University of Lowell
One University Avenue
Lowell, Massachusetts 01854
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February 14, 1985

U.S. Nuclear Regulatory Commission
Director of Nuclear Reactor Regulation
Washington, D. C. 20555

Gentlemen:

The University of Lowell hereby requests that the University of Lowell reactor license, R-125, Docket 50-223, be amended so that the period of licensed operation be extended. Presently, the facility operating license R-125 expires April 20, 1985. The University of Lowell requests that this expiration date be extended for a period of thirty years to April, 2015. This will correspond to the total of 40 year expected lifetime of the facility.

In support of this license amendment, we are submitting information required by 10CFR50, 51, 55 and 73. Specifically, we have included the information requested by Mr. Cecil Thomas, Division of Licensing, USNRC. This information includes a) a Revised Safety Analysis Report, b) a Financial Statement from the University, c) an Environmental Impact Report, d) a copy of our Technical Specifications, e) a copy of our approved Operator Requalification Program, and f) a copy of our submitted revised Emergency Plan.

If you require additional information in support of this amendment, please contact Mr. Thomas Wallace directly at 452-5000 Ext. 2232.

Sincerely yours,

William T. Hogan

William T. Hogan, President
University of Lowell

WTH/TJW/dmm

Signed and sworn before me this 15 day of February, 1985.

Gladys M. Coughlin
, Notary Public

My Commission Expires: 11/25/88

UPDATED SAFETY ANALYSIS REPORT

for the University of Lowell Research Reactor

LICENSE NO. R-125
DOCKET NO. 50-223

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1.0 INTRODUCTION AND SUMMARY DESCRIPTION

1.1 INTRODUCTION

This report presents an updated description and safety evaluation of the University of Lowell open pool research reactor. It is prepared in support of an application to the United States Nuclear Regulatory Commission (USNRC) for a renewal of Reactor License R-125. The reactor is located at the University of Lowell which is situated in the northeastern part of Massachusetts in Lowell as shown in Figure 1.1.

1.2 GENERAL DESCRIPTION OF FACILITY

1.2.1 Facility History

In the late 1950's the decision was made to build a Nuclear Center at what was then Lowell Technological Institute (LTI). Its stated aim was to train and educate nuclear scientists, engineers and technicians, to serve as a multi-disciplinary research center for LTI and all New England academic institutes, to serve the Massachusetts business community and to lead the way in the economic revitilization of the Merrimack Valley. The decision was taken to supply a nuclear reactor and a Van de Graaff accelerator as the initial basic equipment.

Construction of the Center was started in the summer of 1966. Classrooms and offices were in use by 1970, and the Van de Graaff accelerator was put into service in that year. Reactor License R-125 was issued by the United States Atomic Energy Commission (USAEC) on December 24, 1974, and initial criticality was achieved in January, 1975.

The Lowell Technolgical Institute merged with Lowell State

College to form the University of Lowell during 1975. As a result, the reactor facility became the University of Lowell Reactor (ULR).

In the early 1980's, the name of the Nuclear Center was officially changed to the "Pinanski Building." The purpose was to reflect the change in emphasis of work at the center from strictly nuclear studies. At that time, the ULR became part of a newly established Radiation Laboratory. The Lab occupies the first floor of the Pinanksi Building and performs or coordinates research and educational studies in the fields of radiological sciences and nuclear engineering. The remaining two floors of the Pinanski Building are presently occupied by the Computer Science Department.

The Radiation Laboratory is a major research focal point of the University. More than 170 graduate students have used or are using the Laboratory's services; the comparable number for the faculty is in excess of 25. Much research is correlated with safety and efficiency in the nuclear and radiation industries, including public utilities, pharmaceuticals, medical applications, health effects, etc; however, much research also is done by workers in other fields who use the unique facilities as analytical tools.

In addition, the Laboratory's facilities are used in the course work of various departments of the University. It also provides these services to other Universities in the New England area, government agencies and, to a limited extent, industrial organizations in Massachusetts and the New England area.

Figure 1.1 N.E. USA Showing Lowell

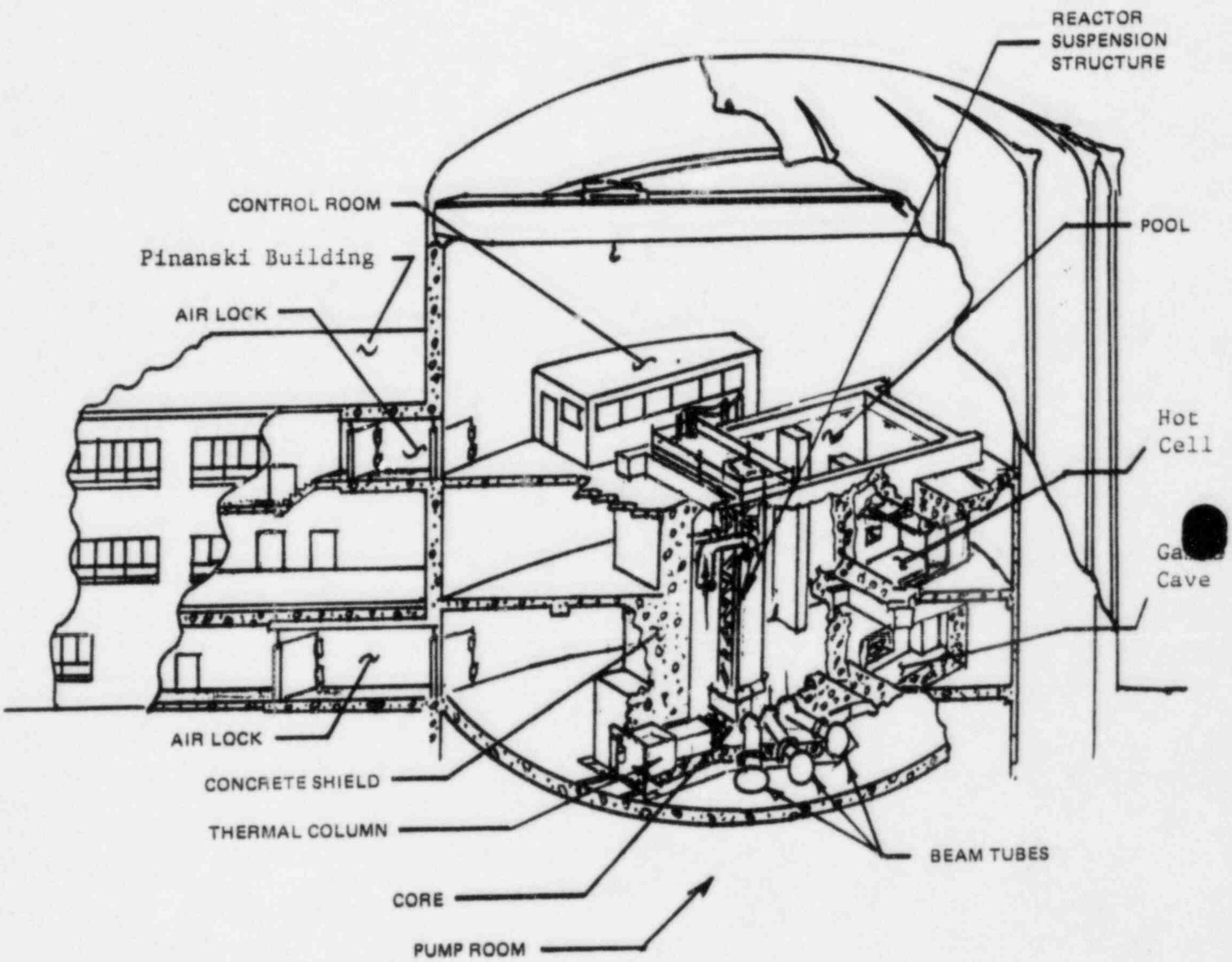


Figure 1.2. The University of Lowell Reactor (ULR)

1.2.2 Reactor

The open pool reactor (Figure 1.2) is a light water moderated and cooled, graphite and water reflected, heterogeneous reactor. The fuel is uranium enriched to 93% ^{235}U alloyed with aluminum to 24 wt% uranium. Major experimental facilities converge toward the core and afford opportunity for simultaneous performance of a number of different experiments.

The core consists of a 9 by 7 rectangular array of spaces capable of containing a 5 by 6 central array of fuel elements surrounded by a row of graphite reflector elements. Core loading is 26 fuel elements, although additional elements may be added to compensate for burnup. However, excess reactivity is limited to 4.7% $\Delta k/k$ (Facility Technical Specifications). Four safety blades and one servo-actuated regulating rod control core reactivity. The blades move vertically within a pair of shrouds extending the length of the core. Core elements are contained in a grid box, enclosed on four sides to confine the flow of cooling water between elements. The grid box and contents, as well as the drive mechanisms, are supported by a suspension frame from the reactor bridge.

A core loading approximately 3.3 kg of ^{235}U produces an average thermal flux of 8.6×10^{12} nV in the core at one megawatt. In the cold, clean condition, there is approximately 4.5% excess reactivity. The safety blades, because of their location and large surface area, have a present total shutdown worth of about 11.3% $\Delta k/k$.

Heat produced in the reactor is removed by natural convection at power levels below 100 kW, and by forced circulation above 100 kW. A double loop coolant system transfers heat from the reactor to atmosphere via the primary coolant system, heat exchanger, a secondary coolant system, and a cooling tower. Pool water make-up and clean-up systems maintain water purity at prescribed values.

The principal reactor design characteristics are tabulated below.

SUMMARY OF REACTOR DATA

Purpose of Reactor	Education and Research
Location of Reactor	Lowell, Massachusetts
Reactor Materials	
Fuel	Uranium (93% ^{235}U)
Moderator	High-purity water
Reflector	Graphite and water
Coolant	High-purity water
Control	Boral
Structural material	Aluminum
Shield	Concrete and water
Dimensions	
Core (active section)	15 in. x 18 in. (less 4 elements) x 24 in. high
Reflector	3 in. thick; 30 in. high
Grid box	9 x 7 array of 3 in. modules
Nuclear Characteristics at Rated Power	
Maximum thermal neutron flux (0.025→0.625 eV)	1.4×10^{13} nV
Average thermal neutron flux	8.6×10^{12}
Maximum epithermal neutron flux (0.625 eV → 5.3 keV)	8.0×10^{12}
Average epithermal neutron flux	5.0×10^{12}
Maximum fast neutron flux (above 5.53 keV)	1.8×10^{12}
Average fast neutron flux	1.2×10^{12}
Critical mass	2.8 kg U-235

SUMMARY OF REACTOR DATA (Continued)

Clean, cold core loading (25 1/2 elements)	3.44 kg U-235
Operating excess reactivity	4.5% $\Delta k/k$
Reactivity in safety blades (shutdown)	11.5% $\Delta k/k$
Temperature coefficient	$-0.88 \times 10^{-4} \Delta k/k/^{o}C$
Void coefficient (core average)	-2.2×10^{-3} $\Delta k/k/\%$ void
Prompt neutron lifetime	7.2×10^{-5} sec

Thermal Characteristics (Based on 26 Installed Elements)

Heat output	1 MW(th)
Hot Channel Factor	2.9
Maximum heat flux	21800 BTU/h-ft^2
Specific power (clean, cold)	285 Watts/gm U-235
Maximum gamma heat in core	0.86 Watts/cc
Coolant flow	1600 gpm
Maximum coolant velocity (in fuel)	3.1 ft/sec
Maximum water temperature (hot channel)	124^{o}F
Maximum fuel surface temperature (rated power)	156^{o}F
Water intake temperature (average bulk)	100^{o}F
Water outlet temperature (average bulk)	104.8^{o}F
Margin to nucleate boiling ($T_{superheat}-T_{surface}$)	86^{o}F
Primary water pressure at heat exchanger	$\sim$50 psig
Secondary water pressure at heat exchanger	$\sim$35 psig
Pressure drop through core	0.2 psi

Control

Safety elements	Four 10.6 inch wide vertical blades
Regulation element	One 2 1/2 inch square vertical rod
Composition	Boral (minimum 35 wt% Boron)
Withdrawal rate of safety blades	3 1/2 inches/minute
Withdrawal rate of regulating rod	78 inches/minute

Fuel

Type	Plate
Number of elements	26
Number of plates per element	18
Plate thickness	0.060 inches

Clad thickness	0.024 inches
Plate width	2.79 inches
Plate length	25 inches
Active fuel length	24 inches
Water gap	0.1 inches
Over-all element length	40 inches
Cladding	Aluminum
Fuel alloy	24 wt.% uranium-76 wt.% aluminum
^{235}U/element	135 ± 4 grams
Reflector	
Type	Graphite; 2.85 inches square, 30 inches long
Can	Aluminum; 0.040 inches thick
Number	24
Experimental Facilities	
Thermal column	One 4 x 4 x 8 ft graphite
Beam ports	Two 8-inch diameter Four 6-inch diameter
Pneumatic tube	Two 2-inch diameter
Radiation baskets	Thirty available
Gamma radiation facility	One dry 7 x 8 x 8 1/2 ft high
Hot cell	One 7 x 8 x 13 ft high
Medical facility (unused)	One 3 ft. square
Reactor Pool	
Stall pool	8 x 16 x 31 feet
Bulk irradiation pool	12 x 6 x 31 feet
Reactivity Requirements ($\Delta k/k$) 1 MWt	
Xenon (equilibrium)	1,7%
Temperature	0.2%
Burnup, experiments, buildup of fission products	2.6%
Total	4.5%

SUMMARY OF REACTOR DATA (Continued)

Design and Operating Characteristics

Void coefficient	Negative
High flux scram limit	125% rated
Maximum operating excess reactivity	4.5% Δk/k
Maximum worth of regulating rod (with aluminum side adjacent to core)	<0.7% Δk/k
Minimum worth of a control blade	3.0% Δk/k
Maximum worth of a control blade	4.1% Δk/k
Total worth of 4 control blades	11.5% Δk/k
Maximum allowable reactivity to be inserted instantaneously	0.5% Δk/k
Startup count rate, minimum	2 counts/second

1.2.3 Auxiliary Systems and Radioactive Waste Management

A 70 kW natural gas fueled motor generator starts automatically upon loss of power. This feeds reactor instrumentation, health physics monitoring and alarm systems, emergency lighting, and other safety oriented components of the facility.

Fuel is stored in the pool in racks designed to maintain a planer array of the elements. This geometry is sub-critical under total water flooding.

The relatively small amounts of liquid radioactive waste are mostly the result of regeneration of ion exchange resins used to cleanup the primary water. The effluent from the cleanup demineralizer goes to a 3000 gallon sump in the pump room of the reactor, and is transferred at appropriate times to storage tanks in the Pinanski Building basement, where it is controlled with no release to the environment until the appropriate analysis is performed to assure conformity with applicable standards.

The gaseous radioactive waste product produced in largest quantity is ^{41}Ar; annual integrated doses, based on stack emitted ^{41}Ar, and an expanding balloon model for operation at one megawatt, have been calculated. Fission product releases from the containment during routine operation will be extremely small and likely not detectable on a routine basis by continuous stack monitoring techniques because of the higher level of the gaseous activity, ^{41}Ar, and the higher naturally occurring particulate activities arising from the decay of ^{222}Rn and ^{220}Rn.

Gaseous radioactivity monitoring is done by an extensive system including a constant air monitor at the pool level and another on the beam floor, as well as a stack gas effluent monitor using beta scintillation detectors to monitor filter tape activities for radioactive particulates in conjunction with shielded gas cylinder containing a twelve inch GM tube for detecting radioactive gases. For routine operations with the exception of ^{41}Ar, all radioactive gaseous releases will be emitted at concentrations orders of magnitude below MPC's.

The amount of solid wastes generated as a result of reactor operations is small. Spent resins from the cleanup demineralizer are drained of liquid, bagged in plastic, and loaded into D.O.T. approved metal drums for pickup by a waste disposal contractor after short-lived activities (e.g., ^{24}Na) have been allowed to decay.

Some solid waste in the form of filters result from normal gaseous waste handling from the various facilities of the reactor such as the hot cell, pneumatic tube exhaust, beamport vent line, etc. Additional solid wastes are produced as a result of experimental and maintenance operations.

Waste from all sources produce five to ten 55 gallon drums per year with the average activity per drum less than one millicurie. The D.O.T. approved drums are turned over to an authorized waste disposal contractor.

1.3 COMPARISON TABLES

1.3.1 Comparisons with Similar Facility Designs

Table 1.1 gives the characteristics of some reactor facilities which have been selected because of their similarity to the ULR. It is worth noting that the ULR has containment capability which is not necessarily requisite for safe operation (as shown by years of operating experience for reactors of this type).

1.3.2 Comparisons of FSAR and Present Designs

The most significant change to the facility since the submittal of the Final Safety Analysis Report (FSAR) was made to the forced convection flow path. The coolant path was changed from the "downcomer" to the "cross-pool" flow mode to eliminate control blade oscillation. A detailed safety analysis of cross pool flow during forced convection operation is included in Chapter 9 of this updated safety analysis report.

1.4. IDENTIFICATION OF AGENTS AND CONTRACTORS

The ULR was designed, manufactured, and installed by the General Electric Company. The building which houses the reactor was designed by the architectural firm of W. Chester Brown Associates in coordination with LTI personnel. The prime contractor for the reactor building was the Wexler Construction Company, Inc., and the welded steel containment shell was furnished, erected and leak tested by the Chicage Bridge and Iron Company.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

A need for further technical information is not required for operation at 1 MW. Should an application for a license to operate at some power greater than 1 MW be desirable at a future date, it would be necessary to submit a new safety analysis for the new power using, insofar as practical, measured input information.

1.6 MATERIAL INCORPORATED BY REFERENCE

Chapter 4:

Reference 1: Hackney, M. R., "Physics Analysis of the Lowell Swimming Pool Reactor," General Electric Co., GECR-4365 (1963).

Chapter 6:

Reference 1: "Application for License for Storage of University of Lowell Reactor Fuel Elements," submitted to USAEC on October 20, 1970.

Chapter 7:

Reference in body of Section 7.3.8: "Environmental Significance of the Projected 5 MW ^{41}Ar Release Rate of 400 $\mu Ci\ sec^{-1}$ from the Lowell Technological Institute Reactor Facility," submitted to the USAEC on September 8, 1970.

Reference 1: Letter dated January 3, 1972, from Rhode Island Nuclear Science Center to USAEC, Assistant

Director for Reactor Operations.

Chapter 8:

Reference 2: Dady, C. E., and Woodward, G. G.,"Health Physics Survey of Five Megawatt Operation at the AMMRC Nuclear Reactor," Internal Report, Army Materials and Research Center, October, 1969.

Chapter 9:

Reference in body of Section 9.2.1.2 and Table 9.9, "Environmental Significance of the Projected 5 MW ^{41}Ar Release Rate of 400 μCi sec^{-1} from the Lowell Technological Institute Reactor Facility," submitted to the USAEC on September 8, 1970.

Reference 2: Hackney, M. R., "Physics Analysis of the Lowell Swimming Pool Reactor," General Electric Co., GECR-4365 (1963).

Reference 10: Answer to Question 14, "Responses to Questions Contained in a Letter from the U.S. Atomic Energy Commission," September 14, 1964.

Reference 15: University of Wisconsin Nuclear Reactor, Docket No. 50-156, License R-74, Amendment 4, dated May, 1963.

Reference 16: Hunt, C. H., and DeBevec, C. J., "Effects of Pool Reactor Accidents," General Electric, Atomic Power Equipment Department, GEAP-3277, November, 1959.

COMP

Facility	Power Rating Max. Thermal Flux	Fuel Type
ULR	1 MW 1.4×10^{13}	Plate type fuel 18 plates/element 0.060 in. thick 0.024 in. clad (Al) 2.79 in. wide, 24 in. lon 0.1 in. water gap 93% enriched 24 w/o U-Al
Rhode Island Open Pool Reactor	1 MW 1.2×10^{13}	Plate type fuel 18 plates/element 0.060 in. thick 0.015 in. clad (Al) 2.79 in. wide, 24 in. lon 0.1 in. water gap 93% enriched 30 w/o U-Al
Batelle Research Reactor (Decommissioned 1975)	1 MW 1.4×10^{13}	MTR type - 18 plate 0.02 in. thick Al clad 3.16 x 2.996 x 24 in. hig 18 x 27 x 24 in. core siz
Ford Nuclear Reactor University of Mich.	2 MW 1×10^{13}	BSF type - 18 plates 14.1% U-Al Al clad 0.02 in.
Naval Research Laboratory Reactor (Decommissioned 1970)	100 - 1000 kW	18 plate MTR type. Core size 21 x 21 x 24 in. high
Pennsylvania State University Reactor	100 kW 0.7×10^{12}	MTR type-10 plates 14.1% U-Al (90% ^{235}U) 0.02 in. Al clad 24 in. long
Industrial Reactor Laboratory (Decommissioned 1975)	5 MW $>2 \times 10^{13}$	MTR type-18 plates (only 16 are fuel) 93% enriched ~190 grams per element

TABLE 1.1

ARISON WITH SIMILAR FACILITIES

	Reflector	Control	Coolant	Containment	Year	Remarks
; (meat)	H_2O Graphite	10.6x0.375 in. wide safeties 1-2 1/2 in. square Reg. Rod. Boral	H_2O Forced and Natural Convection 1400 gpm	Yes	1973	Pool type, stall pool 8 x 16 x 31 ft deep Bulk pool: 12 x 16 x 31 ft deep
g (meat)	H_2O Graphite	10.6x0.375 in. wide safeties 1-2 1/2 in. square Reg. Rod. Boral	H_2O Forced and Natural Convection 1500 gpm	No	1962	Pool type High Power Section 8 1/2 ft diam x 32 ft deep Low Power Section 7 x 7 1/2 x 30 ft deep Fuel Storage Section 7 x 5 x 30 ft deep
h e	H_2O Graphite	4 B_4C safeties & shims 1-ss Reg. Rod.	H_2O-Forced Circulation 900 gpm 75 to 85°F coolant	No	1956	Max clad temp. 147°F
	H_2O Graphite	3-7/8 x 2 1/4 in. B_4C-Pb shims -1 ss Reg. Rod	H_2O-Nat. & Forced Convection	No	1956	Pool 10 x 27 x 27 ft deep
	H_2O Graphite	3 Boron safeties -1 ss Reg. Rod	H_2O-Nat. Convection	No	1956	Pool type 28 1/2 ft deep
	H_2O Graphite	3 B_4C safety -1 ss Reg. Rod	H_2O-Nat. Convection	No	1955	Pool - 15 x 30 x 21 ft deep
	H_2O	5 Boron Safeties -1 ss Reg. Rod	H_2O Forced & Natural Convection 2500 gpm	Yes	1958	Pool capacity 135,000 Gallons

2.0 SITE AND METEOROLOGY

2.1 GENERAL LOCATION

The reactor is located on the North Campus of the University of Lowell, in the city of Lowell, Middlesex County, Massachusetts (see Figures 2.1 and 2.2). The North Campus of the University of Lowell is presently some 60 acres in size and is mainly situated just north of the Merrimack River, although several dormitories and a Student Union building lie south of the river. The entire complex is near the northern edge of the city, and this region of the river is known as Pawtucket Falls.

The reactor is a major facility of the Radiation Laboratory, which, in addition to the reactor, includes a 5.5 MeV Van de Graaff accelerator, laboratories, classrooms, and staff offices. The Laboratory is part of the Pinanski Building which is located south and east of existing classroom buildings, and west of the gymnasium (see Figure 2.3).

The nearest public thoroughfare to the reactor site is the Veterans of Foreign Wars Highway at a distance of 45 meters, but campus access is from Riverside Street or University Avenue (see Figure 2.3). The distance from the reactor to the nearest extant dwelling is about 200 meters as shown in Figure 2.3.

2.2 POPULATION AND ACTIVITIES IN SURROUNDING AREA

Land use in the immediate area surrounding the reactor is primarily residential, with some industry southeast of the campus along the river. The population of Lowell at the 1980 census was 92,418, a decrease of 1,821 from the population registered in 1970 census (see Table 2.1). The city of Lowell is part of the Lowell

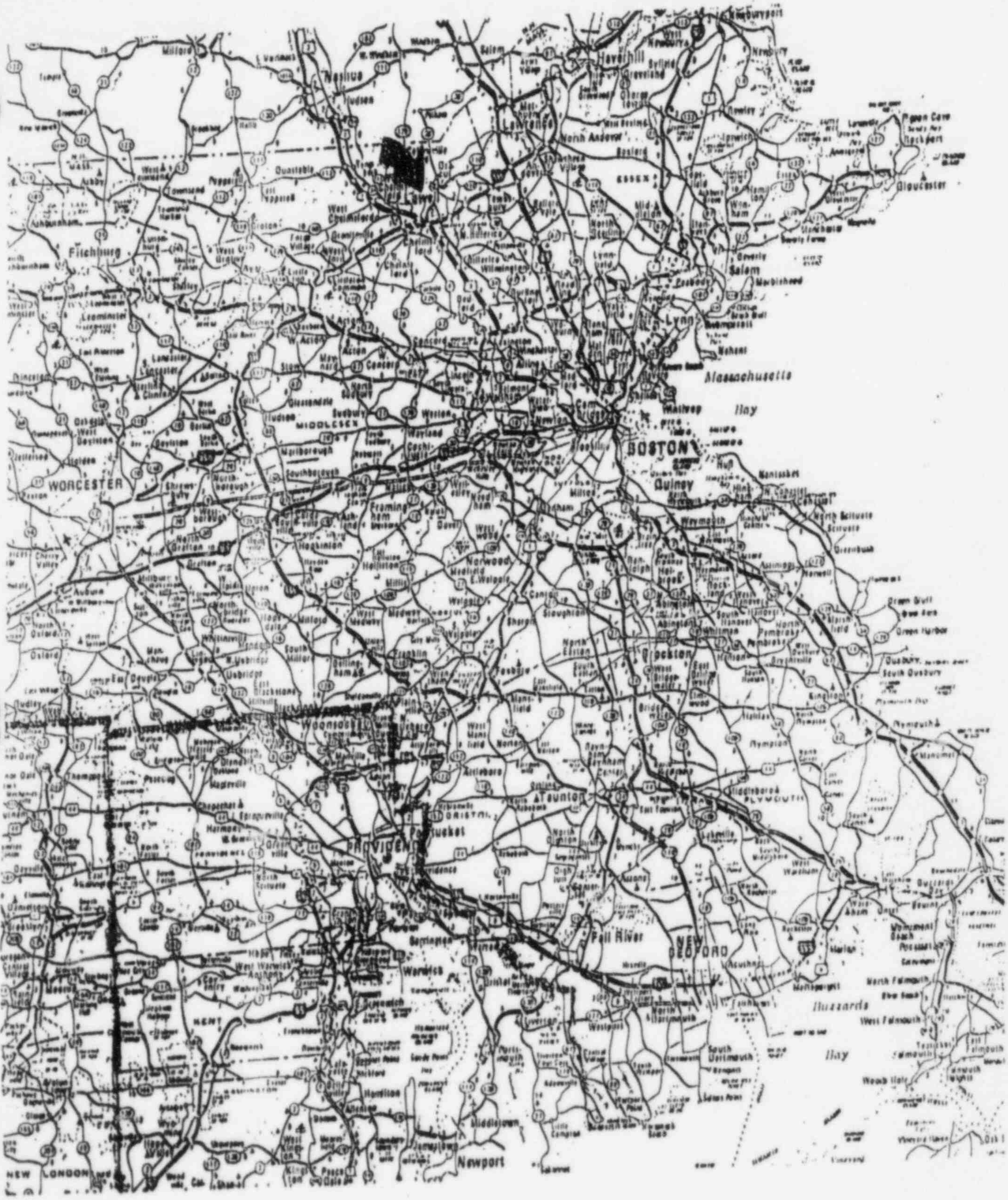


Figure 2.1 Map of Eastern Massachusetts

Figure 2.2 Map of Lowell Area

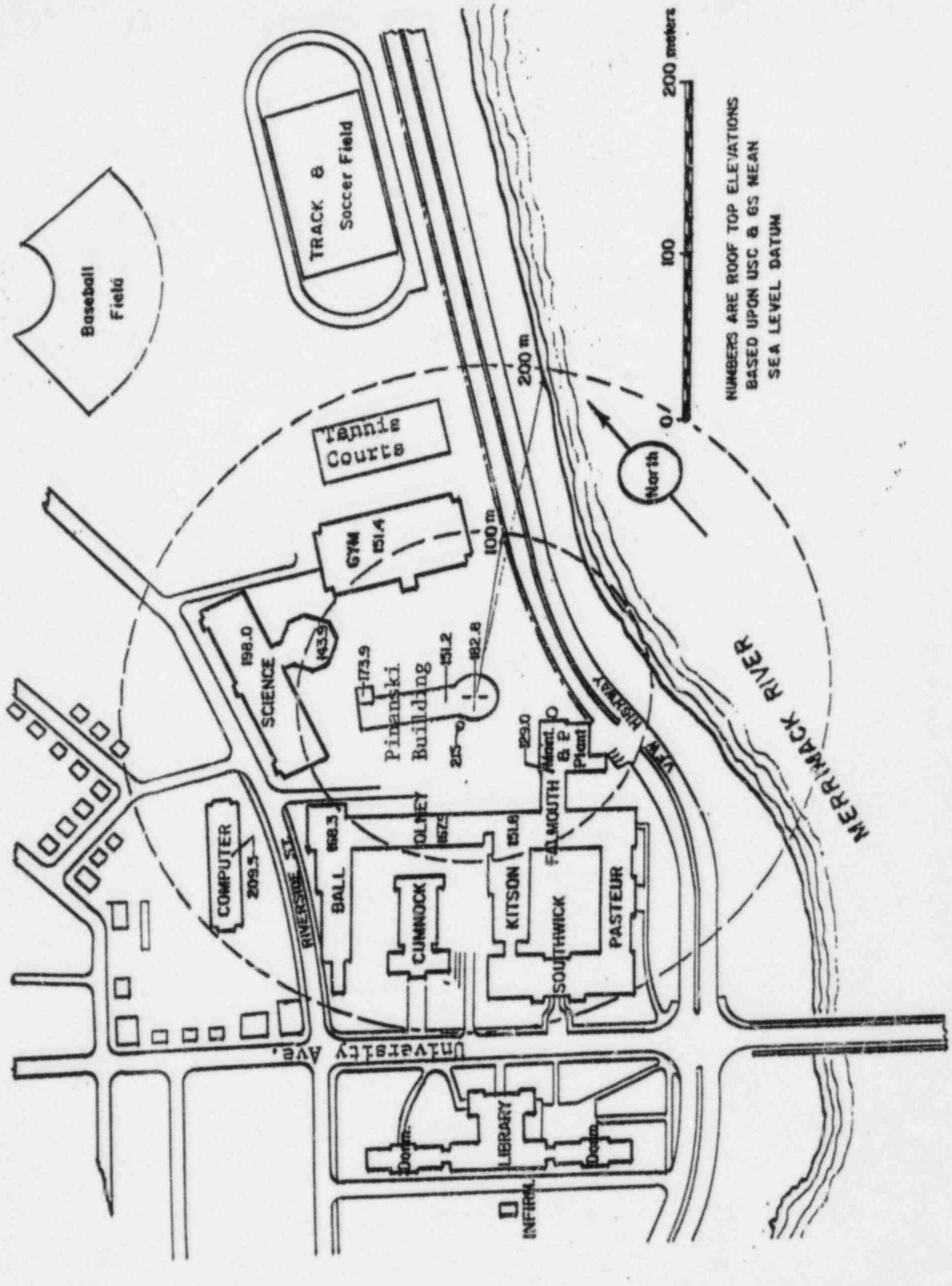


Figure 2.3 Site Map with Building Heights and Reference Radii

TABLE 2.1

Population in the Lowell Area

Town	Tract	Population 1970	Population 1980	Numerical Change	Percentage Change
Lowell	3101	859	1,902	1,043	121.4
	3102	6,117	5,873	- 244	- 39.9
	3103	5,927	5,463	- 364	- 62.5
	3104	3,604	3,233	- 371	- 10.3
	3105	3,436	3,236	- 200	- 5.8
	3106	7,131	9,012	1,881	26.4
	3107	3,825	3,864	39	1.0
	3108	1,759	2,523	764	43.4
	3109	886	794	- 92	- 10.4
	3110	1,438	1,169	- 269	- 18.7
	3111	2,742	2,008	- 734	- 26.8
	3112	3,257	2,839	- 418	- 12.8
	3113	3,929	3,581	- 348	- 8.9
	3114	3,918	4,782	868	22.2
	3115	2,847	2,664	- 183	- 6.4
	3116	5,318	5,020	- 298	- 5.6
	3117	4,327	3,897	- 430	- 9.9
	3118	3,625	2,854	- 771	- 21.3
	3119	4,075	2,507	-1,568	- 38.5
	3120	2,445	3,145	700	28.6
	3121	2,592	2,495	- 97	- 3.7
	3122	4,510	4,165	- 345	- 7.6
	3123	4,264	3,391	- 873	- 20.5
	3124	2,570	2,109	461	17.9
	3125	8,735	8,895	160	1.8
Lowell Total 3101 to 3125		94,239	92,418	-1,821	- 1.9
Tyngsboro	3131	4,204	5,683	1,479	35.2

TABLE 2.1 (Continued)

Town	Tract	Population 1970	Population 1980	Numerical Change	Percentage Change
Dracut	3141	8,003	9,035	1,032	12.9
	3142	5,142	4,797	- 345	- 6.7
	3143	5,069	7,417	2,348	46.3
Dracut Total		18,214	21,249	3,035	16.7
Tewksbury	3151	3,429	3,952	523	15.3
	3152	4,628	5,788	1,160	25.1
	3153	1,259	1,228	- 31	- 2.5
	3154	5,288	5,616	328	6.2
	3155	8,151	8,051	- 100	- 1.2
Tewksbury Total		22,755	24,635	1,880	8.3
Billerica	3161	8,967	11,350	2,383	26.6
	3162	7,423	7,640	217	2.9
	3163	3,713	5,866	2,153	58
	3164	5,198	5,536	338	6.5
	3165	6,347	6,335	- 12	- .2
Billerica Total		31,648	36,727	5,079	5.9
Chelmsford	3171	13,815	14,180	365	2.6
	3172	11,904	11,423	- 481	- 4
	3173	5,713	5,571	- 142	- 2.5
Chelmsford Total		31,432	31,174	- 258	- .8
Westford	3181	4,179	4,856	677	16.2
	3182	2,585	2,599	14	.5
	3183	2,464	5,979	2,375	66
	3184	1,140	5,979	2,375	66
	3185	3,604	5,979	2,375	66
Westford Total		10,368	13,434	3,066	29.5
SMSA Total		114,417	127,219	12,802	6.9

Standard Metropolitan Statistical Area (SMSA) which includes Tyngsboro, Dracut, Chelmsford, Tewksbury, Billerica and Westford (see Figures 2.4 and 2.5). Lowell itself has had a population decrease of over 1.9% in the past ten-year period (1970-1980) while the Lowell SMSA has increased 6.9% in the same period.

2.3 TOPOGRAPHY

The U.S. Geological Survey quadrangle map for Lowell and immediate area is shown in Figure 2.6. The average elevation at the site is 112 feet MSL and the area generally slopes toward the east and north and consists of gently rolling hills whose elevation is 200-350 feet MSL, with numerous ponds in the lower areas. The river valley is quite narrow and rather shallow in most areas and winds through the city in an eastward direction. No abrupt topographical features exist in the 0-5-mile range from the site.

2.4 METEOROLOGY

Temperature and precipitation data for Lowell are listed in Table 2.2. The average annual temperature is 50°F with a maximum of 103°F and a minimum of -29°F, although recent extremes are less severe. The average annual precipitation is 43.3 inches.

Wind speed and direction data taken by the Health Physics Department at ULR for a period of one year were reduced into averages as shown in Table 2.3. In general, the prevailing winds are westerly although the effects of occasional northeasters are discernible. Wind speeds of 10 mph or higher are seen to occur better than twenty five percent of the time, while calm conditions

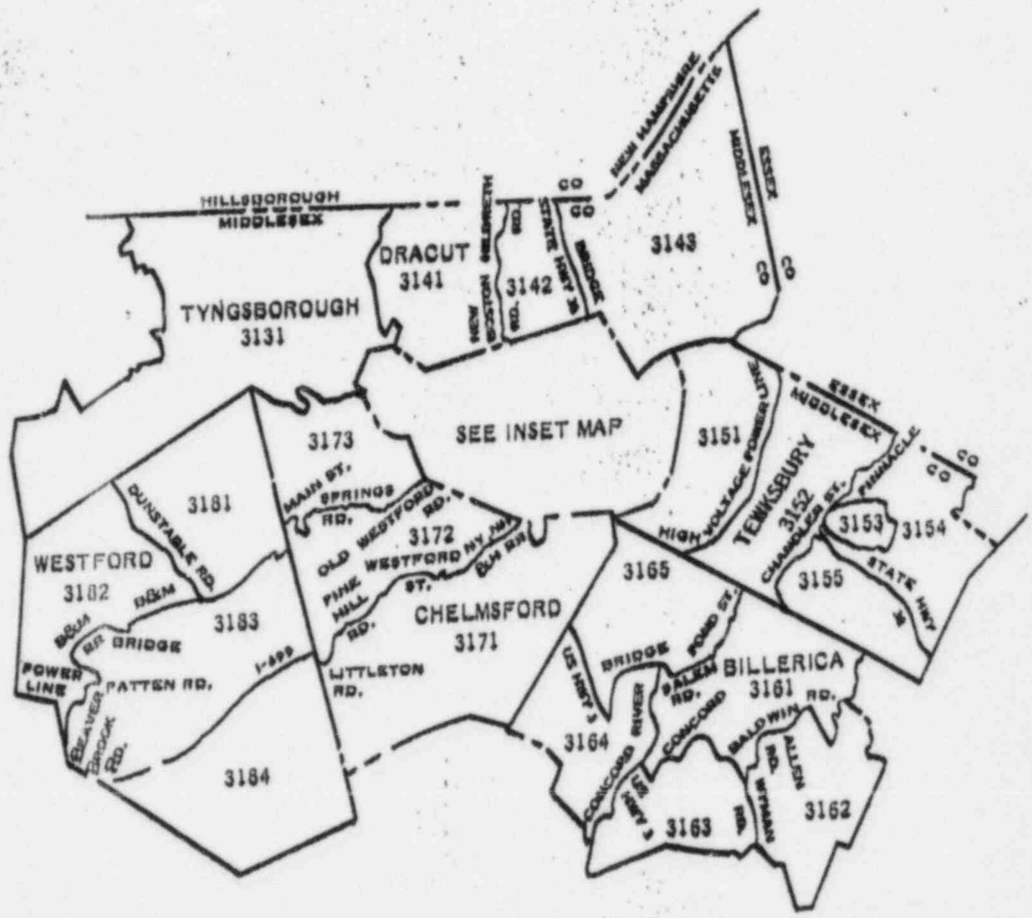


Figure 2.4 Census Tract Map of the Lowell Standard Metropolitan Statistical Area

CENSUS TRACK IN THE LOWELL, MASS, SMSA INSET MAP—LOWELL

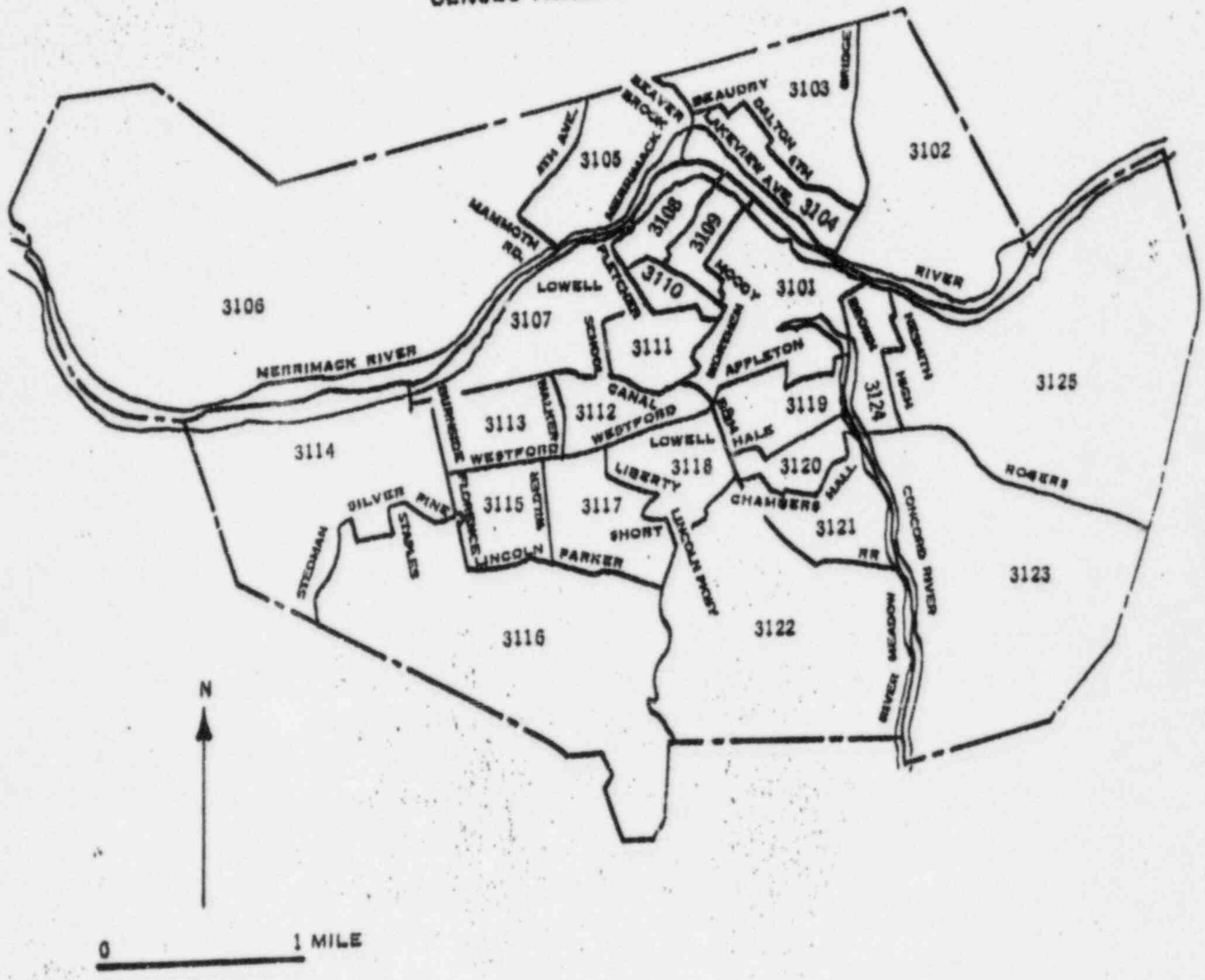


Figure 2.5 Census Tract Map of Lowell

Figure 2.6 Topographical Map of the Lowell Area

TABLE 2.2

SUMMARY OF LOWELL WEATHER CONDITIONS

CONDITION	SURVEY TIME SPAN			
	1957-63	1950-60	1896-1970	1931-60
Maximum Temperature	97°F	102°F	103°F	
Minimum Temperature	-15°F	-15°F	-29°F	
Maximum Rainfall in 24 h period	4.8 in.			
Maximum Snowfall in 24 h period	19.5 in.			
Average Annual Temperature	49.8°F	51.2°F	49°F	50°F
Average Annual Precipitation	43.4 in.			43.34 in.

*Source: U.S. Weather Bureau, Boston, Mass.

TABLE 2.3

Wind Speed and Direction Summary

Univeristy of Lowell Nuclear Center

January 1984 - December 1984

Wind Speed Freqency (%)

Direction	Calm	1-4 mph	5-9 mph	10-15 mph	15 mph	Total
N		2.91	3.07	1.73	0.33	8.04
NNE		1.58	2.44	0.90	0.24	5.16
NE		1.52	2.88	1.47	0.41	6.28
ENE		0.46	1.36	1.30	0.24	3.36
E		0.73	1.55	0.73	0.49	3.50
ESE		0.14	0.65	0.30	0.30	1.39
SE		0.52	0.95	0.65	0.14	2.26
SSE		0.81	1.36	0.60	0.30	3.07
S		1.09	1.36	0.35		2.80
SSW		0.22	0.16	0.03		0.41
SW		0.76	0.71	0.38		1.85
WSW		0.43	0.84	0.81	0.05	2.13
W		3.91	4.27	2.04	0.35	10.57
WNW		1.60	2.64	1.58	0.52	6.34
NW		8.67	11.27	4.84	1.85	26.63
NNW		2.91	4.62	2.47	0.69	10.69
	5.52	28.26	40.13	20.18	5.91	94.48
						5.52
						100.00

(less than 1 mph) occur approximately five percent of the time. The data are presented in wind rose from in Figure 2.7.

Hosler[1] has estimated the inversion frequency to average about 30% of the year with a maximum of 60%. This correlates well with the sum of the calm and 1-4 mph ULR data, which add up to 33.28% of the year.

2.5 GEOLOGY AND SEISMOLOGY

The site as described by geologists from Haley and Aldrich Inc.,[2] is underlain by quartzite whose color varies from gray through greenish-gray to brown, the latter color being imparted by minute particles of red biotite. Thin layers of slaty quartzite and quartz schist also exist. This underlayment is known as the Merrimack quartzite, and is, in the opinion of the consulted geologists, essentially sound, free from weathering, with only minor jointing. In the opinion of Haley and Aldrich, the bedrock has an allowable bearing pressure in excess of 20 tons per square foot, which is about four or five times the maximum bearing pressure beneath the foundation mat for the reactor containment building. Differential settlement of the structure under these conditions will be insignificant. The rock is hard, chemically stable, and insoluble.

