correct for, the operator must manually re-shim to avoid a power slump. 1981-1982

The only other parameter that will normally influence reactivity is core temperature, but this effect is small. ~~The calculated temperature change in re-~~ The calculated temperature change in reactivity from cold core to operating temperature is approximately a negative .002 Δk. 1967-1968

Control system requirements with respect to response time, accuracy and continuity of operation are consistent with those of ordinary open pool type research reactors.

1995

9.2 Operation Control Station

All reactor displays and controls are within a single reactor control room. The reactor control room is on the third level of the reactor containment building at the elevation of the pool surface and the operating bridge. One wall of the control room is windowed and overlooks the walkway around the pool and the open area surrounding it. A door in the north wall leads into the control room from the elevator lobby. ~~A second door in the south wall of the control room leads into a storeroom.~~ DELETE The control room is 17 feet by 28 feet 8 inches with an 8 foot suspended accoustical ceiling. A partition runs across the rear of the room, 4 feet out from the back (east) wall into which the instrument cubical is recessed. Two openings in the floor provide cable access to the control console and the instrument cubicle. 1995

The general layout of the control room is shown in Figure 9.1. The control console is a straight desk type 93 3/4 inches long and 44 inches high (overall) with a sloping face. The instrument cubicle, which stands 4 feet behind the console is 8 feet high and is 9 feet 7 inches wide. The control room windows are to the rear of the reactor operator as he faces the console. 1995

INSERT AT END OF PARAGRAPH

Communication between the control room and other areas throughout the facility is by means of telephone or the intercommunication and paging system.

"One inch steel plating is attached to the lower half of the windows, shielding the operating staff from pool background radiation."

1995
2001
2004

All operating information is displayed or recorded on the control console or the instrument panel. All operating controls are on the control console. The control console layout is shown in Figure 9.2. The instrument panel layout is shown in Figure 9.3. A summary of console displays and controls is shown in Table 9.1. The "number" column in Table 9.1 refers to the identification on Figure 9.2.

TABLE 9.1

CONTROL CONSOLE DISPLAYS AND CONTROLS

(a) Displays

No.	Function	Type
1	Control Rod Position Indication	Digital
2	Control Rod Drive Mechanism "Power On" (4)	Light
3	Control Rod Drive Mechanism "Drive Full In" (4)	Light
4	Control Rod Drive Mechanism "Drive Full Out" (4)	Light
5	Control Rod Drive Mechanism "Magnet Engaged" (4)	Light
6	Control Rod Drive Mechanism "Blade Full In" (4)	Light
7	Regulating Rod Full In	Light
8	Regulating Rod Full Out	Light
9	Regulating Rod 10% Withdrawn	Light
10	Regulating Rod 20% Withdrawn	Light
11	Regulating Rod 60% Withdrawn	Light
12	Source Range Level - Channel 1	Meter
13	Source Range Period - Channel 1	Meter
14	Intermediate Range Level - Channel 2	Meter
15	Intermediate Range Period - Channel 2	Meter
16	Intermediate Range Level - Channel 3	Meter
17	Intermediate Range Period - Channel 3	Meter
18	Power Range Level - Channel 4	Meter
19	Power Range Level - Channel 5	Meter
20	Power Range Level - Channel 6	Meter
21	Wide Range Level	Meter
22	Power Level Set	Meter
23	Pneumatic Tube System Blowers "On" Indication	Light
24	Regulating Rod Position Indication	Digital
25	Auto Shim Engaged	Light | 2004
26	High Power "Warning"	Light

2001

* TABLE 9.1(a) & 9.1(b) REVISED IN IT'S ENTIRETY BY 2001 REV.

1981-1982
1995
2001

Light Digital Readout

P-Tube Blower Light Digital Readout

81 85

TABLE 9.1 (cont'd)

Number	Function	Type
37	Intermediate Range Level	Meter
38	Intermediate Range Period	"
39	Intermediate Range Level	"
40	Intermediate Range Period	"
41	~~Wide Range Level~~ Delete	"
42	Power Range Level	"
43	Power Range Level	"
44	Power Level Set	"
45	High Power "Warning"	Light
55	Reg. Rod 10% Withdrawn	"
56	Reg. Rod 20% Withdrawn	"
57	Reg. Rod $\boxed{60\%}$ Withdrawn	"

A00 →

1981-1982 1981-1982 1981-1982

2001

(b) Controls

Number	Function	Type
12	Control Mode "Manual"	P.B. Switch
13	Control Mode "Auto"	"

Number	Function	Type
12	C	
13	C	
16	R	
17	R	
21	M	
22	M	
23	C	
24	Se	
25	C	to Center
26	C	
27	R	to Center
29	P	
30	Annunciator "Acknowledge"	P.B. Switch
31	Annunciator "Test"	"
32	Annunciator "Reset"	"

No.	Function	Positions	Type of Switch
27	Rod Control Mode	"Manual"	Push Button
28	Rod Control Mode	"Auto"	Push Button
29	Power Schedule Selector	"Raise-Off-Lower"	3 Pos. Spring Ret.
30	Regulating Rod Operate	"Jog In"	Push Button
31	Regulating Rod Operate	"Jog Out"	Push Button
32	Master Control	"Off-Test-On"	3 Pos. Key Lock
33	Power Level Selector	"50kW-5MW-10MW"	3 Position
34	Control Rod Selector	"A-B-C-D-Gang"	5 Position
35	Control Rod Operate	"In-Normal-Out"	3 Pos. Spring Ret.
36	Regulating Rod Operate	"In-Normal-Out"	3 Pos. Spring Ret.

1995 2001

Number	Function	Type
24	Select Wide Range Channel Range	Range Switch
25	Control Blade "In-Off-Out"	3 Pos. Switch-Spr. Return to Center

* TABLE 9.1(a) & 9.1(b) REVISED IN ITS ENTIRETY BY 2001 REV.

2001
2004
2008

TABLE 9.1 (cont'd)

~~26 Control Blade "Rod Run-In" P.B. Switch~~

Number	Function		Type
37	Annunciator Acknowledge	N/A	Push Button
38	Annunciator Reset	N/A	Push Button
39	Annunciator Test	N/A	Push Button
40	Scram	N/A	Push Button
41	Scram Reset	N/A	Push Button
42	Rod Run-In	N/A	Push Button
43	Rod Run-In Reset	N/A	Push Button
44	Magnet Current	"Off-On"	2 Position
45	Reactor Isolation	"Off-On"	2 Position
46	Facility Evacuation	"Off-On"	2 Position
47	Hi/Low Reflector ΔP	"Off-Bypass"	2 Pos. Key Lock
48	Low Pressurizer Pressure	"Off-Bypass"	2 Pos. Key Lock
49	Low Primary Pressure	"Off-Bypass"	2 Pos. Key Lock
50	Vent Tank Low Level	"Off-Bypass"	2 Pos. Key Lock
51	Rod Magnet Contact	"Off-Bypass"	2 Pos. Key Lock
52	Ant-Siphon High Level	"Off-Bypass"	2 Pos. Key Lock
53	Intrusion Alarm	"Off-On"	2 Position
54	Airlock Door Security	"Closed-Open"	2 Position
55	Thermal Column Shutter	"Off-On"	2 Position
56	Pneumatic Tube Blowers	"Off-On"	2 Position
57	Airlock Door Open	N/A	Push Button
58	Range Switch	N/A	18 Position
59	Intercom	N/A	Light
~~60~~	~~Temperature Readout~~	~~N/A~~	~~24 position~~ 2004
60	Bridge Door Unlock	N/A	Push Button 2008

~~77 Fission Chamber Control 3 Pos. Switch~~

Table 9.2 presents a summary of the devices installed in the instrument cubicle. The "Number" column Table 9.2 refers to the identification on Figure 9.3.

TABLE 9.2

INSTRUMENT CUBICLE DEVICES

Item No.	Function
"1	Annunciator
2	Source Range Monitor Level Recorder
3	Intermediate Range Monitor Level Recorder - 2 Pen
4	Wide Range Monitor Level Recorder - 2 Pen
5	Power Range Monitor Level Recorder - 3 Pen
6	Multiscaler

2001

* TABLE 9.2 REVISED IN IT'S ENTIRETY BY 1981-1982 REV. REVISED IN ITS ENTIRETY BY 2001 REV.

1981-1982
1995
2001
2002
2006

TABLE 9.2 (Cont'd)

INSTRUMENT CUBICLE DEVICES

ITEM NO.	FUNCTION
~~6~~	~~Reactor Water Inlet-Outlet Temperature Recorder, 2 Pen~~
7	Neutron Flux Monitor - Signal Processor Drawer No. 1
8	Neutron Flux Monitor - Signal Processor Drawer No. 2
9	Neutron Flux Monitor - Channel No. 6 Drawer
10	Neutron Flux Monitor - Wide Range Monitor Drawer
11	Annunciator & Interlock Relay Drawer
12	Servo Amplifier Drawer
13	Auxiliary Annunciator - Panalarm
14	Non-Coincidence Logic Unit - Reactor Safety System "Yellow Leg"
15	Trip Actuator Amplifier - Reactor Safety System "Yellow Leg"
16	Non-Coincidence Logic Unit - Rod Run-In System
17	Trip Actuator Amplifier - Rod Run-In System
18	Non-Coincidence Logic Unit - Reactor Safety System "Green Leg"
19	Trip Actuator Amplifier - Reactor Safety System "Green Leg"
20	20 VDC Regulated Power Supply Drawer (2PS1)
21	20 VDC Regulated Power Supply Drawer (2PS2)
22	Control Blade Drop Timer Circuit
23	Rod Position Indication Drawer
24	Reactor Safety System Relay Drawer
25	Primary Coolant System Pressure Meter - PT 943
26	Primary & Pool Coolant Demineralizer Flow Recorder - 2 Pen
27	Dual Alarm Unit (EP 920E/F) Primary Low Flow Scram 980B
28	Dual Alarm Unit (EP 953A/B) - Primary Coolant High Temperature Scram
29	RTD Transmitter (EP 903B) - Primary Coolant T_h
30	Isolated Power Supply (EP 911A) - Reactor Safety System "Yellow Leg"
"31	Square Root Transmitter (EP 919A) – Primary Flow "A" Loop"
32	Dual Alarm Unit (EP 920A/B) - Primary & Pool Low Flow Scrams
"33	Square Root Transmitter (EP 919F) – Pool Flow "B" Loop"
34	RTD Transmitter (EP 903A) - Primary Coolant T_c
"35	Adder-Subtractor Module (EP 954) - Primary Coolant Differential Temperature
36	MV/I Transmitter (EP 955) - In-Pool Heat Exchanger Differential Temp.
37	RTD Transmitter (EP 903C) - Pool Coolant T_c
"38	Adder-Subtractor Module (EP 952) - Pool Coolant Differential Temperature
39	RTD Transmitter (EP 903D) - Pool Coolant T_h
40	Square Root Transmitter (EP 919C) - Primary Demineralizer Flow
41	Square Root Transmitter (EP 919D) - Pool Demineralizer Flow
42	Isolated Power Supply (EP 911B) - Reactor Safety System "Green Leg"
43	Square Root Transmitter (EP 919E) – Primary Flow "A" Loop
44	Dual Alarm Unit (EP 920C/D) - Primary & Pool Low Flow Scrams
"45	Square Root Transmitter (EP 919B) – Pool Flow "A" Loop"
46	Square Root Transmitter (EP 919G) - Primary Flow "B" Loop
47	Power Mode I, II, & III Indication Lights
48	Clock
49	"REACTOR ON" Light
50	Annunciator Alarm Power Switch - 3 Position
51	Area Radiation Monitor - Beamport Floor North Wall
52	Area Radiation Monitor - Beamport Floor West Wall
53	Area Radiation Monitor - Beamport Floor South Wall
54	Area Radiation Monitor - Containment Building Exhaust No. 1
~~38~~	~~Noncoincidence Logic Unit Scram/Yellow~~

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198
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2001
200
200

TABLE 9.2 (Cont'd)

INSTRUMENT CUBICLE DEVICES

ITEM NO.	FUNCTION
55	Secondary Coolant System Radiation Monitor
56	Area Radiation Monitor - Reactor Pool Bridge ALARA
57	Area Radiation Monitor - Reactor Pool Bridge
58	Area Radiation Monitor - Beamport Floor East Wall
59	Area Radiation Monitor - Fuel Vault
60	Area Radiation Monitor - Mechanical Equipment Room (Room 114)
61	Area Radiation Monitor - Containment Building Exhaust No. 2
62	Fuel Element Failure Radiation Monitor
63	Circuit Fuses (7)
64	Reactor Pool Bridge Radiation Monitor Upscale Switch & Light
65	Containment Ventilation Isolation Door 504 Stop Push Button
66	Containment Ventilation Isolation Door 504 Close Light
67	Containment Ventilation Isolation Door 504 Open Push Button & Light
68	Containment Ventilation Isolation Door 505 Stop Push Button
69	Containment Ventilation Isolation Door 505 Close Light
70	Containment Ventilation Isolation Door 505 Open Push Button & Light
71	Valve 552A Open Indication Light
72	Valve 552A Closed Indication Light
73	Valve 552B Open Indication Light
74	Valve 552B Closed Indication Light
75	Valve 552B Control Switch - 2 Position – "Open-Normal"
76	Valve 527D Open Indication Light
77	Valve 527D Closed Indication Light
78	Valve 527D Control Switch - 2 Position - "Open-Normal"
79	Valve Control Switches[a] - 2 Position - "Auto-Man"
80	Valve Control Switches[b] - 2 Position - "Open-Close"
81	Pump Control Switches[c] - 2 Position - "Off-On"
82	Cooling Tower Fan Control Switches - 3 Position - "Fast-Off-Slow"
83	Valve 547 Position Indication Light
84	Heavy Equipment Entry (Door 101) Door Ajar Indication Light
85	Primary Coolant T_h - T_c Recorder - 2 Pen
86	Primary Coolant System Temperature Controller (S-1)
87	Pool Coolant T_h - T_c Recorder - 2 Pen
88	Pool Coolant System Temperature Controller (S-2)
89	Primary Coolant System "A" Loop Flow Recorder - 2 Pen
90	Primary Coolant System "B" Loop Flow Recorder - 2 Pen
91	Pool Coolant System Flow Recorder - 2 Pen
92	Closed Circuit Television Monitor
93	Pool Coolant System Differential Temperature Meter
94	Primary Coolant System Differential Temperature Meter
95	Pressurizer Water Level Indication Meter
96	In-Pool Heat Exchanger Differential Temperature Meter
97	Reactor Pool Reflector Region Differential Pressure Meter - PT 917
98	Reactor Core Outlet Pressure Meter - PT 944A
99	Reactor Core Outlet Pressure Meter - PT 944B
100	Primary Coolant HX503A Outlet Temperature Meter - TE 980A
101	Primary Coolant HX503B Outlet Temperature Meter - TE 980B
102	Reactor Core Differential Pressure Meter - DPS 929
~~72~~	~~Vent Duct MO Door 1 Indicator Light and Switch Push Button~~
~~73~~	~~Vent Duct MO Door 2 Indicator Light and Switch Push Button~~

Secondary Coolant Radiation Monitor 1995 52

Room 114 Area Radiation Monitor 57 1995

1981/1982 2001

2006 1995

2006

NEW PAGE

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1990
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2006

TABLE 9.2 (Cont'd)

INSTRUMENT CUBICLE DEVICES

Item No.	Function
103	Valve 16A Closed Indication Light
104	Valve 16A Open Indication Light
105	Valve 16B Closed Indication Light
106	Valve 16B Open Indication Light
107	Fan Failure Alarm Panel
108	Secondary Coolant System Recorder - 3 Pen (Temperatures & Flow)
109	Secondary Coolant System High Temperature Alarm Light
110	Drain Collection System Control Panel
111	Digital Temperature Readout (TE 980/990)
112	Emergency Diesel Generator Alarm Panel
113	Hot Cell Isolation Valve Position Indication & Remote Operator
114	Reactor Power Calculator
115	Reactor Safety System Monitoring Circuit
116	Pool Coolant Flow Bypass Switch (2S40) - 4 Position
117	Primary Coolant Flow Bypass Switch (2S41) - 4 Position
118	Primary Coolant System Conductivity Meter - Demineralizer Inlet
119	Primary Coolant System Conductivity Meter - Demineralizer Outlet
120	Pool Coolant System Conductivity Meter - Demineralizer Inlet
121	Pool Coolant System Conductivity Meter - Demineralizer Outlet
122	Fire Main Low Pressure Alarm Light
123	Door Open Alarm (Room 114, Cooling Tower, Demineralizer Area)
124	Alarm Cutout Switches - Door, Firemain, Secondary pH
125	Domestic Cold Water (DCW) Low Pressure Alarm Light
126	Fire Main Low Pressure Alarm Panel
127	Pneumatic Tube System Irradiation Counter
128	Off-Gas Radiation Monitor Recorder - 3 Pen
129	Off-Gas Radiation Monitor Flow Alarms & Cutout Switch
130	Off-Gas Radiation Monitor Recorder - 3 Pen
131	Secondary Coolant Transmitter Selector Switch
132	Emergency Diesel Generator Elapsed Time Run Meter
133	Dual Alarm Unit (EP 920G/H) Primary Low Flow Scram
134	Square Root Transmitter (EP 919H) - Primary Flow "B" Loop
135	Radioactive Liquid Waste System Alarm Panel
136	Fire Protection System Alarm Panel

[a]Auto/Manual control switches for the following valves: V546A/B, V507A/B, V509, V545, V526, V527A, and V527B

[b]Open/Close control switches and indication lights for the following valves: V546A/B, V507A/B, V509, V543A/B, V527E, V527F, V545, V526, V527A, V527B, and V527C.

[c]Off/On control switches and indication lights for the following pumps: SP-1, SP-2, SP-3, P501A/B, P508A/B, P513A/B, and P533 (Off/Auto).

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TABLE 9.2 (Cont'd)

INSTRUMENT CUBICLE DEVICES

Item No.	Function
173	Mode Selector Switch 2S40
174	Mode Selector Switch 2S41
175	Drain Collection Control Panel
176	980/990 RTD Readout
177	Conductivity Amplifiers
193	Valve 552A Open Light
194	Valve 552A Closed Light
195	Valve 552B Open Light
196	Valve 552B Closed Light
197	Valve 552B Open/Normal Switch
198	Valve 527D Open Light
199	Valve 527D Closed Light
200	Valve 527D Open/Normal Switch
202	903A/B Reactor Water Inlet Temperature
203	903A/B Reactor Water Outlet Temperature
204	953 Reactor Water Outlet Hi Temperature Scrams
205	954 Reactor Water Differential Summer Temperature
207	In-Pool HX ΔT
208	903C/D Pool Water Outlet Temperature
209	903C/D Pool Water Inlet Temperature
210	952 Pool Water Differential Summer Temperature
211	911A/B Power Supply GE/MAC
212	919A/B Reactor Water Flow Square Root Converter
	912C/D Reactor Water Flow Square Root Converter
213	919A/B Pool Water Flow Square Root Converter
	919C/D Pool Water Flow Square Root Converter
214	955 In-Pool HX Temperature Differential
215	911A/B Power Supply GE/MAC
216	919A/B Reactor Water Demin. Square Root Converter
	919C/D Reactor Water Demin. Square Root Converter

DELETE

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TABLE 9.2 (Cont'd)

INSTRUMENT CUBICLE DEVICES

Item No.	Function
217	920A/B Reactor and Pool Water Low Flow Scram
218	919A/B Pool Water Demin. Square Root Converter
	919C/D Pool Water Demin. Square Root Converter
219	920C/D Reactor and Pool Water Low Flow Scram
220	911C Power Supply GE/MAC
221	Blower
222	Room 114 Door Open
223	Cooling Tower Tunnel Door Open
224	DI-200-Di-201 Door Open
225	DI-202-R200 Door Open
226	Cooling Tower Door Open
227	Alarm Cut Out Flasher
228	DCW Low Pressure Alarm
229	Fire Main Low Pressure
230	Channel 5 and 6 Recorder
231	Rod Drop On/Off Power Switch
232	Ro (Not Used) (Not Used) et Push Button
233	Co (Not Used) e Drop Time
234	Ro (Not Used) (Not Used) imer
235	Ro (Not Used) (Not Used) imer
236	Rod C Drop Timer
237	Rod D Drop Timer
238	Alarm Cutout Switches
239	Mode Annunciation
240	RTD 980A Meter
241	RTD 980B Meter
242	DPS 928A Meter
243	DPS 928B Meter
244	DPS 929 Meter
245	919F Pool Water Flow Square Root Converter
246	920F Pressurizer Level Alarm Unit

DELETE

1995

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NEW PAGE

1981-
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1995
2001

TABLE 9.2 (Cont'd)

INSTRUMENT CUBICLE DEVICES

Item No.	Function
247	920G Pressurizer Level Alarm Unit
248	920H Pressurizer Level Alarm Unit
249	Pressurizer Level Indication
255	PS 944A Meter
256	PS 944B Meter
258	V547 Position Indication
259	919E Reactor Water Flow Square Root Converter
260	Nuclepore Exhaust Valves
261	Hot Cell Isolation Valve
263	**Pneumatic Tube Irradiations Counters**

DELETE

1995

1981-1982/2001

1995
1981-1982
2001

TABLE 9.2 (cont'd)

Number	Function
68	13 Valve "Open-Closed" Indicator Lights
69	Demineralized Water Storage Tank (T-300) Level Indicator
70	Demineralized Water Storage Tank (T-301) Level Indicator
71	Vent Duct Door Motor P.B. Switch
72	Vent Duct Door Motor P.B. Switch
73	Vent Duct Door Motor P.B. Switch
74	Vent Duct Door Motor P.B. Switch
75	7 Branch Circuit Protection Fuses

DELETE

1981-1982
2001

9.3 Nuclear Instrumentation

9.3.1 General Description

Reactor power is monitored during startup and operation by six channels of nuclear instrumentation. These are; one startup channel, two intermediate range channels, and three power range channels. The startup channel has no output to the safety system but it must be on scale and indicating at least one count per second before startup can begin. The intermediate range channels provide both period and power level information. **"The intermediate range channels provide trip outputs on short period."** ~~p outputs on both short period and high period power.~~ The power range channels provide an output to the automatic control system and trip outputs to the safety system that actuate on high power.