There is some likelihood that the area is part of a fault zone, particularly where the quartzite meets the Ayer Granite underlayment, northwest of the site. In any case, Haley and Aldrich state that a search of the literature indicates that if major faulting does occur in the area, it has not given rise to significant earthquake activity. The area has been subjected to a

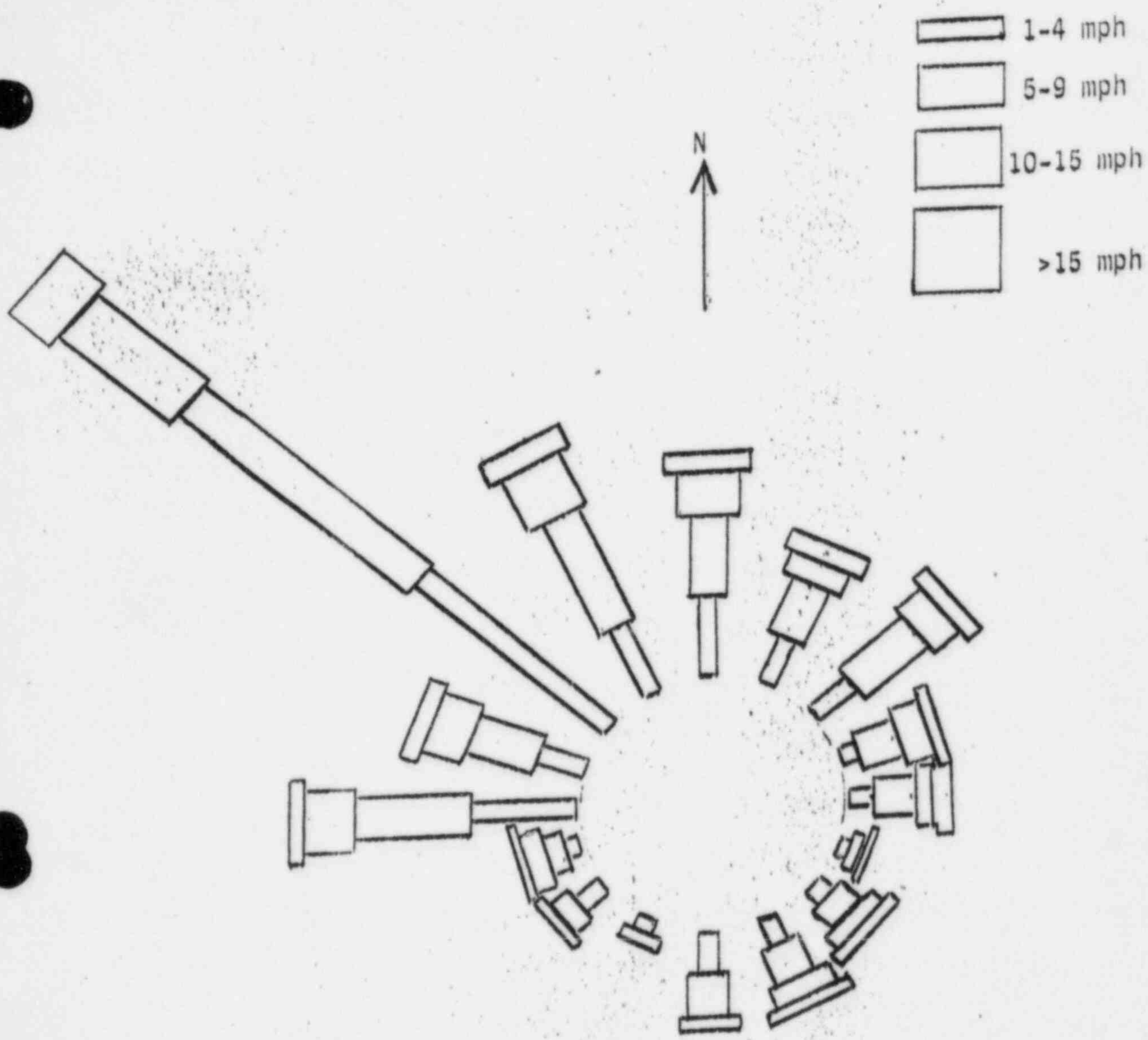


Figure 2.7 University of Lowell Wind Rose for January 1984 - December 1984

few minor earthquakes with intensities ranging from IV to VI on the Modified Mercalli Scale.

The Area Zoning Map of the Uniform Building Code[3] shows the Lowell area included in Seismic Probability Zone 2, in which the Code recommends designing to withstand shocks equivalent to intensity VII on the Modified Mercalli Scale. The center of intensity VII corresponds to an acceleration of 0.1 g[4], and the reactor building foundation, the pool, the reinforced concrete parts of the reactor building, and the steel containment have been designed to withstand 0.1 g acceleration.

2.6 HYDROLOGY

Drainage in the Pawtucket Falls area is directly toward the Merrimack River. Average and minimum flow rates of the river measured at the Lowell Gauging Station between 1925 and 1976 were 6,540 cfs and 181 cfs,[5] respectively. During the record flood of 1936, the flow rate was 157,430 cfs.

The U.S. Army Engineer Division, New England, of the Corps of Engineers[6] has tabulated flood data for the 1936 flood and for the second largest event in September 1938 for locations near the University. These data are reproduced below along with some reactor building elevations (see Figure 3.3 of Section 3).

Location	Elevation in feet above mean sea level	
Upsteam from Moody Street Bridge	89.6	(1936 flood)
	84.2	(1938 flood)
Downstream from Moody Sreet Bridge	82.5	(1936 flood)

Location	Elevation in feet above mean sea level	
(about 200 ft. downsteam of section opposite the reactor)	76.5	(1938 flood)
Reactor Building	113.7	(ground)
	98	(basement)

After the floods of the thirties, the Corps of Engineers completed four flood control reservoirs in New Hampshire to reduce the flood potential of the Merrimack River. The effect of the controls is estimated to be equivalent to a reduction of the 1936 flood level by 6.5 feet and the 1938 level by 5 feet in the reactor site region.

The corps developed a Standard Project Flood which is a synthetic flood reflecting the storm and runoff potential of a river basin. For Lowell, this synthetic flood would cause stages about 3 feet below the 1936 flood levels. Since the basement of the reactor building is some 18 feet above the potential flood level, it is considered that there is no flood risk involved in the site.

REFERENCES

1. Hosler, C.R., Low Level Inversion Frequency in the Contiguous United States, Monthly Weather Review, September, 1961.
2. Haley & Aldrich, Inc., Consulting Soil Engineers, Cambridge, Mass. 02142.
3. Seelye, E.E., Foundations, Design and Practice, pp.3-100, John Wiley & Sons, New York, 1956.
4. Nuclear Reactors and Earthquakes, Chapter 1, TID-7024, USAEC, Washington, 1963.
5. Application for License: Project No. 2790, Lowell

Hydroelectric Project, May 1980.

6. John Wm. Leslie, Chief, Engineering Division, Waltham, Mass., by letter dated October 27, 1964.

3.0 CONTAINMENT

3.1 CONTAINMENT SYSTEM STRUCTURE

3.1.1 Containment System Structure Design

The containment building is a welded steel shell with a flat bottom, cylindrical sides, and a domed top. The structural specifications meet the design criteria listed below.

(1) Design pressure: Internal, 2 lb in.$^{-2}$; External, 0.2 lb in.$^{-2}$

(2) Design internal volume: $(375 \pm 5) \times 10^3$ ft^3

(3) Design internal temperature: 70 ± 3°F

(4) Leak rate: No more than 10% per 24 hr at 2 psig

(5) Roof live load: 40 lb ft^{-2}

(6) Wind load: 20 lb ft^{-2}

(7) Dead load: 150 lb ft^{-2}

(8) Earthquake load: Intensity VII on the Modified Mercalli scale.

(9) Design stresses: In accordance with the ASME Boiler and Pressure Vessel Code, Section 8, "Rules for Construction of Unfired Pressure Vessels."

(10) Design analysis: Based on elastic analysis.

The flat bottom of the shell is lined with two and one half feet of poured concrete and the cylindrical walls are lined with two feet of poured concrete to serve as a ballistics and radiation shield, and to support a fifteen ton polar crane. The inside clear diameter is nominally eighty feet. From the flat steel bottom to the highest point of the domed top the distance is about 95 1/2 ft, of which the lower 28 ft are below grade. The domed or ceiling portion is insulated with two inches of fiberglass held in place by stud welded pins and speed washers, and sealed on the underside with a finish coat of white lagging adhesive to provide a continuous vapor and dust barrier. The outside of the shell is painted with a red lead undercoat and a weather resistant finish coat. A beam level plan view and an elevation view are shown in Figures 3.1 and 3.2.

Beneath the flat steel bottom, a concrete pad rests on the underlying rock which, in turn, is a light gray quartzite. Core borings were taken for ten feet into the rock by the Atlantic Test Boring Company and analyzed by soil engineers.* These tests indicate some jointing, but excellent bearing capacity is concluded by the engineers. The rock slopes generally downward (Figure 3.3) to the Merrimack River bed, with an average slope of 0.086. The containment building foundation is firmly keyed to the bedrock.

The prime contractor for the reactor building, including excavation and foundation work, was the Wexler Construction Company, Inc. The welded steel containment shell was furnished and erected by the Chicago Bridge & Iron Company.

3.1.2 Penetrations

A general schematic drawing of major penetrations is shown in Figure 3.4.

3.1.2.1 Doors. Normal access to the building is through a set of ground floor or beam level airlock doors. A second set, normally used only by operating personnel but available for any emergency

*Haley and Aldrich, Inc., Cambridge, Mass. 02142. See also Chapter 2 of this document.

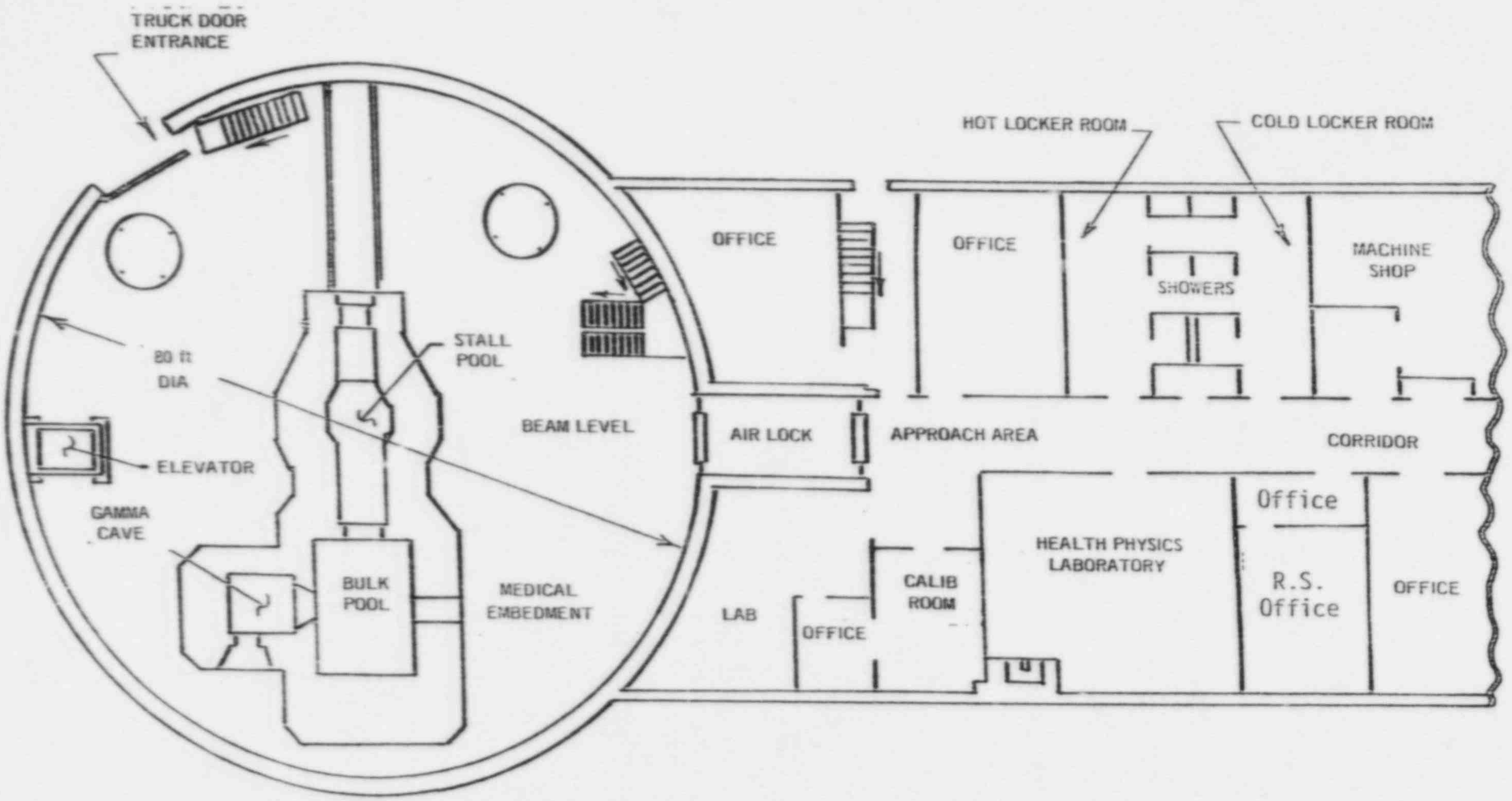


Figure 3.1. Plan View of the Pinanski Building

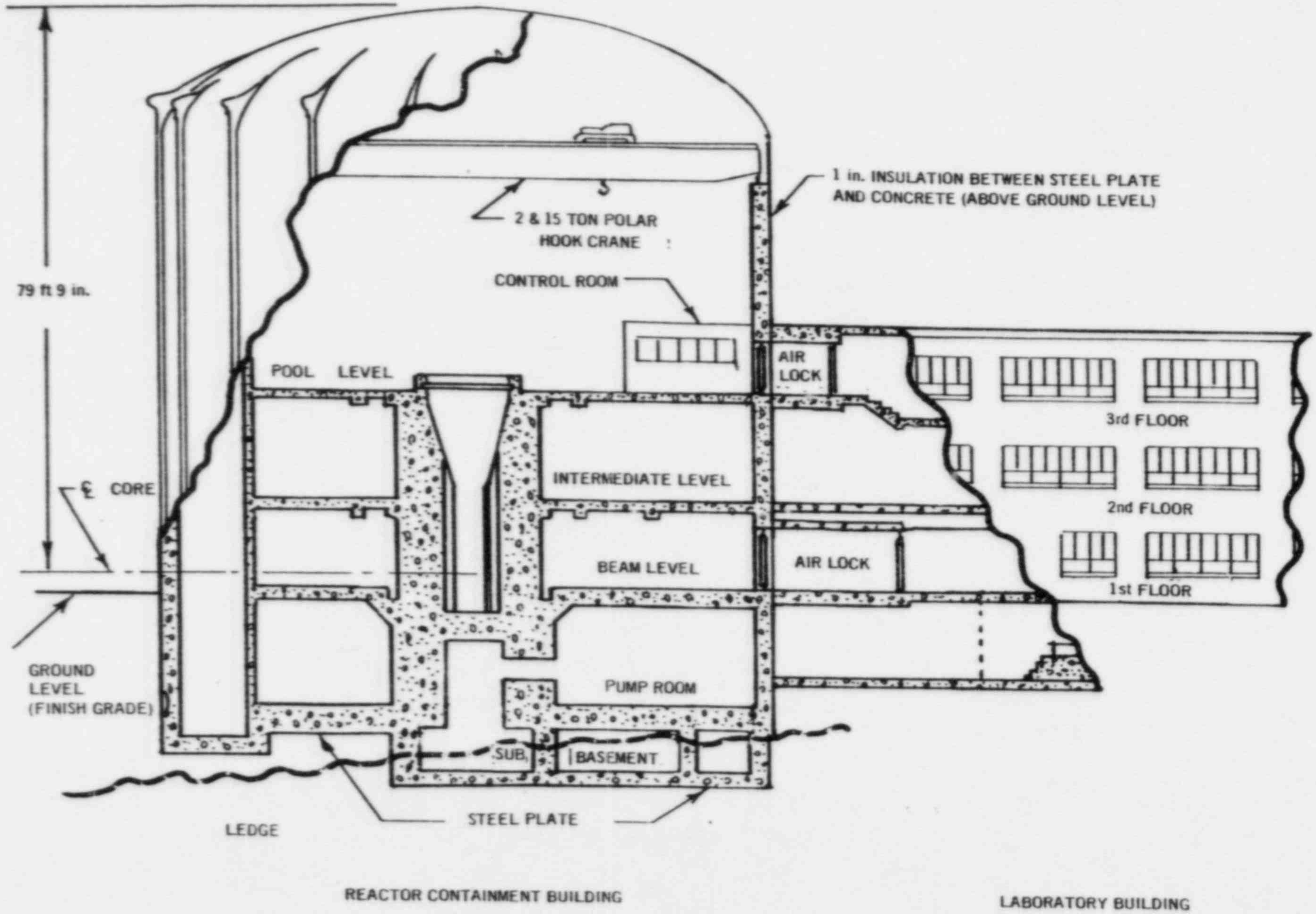


Figure 3.2. Elevation View of the UL Reactor and Pinanski Building

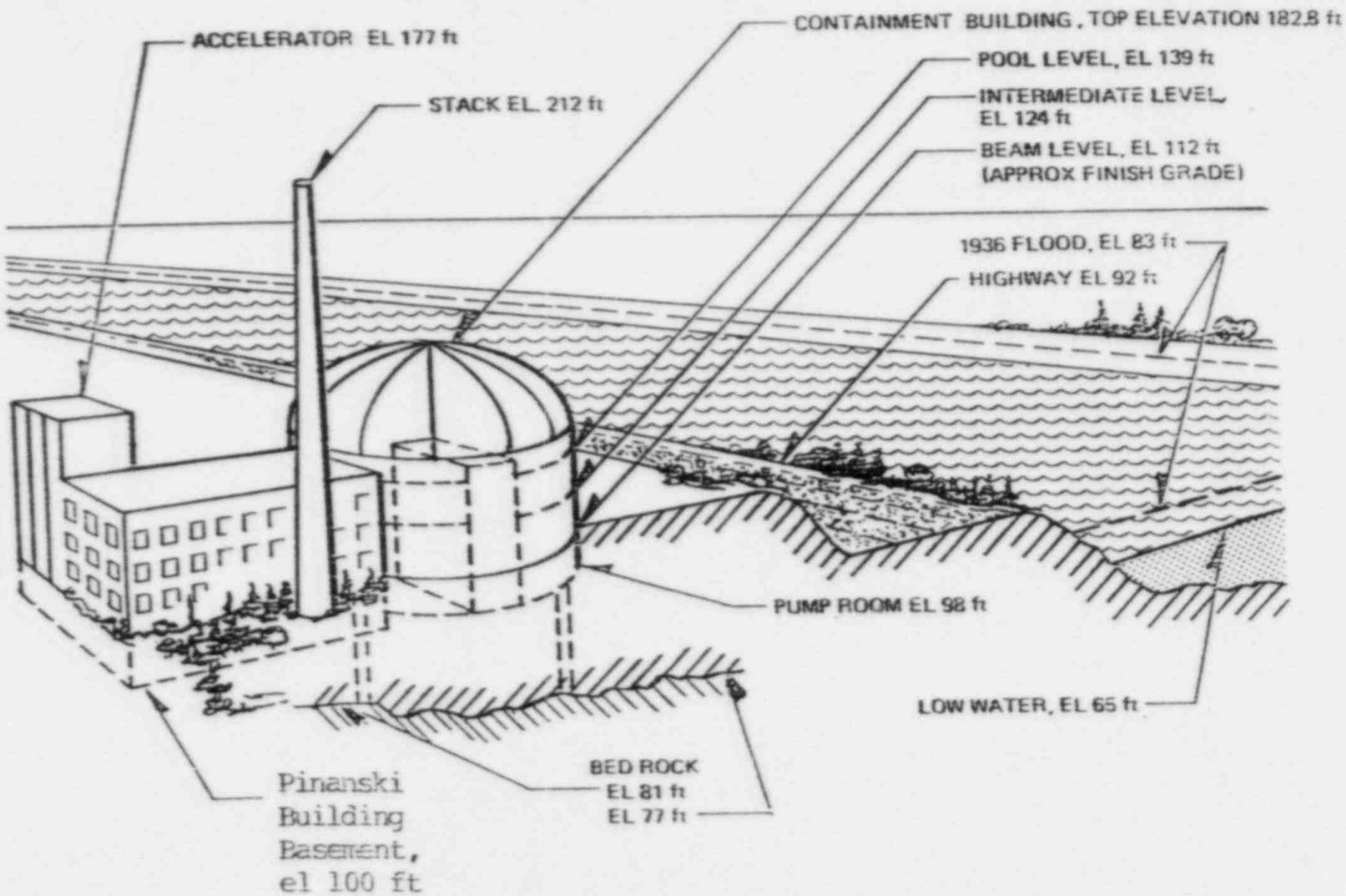
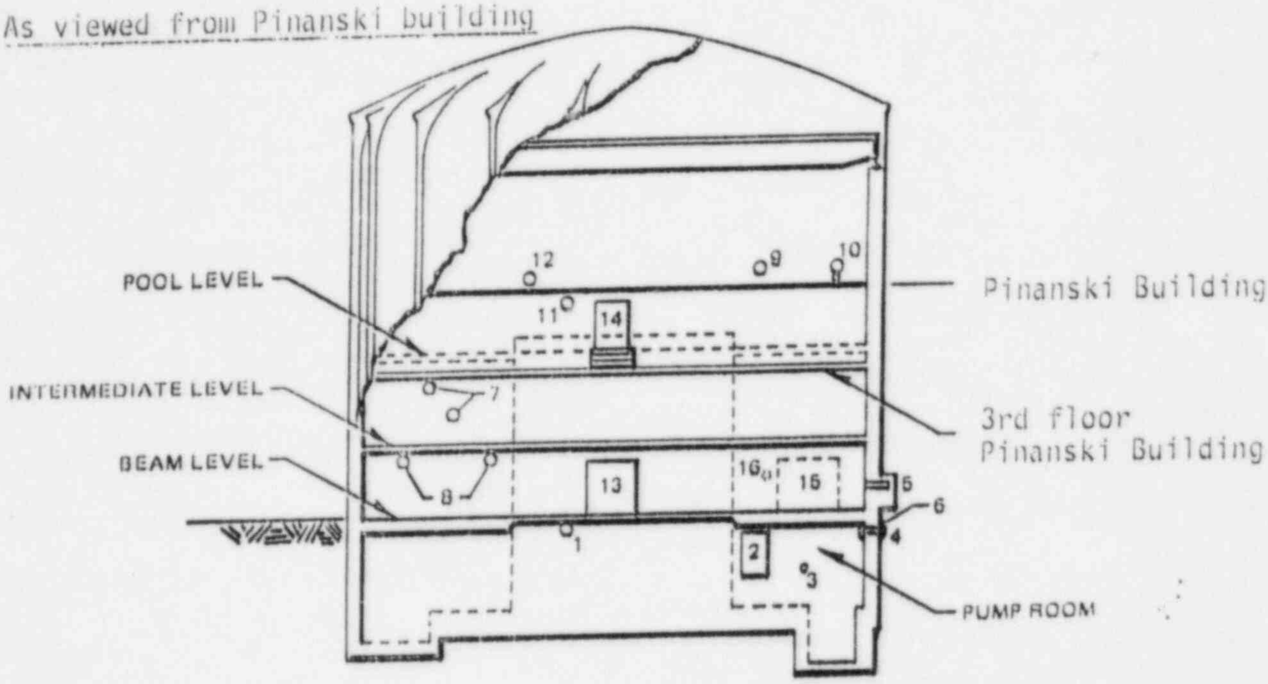
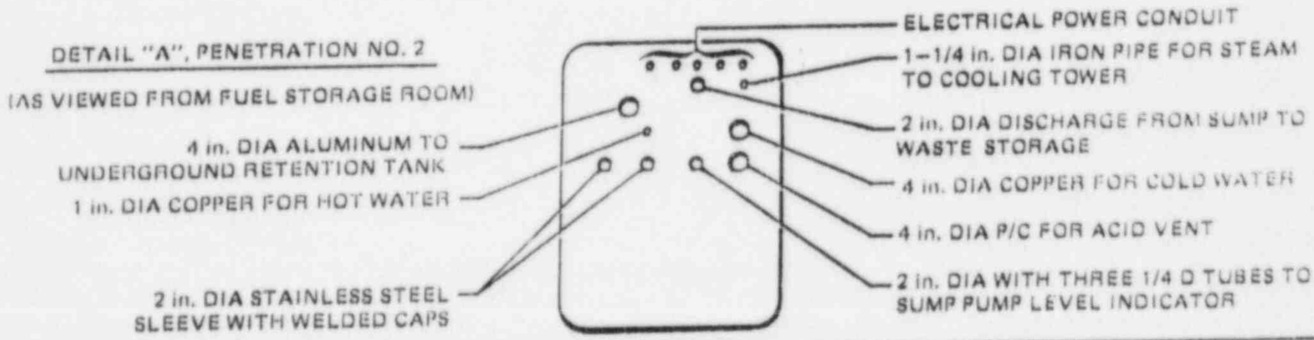


Figure 3.3. Longitudinal Section Through Buildings, Highway and River

As viewed from Pinanski building





LEGEND	DESCRIPTION
1	4 in. DIA SLEEVE CONTAINING WASTE TRANSFER PIPE
2	SEE DETAIL "A"
3	8 in. PIPE TO STORAGE TANK
4	2–10 in. DIA FOR SECONDARY SYSTEM TO COOLING TOWER, STEAM LINE & MAKE-UP WATER
5	CONCRETE LATTICE FOR BEAM PORT PLUG STORAGE
6	2–1-1/4 DIA CONDUIT LOCATED UNDERGROUND LEVEL BETWEEN TRUCK ENTRY AND BEAM PORT STORAGE FOR COOLING TOWER FAN MOTOR
7	4 in. DIA MAIN VENTILATION EXHAUST AND ELECTRICAL CONDUIT PENETRATION
8	3–1 in. DIA ELEC CONDUIT AND 5–SIGNAL CONDUITS FOR HP & AUDIO SYSTEMS
9	4 in. DIA VENTILATION INLET
10	FACILITIES AND EMERGENCY EXHAUST
11	4 in. DIA SEWER VENT
12	6 in. DIA VACUUM RELIEF VALVE
13	AIRLOCK
14	AIRLOCK
15	TRUCK DOOR
16	PENETRATIONS FOR PRESSURE COMPARISON AND FOR PRESSURIZING DURING CONTAINMENT TEST.

Figure 3.4. ULR Containment Penetrations

use, is located at the pool or operations level. The double doors in each set permit entrance to and egress from the building during reactor operation without loss of containment integrity. A truck door which can be used only when the reactor is not operating is provided on the beam level. All these access doors were manufactured by the W.J. Woolley Company, and all door frames are welded to the shell. A general description follows.

Structurally, the airlocks consist of a steel shell, steel bulkheads and steel doors. Sealing is accomplished by a continuous inflatable seal which surrounds the door edge. When the door is closed, the seal inflates outwardly from the door and impinges against a smooth stainless steel sealing surface. Because the seal is on the door rather than the door opening, it will not be damaged by equipment passing through.

Normal passage through an airlock is accomplished by the opening and closing of two doors, at least one of which is closed and sealed. Operation of the airlocks is automatic; push button stations for opening and closing <u>each</u> door are located at the entrance to either end of the airlock and inside the airlock. (Access keys or activation of the "permit" switch by the reactor console operator are required to operate the outside entrance door from outside as indicated in Paragraph 10.6). An electro-hydraulic system commands the inflation or deflation of the pneumatic seal as appropriate, and another swings the door through 90°, as appropriate. There are separate systems for each door, so that each airlock has four 1/3 horsepower electric motors associated with it. These, as well as the air compressor serving the systems, are supplied power by the emergency generator in case of power failure (see Paragraph 6.5). Relief valves in the hydraulic circuit allow the doors to be stopped during automatic operation by application of a nominal force by hand on the door. Interlocks exist so that, in a given airlock system, only one door will open at a time. Simultaneous opening of more than one door in an airlock will scram the reactor.

In the event of failure of all power, the airlocks can be manually operated. To enter the airlock from either the reactor side or the outside of the reactor, a lever mounted on a shaft penetrating the bulkhead is turned 90° in an unsealing direction and the door is then opened by manually swinging the door on its tapered roller bearing hinges. After entering the airlock, the door is swung closed and a second lever on the shaft mentioned above is rotated 90° in the sealing direction. This completes the action at one door, and the same process is followed at the other door. At each bulkhead the operating levers are keyed to the same shaft, and interlocked by a push-pull control lever which interlocks the doors so that the second cannot be unsealed while the first is unsealed. It is possible to override the mechanical interlock by removing a padlock from an access port on the front plate, swinging clear the access port (which sends a signal to the control room), and inserting a pin which prohibits the action of the push-pull control lever.

The truck door has an inflatable seal of the same design as that used in the airlocks. Its operating mechanism is a seal inflating and deflating device similar to the manual device supplied with the airlock; opening and closing of the door is manual. The door is interlocked to scram the reactor if it is opened or the seal is lost during operation.

3.1.2.2 Ventilation and Exhaust Ducts Ventilation ducts which penetrate the containment shell are welded to it and are fitted at the juncture with fast-acting, fail-safe blast valves (Valves A and B of Figures 3.5 and 3.6) which are described in Paragraph 3.4. The normal supply and exhaust ducts are nominally 48 inches in diameter, and the exhaust duct has an additional fail-safe valve located some 90 feet downstream (Valve C of Figures 3.5 and 3.6) to prohibit the release of air that exhausts from the shell after a closure signal and before the shell juncture valve closes.

A duct nominally 20 inches in diameter and fitted with an appropriately sized fail-safe blast valve designed as described above penetrates the shell and serves as an exhaust line for some of the support facilities (see Valve E of Figures 3.5 and 3.6).

A smaller duct, nominally 12 inches in diameter, penetrates the shell, encompasses a smaller version of the fail-safe blast valve at the shell-duct juncture, and serves as the emergency exhaust (Valve D and duct of Figures 3.5 and 3.6).

3.1.2.3 Sewer and Acid Vent Penetration Toilet facilities are located on the pool or operations level. The wastes are carried in a standard sewer line which drops down to the pump room where a deep (about six ft) trap is installed upstream of the containment shell penetration for connection to normal sewer pipes. The sewer line is vented to the outside atmosphere through a four-inch line which has a fail-safe pneumatically operated diaphragm valve

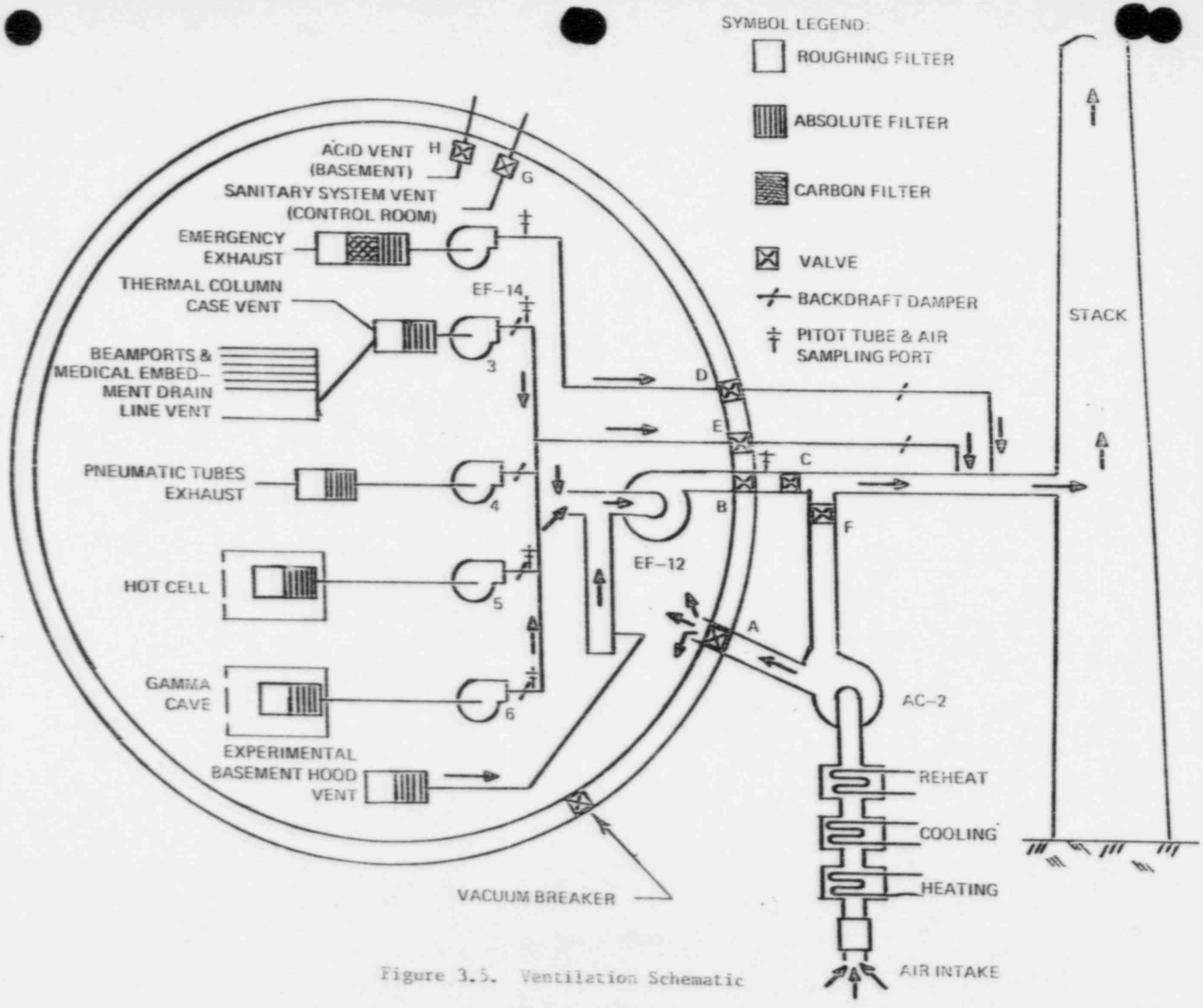


Figure 3.5. Ventilation Schematic

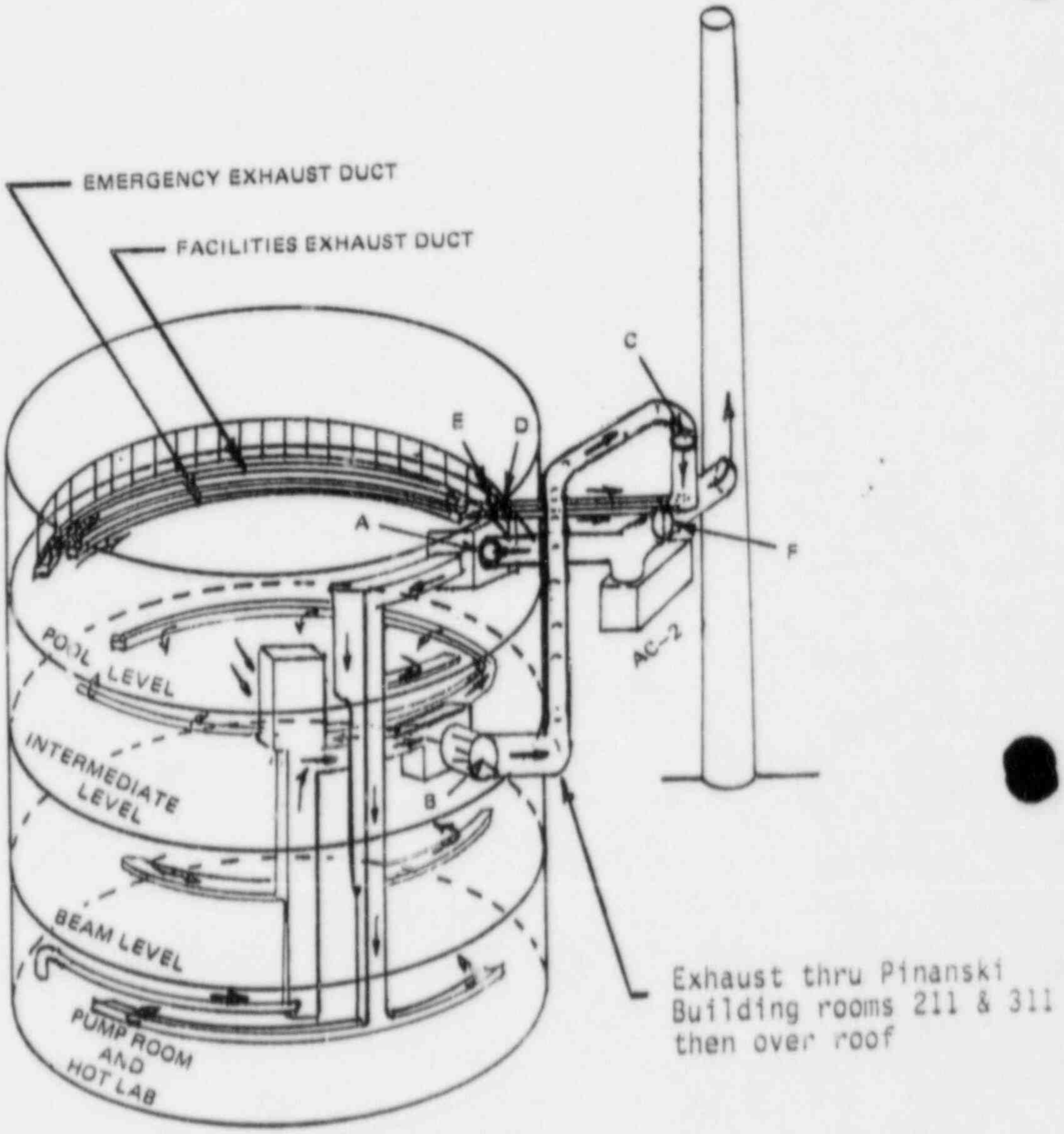


Figure 3.6. Containment Ducting

supplied by an air compressor inside the containment building that also serves the airlock doors.

The high level laboratory in the reactor basement is fitted with a hood, work bench, and several sinks. Although the chemical wastes themselves are emptied into a reactor basement sump, a vent line penetrates the shell after being fitted with a fail-safe pneumatically operated diaphragm valve. The reactor basement sump is connected to waste storage tanks outside the shell by an independent pipe line with a pump and closure valve.

3.1.2.4 Vacuum Relief Valve A six-inch diameter vacuum relief valve is mounted on the shell to preclude the possibility of creating an appreciable vacuum in the shell during a long-term shutdown or "containment closed" condition, because of temperature or atmospheric pressure differentials. This relief-valve is designed to allow air to pass into the shell before the shell achieves a vacuum of 0.5 inch water column.

3.1.2.5 Miscellaneous Penetrations A number of penetrations exist for telephone and signal wires, and for power line conduits. One small penetration is made for comparison of containment building pressure with outside ambient pressure. Air compressors that serve pneumatically operated valves are on the same side of the shell as the valves, and the compressor that furnishes shop air and air-lock door air is inside the containment, so that no high pressure air lines penetrate the containment. Appropriate penetrations are made for the passage of secondary system water to and from the cooling tower.

3.1.2.6 Penetration Seals All pipe and duct penetrations are welded to the steel shell or pass through special fittings designed to be airtight which are welded to the shell. Signal wire penetrations make use of Pyle fittings, and power line conduits are sealed with Duxseal or similar material. Pipes which carry fluid are fitted with check valves, fail-safe valves which close on containment command, or deep traps, as appropriate. Some special purpose penetrations not in use during normal reactor operation, such as the return line to the primary pump from the outside hold tank, are fitted with manual valves.

3.1.3 Containment Structure Design Bases

Paragraph 9.2 of this report describes the conditions and consequences of several assumed accident cases, among which are those discussed below.

(1) Assuming the complete loss of radioiodines and noble gases from a single failed fuel plate to the reactor pool with subsequent diffusion from the pool to the containment and leakage of airborne radioactivity at a rate of 0.10 per day from the containment, the dose rates were calculated. Results show that even under very restrictive assumptions of no atmospheric dispersion of a ground level release, two hour thyroid doses from inhaled iodines would be less than 10 rem, and the two hour whole body gamma dose resulting from exposure to the cloud of radioactive gases outside the containment would be about one rem.

(2) An excursion of 538 MWS would be required to raise the building overpressure to 0.5 psig. This is nearly four

times the 135 MWS achieved in the Borax tests[1,2,3,4]. If an accident leading to an overpressure of 0.5 psig were to occur during barometer readings of 30 inches of mercury just prior to the passage of a hurricane, the eye pressure of which was 27 inches of mercury, the resultant building pressure would be the original 0.5 psig plus the 1.47 psig equivalent to 3 inches of mercury, or 1.97 psig. Even accepting this sequence of unlikely events, the rated 2.0 psig is adequate.

(3) The postulated hurricane of (2), above, was assumed to move at a velocity that would result in a pressure decrease rate of 2 in. Hg h^{-1}. If an accident resulting in no overpressure but requiring containment is postulated immediately prior to the hurricane, and if the further assumption is made that somehow the emergency exhaust blower is able to remove the over-pressure air in the containment building as the hurricane approached (requiring a true linear exhaust rate of 373 cfm through the blower rated at 320 cfm), then no overpressure occurs. However, as the hurricane recedes and atmospheric pressure increases, an underpressure of the containment building is indicated, and this must be relieved by the vacuum breaker. Assuming the atmospheric pressure increases at the rate of 2 in. hg h^{-1} as the storm abates, the vacuum breaker must supply air to the building at a linear rate

of 373 cfm to equalize the containment building underpressure. In the six-inch diameter pipe of the vacuum breaker, this corresponds to an air velocity of 1895 ft min^{-1}, or about 22 miles per hour through the pipe, which indicates the vacuum breaker is adequate for this extreme case.

3.2 VENTILATION VALVES

The ventilation fail-safe blast valves are opened against a compression spring by compressed air, so that they close upon receipt of an appropriate signal from the radiation monitoring system, upon loss of electrical power, or upon loss of the supply air pressure. Normal operation of the valves is commanded from control room push buttons. Because the volume of air which must be exhausted to allow valve closure is larger for the large cylinders and pistons in the forty-eight-inch diameter valves, a dual relief system is used. This allows a relatively small amount of air to be bled (through a quick exhaust valve) from a secondary cylinder to relieve its piston which in turn opens a sizeable port in the main cylinder, thus allowing air to escape and the main piston to move to the valve-closed position (see Figure 3.7). The twenty-inch diameter and twelve-inch diameter valves release their compressed air directly through a quick exhaust valve (see Figure 3.8).

On all the blast valves the valve seat is steel on neoprene and is made with a no-wipe contact to minimize wear by abrasion. Closure times are no greater than 1.5 seconds, and the valves seat in a direction such that building internal overpressure assists in

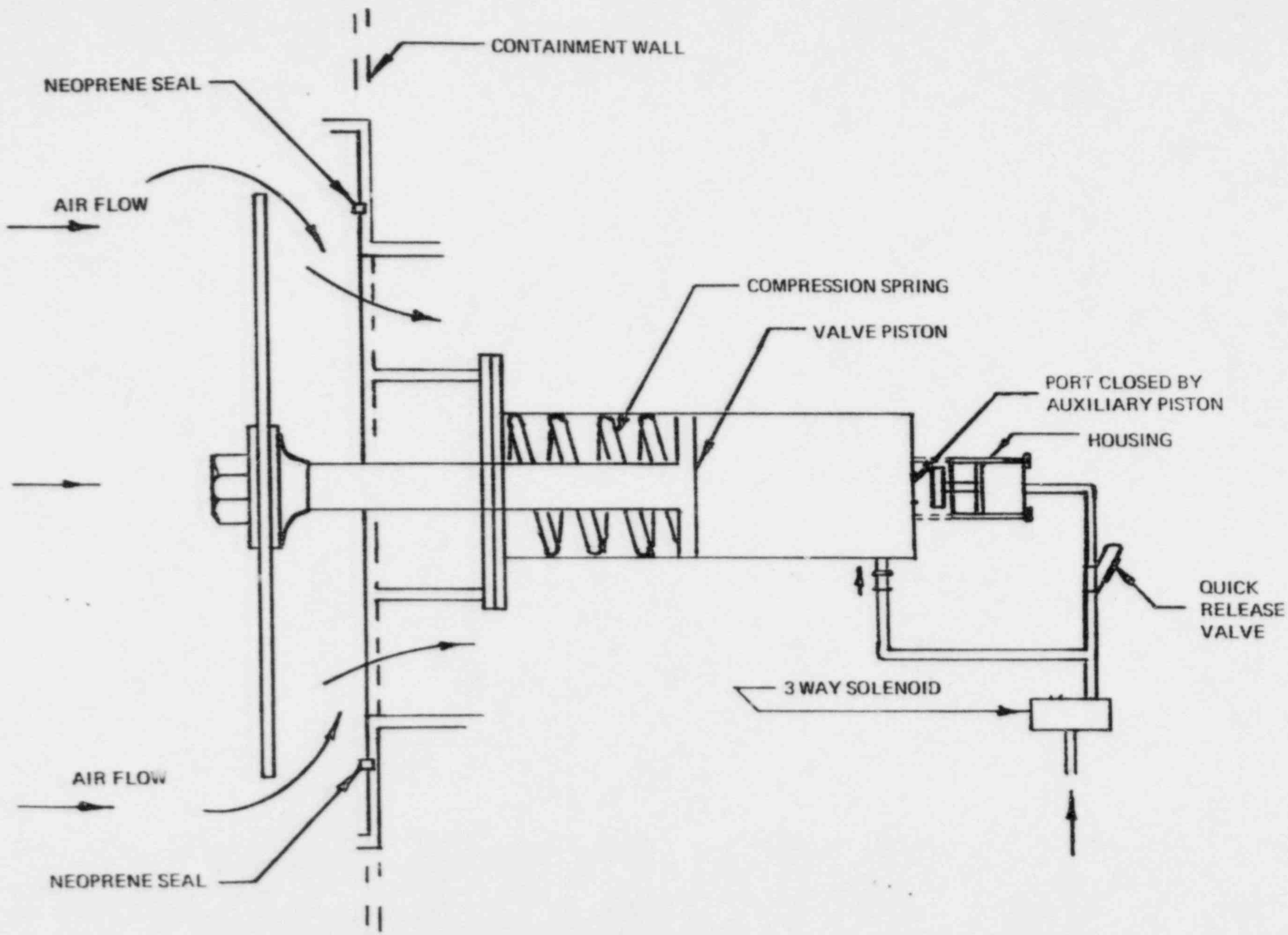


Figure 3.7. Schematic Of Air Control System for 48 inch Diameter Containment Valves

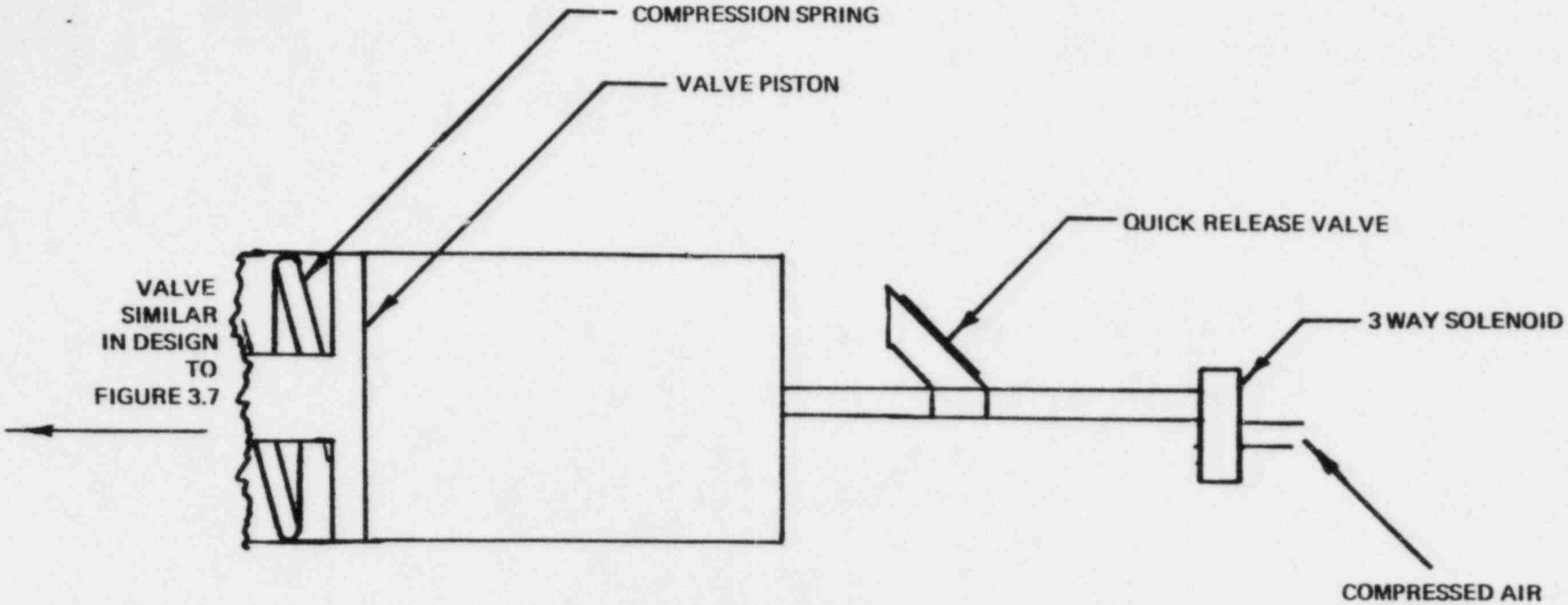


Figure 3.8. Schematic of Air Control System for 20 inch and 12 inch Diameter Containment Valves

making the seal, as shown in Figure 3.7.

3.3 VENTILATION

3.3.1 Building Air Supply and Exhaust

The internal volume of the containment building is about 375,000 cu ft, of which about 40,000 cu ft is occupied by the reactor pool, floors, walls, and columns, and other solid materials, leaving about 335,000 cu ft to be filled with air. Of this air, about 85,000 cu ft fills space below the beam level floor, leaving 250,000 cu ft in the relatively open space from beam level to ceiling. Certainly some mixing of airborne activity can be expected between these areas but because of the intervening floors and doors it is an order or two of magnitude slower than mixing throughout the freely open upper volume.

Air flows through the ventilation system at the nominal rate of 14,500 cfm. Even if the below beam-level air volume is included, this results in a removal rate constant of 0.0442 min^{-1}, which is seven times as great as the radioactive decay constant of ^{41}Ar (0.0063 min^{-1}), the expected principal airborne activity in the reactor room air during normal operation. If just the above beam-level air is considered, the factor of seven increases to nearly ten.

Supply air is heated and cooled or dehumidified by appropriate heating or cooling coils after it is filtered (see Figure 3.5). The externally located supply blower (AC-2) furnishes air nominally at 14,500 cfm which is distributed several levels, including a duct into the control room.

The exhaust blower (EF-12 of Figure 3.5) is inside the shell and set to exhaust air nominally at 15,000 cfm so that a slight negative pressure is normally maintained. A large building-exhaust inlet is located near one end of the pool with smaller inlets from the basement pump room and high level hood. The exhaust air is discharged to the atmosphere from a 100 foot high facility stack. The exhaust fan is interlocked so that if the supply fan stops, the exhaust fan cannot operate.

3.3.2 Facilities Exhaust

Several facilities, including (1) the hot cell, (2) pneumatic tube exhaust, (3) beamport and thermal column vent lines, and (4) the gamma cave are exhausted through absolute* filters by in-line blowers rated at (1) 600 cfm, (2) 230 cfm, (3) 600 cfm, and (4) 600 cfm, respectively. These lines exhaust into a common duct leading to the 20-inch diameter blast valve and external duct which goes to the stack as shown in Figure 3.5; control of these devices is from the control room. See Figure 3.9 for filter details.

The basement high level experimental hood exhausts air passing through absolute filters into a duct which connects directly to the main building exhaust system, as shown in Figure 3.5.

*The term "absolute filters" as used in this chapter refers to filters whose design specifications include: "...a penetration not exceeding 0.05% for 0.3 micron diameter homogeneous particles of dioctyl phthalate. The filters shall withstand 100% relative humidity air, determined dynamically, at a temperature of 70-100°F for periods of not less than 12 hours. The filter resistance shall not exceed 0.4 inch water gauge static pressure drop initially when operated at rated atmospheric conditions..."

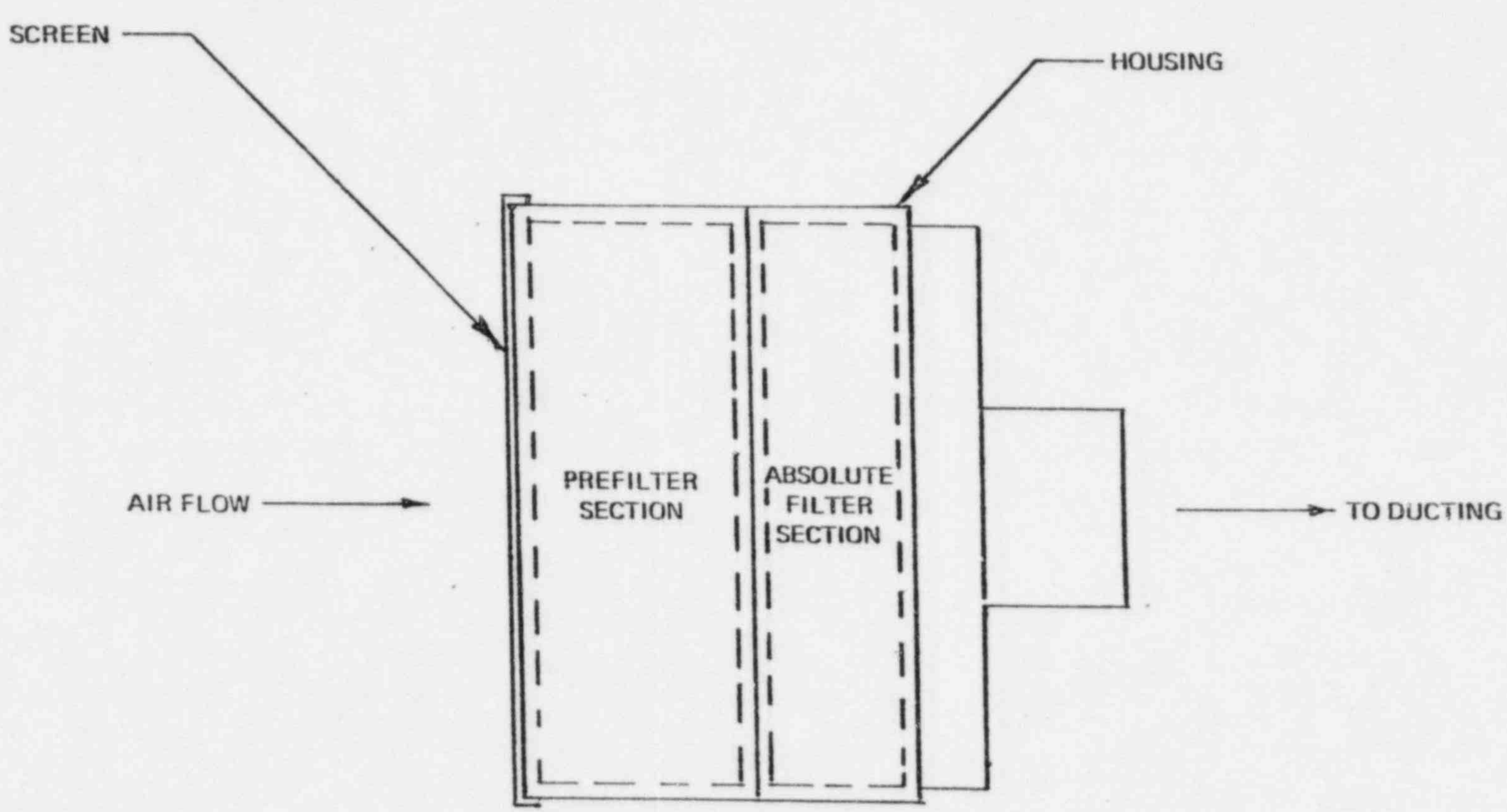


Figure 3.9. Schematic Of Absolute Filter System

3.3.3 Emergency Exhaust

An emergency system draws air through charcoal filters along with absolute filters (see Figure 3.10 for details) into a separate duct which passes through the containment shell where the integral blast valve is located, and then connects to the main exhaust down-stream from all other valves in order to allow air passage up the stack. This system is shown in Figure 3.5; EF-14 is a 320 cfm blower fan and Valve D of an appropriate differential pressure signal as described in Paragraph 3.4.2 or it can be operated manually from the console.

3.4 SYSTEM OPERATION

3.4.1 Normal Operation

Air is drawn in through an intake grille by the intake blower (Figure 3.5) at nominally 14,500 ft^3 min^{-1} and is filtered, cooled, or heated before entering the building through Valve A, from which it is ducted around the periphery at several levels.

Exhaust air is drawn from the basement pump room, the experimental hood in the basement, and the large grille located near the rear of the pool through Valves B and C, and thence to the 100 ft stack.

Each facility that is fitted with its own blower, such as the hot cell, the gamma cave, the pneumatic tube system, etc., is operated independently when indicated at the discretion of the operating staff under appropriate operating procedures.

The emergency exhaust system is not operated during normal conditions although, of course, it is in the "automatic" mode and potentially capable of operating as described in Paragraph 3.4.2.

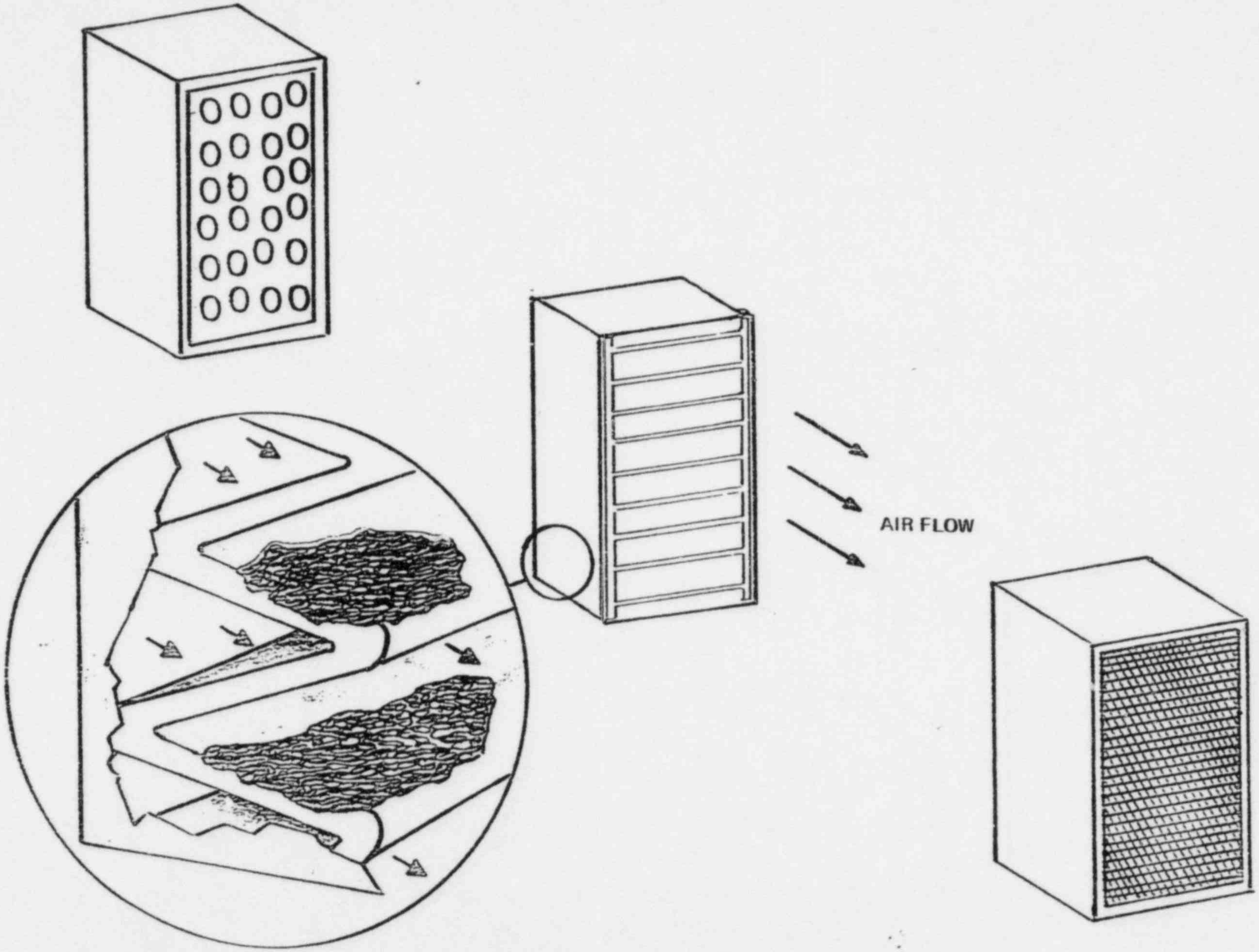


Figure 3.10. Schematic Of Charcoal Filter System

A manual mode does exist so that the functioning of this system can be checked, and so that it can be operated if such operation is indicated under extraordinary circumstances.

3.4.2 Emergency Operation

3.4.2.1 System Closure Closure of the reactor ventilation system is initiated in a number of ways as outlined below.

Signals which achieve a General Reaction in the Ventilation System (GRVS) are initiate:

(1) by activating the LREA or GREA* in the control room;

(2) by activating the GREA in the Reactor Supervisor's office;

(3) by the proper combination of switch options available in the control room (this amounts to achievement of the GRVS condition through mode control of the components making up the ventilation system);

(4) by loss of power; and

(5) by activation of ventilation freeze alarm.

3.4.2.2 Response to Initiation of System Closure When the GRVS condition is activated, the various control room switch options are overridden, and the following events occur (refer to letter and number designations on Figure 3.5).

(1) Valves A, B, C, E, G, and H close and Valve F opens. These are the "normal" positions of the valves, i.e., the condition when there is no power or actuating medium

*See Chapter 10 Appendix for those conditions which lead to the General Radiation Emergency Alarm (GREA) or the limited Radiation Emergency Alarm (LREA), both of which initiate GRVS. Also combinations of monitor signals which indicate a potential GREA automatically initiate GRVS.

delivered to the valves.* Clearance of the GRVS signal opens Valves G and H but the other valves operate only after a separate command from the Process Control Cabinet in the reactor control room.

(2) Fan blowers EF-12 and Numbers 3, 4, 5, and 6 cease to operate. Fan AC-2 continues to operate, except for the case where power is lost, in which case a short delay occurs (less than thirty seconds) before power is available from the emergency generator.** Clearance of the GRVS signal does not reactivate the affected fans until a separate command is given from the Process Control Cabinet.

3.4.2.3 Emergency Exhaust Operation The emergency exhaust operates independently and is not affected by a GRVS signal or lack of signal, by any "on" or "off" mode of the various Process Control Cabinet switches, nor by the status of the reactor control circuitry.

Normally the emergency exhaust will be in the automatic mode during reactor operation. In this mode, Fan EF-14 starts and Valve D opens if the differential pressure between the containment building and ambient outside ($P_{containment} - P_{ambient}$) reaches or exceeds a positive 0.25 $\pm$ 0.05 inch water column. Operation continues until either:

*Valve F is opened and closed by compressed air; the opened position being commanded by the solenoid upon proper signal or loss of power.
**The emergency generator feeds Fan AC-2, as well as Fan EF-14, Valve D, and control signals in the emergency exhaust system.

(a) the differential pressure drops to negative 0.25 ± 0.05 inch water column, or,

(b) the differential pressure rises to or exceeds positive 0.50 ± 0.05 inch water column.

If either condition (a) or (b) is met, EF-14 stops and Valve D closes. If (a) is the condition, the emergency exhaust remains shut down unless the differential pressure again rises to positive 0.25 ± 0.05 inch water column.

If (b) is the condition, the emergency exhaust remains shut down until the pressure drops to positive 0.50 ± 0.05 inch water column, at which point EF-14 starts and Valve D opens.

The emergency exhaust system is intended to relieve small overpressures accompanied by airborne radioactivity in the containment building by passing contaminated air through high efficiency particulate filters and then through charcoal filters before releasing the air to the stack. Emergency exhaust air carried to the stack is diluted by the high volume (nominally 14,500 cfm) of air being fed up the stack from Supply Fan AC-2 through the by-pass Valve F. The emergency exhaust system is interlocked with AC-2 to ensure proper operation.

Following a substantial pressure surge, unloading of the excess pressure through the charcoal filter is prohibited, since the capacity and effectiveness of the filter is not intended for large volume releases. Thus, an overpressure in the building of greater than a half inch water column prohibits release through the stack, and full reliance is placed on the containment.

3.5 CONTAINNMENT SYSTEM

3.5.1 Initial Acceptance Test

After the airlock doors, truck entrance door, vacuum relief device and miscellaneous penetrations for mechanical and electrical services were installed, the Chicago Bridge & Iron Company performed an acceptance leak test of the containment building. All welds in the bottom and all welds in the cylindrical shell were tested using vacuum box and soap film, and the airlock door systems were tested using a halogen leak detector. After appropriate instrumentation was installed, the containment building was sealed, air was pumped in to raise the internal pressure to 2-1/2 psig; the pressure was then reduced to 2 psig and the building maintained thus for 3 days with leakage measurements being taken during the period. The resulting measured leak rates varied from 0.159% per 24 hours to 0.046% per 24 hours, with an arithmetic mean of 0.089% per 24 hours.

3.5.2 Final Acceptance Test

Upon the completion of the reactor construction, LTI performed the final leak test. The resulting leak rate varied from 3.27 $\pm$ 0.33% to 4.21 $\pm$ 0.34% building volume per day at 2.0 psig. The test ensured that the containment building met the specification of less than 10% of the building air volume per 24 hours at 2 psig overpressure.

3.5.3 Continuing Tests

After the final acceptance test was successfully completed and the facility accepted, appropriate testing continues for the life of the facility. Part of this requirement is met by stipulating verification of negative building pressure (compared to ambient outside pressure) at least once every 8 hours during reactor operation. In addition to the daily verification of building integrity, the isolation of the building by proper valve closure in less than 2.5 sec (half the time for air under normal ventilation to travel from Valve B to Valve C in Figure 3.5) after an initiation signal is verified every 6 months, and a building leak test is performed nominally every two years. After the acceptance test, leak tests are performed at 0.5 psig, with results extrapolated to a leakage rate at 2 psig where the rate must be less than 10% per 24 hours. The test at lower pressure is considered acceptable because an overpressure of 0.5 psig would result from an excursion of 538 MW sec, and this is nearly four times the energy release of 135 MW sec achieved in the Borax[1,2,3,4] tests. The most current leak test resulted in measured leak rates ranging from 1.99 ± 0.44% to 2.38 ± 0.43% building volume per day at an extrapolated 2.0 psig overpressure.

3.6 REFERENCES

(1) J.R. Dietrich, "Experimental Investigation of the Self-Limitation of power Driving Reactivity Transients in a Subcooled, Water-Moderated Reactor," The Argonne National Laboratory Report ANL-5323, 1954.

(2) Reactors, Nucleonics 13, 40 (September 1955).

(3) J.R. Dietrich, "Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors," Proc. Intern. Conf. Peaceful Uses At. Energy, Geneva, 1955, 13 88 (1955).

(4) W.H. Zinn, et al., "Transient and Steady State Characteristics of a Boiling Reactor, the Borax Experiments," 1953, the Argonne National Laboratory Report ANL-5211, 1954.

4.0 FACILITY DESCRIPTION

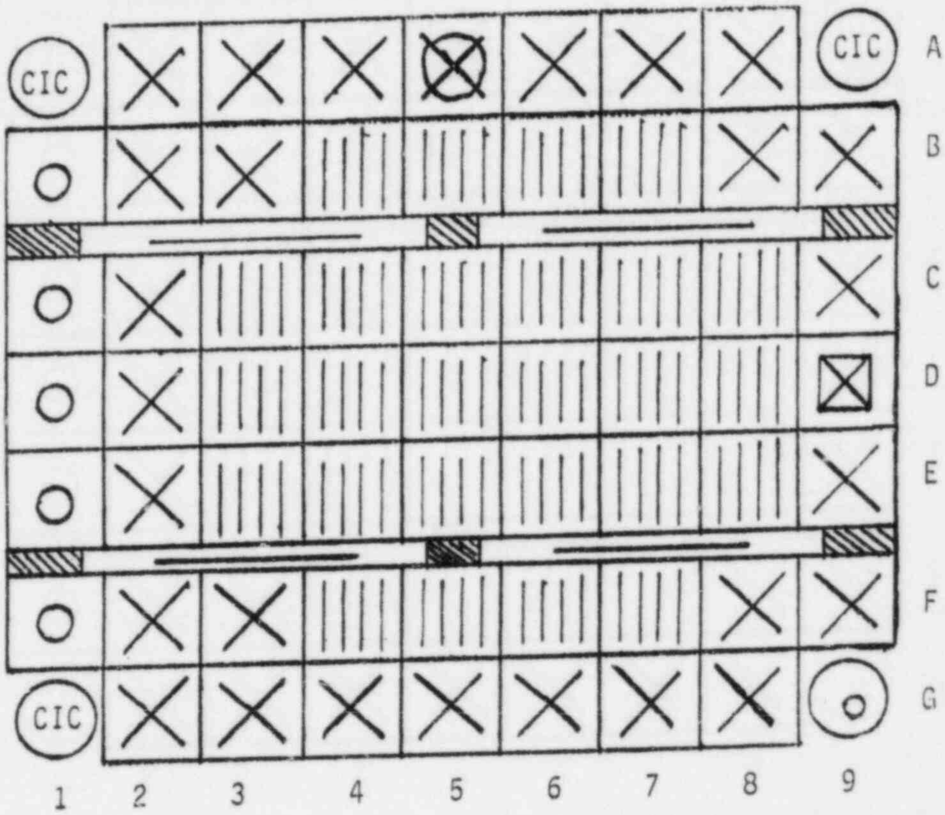
4.1 REACTOR DESCRIPTION

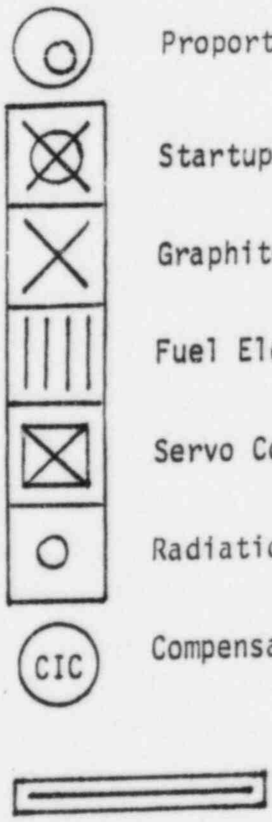
4.1.1 Core Arrangement (Figures 4.1 and 4.4)

The core consists of a 9 x 7 array of 3-inch square modules with the four corners occupied by posts. The initial loading was 26 fuel elements and will likely be expanded during operation as fuel burns up to include a 6 x 5 array in the center of the core. Four safety blades subdivide the fuel element array into three sections. The modules surrounding the fuel array may be utilized for graphite reflectors or radiation baskets and are filled with one or the other to ensure proper flow distribution with forced circulation.

4.1.2 Fuel (Figures 4.1 and 4.2)

The fuel is of the flat plate type. Each element consists of two aluminum side plates and 18 equally spaced flat fuel plates. The meat of the standard fuel plate is a uranium-aluminum alloy, about 24 wt% uranium-enriched to 93% U_{235}, sandwiched between 24-mil aluminum cladding on each side. This results in a fuel element loading of approximately 135 grams of U_{235}. The partial fuel element has a cladding thickness of 27-mil of aluminum with a fresh element loading of approximately 67 grams of U_{235}. Each fuel plate is blister tested to ensure that bonding is complete between the fuel and clad. This test is run at 940°F for one hour. Any observable blister is cause for rejecting the plate. The plate may also not be reduced more than 25 percent in thickness by rolling after the tests have been made. The plate is also fluoroscoped to assure proper positioning of the





Proportional Counter

Startup Source

Graphite Reflector Element

Fuel Element

Servo Control Element

Radiation Basket

Compensated Ionization Chamber

Control Elements

Figure 4.1. Core Arrangement

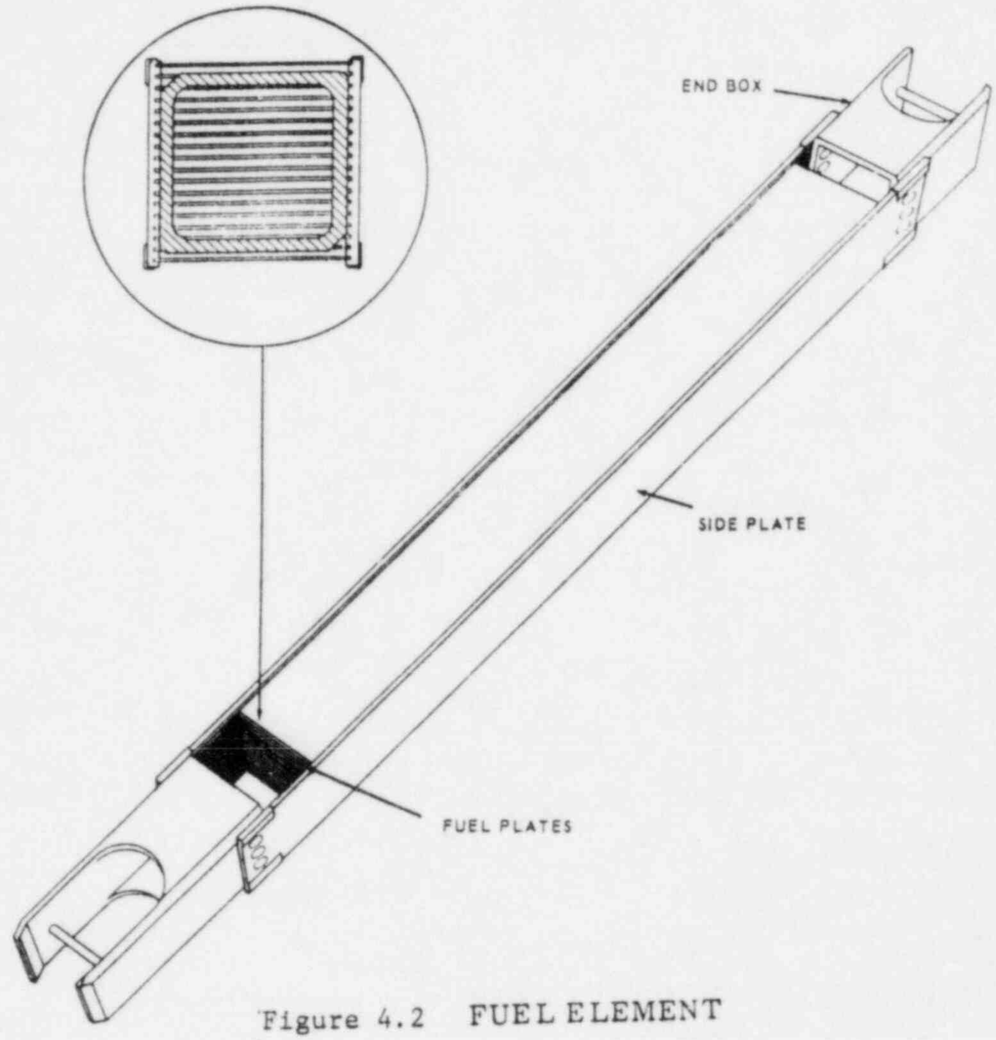


Figure 4.2 FUEL ELEMENT

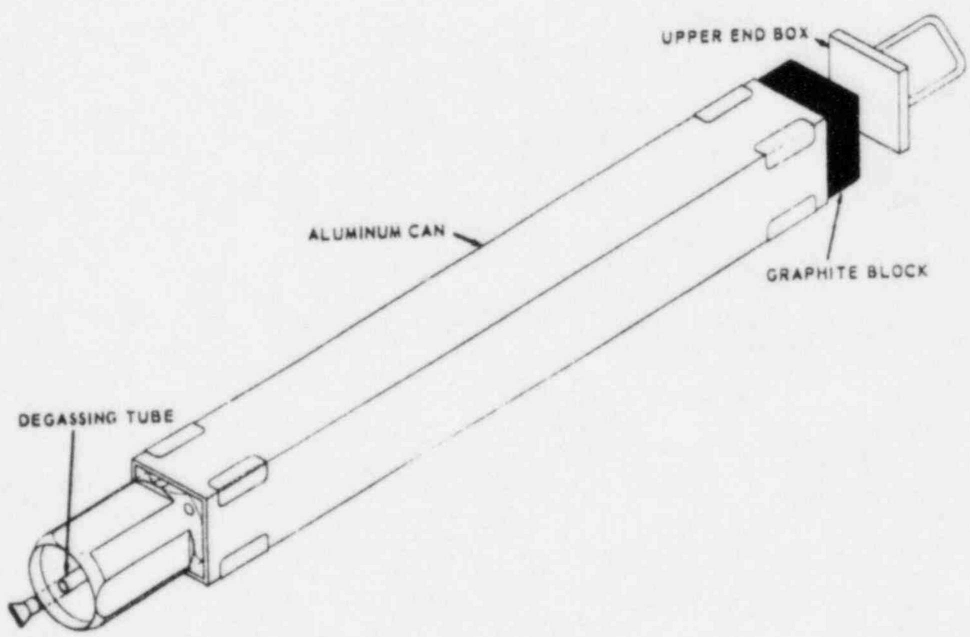


Figure 4.3 REFLECTOR ELEMENT

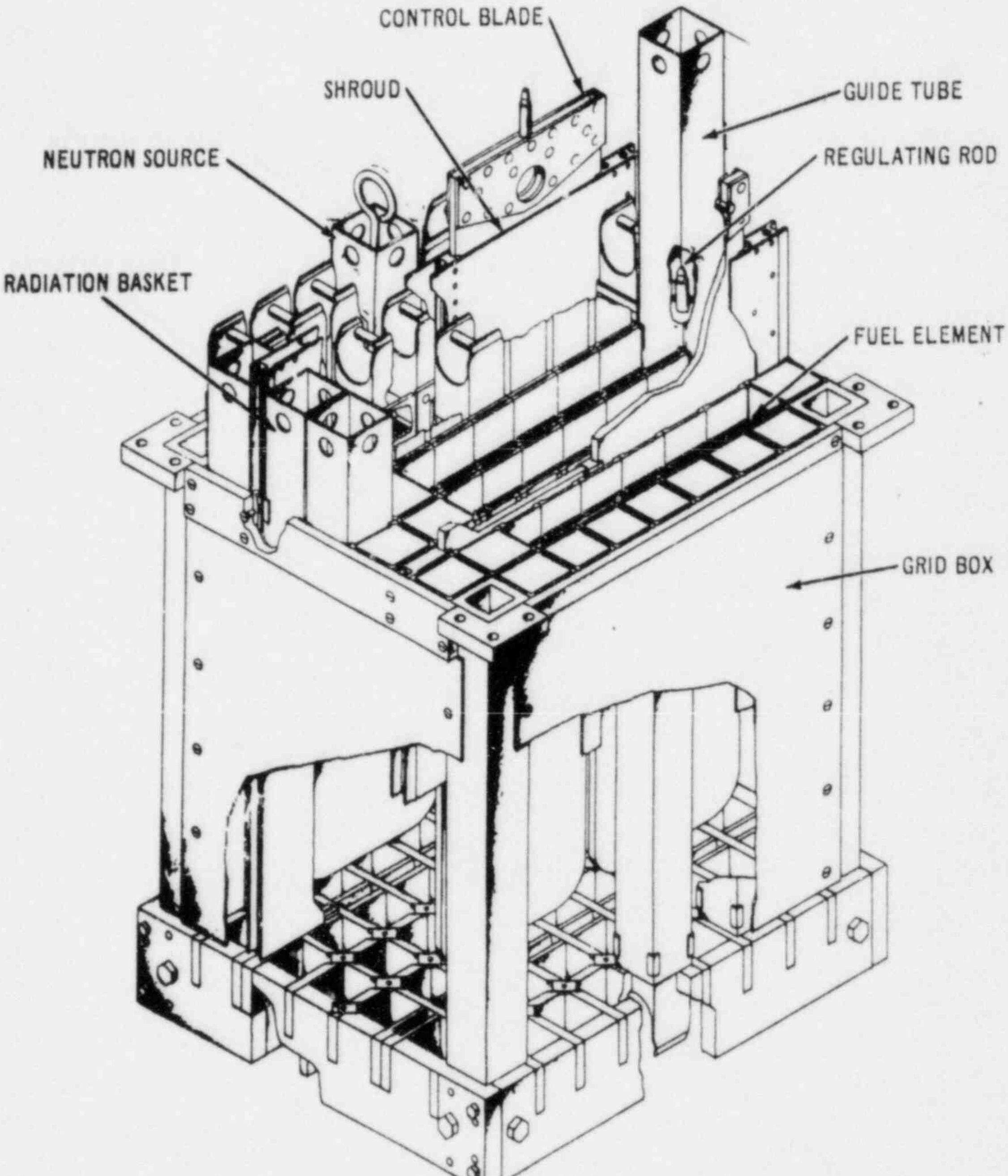


Figure 4.4 REACTOR CORE

fuel meat region.

The fuel plates are 0.060 inch thick, 2.79 inches wide, and 25 inches long. The active fuel bearing length is 24 inches. Then assembled in the fuel element, plates are separated by a 0.1 inch gap for water passage. Two identical end boxes position the fuel element in the grid and provide handles for refueling. Including end boxes, the elements are nearly 40 inches long. The elements may be inverted and rotated to achieve more efficient utilization of fuel.

4.1.3 Reflector (Figure 4.3)

The reflector element is a reactor grade graphite block contained in a 3-inch square aluminum can. By reducing neutron flux leakage from the active core, the reflector elements increase the neutron flux at the core perimeter and improve utilization of the fuel. The graphite log in the reflector element extends about 3 inches above and below the active 24-inch length of the adjacent fuel elements. The thin-walled aluminum can is evacuated to collapse the walls onto the graphite and thus provide good heat transfer to the pool water.

The design of the reflector element allows for thermal expansion, and for an increase in graphite dimensions of 1.1% due to irradiation growth and gas evolution from an integrated flux of 2×10^{21} nvt. Irradiation tests in the Hanford reactors and at the MTR, in environmental conditions at least equivalent to those at this reactor, have revealed no other significant changes in the graphite properties.

4.1.4 Start-Up Source

A 5 curie americium-beryllium neutron source is provided for routine startup of the reactor.

This source is located inside a source holder which in turn is seated in the reflector region of the core grid. This ensures that the source is in the active region of the core. The source holder is an aluminum shell, 3 inches square and 42 inches long, normally water filled. This source would be removed for operation of the core above 10 kW.

In addition to the main neutron source described above, an antimony-beryllium neutron source is available. This source may be placed in a small source holder which is inserted in a standard radiation basket in the reflector region of the core.

4.1.5 Control Blade (Figure 4.4)

Reactor control for startup and shutdown is accomplished by four blade-type control elements, with a total present worth of 11.5% $\Delta k/k$. The poison section of boron carbide and aluminum is sandwiched between aluminum sideplates. It is 40.5 inches long, 25 inches providing active control of the core. The remaining 15.5 inches connect the poison section to the drive tube.

The shrouds act as guides for the control blades throughout blade travel while in the core. Each shroud consists of two thin aluminum plates 38 inches high, separated by aluminum spacers to provide about a 1/8-inch water gap around the blade. The shroud is fastened to the sides of the grid box with screws. Small flow holes at the bottom of the shroud aid in minimizing the effect of viscous damping on the scram time.

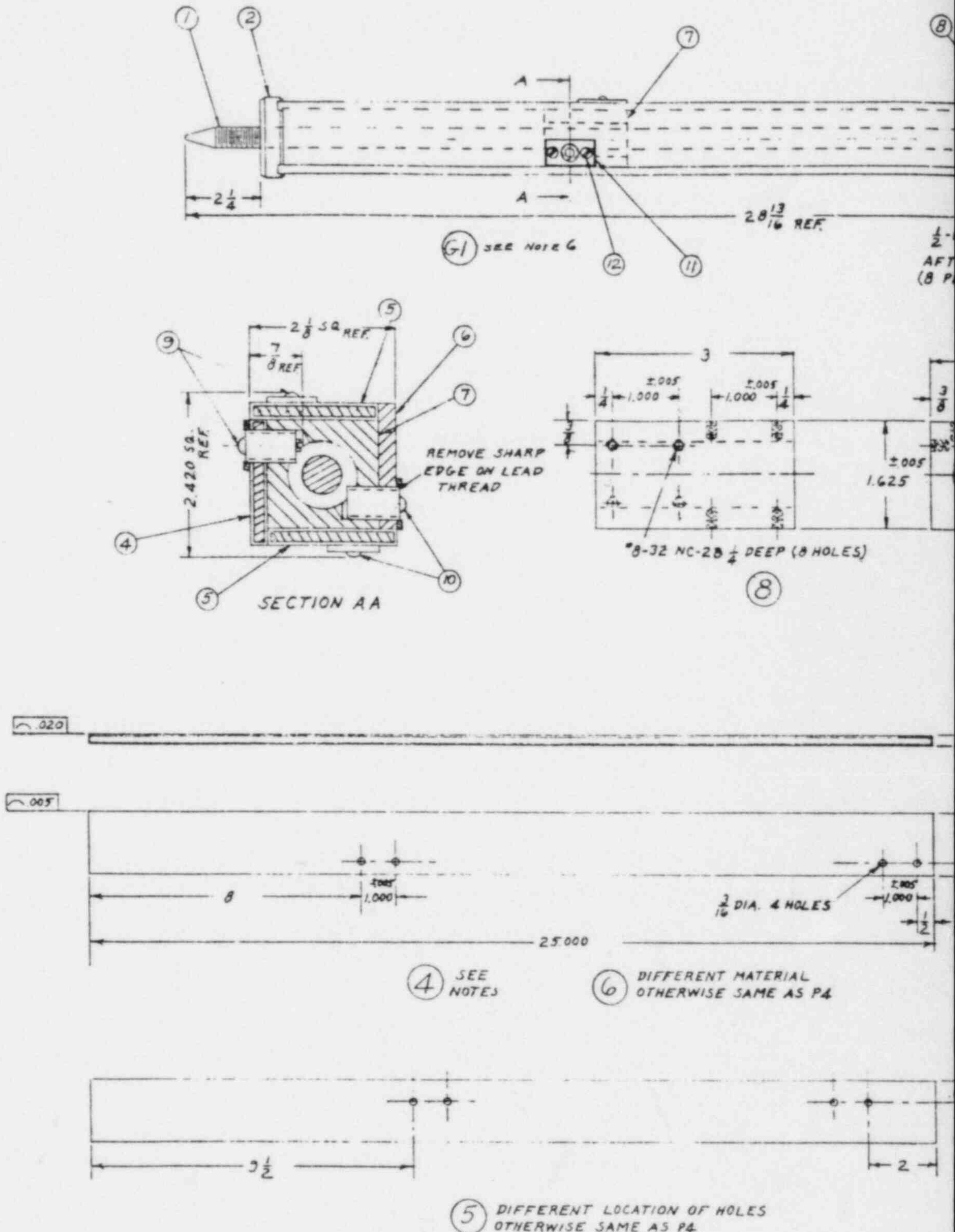
4.1.6 Servo Regulating Rod (Figure 4.5)

The servo regulating rod provides continuous control of the reactor by actuation of an automatic servo control system to compensate for small changes in reactivity. The regulating rod has a reactivity worth of about 0.5% $\Delta k/k$ when the aluminum side is next to the core, and of about 0.7% $\Delta k/k$ when the aluminum side is farthest from the core. The regulating rod is positioned with the aluminum side next to the core to ensure a worth less than 0.7% $\Delta k/k$. The regulating rod is fabricated of a 25-inch long, 2-1/2-inch square boral tube of 1/4-inch wall thickness which is lock-screwed to the servo regulating rod drive shaft. A 3-inch square aluminum shell guide tube is seated in the reflector region of the grid to ensure proper rod travel.

4.1.7 Control Blade Drives (Figure 4.6)

Each of the four control blades is actuated by an electro-mechanical drive mechanism which can position, hold, or scram its respective blade. The drives, mounted on the locating plate on the underside of the bridge above the core, are coupled to the control blade extension through electromagnets. Gravity scram results when the electromagnets are deenergized. The control blade drive mechanism includes a drive motor, worm-gear reducer, slip clutch, ball bearing screw assembly, limit switches, scram magnet assembly, housing, and related mounting and connecting hardware. A position transmitter is provided to give a continuous position indication within $\pm$ 0.02 inch.

The drive motor is a reversible electric motor, with an integral worm-gear assembly which provides speed reduction and



A
A
2 1/4
28 13/16 REF.
G1 SEE NOTE 6
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7/8 REF
2.420 SQ. REF.
REMOVE SHARP EDGE ON LEAD THREAD
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1.000
1/4
±.005
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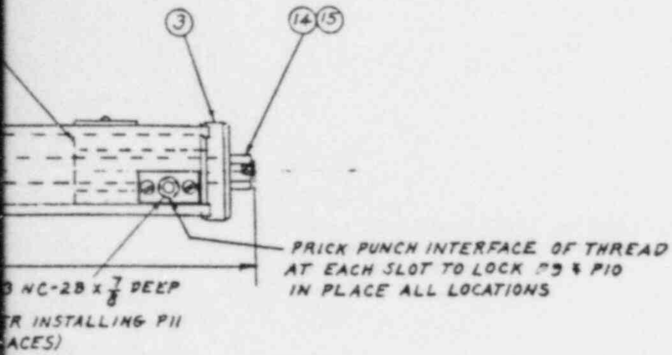
3/8
#8-32 NC-2B 1/4 DEEP (8 HOLES)
.020
.005
8
1.000
3/16 DIA. 4 HOLES
25.000
1/2
SEE NOTES
DIFFERENT MATERIAL OTHERWISE SAME AS P4
9 1/2
2
DIFFERENT LOCATION OF HOLES OTHERWISE SAME AS P4



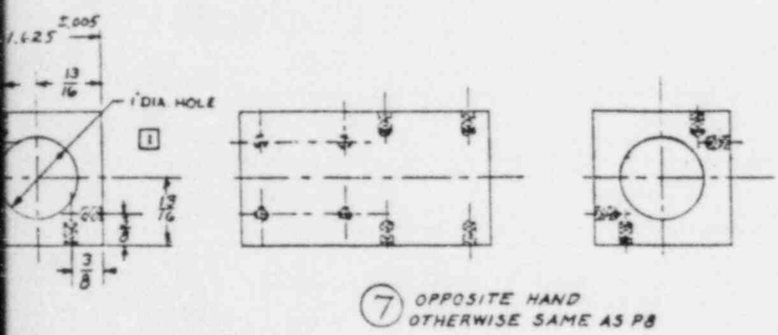
FCF ON 105X376
105X339

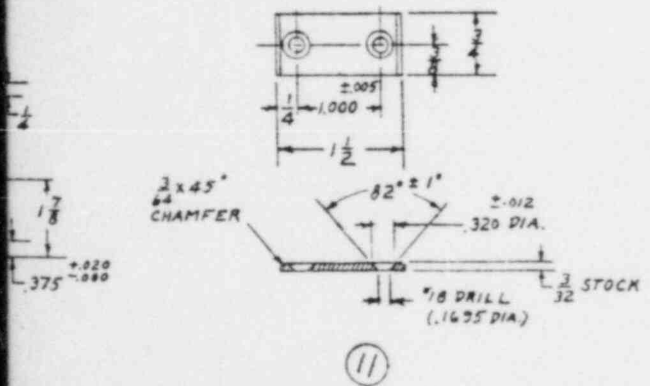
NOTES:

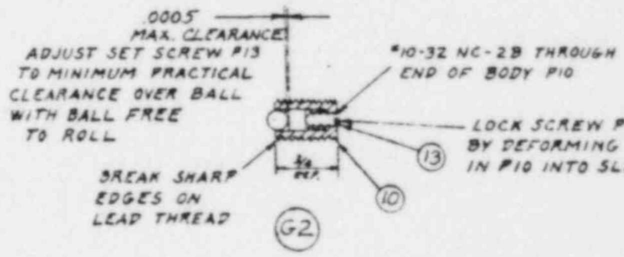
3 1. MATERIAL OF P4 & P5 TO BE FLUSH EDGE CLAD BORAL SHEET (ROLLED CHANNEL EDGE CLADDING). 35% ± 10% B4C. SIMILAR TO BORAL SHEET AS PURCHASED FROM BROOKS & PERKINS, 1950 WEST FORT ST. DETROIT 16, MICHIGAN OR ENG. APPD. EQUIV.

2. VENDOR SHALL CERTIFY THAT ALL CLADDING, INCLUDING EDGE CLADDING, SHALL BE CONTINUOUS AND UNBROKEN BY BORON CARBIDE OR OTHER INCLUSIONS. THIS CONFORMANCE SHALL BE CONSTRUED AS ACCEPTABLE ON BASIS OF FOLLOWING MINIMUM REQUIREMENTS:
   A. COMPLETE VISUAL EXAMINATION, WITH ANY QUESTIONABLE AREAS INSPECTED WITH THE USE OF 10X MAGNIFICATION.
   B. ACTUAL CLADDING THICKNESS SHALL BE MEASURED AND ESTABLISHED AS ACCEPTABLE USING A MINIMUM OF ONE (1) TEST COUPON PER BORAL EDGE CUT FROM AN EXCESS PIECE IN ACTUAL FABRICATION PROCESS. AT LEAST TWO OF THESE COUPONS SHALL BE EDGE CLAD. SHEET CLADDING AND EDGE CLADDING SHALL NOT VARY MORE THAN PLUS OR MINUS 50% OF NOMINAL SHEET CLAD.