1995

The nuclear instrument main chassis and their shared power supplies are mounted on the instrument panel in the control room. Channel outputs are displayed both locally and on the control console.

1995

9.3.2 Startup Channel

~~The startup channel detector is a fission chamber. The fission chamber is mounted in a water tight container and located in the pool outside the reflector region at approximately the elevation of core center line. Fission chamber position is adjustable both vertically and radially in the pool. The fission chamber is connected to a drive system so that it maybe operated in any one of three predetermined positions. When not in use fission chamber is withdrawn into a thermal neutron shield and deenergized.~~

"The startup channel detector is a fixed position fission chamber with 45 inches of sensitive length. The fission chamber is mounted in a water tight drywell enclosure and located in the pool outside the reflector region approximately centered at the elevation of core centerline.

Pulses from the fission chamber are first received by a remote amplifier mounted on the biological shield at the mezzanine level. The amplifier assembly contains the signal conditioning circuitry, dc power supplies and the high voltage detector excitation supply.

From the amplifier the conditioned signal is fed into the signal processor assembly in the rack mounted instrument drawer in the control room. The signal processor provides further processing of the detector signal into signals which are a measure of the logarithm of count rate and rate-of-change of count rate. The log count rate signal drives a local panel indication, a remote meter mounted on the control console, and a log count rate recorder. The period indication is displayed on a local indication and a remote period meter mounted on the control console.

The discriminated output of the source range channel is also delivered to a panel mounted scaler. The scaler provides count rate information that is used to measure subcritical multiplication during a startup."

~~differentiates the log count rate signal and drives a local amd remote period meter. The remote period meter is mounted on the control console.~~

~~The discriminated output of the pulse amplifier is also delivered to a panel mounted scaler. In addition to receiving pulses from the fission chamber, the scaler drives an audio pulse amplifier which is mounted on the control console, thus giving the reactor operator an audible indication of fission chamber counting rate.~~

1995-

1996

9.3.3 Intermediate Range (Log N)

~~Two identical intermediate range channels are functioning during all phases of reactor operation. The intermediate range neutron detectors are compensated ion chambers mounted in the pool outside the~~

Replace with:

"Two intermediate range channels are required to be operable during all phases of reactor operation. The intermediate range neutron detectors may either be compensated ion chambers (CIC) or fission chambers mounted in the pool outside the reflector region at approximately core centerline elevation in water tight drywells. The intermediate range detector locations are designed such that detector locations may be varied both vertically and radially. The compensated ion chamber (CIC) based intermediate range channel delivers a d.c. signal proportional to neutron flux to an intermediate range monitor chassis which is mounted on the instrument panel. The intermediate range monitor develops an output which is proportional to the logarithm of ion chamber current. The logarithmic output is delivered to local and remote level indicators and to a recorder. The logarithmic output also drives a period amplifier which delivers a period signal to two independently adjustable trip circuits and to a local and remote period indicator. The remote period and level indicators are located on the control console.

The fission chamber based intermediate range channel is part of wide range monitor that operates in both the pulse counting mode and the mean-square-voltage (MSV) mode.

The Wide Range signal is a series of randomly spaced pulses superimposed on a d.c. voltage. The pulse signal is processed by one of the log count rate and rate-of-change circuit boards and the d.c. voltage signal is processed by the log amplifier, rate-of-change, and auctioneer circuit board. The log count rate circuit provides an output that is proportional to the logarithm of the average count rate of pulses of the Wide Range signal over the range of about $10^{-8}\%$ to $3 \times 10^{-2}\%$ power. The log amplifier circuit provides an output that is proportional to the logarithm of the mean square variation of the Wide Range signal over the range of about five decades, from $10^{-3}\%$ to 200% of reactor power.

The two signals are combined by the auctioneer circuit to provide one continuous output over the range of $10^{-8}\%$ to 200% of reactor power. Two rate-of-change circuits associated with these log signals provide outputs that are also combined to provide one continuous rate-of-change signal over the full reactor flux range. The combined log output and period output are displayed locally on the plasma displays on the front panel and remotely on the control console.

The two intermediate range level indications will be capable of being recorded, although only one IRM level indication is required to be recorded."

~~A single intermediate range power level recorder serves both intermediate range monitors. The channel to be recorded is switch selected by the operator at the control console.~~

1996

1995
1968-
1969
2001

9.3.4 Power Range (Linear Level) Channels

Two of the three power range channels are identical and monitor reactor power from shutdown condition to full power in a single range. The detectors for thes... cor... ...ers and ... out- put ... to loca... reco... ...p circ... modi...

"Three power range channels are required to be operable and provide scram and rod run-in trips during all phases of reactor operation. The power range neutron detectors may be either compensated or uncompensated ion chambers or fission chambers mounted in the pool outside the reflector region at approximately core centerline elevation in water-tight drywells. The power range detector drywells are designed to allow both vertical and radial adjustment, if necessary.

One power range channel will monitor reactor power from shutdown to full power in discrete linear ranges. The output of the Wide Range Monitor is delivered to the servo amplifier system and provides the controlling signal for the regulating blade when the reactor is operating under automatic control. This channel may be one of the three power range channels required to provide scram trips. If this channel does not provide scram trips, a total of four power range channels will be required (three others with scram trips).

The output of the power range level monitors is delivered to local and remote meters, power level recorders, and for three of the monitors to two adjustable trip circuits (scram and rod run-in). The power level remote meters are located on the reactor control console."

The... **"Th... pov... thi...** ...eactor ...or for Pow... which is located near the uncompensated ion chambers.

1961

2001

The output of the Wide Range Monitor is delivered to two independently adjustable trip circuits, remote and local level indicators, and to the power level recorder. The remote power level indicator is mounted on the control console.

Power level, as measured and indicated by any one of the three power level monitors is continuously recorded. Power level indication from the power range monitor units is recorded on a dual unit recorder located in an auxiliary panel next to the instrument cubicle. The wide range monitor power level indication is recorded on a recorder located on the instrument cubicle in front of the control console.

Po... of... Th... th... ...ne ...ed. ...n

1968-1969

The output of the Wide Range Monitor is delivered to the servo amplifier system and provides the controlling signal when the reactor is operating under automatic control.

1995

9.4 Safety System

The reactor safety system includes scram logic circuits and trip actuator amplifiers, rod run-in logic and trip actuator amplifiers, and the alarm and annunciator circuits.

9.4.1 Scram Logic and Trip Actuator Amplifiers

The four reactor shim rods are attached to the rod drive mechanisms by d.c. electromagnets. Magnet current is controlled by the trip actuator amplifiers. Two trip actuator amplifiers operate in parallel, each controlling current to two magnets. The magnets are deenergized by the trip actuator amplifiers in response to a signal from the scram logic circuit. ~~The trip actuator~~

"The trip actuator amplifiers can be reset when the scram input signal is cleared and the "Scram Reset" button is depressed." 1995

There are two logic units, each driving both trip actuator amplifiers. ~~The logic units initiate a "scram" whenever any one of nine inputs is~~

"The logic units initiate a "scram" whenever any one of its inputs is interrupted." 1995

~~connected in parallel.~~

The scram system is depicted in Figure 9.4 showing the conditions which will initiate a scram and how some of these may be bypassed.

9.4.2 Rod Run-In Logic and Trip Actuator Amplifiers

Rod run-in is initiated through a trip actuator amplifier by removing current to two rod run-in relays. The trip actuator amplifier receives its signal from two nine input logic units operated in parallel.

1995
2007

"Once tripped, the trip actuator amplifier must be reset by depressing the "Rod Run-In" reset button after the rod run-in input signal is cleared." 1995

~~button.~~ Rod run-in is initiated manually by removing a.c. power to the trip actuator amplifier.

The rod run-in system is depicted in Figure 9.5 showing the conditions which will initiate a run-in and how certain of these may be bypassed.

9.4.3 Alarm and Annunciate System

The reactor operator is alerted to any abnormal condition by annunciator action. The annunciator panel is a 6x10 array of 3 5/8x3 3/8 inch windows. Each window is back lighted and illuminates red for scram conditions, blue for rod run-in conditions and white for simple alarm conditions. To ensure that the annunciator panel remains operable, an out of normal or loss of power condition is indicated on the auxiliary annunciator. 2007

The annunciator sequence is shown in Table 9.3 below.

TABLE 9.3

ANNUNCIATOR SEQUENCE

Condition	Illumination	Audible Signal
Normal	Off	Off
Alarm Input	Flashing	On
Alarm Input Acknowledged	On Bright	Off
~~Retu~~ "Return to Normal	Flashing Dim	On" ~~Off~~
Reset	Off	Off
Test	Flashing	On

1995

1981-1982
1995
1970-1971

The annunciator sequence is initiated by any of the following conditions.

(1) Any scram or rod run-in condition.

(2) Startup channel at less than 1 count per second.

(3) High activity in the off-gas stack.

(4) High activity in the secondary coolant.

(5) High activity in the reactor coolant loop.

(6) Nuclear instrument anomaly.

(8) "Water tank T-301 Hi or Low Level" ~~ater reserve tank.~~ 1981-1982

(9) "Water tank T-300 Hi or Low Level" ~~er storage tank.~~

(10) Low water level in the utility seal trench.

(11) Low pressure in the nitrogen system.

(12) Rods not fully inserted.

(13) Regulating rod less than "20% withdrawn". 1981-1982

(14) Regulating rod 60% withdrawn. 1970-1971

(15) "Jumper board in use" ~~pressure across reflector region.~~ 1981-1982

(16) "Thermal column door open" ~~r.~~

(17) "Reactor loop high temperature." ~~hanger high inlet temperature.~~ 1995

(18) Low flow in the reactor loop.

(19) Low flow in the reactor or pool coolant cleanup loop.

(20) Reactor coolant temperature control valve on a limit.

(21) Reactor convective cooling loop valve not fully closed.

(22) Reactor coolant pump on with isolation valves closed.

(23) High temperature in the pool loop.

(24) Low flow in the pool loop.

(25) "Fuel vault intrusion" ~~emperature across pool loop heat exchanger.~~ 1981-1982

-1995
-1987-1988
-1970-1971
-1968-
-1969
-1998

"(26) Anti-siphon tank high or low pressure." 1995

(27) Reactor or pool loop high conductivity.

(28) Low pressurizer pressure.

(29) High pressurizer pressure.

(30) High pressurizer water level.

(31) Low pressurizer water level.

1970-1971 1995-

(32) Reg blade out of AUTO ← 1968-69

(33) Rod control power unavailable

(34) Channel 4, 5, or 6 downscale

(35) Secondary low flow"

"(36) **Reflector valve 547 off open.**"

ADD TO LIST

9.5 Rod Withdrawal Prohibit Circuits 1987-1988

Before control rods can be pulled to start the reactor, several conditions must have been met. These conditions are listed below.

(1) Master switch in the "ON" position.

"(2) **"Rod Run-In" and "Scram" switches are reset."** 1995

(3) No nuclear instrument anomalies.

(4) Shim rods bottomed and in contact with magnets.

"(5) Startup channel more than 1 count per second or IRM recorder indication greater than 1×10^{-5}% power. ~~ond.~~ 1995 1998

(6) Thermal column door closed.

In condition (4) above, once the rods have bottomed and contact with magnets is made, rods may be withdrawn. If a rod loses contact with a magnet it must be bottomed before it can again be withdrawn.

The ~~startup interlock~~ rod withdrawal prohibit circuits may be bypassed if these conditions are met: 1987-1988

(1) Master switch in "Test" position.

(2) Rod magnets are deenergized.

9.6 Rod Control System

Reactor power is controlled either manually or automatically. The reactor is taken from the shutdown condition to operating power manually. The reactor may be put into automatic control after a minimum power has been reached.

1981-
1982

9.6.1 Manual Control

After all preoperational conditions have been met shim rods or the regulating rod are withdrawn by operating a "pistol-grip" 3 position spring-return-to-center switch.

The regulating rod operates at a speed of approximately "40" inches per minute either inserting or withdrawing. 1981-1982

The four shim rods may operate either individually or ganged. These rods drive out at a rate of 1 inch per minute and drive in at a rate of 2 inches per minute.

9.6.2 Automatic Control

When the Wide Range Monitor range switch is set for a full scale of greater than 1000 watts, the power range recorder is upscale beyond a set point, the reactor period is greater than the period auto-control interlock set point, and the reactor is shimmed so that the regulating rod is at least "60%" withdrawn the reactor may be placed in automatic control by depressing a console mounted push button. 1981-1982

The system will remain in automatic control until any one of the following actions occur:

(1) Control mode transferred to manual.
(2) Regulating rod control switch actuated.
(3) Regulating rod reaches less than 10% withdrawn.
(4) Any scram or rod run-in occurs.
(5) Regulating rod reaches its lower limit.

1981-
1982

When in automatic control, the controlling parameter is reactor power level as measured through the neutron flux detected by a compensated ionization chamber. The ionization chamber output current drives the Wide Range Monitor whose output voltage is proportional to ion chamber current.

The output of the Wide Range Monitor is delivered to the Servo Amplifier which senses the error between reactor power and the power demand set point and actuates relays which cause the regulating rod to withdraw or insert in response to the sense of the power error and in a direction to reduce the error.

The power set point is selected by the reactor operator. A power set potentiometer is motor driven through a console mounted switch. The power demand set point is indicated on a console mounted panel meter.

In automatic control the shim rods also adjust automatically but only in a direction to reduce core reactivity. The automatic shimming circuit is activated when the regulating rod inserts to the 20% withdrawn position. At this point the shim rods drive in for a preset time and then stop. After a "wait" period has elapsed the cycle is repeated. The alternate "drive and wait" action will continue until the regulating rod, which remains on servo control, has returned to the "60%" withdrawn position. At this point the automatic shimming circuit is deactivated until the regulating rod has again inserted to the 20% withdrawn position. The "drive" and the "drive 1981-1982

1990 - 1991
1995
2000

plus wait" periods are independently adjustable to facilitate setting an optimum automatic shimming cycle.

9.7 Radiation Monitoring Systems

9.7.1 Area Radiation Monitoring System

The area radiation monitoring system monitors radiation at the locations and in the ranges as shown in Table 9.4. With the exception of the detectors located in the building exhaust air plenum, each detector is equipped with a high level alarm light.

~~plenum and on the reactor bridge, each detector is equipped with a high level alarm light.~~

TABLE 9.4

AREA RADIATION MONITORING SYSTEM

DETECTOR LOCATION	DETECTION RANGE
BEAM PORT FLOOR SOUTH WALL	0.1 -10K mR/hr
BEAM PORT FLOOR WEST WALL	0.1 -10K mR/hr
BEAM PORT FLOOR NORTH WALL	0.1 -10K mR/hr
FUEL STORAGE VAULT	0.1 -10K mR/hr
BUILDING EXHAUST AIR PLENUM #1	0.1 -10K mR/hr
BUILDING EXHAUST AIR PLENUM #2	0.1 -10K mR/hr
REACTOR BRIDGE	0.1 -10K mR/hr
"Beam Port Floor East Wall"	0.1 -10K mR/hr
REACTOR BRIDGE ALARA	1 - 100K mR/hr"
COOLING EQUIPMENT ROOM	1 - 100K mR/hr

1990-1991
2000
1995

~~recorded on a multipoint radiation recorder on the control room instrument panel. Each of the~~

"The signal from each of the remotely located detectors is displayed on panel meters on the control room instrument panel. Each of the panel meters is equipped with an adjustable alarm contact which illuminates a lamp at that meter and initiates an audible alarm whenever the detected radiation level exceeds the setpoint.

~~radiation level exceeds the set point.~~

1990 -
1991
1995

~~Additionally, when the radiation level sensed by the building exhaust air plenum detector exceeds~~

Additionally, when the radiation level sensed by either of two containment building exhaust air plenum detectors or either of two reactor bridge monitors exceeds the trip level a "reactor isolation" is initiated. In a reactor isolation:

(1) The reactor is scrammed;
(2) The containment building is sealed (all ventilation openings are closed); and
(3) The containment building exhaust and supply fans are de-energized."

~~openings are closed), and~~
~~(3) The exhaust and supply fans are deenergized.~~

1995

9.7.2 Fuel Rupture Monitoring System

~~The fuel rupture monitoring system monitors~~

The fuel rupture monitoring system monitors reactor coolant water for the presence of the fission product activity. The method employed utilizes the technique of measuring the short-lived iodine isotopes. A scintillation detector measures the activity deposited on an anion exchange resin column and subsequent electronic circuitry selects that activity due to radioiodine.

~~that activity due to radioiodine.~~

~~A portion of the reactor coolant bypass cleanup~~

A portion of the reactor coolant bypass cleanup loop is taken off upstream of the demineralizer and passed through a filter, a cation column and an anion column. The scintillation detector views the anion column and senses any collected activity. Detector output is fed to a channel of the Eberline RMS II, where the signal is processed. This

"This information is indicated on a control room panel meter that is equipped with an adjustable high level alarm, which also initiates an annunciator alarm."

1995

~~If the discriminated detector count rate exceeds a preset level an annunciator alarm is initiated.~~

1990 - 1991

1990-1991
1995
1981-1982
1981
1999

9.7.3 Off-Gas Radiation Monitoring System

"The Off-Gas Radiation Monitoring System consists of a three-channel radiation detection system designed to measure the airborne concentrations of radioactive particulate, iodine, and noble gas in the exhaust air which is sampled by an isokentic probe located in the facility ventilation exhaust plenum. The radiation detection system is a self-contained unit consisting of a fixed filter monitored by a beta scintillation detector, a charcoal cartridge monitored by a gamma scintillation detector, and a gas chamber monitored by a beta scintillation detector shielded by three inches of lead (4π configuration). One-inch lead shields separate the individual detectors. The output from each radiation detector is displayed on a local meter in counts per minute (cpm) and on a strip-chart, three-pen recorder mounted on the instrument panel in the reactor control room. An audible and visual alarm alerts the operator to high activity or abnormal flow through the radiation detection system." 1980-1981 1995

9.7.4 Secondary Coolant Monitoring System

~~The secondary coolant monitoring system is~~
A scintillation detector is installed in the return leg of the secondary piping. The output of the scintillation detector is fed to a channel of the Eberline RMS II, where the signal is processed. ~~This information is indic~~ "This information is indicated on a control room panel meter that is equipped with an adjustable high level alarm, which also initiates an annunciator alarm." 1995 1990-1991

9.8 Process Instrumentation and Control

ADD AFTER HOR 1981-1982

Plant process parameters that are monitored and controlled are; temperature, flow and pressure. Other parameters which are simply monitored or have an abnormal condition are; coolant conductivity, demineralized water storage and reserve tank levels, utility seal trench level, and low nitrogen pressure.

See Sections 9.14 and 9.15 for 10MW upgrade modifications.

9.8.1 Temperature Measurement and Control

9.8.1.1 Reactor Loop Coolant

Reactor coolant temperature is measured at these points:

1995
1981-1982
1968-1969
2001

(1) At reactor outlet just downstream from the outlet isolation valve (Reactor Outlet Temperature).

(2) At the reactor loop heat exchanger outlet (Reactor Inlet Temperature).

These temperatures are recorded on a two pen recorder mounted on the instrument panel. Temperature difference is displayed on a panel meter.

If either sensor detects a temperature of more than "120%" of the normal value an alarm is initiated. ~~If th~~ "If reactor outlet temperature exceeds 125% of normal, a scram is initiated." ~~is initiated.~~ 1981-1982 2001

Reactor coolant temperature is controlled by controlling inlet (heat exchanger outlet) temperature. Control is achieved by varying secondary flow through the heat exchanger. ~~Whenever the control valve reaches either limit, indicating~~ "Whenever the control valve approaches either limit, indicating that little control is available, the reactor operator is alerted by annunciator action." ~~operator is alerted by annunciator action.~~ 1995

In addition to the normal flow temperature measurements, the temperature differential across the in-pool heat exchanger (reactor convective loop) is measured and displayed on the control room instrument panel.

9.8.1.2 Pool Loop Coolant

Pool coolant temperature is measured at the heat exchanger inlet and outlet. ~~and at a point approximately 12 inches below the surface of the pool.~~ DELETE 1968-1969

2006 1981-1982 1995 1970-1971

Heat exchanger outlet temperature and the pool temperature are displayed on a two pen recorder on the control room instrument panel. The heat exchanger differential temperature is indicated on a control room panel meter.

"An annunciator alarm occurs if the heat exchanger inlet temperature is abnormally high." 1970-1971

~~temperature are abnormally high.~~

Pool loop temperature is controlled by varying heat exchanger secondary flow. The controlling parameter is heat exchanger outlet temperature.