3. VENDOR SHALL SUBMIT ALL TEST COUPONS WITH COMPLETED PRODUCT FOR APED VERIFICATION AND RECORD PURPOSES.







NOTES: (CONTINUED)

4. VENDOR SHALL CERTIFY BORON CARBIDE CONTENT TO BE 35% ± 10% BY WEIGHT AS APPLICABLE.

5. VENDOR SHALL PROVIDE CERTIFICATION OF CLADDING MATERIAL.

6. THIS ASSEMBLY MAY BE TESTED DIMENSIONALLY BY TRIAL INSERTION IN GUIDE TUBE. SEE MASTER PARTS LIST FOR PROPER DWG. NO.

Figure 4.5. Servo Regulating Rod

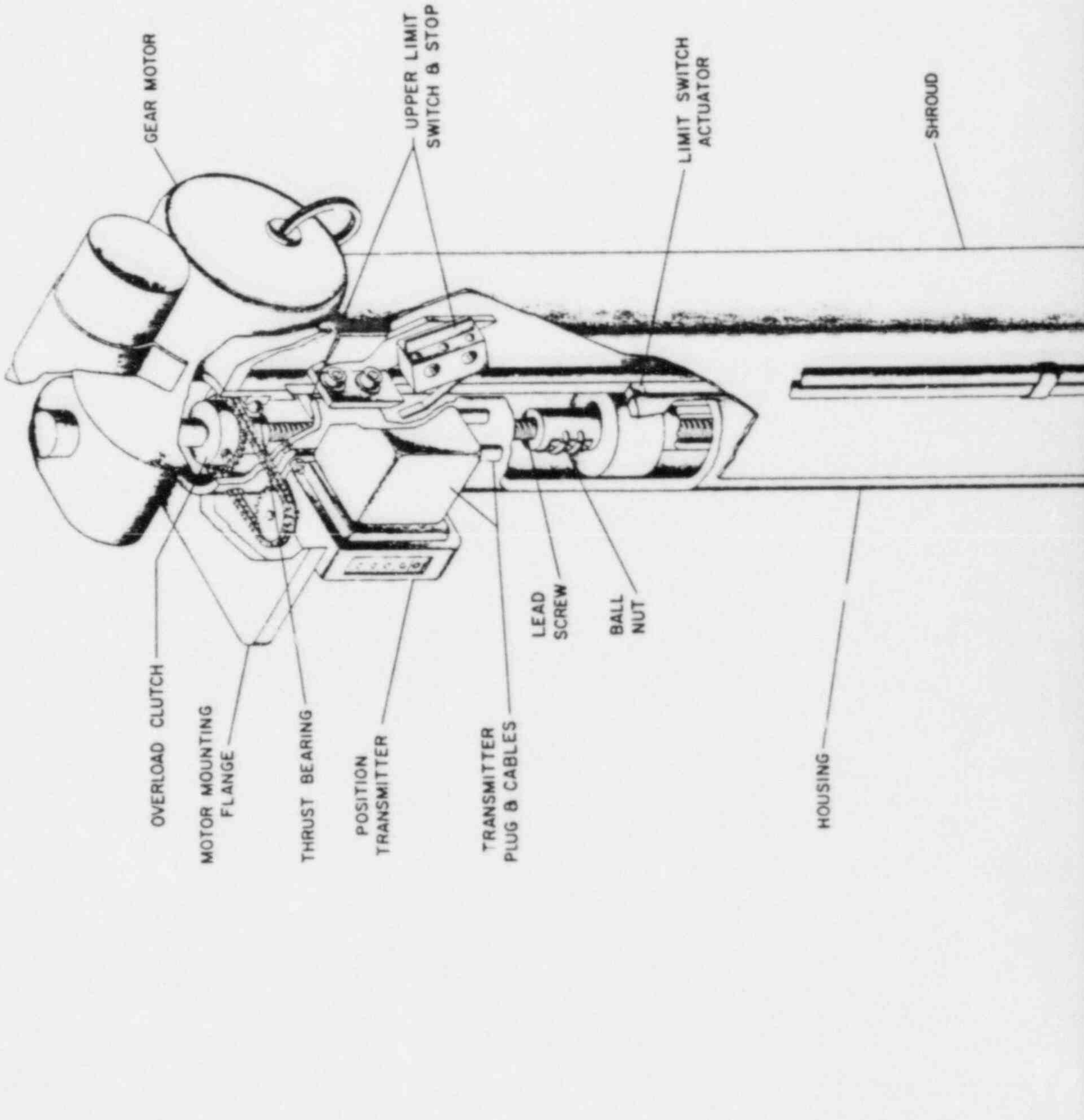
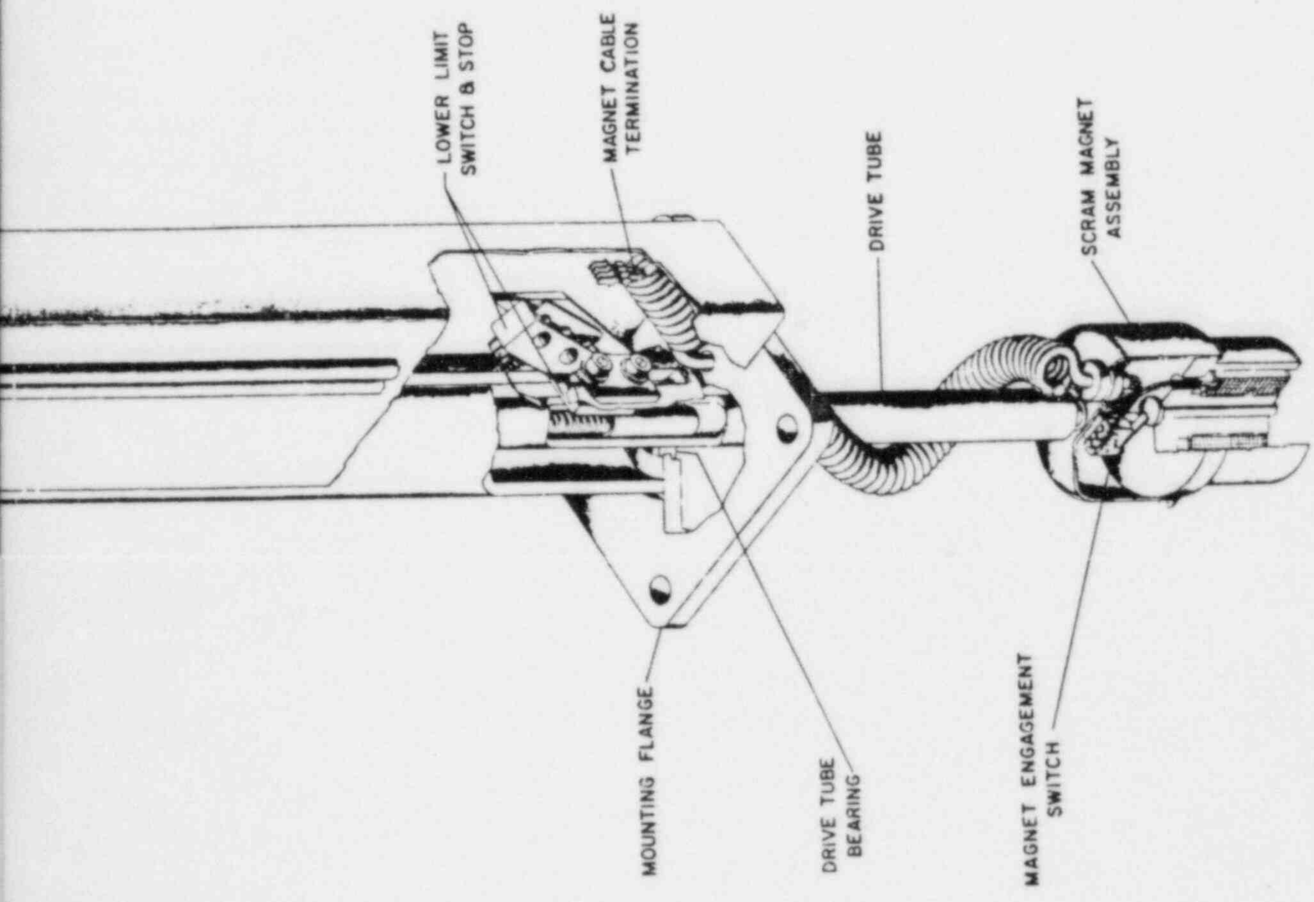


Figure 4.6. Control Blade Drive Mechanism


LOWER LIMIT
SWITCH & STOP
MAGNET CABLE
TERMINATION
DRIVE TUBE
SCRAM MAGNET
ASSEMBLY
MOUNTING FLANGE
DRIVE TUBE
BEARING
MAGNET ENGAGEMENT
SWITCH

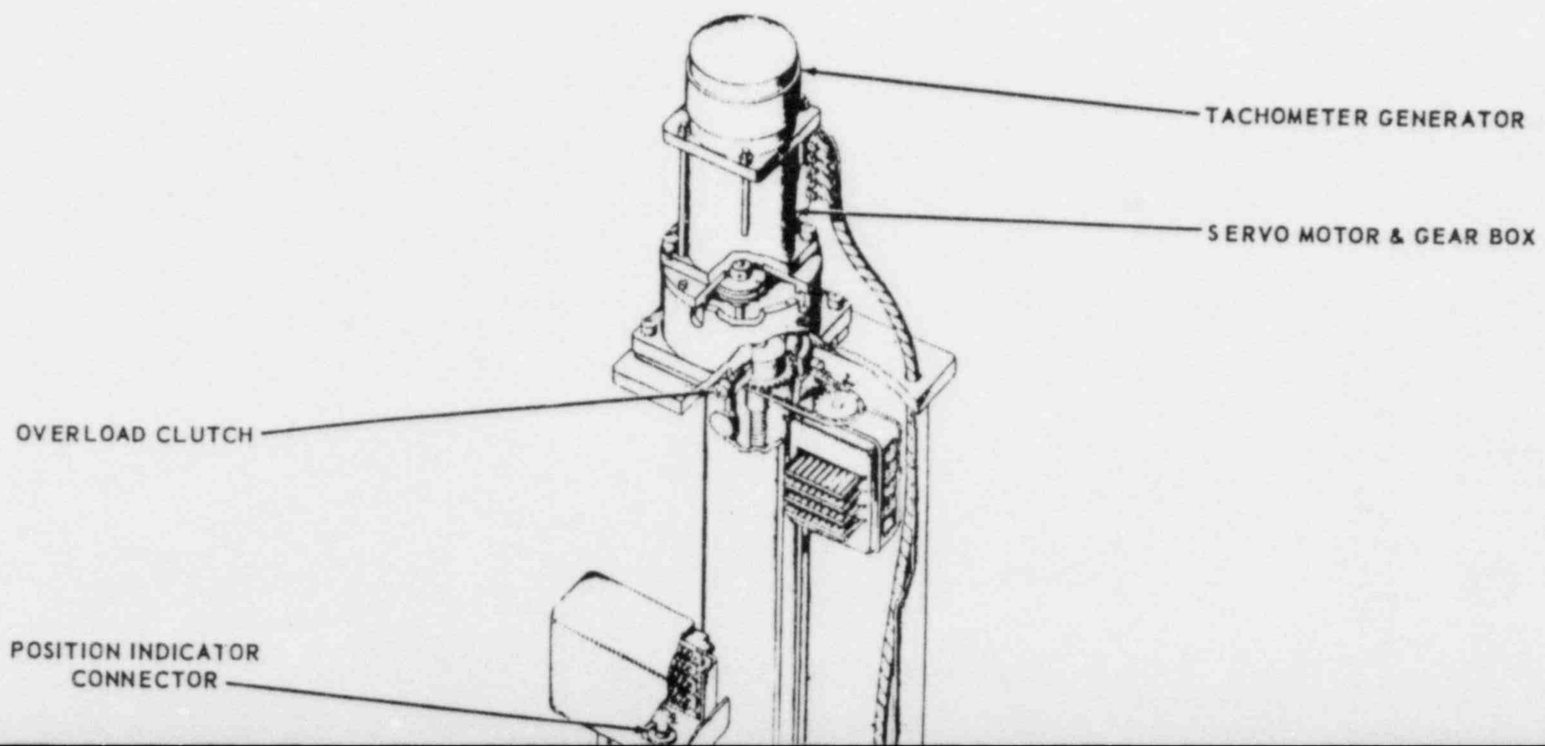
prevents possible control blade drift. Overtravel of the mechanism is limited to less than 0.03 inch in the up direction and 0.04 inch in the down direction. A friction-disc type overload clutch, located on the drive motor output shaft limits the torque applied to the lead screw so that the force on the related control blade is limited to approximately 75 pounds. Lead screw rotation causes the ball nut and drive tube to raise and lower the control blade. Each of the control blade drive mechanisms operates through a stroke of 32 inches at a present operating speed of 3.7 in/min. This speed corresponds to a maximum reactivity addition rate of .017% $\Delta k/k$/sec based on a blade worth of 3.6% and the assumption that the maximum blade worth is twice the average. The limit switch actuator, located at the base of the ball nut, actuates adjustable limit switches at the top and bottom of the 32-inch stroke, which shuts off the drive motor and actuate control blade position indicator lights located in the reactor control room. Mechanical stops are provided at the upper and lower limit switch positions to prevent damage to the drive tube and control blade in the event that either limit switch fails to operate.

The scram magnet assembly, located on the lower end of the drive tube above the pool water, when energized provides a coupling between the drive tube and control blade. A magnet engagement limit switch in the scram magnet assembly actuates an indicator light in the reactor control room to indicate blade engagement. To scram the reactor, the magnet is deenergized to release the extension shaft and control blade allowing the control blade to gravity-fall into the fall within 600 milliseconds after the voltage applied to the scram circuit reaches zero. Likewise, in the event of facility power failure the scram circuit voltage will drop to zero, and an automatic scram will result. Recovery of the control blade after scram is achieved by running the drive mechanism fully down and engaging the scram magnet to the top of the control blade shaft.

A mechanical position indicator on each control blade drive mechanism is synchronized through a position transmitter to an indicator in the reactor control room to provide blade position in the core relative to the fully inserted (or scram) position. The indication presented is in digital form in increments of 0.010 inch to an accuracy of 0.020 inch.

4.1.8 Servo Regulating Rod Drive (Figure 4.7)

The servo-controlled regulating element drive actuates the servo element allowing regulations of reactor power within closer limits than those attainable with use of the control blades alone. The servo drive mechanism is similar to the control blade mechanism in that the two units have an identical slip clutch, ball bearing screw assembly, limit switches, housing, and position indicator. However, a solid coupling replaces the scram magnet assembly, negating the scram provision. The servo control drive is operated by a servo motor through a spur gear train through a total stroke of 26 inches at a maximum travel speed of 78 in./min, which corresponds to a maximum reactivity addition rate of 0.054% $\Delta k/k$/sec based on a rod worth of 0.5% $\Delta k/k$, an effective length of 24 inches, and the assumption that the maximum rod worth is twice


TACHOMETER GENERATOR
SERVO MOTOR & GEAR BOX
OVERLOAD CLUTCH
POSITION INDICATOR CONNECTOR

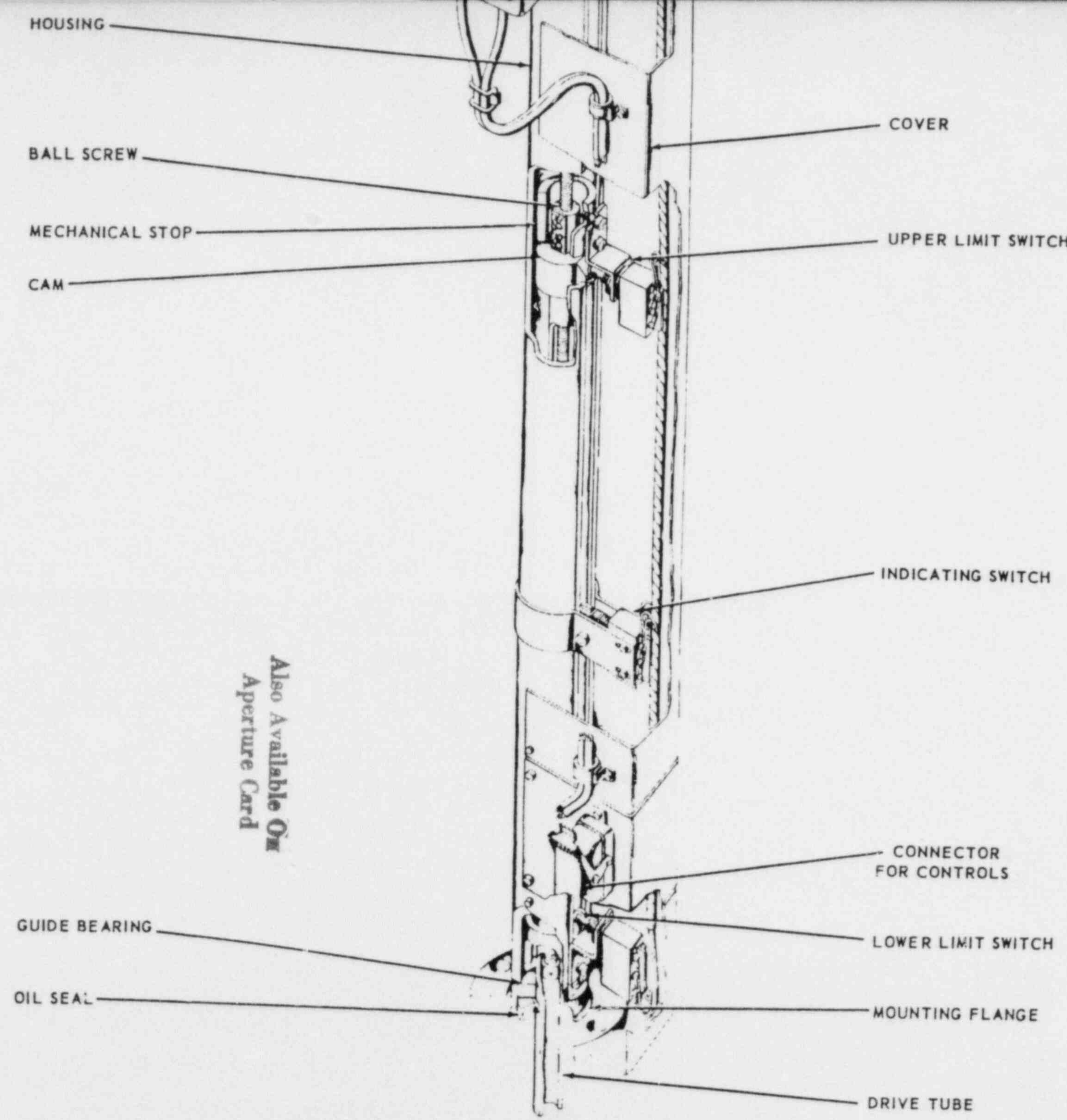


Figure 4.7. Servo Control Drive Assembly

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the average.

The servo drive motor is a fast response, two-phase, servo-type electric motor which provides the power required to insert and withdraw the servo control element. A tachometer, or rate generator, integral with the motor, provides an a-c feedback signal for damping the automatic power level channel. A precision spur gear train provides speed reduction and furnishes driving power to the ball bearing screw assembly. Limit switches and mechanical stops at each end of the 26-inch stroke are similar to those on the control blade drive mechanisms. A 3/16-inch diameter by 2-inch long stainless steel cotter pin provides connecting means for the drive shaft and servo element shaft. Removal of the cotter pin allows the drive and element to be disconnected and reassembled without affecting drive adjustment.

4.1.9 Startup Counter Drive and Assembly

The startup counter drive allows remote control of the startup chamber with respect to the reactor core. The startup counter drive mechanism is identical to the servo regulating rod drive mechanism except that operating control is not performed in an automatic mode and hence the startup counter drive is not equipped with the servo motor. The mechanism, which is geared to operate at the drive speed of 6.8 in./min, enables the startup counter assembly to be positioned anywhere along its 31-inch stroke. A position indicator in the reactor control room indicates the chamber position.

The startup counter assembly consists of a tubular aluminum rod connected to the startup counter drive shaft and a coupling adapter which in turn is connected to the startup counter container. The connections between the drive shaft, coupling, and the container are made with 3/16-inch diameter by 2-inch long stainless steel cotter pins. The cotter pins facilitate disconnection of the drive and startup counter without affecting drive adjustment. The startup counter container provides mounting for a proportional counter and its related connecting hardware.

4.1.10 Reactor Pool (Figure 4.8)

The reactor pool is comprised of two principal sections, a stall pool and a bulk irradiation pool. Each pool is equipped with primary cooling connections necessary for operation at rated power. The overall dimensions of the reactor pool are approximately 31 feet deep by 32 feet long. The pool walls are constructed of heavy aggregate concrete and ordinary concrete as required to provide adequate biological shielding. Penetrations to the pool are seal-welded to a 1/4-inch thick aluminum liner. There are no penetrations made in the pool floor. Penetrations in the pool wall include six beam ports, the unused medical embedment, thermal column, and hot cell. These are discussed in Paragraph 4.3 of this report. Fuel storage racks are provided along the walls of the stall pool and bulk irradiation pool as discussed in Paragraph 6.1.2.

High power operation is conducted in a section of the stall pool approximately 8 feet in diameter and 31 feet deep. The beam ports, pneumatic tubes, and thermal column are located in the stall pool. The stall pool is separated from the bulk irradiation pool by a pool divider gate. The pool divider gate is constructed

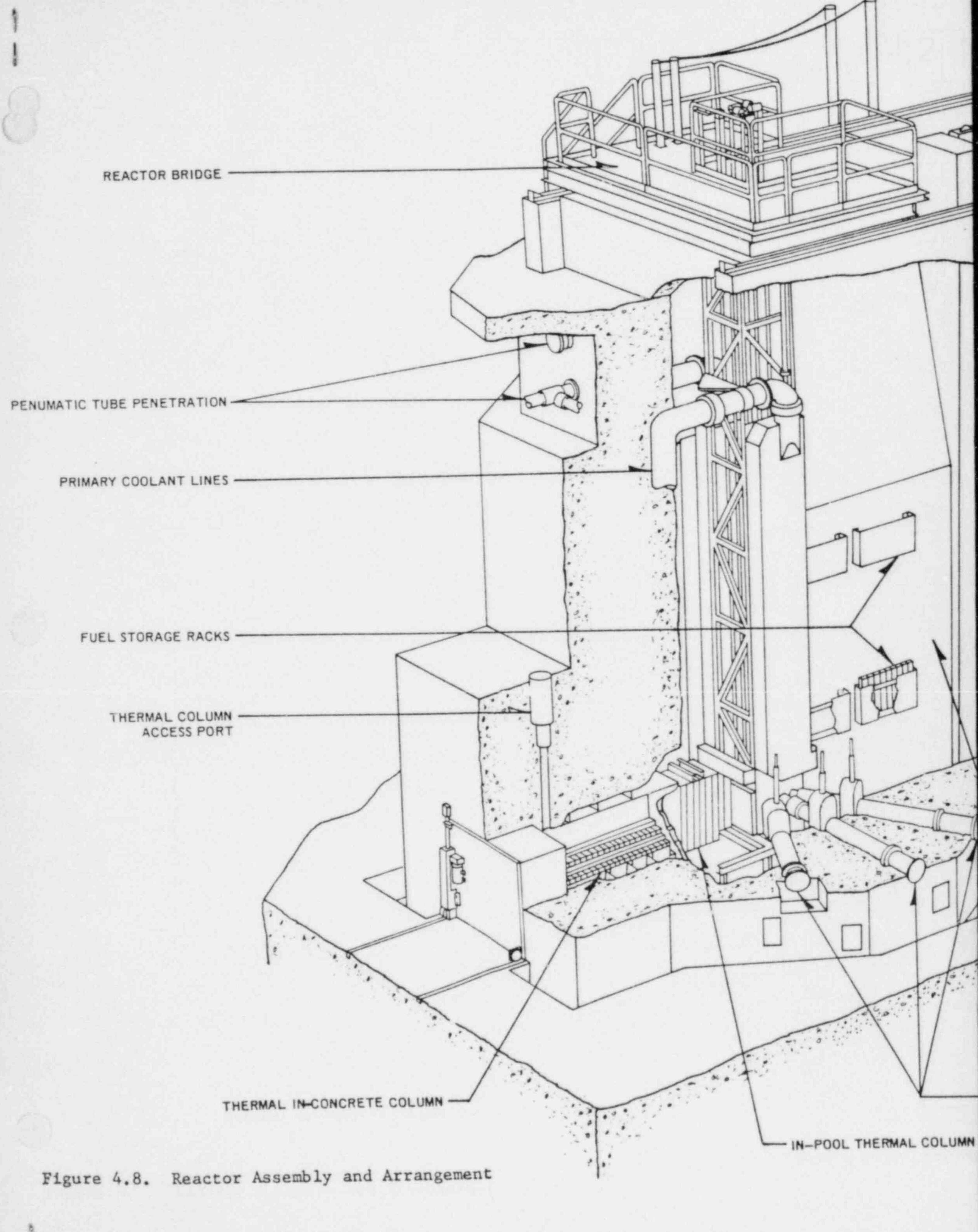
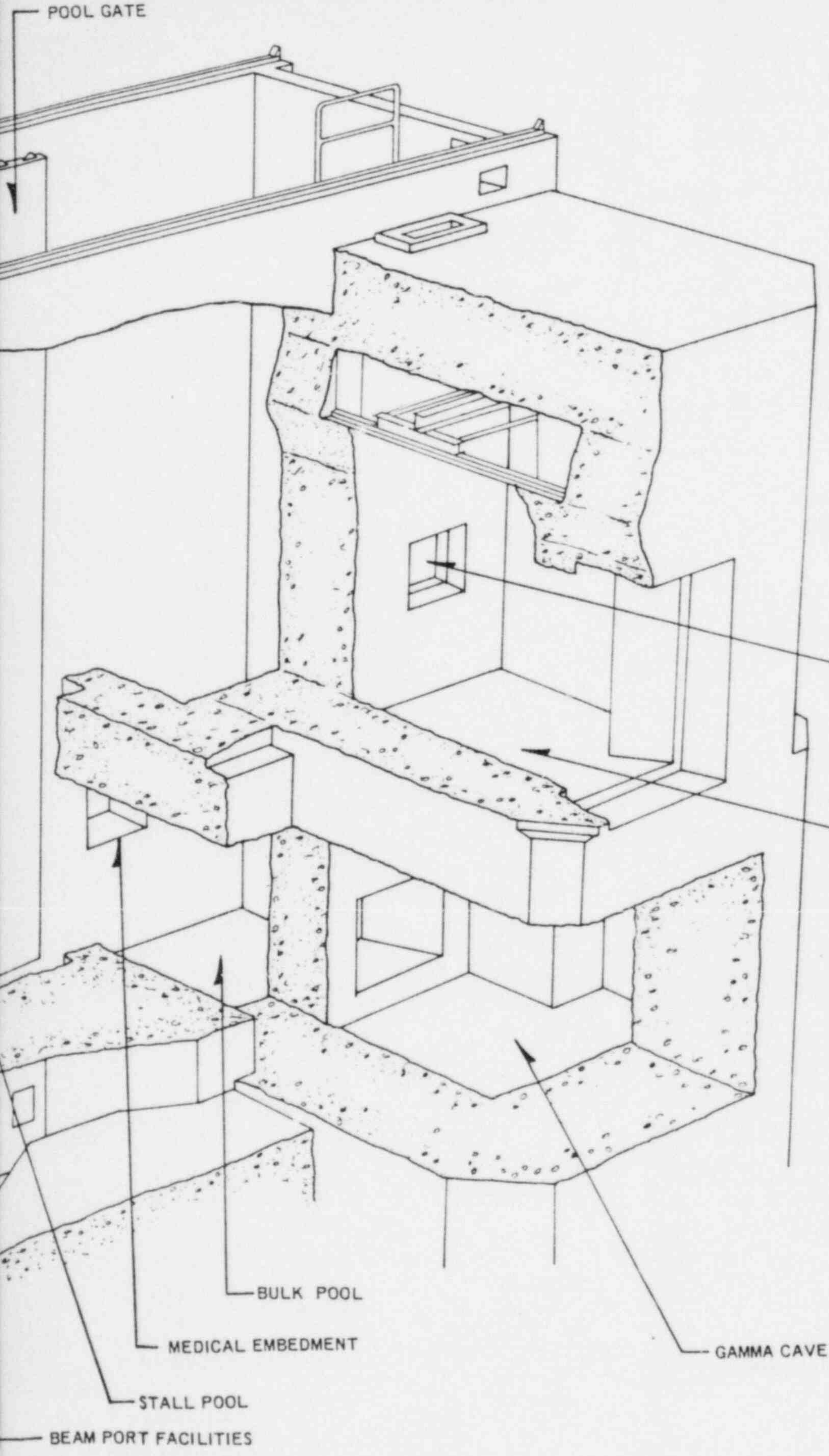


Figure 4.8. Reactor Assembly and Arrangement


POOL GATE
TRANSFER PORT
HOT CELL
BULK POOL
MEDICAL EMBEDMENT
GAMMA CAVE
STALL POOL
BEAM PORT FACILITIES

of aluminum and is approximately 5-1/2 feet wide by 27 feet high. A rubber gasket located about the edges of the gate provides a watertight seal. The gate permits independent drainage of either pool section.

4.1.11 Reactor Bridge (Figure 4.8)

The reactor bridge is provided as a means of supporting the reactor core and core suspension frame, as well as serving as an access platform. The reactor bridge spans the entire width of the reactor pool. The bridge consists of two separate sections of structural frame-work set horizontally one above the other and supported on each side of the pool by a two-wheel, rail-mounted truck assembly. The truck assembly allows the reactor bridge to be positioned at the desired location within the reactor pool. The upper section, or upper bridge, is supported independently over the bridge wheels and allows easy access to all the reactor. The lower section, or lower bridge, supports the weight of the suspension frame and core.

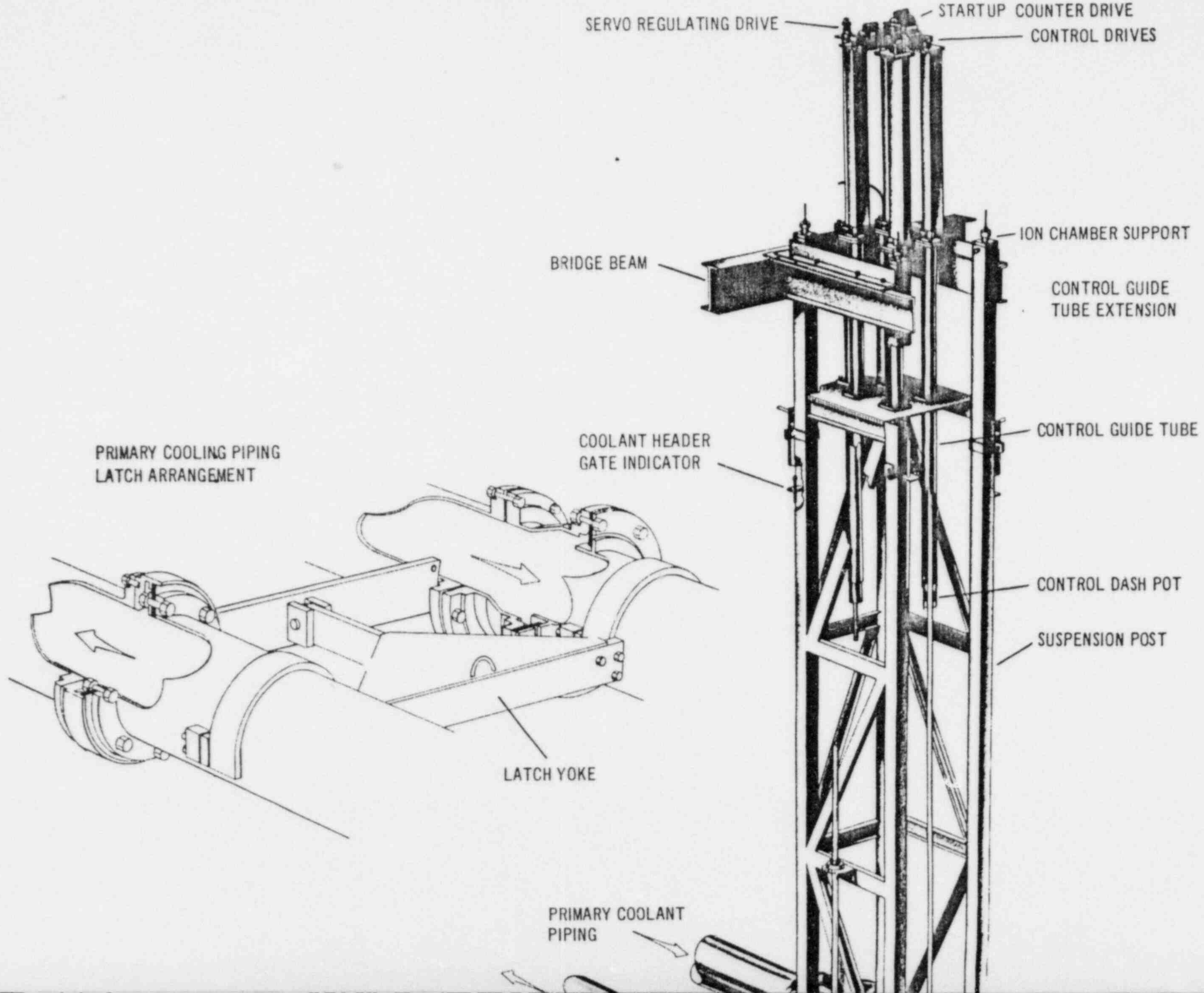
A hand crank and gear drive are provided for moving the bridge on the pool rails at a rate of approximately 1-1/2 inches per full turn. The bridge is equipped with a brake assembly to allow securing the bridge in the desired operating or inspection position. The bridge is interlocked to prevent any movement while the reactor control blades are withdrawn. Limit switches are provided to ensure that power operation above 100 kW is limited to the high-power operating positions at each end of the reactor pool.

4.1.12 Core Suspension and Support (Figures 4.8 and 4.9)

The core suspension frame is suspended from the lower reactor bridge. The suspension frame is an aluminum rectangular column built of four square corner posts forming a rigid structure. The core box is attached to the lower end of the suspension frame. Cross braces and stiffeners are utilized to provide structural rigidity and alignment in the upper half of the suspension frame. Coolant flow channels are used to provide this function in the lower half of the suspension frame. Stiffeners are provided on three sides of the frame while the fourth side is open to provide access to the core. The open side of the frame also allows manipulation of fuel elements during refueling operations.

The manipulation is performed using a manual grapple thus enabling the operator to position an element in any one of the spaces in the grid box. An ion chamber is located in each of three corner posts of the suspension frame. The startup counter is located in the fourth corner post of the frame. The ion chambers are electrically insulated from the suspension frame to elminate possible ground current effects. The three ion chambers are suspended from the corner posts by cables and held in place by cable clamps at the bridge level. Positioning of the ion chambers is performed manually. A locating plate which spans the upper end of the suspension frame serves as a mounting for the startup drive, the servo regulating rod drive, and the control blade drives. The control blade guide tubes are flanged to the bottom of the locating plate.

The core suspension system is designed primarily to support


SERVO REGULATING DRIVE
STARTUP COUNTER DRIVE
CONTROL DRIVES
ION CHAMBER SUPPORT
BRIDGE BEAM
CONTROL GUIDE TUBE EXTENSION
CONTROL GUIDE TUBE
COOLANT HEADER GATE INDICATOR
PRIMARY COOLING PIPING LATCH ARRANGEMENT
CONTROL DASH POT
SUSPENSION POST
LATCH YOKE
PRIMARY COOLANT PIPING

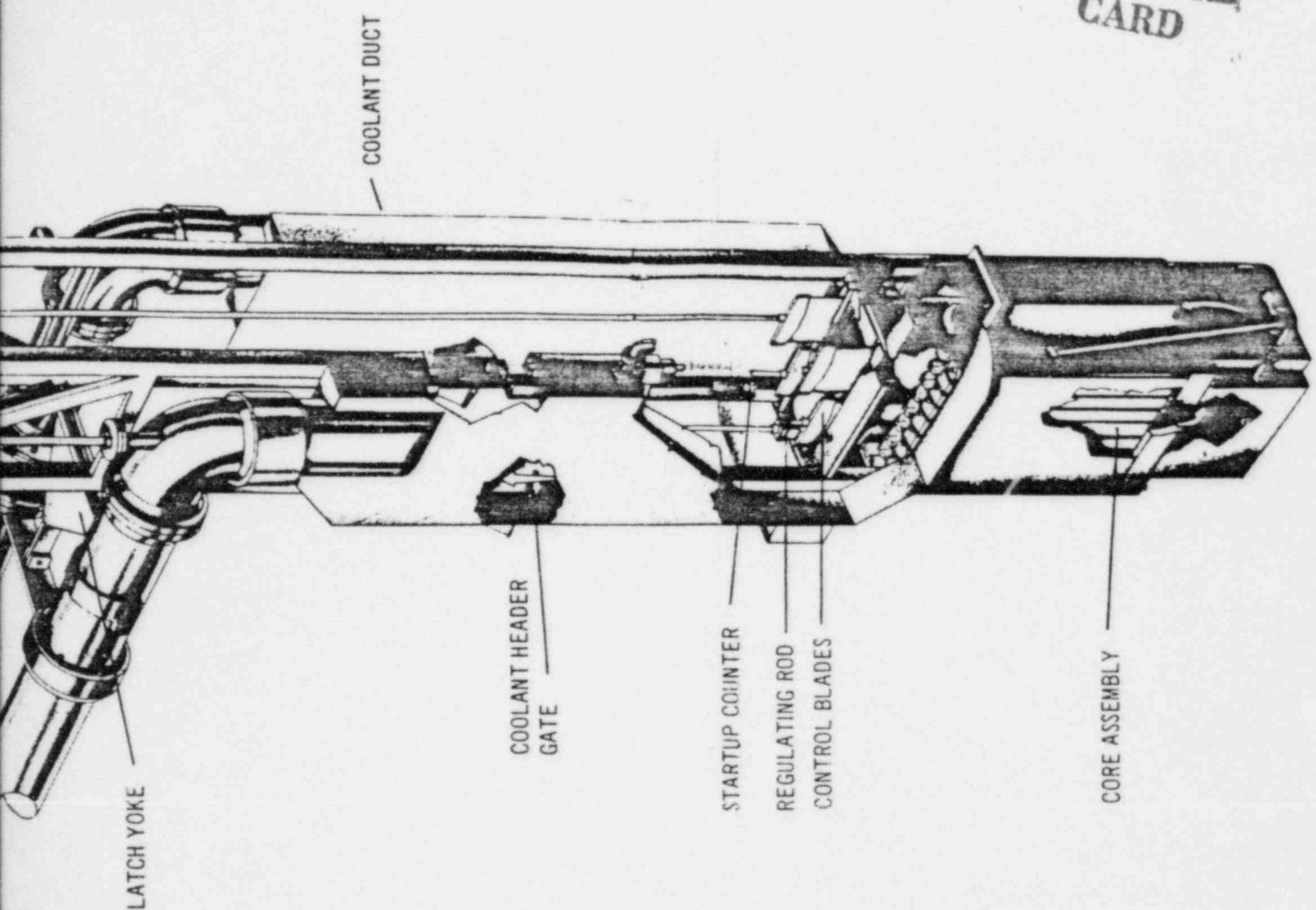


Figure 4.9. Core Suspension

the reactor core, to provide the means for moving it along the major axis of the pool, and to secure the core in any desired operating or service positions, including one readily reproducible position with respect to the coolant header and experimental facilities. The core suspension is also arranged to ensure positive alignment of the shafts between the control and servo drive mechanisms and the respective driven elements as well as facilitating access to the core for servicing elements and experiments.

4.2 REACTOR COOLING AND WATER CONDITIONING SYSTEMS

4.2.1 General Description

The figures on the following pages show the reactor pool and details of the cooling systems (Figures 4.10 and 4.11).

The systems include:

(1) Primary Coolant System which transfers heat from the reactor to the Secondary Coolant System at the heat exchanger.

(2) Secondary Coolant System whose water carries heat received at the heat exchanger to the cooling tower.

(3) Makeup Demineralizer System which provides demineralized water to fill the pool and compensate for operating losses.

(4) Cleanup Demineralizer System to remove impurities from the pool water.

(5) Waste Disposal System for removal of reactor water and for storage of hot waste.

4.2.2 Primary Coolant System

The primary coolant system functions to remove heat and maintain core temperature below a predetermined level. The

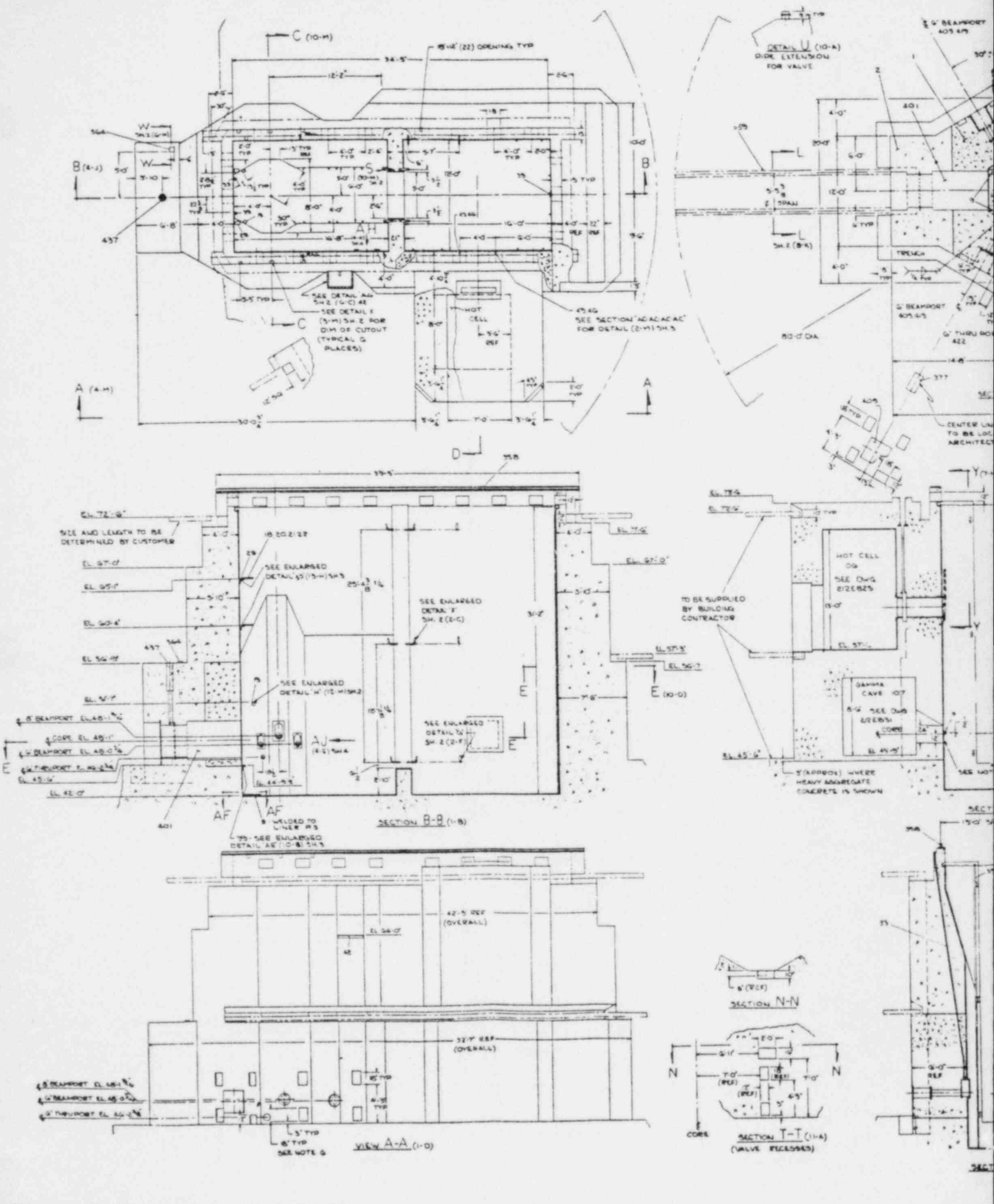
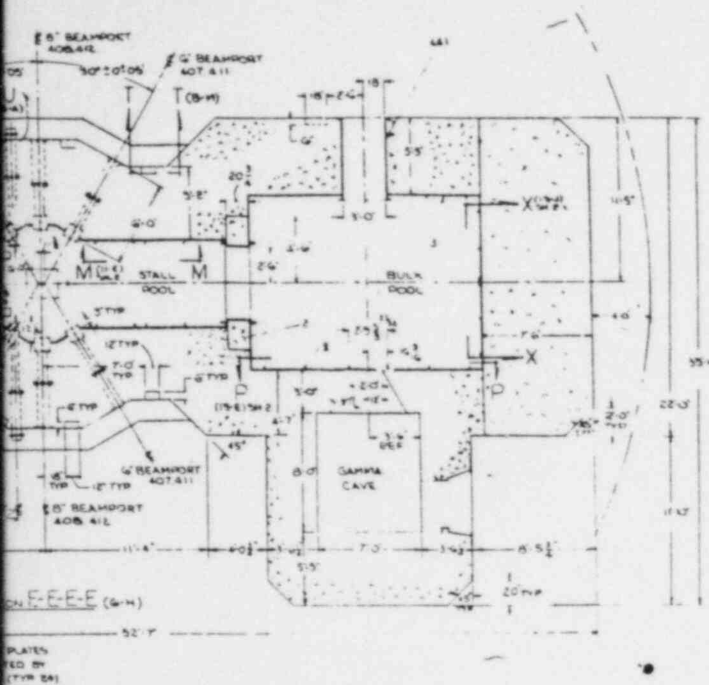
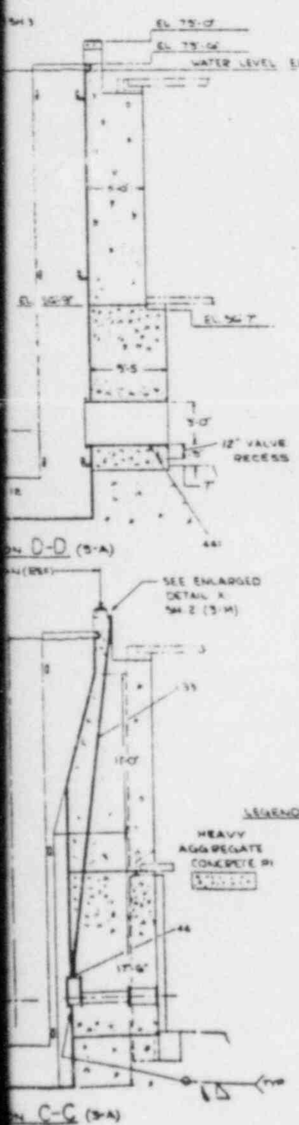
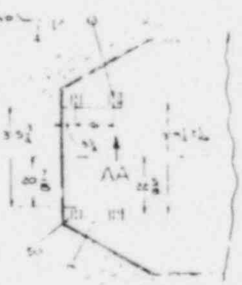


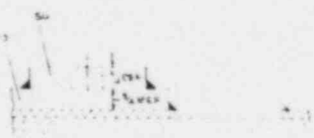
Figure 4.10. Reactor Pool Outline







VIEW AF-AF
SHOWING LOCATION OF STUD PLATE PT 50



VIEW AA

NOTES

1. REINFORCE CONCRETE UNDER REACTOR BRIDGE RAILS FOR 9000 LB LOAD TOTAL WEIGHT OF LOADED REACTOR BRIDGE
2. REINFORCE CONCRETE UNDER THERMAL COLUMN DOOR RAILS FOR 85,000 LB LOAD TOTAL WEIGHT THERMAL COLUMN DOOR
3. LINER TO BE INSTALLED LEVEL AND PLUMB. ALL JOINTS AND LINER PENETRATIONS TO BE SEALED WATER TIGHT. ANY PLATES AND COLLARS NECESSARY TO SEAL THESE JOINTS AND PENETRATIONS TO BE SAME ALLOY AS LINER
4. ALL POOL DIMENSIONS ARE TO INNER FACE OF POOL LINER
5. ALL CONCRETE REINFORCEMENT OR OTHER STEEL SUPPORTS TO BE CLEAR AND FREE OF IN CONCRETE ALUMINUM PIPING ETC
6. THESE RECESSES FOR BEAMPORT VALVES
7. FOR 3 DIGIT NUMBERS SEE MASTER PARTS LIST 105/553. FOR ALL OTHER PART NUMBERS SEE PL OF THIS DWG
8. TOP OF RAILS TO BE LEVEL WITHIN 1/64. RAIL GAGE (PARALLELISM) TO BE MAINTAINED WITHIN 1/32
9. ALL ALUMINUM WHICH COMES IN CONTACT WITH CONCRETE SHALL BE PAINTED WITH TWO COATS OF BITUMASTIC SUPER SERVICE BLACK KOPPERS CO INC, PITTSBURGH PA - OR APPROVED EQUIV
10. ALL CONCRETE SURFACES ON INTERIOR OF POOL WHICH ARE NOT COVERED BY ALUM LINER SHALL BE PAINTED WITH 6 COATS VINYL RESIN (AMERCOAT NO. 33) 6 1/2 MILS THICK AS MANUFACTURED BY AMERCOAT CO. OR ENG. APPROVED EQUIV
11. ALL LINER PENETRATIONS TO BE DYE PENETRANT CHECKED AS PER CUSTOMER SPECIFICATION
12. REINFORCE WINDOW WITH ALUM PLATE. NO RIB STRUCTURE ALLOWED MUST BE FLUSH WITH INSIDE OF POOL LINER

LEGEND

HEAVY AGGREGATE CONCRETE P1

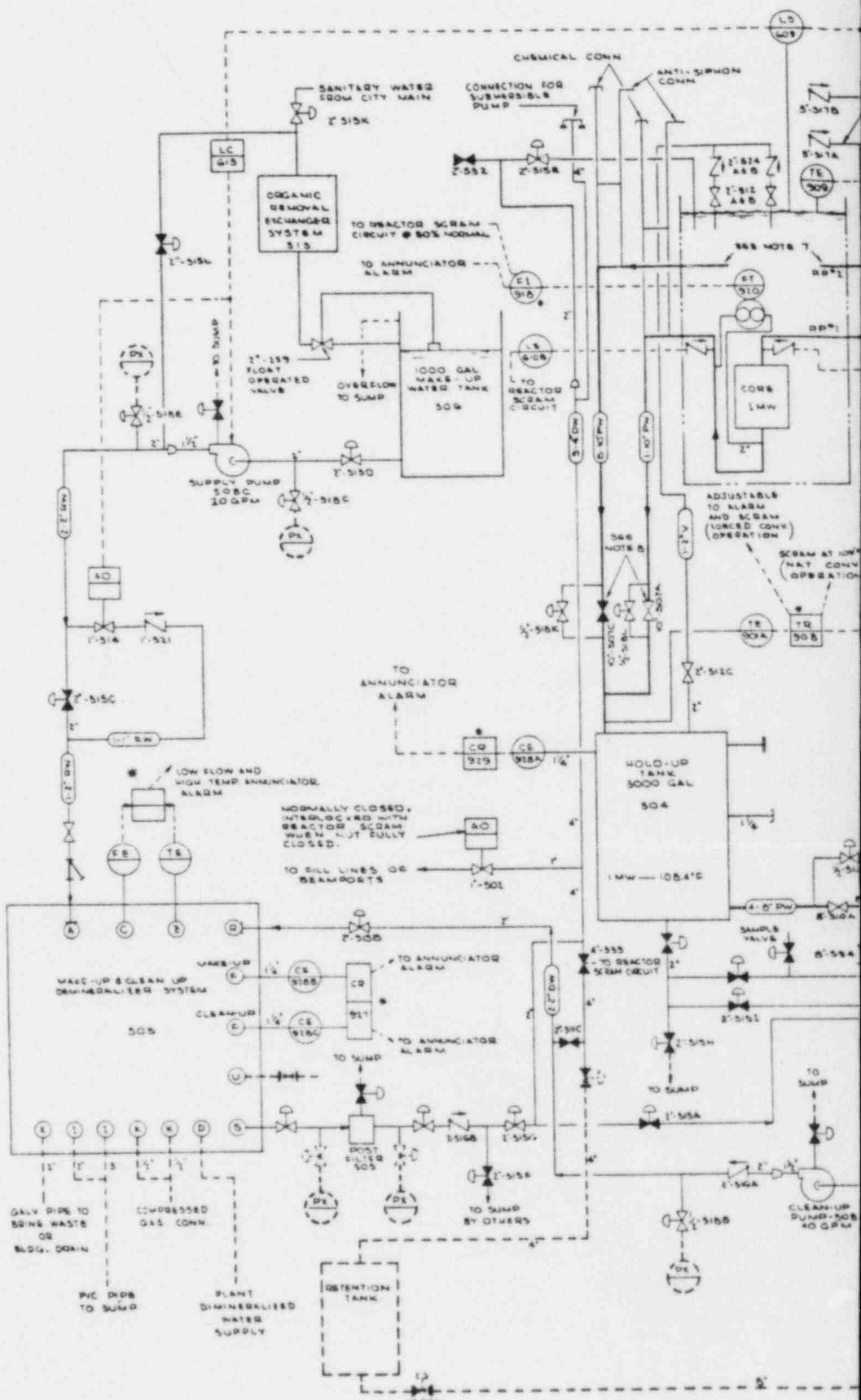
ORDINARY CONCRETE P2

SCALE 1/4"=1'

TI APERTURE CARD

Also Available On Aperture Card

8502280372-07



CHEMICAL CONN
ANTI-SIPHON CONN
SANITARY WATER FROM CITY MAIN
CONNECTION FOR SUBMERSIBLE PUMP
ORGANIC REMOVAL EXCHANGER SYSTEM 515
TO REACTOR SCRAM CIRCUIT @ 50% NORMAL
TO ANNUNCIATOR ALARM
FLOAT OPERATED VALVE
OVERFLOW TO SUMP
1000 GAL MAKE-UP WATER TANK 506
TO REACTOR SCRAM CIRCUIT
SUPPLY PUMP 508C 20 GPM
CORE 1MW
ADJUSTABLE TO ALARM AND SCRAM (FORCED CONV OPERATION)
SEE NOTE 8
TO ANNUNCIATOR ALARM
HOLD-UP TANK 3000 GAL 504
1MW — 108.4°F
LOW FLOW AND HIGH TEMP ANNUNCIATOR ALARM
NORMALLY CLOSED, INTERLOCKED WITH REACTOR SCRAM WHEN NOT FULLY CLOSED.
TO FILL LINES OR BEAMPORTS
SAMPLE VALVE
MAKE-UP & CLEAN UP DEMINERALIZER SYSTEM 505
MAKE-UP
CLEAN-UP
TO SUMP
POST FILTER 505
TO SUMP BY OTHERS
CLEAN-UP PUMP 508 40 GPM
GALV PIPE TO BRINE WASTE OR BLDG. DRAIN
COMPRESSED GAS CONN.
PVC PIPE TO SUMP
PLANT DEMINERALIZED WATER SUPPLY
RETENTION TANK

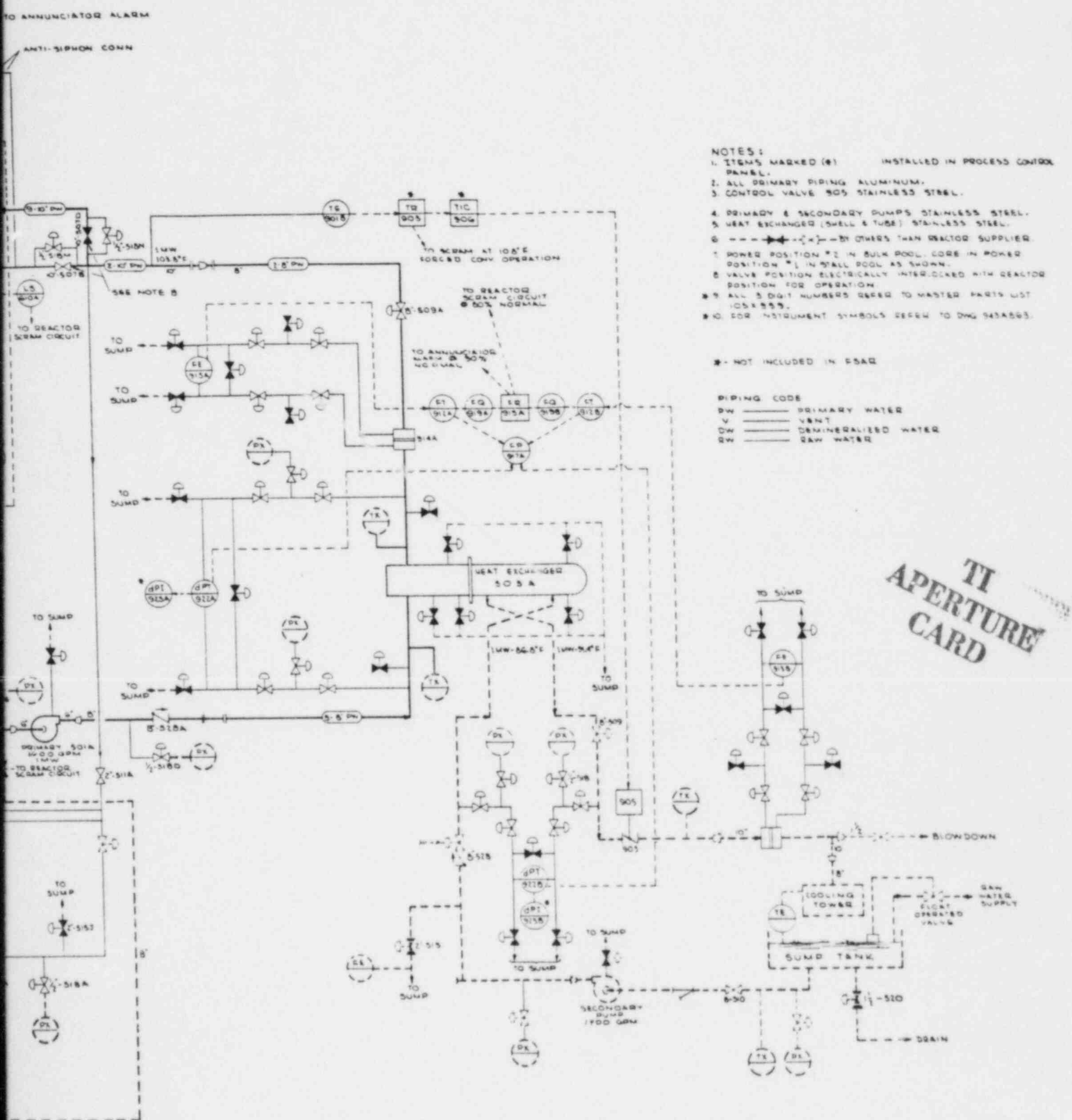


Figure 4.11. Piping and Instrumentation Diagram

primary coolant is moved through the reactor core, through a holdup tank to decay N^{16} and O^{19} isotopes, and then through a heat exchanger and back to the core in an essentially closed system. Primary coolant connections are located at each of the two high power positions within the reactor pool, power position No. 1 in the stall pool and power position No. 2 in the bulk pool. Normal high power operation of the reactor occurs at power position No. 1. At this position, two modes of core cooling are available. In the downcomer mode, the primary coolant systems valves are aligned so that the cooling water is supplied to the core by a 10-inch aluminum line connected to the inlet flow channel forming one side of the suspension frame. Water is fed from the flow channel into a plenum over the core. From here it is forced downward through the core flow channels. In the cross-stall mode, the primary coolant valves are aligned so that the cooling water is supplied by the primary inlet piping in the bulk pool. The cool water then transverses the pool and is drawn downward throught the core flow channels and exits the pool through the stall pool core outlet plenum. The latter of these two cooling modes is preferred due to reduction in core vibration in the cross-stall mode. All core components are fitted with end boxes which fit into holes in the grid plate. This grid plate provides support and orientation for these components. It should be noted that the grid plate is specifically designed for high leakage and minimum restriction of flow. The grid plate is located at the bottom of the grid box which provides the basic flow channel for the entire core.

The side plates on a fuel element are longer than the fuel plates. This additional length provides an opening , or plenum, above and below the fuel zone. Core flow is directed from the plenum above the fuel plates to the plenum below through the core flow channels established by:

(1) the spacing of fuel plates in each fuel element

(2) the space between side plates maintained by the positioning pads on each side plate,

(3) the space between the fuel bearing plates of adjacent elements which is also maintained by the positioning pads.

Proper orientation of the fuel elements is accomplished by two chamfered edges on the fuel element end boxes which fit into matching openings in the core grid. The end boxes only approximately align the fuel elements in the grid. Final alignment and spacing is assured by pads embossed in each corner of the side plates. All other core components have similar positioning pads. Operation under forced circulation requires a full grid box to maintain flow channels and patterns in the core. The control blades operate within permanent shrouds eliminating both contact with the fuel and mechanical interference between moving control blades and core components. In addition to the primary flow through the fuel elements, a small percentage of primary coolant flow passes into the shrouds through holes in the shroud in the plenum area, and then passes downward between the shroud and the control blades. All water then passes into the outlet flow channel which forms the opposite side of the suspension frame, and from there passes through a 10-inch aluminum discharge line back to the primary pump and heat exchanger.

A controlled volume of water is allowed to flow from the pool into the primary loop through the core via the cleanup system. This flow is controlled by the cleanup pump at 40 gpm and it is returned to the pool through a 2-inch aluminum line after demineralization and filtration. The pool and primary coolant loop are filled and kept to normal operating level by the makeup demineralizer system, which is described in Paragraph 4.2.4.

The primary coolant connection between the suspension frame and the 10-inch (in concrete) primary coolant lines is made by a pair of male-female gasketed couplings. The couplings seal water tight by the application of a slight differential pressure between the coolant line and the pool. This differential exists when the reactor is being operated using forced circulation. Since the reactor may be operated in either the No. 1 position in the stall pool, or the No. 2 position in the bulk pool, electronic interlocks are provided to ensure correct line up of the primary coolant lines for the particular coolant mode desired. Other interlocks prevent operation above 100 kW unless forced circulation flow has been established.

An automatic antisiphon line is provided in the discharge line of the primary coolant loop. This line is equipped with a break valve located above the reactor pool water level. The break valve will normally be closed by pressure during forced circulation. Loss of pressure will cause the break valve to open. A second antisiphon open standpipe line is connected to the return primary coolant line. Both antisiphon risers which connect to the primary system are 3-inch aluminum pipes imbedded in the concrete, which are connected to the 10-inch (in-concrete) primary lines. The antisiphon lines extend out of the concrete just above the surface of the reactor pool water. In the event of a primary piping failure, the pool will start to siphon. After the pool level has dropped to the level of the primary pipes, air admitted to the antisiphon risers will prevent a continuing siphon effect.

At power levels up to 100 kW, the reactor may be operated any place in the reactor pool when core cooling is maintained by natural convection. When the pool temperature reaches approximately 107°F, the power must be reduced, or cooling with the primary cooling system should be introduced. Primary coolant temperature is measured by a temperature sensor located in the primary coolant loop upstream from the holdup tank. Pool water temperature is measured by a temperature element in the pool.

4.2.2.1 Primary Pump A primary pump is located in the primary loop between the holdup tank and the heat exchanger in a section of piping which can be isolated from the system for maintenance or repair by a manually-operated 8-inch gate valve and an 8-inch check valve. The primary pump is a horizontally-mounted centrifugal pump capable of delivering 2000 gpm (design value) through the primary loop. The pump casing, shaft and impeller are made of stainless steel. The 50 HP motor is a drip-proof, induction type operating on a 440-volt 3-phase, 60 hz electrical system. The location of the primary pump, between the holdup tank and the heat exchanger, results in minimizing the radiological effects of any pump seal leakage. It is anticipated that leakage through pump seals will be of relatively small volume, and the hazard attendant to such leakage will be negligible as a result of the holdup time provided by the holdup tank for decay of short half-life radioisotopes.

4.2.2.2 Heat Exchanger A shell- and tube-type heat exchanger, constructed of Type-304 stainless stell and capable of removing 2.5 MW of heat, is provided with the primary cooling loop. The unit is of the U-tube type, with the tubes rolled and seal welded to the tube sheet. The exchanger was supplied by Southwestern Engineering Co., Los Angeles, California, and complies with Section VIII of the ASME Boiler and Pressure Vessel Code. The factory specified design conditions of the heat exchanger are shown in Table 4.1.

Table 4.1

HEAT EXCHANGER OPERATING CONDITIONS

Item	Primary (Tube) Side	Secondary (Shell) Side
Inlet Temperature (^{o}F)	116	80
Outlet Temperature (^{o}F)	104	91.4
Flow Rate (gpm)	1600	1500
Operating Pressure (psig)	50	50
Pressure Drop (psi)	6	10

4.2.2.3 Holdup Tank The holdup tank is an aluminum tank with a 3000 gallon capacity. The tank is provided to allow decay of short-life N^{16} and O^{19} isotopes produced in the primary coolant. The holdup tank is provided with a vent. The vented air is bled off under water in the reactor pool. Significant radiation levels as a result of N^{16} decay will be extant in the vicinity of the holdup tank.

4.2.3 Secondary Cooling System

The secondary cooling loop functions to transfer heat from the primary coolant in the heat exchanger to the atmosphere at the cooling tower. The secondary cooling water is neither activated by direct contact with the reactor core, nor contaminated by mixture with primary coolant in the heat exchanger; therefore, the heat being carried by the secondary cooling water can be reasonably and safely dissipated by evaporation of a portion of the water in a conventional cooling tower. A substantial amount of makeup water is required to replenish the resulting loss to the atmosphere. As evaporative losses tend to augment the concentration of solid impurities in the system, a small percentage of the circulating coolant may be continuously removed by blowdown. Losses due to evaporation and blowdown are automatically made up through a float-actuated valve from the city water supply. The secondary cooling loop includes the heat exchanger (as previously described) and temperature control valve, the cooling tower with cooling tower basin, a secondary pump, and required flow sensor.

4.2.3.1 Cooling Tower The cooling tower is a forced draft, vertical discharge single-flow tower designed to cool 1465 gpm from 90°F to 80°F at a 72°F wet bulb temperature.

There are two fans with positioning inlet dampers for temperature control. A motor-operated valve can be used to control the primary coolant temperature by throttling the secondary coolant flow. This valve is controlled from the reactor control room.

4.2.3.2 Secondary Pump The secondary pump is located between the cooling tower and the heat exchanger in a section of piping which can be isolated from the system for maintenance and repair by a manually operated 8-inch gate valve and an 8-inch check valve. The secondary pump is a single-stage, 40 H.P. centrifugal-type capable of delivering 1500 gpm. The pump has a steel casing, impeller and shaft. The pump is driven by a 440-volt, 3-phase, 60 hz, drip-proof induction-type motor.

4.2.3.3 Secondary Makeup and Chemical Treatment Raw makeup water is admitted to the cooling tower basin by a float-operated valve set to maintain a fixed water level in the basin. Chemical treatment of the secondary coolant is used to protect the various materials of the system and to control the growth of algae.

4.2.4 Makeup System

The makeup system provides the water required to fill and maintain the reactor pool at the proper level. A float-operated valve admits water from municipal sources through an organic removal exchanger through an air gap to a 1000-gallon makeup water tank. A supply pump delivers water from this tank to a mixed-bed (single shell) regenerative demineralizer and then to the cleanup system upstream of the postfilter as described below. The demineralizer, containing approximately 12 cubic feet of nuclear grade high capacity resins, can deliver up to 20 gpm of very high purity effluent containing less than 1 ppm total dissolved impurities. Chemical Tanks and valves required for regeneration are furnished with the demineralizer.

4.2.5 Cleanup System

The cleanup system removes impurities that enter into the reactor pool including those resulting from water reacting with its environs and those caused by mechanical wear and damage. The cleanup system includes a pump, as mixed-bed (single shell) regenerative demineralizer designed for a normal 40 gpm flow rate, and a post filter. Manual shutoff and check valves control flow. Water is removed from the primary cooling loop after the heat exchanger, cleaned and returned to the pool.

The cleanup demineralizer is a rubber-lined, carbon-steel tank with attendant stainless steel fine mesh filter installed in the effluent piping to prevent resin fines from entering the effluent stream. The pool cleanup demineralizer will produce an effluent having a specific resistance exceeding 1×10^6 ohm-cm when treating an influent containing approximately 1.0 ppm of ionizable impurities. Regeneration is accomplished by completely isolating the demineralizer from the system.

4.3 EXPERIMENTAL FACILITIES

Experimental facilities make the radiation produced by the reactor available for experimental work and are designed to accomplish this without jeopardizing the safety of personnel. The experimental facilities provided in the stall pool section include a thermal column, two 8-inch and four 6-inch beam ports and two pneumatic tubes. In-core radiation baskets are provided which may be located in the periphery of the core. In the bulk irradiation section of the pool, a dry gamma radiation facility, a hot cell, and an unused medical facility are incorporated in the walls of the pool. All of the experimental facilities are exhausted to the 100 foot plant stack after the air is filtered by one or more prefilters and absolute filters as described in Paragraph 3.3.2.

4.3.1 Thermal Column

The thermal column provides neutrons in the thermal energy range for experimental application. The thermal column is comprised of two separate assemblies. One assembly, a 4 x 4 foot square column, is embedded within the reactor pool biological shield. The other assembly, the thermal column extension, is located between the pool liner and the nuclear core, and is supported by a structural member which is firmly attached to the pool structure.

The pool liner is continuous in front of the thermal column penetration of the pool wall, and the pool liner is reinforced at the opening by a 1-1/4 inch thick aluminum plate. The center line of the thermal column is aligned with the center line of the core when the core is located in the stall pool.

The 4 x 4 foot square embedded column is filled with graphite. Some center graphite stringers are removable to provide for experimental samples. A lead gamma shield, located on the front end of the thermal column extension, is cooled by natural convection to the reactor pool.

The section embedded in concrete is encased in a double shell through which cooling water passes by natural convection. Heat removal from the thermal column must be satisfactory to hold the graphite below that which would have any deleterious effects on the graphite or its container. A generally recognized temperature limit for graphite is 800°F (420°C) below which there is negligible oxidation. The calculated operating temperature of the thermal column graphite is below 200°F at 5MW. This operating temperature is materially reduced at the present operating level of one megawatt.

A heavy, steel thermal column door is provided as a shield to protect operation personnel against gamma radiation. The thermal column door moves on rails, set into the concrete floor perpendicular to the shield face, by means of a 110-volt, single-phase, 60-hz, 1/2-hp ratiomotor which drives two of the four door wheels. The drive motor is operated with a door-mounted starter switch. The drive motor will move the door at a rate of 1 ft/min in either open or close travel. Four access ports are provided in the face of the door. Each port is fitted with four separate boral-faced plugs. Each plug is drilled and tapped to accommodate a plug removal tool which is used to insert and remove plugs. A 2-foot deep experimental air chamber between the face of the graphite and thermal column door provides location for air, water and electrical service connections to the biological shield face. A line common with other facility vent lines is located in the floor at the center of the graphite block, and serves as a drain and ventilation line to remove condensation and radioactive gases. A scram interlock prevents reactor operation when the thermal column door is opened.

4.3.2 Beam Ports (Figure 4.12)

Two 8-inch beam ports and four 6-inch beam ports are included in the reactor experimental facilities. The beam ports provide leakage neutrons with energies throughout the fission energy spectrum for experimental application. Each beam port is an air-filled aluminum tube extending from the reactor core face nearly through the biological shield and then flange-coupled (with a nonconducting plastic gasket) to a stainless steel end fitting which extends through the shield face and terminates at a flange for experimental access. Radiation protection is provided by a lead shutter and removable shield plugs. The shield plugs are aluminum castings filled with ferro-phosphorous concrete, lead, and steel. The plugs are fitted with a spiral-type conduit to accommodate instrument leads. A common drain line is provided for port drainage and ventilation assuring adequate removal of seepage and radioactive gases. Valving on individual lines allows balancing of ventilation throughout the system.

Air, gas, water or experiment lines are introduced to the beam ports through four conduit lines on each beam port extending from the biological shield face to a point within the shield. In addition to the conduit lines, each beam port is fitted with a separately-valved demineralized water line. The water is supplied to the beam ports through a common header which has a solenoid controlled valve interlocked to prevent opening and flooding the ports during reactor operation.

Each beam port has a thimble consisting of an aluminum section extending from the reactor core face to the inner tube to

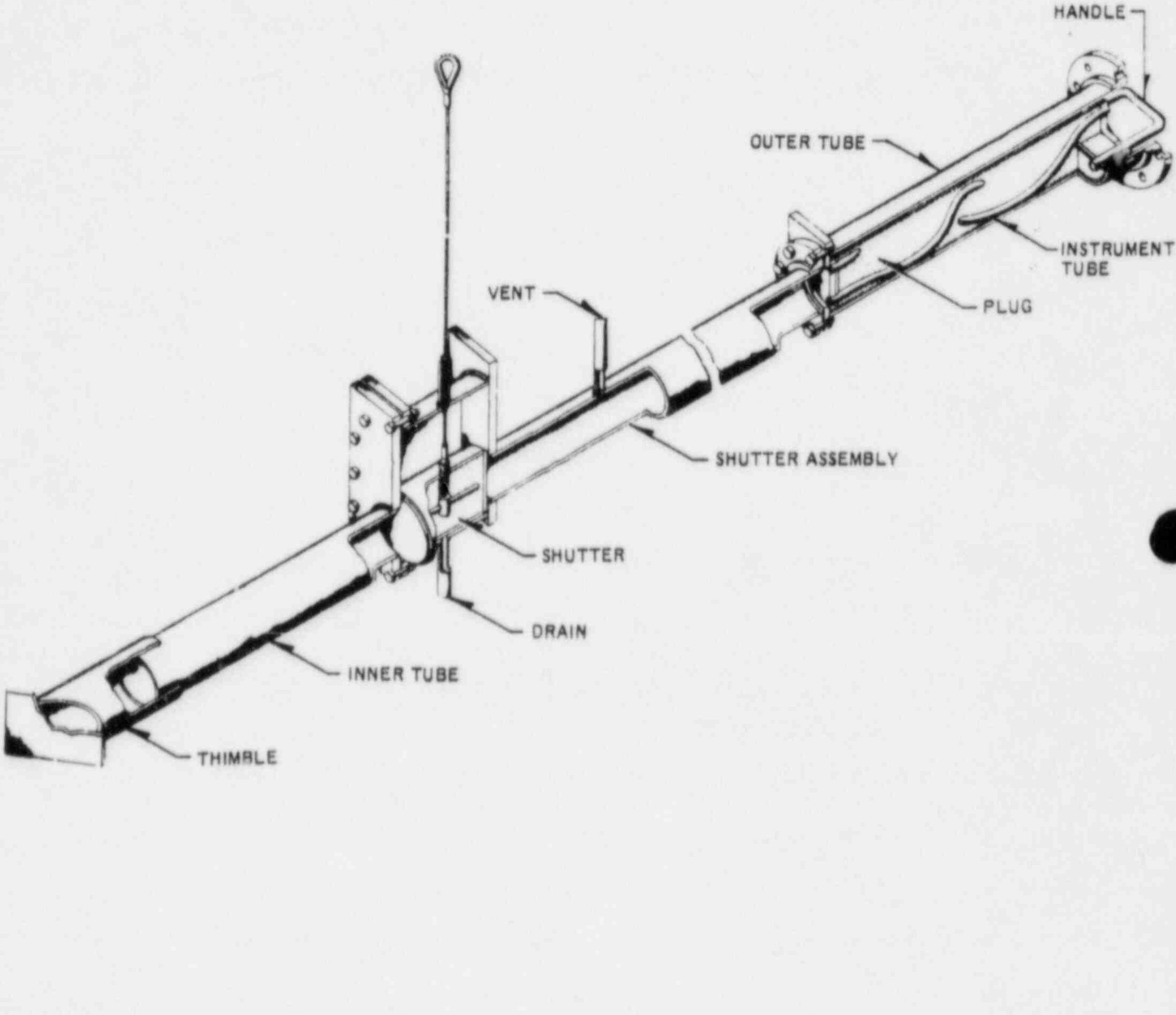


Figure 4.12. Typical Beam Port

which it is welded. The inner tube forms the transition between the thimble and the shutter housing. The inner tube is connected to the shutter housing by a flange-bolted connection.

The shutter housing forms the intermediate portion of the beam port between the inner and outer tubes. The beam port shutter is contained in the flanged housing at the inner end of the shutter housing assembly. The outer tube extends from the flanged connection of the shutter housing assembly to the stainless steel end piece which goes through the outer face of the biological shield. The tube is flange-fitted at both ends.

4.3.3 Pneumatic Tube System

The dual pneumatic tube system rapidly moves small experimental samples from remotely located stations in the facility to symmetrically located terminals in the flux region adjacent to the nuclear core. This operation allows irradiating samples for short, controllable periods, and the irradiated samples of short half-life may then be conveyed to a receiving station. During such operations, the core is located in the stall pool section of the reactor pool. The pneumatic tube system includes the following:

(1) Control System - The control system is operated at 110-volts, 60-cycle normal power. Pilot lights are provided to indicate when the system is in operation , and when a specimen carrier capsule is located at a reactor terminal. The control system can be operated on an automatic mode with a 2-second to 20-minute cycle. The manual mode may be used for a longer operating cycle. Two control units are provided with the

following push-button and selector functions.

a. START-STOP push button and indicating light for centrifugal exhauster ON operation.

b. MANUAL-AUTOMATIC selector switch

c. RABBIT DISPATCH push button

d. MANUAL RETURN push button

e. EMERGENCY RETURN push button

f. RABBIT IN REACTOR indicating light.

(2) Centrifugal Exhauster - A centrifugal exhauster provides the means of creating system vacuum. The speed of the exhauster is variable, allowing the vacuum to be adjusted, thus providing a means of controlling the carrier capsule speed. The exhauster is rated at approximately 230 cfm at 5 in. Hg. The exhauster is connected directly to a drip-proof induction motor rated at 5 hp, 3600 rpm and operating from a 220/440, 3-phase, 60 hz source.

(3) Wind Gate Cabinet - A solenoid-operated wind gate cabinet provides the means of changing the direction of air flow, and thus the means of determining rabbit direction. The wind gate cabinet operates from a 115-volt, single phase, 60 hz source.

(4) Reset Timers - Two reset timers are provided as a means of adjusting the rabbit "In-Reactor" time. The reset timers have a variable 2-second to 20-minute range and operate from a 115-volt, single-phase, 60 hz power source.

(5) Slide Gate Valves - Four manual slide gate valves (or siphon breakers) are located in all lines (near pool exit) leaving the reactor pool.

(6) Transit Tubing - The transit or carrier guide tubes are

formed of 2-1/2 inch OD, NO. 20 BWG (0.035) galvanized steel for all outside of pool concrete penetrations, and of the same dimension aluminum tubing for all inside of pool concrete penetrations.

(7) Receivers - Two lead-lined sealed receivers are provided at two receiving stations. An additional receiver (short-chute) is provided in the hot cell. The receivers are utilized to perform the carrier recovery function.

(8) Deflector - One horizontal deflector is provided in the transit tubing of one side of the dual system. The deflector functions to direct the rabbit to the hot cell or to one of the receiving stations as desired.

4.3.4 Radiation Baskets

Aluminum 3-inch square radiation baskets provide chambers for in-core radiation experiments. The baskets are interchangeable with fuel and reflector elements and are therefore adaptable to placement in any core position. An aluminum shell in each basket provides a chamber approximately 1-7/8 inches ID by 33-inches long to accommodate experimental materials.

An aluminum orifice plate located in the bottom of each radiation basket permits primary cooling water to flow through the basket, and thereby cool the baskets and the experimental contents. A lifting rod is integral with the basket to facilitate handling and lifting with the tweezer grapple which is discussed in Paragraph 6.1.3.

4.3.5 Gamma Cave

A gamma radiation facility, adjacent to the lower section of the bulk pool, is a "dry room" providing opportunity for bulk irradiation of experiments in air. An approximate 2 ft x 2 ft opening is provided in the pool wall with the center line of the opening directly opposite the center line of the core when in the operating position in the bulk pool. The pool liner is continuous in this area, thus keeping pool water from this facility. In addition, a heavy aluminum plate is used to reinforce the pool liner at the opening in the concrete.

Access to the gamma cave is via a heavy hinged door which is padlocked closed during reactor operation. An interlock system prevents access to the gamma cave if unsafe radiation levels exist.

In addition, the gamma cave can be used for gamma irradiation of specimens via spent fuel elements or another gamma emitting source of radiation. As spent fuel elements become available, they may be stored in a storage rack mounted on the gamma cave window.

4.3.6 Hot Cell Facility

The hot cell is located directly above the gamma cave facility. The hot cell is connected to the pool by means of a 2 ft x 2 ft transfer port (cavity) in the pool wall . There is a watertight door at each end of the cavity which may be raised or lowered by an operator at the pool surface level. The doors are interlocked in such a manner that only one door can be opened at a time. A hinged access door with a warning light is provided and is similar in design to the gamma facility access door. The hot cell contains a viewing window, manipulators, bridge crane, and electrical outlets which can be used in the performance of experiments.

4.4 CONTROL AND INSTRUMENTATION

4.4.1 General Features

The controls and instrumentation of the reactor include the following:

(1) Nuclear instrumentation to measure neutron flux at the core, and to supply signals for alarms and scram circuits.

(2) Area radiation monitors outside the pool.

(3) Controls for the safety blades and regulating blade.

(4) Process system instrumentation.

(5) Alarm and indicator system.

Each area and the components within are defined in the following paragraphs. The controls and instrumentation for the reactor are outlined in block form in Figure 4.19. The reactor controls and instrumentation are assembled in two cabinets: the reactor control console and the amplifier cabinets. The process control and motor control center are described separately.

The control console serves as an assembly point for location of operating controls and instruments. The operator is provided with a vantage point from which to conveniently observe reactor performance and adjust operation to varying requirements when needed for tests, experiments, and other operations.

4.4.2 Control Console (Figure 4.13 and Table 4.2)

The control console consists of a desk-type cabinet 71 inches wide, 44 inches high, and 31 inches deep. The controls and instruments required for operation of the reactor are contained in a control panel which slopes upward from the rear of the desk. Located on the right side of the console are the control blade selector switch, manual rundown switch, control blade manual control switch, and the control blade position indicators. Mounted with the position indicators are the power ON indicator, and the blade limit indicators for each control blade.

TABLE 4.2

CONTROLS AND INSTRUMENTS - CONTROL CONSOLE

Description	Reference Symbol (see Figures 4.13 and 4.14)	Function
Control Blade Group		
BLADE SELECTOR Blade No. 1,2,3,4 Four Position Switch	7S4	Selects proper blade to be raised or lowered.
CONTROL BLADE Three Position Manual Control Switch with Spring Return to Off	7S3	IN (Decrease Reactivity) Position completes circuit to energize control blade motor to withdraw blade.
MANUAL RUNDOWN Two Position Switch	7S1	RUNDOWN runs the four control blades down simultaneously. OFF opens switch.
BLADE 1 BLADE 2 BLADE 3 BLADE 4 Digital Position Indicators	7Z1 7Z2 7Z3 7Z4	Indicates blade position to second decimal place with respect to zero at fully inserted position.
IN Blade Inserted Light	DS2	Indicates blade has reached the limit of travel in the direction of decreasing reactivity and is fully inserted in the reactor core.

TABLE 4.2 (Continued)

Description	Reference Symbol (see Figures 4.13 and 4.14)	Function
OUT Blade Withdrawn Light	DS1	Indicates blade has reached the limit of travel in the direction of increasing reactivity and is fully withdrawn from the reactor core.
ON Blade Circuit Energized	DS3	Indicates control blade circuit is energized.
MAG ENG D (Magnet Engaged Light)	DS4	Indicates scram magnet on the control drive is in proper mechanical contact with the armature disc at the top end of control shaft.
Servo-Controlled Regulating Blade Drive Group		
REG BLADE Servo-Controlled Regulating Blade, Three Position Switch Spring Returned to OFF (Manual Control)	9S4	IN (Decrease Reactivity) energizes servo control drive motor to insert blade.
WITHDRAWAL IN INCHES Digital Position Indicator	9Z1	Indicates blade position to second decimal place with respect to zero at fully inserted position.
REGULATING BLADE	S1 (Part of 9Z1)	"MAN" permits manual operation of the servo control drive through the "Reg Blade" contro[l] switch.

TABLE 4.2 (Continued)

Description	Reference Symbol see Figures 4.13 and 4.14)	Function
	S2 (Part of 9Z1)	"AUTO" integrates the servo control drive into the automatic power level channel.
	DS1 (Part of 9Z1)	"IN" light indicates blade has reached the limit of travel in the direction of decreasing reactivity and is fully inserted in the reactor core.
	DS2 (Part of 9Z1)	"OUT" light indicates blade has reached the limit of travel in the increase reactivity direction and is fully withdrawn from the reactor core.
	S4, S5 (Part of 9Z1)	JOG IN and JOG OUT switches allow regulating blade to be raised or lowered in short increments of travel by use of a time-delay relay.
POWER SCHEDULE Three Position Switch Spring Returned to OFF	9S7	RAISE position energizes reversible motor in servo amplifier to drive the wiper of the reference potentiometer in the direction of increasing reactor power output.

TABLE 4.2 (Continued)

Description	Reference Symbol (see Figures 4.13 and 4.14)	Function
		LOWER position energizes reversible motor in the servo amplifier to drive the wiper of the reference potentiometer in the direction of decreasing reactor power output.
% FULL POWER	9M1	Indicates power level desired during automatic operation.
Startup Counter Control		
STARTUP COUNTER Proportional Counter Position Indication	10Z1	IN light indicates counter has reached the limit of travel in the direction of increasing neutron flux, and is in the lowermost position with respect to the core midplane. OUT light indicates counter has reached the limit of travel in the direction of decreasing neutron flux, and is in its uppermost position with respect to the core midplane.
STARTUP COUNTER Three Position Switch	10S1	IN-energizes drive motor in direction to insert proportional counter.

TABLE 4.2 (Continued)

Description	Reference Symbol (see Figures 4.13 and 4.14)	Function
		OUT-energizes drive motor in direction to withdraw proportional counter.
		CENTER indicator light, when lighted, indicates counter is in center position.
Safety Channel Group		
POWER RANGE Picoammeter Switch	12Z1 12Z2	Sets full scale range of 12AR1 or 12AR2.
INDICATORS	12M1 12M2	Indicates output in watts from stable picoammeters.
WARNING Indicator Light	7DS1	Lights when power level is in the last multiplier set range. When lighted do not use next higher range.
Log N Period Channel		
POWER LEVEL Indicator	11M1	Indicates output of log N amplifier.

TABLE 4.2 (Continued)

Description	Reference Symbol (see Figures 4.13 and 4.14)	Function
PERIOD SECONDS Indicator	11M2	Indicates reactor period as obtained by differentiating the logarithmic output of the log N amplifier 11AR1.
Startup Channel		
COUNTS PER SECOND Log Count Rate Indicator	10M1	Indicates output of log CR meter 10AR2 as the logarithm of the fission rate at the counter.
Alarm and Indicator System		
ANNUN. RESET ANNUN. ACKN. Annunciator and Annunciator Lights	5Z1	An abnormal condition in the reactor system is indicated by a buzzer and the relevant light becoming lighted on the amplifier cabinet.
SCRAM RESET ANNUN. TEST		Pressing the Annun. Ackn. button turns off the buzzer. When the difficulty is corrected the annunciator and scram reset switches are pressed, lighting the scram reset light and returning the annunciator lights on the amplifier cabinet to normal dim condition. To test one of the alarm and indicator system points, press the ANNUN. TEST button.

TABLE 4.2 (Continued)

Description	Reference Symbol (see Figures 4.13 and 4.14)	Function
SCRAM Scram Switch	6S2	De-energizes scram relays and produces a relay scram at the discretion of the operator. Relay scram protection is re-instituted by pressing the scram reset button.

The central portion of the control board is occupied by the startup counter indicator panel control, the manual scram switch and the timer. Mounted below and to the left of the timer are the annunciator reset switch, annunciator acknowledge switch, annunciator test switch, and scram reset switch. Below the startup counter control in the center portion of the control panel is the regulating blade position indicator panel and control switch. On this panel are located, below the digital readout, the mode transfer switches, and the regulating element limit indicators. The log count rate (startup channel), log N, period, power schedule, and the power level meters are located on the left side of the control panel. Below these indicators are mounted the picoammeter range switches and power schedule selector switch.

4.4.3 Nuclear Instrumentation Cabinet (Figure 4.14 and Table 4.3)

The amplifier cabinet, located to the left of the control console, serves as an assembly point for amplifiers, scalers, recorders, and other components required for measurement and control of reactor power. The instruments are mounted in a three-bay, relay-rack type cabinet. The upper portions of the

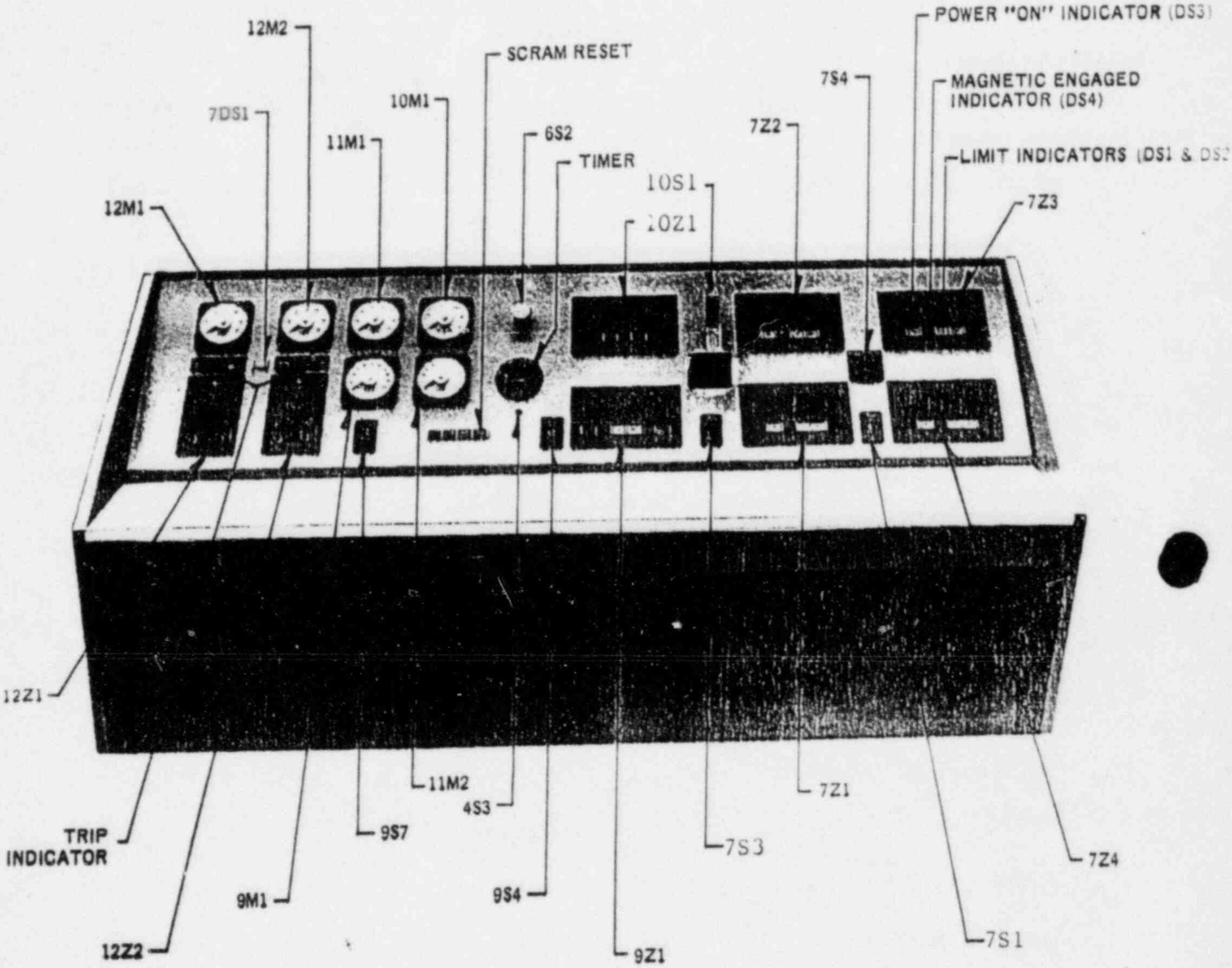


Figure 4.13. Control Console

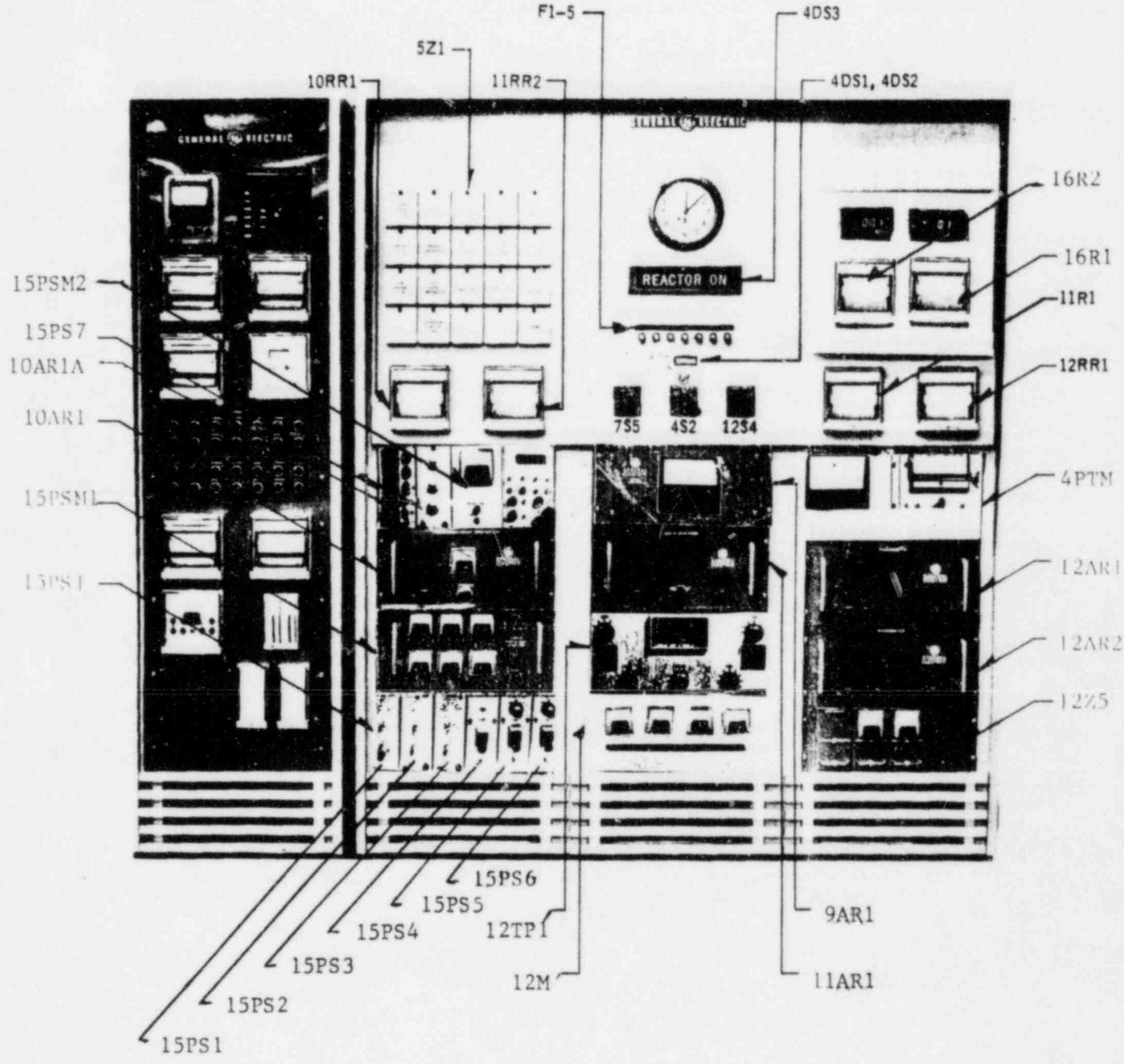


Figure 4.14. Nuclear Instrumentation (Amplifier) Cabinet

TABLE 4.3
CONTROLS AND INSTRUMENTS - AMPLIFIER CABINET

Description	Reference Symbol (see Figure 4.14)	Function
Power Distribution Group		
CONTROL POWER Circuit Breaker (Mounted in rear of cabinet)		Protects reactor control circuit.
115 V AC ON Indicator Light	4DS1	Indicates reactor control circuit is energized through control power circuit breaker and master switch 4S2.
24 V DC ON Indicator Light	4DS2	Indicates 24 volts d-c power supply circuit is energized from power supply.
REACTOR ON Light	4DS3	The light is lit when the master switch (4S2) is in the "ON" position.
MASTER SWITCH	4S2	OFF position deenergizes two fuse-protected branches of the reactor control circuit. TEST position energizes reactor control circuit without enegizing scram magnets, thereby allowing test of control drives with blades disconnected.

TABLE 4.3 (Continued)

Description	Reference Symbol (see Figure 4.14)	Function
		ON position energizes reactor control circuit and scram magnets; sounds warning horn for first ten seconds during which control drives are de-energized to prevent withdrawal of blades.
Fuses	F1, F2, F3, F4, F5	Protect branches of reactor control circuit.
High Voltage Power Supply	15PS1,15PS2 15PS3,15PS4 15PS5,15PS6	Converts 115-volt, 60 Hz input to 0-800 volt d-c output required by ion chambers for operating voltage and gamma compensating voltage.
High Voltage Power Supply	15PS7	Converts 115 volt, 60 Hz input to 700 volt d-c output required by the startup prop. counter for operating voltage.
High Voltage Power Supply Monitor	15PSM1	Monitors operating voltage supplied to compensated ion chambers and initiates a scram if power drops below 100 volts of normal operating voltage.