9.8.2 Flow Measurement and Control

9.8.2.1 Reactor Loop

~~The reactor loop flow measurement system includes one orifice plate. and three flow switches.~~ 1981-1982

~~Flow, as measured by the orifice plate, is recorded in the control room on the instrument panel. The~~

~~"The re... heat exchan... Flow, a... control room... The ori... normal flow... valves oper... e in each of two... s recorded on a... w. At 85% of... phon break~~ 1995

"The reactor loop flow measurement system includes two orifice plates, one in each of the two heat exchanger legs. Flow, as measured by two transmitters connected to each orifice plate, is recorded on two, two-pen recorders mounted on the control room Instrument Panel. The orifice plate flow measuring channels will alarm at approximately 95% of normal flow. At approximately 90% of normal flow the core isolation valves, reactor convective loop valves, and the siphon break valves operate as the reactor scrams." 2006

~~The orifice plate flow measuring channel will alarm at 90% of normal flow. At 85% of normal flow the core isolation valves, reactor convective loop valve and the siphon break valves operate and the reactor scrams.~~

1995
1981-
1982

~~The three flow switches operate when the coolant flow drops to 50% of normal. Operation of any of these switches causes the core isolation valves to close, the convective loop and siphon break valves to open and the reactor to scram.~~ DELETE 1981-1982

9.8.2.2 Pool Loop

Pool loop flow is monitored by measuring the pressure drop across the reflector region, and by an orifice flow meter on the outlet side of the pool loop heat exchanger.

~~Flow as measured by the orifice plate, is recorded on the control room instrument panel mounted two pe~~ "Flow, as measured by two transmitters connected to this orifice plate, is recorded on a control room instrument panel mounted two pen recorder." ~~ugh a c when the cooling plant is expanded for 10 megawatt peak power operation.~~ 1995

An annunciator alarm occurs when flow through the pool heat exchanger drops to 90% of normal.

~~If t~~ "If the flow reaches 70.8% of normal, the reactor scrams." ~~tor is scrammed.~~ 1995

"When the pressure drop across the reflector region becomes high or low a reflector high/low differential pressure scram occurs." ~~becomes excessive an~~ 1995

~~In addition to the flow monitoring devices the pool coolant loop has a pressure switch on the~~

"In addition to the flow monitoring devices the pool coolant loop has a pressure switch on the discharge side of the coolant circulation pumps. When the pumps discharge dynamic head decreases to the switching level, the pumps shut off, the pool isolation valve closes, and the reactor scrams."

~~pool isolation valve closes and the pool natural convection valve opens, and the reactor is scrammed.~~ 1995

1995
1967-
1968

9.8.3 Pressure Measurement and Control

9.8.3.1 Low Pressure Safety Action

~~Two pressure switches, electrically connected in~~

"Two pressure switches are installed in the outlet leg of the reactor coolant loop. When the loop pressure at these switches decreases to approximately 25 psig the following action takes place." 1995

~~13.7 psig the following action takes place.~~

(1) Reactor loop isolation valves close.

(2) Anti-siphon valves open.

(3) Reactor convective loop valve opens.

"(4) Reactor coolant circulation pumps stop." ~~on pump stops.~~ 1995

(5) Reactor Scrams.

9.8.3.2 Reactor Pressurizer System

The pressurizer system functions to maintain the core inlet pressure at 65 psia. Pressurizer pressure is displayed on the control room instrument panel.

Six pressure switches installed on the pressurizer perform the following functions:

(1) At 85% of normal pressure the reactor scrams.

(2) At 90% of normal pressure the annunciator alarms.

(3) At 95% of normal pressure nitrogen gas is let into the pressurizer.

(4) At 105% of normal pressure nitrogen gas is released from the pressurizer.

(5) At 110% of normal pressure the annunciator alarms.

(6) At 115% of normal pressure the reactor scrams.

I **"Pressurizer water level is maintained by adding water with a positive displacement pump if the level is low, and dumping water to the drain collection system if the level is too high."** 1995
~~is low and dumping water to the hot waste system~~
~~if the level is too high.~~ If the pressurizer water level becomes very low it is isolated from the balance of the system. ~~Very high or~~ Very low pressurizer water level will scram the reactor. 1967-1995

2001

[redacted]

PLAN VIEW — REACTOR CONTROL ROOM

Figure 9.1. Control Room Layout

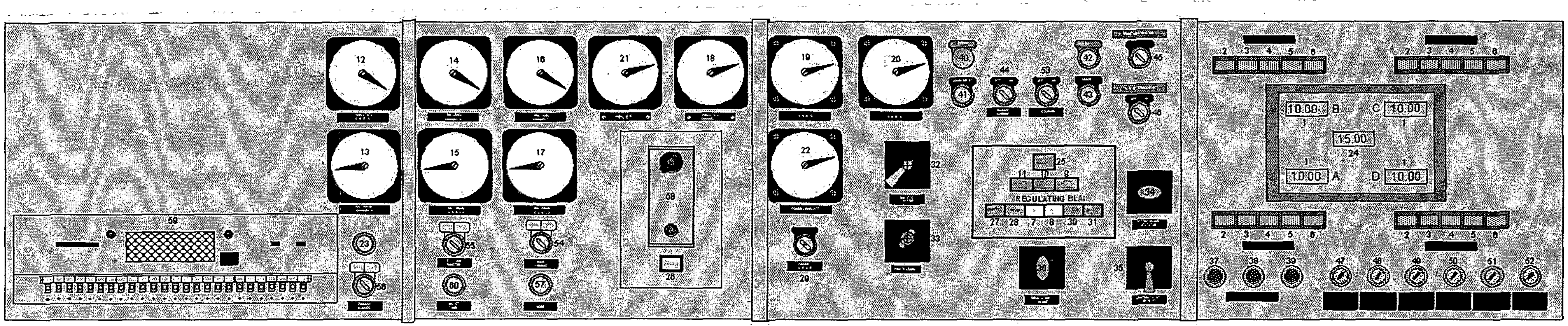
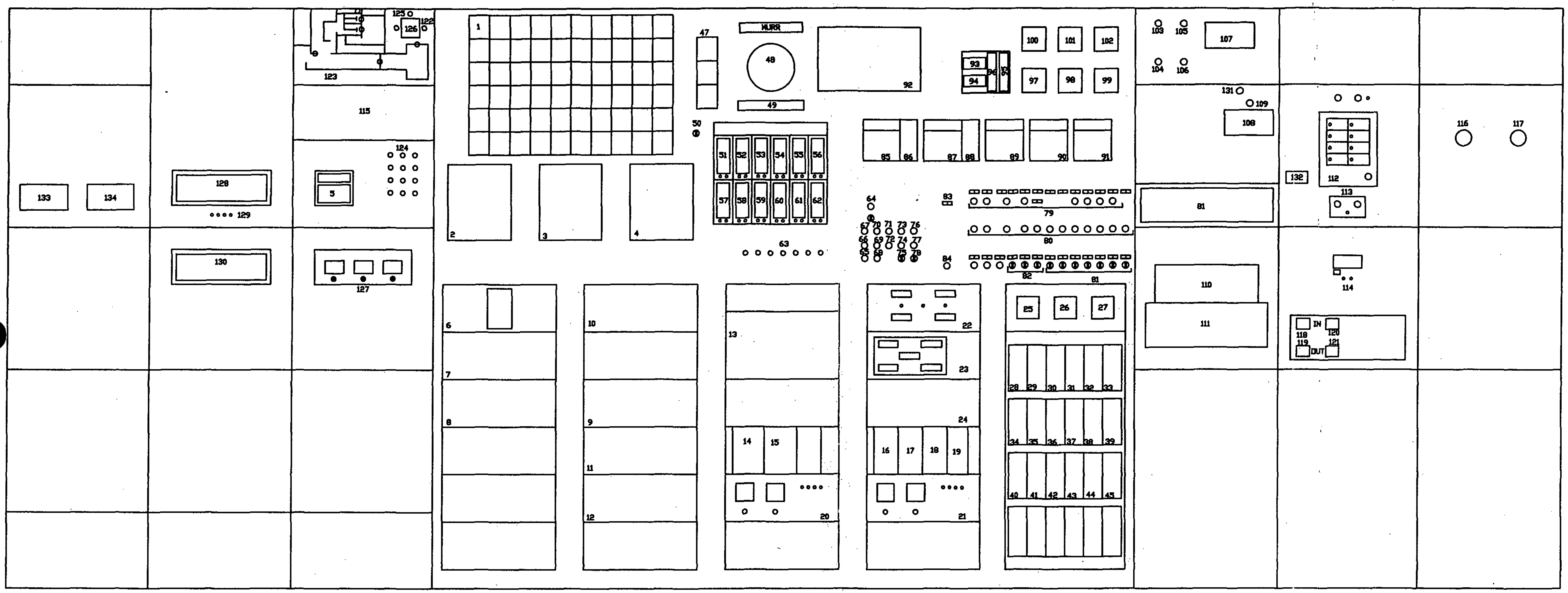


Figure 9.2 Control Console Layout
7/10/08



REV	DESCRIPTION	DATE	BY
9	Removed # 46 and reconfigured panel	5/4/06	DN
8	added 133 & 134, removed 81 switches, added 81 panel	12/27/05	DN
7	UPDATED 115	1/26/05	CHJ
6	ADDED 131, 132	2/4/02	JCA
5	UPDATED 79,80,108,115	8/27/01	JCA
4	UPDATED AND REDRAWN	10/19/00	JCA

seal
2006

DATE: 10/19/00
DRAWN BY: j.c.a.
CHECKED BY:
ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

FIGURE 9.3

UNIVERSITY OF MISSOURI–COLUMBIA
FACILITIES OPERATIONS
research reactor facility

INSTRUMENT CABINET

MURR NUMBER: 74

SHEET 12 of 12

SCRAM RESET

[redacted]

1S20, 21 & 22

CONTACT	OFF	BYPASS
1		X
2	X	
3		X
4	X	

CR2940

(KEY LOCK WITH KEY REMOVABLE IN OFF POSITION ONLY)

REFERENCE DRAWINGS

1. NEUTRON MONITORING SYSTEM ELEM. DIAG. – DWG. 40
2. REACTOR CONTROL ELEM. DIAG. – DWG. 42
3. ANNUNCIATOR CONTROL ELEM. DIAG. – DWG. 138
4. PROCESS INST. INTERLOCK ELEM. DIAG. – DWG. 41
5. RELAY FUNCTIONS – DWG. 229

BEFORE THE ROD RUN-IN SYSTEM CAN BE RESET.

REV. NO.	DESCRIPTION	DR. BY	DATE
26	MOD. PKG 04-1 ADDENDUM 1, 942 LFE TO BEEDE	THS	04/09/07
25	MOD. PKG 05-8 ADDENDUM 1	THS	12/26/06
24	MISCELLANEOUS CORRECTIONS	BJN	10/13/03
23	980 A/B MOD 01-3 ADDENDUM 1	JJL	10/10/02
22	MOD PKG. 01-3	JCA	4/5/01
21	REPLACED 928A, 928B, AS PER MOD 97-1	JCA	12/19/00
20	ADDED JUMPERS TO GRN & YELLOW LEGS AS PER MOD-75-1 ADDI	JCA	11/23/00

seal

2007

DATE: 3/3/98

DRAWN BY: J.RILEY

CHECKED BY:

ENGINEER: LPF

CODE:

REVISION NUMBER: 26

REVISION DATE: 04/09/07

FIGURE 9.4

UNIVERSITY OF MISSOURI-COLUMBIA

FACILITY OPERATIONS

research reactor facility

SAFETY SYSTEM

MURR NUMBER: 139

SHEET 1 OF 1

[redacted]

seal

2001

DATE: 3/16/98

DRAWN BY: J.RILEY

CHECKED BY: ENGINEER:

CODE:

REVISION NUMBER: 13

REVISION DATE: 3/16/98

FIGURE 9.5

UNIVERSITY OF MISSOURI–COLUMBIA

FACILITY OPERATIONS

research reactor facility

ROD RUN-IN SYSTEM

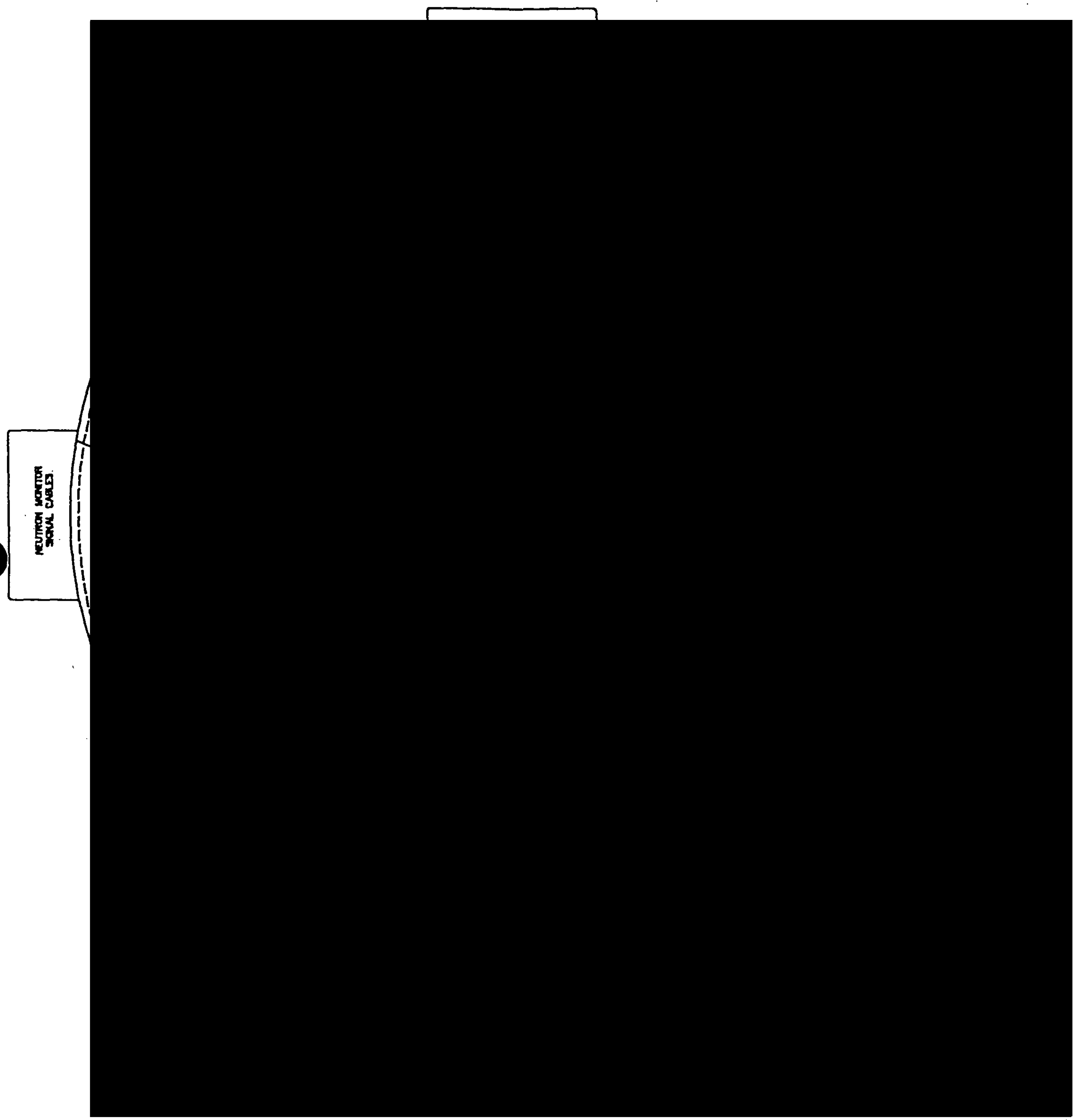
MURR NUMBER: 140

SHEET 1 OF 1

REV. NO.	DESCRIPTION	DR. BY	DATE
13	REDRAWN ON CAD	JDR	3/16/98

1996 1996

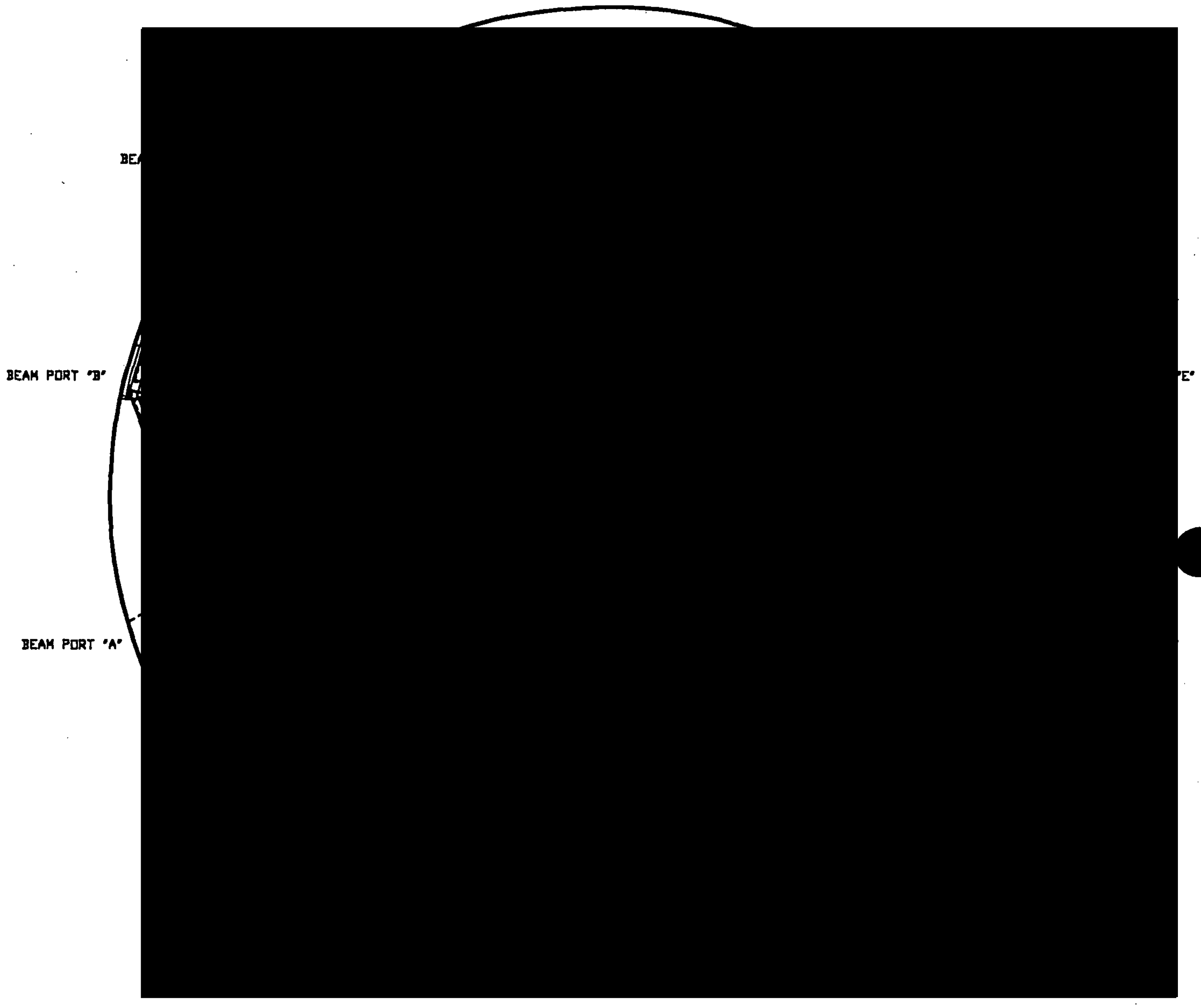
Figure 9.6



[redacted]

PLAN VIEW – OPERATING BRIDGE
(Rev. 2/7/97)

1995



[redacted]

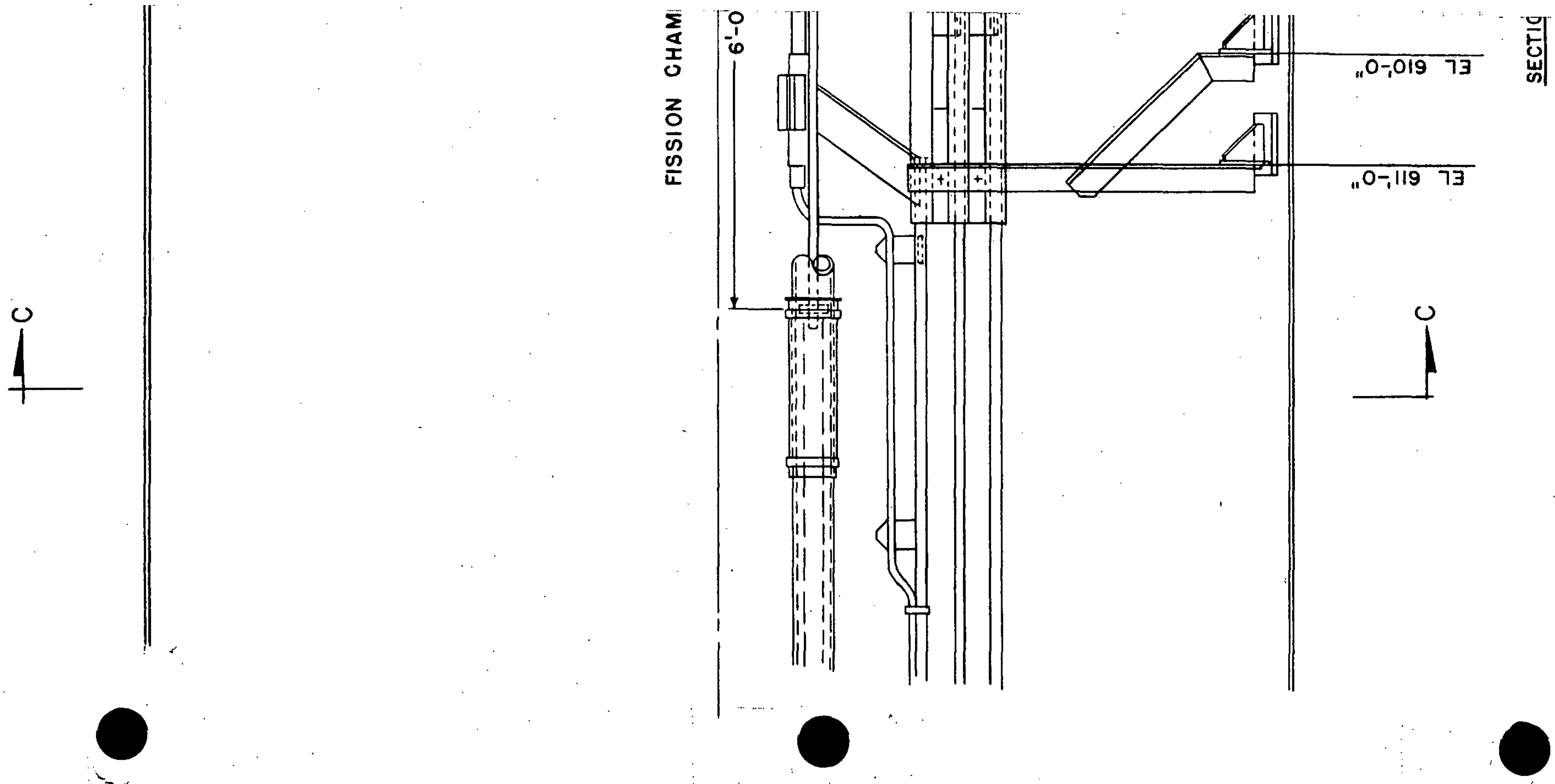
Figure 9.7 Nuclear Instrument Detector Drywell Locations
(Rev. 1/3/95)

FIGURE 9.8 IS DELETED AND FOLLOWING NOTE IS ATTACHED TO THIS PAGE

1995

<u>Delete</u> Figure 9.8 and <u>Add</u> the following note:

> **"The relative plan view location of the Nuclear Instrument Drywells is depicted by Figure 9.7. Each detector elevation is typically centered on core centerline (± 6") and are adjusted as necessary to minimize rod shadowing effects and occasional flooding of beamtubes for maintenance."**



1994

10.0 MAINTENANCE

~~Maintenance on the reactor and systems will normally be performed by personnel permanently assigned to the Research Reactor Facility for this purpose. The~~

"Maintenance on the reactor and process systems will normally be performed by personnel permanently assigned to the Research Reactor Facility for this purpose. There will be at least one mechanical and one electrical technician permanently assigned at the facility. These people will receive training on the reactor and process systems similar to that of a reactor operator. They will be thoroughly indoctrinated with appropriate radiation health and safety practices."