TABLE 4.3 (Continued)

Description	Reference Symbol (see Figure 4.14)	Function
	15PSM2	Monitors high voltage supplied to start up proportional counts and initiates a scram if power drops below 100 volts of normal operating voltage.
Power Line Transient	4PTM	Monitors city power line transients and gives an indication if a transient occurs.
High Voltage Monitor (Mounted in rear of cabinet)	15PSM1	Provides protection against CIC high voltage failure.
Startup Group		
PREAMPLIFIER (Mounted on Reactor Bridge)	10AR1	Amplifies signal received from proportional counter and transmits it to the log CRM linear pulse amplifier.
LINEAR PULSE AMPLIFER	10AR1A	Amplifies pulses from pre-amplifier and transmits signals to the log count rate amplifier 10AR2 and the scaler 10AR3.
SCALER	10AR3	Indicates directly the number of events occuring at the proportional counter during a preset time interval, or the time interval required to register a present number of events.

TABLE 4.3 (Continued)

Description	Reference Symbol (see Figure 4.14)	Function
LOGARITHMIC COUNT RATE AMPLIFIER	10AR2	Connects signals from the pulse amp to a DC level proportional to the log of the pulse count rate. The LCR amp output drives the LCR meter (10M1).
LOG CR (Log Count Rate Recorder)	10RR1	Records output of log CR meter 10AR2.
LCR METER	10M1	Indicates LCR level.
Log N and Period Group		
LOG N AMPLIFIER	11AR1	Amplifies signal received from the compensated ion chamber and transmits it to the log N recorder and meter.
PERIOD AMPLIFIER		Differentiates Log N amplifier output and generates signals for period meter and recorder blade control circuits and annunciators.
Log N Amplifier Calibration Switch and Calibration potentiometers		Used for internal calibration of the Log N d-c signal amplifier.
LOG N Recorder	11RR1	Records output of log N amplifier.
LOG N Meter	11M1	Indicates Log N level.
PERIOD Recorder	11RR2	Records output of the period amplifier.
PERIOD Meter	11M2	Indicates period level.
Flux Level Safety Group		
LINEAR PWR Recorder	12RR1	Records output of the stable picoammeters 12AR1 or 12AR2.

TABLE 4.3 (Continued)

Description	Reference Symbol (see Figure 4.14)	Function
CHANNEL SELECT Selector Switch	12S4	No. 1 position directs output of 12AR1 stable picoammeter to LINEAR PWR recorder 12RR1. No. 2 position directs output of 12AR2 stable picoammeter to LINEAR PWR recorder 12RR1.
POWER LEVEL Selector Switch	7S5	100 kW low position relay is energized by a circuit through the stable picoammeter remote range switches, and coolant gate switch (when coolant gate is open). Relay will be opened and cause a scram if the operator attempts to raise the power above 100 kW.
STABLE PICOAMMETER	12AR1	Receives the signal from a compensated ion chamber and generates DC level and control signals for the trip and annunciator circuits, the servo amplifier, and power level meter (12M1).
STABLE PICOAMMETERS	12AR2	Performs same function as 12AR1 except for signal to servo circuit.

TABLE 4.3 (Continued)

Description	Reference Symbol (see Figure 4.14)	Function
ACTUATOR AMPLIFIERS Trip Amplifier	12Z5	Receives signals from stable picoammeters 12AR1 and 12AR2 and the log N period amplifier, and cuts off power to the safety magnets on signals of excessive neutron flux or short period.
SERVO AMPLIFIER	9AR1	Receives signals from stable picoammeter (12AR1)and compares them to a reference voltage adjusted through the POWER SCHEDULE SET switch and amplifies the resulting d-c error signal and transmits it to the servo regulating element drive motor.
Alarm and Indicator System Annunciator and Annunciator Lights HIGH COOLANT TEMP. LOW COOLANT FLOW BRIDGE UNLOCKED COOLANT GATE OPEN LOW POOL LEVEL REG. BLADE LIMIT CONT. BLADE DISENG'D HIGH COND. HIGH NEUTRON FLUX	5Z1	Monitors abnormal conditions at sixteen critical points in the reactor system. An abnormal condition is indicated by the relevant light becoming lighted. Pressing the "Annunciator Acknowledge button" on the control console turns off the buzzer.

TABLE 4.3 (Continued)

Description	Reference Symbol (see Figure 4.14)	Function
SHORT PERIOD SEISMIC TRIP SAFETY CHAIN SCRAM ACCESS DOOR OPEN REACTOR CORE LOW FLOW HIGH VOLTAGE FAILURE HIGH RADIATION		When the difficulty is corrected the annunciator and scram reset switches are pressed on the control console, lighting the scram reset light and returning the annunciator lights to normal dim condition.
Miscellaneous Group		
Cooling Fans		Cools electronic components in the left, center, and right sections of the amplifier cabinet.
Surveillance Test Panel	12TP1	When connected, used in determining various timed parameters inherent to the reactor such as -Blade drop times -Blade flight times -Ventilation valve closing times -Electronic and safety chain scram initation times.
Scram Magnet Monitor Panel	12M	Monitors individual magnet current to the four control blade scram magnets.

TABLE 4.3 (Continued)

Description	Reference Symbol (see Figure 4.14)	Function
Delta Temperature Recorder	16R1	Records the differential temperature across the reactor core utilizing thermocouples. (Used only in forced flow mode.)
Nitrogen-16 Recorder		Records N-16 production from a detector located above the primary core exit coolant pipe prior to entrance to the N-16 decay tank, (operable only in forced flow mode).

cabinet contain the annunciator lights and the core t and N-16 recorders. Below the upper portion of the cabinet are the period, linear power, log count rate, and log N recorders. The clock, fuses, "reactor on" indicator, channel selector, power level selector, and master switches occupy the center portion of the upper panel.

The right bay contains the logic and trip amplifier, two picoammeters, and a power line transient monitor. The center bay contains a log N amplifier, a servo amplifier, a surveillance test panel and the Scram Magnet Monitor. The scaler, linear pulse amplifier, log count rate amplifier, HV power supplies and HV power supply monitors are mounted on the left bay of the cabinet. Cooling fans are mounted at the lower end of each bay to provide circulation of cooling air to the components.

4.4.4 Power Distribution System

Power for operation of the reactor equipment is supplied through the control - power circuit breaker and the master switch from a 115-volt, 60 hz unregulated supply. The master switch is key-locked in the "off" position. The switch also provides an "on" for normal operation, and a "test" position in which the control drives may be exercised without energizing the scram magnets and withdrawing the blades. Whenever the master switch is turned ON from TEST or OFF, an interlock causes the alarm bell to sound for 10 seconds to warn personnel of impending startup. A time delay mechanism prevents the control drive motors from withdrawing blades during the 10-second delay period.

4.4.5 Unregulated Control Power Supply

Unregulated 115-volt, 60 hz control power is supplied to the operating components of the reactor control system. This unregulated power is supplied to the cubicle blowers, to the input of the 24-volt d-c power supply, and to the master control components.

A master switch located in the center upper portion of the amplifier cabinet controls the unregulated power to the control blade drives, power level interlocks, startup counter drive, regulating blade drive, position indicators, annunciation, power level control, and trip actuation circuits. The 115-volt unregulated power is supplied to the input of the regulating transformer which provides the reactor instrumentation channels with constant 115-volt, 60 hz voltage.

4.4.6 Regulated Instrumentation Power Supply

The amplifiers, meters, and associated precision components of the reactor instrumentation channels, and the high voltage power supplies receive constant 115 volt, 60 hz voltage from the regulating transformer.

4.4.7 High Voltage Power Supplies

Adjustable high voltage d-c power to the ion chambers and proportional counter is furnished by a series of high voltage power supplies. High voltage monitors which are tied to the high voltage system, provide protection against loss of power to the reactor instrumentation detectors.

4.4.8 Startup Channel (Figure 4.15)

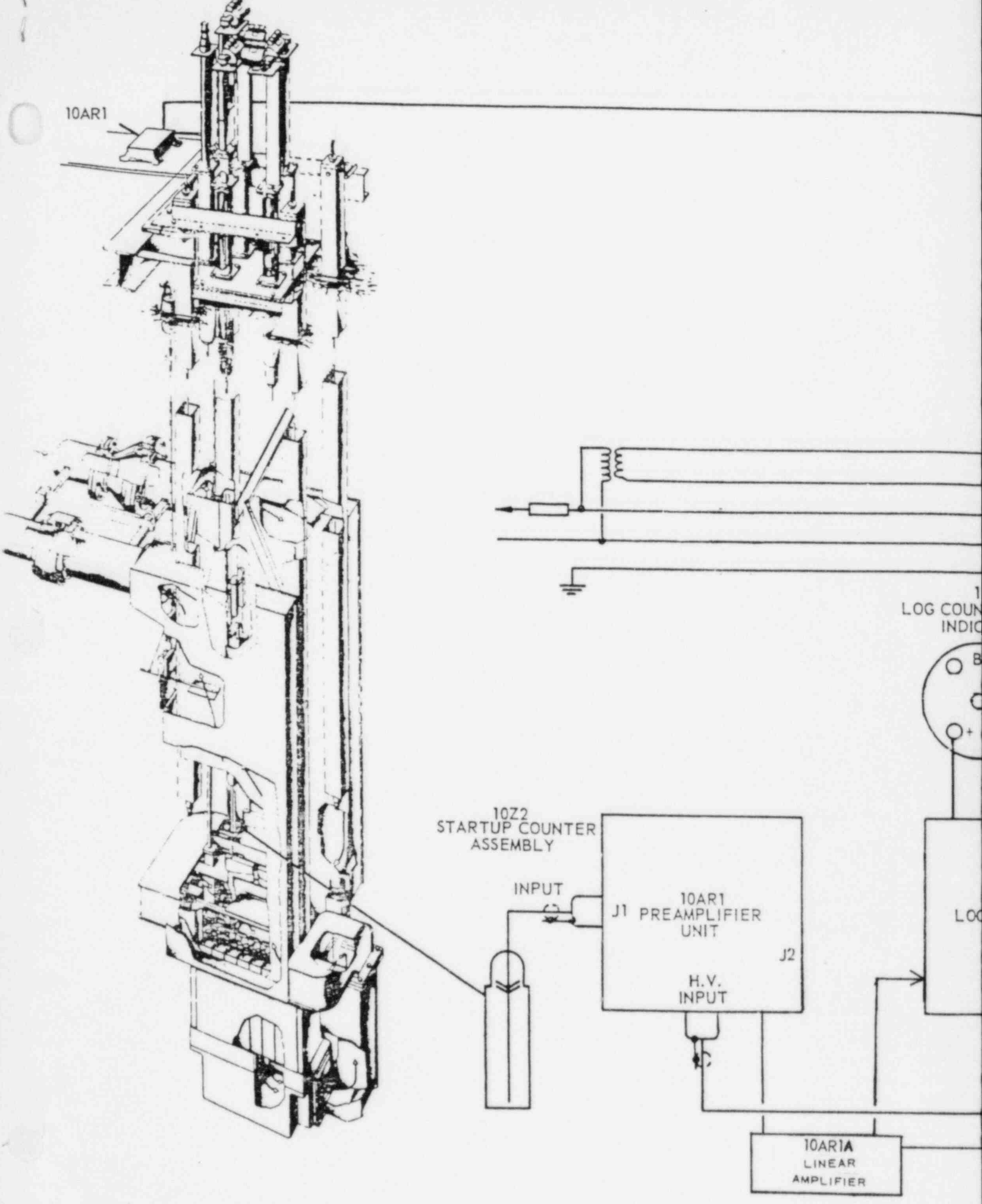
The units composing the startup channel include the proportional counter, preamplifier, linear pulse amplifier, log count rate amplifier, log count rate recorder, and the LED scaler. The startup channel can be used to monitor flux levels from a minimu effective low range of 0.12 nv to a high of 5×10^4 nv.

The pulse output from the proportional counter, which serves as a sensing element, is applied to the preamplifier, which adjusts the impedance level for optimum transmission of the pulse to the linear pulse amplifier. The output of the pulse amplifier drives the LCR amplifier and the scaler. The scaler gives a direct reading of the number of counts occuring during a preset time interval, or conversely, the time required to register a preset number of counts. The scaler is intended for the lowest flux levels.

The log count rate meter provides a useful indication at higher levels, since its scale is proportional to the logarithm of the number of counts per second. This output is recorded.

4.4.9 Control Blade Drive System

The four safety blades are normally controlled by two



10AR1
10Z2
STARTUP COUNTER
ASSEMBLY
INPUT
J1
10AR1
PREAMPLIFIER
UNIT
J2
H.V.
INPUT
10AR1A
LINEAR
AMPLIFIER
LOG COUN
INDIC

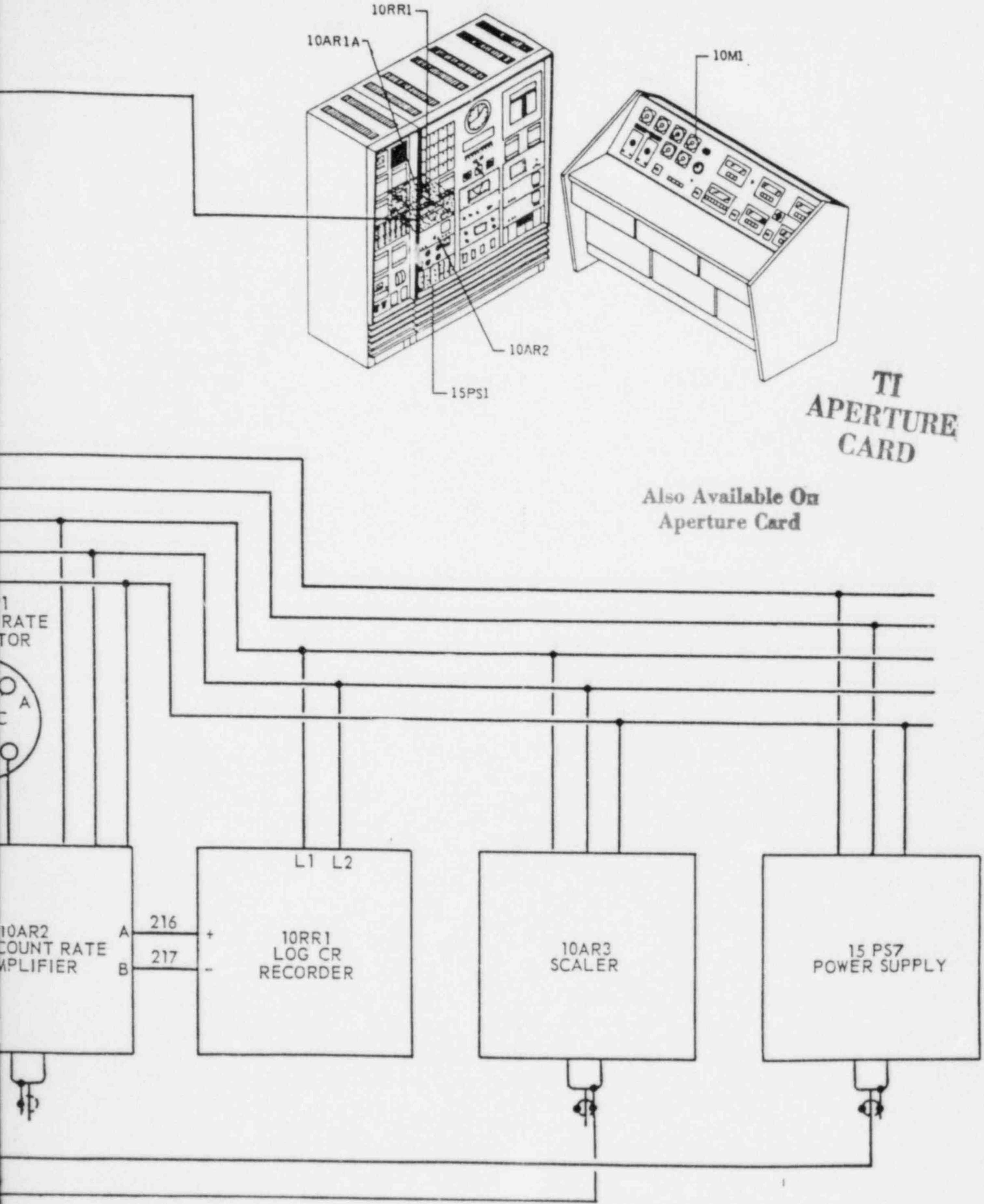


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Figure 4.15. Startup Channel

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switches: one selects the blade to be moved; the other, a switch with a spring return, has positions RAISE, OFF, and LOWER and actuates the selected safety blade. Only one blade may be raised at a time. Digital readouts indicate the position of each safety blade.

Indicator lights on the reactor control console show when a safety blade has reached either limit of its travel. In the case of a scram, the safety blade normal control is overridden by an automatic or manual scram. Control blade drive motors are interlocked against withdrawal of blades during a 10-second delay period subsequent to reactor startup, when either picoammeter is downscale, during a short period or high flux, and when the log count rate meter is below the minimum number of counts per minute.

A position indicator is provided for each control blade drive, to show drive position relative to the fully inserted position. Digital indication to the accuracy of 0.02 inch is furnished by number wheels on a mechanical counter which is chain driven from the ball-bearing screw of the drive. The indication is transmitted electrically through a segmented commutator in the counter to a "withdrawal in inches" indicator on the control console.

A "Manual Rundown" switch is also provided which, when put in the "IN" position, inserts all control blade drives simultaneously. This switch is located on the control console.

4.4.10 Startup Counter Drive System

The startup counter drive circuit consists of a positioning switch, two limit switches, three indicating lights, and a reversible control drive motor. The startup counter position is established at the control console by pushing the positon switch towards IN or OUT. The travel of the drive is established by the limit switches on the drive, which are in series with the motor drive circuit. As the drive reaches either the in or out position, the respective limit switch is opened, stopping the drive at the desired position. Mechanical stops prevent damage in case of failure of the limit switches at the IN and OUT limits of travel.

4.4.11 Servo Controlled Regulating Element Drive System

The regulating element system provides automatic control of reactor power level by a servo system that responds to changes of neutron flux level in the reactor. The automatic system is composed of the following:

(1) A gamma-compensated ionization chamber and picoammeter.

(2) A servo amplifier which responds to the picoammeter signal and controls the speed and direction of the regulating element motion through a servo motor.

(3) A power schedule device which sets the reference power level.

The regulating element may also be controlled manually at the operator's discretion by a control switch on the reactor control console. The automatic control circuit remains "locked-in" until the operator returns to manual control by means of the selector switch or will automatically switch to manual control on any reactor scram or excessive servo deviation error. Indicators on the reactor control console indicate the following:

(1) Regulating element position in the core.

(2) Scheduled power.

(3) Regulating element at either end of travel.

(4) Servo error.

The system is powered by a conventional, two-phase servo motor driven by an a-c output from the servo amplifier in a direction dependent upon output phase. The motor is provided with an interlock against initiation of automatic control if the reactor period is less than 30 seconds. The method of position indication is identical to that of the control drives.

4.4.12 Automatic Power Level Channel (Figure 4.16)

The purpose of the automatic power level channel is to hold the reactor power at a preselected flux level. The units forming the channel include the power schedule set switch, the power schedule indicator (% FULL POWER), and a servo-amplifier. The compensated ion chamber, suspended within the suspension frame corner post, provides input to a closed-loop servo-system which monitors the flux level. When the mode transfer circuit is set for automatic operation, the servo-system operates the servo control drive.

The desired power level is selected by means of the power schedule set switch and is indicated on the power level indicator. This establishes a reference signal for the servo-amplifier, which is compared with the actual power level signal from the compensated ion chamber. The difference between the two signals generates an error signal. The error signal drives the regulating element drive motor in the proper direction to adjust the reactor power level to the desired values. The servo-amplifier in the

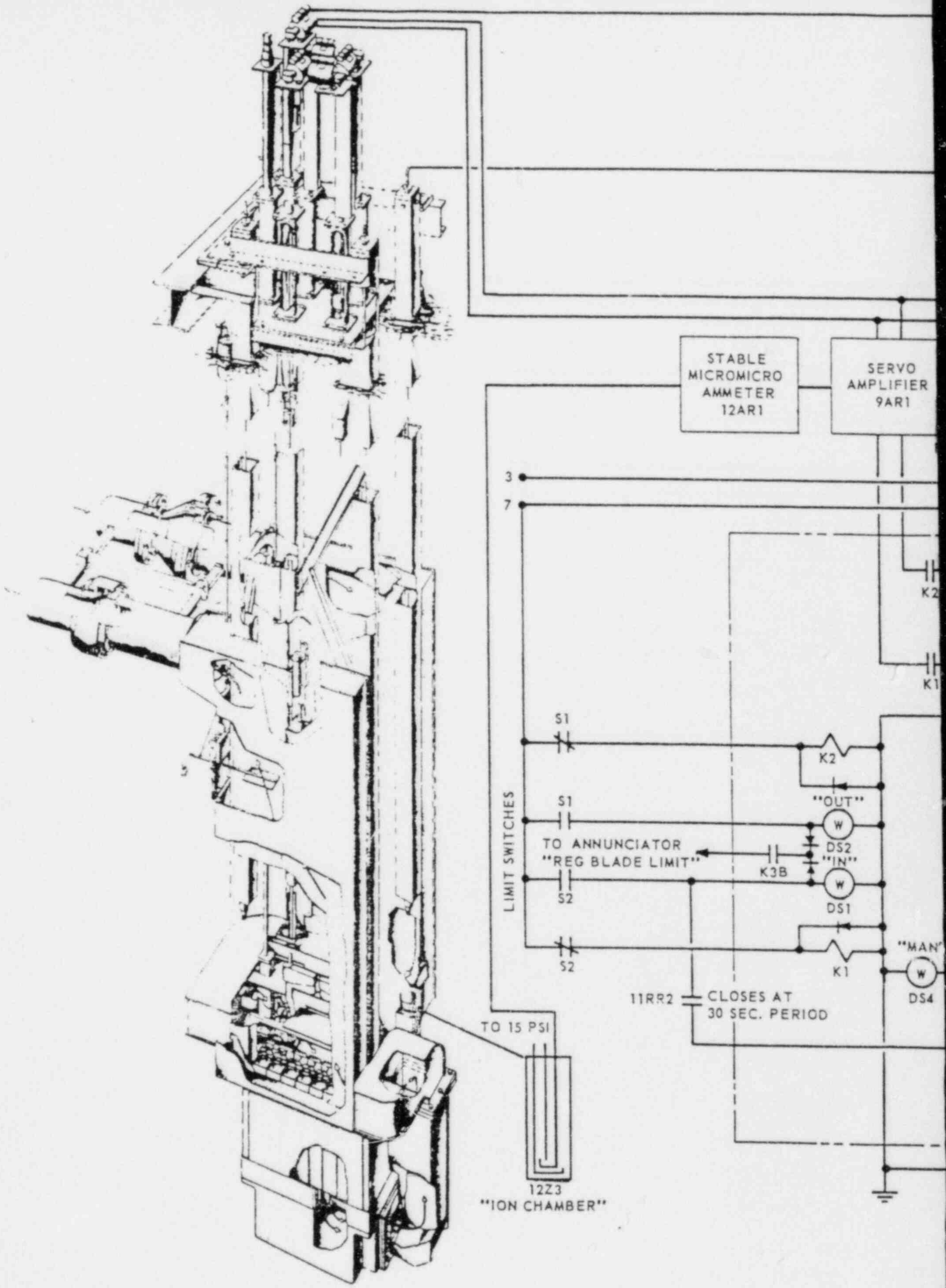
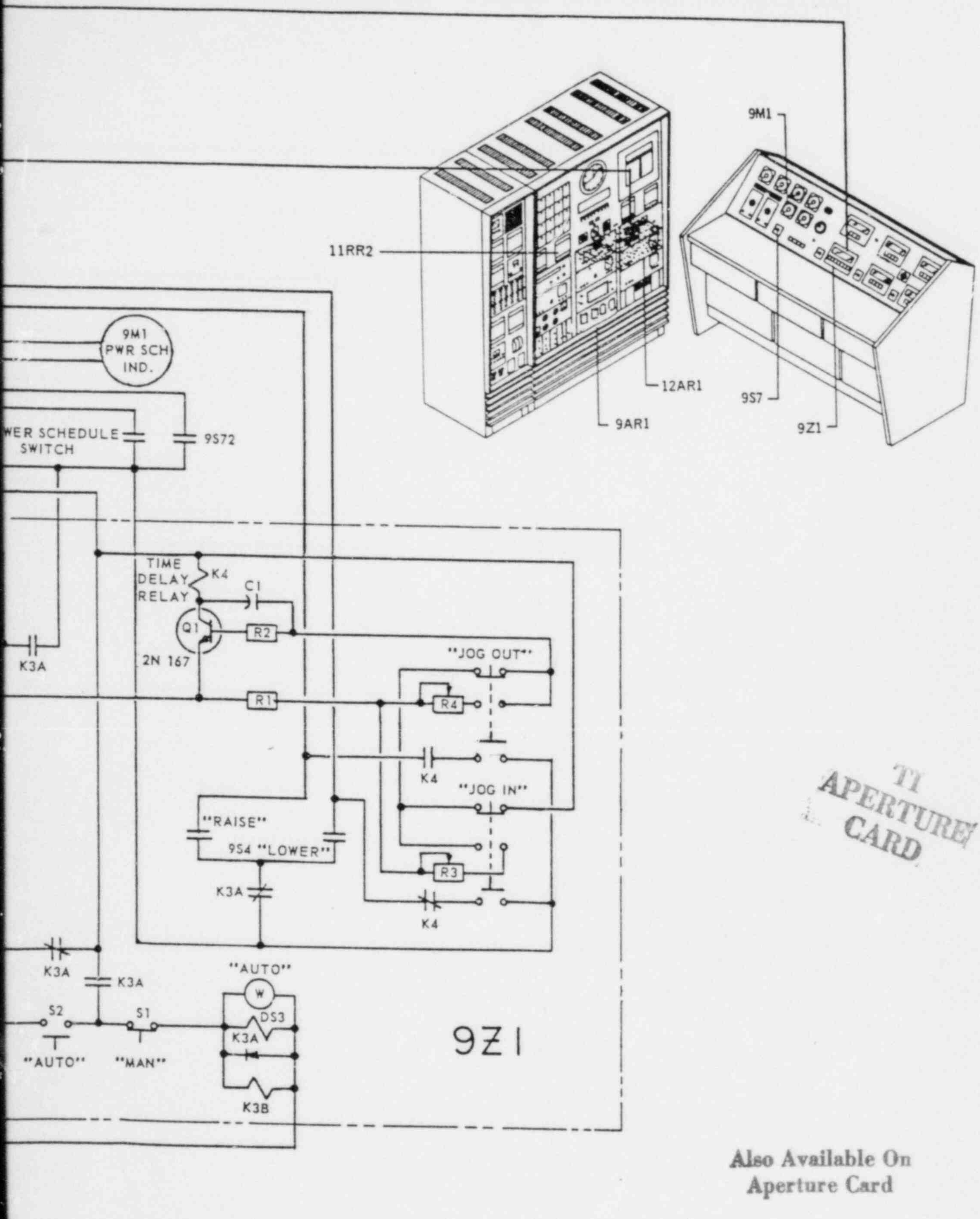


Figure 4.16. Automatic Power Level Channel



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channel provides the following:

(1) Reference voltage which serves as the system power level demand.

(2) Summing junction which compares the demand signal to the actual power level and indicates any voltage error that occurs between the two.

(3) Servo deviation meter.

(4) Power gain necessary to operate the 25-watt regulating element drive motor.

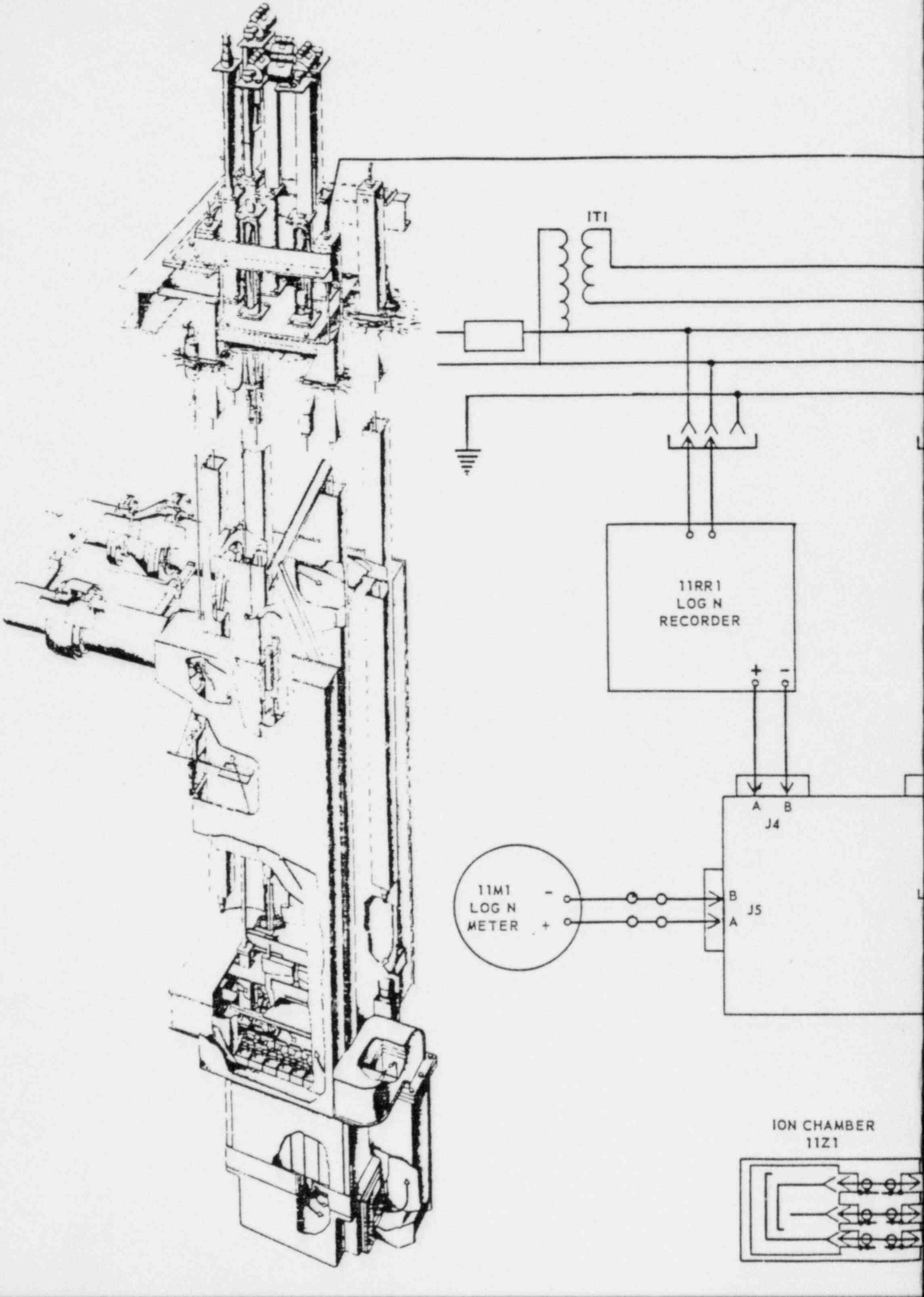
(5) Signal to display power level demand at a remote readout.

The automatic power level channel may be activated at any time provided the reactor period is greater than 30 seconds and the power schedule switch reading is within 20 percent of the CIC signal reading. Once placed in automatic control, the system remains in automatic as long as error voltage is within the servo deviation meter set points or until the mode transfer (MAN) button is pressed, or the reactor scram relays are deenergized.

4.4.13 The Intermediate Channel (Figure 4.17)

The intermediate channel is made up of the log N and period channels and monitors the power level of the reactor over the range of approximately 0.1 watt to 10^6 watts (flux level from 3×10^2 nv to 3×10^9 nv at the chamber location). A compensated ion chamber, suspended within a suspension frame corner post, provides input to the log N amplifier, which feeds the log N meter at the console and the log N recorder on the amplifier cabinet.

A second amplifier differentiates the log N signal to give


IT1
11RR1
LOG N
RECORDER
+
-
A B
J4
11M1
LOG N
METER
B
J5
A
ION CHAMBER
11Z1

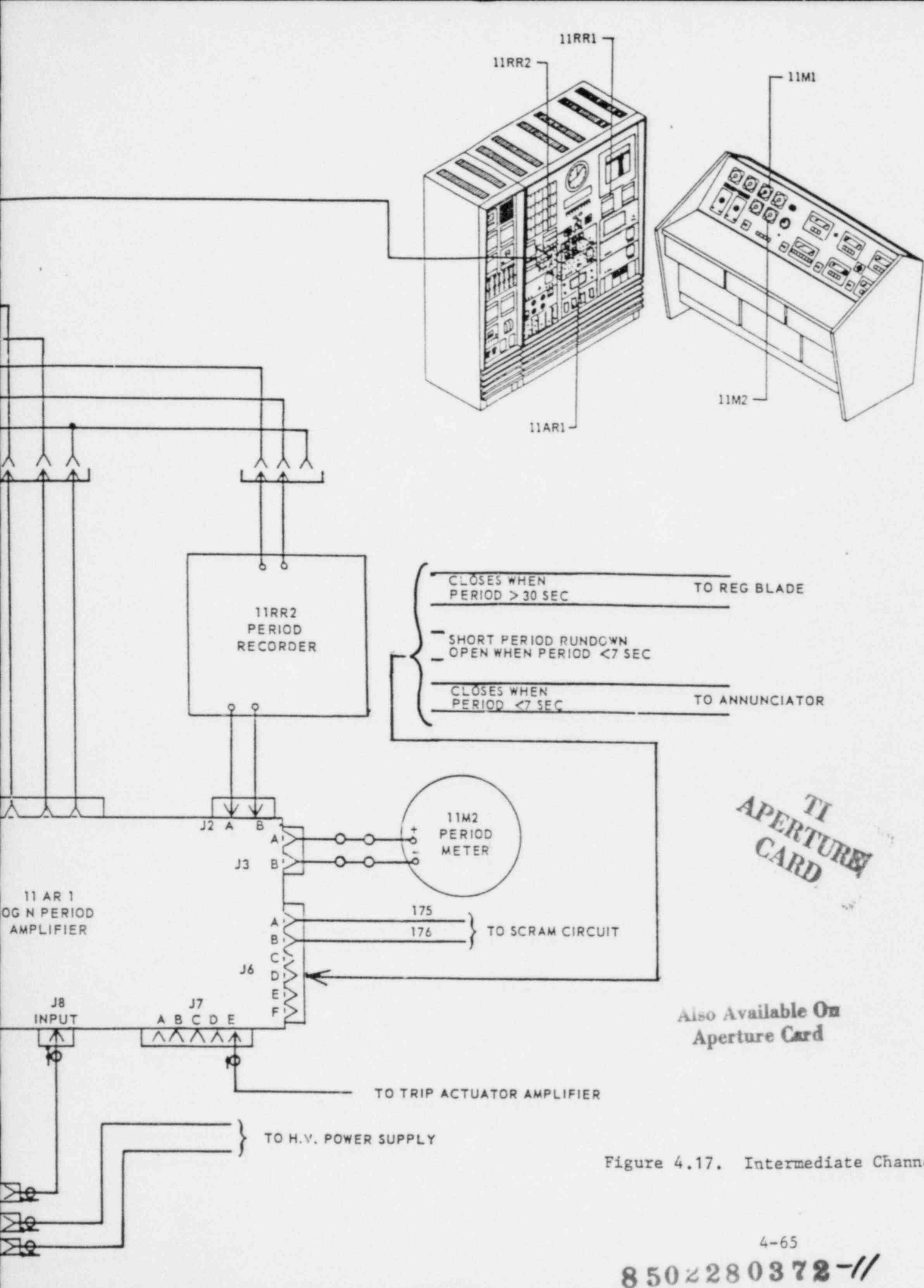


Figure 4.17. Intermediate Channel

the reactor period. Two recorders permanently record log N and reactor period. In addition to monitoring flux levels, the channel supplies signals to: a trip amplifier which can send a scram signal to the actuator amplifiers, the safety chain and the annunciator circuit.

4.4.14 Flux Level Safety Channels (Figure 4.18)

The units that form the safety channels are two compensated ion chambers mounted in corner posts of the suspension frame, two picoammeters (including the two power level indicators and range selector switches mounted on the control console), a trip amplifier, a safety selector switch, and a linear power recorder. The channels monitor power level over the flux range of 3 nv to 7 x 10^9 nv. They therefore overlap the startup channel range and cover the log N and period channel range.

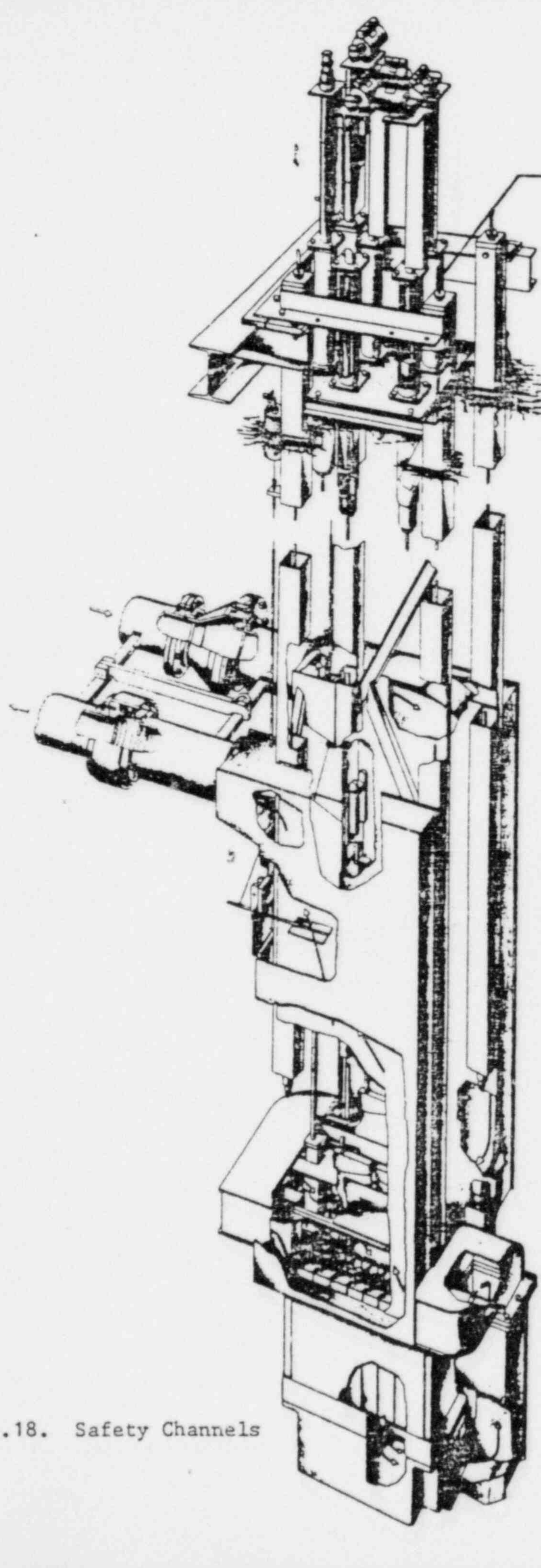
The picoammeters are each equipped with the following trip outputs:

(1) An upscale trip providing a 0 Vdc output when tripped and a 12 Vdc output when untripped to the scram logic unit.

(2) An upscale trip providing an open relay contact output when tripped and a closed contact when untripped. The relay coil associated with this contact is deenergized in the tripped condition. This trip contact is used in the scram safety chain. An additional relay contact actuated by this trip is used to initiate the annunciator.

(3) The third trip is identical in type and operation as (2) above, but the contact is used in the rod withdrawal prohibit circuit.

(4) The fourth trip in the picoammeters is a downscale trip



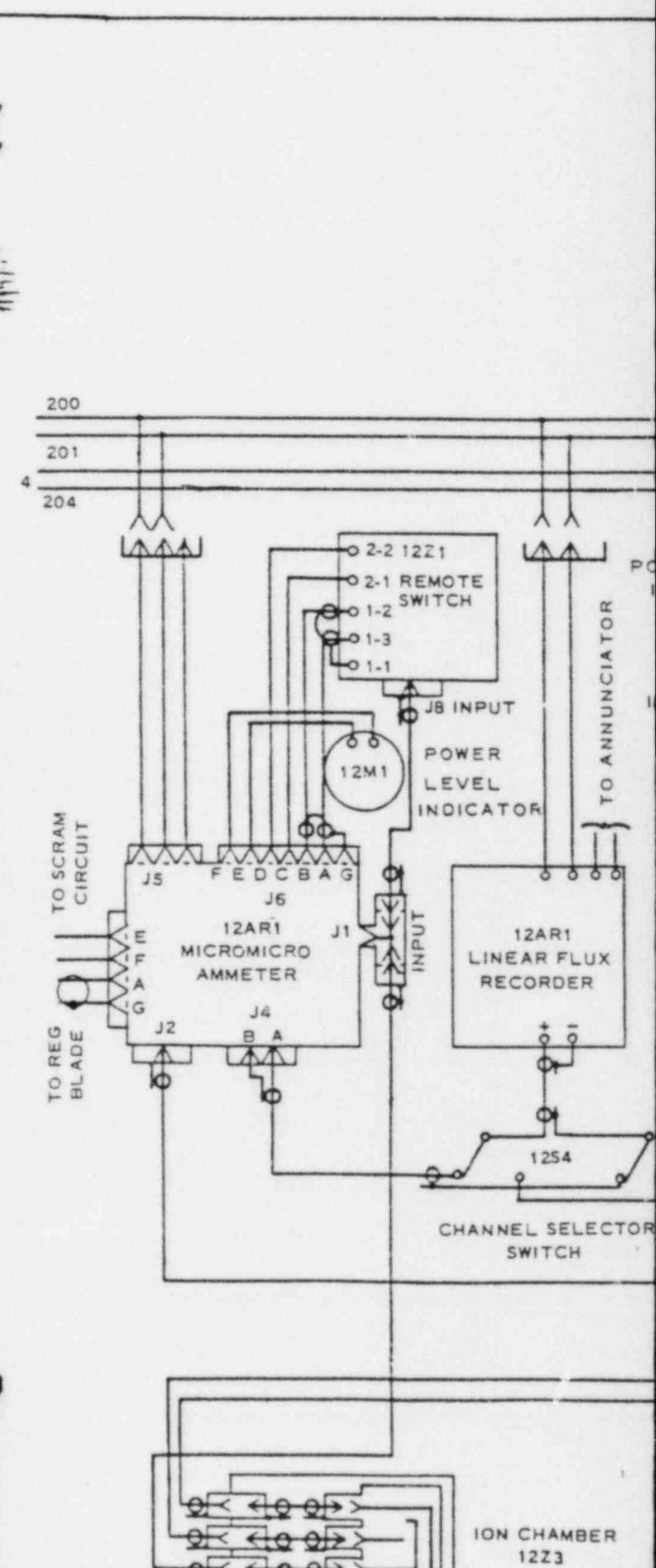
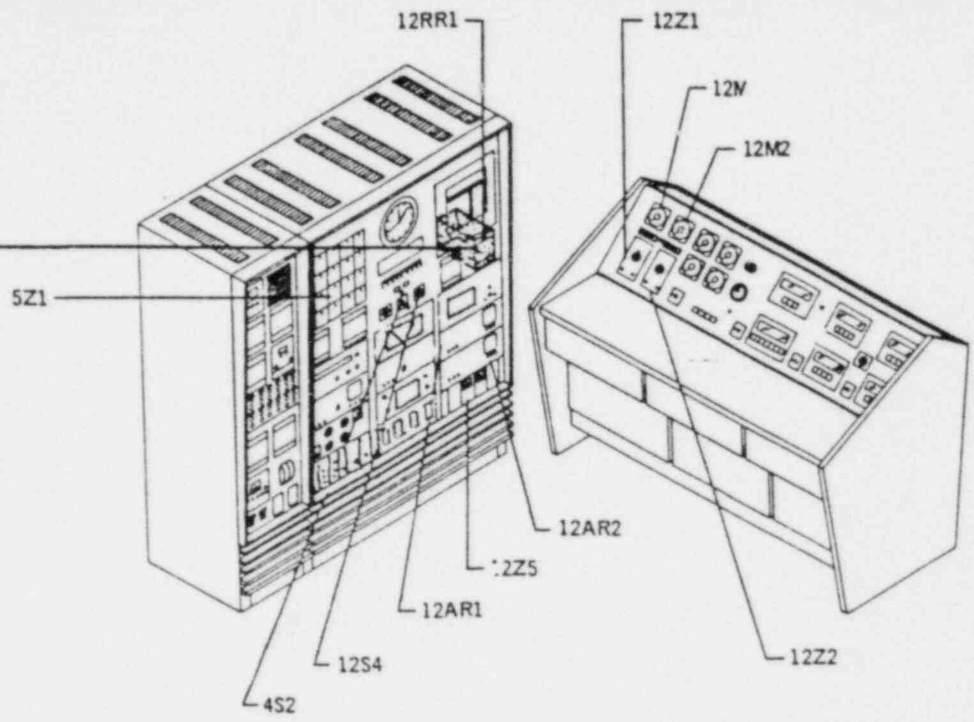
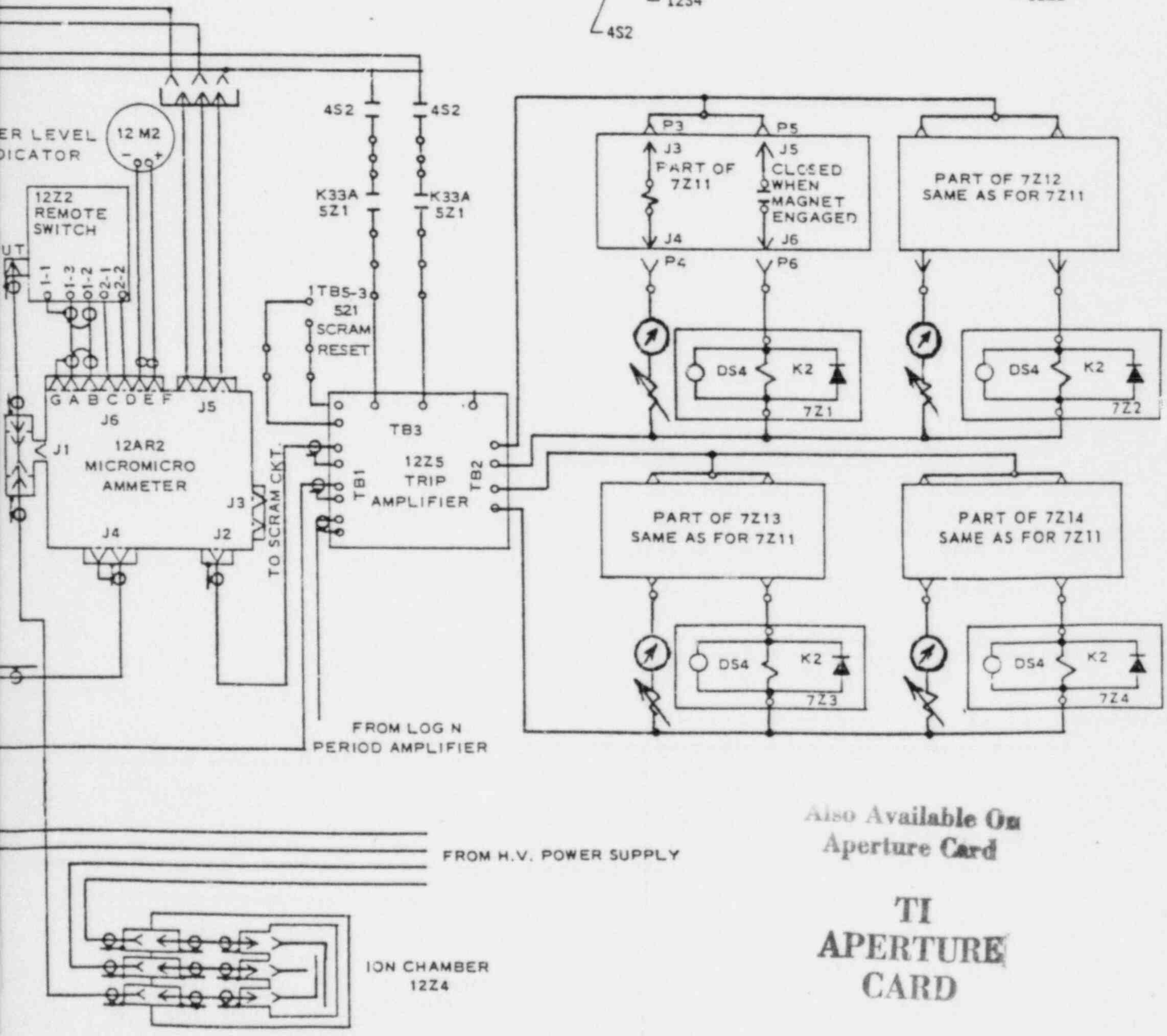


Figure 4.18. Safety Channels





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which is actuated when the d-c amplifier output is below the predetermined set point. It provides an open relay contact when tripped and a closed contact when untripped. The relay coil associated with this contact is deenergized in the tripped condition. This trip is used in the rod withdrawal prohibit circuit to prevent rod withdrawal if the picoammeter reads downscale when the reactor is operating in the rated power region. Provisions were made for bypassing this interlock during cold-clean core startup. The bypass feature is accomplished by effectively shorting contacts - one for each picoammeter in the rod withdrawal prohibit chain, using the picoammeter range switch auxiliary contacts. These switch contacts would be closed when the remote range switch is set on the most sensitive range during startup. When the range switch has been upscaled from the minimum operating range, the auxiliary contacts are open and the picoammeter downscale trips become effective in the rod withdrawal inhibit circuit.