~~and systems equivalent to that of a reactor operator and may become licensed reactor operators. They will be thoroughly indoctrinated with appropriate radiation health and safety practices.~~ 1994

All maintenance activities on the reactor and process systems will be directed by a licensed reactor operator.

Following maintenance on the reactor and process equipment this equipment will be thoroughly checked for safe and proper performance prior to continued reactor operation.

10.1 Facilities

~~The facilities provided at the Research Reactor~~

"The facilities provided at the Research Reactor Facility for performing maintenance tasks include:

(1) One 15 ton overhead rectilinear crane and one 500 lb. jib crane in the reactor containment building.
(2) A mechanical maintenance shop within the site boundary.
(3) An electrical and instrument shop within the site boundary.

The two shops (items 2 and 3 above) are equipped with the machine tools, small tools, electrical and instrument equipment necessary to perform the normal maintenance tasks required by the Research Reactor Facility." 1994

~~containment~~

1994

~~(4) A large decontamination area on the below grade level.~~ DELETE

~~(5) A mechanical maintenance shop on grade level of the laboratory.~~

~~(6) An electrical and instrument maintenance shop on grade level of the laboratory.~~

DELETE

~~The two shops (items 5 and 6 above) are equipped with the machine tools, small tools, electrical and instrument equipment necessary to perform the normal maintenance tasks required by the laboratory and the reactor.~~ 1994

10.2 Reactor Servicing and Maintenance in Contaminated Areas

~~Subsequent to reactor operation and prior to entry into contaminable areas a complete radiation survey~~

"Subsequent to reactor operation and prior to entry into potentially contaminated areas, a complete radiation survey will be made to establish dose rates. Also, following any change of conditions in which the dose rates may be effected, such as a change in pool water level, a survey will be conducted to ensure dose rates are within acceptable limits."

~~a survey will be made to assure acceptable exposure rates.~~ 1994

Routine health physics monitoring will assure that the area surrounding the reactor pool is kept free of smearable contamination. During reactor maintenance periods special precautions will be utilized to prevent the spread of contamination. These precautions include the use of floor coverings, the use of isolation ropes, protective clothing, and continuous air sampling.

1994

Maintenance tasks within the pool include:

(1) The rod drives
(2) The control rods
(3) In-pool valves
(4) Refueling
(5) Neutron detectors
(6) Pressure vessel closure
(7) Experimental Facilities

All maintenance activities on in-pool equipment will be performed with the reactor in a shutdown condition. To prevent accidental reactivity transients, any maintenance on the control rods requiring rod movement will be done with the reactor defueled. To protect against reactivity transients during rod drive maintenance a control interlock is provided which prevents drive movement if the magnet current is on. In addition, during initial movement of the drive the technician will visually observe that the control rod is disconnected from the drive. Checkout of the entire drive assembly will not be done unless the reactor is defueled or instrumentation as required for reactor startup is operational.

~~To prevent endangering personnel during maintenance activities on electrical equipment a tag out proce-~~

"To prevent endangering personnel during maintenance activities, a tag out system will be utilized. (Refer to Section 10.7 for tag procedure).

~~electrical breakers will be de-energized and tagged (Refer to Section 10.7 for tag procedure).~~ 1994

1995
1994

~~Maintenance tasks other than routine (i.e. removal of graphite reflector, etc.) will be performed in~~

Non-routine maintenance tasks (i.e., removal of graphite reflector, etc.) will be performed in accordance with special written procedures. These special procedures will establish the condition of the reactor, include special tools required, describe the monitoring and personnel protection requirements, and include a step by step guide for completion of the maintenance activity. These special procedures will be reviewed and approved by the Reactor Manager."

~~procedures will be reviewed and approved by the reactor supervisor.~~ 1994

10.3 Reactor Process Equipment Maintenance

The areas in which a high radiation level may exist are:

(1) Main equipment room.
(2) Reactor pipe tunnel.
(3) Demineralizer area.

Figure 10.1 shows the equipment layout in the equipment room and pipe tunnel. Access to this area is not permitted during reactor operation due to high radiation levels expected from N^{16}. It is not expected that radiation levels following reactor shutdown and a short decay period will be high enough in these areas to prevent maintenance activities. Prior to entry, the area will be surveyed and dose rates established. ~~Poi~~ **"Point of control will be at the entries** ~~the ent~~ **designated by Health Physics."** ~~ire 3.2.~~ Suitable containers will be provided at the control points for discharge of potentially contaminated materials worn or used in the area. Maintenance work in the area will be directed by a licensed reactor operator and ~~a tag~~ **a tag out procedure will be followed for affected systems and equipment."** 1994 1995

This area is serviced by the ventilation system and air-borne activity is not anticipated under normal conditions. ~~In the event that the maintenance activity may create air-borne activity continuous~~

"In the event that the maintenance operation may create air-borne activity, continuous air monitoring capability, as necessary, will be prescribed by Health Physics." 1994

~~will be provided.~~

Waterproof protective clothing will be provided to personnel when it becomes necessary to perform work on systems which may involve contact with water from the process piping or equipment.

When not occupied, access doors to this area will be locked and key controlled with the reactor supervisor.

Long-lived radioactivity may limit entry into the demineralizer rooms. ~~Regeneration of the demineralizers is accomplished as described in Section 7.1.9 in areas shielded from the demineralizers. At intervals of approximately 18 to 24 months the resins will~~

"Regeneration or replacement of the demineralizer resin is accomplished as described in Section 7.1.9 in areas shielded from the demineralizers. At intervals of approximately two to four months the resins will require changing. They will be discharged into containers and discarded as radioactive waste. Monitoring, as prescribed by Health Physics, will be provided to personnel during the resin transfer procedure. Personnel will be provided proper protective clothing as necessary." 1994

~~health physics monitoring will be provided to personnel during the resin transfer procedure. Personnel will be provided proper protective clothing when necessary.~~

10.4 Laboratory and Experimental Maintenance

The research facility ventilation system and laboratories represent the balance of the facility in which radioactivity will normally be encountered.

1994

Replacing the filters in the ventilation system will require monitoring for radioactivity and establishing dose rates. ~~Protective clothing and respiratory protection will be provided when handling~~ **"Protective clothing will be prescribed by Health Physics when handling contaminated filters. Immediately following removal of a filter from the system it will be placed in a plastic bag or other suitable air tight container and sealed."** ~~placed in a plastic bag or other suitable air tight container and sealed.~~ Disposal of the filters will be as radioactive material. 1994

Maintenance in the laboratories and on experimental equipment will be as the need requires. Thorough monitoring and protective measures will be taken prior to and during maintenance activities.

10.5 Routine Systems Maintenance

~~A~~ **"A periodic maintenance system will be in effect on all reactor and process equipment."** ~~a [illegible].~~ The purpose of this system will be to provide a preventive maintenance program assuring that all equipment is routinely checked and properly cared for. ~~A card f~~ **"A record will be maintained providing the date and nature of maintenance activities on all equipment."** ~~ure of maintenance activities on all equipment.~~ It will indicate when the next inspection on any piece of equipment is due and the type of periodic maintenance to be performed. 1994 1994

Equipment not previously covered in this section such as the emergency diesel generator, secondary system pumps, fans and blowers, pneumatic air system equipment, air compressors, vacuum pumps, and refrigeration equipment is not expected to require special maintenance considerations and their servicing will be part of the

1994
2002

periodic maintenance system to assure reliable operation at all times. The emergency generator and air compressor will be operated at least once per week for not less than 30 minutes to assure operability.

~~Acid resistant protective clothing and face shield~~

"Acid resistant protective clothing and face shield will be worn by personnel performing work on the acid system. An eye wash and shower are provided in the cooling tower building for use in the event of an emergency."

~~in the event of an emergency.~~ 1994

A tag out procedure will be applicable to all maintenance activities.

10.6 Instrument and Electrical Maintenance

Servicing of instrument and electric equipment will be performed only by technically competent personnel. Maintenance of components such as ion chambers, transmitters and controllers, located in areas described in Section 10.2, 10.3, and 10.4 will require that appropriate protective measures be taken by personnel as prescribed for radiation areas.

Instrument and electrical components will be part of the periodic maintenance program.

"A source check of the radiation monitoring modules which are connected to the Safety System will be performed at least ~~a~~ monthly ~~week~~ to ensure proper operation. A source check of all the radiation monitoring modules will be performed quarterly to ensure proper operation. The calibration of instruments will be included in the maintenance program." 1994 2002

1994
2002

10.7 Tag Out Procedure

To prevent inadvertant damage to equipment or injury to personnel, a tag out system will be utilized.

~~"(1) A danger tag will be placed on a piece of equipment to signify that operation of the equipment presents a hazard to personnel and/or potential damage to the equipment. The tag will be numbered and contain information as to the date it was initiated, the piece of equipment involved, and the reason for the tag."~~ ~~ch or~~ ~~ies on~~ ~~h damage~~

~~to equipment may occur if the switch is activated. The tag will be numbered and contain information as to the date it was initiated, the piece of equipment involved, and the potential hazard that exists.~~ DELETED IN ITS ENTIRETY 1994 2002

~~(2) A red tag will be placed on the switch or controller of a piece of equipment to signify that operation of the equipment presents a hazard to personnel. The tag will be numbered and contain information as to the date it was initiated, the piece of equipment involved and the hazard that exists.~~ 1994

(2) A log book will be maintained containing the 2002

"A log book will be maintained containing the tag number, the date initiated, the equipment involved, the date of the tag removal, and the initials of the person responsible for tag placement and removal." 1994

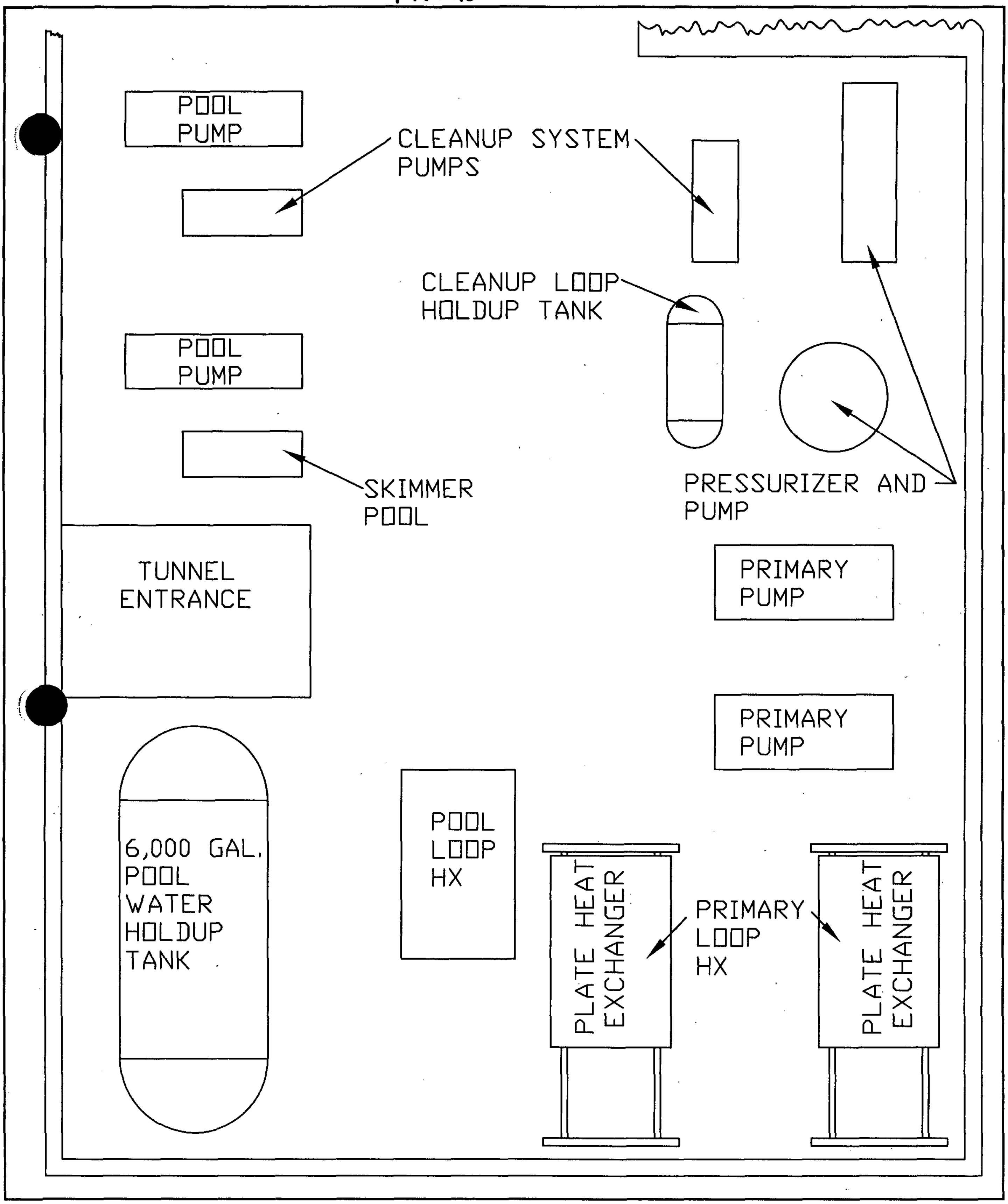
initials.

(3) The removal of any tag will be with supervisory approval only, and only after it is assured that maintenance activities are completed and personnel are removed from the area. The tag will be destroyed after use. 2002

(4) In the case of key lock switches or electrical breakers which can be locked out, the equipment will be placed in the locked out condition. 2002

FIGURE 10.1

2008



MAIN REACTOR EQUIPMENT ROOM

FACILITY OPERATIONS
research reactor facility
UNIVERSITY OF MISSOURI–COLUMBIA

date
drawn by J RILEY
checked by: engineer
code
revision 1
rev.date 8/8/08

murr number 2463

sheet 1 of 1

1973-
1974
1974-
1975

11.0 INITIAL TESTS AND OPERATING PROCEDURES

To assure the ability of the reactor and systems to operate safely and perform within specifications, certain tests and procedures will be followed during initial testing and routine operation.

There are six major steps in bringing the reactor to routine 10 MW operation. They are:

(1) Preoperational systems tests.
(2) Initial reactor fuel loading.
(3) Zero and low power tests.
(4) Tests at increasing power.
(5) 5 MW power operation.
(6) Tests to increase power to 10 MW as outlined in the 10 MW Preoperational Test Procedures document as approved by the Procedures Review Subcommittee. 1973-1974

In addition there are these formalized routines:

(1) Operating procedures.
(2) Routine operational tests and checks.
(3) Corrective action and emergency procedures.
(4) Tabulations of reports and records.
(5) Security Plan and Security Procedures 1973-1974

(6) SNM Control and Accounting Procedures

1974-
1975

11.1 Preoperational System Tests

These tests are designed to prove that the systems perform the functions for which they are intended and that they meet established safety and reliability criteria. Data will be accumulated from the tests on system variations to establish span of control and to determine maximum and minimum system capabilities.

The preoperational tests will be performed by the reactor contractor as part of the reactor acceptance program. The tests include, where applicable, hydrostatic pressure checks, checks of flow, reliability, auto control operation, instrumentation, and efficiency.

The method and content of the test to be performed on each system will be contained in a testing manual mutually prepared by the reactor contractor and University of Missouri personnel. All components and systems will be subjected to functional testing.

11.2 Initial Reactor Fuel Loading

Prior to loading fuel into the reactor core, subcritical multiplication will be determined on the fuel storage rack using the reactor fuel. The rack configuration is such that it is well below critical. However, multiplication studies will be run to assure this fact, to provide the operating crew with experience in working as a team and for familiarization with the instrumentation to be used for loading the core.

Several hazards associated with the initial loading of the core are recognized.

(1) Dependability of calculations of core reactivity and control. Although accurate to best ability using established techniques, the loading of the core must provide for possible inaccuracies in calculations.

(2) Possibilities of error in fuel element loading.
(3) Inadvertant rapid insertion of reactivity such as by dropping a fuel assembly or rapidly flooding the core.
(4) Instrument malfunction.
(5) Misinterpretation of data.
(6) Communications difficulties.
(7) Possible radioactive exposure to personnel.

Recognizing these hazards, adequate protective measures are to be provided in the following manner. The core will be loaded in a subcritical configuration without moderator. While loading, the procedure will follow precautions similar to those followed if the moderator were present. The loading crew will be thoroughly indoctrinated to the loading procedure and dry runs of the loading operation will have been performed. Two extra source level channels will be added to the instrumentation to provide additional data during loading. Instrumentation will be thoroughly checked out and in proper operating condition before loading the core. The utmost care and caution will be exercised during the loading of the fuel elements and subsequent addition of moderator.

The core will not be allowed to reach critical during the fuel loading. A detailed procedure will be followed during the loading process. All members of the loading team will be thoroughly briefed on the procedures and their particular assignments. During the time the core is being loaded, occupancy of the reactor enclosure will be limited to those persons on the loading team. There will be no "defeats" on the reactor safety system or the building isolation system. The core will be loaded with no circulation in the process systems. The normal complement of reactor instrumentation will be in operation during loading and in addition two BF_3 counter chambers will be installed to monitor neutron population. This provides a fission chamber, and two BF_3 chambers to record the source level neutron population and multiplication while installing the fuel.

As fuel is added to the core the critical loading will be predicted by plotting the reciprocal of the neutron population, as indicated by the three pulse counters, vs. wt. of U-235 as it is added to the core and the curve extrapolated to zero, at least two separate plots will be maintained.

The major procedures involved in loading the core are:

(1) Fill the pool and center test hole with water to a level approximately three feet below the top of the reactor pressure vessel.

(2) The water level in the core will be below the bottom of the fueled regions.

(3) Install the source and BF_3 counting chamber in the center test hole.

(4) Check out instrumentation, control, and safety systems.

(5) Position all counting chambers to indicate not less than 6 counts per second of source neutron population.

(6) Load four fuel elements in dry pressure vessel, one at a time, in alternate core positions. The control rods are to be full out. The plot of reciprocal multiplication maintained at the loading of each element and data analyzed before the next element is loaded. The effect of control rods on neutron population is to be determined after four elements have been loaded.

(7) After the fourth element is loaded the core will be flooded in small increments. Reciprocal multiplication will be plotted vs. water height at each incremental water level.

(8) Each succeeding fuel element will be loaded with the core dry then the core flooded, with multiplication data plotted, after the loading of each element. The effect of control rods will be determined at each loading and they will be positioned to provide shutdown margin and yet keep the reactor below critical preceding each successive loading.

After the core fueling is completed, the water level in the pool and core will be raised to the refueling level. Then, after substantiating that the data verifies the calculated characteristics of the core, the reactor will be taken to critical condition. This completes the core loading and the reactor will then be shut down.

The flooding will be done manually by an operator at the location of loading operations. The fill line will be sized to prevent rapid flooding of the core. The fuel grapple will have a positive lock on the fuel element. Personnel are under advisement of a qualified health physicist at all times and no operators will be at the refueling level of the pool during flooding of the core.

Core loading operations will be discontinued if there are not at least two source level counting channels in proper operation, when conflicting data is obtained, and when a condition arises which, in the opinion of the loading supervisor, presents a hazard to safe continuance of fuel loading.

11.3 Zero and Low Power Tests

These tests are to be made to verify or establish the physics characteristics of the reactor. The tests to be performed include the following:

(1) Power Level Determination. Determine power level by foil activation methods.