Failure of the power source or internal power supplies will cause all the trips to assume their tripped state. Relay contacts associated with trips (2), (3), and (4), above, will open and the voltage output from (1) will drop to zero. Reactor scram and rod withdrawal prohibit will result.

A component failure in the amplifier section causing a false upscale signal output will cause trips (1), (2), and (3) to be actuated. Relay contacts associated with trips (3) and (4) will open and the voltage output of trip (1) will drop to zero. Reactor scram and rod withdrawal inhibit will result.

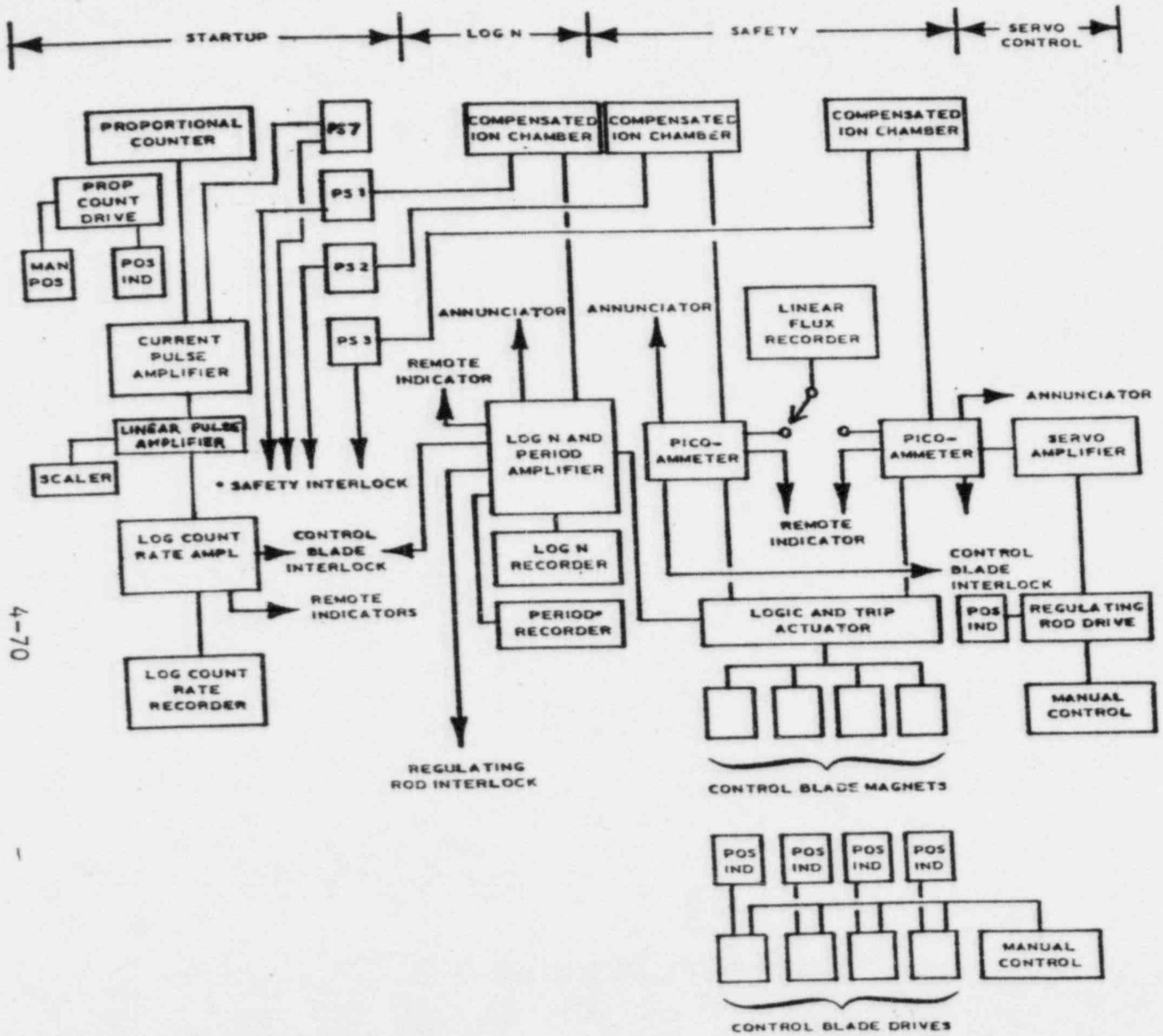
A component failure in the amplifier section causing a false downscale output will cause trip (4) to be actuated. Rod withdrawal will be prevented when the range switch is set on or above the minimum operating range. Since both scram trips from each picoammeter are operated in a 1-out-of-2 mode, reactor scram protection is provided by the operating amplifier.

A component failure in the amplifier section which would cause the output to neither increase or decrease, yet not respond to an increase or decrease of signal input, is extremely remote. In this case, however, reactor protection is provided by the second picoammeter.

The safety channel signals are combined with signals from the log N and period amplifier in a trip amplifier. The trip amplifier scrams the reactor on high flux or short period. The scram setting is 125 percent power on any selected range from either picoammeter channel and is 3 seconds (minimum setting) period from the Log N. Separate high voltage power supplies are provided for each safety channel detector.

4.4.15 Scram Circuits (Table 4.4 and Figures 4.19 and 4.20)

The scram circuits initiate either a relay scram or an electronic, or "fast," scram. The fast, electronic trip from both picoammeters and the Log N period amplifiers is backed up with a slower relay trip. Both types of trips are automatic. The electronics trip deenergizes the scram magnets via the logic unit and trip actuator amplifier. The relay trips listed in Paragraph 4.4.15.1 are series-connected in the safety chain which leads to the two master scram relays. Contacts from the scram relays



* SCRAMS ON LOSS OF HIGH VOLTAGE

Figure 4.19. Reactor Controls and Instrumentation Outline

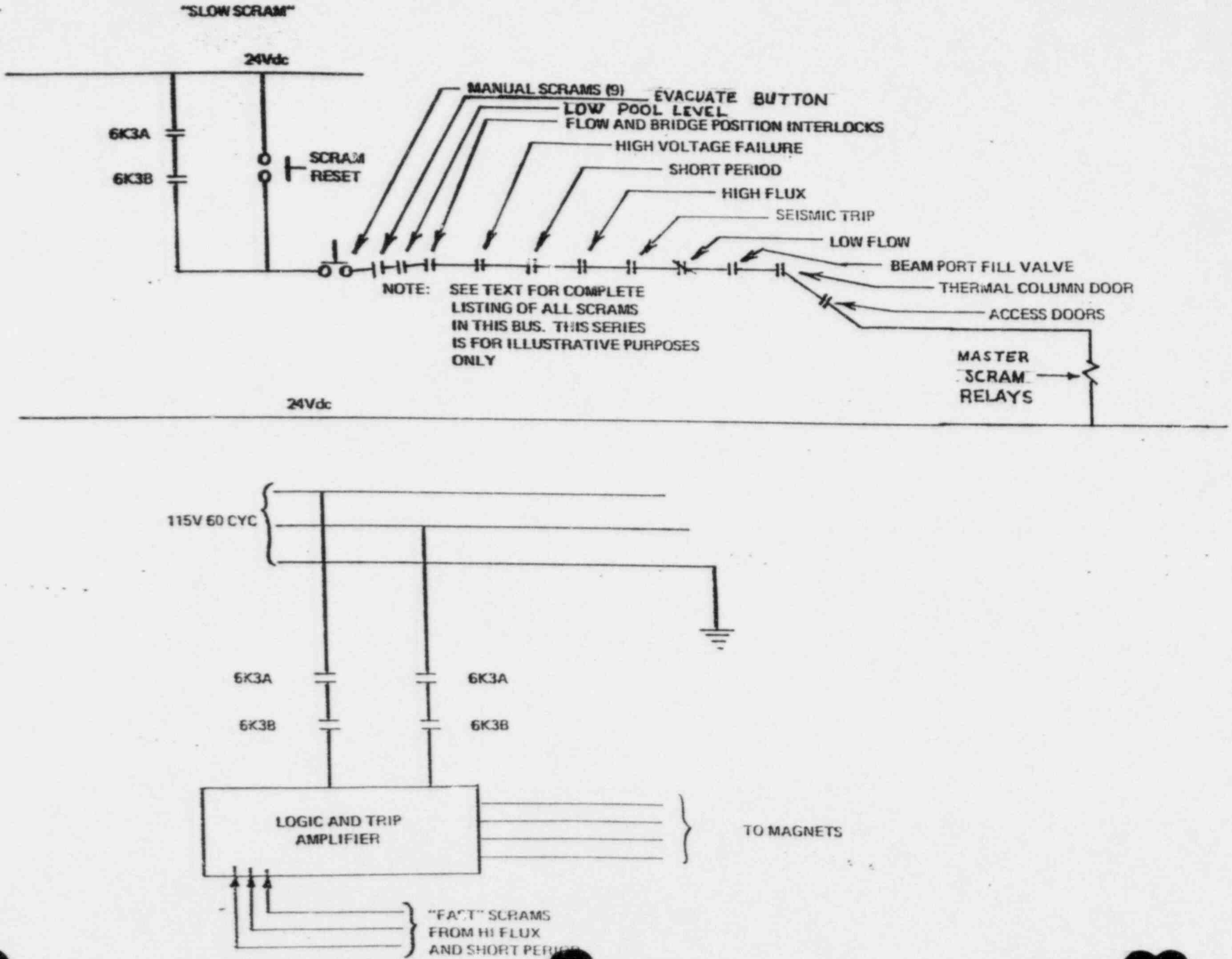


Figure 4.20. Reactor Scram Circuit

appear in the power supply to the trip actuator amplifier.

The trip actuator amplifier is designed to supply and control the current to the scram magnets. The control function is accomplished by solid-state switching circuits. The two series connected bistable amplifiers and a power supply furnish a continuous output of one ampere at a nominal 24 Vdc. The control of the switching action is provided by supplying 12 Vdc signals at two inputs. These signals originate from the logic unit. Loss of the signal at either input trips the amplifier. Interruption of the 115-volt, 60-cycle power supply to the trip actuator amplifier also serves as a scram signal source which will deenergize the magnet coils.

4.4.15.1 Relay Scram Circuit The relay scram circuit controls input power to the trip amplifier. Interruption of this power deenergizes the scram magnets and automatically shuts down the reactor. The manner in which the relay scram circuit is acutated under certain conditions is summarized below.

TABLE 4.4

PRESENT REACTOR SAFETY SYSTEM SETPOINTS

Control	Level	Scram	Alarm
Pool Temperature (natural convection mode)	104°F	X	
Manual action		X	X
Short period (1 detector)(1)	15 sec		X
	7 sec adjustable	X	X
High flux (2 detectors)	1.1 ratio		X
	1.20 ratio	X	X
	Off-scale	X	X

TABLE 4.4 (Continued)

Control	Level	Scram	Alarm
High temperature of primary coolant entering core	104°F	X	X
	100°F		X
High temperature of primary coolant leaving core	108°F	X	X
	104°F		X
Low flow rate of primary coolant(2)	0.9 ratio		X
	0.8 ratio	X	X
High conductivity of primary coolant			X
Low pool water level	4 in. below normal		X
	24.25 ft above core center line	X	X
Seismic disturbance	IV on MM scale	X	X
Bridge misalignment or movement		X	X
Coolant gates open		X	X
Safety blade disengaged from magnet			X
High voltage failure	80 to 90% normal	X	X
Regulating rod at either limit of travel			X
Thermal column door open		X	X
Containment air lock doors open		X	X

NOTE

(1) Electronic scram occurs for periods shorter than 7 seconds.

(2) Alarm and scram signals are provided by two different and independent devices.

(1) Manual - Manual scram is initiated at the operator's discretion by actuation of the manual scram push button which breaks the relay scram circuit and deenergizes the scram relays. A scram is also initiated by pushing the radiation emergency evacuate button on the ULR Radiation Monitoring Cabinet (Paragraph 4.4.19).

(2) Period - The period relay scram occurs when the period alarm relay in the log N period amplifier is deenergized by a reactor period less than the preset value.

(3) Bridge, Gate or Coolant - Relay scram occurs if the bridge is moved out of position, if the coolant gates are opened, if the primary coolant temperature or pressure is abnormal, or if the pool water level is low.

(4) Neutron Flux - Relay scram occurs when the high flux relays in either picoammeter are deenergized by high flux signals in either safety channel. This scram occurs when the flux level signal reaches a level of 120 percent of the range setting being used.

(5) Seismic Disturbance - Relay scram occurs when a seismic disturbance closes the seismic trip detector contact which short circuits the seismic trip relay coil.

(6) High Voltage Failure - Relay scram occurs when the relays in the CIC chamber high voltage monitors are deenergized by a failure in the high voltage power supply circuits.

4.4.15.2 Electronic Scrams - An electronic scram is caused by a flux level exceeding a preselected value from 0 to 120 percent of any picoammeter range in either safety channel, or by a reactor period of less than 7 seconds. The electronic scram is effected by electronically cutting off the current in each scram magnet so that the weight of the control blade will cause it to fall.

4.4.16 Alarm and Indicator System (Table 4.4)

The alarm system is divided into two sections: one for coolant variables and the other for nuclear variables. The section used for cooling system alarm will be operative with forced cooling. When an abnormal condition develops, a buzzer sounds and the appropriate light goes on. The operator may press the acknowledge button to silence t buzzer. When the alarm condition is corrected, the ligh may be reset. The following conditions will actuate the alarm system:

(1) Short period inhibit
(2) High neutron flux inhibit
(3) Safety chain scram
(4) Blade disengaged
(5) Low pool level
(6) Bridge unlocked
(7) Access doors open
(8) Coolant gates open
(9) Seismic trip
(10) Low coolant flow (2 sensors)
(11) High coolant temperature (3 sensors)
(12) High conductivity
(13) High voltage failure
(14) Regulating blade at limit
(15) Reactor core low flow

(16) Demineralizer high temperature and low flow

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

(1) REACTOR ON

(2) 24 VDC ON

(3) Regulating rod at either end of travel

(4) Regulating rod automatic control

(5) Control blade magnet engaged

(6) 115 VAC ON

(7) Control blades in (separate light for each)

(8) Control blades out (separate light for each)

(9) Start up counter OUT

(10) Start up counter IN

Summarized in Tables 4.3 and 4.4 are the rea conditions for which the safety system produces a reactor scram and/or alarm.

4.4.17 Process Control and Instrumentation (Figure 4.21 and Table 4.5)

Control of the cooling system and the monitoring of its operating conditions are accomplished through the process control and instrumentation system. Except for sensing elements, instrumentation for the system is concentrated in the process control cabinet. Motor control unit relays and circuit breakers are on the panel of the motor control center with remote start-stop buttons mounted on the process control cabinet. The process control cabinet is installed beside the amplifier cabinet and is accessible to the operator in the control room.

4.4.17.1 Process Control Cabinet The process control cabinet is a sheet metal cabinet with front panel and rack-type mounting

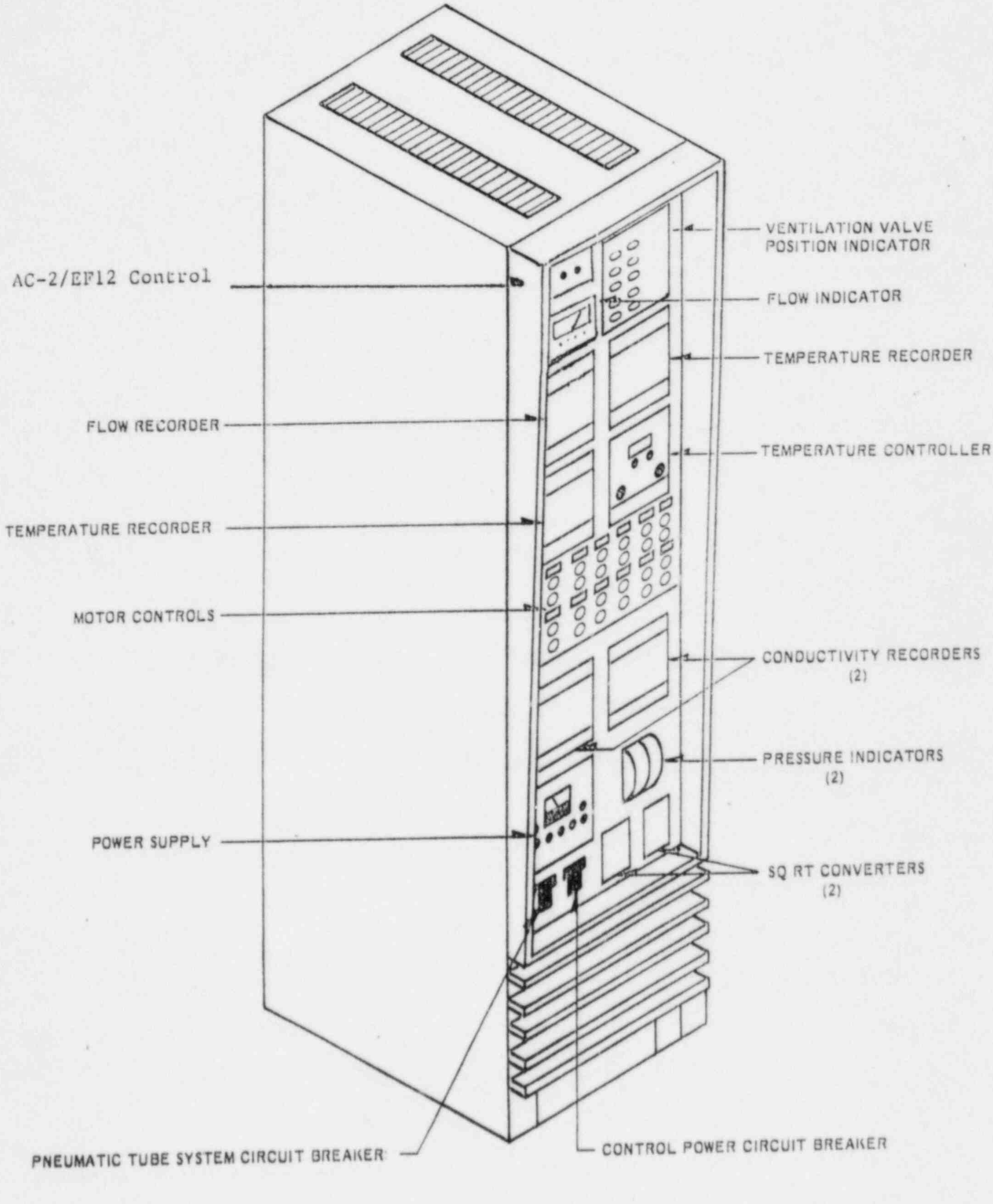


Figure 4.21. Process Control Panel

provided for process, cooling and ventilation system instrumentation. The cabinet serves as an assembly point for location of a flow indicator, five recorders, a temperature controller, fourteen motor push button indicating controls, two pool level indicators, two square root converters, a power supply, a valve position indicator panel and two circuit breakers. One circuit breaker controls the instrument power to the process control cabinet and the other circuit breaker controls power to the pneumatic tube system.

TABLE 4.5

PROCESS CONTROL CABINET COMPONENTS

Description	Reference Symbol (Figure 4.11)	Function
Conductivity Channel		
Conductivity Recorder	927	Records ouput from conductivity cells 928B and C. Actuates annuciator relay if conductivity of coolant exceeds preset limits.
	929	Records output from conductivity cell 928A. Actuates annunciator relay if conductivity of coolant exceeds preset limits.
Temperature Channel		
Temperature Recorder	903	Receives and records signal from the resistance temperature element 901B. Actuates scram if temperature of coolant rises to 104 °F as indicated by signal input from element 901B.

TABLE 4.5 (Continued)

Description	Reference Symbol Figure 4.11	Function
	908	Receives and records signal from the resistance temperature elements 901A and 909. Actuates annuciator relay if temperature of coolant rises to 110 percent of normal as indicated by signal input from element 901A. Actuates scram relay if temperature of coolant rises to 108°F as indicted by signal input from element 909 (natural convection mode).
Temperature Controller	906	Actuates motor-operated flow control valve 905 upon receipt of input from element 901B in excess of set point.
Flow Channel		
FLow Transmitter (Located in Lines) Primary Secondary	912A 912B	Detects changes in differential pressure between the upstream and sides of the calibrated flow orifice, and converts it to a proportional d-c signal which is transmitted to the square root convertor.
SQUARE ROOT CONVERTOR		Takes square root of flow transmitter DC signal and sends signal to the flow recorder.
Flow Recorder (2 Pen) Primary	915A	Actuates annunciator relay if flow is primary loop drops to 90 percent of normal as indicated by square root converter 919A. Actuates scram

TABLE 4.5 (Continued)

Description	Reference Symbol (Figure 4.11)	Function
		relay upon receipt of signal indicating flow has dropped to 80 percent of normal. Records input from flow transmitters 912A and 912B.
Secondary		Actuates annunciator relay if flow in the secondary loop drops to 75 percent of normal as indicated by signal input from square root converter 919B.
Power Supply Unit	917A	Supplies d-c power to flow transmitters 912A and 912B.
Delta Pressure Indicator Primary	923A	Indicates Output from Delta Pressure Transmitter 922A.
Secondary	923B	Indicator Output from Delta Pressure Transmitter 922B.
Flow Indicator	918	Indicates output from flow meter 920. Actuates scram relay upon receipt of signal indicating flow has dropped to 80 percent of normal.
Ventilation Valve Position Indicators		Indicate ventilation valve positions.

4.4.17.2 Motor Control Panel Each of a number of process motors is controlled by a set of remote RUN and STOP indicating push buttons on the process control panel located in the control room. These motors include the primary and secondary pumps, secondary cooling tower fan, all ventilation exhaust fans, the makeup pump, the cleanup pump and beam tube supply valve.

4.4.17.3 Coolant Monitoring Channel To monitor the conditions of flow, temperature and conductivity of the coolant system, the process system is provided with sensing, transmitting, and recording instruments for each channel.

4.4.17.4 Conductivity Channel Coolant conductivity is measured by three cells: one in the holdup tank, one in the pool cleanup system, and one in the makeup system. Outputs of the three cells are recorded on the process control panel. The recorder is calibrated to give conductivity readings in micromho/cm.

The primary coolant is examined for dissolved solids by measuring resistivity of the water in the holdup tank. The conductivity cell forms one leg of an a-c bridge in the conductivity meter. When resistance of the coolant changes, an error signal is established, which if above the prescribed limits, initiates an alarm. A continuous record of the conductivity of the coolant circulating in the primary systems is maintained by the conductivity recorder.

4.4.17.5 Temperature Channel The temperature sensing resistance elements operate on the principle that resistance in a wire varies in relation to change of temperature. A resistance element (RTD) is connected to the recorder input as one arm of a DC bridge circuit. The element resistance valve determines the recorder pen deflection. The recorder scale is calibrated in ^{o}F.

4.18 Seismic Detector

The seismic detector is an oil-damped, approximately 1-second period pendulum with platinum-coated cone and ring contacts near the center of oscillation. Any horizontal earthquake acceleration period combination that causes a differential motion of the pendulum center of oscillation equal to contact spacing will close the switch. The resultant signal is integrated into the scram circuit via the safety chain (see Paragraph 4.4.15.1(5)). The U.S. Coast and Geodetic Survey, originator of the switch, has used the device for over 25 years to activate strong-motion seismographs in time to record potentially damaging earthquake motion. The switch is integrally mounted to the pool concrete on the top of the thermal column and is set to respond to earthquake intensities corresponding to a setting of IV on the Modified Mercalli scale. This intensity earthquake would be qualitatively described as follows:

> "During the day, felt indoors by many, outdoors by few; at night some awakened. Dishes, windows, door disturbed; walls making cracking sound. Sensation like heavy truck striking building. Standing motor cars rocked noticeably."

4.4.19 <u>ULR Radiation Monitoring Cabinet</u> (Figure 4.22)

This cabinet houses the remote readout indicators of the various radiation detectors described in Chapter 10, Appendix. Depending on radiation levels at various monitoring stations, the potential for either a Limited Radiation Emergency Alarm (LREA) or a General Radiation Emergency Alarm (GREA) is indicated here by lights and audible signals. The push buttons which actuate either an LREA or GREA are also on this cabinet. An LREA or GREA scrams the reactor and initiates containment isolation.

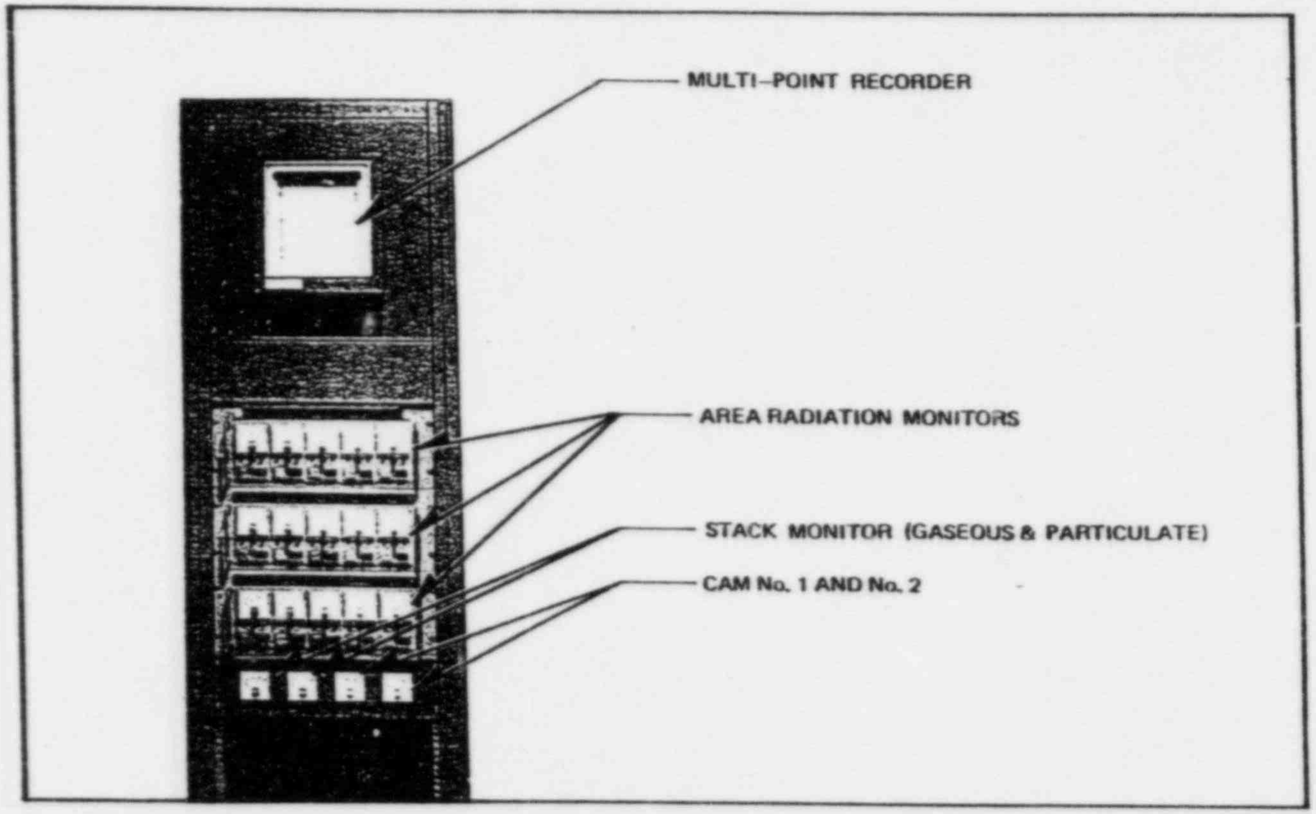


Figure 4.22. Radiation Monitoring Cabinet

4.5 FACILITY DESIGN OPERATING CHARACTERISTICS

This section summarizes the ULR operating characteristics as calculated[1] or as measured for the ULR.

4.5.1 Critical Mass and Loading

The critical mass for a water reflected core was calculated to be 3 kg of ^{235}U or about 22.5 elements, and the critical mass for a core reflected by a ring of graphite elements and then water was calculated to be 2.8 kg of ^{235}U or about 20.5 elements. Loaded with 3.5 of ^{235}U, i.e., a 26 element core, the ULR will have an excess reactivity of about 4.5% $\Delta k/k$.

4.5.2 Neutron and Gamma Flux

Figure 4.23 shows the two-dimensional thermal neutron flux distribution in the reactor core when operated at a level of 1 MW. Figures 4.24 and 4.25 show the three group neutron flux distributions in the X and Y directions. The average thermal neutron flux in the reactor core is 8.6×10^{12} nv, and the average epi-thermal flux is 1.2×10^{13} nv. The gamma heating in the core has a peak value of less than 1 watt/cm^3.

Figure 4.26 shows the gamma dose rate in the thermal column. The fast neutron flux at the core end of an 8-inch beam port is computed to be 2×10^9 nv. The calculatied gamma flux entering the beam port is 5.8×10^{11} meV/cm^2-sec., which corresponds to a dose rate of 9×10^5 R/h. at 1 MW.

4.5.3 Reactivity Requirements

1.7% in reactivity is required to compensate for the buildup of Xe and Sm at the 1 MW operating level. An additional 0.2% compensates for temperature rise during operation. With 4.5%

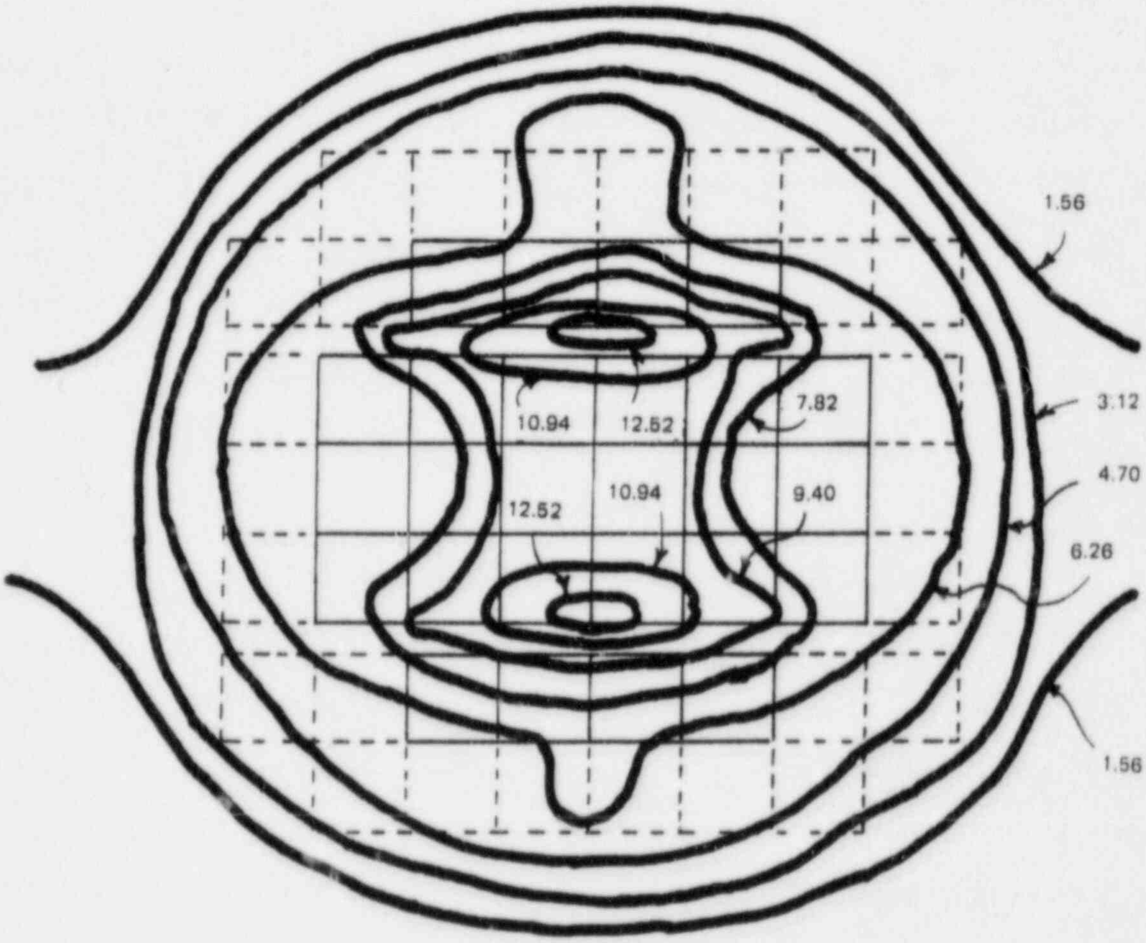


Figure 4.23. Thermal Neutron Flux Distribution At 1 Megawatt

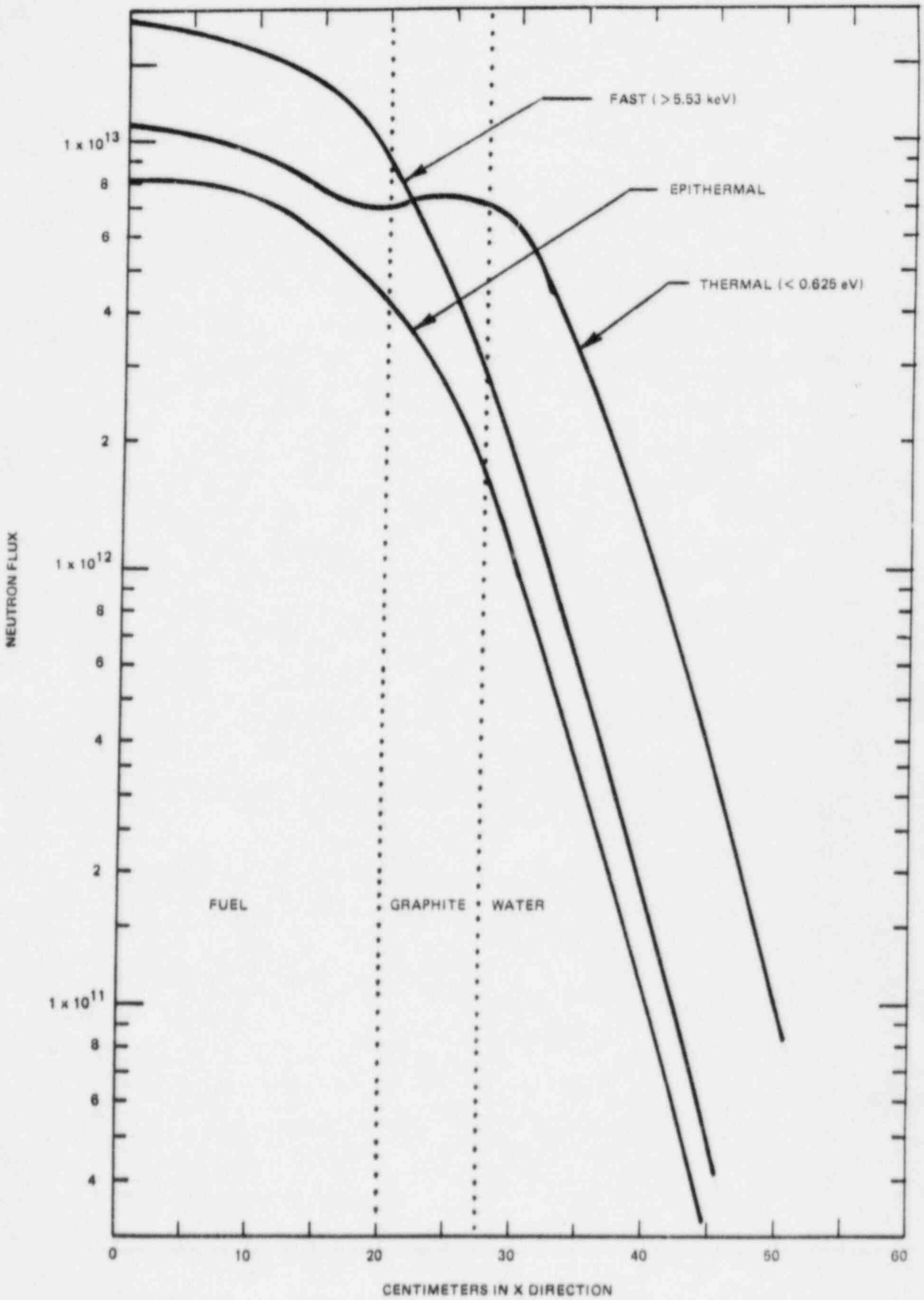


Figure 4.24. Neutron Flux Distribution In X Direction At 1 Megawatt

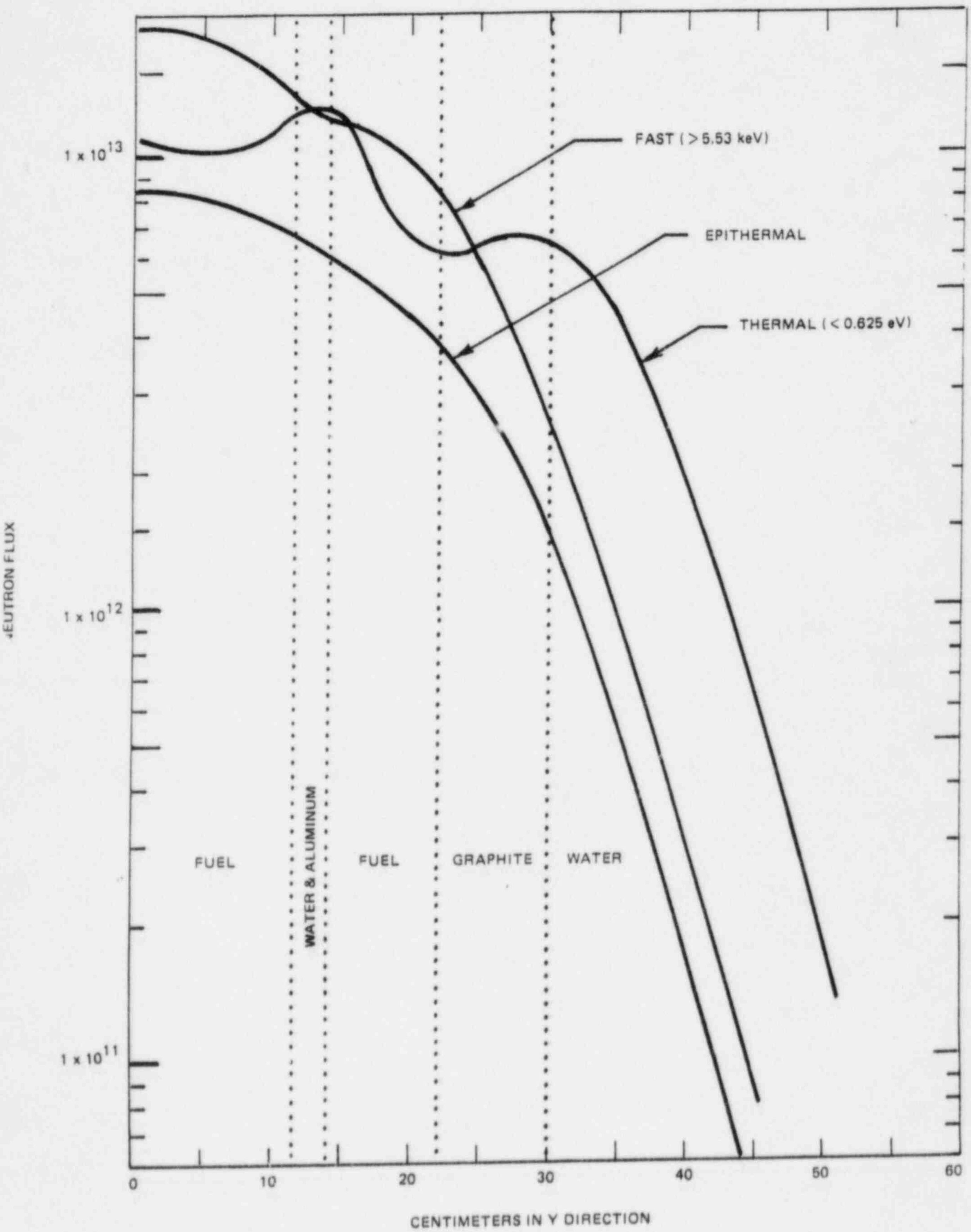


Figure 4.25. Neutron Flux Distribution In Y Direction At 1 Megawatt

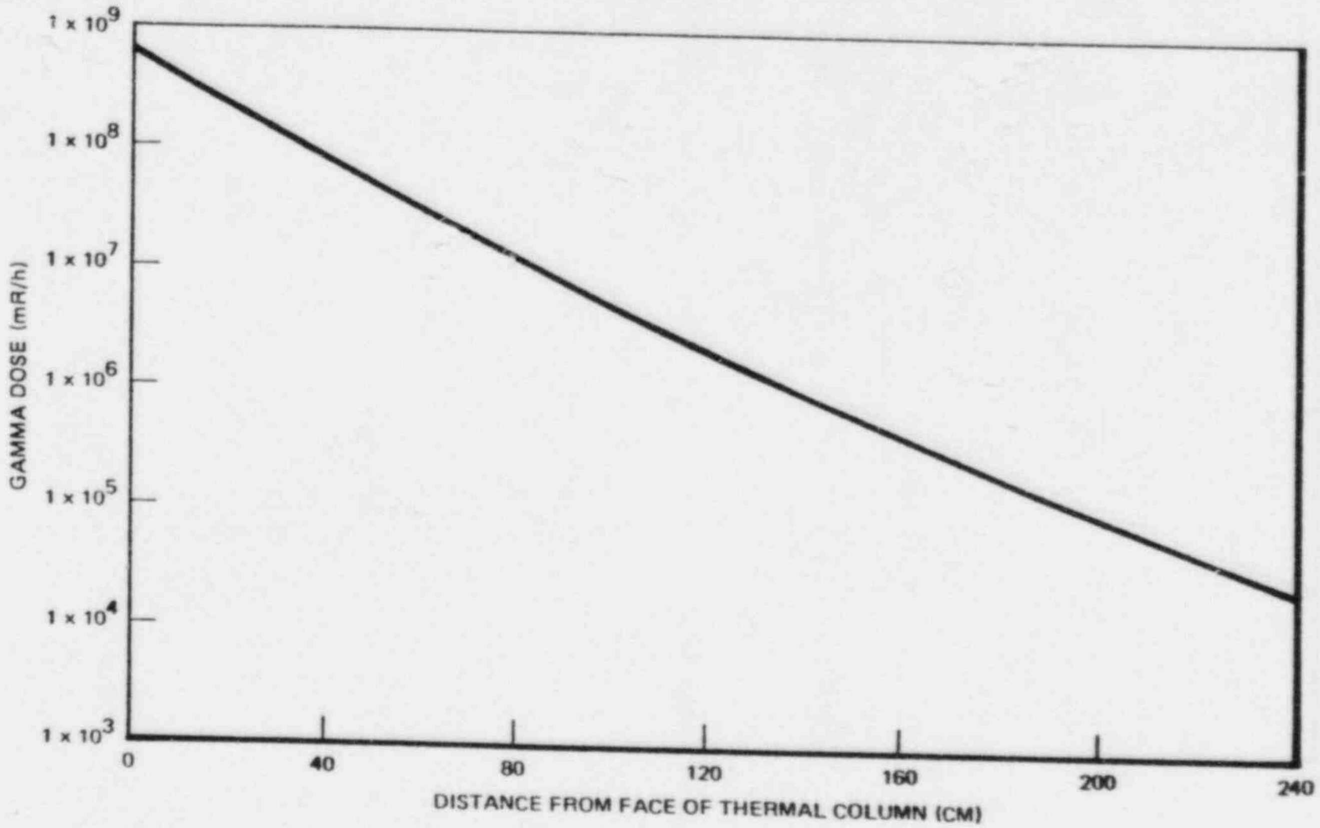


Figure 4.26. Gamma Dose In Thermal Column At 1 Megawatt

total excess reactivity in the core, 2.6% remains to compensate for burnup of fuel, buildup of fission products, experiments, and to provide a margin for control.

4.5.4 Control System

The four control blades (Figure 4.1) divide the core into three sections. The combined control effectiveness is about 11.5% $\Delta k/k$. A schematic of the control blade is shown in Figure 4.27.

A servo-controlled regulating rod worth 0.5 to 0.7% $\Delta k/k$ is used to compensate for small reactivity changes. It is located in the reflector region of the core and may also be used for calibration and measurement of small reactivity changes. A schematic drawing of the regulating rod is shown in Figure 4.27.

4.5.5 Temperature and Void Coefficients

The void coefficient was measured in the THOR[2] reactor and found to be -2.2×10^{-3} $\Delta k/k/\%$ void. This reactor has a 1 MW core very similar to the ULR.

The temperature coefficient was measured in the ULR over the range of 20°C to 32°C and found to be -0.705×10^{-4} $\Delta k/k/°C$.

4.5.6 Neutron Lifetime

The neutron lifetime was calculated as 7.2×10^{-5} seconds. This value may be compared to the measured value of 6.5 $\times 10^{-5}$ in the Borax I reactor.