(2) Control Rod Calibration. Determine reactivity worth of the regulating rod by positive period method. Calibrate shim rods against regulating blade.

(3) Map Flux Distribution. Part of this test will be accomplished in determining flux distribution in core for low power determination. These tests will be expanded to determine flux in other critical areas.

(4) Determination of Reactor Temperature and Void Coefficients. These coefficients will be determined in reflector, core, and central test position regions independently.

(5) Determine Reactivity Effects Associated With Flooding Beamports and Pneumatic Tube Operation.

(6) Operational Test of the Automatic Control System.

(7) Operational Characteristics of Reactor With Cooling System on Line.

These tests, with the exception of (7) will be performed without forced convection cooling. The pressure vessel will be open for instrument access when necessary. The nuclear safety trip settings will be adjusted to 25 per cent above the value corresponding to just critical until the power calibration has been determined. The trip settings will then be adjusted to 125 per cent of the 100 KW power reading on the corresponding instrument. The period safety trip setting will be maintained at 50 seconds.

Power during these tests will not exceed 100 KW. The 100 KW master control prohibit circuit will be in effect during the tests.

11.4 Tests at Increasing Power

The purpose of these tests is to evaluate reactor performance in incremental steps to 5 MW and to predict the ability of the reactor to operate as designed at higher power level.

Power increase to 5 MW will be made in five one megawatt steps plus an intermediate step at 500 KW. Additional intermediate steps may be made if necessary. The checks to be made at each power step include:

(1) Radiation surveys
(2) Reactor control and stability tests
(3) Air and water activity analysis

Power at 1 MW and above will be determined by heat balance on the reactor cooling systems and at one intermediate step will be compared with power determined by foil activation.

The power level at which the scheduled test is to be run will be considered full power and nuclear safety trips set accordingly.

Subsequent to operation at design temperature, but before accumulation of excessive quantities of fission products, at least one fuel assembly will be removed and given a detailed visual inspection to determine general condition.

11.5 Five Megawatt Power Operation

During the interval of time after reaching 5 MW and before increasing power to 10 MW considerable data will be accumulated at 5 MW operation. This data will be used to predict conditions to be expected at 10 MW. In addition to the specific tests called for in Section 11.4 (Tests at Increasing Power) additional tests will be performed, mostly relative to experimental facilities. Included will be a determination, by foil activation, of absolute flux available in the experimental positions.

1995

11.6 Tests to Increase Power to 10 Megawatts

The ability of the reactor to operate at 10 MW is dependent upon additional cooling equipment being installed. The tests involved will determine the reliability, adequacy, and safety of the new equipment in a similar manner as described in Section 11.1.

After the necessary additional cooling equipment has been installed and tested, reactor power will be increased from 5 MW to 10 MW. Power will be increased in steps and tests performed as outlined in Section 11.4.

It is expected that the fuel loading of the reactor will be increased from 5.2 to 6.2 Kilograms of U235 for 10 MW operation. It will not be necessary to use the same method to load the core as was used in initial loading. The operating history of the reactor will provide sufficient information of the characteristics of the reactor and control system to enable a determination of the safety of loading the core with moderator present. A supplement to the Hazards Summary and Technical Specifications will present the proposed method for increasing the core loading.

11.7 Operating Procedures

Written Standard Operating Procedures will give detailed information pertaining to the operation of the reactor and systems. These procedures are designed to assure operation within the limits of the Technical Specifications and will be approved by the ~~rea~~ Reactor Manager ~~sor~~ and reviewed by a committee designated by the facility director. Included in the procedures are requirements for reactor startup, normal operation, shutdown, system operation, experimental operations and emergency procedures. 1995

1995
1981-
1982

~~Appendix A lists general requirements for reactor operation to be included in the standard operating procedures.~~ DELETE 1981-1982

The master copy of the standard operating procedures will be maintained by the Reactor Manager with a duplicate copy in the reactor control room. Additions and revisions to these procedures will be by authority of the Reactor Manager following review as above. 1995 1995

11.8 Routine Operational Checks and Tests

The periodic maintenance program (Section 10.0) will schedule routine checks and calibrations on equipment and instrumentation. These schedules will be based on manufacturers recommendations and experience.

11.8.1 Startup Routines

Startup of the reactor will be preceded by the completion of a startup check sheet listing items of the process, safety, instrument, and radiation monitoring systems which require verification of proper operation before reactor operation. General information on the startup check sheet will be checked within one hour prior to startup. This check sheet will also contain a section relative to scram checks which is to be completed immediately before beginning the approach to critical.

11.8.2 Shutdown Routines

Before the reactor is left unattended at the end of the operating period a shutdown check sheet will be completed to assure that the reactor and systems are in the proper status.

1996
1995

11.8.3 Patrol Routines

"Daily patrols (Monday - Friday) are established for a mechanical technician and a member of the health physics group. Reactor Operations performs routine patrols every day the reactor operates, including weekends and holidays." ~~operator and the health physicist.~~ These patrols are designed to check the operation of equipment not under direct observation at all times. Items to be monitored by these patrols include the following: 1996

- Cooling tower equipment
- Waste tank levels
- Ventilation system equipment
- Air compressors
- Air conditioning equipment
- Water analysis
- Nitrogen header
- Radiation detection system
- Fission product monitor
- Experimental equipment

11.9 Corrective Action and Emergency Procedures

The reactor will be shut down either automatically or manually in the event maximum permissible operating conditions are exceeded without corrective action.

The occurrence of an alarm condition need not require reactor shutdown. In many instances the alarm condition may be corrected without affecting reactor operation. If an alarm condition exists which cannot be corrected without shutdown of the reactor, the condition will be evaluated. Continued operation will be dependent upon this evaluation. If operation is continued, the condition will be corrected following scheduled reactor shutdown. The ~~re~~Reactor Manager ~~visor~~ or his designated representative must authorize the continued operation. 1995

1996
1995
1985-
1986
1981-
1982
1967-
1968

The standard operating procedures will include detailed emergency procedures for specific conditions. In general these procedures will consist of steps required to prevent injury to personnel and to place a system or piece of equipment in such a state as to minimize hazards and prevent further adverse effects.

~~General Emergency procedures are included in Appendix B.~~ 1981-1982

11.10 Records

Strip chart recorders provide a permanent record of operating information. The parameters recorded are:

- Log count rate.
- Log power (one of two channels to be selected).
- ~~Linear power (one of three channels to be selected).~~ "Linear power" 1996
- ~~Area radiation.~~ Off Gas Activity 1985-1986
- Reactor water inlet and outlet temperature.
- Pool water inlet and outlet temperature.
- Reactor coolant flow rate.
- Demineralizer flow rate.
- Secondary coolant flow rate.
- Secondary coolant temperatures.
- "Pool coolant flow rate" 1995

The charts will be put in permanent storage to serve as a reference to past operating experience.

Certain data will be taken manually for a record of reactor performance information not recorded by instrumentation. A nuclear and a process data sheet will be used to record this information.

1995
1981-
1982

Nuclear data ~~will be recorded as specified in Appendix A and~~ DELETE consists of the following items: 1981-1982

Time

Power level

"Nuclear channel 1"

Nuclear channel 2 1995

Nuclear channel 3

Nuclear channel 4

Nuclear channel 5

Nuclear channel 6

Reg. rod position

~~Shim rod A position~~

~~Shim rod B position~~

"Average Shim Rod Position"

~~Shim rod C position~~ 1995

~~Shim rod D position~~

Total accumulated MWD (to be calculated)

Calculated power (from cooling system)

~~Mode of operation (Auto-Manual)~~ DELETE 1995

~~Operator initials~~ DELETE

Remarks

Process data, ~~to be recorded as provided in Appendix A,~~ DELETE consists of: 1981-1982

Time

All flow indicator readings

All temperature indicator readings

Tank levels (T-300 and T-301)

Area radiation levels

All pressure indicator readings

All differential pressure readings

All differential temperature readings

Water conductivities

1995

Fuel rupture monitor indication
Secondary coolant activity
Operator initials
"Pressurizer level"
"Off-gas stack activity"

1995

A console operator log will be maintained in the control room providing a detailed diary of reactor operation. This log, as well as the routine check sheets, will become a permanent record of reactor operation.

Scram, incident, accident and equipment malfunction reports will also be maintained on permanent record.

1996

12.0 ORGANIZATION AND ADMINISTRATION

12.1 General Description

~~The reactor and laboratory facilities will be available to any faculty member or graduate student interested in pursuing research involving radiation, radioisotopes, or the reactor. The diverse research programs initiated will be coordinated and health and safety supervision provided by the permanent staff employed to operate the facilities. The research staff, composed of faculty members and graduate students, will be semi-transient, since no permanent assignment of space or facilities will be made. Administration of the reactor and laboratory~~

Replace section with (changes bolded):

"The reactor and laboratory facilities **are** available to faculty **members** or graduate **students** interested in pursuing research involving radiation, radioisotopes, or the reactor. The diverse research programs **are** coordinated and health and safety supervision provided by the permanent staff employed to operate the facilities. **Part of the** research staff, composed of faculty members and graduate students, will be semi-transient, **with** no permanent assignment of space or facilities. Administration of the reactor **is separate** from the research programs to eliminate the possibility of a compromise in safety for the sake of experimental expediency.

The Research Reactor **operations** staff is divided into two groups. One group, the reactor **operations** group, has responsibility for the routine operation and maintenance of the reactor. The second, the **facility** operations group, has responsibility for the supervision and maintenance of the laboratory facilities associated with the reactor. Coordination and overall administration of these groups is performed by the **Facility Director**. A current table of organization for the Research Reactor Facility based upon continuous operation 7 days a week is presented in **Figure 6.0, Technical Specifications. The Reactor Operations** staff requirements for operation of these facilities **is expected to** be twelve to sixteen people. These people provide operation and supervision of the facilities for all research personnel.

1996*

~~facilities associated with the reactor. Coordination and overall administration of these groups is performed by the director of the project. A table of organization for the reactor facility is presented as Figure 12.1. It is estimated that the staff requirements for operation of these facilities on an eight-hour day, five-day week, will be twelve to sixteen people. These people will provide operation and supervision of the facilities for all research personnel, expected to number from thirty to sixty people.~~

* DELETES A 1982 REV.

1996
1981-1982
1997
2002

"The Research Reactor Facility fits within the organizational structure of the University of Missouri as part of the Columbia campus." 1996

~~as a separate entity.~~ The various operating and service groups of the University can be divided into three major categories. One of the major categories is the academic departments, each of which is supervised by a dean. A second group of operating subdivisions within the University are classified as academic service organizations. ~~This group includes the Univer... ision station, University Press, etc., and includes the Research Reactor Facilities.~~ "This group includes the Research Reactor Facility." 1997 Each of these departments has an assigned director who is responsible for the administration of his particular division. The third category is termed non-academic services and includes such things as physical plant, budget, construction, auditing, etc.

The Director of the Research Reactor Facilities reports to the "Office of the Provost" ~~ident Academic Affairs on~~. ~~The~~ 1981-1982 2002 ~~De~~ "The Office of the Provost reports to the Chancellor, UM-Columbia." ~~the President of the University.~~ 2002 ~~This chain of command is s~~ "This chain of command is maintained in Figure 6.0, Technical Specifications." 1997

12.2 Organization

12.2.1 The Director

The Director of the Research Reactor Facilities reports to the ~~Associate~~ "Office of the Provost" ~~mic Affairs on~~. 1981-1982 2002 The Director has overall responsibility for the reactor, and all associated laboratories. He will supervise, through his assistants, the operation, as well as the utilization and maintenance of these facilities. He will prepare and administer the budget, he will

1996
1981-1982
1967-1968
1977
1997

negotiate with, and employ the necessary staff for operation of these facilities. He will be available to faculty members for consultation on research proposals for the utilization of the reactor or associated laboratory facilities.

12.2.2 The Reactor Advisory Committee

~~The Reactor Advisory Committee has functioned during~~

The Reactor Advisory Committee shall review:

"(1) Proposed changes to reactor equipment or procedures when such changes have ~~anges have~~ safety significance, or involve an amendment to the operating license, a change in the Technical Specifications incorporated in the license, or an unreviewed safety question pursuant to 10 CFR 50.59. Changes to procedures that do not change their original intent may be made by the Reactor Manager or a designated alternate who is a senior licensed operator. All such changes to the procedures shall be documented and subsequently reviewed by the Reactor Advisory Committee. 1997

1996

(2) Proposed tests or experiments significantly different from any previously reviewed or which involve an unreviewed safety question pursuant to 10 CFR 50.59. 1997

(3) The circumstances of all abnormal occurrences and violations of Technical Specifications and the measures taken to prevent a recurrence.

The Reactor Advisory Committee may appoint subcommittees consisting of students, faculty and staff of the University when it deems necessary in order to effectively discharge its primary responsibilities. ~~When subcommittees are appointed they are to consist of no less than three members with no more than one student appointed to each committee.~~ DELETE The subcommittees may be authorized to act in behalf of the parent committee. ~~The Reactor Advisory Committee and its subcommittees are to maintain minutes of meetings in which the items considered and the committees recommendations are recorded.~~ Independent actions of the subcommittees are to be reviewed by the parent committee at the next regular meeting. A quorum of the committee or the subcommittees consisting of at least fifty percent of the appointed members must be present at any meeting to conduct the business of the committee or subcommittee. 1997 1977

INSERT "The Reactor Advisory Committee and its subcommittees are to maintain minutes of meetings in which the items considered, actions taken, and the recommendations made are recorded." 1997

The Reactor Advisory Committee shall meet at least once during each calendar quarter.

A meeting of a subcommittee shall not be deemed to satisfy the requirement of the parent committee meeting quarterly.

Members are selected for appointment to the Committee because of their expert knowledge of experimental activities, reactor operations, University business policy, or related subjects. Members of the Committee, its Chairman, and its Vice Chairman are appointed annually during the Fall semester to serve one year. Members may be reappointed indefinitely.

~~Research for the University.~~ 1997

12.2.3 The Reactor Manager ~~sor~~

The Reactor Manager ~~sor~~ has responsibility for operating the reactor in a safe, reliable, efficient, and economical manner, and in a manner such as to make 1996

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the fullest possible services available to experimenters. He will be responsible for instructing personnel in their duties, for issuing the operating schedule, for personally supervising non-routine reactor operation, for establishing a procedure for the accommodation of experiments, for the review and processing of applications for radiation previously approved by the advisory committee, for advising on safety aspects of proposed experiments, and for the provision of information to technical visitors regarding the reactor facility and its operation. He will also be responsible for the safety of experimenters, students in laboratory courses, and students pursuing graduate research. He will be assisted in the performance of these duties by the Associate Reactor Supervisor.

Add sentence to end of section:

"He will be assisted in the performance of these duties by the Assistant Reactor Managers and the Lead Senior Reactor Operators." ~~erations Engineer and Reactor~~ 1996 2002

2.2

~~The Laboratory Supervisor will supervise the activities being conducted in the various laboratories associated with the reactor. He will teach and supervise the use of the gamma facility, and the pneumatic tubes. He will assist experimenters with 1) the design of their experiments, 2) the selection of proper instrumentation, and 3) the evaluation of radiation hazards. He will cooperate with the Health Physicist in the decontamination, packaging, and disposal of radioactive waste. The laboratory supervisor will have a background in Health Physics, Radiochemistry or Radiobiology.~~

DELETED IN ITS ENTIRETY

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12.2.5 Reactor Health Physics Manager

~~The Reactor Health Physicist is not part of the Research Reactor Facility staff. He is responsible to the Radiological safety officer who, in turn, reports to the Dean of Research Administration. The~~

DELETE

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~~Reactor Health Physicist is permanently housed at~~

Add to beginning of second paragraph:

"The Reactor Health Physics Manager is part of the Research Reactor Facility staff." 1996

The primary aim of the Reactor Health Physics Manager is to achieve safety in work with radiation and radioactive materials with a minimum of interference in the conduct of that work. To accomplish this, the Reactor Health Physics Manager must establish operational 1996 and administrative guides consistent with presently accepted standards. He must insure that these guides are observed and that personnel adhere to them.

It is necessary that routines be established by which the Reactor Health Physics Manager can insure that 1996 radiation safety is being maintained within the facility. Typical of these routines are the following:

(1) Daily surveys of reactor-associated experimental setups;
(2) Daily surveys of the laboratories using radioactive materials;
(3) Weekly surveys of the permanently mounted reactor-associated radiation monitoring equipment;
(4) Monthly surveys of the facility, both smear and instrument;
(5) Maintenance and review of personnel exposure records;
(6) Radioactive analysis of the in-pool water and the primary loop water.

Other duties and responsibilities of the Reactor Health Physics Manager will include the following: 1996

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(1) Act as consultant to the Director and assist him in the preparation of the yearly report;
(2) Serve on the Reactor Advisory Committee;
(3) Provide radiation safety supervision for research personnel;
(4) Monitor operations involving the opening of the beamports and the thermal column, removal of material from the pool, and all new experimental setups;
(5) Periodically calibrate all survey instruments;
(6) Conduct training and orientation programs in radiation safety and applied health physics for all personnel;
(7) Maintain records pertaining to personnel exposure radiation and contamination control, radioactive waste control, on- and off-site environmental control, and off-site transfers of radioactive materials.

12.2.6 Reactor Operators

A sufficient number of reactor operators will be employed to provide for at least two to be on duty during preoperational checkout, operation and shutdown procedures. The reactor operators will be trained and examined by senior members of the reactor staff. During work periods when they are not operating the reactor these individuals will continue their training and will assist in routine maintenance and repair operations, **and sample handling**." 1996

12.2.7 Shops

There is an electronics shop and a machine shop for maintenance of the reactor and laboratory equipment,

1986
1974

as well as to meet the needs of research personnel utilizing these facilities. ~~The shops will be under~~

"The shops will be under the direct supervision of the **Facilities Operations Manager**."

He will approve expenditures and make periodic evaluations of operations. The shop personnel will construct, or supervise the construction of, complex pieces of research equipment. They will also provide routine maintenance as required for the reactor and all instrumentation in the various laboratories associated with the reactor. 1986*

12.3 Staff Training

~~The reactor operators will be hired and a teaching and training program completed at the University of Missouri prior to reactor startup. In this manner, these individuals can procure valuable experience in the final construction and installation of machinery, as well as with the electronics associated with the control of the reactor. The machinist and electronic technician will also be trained in the first group of reactor operators to permit them to procure an operator's license so that they might be better prepared to do maintenance on the reactor.~~ 1986

DELETE

~~The training program for operators will consist of a series of lectures on reactors, reactor safety,~~

Replace with (changes bolded):

"The training program for operators will consist of **guided self-study** on reactors, reactor safety, general health physics, reactor control systems, and postulated reactor malfunctions **and emergency response**. **This study** will be coupled with **on-the-job training** which will demonstrate the utilization of health-physics equipment, the operation of components of the reactor systems, and the operation of the reactor controls. This program of training will prepare these people for the **NRC** operator's license examination."

"Following licensing by the NRC all operators and senior operators will take part in an operator requalification program." 1974 1986

* DELETES A 1968 REV

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~~people for the A.E.C. operator's license examination.~~ 1996

12.4 Visitors

It is expected that the Nuclear Reactor Facility will be an attraction for social and professional organizations in the State of Missouri. It is intended that the reactor and related laboratories have been designed in such a manner that these guests can be handled. A lobby is provided which will receive 50 guests, permitting the registration and handling of these people and the storage of their coats and hats. The various laboratories have windows so that it is possible for visitors to view operations in these rooms.

~~There will be one afternoon of each week open for visitors between the hours of 1 and 5 p.m. Non-technical visitors will be guided through the~~

"Nontechnical visitors will be guided through the facility by trained **guides**. These **guides** will be equipped with the normal personnel-monitoring equipment and he will be instructed to stay with the group of **10 to 15** people assigned to **them**. They take the group on a preassigned path through the facilities." 1996

~~equipment and he will be instructed to stay with the group of 12 to 25 people assigned to him. He will take the group on a pre-assigned path through the facilities.~~ Variations will be made in this path to insure that no visitor enters an area in which there is loose contamination or a possibility of a radiation exposure. In instances where there is experimental work being carried out on the beam hole floor, those areas which represent exclusion areas to visitors will be roped off to eliminate the possibility of a visitor inadvertently entering.

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Technical visitors will be accompanied through the facilities by a member of the staff. Those technical visitors desiring to view a particular experiment or radiation facility will be equipped with film badges at the front desk in the lobby of the laboratory facility. ~~Technical visitors who have no desire to view the facilities but who wish only to speak to members of the staff or use the library facility, will not be required to have a film badge for intermittent visits. If it is their intent to work in the library or any of the facilities for an extended period of time, they must comply with the established regulations for use of the facilities.~~ DELETE 1996

12.5 Policy with Regard to Reactor Utilization

~~The research reactor facilities shall be available for research utilization by any member of the faculties of the University of Missouri and of the Universities comprising the Mid-America Association of State Universities. Priorities for the use by~~

Replace with:

"The research reactor facilities shall be available for research utilization by **members** of the **faculty** of the University of Missouri and **other** universities. Priorities for the use of any specialized facilities on the reactor shall be established by the **Facility Director.** In the event that questions arise as to the advisability of a priority assignment, these questions will be **reviewed by** the Reactor Advisory Committee, and their **recommendations made to the Facility Director**."

~~priority assignment, these questions will be negotiated with the Reactor Advisory Committee, and their findings will be final.~~ 1996

All reactor utilization by faculty members will be subject to the supervision of the ~~R~~ Reactor Manager ~~or,~~ and the Reactor Health Physics Manager Their decisions and recommendations having to do with radiation safety or safety of reactor operation may be negated 1996

1996

only by appeal to the Reactor Advisory Committee. With respect to faculty, graduate students, other educational institutions and industrial contract research, the priority for use shall be as follows:

(1) Faculty and graduate students of the University of Missouri and of other universities in the Mid-America Association of State Universities.
(2) Faculty and graduate students of other educational institutions in the state.
(3) Faculty and graduate students from out-state educational institutions.
(4) Industrial contract research.

Graduate students, under the direct supervision of a faculty member of the University of Missouri, will be urged to utilize the research reactor facilities in their research programs. ~~Space and radiation facility assignments will be made by the Reactor Supervisor in consultation with the chairman of the~~

"Space and radiation facility assignments will be made by the Director in consultation with the graduate student's **MURR radiation work supervisor.** Routine irradiations for graduate students who have demonstrated competency in the handling of radioactive materials may be performed **when authorized by the MURR supervisor.**"

~~radioactive materials may be performed without the approval of his faculty advisor.~~

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The basic necessities for research will be provided by the reactor staff. These necessities will consist of the necessary utilities for driving the experiment, the handling tools associated with the reactor for the transfer or handling of radioactive materials, and radiation experiment supervision by the reactor staff. There will be a stockroom in the laboratory building from which the experimenter may requisition glassware, chemicals, and other assorted items of research equipment. Specialized equipment for the utilization of the reactor in a research program may be constructed in the facility shops or may be provided by the department sponsoring the particular graduate student.

~~Faculty members or graduate students from other~~
"Faculty members or graduate students from other educational institutions in the state may utilize the research reactor facilities upon written request to the Director."
~~the research reactor facilities upon written request to the Director of the laboratories.~~ They will be subject to the same supervision and control as members of the University of Missouri faculty. While in attendance in the reactor facility, they will be registered at the University of Missouri as either students, guests, or trainees. ~~This registration must be established through the University Admissions Office.~~ DELETED 1997

Whenever possible the research reactor and its associated facilities will be available to industrial users. ~~Industrial users shall make application to t~~ "Industrial users shall make application to the Director for the use of these facilities." ~~facilities.~~ This application should include a detailed description of the proposed experiment, together with a suggested time schedule and a listing of the necessary equipment that it is expected would be furnished by the University of Missouri. ~~The Director of th~~ "The Director will provide . . . ~~ies will provide~~ interested concerns with a price schedule for reactor services. This price schedule will reflect costs to the University for the operation of the research facilities. It should be pointed out that the industrial users are members of the last priority group, which is to say that all educational research

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activities take precedence over industrial research. It should further be pointed out that the University of Missouri, as an educational institution, is interested primarily in the procurement of industrial research which will further the teaching and academic programs.

12.6 Internal Safety Review

In order to make the reactor readily available to anyone qualified to use it and still maintain control of reactor experiments a straight forward internal safety review procedure has been established.

Anyone desiring to use the reactor facilty must complete a Reactor Utilization Request form. This form is comprehensive yet flexible enough to provide for a wide variety of reactor experiments. The requestor is urged to seek the assistance of the reactor facility staff in the preparation of his request. This will provide him and the reactor staff with a quick review of his proposed experiment and any possible hazards. "It is anticipated that difficulties . . . ~~many difficulties~~ will be discovered and corrected at this stage. 1997

Upon completion of the utilization request form it is submitted to the reactor supervisor who reviews and in turn submits it to a laboratory supervisor and the Reactor Health Physics Manager for review. If these ~~three~~ individuals concur that the experiment 1997 can be completed in a safe manner the requestor may proceed. 1996

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If any one of the reviewers desires, the requestor may be asked to meet with the ~~three~~ reviewers to provide further information or to resolve difficulties. 1997

~~If~~ "If at this point **either** of the reviewers . . ~~e reviewers~~ feels that the experiment should not be conducted as designed, and the experimenter cannot redesign his experiment, the ~~re~~ Reactor Manager ~~or~~ may request, through the facility director, that the request be presented to the reactor advisory committee. Under special circumstances the reactor advisory committee may be convened to review a single problem. ~~Normally, however, problems to be reviewed by the reactor~~ "Normally, however, problems to be reviewed by the Reactor Advisory Committee will be taken up during normal sessions." ~~monthly sessions.~~ 1997 1996 1997

The reactor advisory committee may find that additional information must be provided, that the experiment must be redesigned or that the experiment must be rejected.

1973-1974

13.0 ACCIDENT ANALYSIS

13.1.1 Introduction to the Original 5 MW Accident Analysis

1973-1974

It is difficult to define a maximum credible accident for the reactor under consideration. Every effort has been made to include engineering features which preclude the occurrence of an extensive accident. However, in an effort to develop a situation which

ssible set of cir-
of 10 per cent fuel
of fission products
postulated.

ing-pool type reactors
rational safety record
is partially inherent
ater moderated reactor
by slight changes in
additions are readily
l and temperature
this type of reactor.

it the laboratories
have demonstrated
ics of water moderated-

zard analysis,
linary swimming-pool
e inherent safety
device.

1973-1974

This section (13.1) is modified by inserting the following after the section title:

"For 10 MW operation, the University of Missouri proposed two design modifications which affect the accident analysis for the reactor. As a result of a review of the instrumentation and safety systems, they were modified as shown in Appendix A of the fourth addendum to HSR to conform with applicable criteria contained in General Design Criteria 20 through 25 of 10CFR50 and in IEEE Standard 279. Also, the primary coolant anti-siphon system was modified so as to maintain the integrity of the primary system under all conditions. It was the anti-siphon system that provided a path for fission products released in the primary system to pass into containment. A description of the modified anti-siphon system is presented in Appendix B of HSR Addendum 4.

Appendix C of HSR Addendum 4 discusses the reduction of the probability of occurance and the consequences of releases of radioactivity to acceptable levels."

2000

It is significant to point out that this reactor is intended exclusively for research programs conducted by faculty and graduate student personnel. The device will not be used for reactor experimentation or for student training. Secondly, it is pertinent to point out that no particularly hazardous types of experiments are intended in this reactor, such as fuel element testing or circulating fuel experiments.

The initial use of the center test hole will be limited to experiments which can be moved or altered only after reactor shutdown. It is intended however that at some future date techniques for an alternate method of use of this facility will be installed. A technique such as the installation of a hydraulic rabbit or pneumatic tube facility will be considered some time after evaluation of parameters for the flux trap position. Only after adequate data is in hand will a request be submitted to the Atomic Energy Commission for an addendum to the operating license to alter the mode of sample entry into the flux trap position.

ADD TO END OF PARAGRAPH

"Technical Specification Amendment 31, which allows movable and unsecured experiments in the flux trap, was approved by the NRC on September 20, 1999."

13.2 General Considerations

In those sections which follow certain items of concern to overall reactor safety will be enumerated and discussed. As was pointed out in earlier paragraphs, it is difficult to define a set of circumstances which would lead one to a maximum credible accident condition. However, general characteristics of flux trap reactors and specific characteristics of this particular device will be discussed under this heading of general considerations.

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1982
1967-
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13.2.1 Void Coefficients

The reduction of water density in the core of this reactor causes reactivity to decrease, as occurs in other water-moderated pool-type reactors. However, reduction of water density in any region other than the core causes a relatively smaller but positive reactivity change, unless it is accompanied by a similar density reduction in the core.

This void coefficient behavior is an unavoidable characteristic of flux trap reactors. In particular it expresses itself in those which are partially reflector moderated and which contain light water in regions other than the core.

The design has been made to eliminate concern with boiling in the island by using pool coolant water through this region. The design incorporates protective circuitry to minimize the probability of boiling in any region.

The calculated reactivity changes accompanying density changes of the water in each region are presented as void coefficients in Table 13.1.

TABLE 13.1

VOID COEFFICIENTS

Region	Void Coefficient Δ k/% void
Island	$+.773 \times 10^{-4}$
Core	-2.51×10^{-3}

It will be noted that though the island displays a positive void coefficient it is of small value compared to that negative coefficient displayed by the core. The region of greatest concern with respect to a positive void coefficient is the island.

13.2.2 Boiling Within the Island

The island flow is in parallel with the reflector, rod gap and test hole flow. The island is cooled by pool water flow.

Loss of flow would not normally result in boiling because this condition scrams the reactor, and loss of flow activates the convective flow within the pool which is sized to prevent the occurrence of boiling when the reactor is shut down. Boiling within the island or any other region, of course, is not a hazard (though certainly undesirable) after shutdown because the rods provide an adequate shutdown margin.

Making the assumption that the reactor were somehow operated at reduced flow through malfunction or deliberate bypassing of safety circuits, boiling should occur first in the core, causing a reactivity reduction.

Calculations indicate that a flow reduction of 20 per cent is required at 10 MW power level to cause surface boiling at the hottest fuel plate. It may by somewhat greater than this depending on the power distribution and the validity of the hot channel calculation factor. The water in the island however will boil only through volumetric boiling induced by conductive heat transfer from the core region or by radiation absorption in the water. During normal operation this heating causes a temperature rise in the island of only 6°F at design conditions. A temperature rise of at least 100°F would be required for boiling.

During low power operation of the reactor the pool coolant loop may or may not be operating. The reactor core is not pressurized and core cooling is by natural convection. If a power excursion without a scram were to occur, boiling conditions would be attained only in the core region, not in the island. If the pool coolant system is in operation adequate cooling is available for any condition. If the island is being cooled by natural convection the entire pool volume (approximately 20,000 gallons) is available as a heat sink. Under these conditions the core's negative temperature and void coefficients would control and limit reactor power.

13.2.3 Boiling Within Other Regions

The reflector, the test hole, and the rod gaps are also cooled by pool water, hence loss of flow in the pool cooling circuit could conceivably cause boiling without a compensating density change in the core.

Surface boiling in these regions is of greatest concern in the reflector immediately surrounding the rod gap. At the hottest spot on the inner face of the beryllium reflector, a velocity of about 3.5 feet per second is necessary to avoid boiling at 10 MW. The heat production is greatly attenuated in the rest of the reflector so this maximum flow requirement exists only at the inner face. However, the average velocity of coolant through the reflector region is 6 feet per second.

Inadequate flow would cause surface boiling to occur first at this inner face of the reflector. With surface boiling alone, the bulk liquid would still be subcooled and the average density reduction would be small. A reduction of flow by a factor of about fifteen is required to produce bulk boiling at the exit of this inner face. Essentially complete flow stoppage is thus required to approximate the condition of total water loss. Expulsion of all the water around the beryllium is calculated to result in the insertion of less than a prompt critical amount of reactivity.

Heat fluxes from the control rods and from nonfuel samples in the center test hole will be low. Flow reduction nearly sufficient to cause bulk boiling must occur before surface boiling results. With the design temperature rise of 6°F through these regions, boiling would result only with large flow reductions. Flow reduction in either region would cause the bulk temperature to rise and dump considerable heat into adjacent regions; however, with flow stoppage the heat transfer coefficient on one side of the wall would be greatly reduced, effectively insulating the region, and boiling would occur.

The occurrence of boiling in the regions cooled by the pool flow is prevented during operation by: (1) designing for much greater flow than is required to prevent boiling, (2) providing a low pool flow alarm and reactor scram, (3) monitoring the pool inlet water temperature, (4) monitoring temperature differential across pool inlet and outlet, and (5) requiring appropriate flow restrictions on experiments placed within the reflector irradiation pieces and being certain that all reflector positions are filled prior to startup to assure an adequate pressure drop across the reflector region.

Following shutdown, boiling in these regions is of concern only in the sense that some damage could conceivably result from the higher temperatures. To prevent boiling during shutdown, a convective coolant valve is provided which opens automatically upon loss of pump outlet pressure in the pool system. When open, this valve permits the pool water to circulate by natural convection.

13.2.4 Shearing a Beamport

A heavy object dropped from the crane or the bridge could shear one of the beam tubes extending into the pool. This would cause the loss of pool water if the beamport did not contain its shield plug and the closure flange. It is pertinent to point out that any beamport not being used for the extraction of a neutron beam will be closed, such that a complete shearing of a beam tube would still not produce a condition where water would be lost, since the liner and concrete shield plug are both sealed to withstand the pool pressure without leakage. In the case of the shearing of a beam tube with an open collimator installed or an experiment open to the beam hole floor one can postulate the loss of pool water. The closure of the beamport shutter would serve as a partial shutoff valve, but of course the shutter could only be closed if a collimator, or lines from an experiment, were not blocking its path. However, any obstruction to the flow would serve to decrease the rate of water loss from its maximum rate.

Provision is made to gravity feed and pump the 7000 gallon content of one demineralized water storage tank into the pool in the event of pool water loss. If the leak cannot be compensated by this means a raw water hydrant has been provided at the top of the pool with a quarter turn valve located in the floor immediately adjacent to the control room. Raw water can be dumped from this large hydrant at flow rates in excess of 1000 gallons per minute. It is calculated that at a rate of water addition of 1000 gallons per minute adequate water would be provided to retain more than three feet of water above a completely severed six inch beamport with no impediments in the port. A ~~re~~reduction to less than 23'~~es~~ in the pool water level scrams the reactor and initiates an alarm. The raw water hydrant is connected directly to the water main supplying the total research reactor facility and is fed from a 500,000 gallon underground reservoir located less than one mile distant from the reactor facility.

13.2.5 Rupture of Pool Loop

A serious leak in the lower plenum, or failure of the convective coolant valve on this plenum to close, would cause a reduction of flow to the test hole, island, reflector and rod gaps. This condition would be annunciated by an alteration of the pressure differential across the reflector region.

A break in the pool loop external to the shield would cause a rate of loss of water less than that from a sheared beamport since the line is six inches in diameter and loss from both ends is precluded by the installation of a check valve on the supply line. The pump flow of the pool loop is low relative to that which could result from the gravity head; hence in the event of a break beyond the pump, which might leave the pump running, the rate of loss would still be less than that from a sheared beamport.

The reactor is scrammed either by reduced flow in the pool cooling system by reduced pool level resulting from a water loss, or by low pool coolant pump outlet pressure.

If not already done automatically, sufficient time is available, following a low pool level alarm and resultant reactor scram, to trip the pool pump and close the pool stop valve before serious loss of pool water could occur. The demineralized storage tank and the large raw water hydrant are also available in an emergency to compensate for a water loss.

In the event that the pool is drained and the heat exchanger within the pool is exposed, the operation of the convective coolant loop for the reactor core is impaired. However, the reactor loop pump continues to run following the scram and the reactor loop removes the decay heat.

Also, the heat production in the reflector and rods due to fission product decay gammas will not cause meltdown of these members even with total loss of water.

13.2.6 Rupture of Reactor Loop

The first line of protection against loss of core water resulting from rupture of the reactor coolant loop is provided by the check valve on the inlet line and by the invert loop on the exit line. A vacuum breaker system at the top of the invert loop provides a siphon break and electrically operated flow stop valves are installed on the exit and supply line. These valves are located immediately outside the shield wall so that a line rupture between the core and the valve is improbable unless it occurs within the pool.

A major rupture anywhere within the pool for which the pressurizer system cannot compensate causes the reactor to scram and causes the primary loop to approach pool pressure. The loop remains filled because the external loops are presumably still intact. Regardless of the location of an internal break, the core is cooled adequately by the pump flow or by the convective loop within the pool, depending upon whether or not the reactor has scrammed and the reactor coolant loop isolated.

A rupture external to the biological shield will cause loss of pressure or a sudden decrease in flow. Either of these conditions will initiate these actions:

(1) Scram the reactor,
(2) Close the reactor loop isolation valves,
(3) Open the in-pool reactor convective loop valve, and
(4) Open the anti-siphon valves.

If the rupture is on the pump and heat exchanger side of the isolation valves the reactor loop in-pool piping remains flooded and core decay heat is removed by natural convection. If the rupture occurs between the shield and the isolation valves, or if the valves fail to function, the invert loop down-leg and the in-pool heat exchanger will be drained. In this case the core water may boil, but two pressure operated check valves installed through the side of the pressure vessel permit water to drain into the core preventing it from boiling dry.

13.2.7 Pump Failure

Failure of the reactor loop pump causes scram due to loss of flow and activates the convective coolant loop to remove the shutdown heat. An emergency pump is not provided.

Opening of the lower plenum convective loop valve permitting pool water to circulate up through the reflector regions is initiated by the closing of the pool supply isolation valve (509). This isolation valve is closed automatically upon loss of pressure in the pool supply line.

Failure of a secondary pump leads to higher water temperatures in the core and if these temperatures are excessive they cause reactor scram. In the event of total loss of the heat exchanger, the pool absorbs the shutdown heat, primarily through the convective loop.

13.2.8 Power Failure

In the event of power failure the reactor is scrammed. Pump failure due to power loss or for any other reason is handled adequately by the convective cooling systems.

13.2.9 Flooding of Beamports

A calculation of the beamport volume water addition or water homogenized in the graphite reflector indicates a small reduction in reactivity upon flooding of the beamports. In light of the small value resulting from this calculation it is concluded that the effect of flooding or emptying a beamport will be slight. Operating procedures preclude the flooding or draining of a port at any time during reactor operation.

13.2.10 Sample Movement

Results of reactivity calculations performed for various compositions of the outer (graphite) reflector suggests that reactivity changes resulting from motion of experimental absorbers in the graphite reflector region or in the beamports will be minimized by the 2.71 inch layer of beryllium interposed between the core and the graphite reflector region. It is not intended that this layer of beryllium will be removed or moved in any manner, nor will experiments be located within this beryllium ring.

As is pointed out in Section 8.1, the beryllium reflector surrounding the core of this reactor provides a very effective decoupling of experiments from the core. In essence, the alteration of reactivity which can result from insertion or removal of samples external to the beryllium annulus is minimized.

13.2.11 Refueling Accident

Procedures for refueling of the reactor are described in detail in Section 7.1.8 and Appendix A. Appropriate tools have been provided to facilitate replacement of fuel assemblies. Fuel assemblies are removed by a special grappling tool which positively connects to a lifting bracket designed as part of each fuel assembly.

In the refueling operation the pressure vessel lid is removed and fuel assemblies are transferred one at a time from the core to a temporary storage rack located on the pool floor. These assemblies are transferred from the storage rack on the pool floor into the permanent spent fuel storage racks only after the pool level has been raised back to its normal operation level.

The transfer of fuel assemblies to a shipping cask is performed under water with the cask located on the shelf behind the weir. Transfer of spent fuel to a shipping cask will not be made until after adequate decay time. This time will be such that a full assembly in a horizontal position can be cooled by heat transfer in air, thus in the event of an accident involving a dropped cask or loss of water coolant from a cask, the fuel assembly would represent only a direct gamma hazard and fission products would not be released by a meltdown accident. Tests performed on ORR fuel assemblies indicate that several days cooling time are required following 10 MW continuous reactor operation.

All storage racks have been designed to be safe with regard to criticality.

13.3 Reactivity Considerations

At 10 MW the reactivity worth of Xenon is considerably greater than in low powered pool reactors. With continuous operation the equilibrium of Xenon and samarium are computed to be 4 per cent in reactivity; with cyclic operation (eight hours on and sixteen hours off) the maximum value is 7.1 per cent. In general the reactivity worth of equilibrium Xenon tends to be greater for a flux trap than for an equivalent swimming-pool type core because of the higher neutron leakage.

It is concluded that the reactivity requirements of the flux trap design are typical of fully enriched aluminum water reactors operated at 10 MW. They are less than for an equivalent swimming-pool type core when said core is operated on an eight hour per day basis at 10 MW. The total reactivity of a six kilogram core for 10 MW operation on an eight hour day may be summarized as follows:

Sm and maximum cyclic Xe	0.071
40-day continuous fuel burnup (529 gm U^{235}) plus fission products	0.0324
Temperature	0.0048
Subtotal	0.079
Excess available for experiments with 6-Kg loading	0.038
Total with 6 Kg	0.117

The total reactivity installed with the initial core of 5.2 kilograms based on eight hour daily operation is as follows:

Available reactivity	0.0973
Peak Xenon (Friday morning)	0.0329
Peak samarium (Friday morning)	0.0079
Fuel burnup and fission products after 400 MWD operation	0.0324
Temperature increase (20°C to 65°C)	0.0048
Experiments	0.0193
Total	0.0973

13.4 Containment Considerations

13.4.1 Introduction

There are a number of means for preventing fission product release exclusive of the containment enclosure. The reactor core is contained within an aluminum pressure vessel designed to withstand a 150 psi pressure. The core coolant loop has an integral in-pool heat dump. This latter safeguard minimizes the probability of core meltdown. The core assembly should never go dry due to the presence of the invert loop. The pressure vessel, core, invert loop, in-pool heat exchanger are all contained within a 10 foot diameter by 30 foot deep water filled pool. There exists then a number of barriers to leakage of fission products prior to the utilization of the containment structure as a final barrier.

13.4.2 Containment Structure

The reactor building and its penetrations are described in detail in Section 3.0. The concrete walls of the building have been designed to withstand a 2 psi differential pressure. The building has been

1967-
1968

designed to provide less than ~~1~~ ~~p~~10% ~~cent~~ leakage of the contained air volume in a twenty-four hour period. 1967-1968

A water seal is provided where the utility lines enter from the laboratory building into the reactor containment structure. The laboratory building is at atmospheric pressure and the water leg is designed such that a differential pressure of 2 psi elevates the water in the trap. This 2 psi differential pressure in either direction across the water leg will cause the pressure to be relieved through the water leg. When the pressure is relieved, the water level drops thus resealing the containment.

The water seal is provided to avoid the rupture of the enclosure in the event of a pressure differential in excess of 2 psi. This value exceeds the over-pressure which could result from a nuclear accident in which case no leakage would occur through the seal. However, in the most extreme condition under which one might postulate a quantity of air leaking through the seal for pressure relief, some degree of scrubbing would accompany this release depending upon how quickly the pressure built up. The building is designed to withstand all pressure rises and to prevent any release at the pressure anticipated (0.5 psi) from a most adverse nuclear incident.

Detailed attention has been given in the construction of the facility to assure that all penetrations of the enclosure are of high integrity.

The pedestrian entry is of the air-lock type and is sealed by an inflatable gasket. The heavy equipment entry is a single door, closed at all times during reactor operation, sealed with an inflated gasket. The exhaust lines from the building are closed by quick-closing positive valves actuated by either automatic signals from the radiation monitoring system or by a building evacuation or isolation alarm initiated by the reactor operator. All electrical lines enter the building through a steel plate positioned in the wall of the containment structure. Each electrical connection is made through a gas-tight seal gasketed to the steel plate.

13.4.3 Pressure Loading on Containment Structure

It is difficult to envision a pressure buildup within the containment because the maximum design temperature of the bulk water in the reactor is only 160°F. In the case of loop ruptures the core water would still be subcooled relative to atmospheric pressure and the water would not flash to steam as in the case of a highly pressurized water reactor.

In the event of a runaway leading to steam formation and loop rupture within the pool the pool water would probably quench any steam releases. It is calculated, for example, that the one foot diameter column of water immediately above the one foot diameter reactor vessel would absorb 108 MW seconds of energy before reaching 212°F.

The energy release necessary to cause a 2 psi equilibrium loading within the reactor vessel or within the reactor containment structure has been considered. This calculation neglected all heat sinks.

With initial conditions of 75°F air at 50 per cent relative humidity and a pool water temperature of 100°F, and with saturation of air assumed for the final condition, the amount of steam that must be released from the pool into the 240,000 cubic foot building volume to cause a 2 psi rise corresponds to 1,040 MW seconds of energy release. Under these conditions the final equilibrium temperature within the containment structure would be 113°F. This energy release of 1,040 MW seconds is far greater than the maximum energy release of 135 MW seconds observed in the Borax experiments.

13.4.4 Effect of a Possible Aluminum-Water Reaction

There is a possibility that an extreme fuel meltdown would cause a molten aluminum-water reaction releasing considerable amounts of energy. The conditions under which such a reaction might occur are widely discussed in the literature pertaining to reactor hazards.

It is currently believed that the occurrence of a rapid reaction requires first, that the metal be present in finely divided form in droplets or particles of less than 500 microns, and second, that the aluminum temperature be above its melting point.

The fuel plates for this reactor contain 29.4 Kg of aluminum, over half of which is cladding. If this quantity of aluminum were to react completely with water it would produce 441 MW seconds of energy and release about 1,340 cubic feet of hydrogen at standard conditions. If the hydrogen were completely recombined an additional 523 MW seconds of energy would be released. This total release of 964 MW seconds is less than the corresponding energy in steam which would have to be released to cause a 2 psi equilibrium overpressure.

15 MW sec/KG

To postulate the total reaction of the aluminum and the recombination of the evolved hydrogen is an excessively conservative assumption. A more realistic approach might be that 10 per cent of the metal would react consistent with an assumption of 10 per cent meltdown as used in the dosage calculations of this section.

On the basis of these observations it is concluded that the assumption of 2 psi maximum overpressure is completely adequate.

13.5 Maximum Credible Accident

13.5.1 Introduction

In the assumption required to provide the conditions for initiation of the maximum credible accident there are some which are extremely remote, others which involve multiple simultaneous failures, and others which ignore the design efforts expended to minimize a core meltdown accident. However, the only credible accident of significance in terms of public hazard must involve the release of an appreciable fraction of the fission product inventory. Anything less might endanger the operating crew and the experimenters but would not represent a public hazard. Then the exercise which will be performed in this section will involve a partial core meltdown accompanied by fission product release.

The most serious accident which one can postulate for any type of reactor is a loss of coolant accident with consequent loss of heat transfer. Under these conditions the heat generation from fission product decay continues after the reactor is scrammed and the fuel temperature will rise. This temperature increase may be sufficient to melt the fuel elements. An accident of this type, that is plain loss of coolant accident, is not considered credible for this reactor. Instead, one must construct a model wherein a driving force is provided to continually expulse the water from the core. The energy required to achieve this must be provided from a much larger source than just decay heat, namely an operating core.

13.5.2 Pre-accident Conditions and Assumptions

(1) Assume the reactor has operated continuously at 10 MW for 40 days.
(2) Assume the reactor is at 10 MW power.
(3) Assume a loss of flow without a pressure loss in the reactor core loop.
(4) Assume failure of the flow monitor to scram the reactor at 85 per cent of flow.
(5) Assume failure of annunciators to distinguish and annunciate low flow and increasing temperature.
(6) Assume failure of reactor operator to discern flow reduction or temperature rise.

The pre-accident situation established at this point consists of a core operating at reduced flow (i.e. less than 85 per cent of normal but greater than 50 per cent of normal). A condition which should have initiated a scram, but didn't. At this time, and under these conditions, the temperature will rise due to the reduced flow. The rate of rise will be a function of the flow rate, but boiling will not take place so long as the flow rate is 50 per cent or more of normal. It now becomes necessary to postulate the failure of another system.

(7) Assume failure to scram on the temperature signal from the detector in the 12 inch coolant line from the core. This is set to initiate scram at 110 per cent of normal.

We now have a rapidly deteriorating situation. We are suffering an ever decreasing rate of coolant flow. The reactor continues at 10 MW power. The temperature is rising in the core which provides some negative reactivity, but assume a negligible power slump. Now the coolant flow rate has dropped to 50 per cent of normal.

13.5.3 Accident Conditions and Assumptions

When the flow rate through the core drops to 50 per cent of normal the core is isolated and the anti-siphon line opens. This is accomplished automatically at 50 per cent flow. The sequence is as follows:

(1) The core coolant pump shuts down.
(2) The isolation valves in the core coolant loop are closed which isolates the core from the coolant loop piping, pump, demineralizer, and heat exchanger.
(3) The two anti-siphon valves open, venting the in-pool portion of the reactor core coolant loop to the building atmosphere.
(4) The in-pool heat exchanger opens to permit convective flow of the core water through the in-pool heat exchanger.

These systems were designed to perform the enumerated sequence of functions in the event of a loss of flow or loss of coolant accident, and would, under normal (scrammed) conditions, provide (a) decay heat removal through the in-pool heat exchanger and (b) anti-siphon action of the invert loop.

However, in this accident analysis it is assumed that temperature, flow, and operator scrams have failed. Then our reactor is operating at 10 MW with no coolant flow and vented to the reactor pool surface through the anti-siphon leg.

The consequences of the postulated conditions are boiling and expulsion of steam and water through the anti-siphon leg. Loss of moderator would effectively scram the reactor, but the core would be dry and partial meltdown would occur.

This accident assumes that steam formation, followed by steam-water expulsion, occurs at a rate such that the 4 inch anti-siphon line can handle the volume flow. In the event that this was not the case the rapid increase in pressure will rupture the 150 psi rupture diaphragm which closes the 12 inch convective flow standpipe coming off the top of the invert loop. If this diaphragm is ruptured the core is immediately flooded by pool water. The core would oscillate through a boiling-shutdown cycle, assuming the rods remain out.

However, the assumption first made above, that is a dry core, leads to the worst conditions.

(1) Assume ten per cent fuel meltdown.

(2) Assume fractional release of fission products as follows:

    Noble gases - 75%
    Halogens - 25 %
    All others - 1%

(3) Assume immediate release of the fission products within the reactor building. The entrainment or sublimation of iodine is

neglected in the maximum dose calculations, but the dose reduction resulting from entrainment has been calculated and is included in the summary of results.

(4) The reactor building is assumed to leak at a constant rate of 1 per cent day. Containment rupture is not considered.

13.6 Accident Consequence

13.6.1 Fission-Product Sources

The total fission product gamma activity is presented in Table 13.2 according to energy and time after reactor shutdown, following 40-day continuous operation at 10 MW. These values are based on the data of Mr. M. R. Smith (4) which were obtained by integrating the fission-pulse data of Putnam and Knabe (5). In the latter work, the results of spectral measurements performed at short times after fission were used in conjunction with the total spectral data obtained by calculation of individual species of fission products. By this method the gross activity can be determined down to short times following release or shutdown.

TABLE 13.2
TOTAL-DECAY GAMMA ACTIVITY

Energy Group	Energy Mev	Activity, 10^{16} Mev/sec. Seconds After Release					
		10^0	10^2	10^3	10^4	10^5	10^6
1	0.1-04	11	6.7	4.5	2.6	1.23	0.33
2	0.4-0.9	49	30	19.7	11.2	5.0	1.58
3	0.9-1.35	36	20	9.5	3.0	1.2	0.53
4	1.35-1.8	32	18	9.8	4.3	2.0	1.1
5	1.8-2.2	21	11	5.9	2.6	0.28	0.017
6	2.2-2.6	17	8.7	4.0	0.95	0.20	0.080
VII	>2.6	38	11	2.2	0.12	0.0075	0.0047
	Totals	204	105	55.6	24.8	9.92	3.64

The total activity of those noble gases and halogens which contribute significant gamma activity or ingestion hazard are tabulated in Table 13.3. These data are obtained from the curves of ORNL-2127 (6) and include the effects of burn-up of individual nuclides. The thermal flux is assumed to be 5.56×10^{13} and the number of U^{235} atoms 14.37×10^{24}.

TABLE 13.3
TOTAL GAMMA ACTIVITY FROM NOBLE GASES AND HALOGENS IN 10^{16} MEV/SEC

	Seconds After Release	Gamma Activity, 10^{16} Mev/sec					
		10^0	10^2	10^3	10^4	10^5	10^6
Iodine	131	.317	.317	.317	.314	.287	.117
	132	2.274	2.274	2.274	2.274	1.93	.204
	133	.847	.847	.838	.772	.235	----
	134	2.766	2.705	2.218	.304	-----	----
	135	3.106	3.106	3.019	2.323	.171	----
	136	.202	.085	-----	-----	-----	----
Bromine	83	.007	.007	.007	.002	-----	----
	84	.813	.813	.813	.016	-----	----
	87	3.237	1.00	-----	-----	-----	----
Total Halogens		13.569	11.154	9.486	6.005	2.623	.321
Krypton	88	.244	.244	.244	.12	-----	----
	133	.163	.163	.163	.163	.163	.042
Xenon	133m	.002	.002	.002	.002	.002	----
	135	.061	.061	.061	.12	.09	----
	135m	.302	.302	.302	.21	.018	----
	137	2.28	1.60	1.14	-----	-----	----
	138	2.22	2.22	1.11	.002	-----	----
Total Noble Gases		5.272	4.592	3.022	.617	.273	.042

Use of these curves required a composite correction factor of 0.682 to account for reduced thermal fission cross section and for epithermal fission. It should be noted that the total activities in Table 13.3 are probably underestimated at times less than 10^3 seconds because of short-lived activities for which decay schemes are unknown.

It is assumed that 10 per cent of the fuel melts down and that 75 per cent of the noble gases, 25 per cent of the halogens, and 1 per cent of all other fission products present in the molten fraction are released immediately to the containment.

The energy spectrum of the total fission-product gamma activity is available from Table 13.3 For calculation of external gamma exposure, it is assumed that the gamma energy spectrum of both the halogens and noble gases is identical to that of the total gamma activity at each time of interest. The assumed composition is largely halogens and noble gases; hence, the assumption is not strictly valid. However, spot calculations based on the dominant halogen and noble gas activities indicate that little error is introduced in the cloud calculation by this assumption, primarily because the dose rate per unit energy flux is nearly independent of energy. The dose from the containment is more sensitive to the spectrum chosen, but the error again appears small because of the low attenuation factors involved.

Accordingly, an effective percentage of the total gamma activity is determined at each time after release, based on the total Mev/sec for each class of fission products and their assumed release rates. For example, at 10^5 seconds, when this effective fraction is a maximum, the halogen energy fraction is .264 of the total (see Tables 13.2 and 13.3). The effective fraction of the total energy release is then:

Noble gas equivalent	=	(.028 x .75 x .1 meltdown)
+ Halogen equivalent	=	(.264 x .25 x .1 meltdown)
+ All others	=	(.708 x .01 x .1 meltdown)
Effective Fraction		.0094

The effective fraction at each time decade is applied to the total gamma energy available, given by Table 13.2, to obtain the gamma activity present in the containment as a function of time and energy.

The method tends to underestimate the initial dose rates because of the underestimate in Table 13.2 at short times. However the time integrated doses are not affected significantly.

13.6.2 Direct Gamma Radiation From Containment

The walls of the reactor building are of ordinary concrete and have a minimum thickness of 12 inches.

Dose rates and time-integrated doses were first computed for an observer at the exterior face of the building, with the assumed fission-product source uniformly distributed within the building. With the containment regarded as an infinite slab 60 feet thick, shielded with one foot of concrete, the observer receives a dose rate of 59 r/hr. at 100 sec after release and accumulates a dose of 53 r during the first two hours. With iodine eliminated by entrainment, these values would be reduced to 40 r/hr. at 100 sec and 23 r in 2 hours. The estimate is conservative in that the effective source volume would probably include only the space between the building wall and the pool structure. The dose under these conditions might be one-third as great.

At large distances the containment is regarded as a point source shielded by one foot of concrete. With air attenuation and appropriate buildup factors applied to each energy group, the dose rates and time-integrated doses are as shown in Figure 13.1. The dashed line indicates the dose reduction if all iodine were removed by entrainment in the pool water.

The radiation 1000 feet distant from the reactor would be less than that at 300 feet by a factor of about 30.

Shielding by the laboratory building, which envelops the lower third of the reactor building, has been neglected.

13.6.3 Dosage Resulting From Containment Leakage

The fission-product source described in Section 13.6.1 is assumed to be distributed uniformly within the containment, and the containment is assumed to leak at the uniform rate of 1 per cent of its volume in 24 hours. Rupture of the containment is not considered.

Cloud gamma dosage and thyroid dosage resulting from iodine ingestion are determined at downwind distances of 300 feet, 1000 feet, and 5000 feet.

13.6.3.1 Diffusion Calculations

The activity is assumed to diffuse as from a continuous point source on the ground, but with corrections to include the effects of building eddies and wind variability. The downwind concentration on the plume centerline is given by Sutton's equation multiplied by appropriate reduction factors:

$$\chi = \frac{2Q}{\bar{u}\pi C_y C_z x^{2-n}} C_1 C_2 C_3$$

where χ = centerline concentration, μc/cc or Mev/cc-sec,

Q = release rate, μc/sec or Mev/sec^2,

$\bar{u}$ = wind speed, m/sec,

C_y C_z = horizontal and vertical diffusion coefficients,

x = distance downwind in meters, and

n = stability parameter.

Two wind conditions are considered, as described in Table 13.4, but the results presented are primarily for the case of inversion.

TABLE 13.4

METEOROLOGY PARAMETERS

	Inversion	Lapse
$\bar{u}$, m/sec	1	6
n	0.5	0.25
C_y, $(m)^{n/2}$	0.4	0.25
C_z, $(m)^{n/2}$	0.07	0.25

(1) Dilution by Building

The additional downwind diffusion caused by the plant structure is included by a correction recommended by the U.S. Weather Bureau staff for a similar application:

$$C_1 = \frac{\pi \quad C_z \, C_y \, x^{2-n}}{\pi \quad C_z \, C_y \, x^{2-n} + A/2}$$

Where A is the minimum projected cross-sectional area of the structure, taken as 465 square meters.

It is noted that this correction gives essentially the concentration from an equivalent disk source having the same total source strength spread uniformly over its area A. A better analogy might be a half-disk, also having area A, in which case the term A/2 in the equation should be replaced by A and greater dispersion is indicated.

(2) Wind Variability

Observation and photographs indicate that for a period of time beyond a few minutes there is an inherent variability of wind direction, whereas Sutton's equation yields essentially instantaneous downwind concentrations. At the suggestion of the Special Projects Group of the U.S. Weather Bureau, previous hazard studies have included a concentration reduction factor for this variation in wind direction (7). This is being proposed as standard by Subcommittee N-6 of the ANS, and it has support from experiments (8).

Assuming Gaussian distribution of the wind in any 45^{o} sector with a ratio of occurence at centerline to edge of 1000-to-1, the following correction factor was derived and used:

$$C_2 = \left[1 + \frac{x^n \tan^2 (\pi/16)}{C_y^{\ 2} \ln 1000} \right]^{-1/2}$$

This factor is 0.65, 0.50 and 0.38 at 300 feet, 1000 feet, and 5000 feet, respectively, for the inversion condition.

(3) Time of Flight Decay

The activity concentration at each downwind station is reduced by exponential decay for individual bulk activities, to include decay during the time of cloud transit, $t = x/\bar{u}$. This represents correction factor C_3.

13.6.3.2 External Gamma Dosage

The average plume centerline concentrations are determined for each energy group based on the energy sources of Section 13.6.1 and a 1 per cent per day leak rate, and using the cloud dilution calculations described in Section 13.6.3.1. The average centerline concentration is assumed to persist uniformly throughout the hemisphere above the observer and build-up is neglected. The dose rate from each energy group is given by:

$$r/hr = \frac{C \chi}{2 \mu}$$

where χ = concentration, Mev/cc-sec

μ = energy absorption coefficient for air, cm^{-1}

C = conversion factor; energy flux to dose rate.

The downwind doses and dose rates are presented in Figures 13.3 and 13.4 for the case of inversion conditions. Neglect of the iodine activity causes the dose reduction indicated by the lower curves in Figure 13.2.

It is noted that the assumption of a uniform cloud having the average centerline concentration overestimates the gamma dose, particularly at short distances and under stable wind conditions.

13.6.3.3 Iodine Ingestion

Of the ingested isotopes, only iodine is considered because the iodine dose to the thyroid is generally the most severe of the doses calculated.

All the iodine activities of Table 13.3 are included, and the average plume centerline concentrations are determined by the methods of Section 13.6.3.1. For each integrated time of exposure the initial concentrations are reduced to correspond to the average value of the exponential decay during the time of exposure. Dose rates are then determined by the specifice exposure values (mrem/inhaled microcurie) presented by Burnett (9) for each iodine activity. A breathing rate of 500 cc/sec is assumed.

The resulting dose rates and doses are plotted in Figure 13.4 and 13.5 for the case of inversion. The effect of iodine retention by the pool water is not included in these figures.

13.6.3.4 Summary of Calculated Dosage

The assumption used throughout the calculations may be summarized as follows:

(1) 10 per cent fuel melting
(2) 75-25-1 release of noble gases, halogens, and other fission products from the molten fuel,
(3) building leakage at 1 per cent per day,
(4) Inversion weather with 1 m/sec wind.

Reduction factors in the cloud calculations include decay during the time of flight, added diffusion by the building, and a small correction for wind variability (0.65 to 0.38 with inversion).

The initial dose rates and the time-integrated doses are tabulated in Table 13.5 where reduction due to iodine retention by pool water is neglected. These values represent the consequences of the maximum credible accident.

TABLE 13.5
DOSAGE SUMMARY WITHOUT IODINE RETENTION BY THE POOL WATER - INVERSION CONDITIONS ASSUMED

Downwind Distance	300 feet	1000 feet	5000 feet
Initial *Dose Rate, rem/hr			
Direct γ	0.22	0.008	--------
Cloud γ	0.027	0.011	0.001
Thyroid	3.92	1.73	0.214
2-Hour Dose, rem			
Direct γ	0.14	0.005	--------
Cloud γ	0.033	0.014	.001
Thyroid	7.32	3.24	0.40
2-Day Dose, rem			
Direct γ	0.94	0.034	---------
Cloud γ	0.336	0.135	0.012
Thyroid	106	47	5.80

TABLE 13.5 (cont'd)

10-Day Dose, rem			
Direct γ	1.8	0.064	--------
Cloud γ	0.692	0.279	0.025
Thyroid	166	73.7	9.09

*Direct gamma dose rates at 100 sec; others at 100 sec plus time of flight.

Under lapse conditions the cloud gamma and thyroid doses indicated in Table 13.5 would be reduced substantially. Including the changes due to different time of flight and wind variability, the lapse values would be less by factors of 0.100, 0.0047, and 0.0021 at 300, 1000, and 5000 feet, respectively.

13.6.4 Retention of Iodine in Pool Water

The effectiveness of the pool water in retaining the iodine nuclides can be estimated from the results of experiments described by Whelchel and Robbins (1). In these experiments Xe, Kr, and I were released under pressure to a pool with the discharge very mear the surface of the pool. The fraction of each element which escaped to a containment above the pool was then determined. A large fraction of the noble gases escaped, but the pool water was found to retain nearly all the iodine.

Shortly after release, the fraction of Iodine in the containment was 5.8×10^{-7}; five hours later, it was 7.2×10^{-6}.

Assuming that the mode of release was a pulse followed by a constant release rate from the pool, the transient fraction of iodine in the containment can be described by the following equation:

$$\frac{dC}{dt} = R - \alpha C$$

where C = fraction of iodine in containment at time t,

R = release rate from pool = $\underline{7.2 \times 10^{-6} - 5.8 \times 10^{-7}}$ = 13.24×10^{-7} hrs,$^{-1}$

α = release rate to atmosphere = $\frac{.01}{24 \text{ hr.}}$ = 4.17×10^{-4} hr $^{-1}$.

If, at time zero, the fraction of iodine in the containment is just that in the pulse. ($Co = 5.8 \times 10^{-7}$) the solution is:

$$C = \frac{R}{\alpha} (1-e^{-\alpha t}) + Co\, e^{-\alpha t}$$
$$= 3.18 \times 10^{-3} (1-e^{-\alpha t}) + 5.8 \times 10^{-7} e^{-\alpha t}$$

Accordingly, the maximum iodine content would increase with time, approaching 3.18×10^{-3} of the total. Ten days are required to reach one-tenth of this value.

The build-up with time is offset by the iodine decay. For example, the composite effect of decay is such that the total dose from time zero to infinity is reduced for each iodine activity by the factor:

$$\frac{R}{\alpha} \left[\frac{\alpha}{\alpha + \lambda}\right]$$

where λ is the decay constant for the particular activity.

This factor varies from 3.3×10^{-4} for I^{131} to 4.6×10^{-5} for I^{136}. Consequently the thyroid dose is negligible if the iodine is released through the pool water and provided the effectiveness of retention is equal to that observed in the experiment.

The consequent reductions in dosage are indicated in Table 13.6 where all contributions from iodine are neglected.

TABLE 13.6

DOSAGE SUMMARY-INVERSION CONDITIONS-NO IODINE

Downwind Distance	300 feet	1000 feet	5000 feet
Initial Dose Rate*, rem/hr			
Direct γ	0.145	0.005	---------
Cloud γ	0.018	0.007	0.001
2-Hour Dose, rem			
Direct γ	0.068	0.002	--------
Cloud γ	0.012	0.005	0.001
2-Day Dose, rem			
Direct γ	0.30	0.011	--------
Cloud γ	0.071	0.028	0.003
10-Day Dose, rem			
Direct γ	0.64	0.023	--------
Cloud γ	0.137	0.054	0.006

*Direct gamma dose rates at 100 sec; others at 100 sec plus time of flight.

13.7 Conclusions

The maximum credible accident postulated in this report involves multiple failures of safety instrumentation and human error. While credible, the accident postulated is certainly improbable.

The resultant consequences of the accident are pessimistically estimated. These estimates are identical to those of the hypothetical accident posed in the Premiliary Hazard Report.

It is the opinion of the authors that the facilities described can be operated without undue risk to the health and safety of the public.

13.8 References for Section 13.0

1. Whelchel and Robbins, Pressure Suppression Containment for Nuclear Power Plants, ASME-59-A-215.

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6. Blomeke, J. O., and Todd, M. F., Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time, ORNL-2127.

7. Preliminary Hazards Report for the RCPA Elk River Reactor, ACF Ind., Washington D. C., March, 1959.

8. HW-56605, Classified.

9. Burnett, T. J., Reactor Hazard vs. Power Level, Nuclear Science and Engineering, 2, 1957.

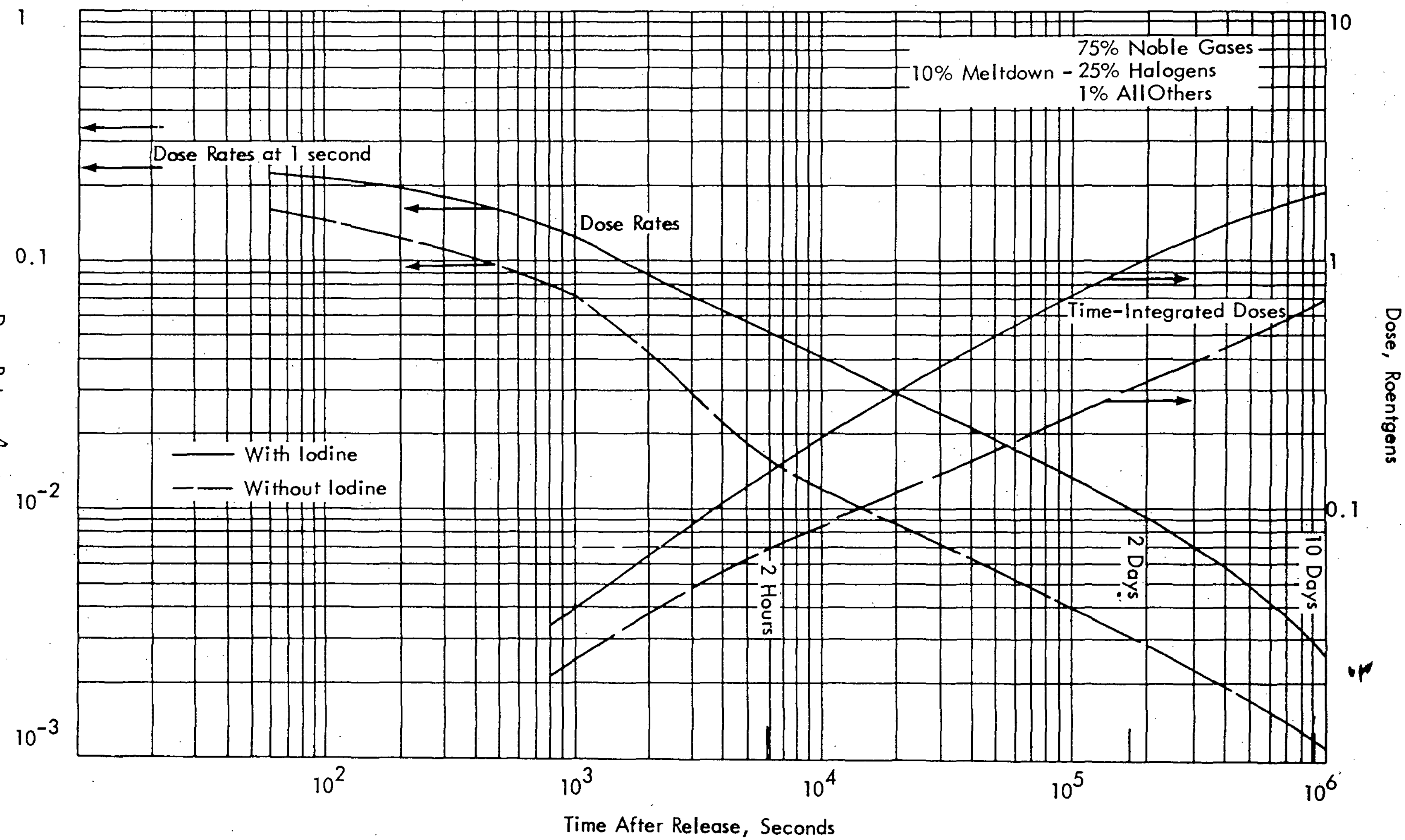

Figure 13.1 Direct Gamma Dose Rate and Dose at 300 Feet From Containment

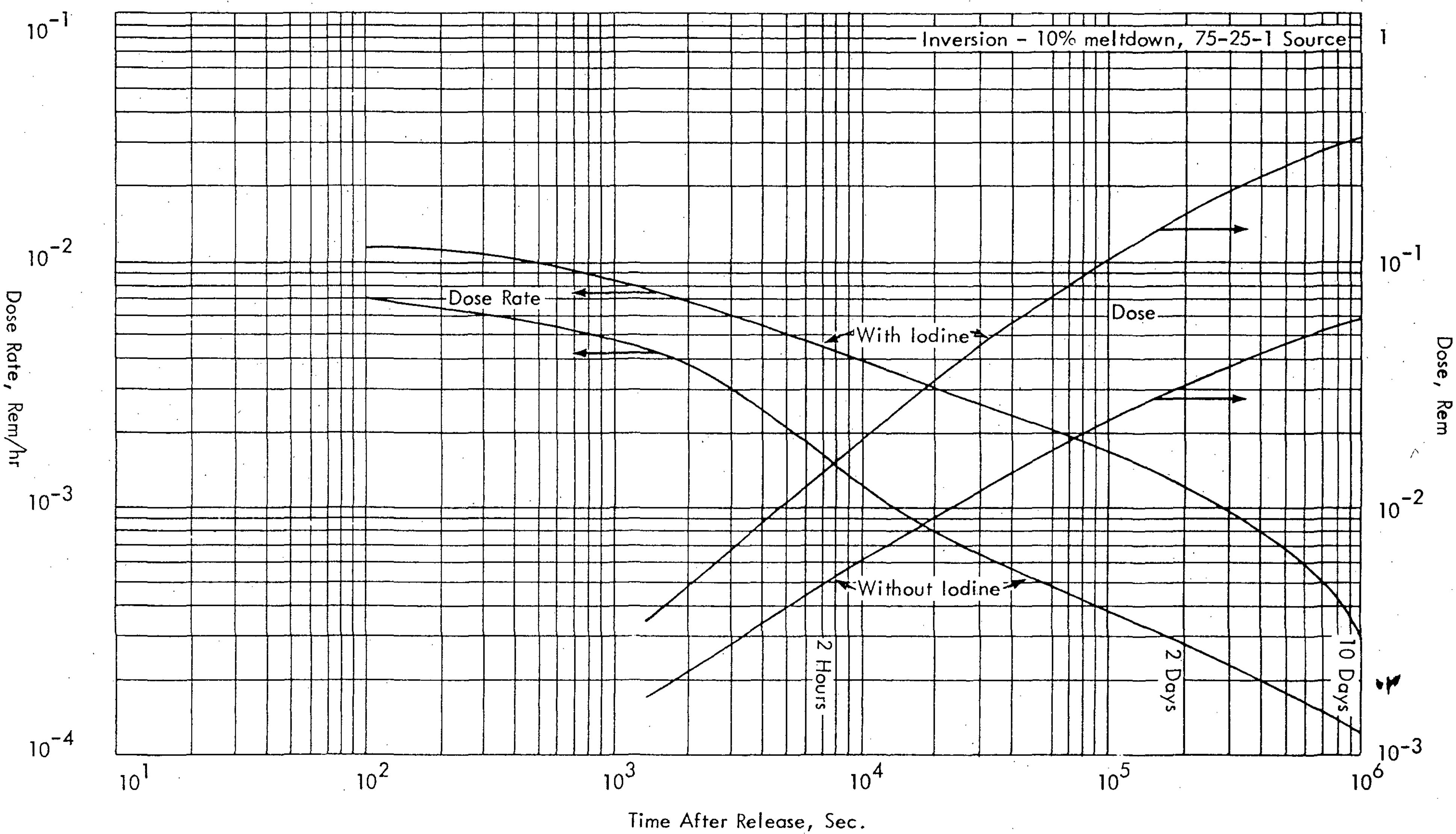

Figure 13.2 External Gamma Exposure From Cloud at 1000 Feet

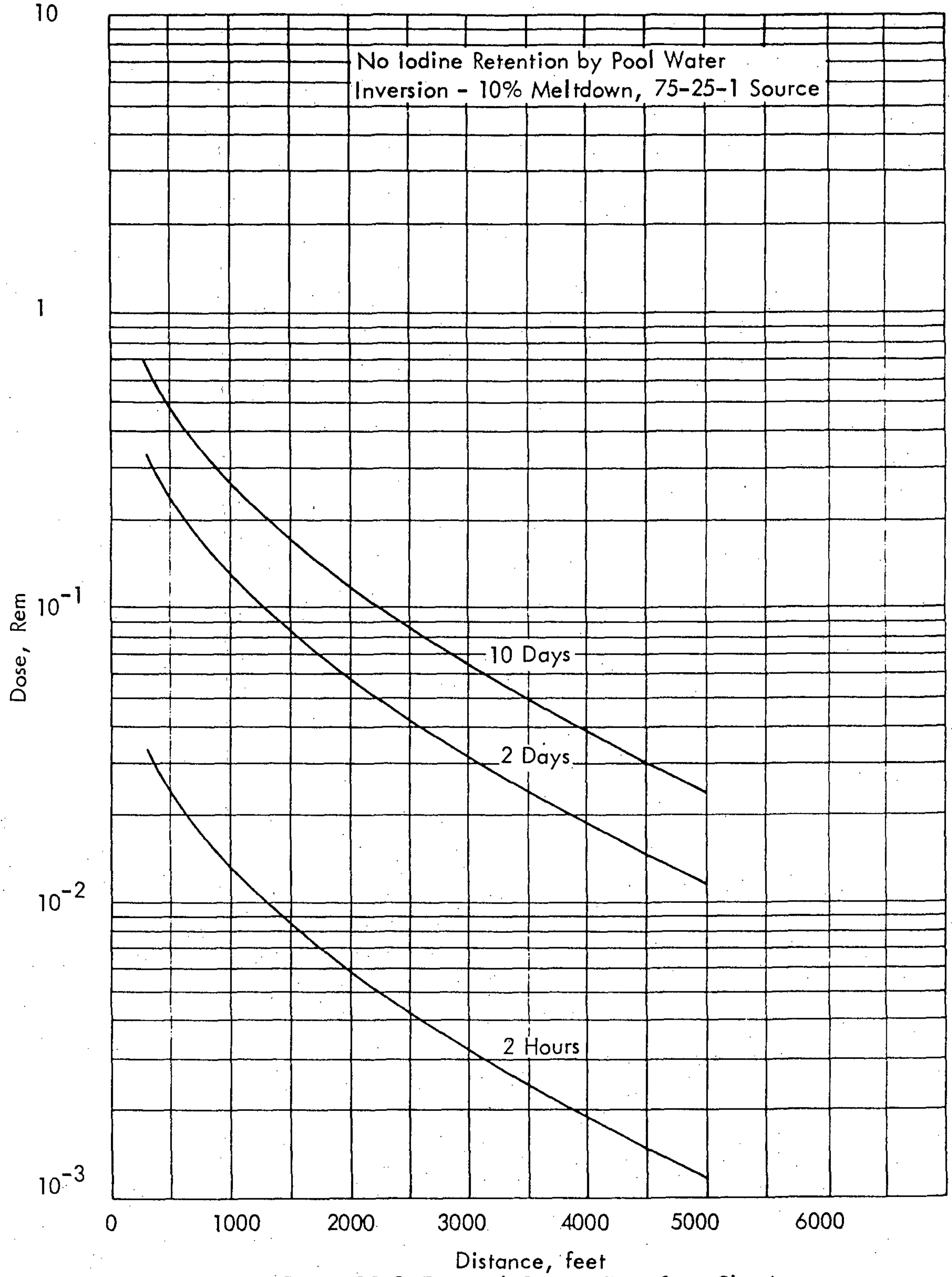

Figure 13.3 External Gamma Dose from Cloud

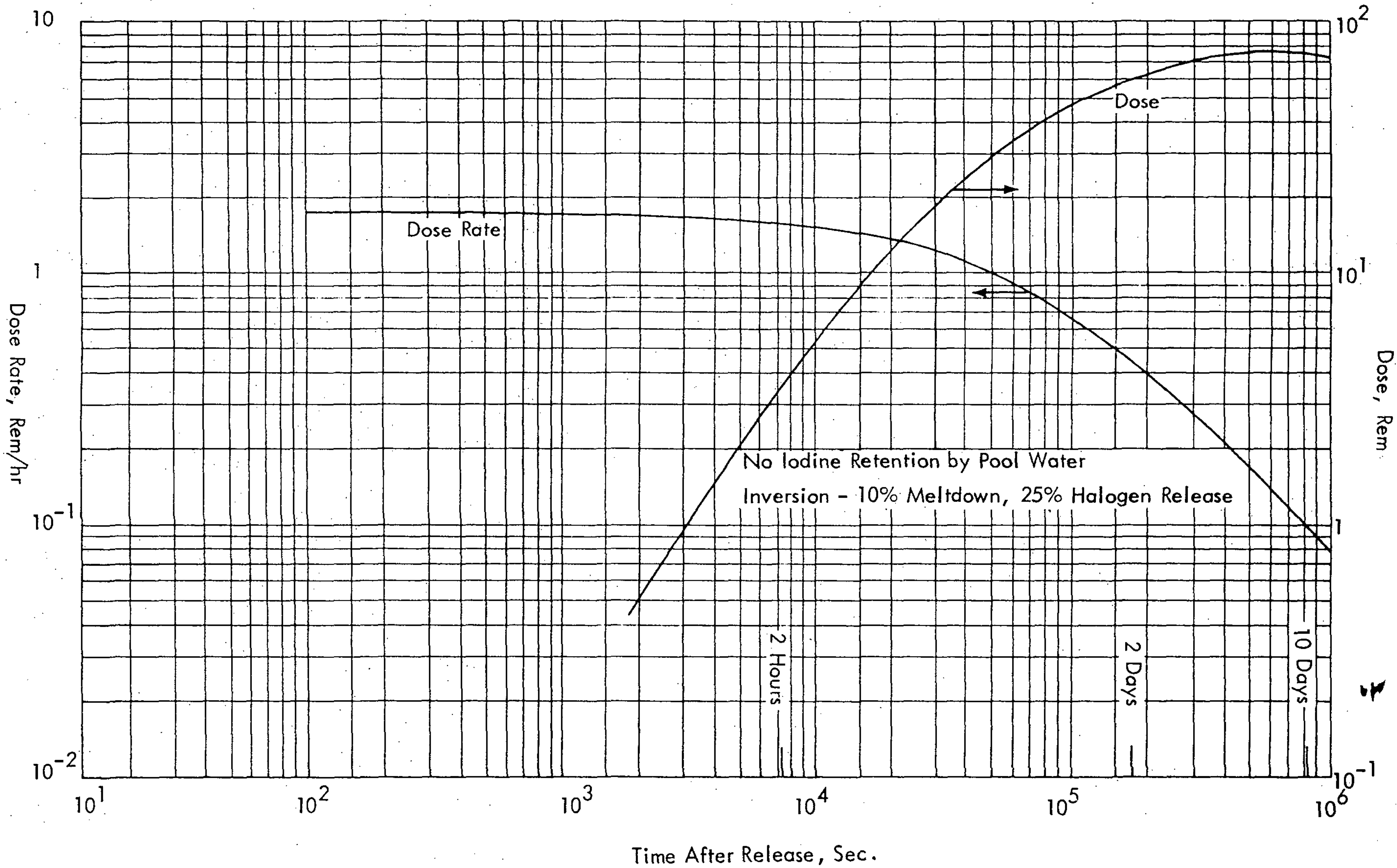

Figure 13.4 Thyroid Exposure at 1,000 Feet Due to Ingested Iodine

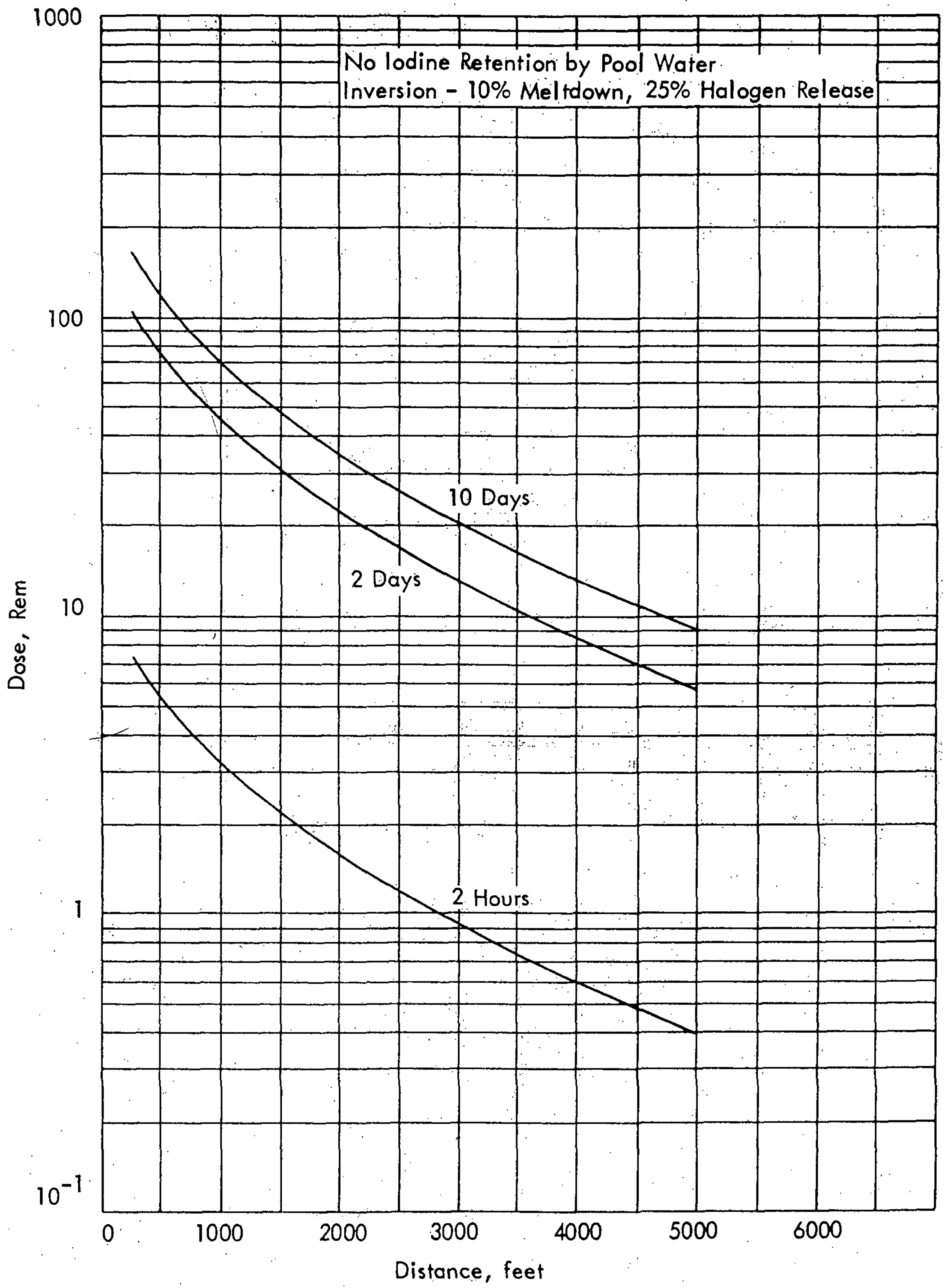

Figure 13.5 Thyroid Exposure