4.5.7 Reactivity Effects of Core Geometry

The effect on reactivity of altering the core geometry reported by Hackney[1] is summarized below. If the reactor is loaded to a 26 element array of fuel surrounded by graphite

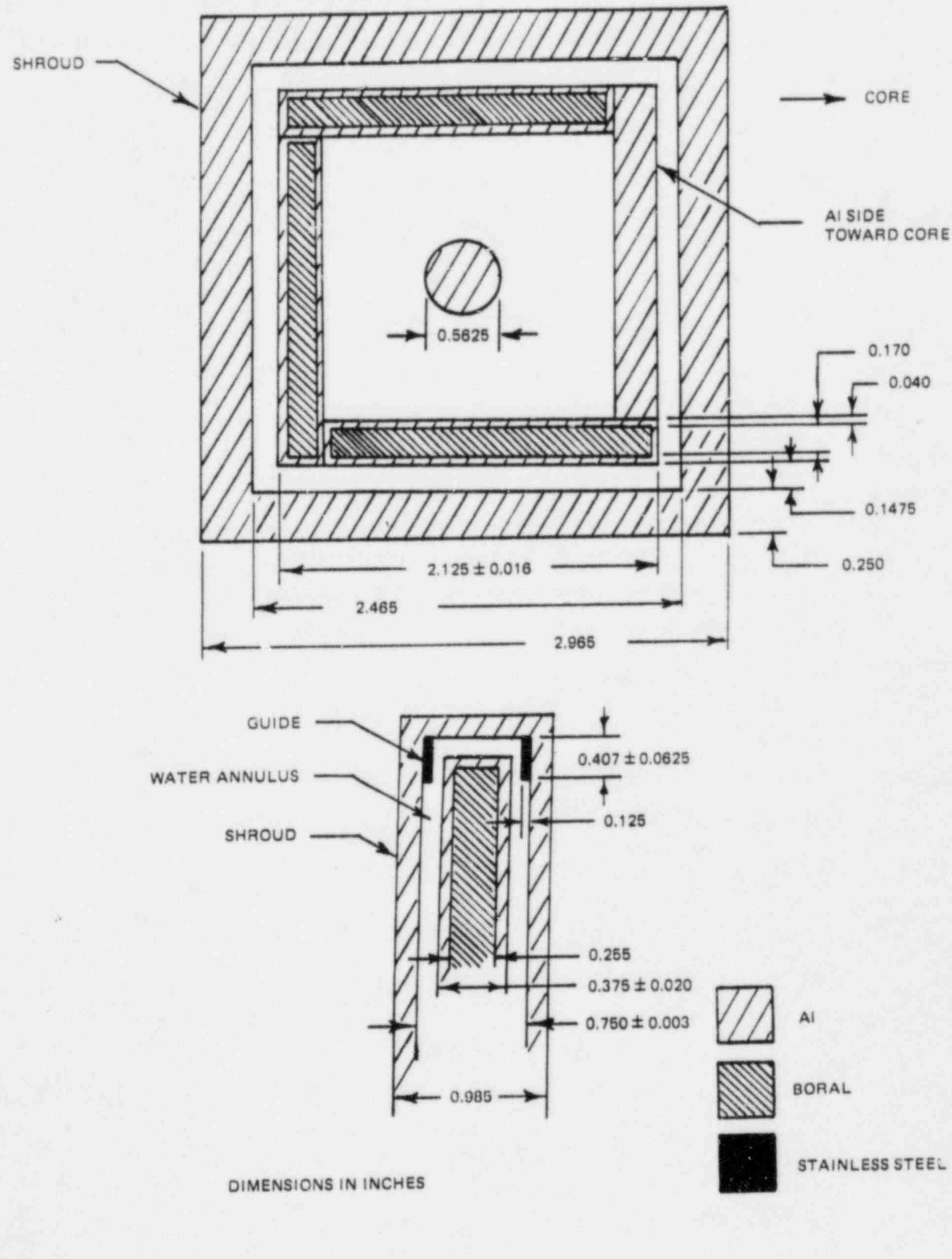


Figure 4.27. Sectioned View Of Regulating Rod (Top) And Control Blade

reflector, the following changes in reactivity occur:

		$\% \Delta k/k$
1.	Removal of one graphite element	-0.1 to 0.3
2.	Removal of graphite reflector	-3.3
3.	Replacing center fuel element with water	-3.4
4.	Replacing center fuel element with graphite	-2.8
5.	Adding one fuel element to the core	+2.0
6.	Flooding of one beamport	+0.06

4.5.8 Fluid Flow

Using the conservative assumptions of Chapter 9 at 1400 gpm primary coolant flow the core flow is as listed below for a 26 element core.

Fuel elements	1022 gpm
Leakage at control blade shrouds	185 gpm
Graphite reflectors	90 gpm
Radiation baskets	92 gpm
Neutron source holder	11 gpm

The flow velocity in the fuel channels is 2.7 ft/sec at full flow. The flow through the fuel elements will always be turbulent, since the Reynolds number at nominal conditions is 10,800.

At full flow, pressure drop across the core is 0.2 psi. Friction factors corresponding to smooth pipes were assumed.

4.5.9 Heat Transfer

The heat produced in the reactor is removed by the primary water flowing through the core at the rate of 1600 gpm. An overall hot channel factor of 2.94 was used in the calculations.

The steady-state operating conditions in the core are shown in the data summary in Chapter 1.

4.6 REFERENCES

1. Hackney,M.R., "Physics Analysis of the Lowell Swimming Pool Reactor," General Electric Co., GECR-4365 (1963).
2. Staff Members of the Reactor and Radio-Isotope Department, Institute of Nuclear Science, National Tsing Hua University, Taiwan, China, "One and a Half Year Experience with the Tsing Hua Open-Pool Reactor (THOR) and Its Utilization," Preliminary Draft 1962, NP14832.

5.0 ELECTRICAL SUPPLY AND DISTRIBUTION

5.1 ELECTRICAL SUPPLY TO ULR

Electricity is supplied at 13,800 V from a pole on Riverside Street, down to underground conduits leading to switch and metering gear near the Power Plant (see Figure 5.1), from which point it is distributed to the area buildings.

The Pinanski Building is fed by a 4160 V line which runs in underground conduit from the Power Plant, as shown in Figure 5.1.

5.2 PINANSKI BUILDING ELECTRICAL DISTRIBUTION

The incoming 4160 V supply is fed through two transformers in the Pinanski Building to two main distribution switchboards. The first transformer is rated at 750 kVA and supplies 277/408 V, 3 phase output to the first distribution switchboard; the second transformer is rated at 300 kVA and supplies 120/208 V, 3 phase output to the second distribution switchboard.

The 120/208 V main distribution switchboard has only one breaker to reactor equipment which supplies power to panel PPL-R1 inside the containment as shown in Figure 5.2. Other breakers feed other parts of the Pinanski Building.

The 277/408 V main switchboard supplies a number of components associated with the reactor including the Motor Control Center #1 and Motor Control Center #2, both of which are located in the basement of the containment building. The 277/408 V switchboard, in addition to feeding other parts of the Pinanski Building, also supplies normal power to the Emergency Distribution Switchboard which feeds the fire alarm, the Pinanski Building radiation monitoring system, the reactor ventilation supply fan, and the

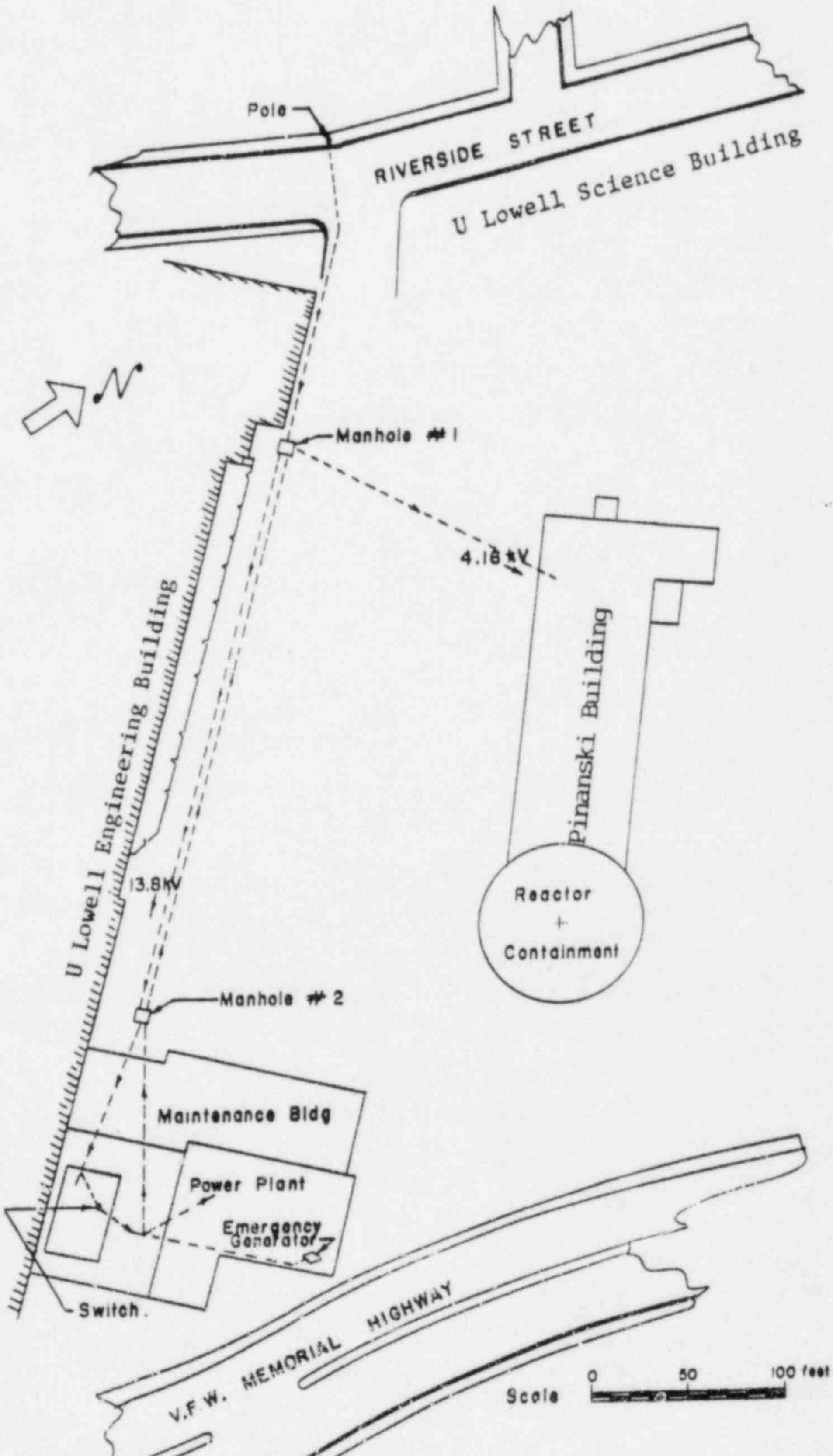


Fig. 5.1. ULR Electric Power Supply

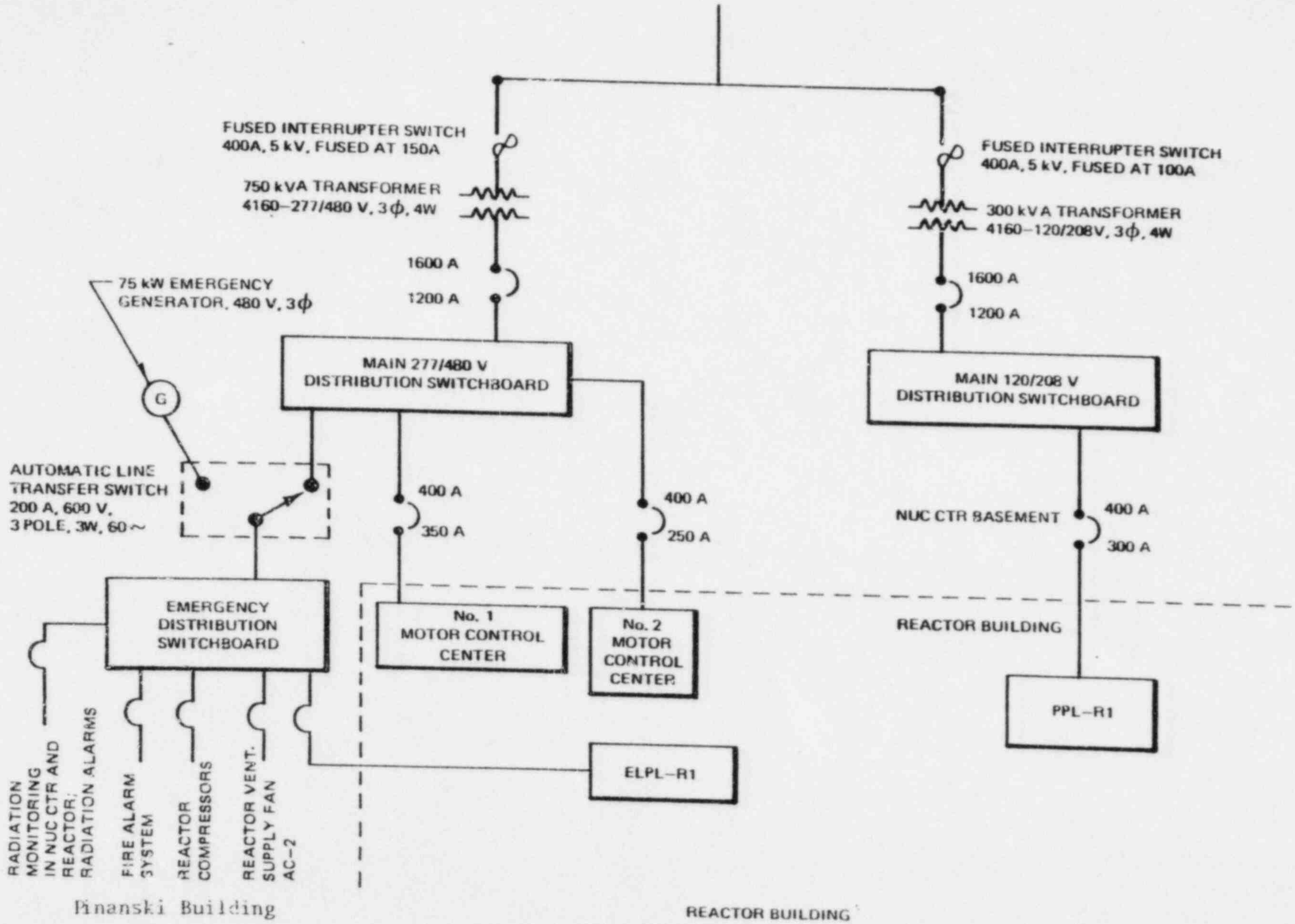


Figure 5.2. Pinanski Building Electrical Distribution

monitoring system, the reactor ventilation supply fan, and the reactor emergency panel ELPL-R1 (see Figure 5.2).

5.3 REACTOR ELECTRICAL POWER DISTRIBUTION

5.3.1 Normal Power

Under normal conditions power for the reactor facility is distributed from:

1) Motor Control Center #1;

2) Motor Control Center #2;

3) PPL-R1

4) ELPL-R1

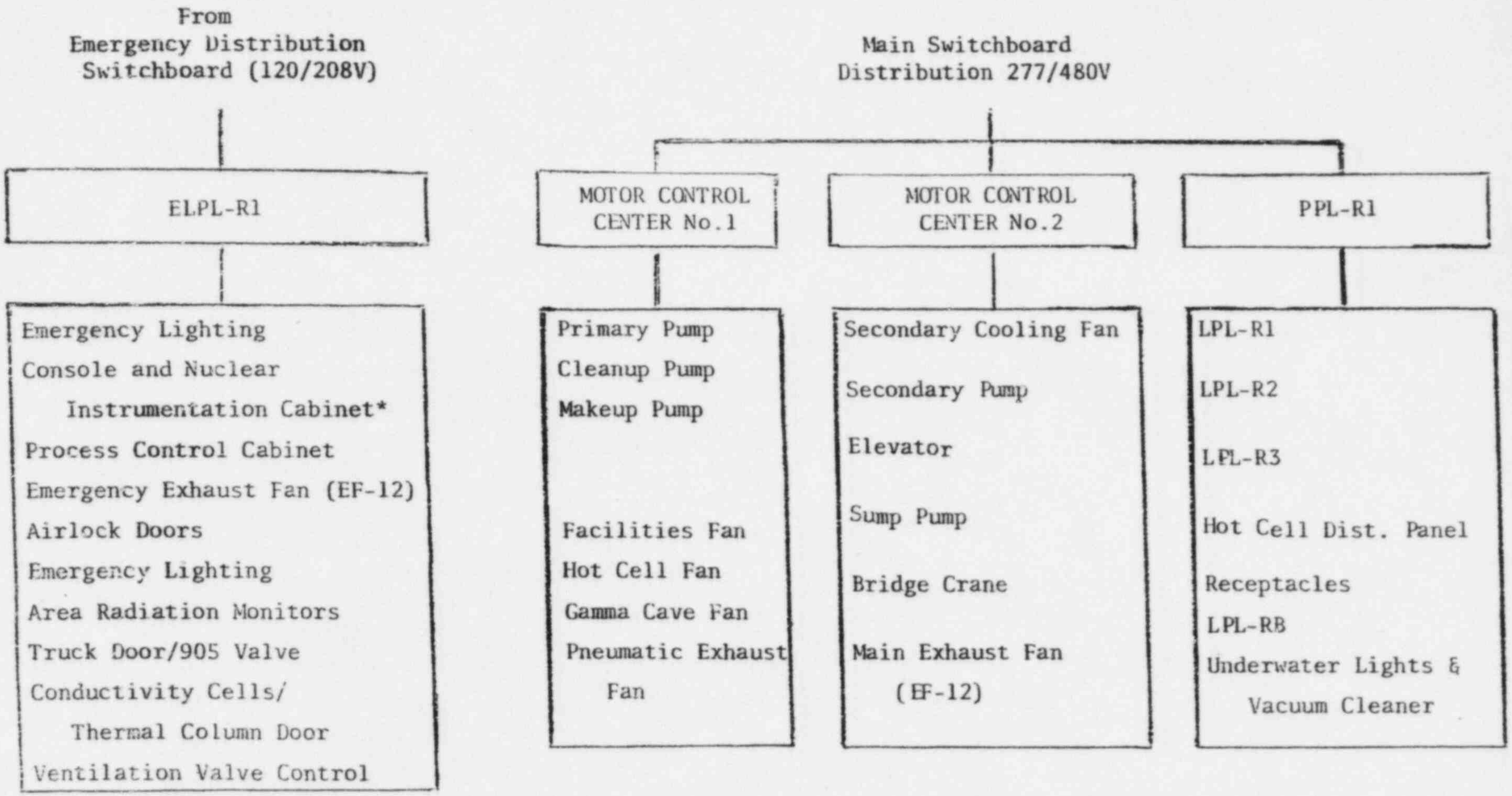
to the various process functions and lighting panels required for operation. The branch lines from these distribution centers to various lighting and receptacle distribution panel (designated LPL-R-) and terminal points are shown in Figure 5.3.

5.3.2 Emergency Power

Emergency power is supplied from the emergency generator to the equipment specified in Figure 5.2, and distributed inside the containment building from ELPL-R1 as stipulated in Figures 5.3 and 5.4. See also Paragraph 6.5, EMERGENCY GENERATOR SYSTEM.

5.4 REACTOR ELECTRICAL SIGNAL AND CONTROL WIREWAYS AND DUCTS

A system of open-topped wireways is situated underneath the Control Room floor and carries the multitude of cables needed for control of the facility (see Figure 5.5). These wireways connect the various penetrations through the floor beneath the control console and instrumentation cabinets and extend over to two standpipes which penetrate the pool level floor near the pool edge. One of the standpipes carries signal cable and the other



*See also figure 5.4

Figure 5.3. Reactor Building and Emergency Electrical Distribution

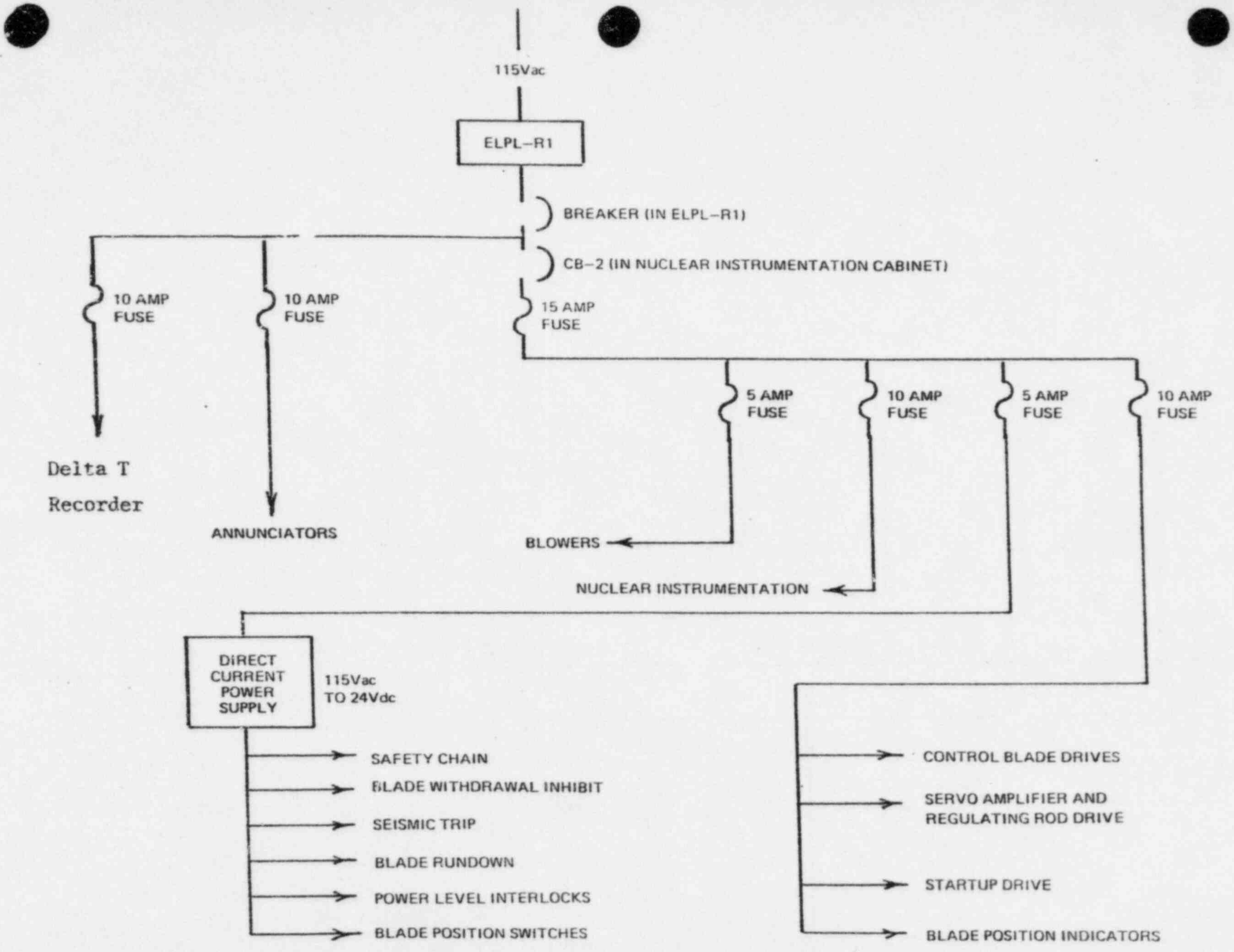


Figure 5.4. Schematic Diagram of Some Console and Nuclear Instrumentation

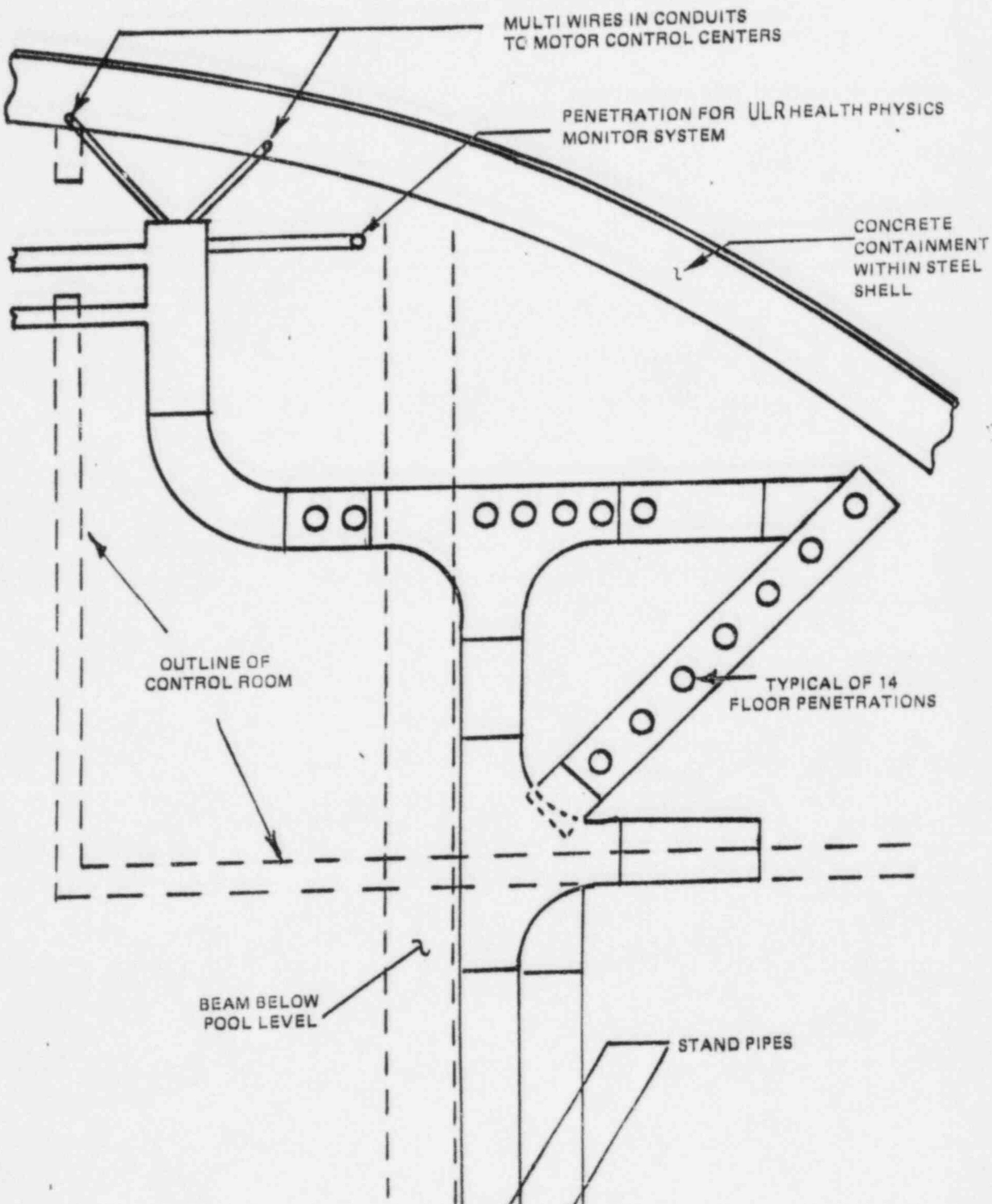


Figure 5.5. Open Wireways Beneath Control Room

power cable, in general, and two companion standpipes on the reactor bridge accept cables looped from the first set of standpipes.

The wireways lead to a dual 6 inch closed duct system which extends from the area under the Control Room to the open area of the containment building and then down along the containment wall through the beam level floor and into the pump room beneath (see Figure 5.6). The dual duct system (and wireway system) is designed to separate signal and power cables.

A small 4 inch closed duct extends halfway around the periphery of the containment building from the 6 inch ducts as shown in Figure 5.7.

From the main trunk lines described above, conduits lead to the various terminal points for process signal and control functions.

5.5 SAFETY SYSTEM WIRING

5.5.1 Scram System

The reactor scram system meets the single failure criterion for those channels which perform a protective action needed to prohibit violation of a Safety Limit under the conditions considered in Chapter 9, Safety Analyses.

Channel independence for the two redundant high level (120% of power)CIC ion chamber circuits from the console and wireway system through the standpipe to the reactor bridge has been achieved by use of flexible armored cable covered with a plastic coating. Hard conduit cannot be used because of the flexing requirements with bridge movement.

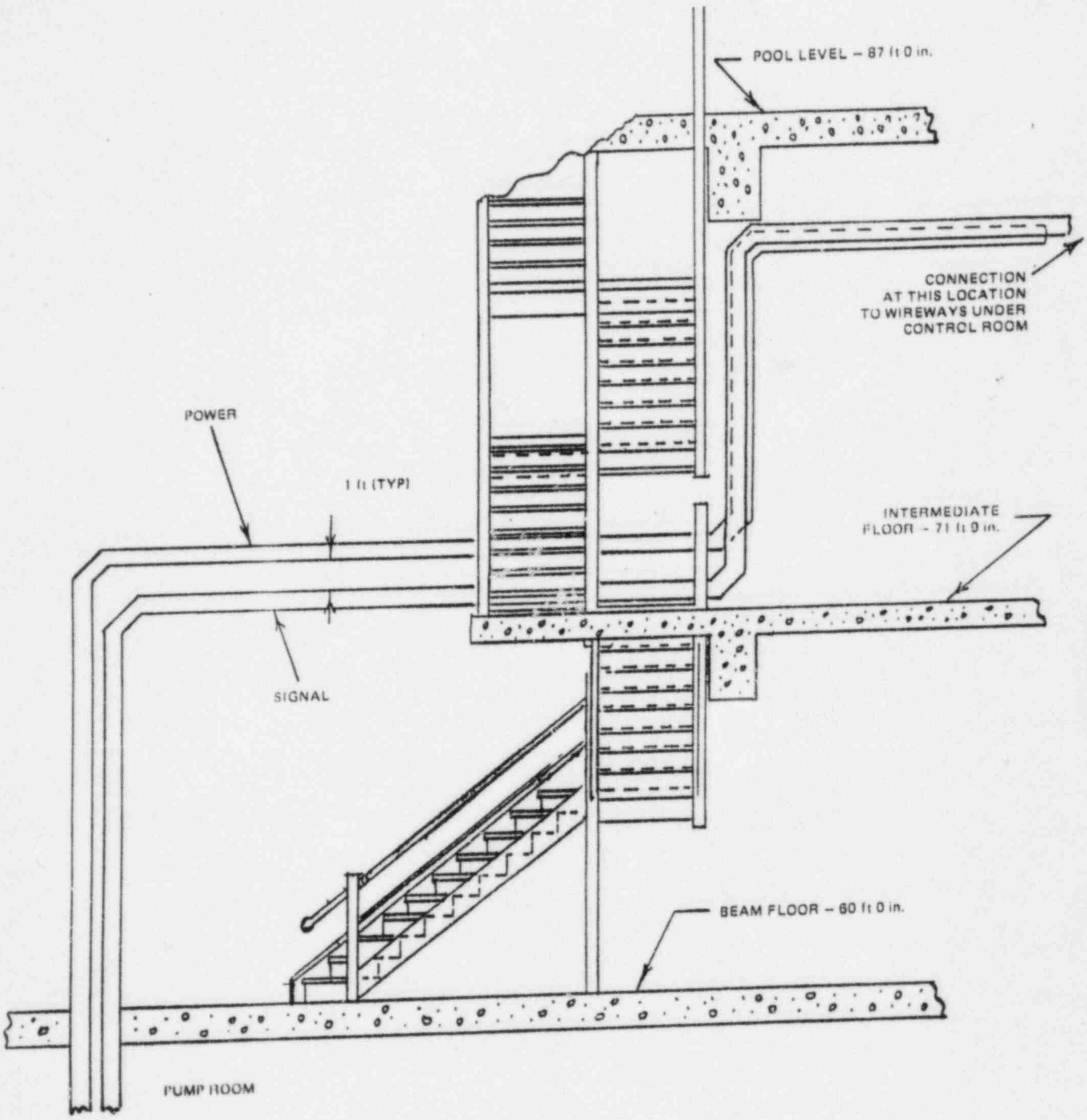


Figure 5.6. Six Inch Square Closed Electrical Duct System

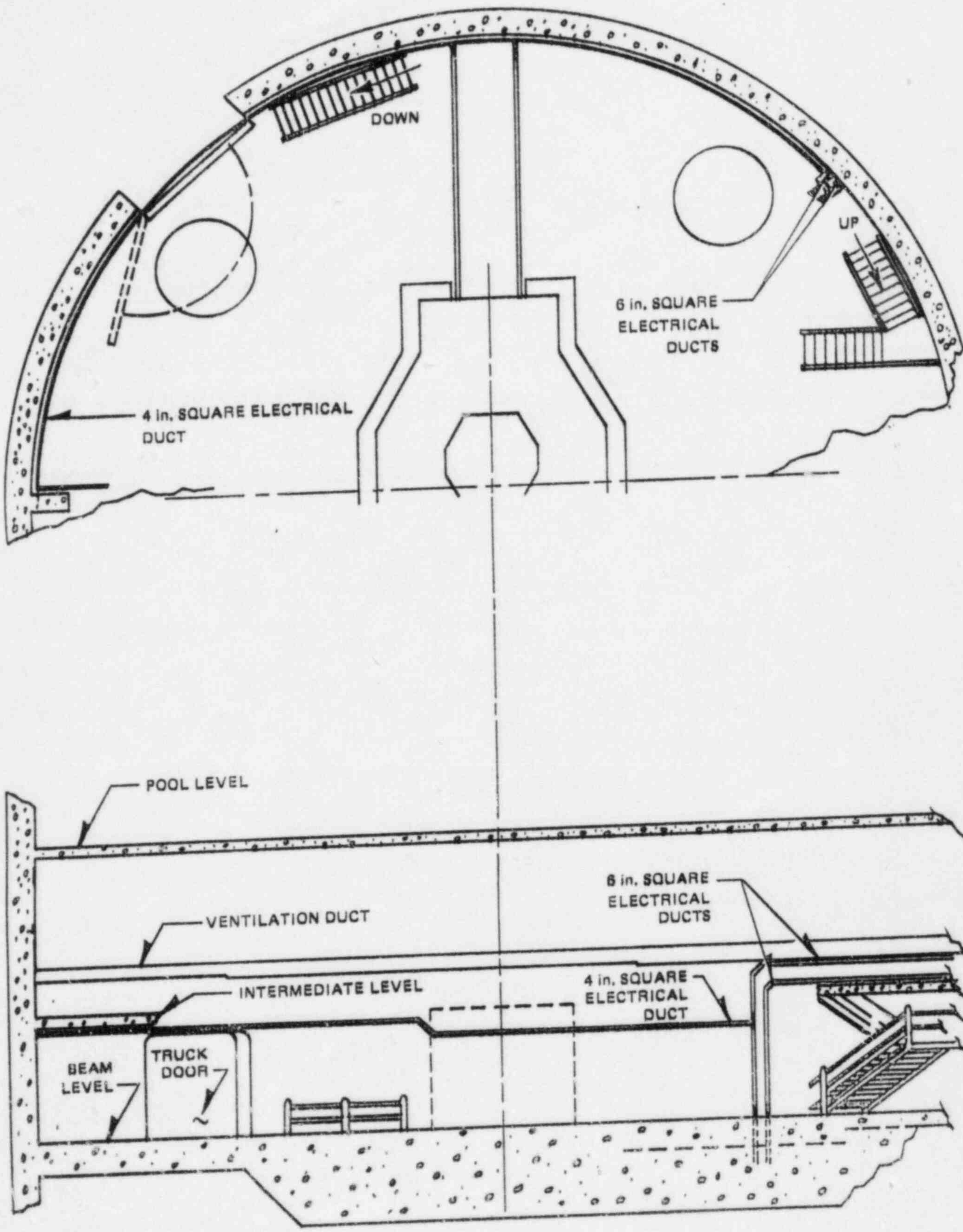


Figure 5.7. Four Inch Square Closed Electrical Duct System

Failure of any component of any protective function channel cannot lead to failure of the protective system. Loss of power initiates protective action by opening the scram bus.

A malfunction in the trip amplifier cannot prohibit the functioning of the scram bus and scram bus relays which furnish power to the trip amplifier which in turn powers the blade magnets.

The signal to the regulating rod servo system from one of the picoammeters could be affected by some feedback malfunction; this would cause failure of just that channel's protective function and not the failure of the protective system. The signal from the second picoammeter is not sent to the servo system, so there is no possibility of feedback from the servo system affecting the scram capability of the second picoammeter high level channel.

5.5.2 Containment System

The containment system meets the single failure criterion since there is no credible single failure that would incapacitate the building isolation described in Chapter 10 Appendix.

Power failure results in initiation of protective system action.

The personnel airlocks have double doors with interlocks to prevent opening both doors concurrently. A scram occurs if both doors open or both inflation seals are broken concurrently, so that double failure would be required to open both seals. The truck door has a single scram contact if the seal is deflated, but this door is padlocked and not used during operation. Even if a scram did not occur, loss of the seal would be recognized by difficulty in maintaining building negative pressure.

The actuation of the General Radiation Emergency alarm system or the Limited Radiation Emergency alarm system (see Chapter 10 Appendix), will initiate emergency action including closure of the containment building, reactor shutdown, containment building evacuation, and formation of the Emergency Team. Both alarm systems as well as the Public Address system are fed by emergency power from the emergency generator (see Table 6.2 of Chapter 6).

Containment closure is achieved by deenergizing solenoids in the various building valve air supplies. This releases the air holding the valves open and allows containment closure. Because the operation of the emergency exhaust system requires operation of its own unique valve, power is fed to this system's solenoid while containment is in effect; the power feed line to the emergency exhaust valve solenoid is isolated in a separate conduit from the power feed lines to other containment valves to ensure that a single failure cannot result in inadvertant opening of containment valves.

6.0 AUXILIARY SYSTEMS

6.1 FUEL STORAGE AND HANDLING

6.1.1 New Fuel Storage

New fuel elements are kept in storage racks located in the reactor pool. Each rack holds nine (9) fuel elements in a planar array. Analysis of this geometry assures that there is no possibility of inadvertant criticality.

At the present time, there are six (6) fresh fuel elements located in storage racks in the reactor pool. The six elements consist of four regular fuel elements, one element with regular loading but removable plates, and one element with half the normal fuel loading per plate and per element. New lower enriched fuel elements for the UL reactor are expected in the near future. Storage for these elements will be addressed at a later date.

6.1.2 Spent Fuel

Fuel storage racks for 72 elements are provided in the pool. Each rack holds 9 elements in a planar array so that there is no possibility of inadvertant criticality with this kind of storage. Positions for eight racks are located along the walls of the stall pool and for eight more along the walls of the bulk pool so that it is possible to locate all of the storage racks in either end of the pool. Figure 6.1 shows the possible locations of storage racks.

6.1.3 Fuel Handling System

New fuel elements will be ordered and received to replace spent fuel elements. New elements will be shipped in approved shipping containers, which will be opened to remove and inspect the elements in preparation for storage in the reactor pool. No more than four (4) fuel elements will be out of mechanically controlled geometry at any one time, to assure no inadvertant criticality. Fueling of the core is done manually from the top of the use of a special handling tool with a bayonet type fitting, (see Figure 6.2), which is kept under lock and key when not in use. The Reactor Supervisor or Senior Reactor Operator must authorize movement of any fuel element, and must be present during the movement. A detailed set of procedures will be written for the receiving and handling of new fuel when future needs arise. In addition to the fuel handling tool, a general purpose grapple consisting of a long handled hook can be used for the movement of radiation baskets and graphite reflectors. Overall length of this grapple is about 32 feet (Figure 6.2)

6.2 COMPRESSED AIR SYSTEMS

Compressed air to operate the various containment valves is supplied by a two compressor system located in the Fan Room exterior to the reactor. The two compressors feed a 6 ft^3 storage tank with 200 psi air, and air from both supplies is rough filtered and then cleaned and dehumidified by an Ingersoll-Rand condenser. All valves fed by this compressor are of fail-safe design so that loss of air causes them to close, except the by-pass valve (Valve F of Figure 3.5), which can be operated several times by the reserve air supply in the storage tank (Valve F can also be manually operated).

A separate compressor and tank is located inside the reactor on the intermediate level floor. This supplies air to operate the

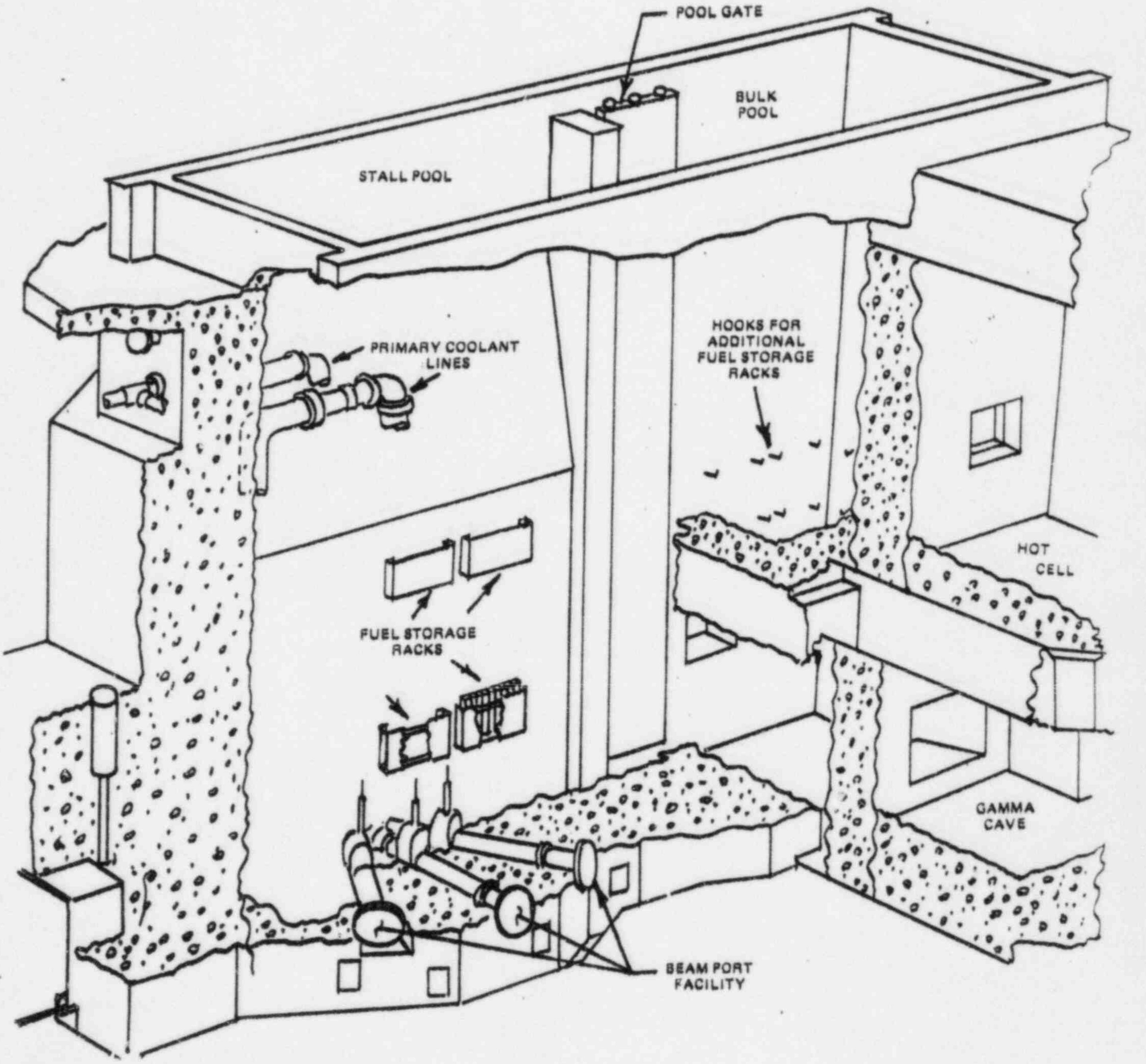


Figure 6.1. Location of Fittings for Fuel Storage Racks

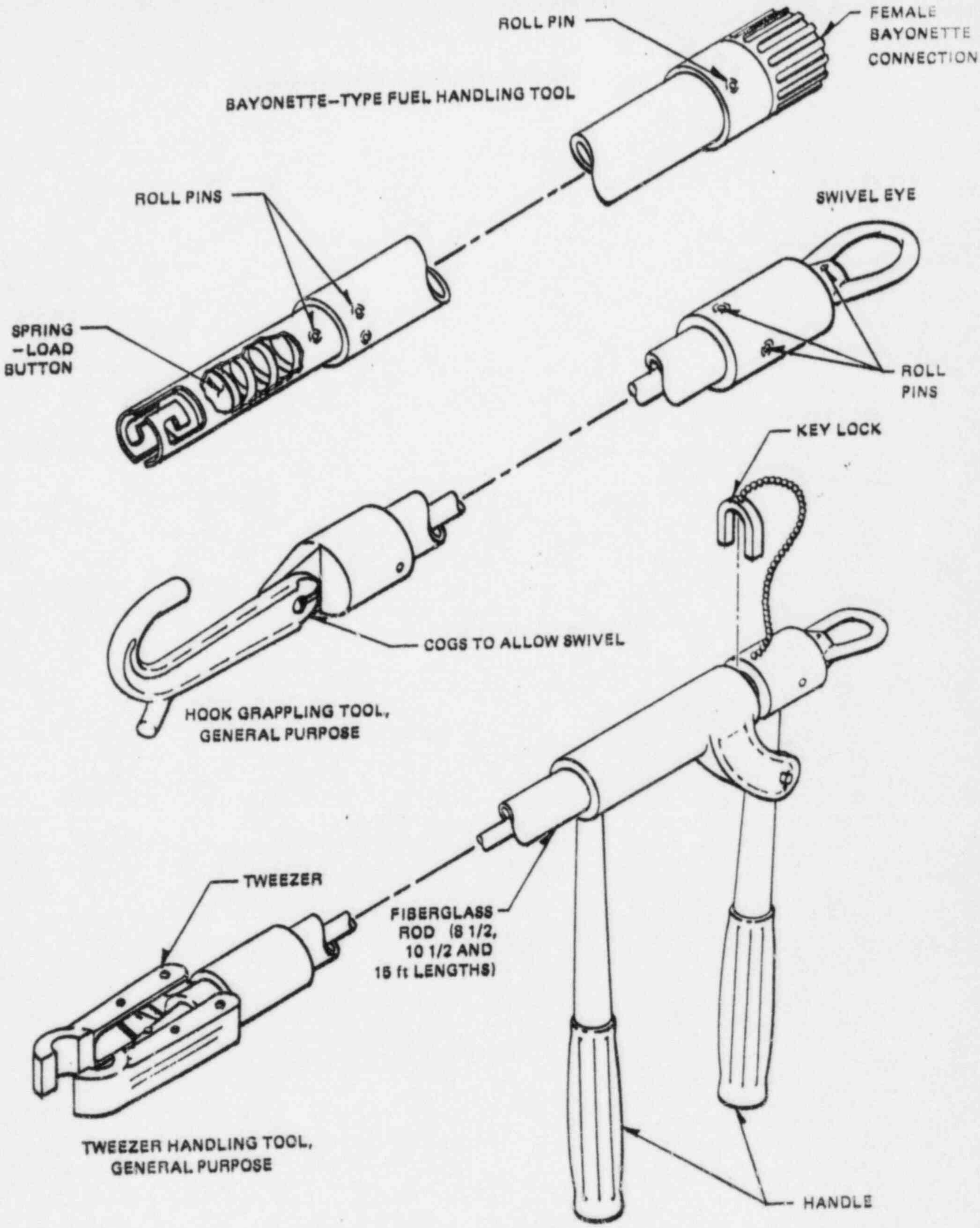


Figure 6.2 ULR Handling Tools

air lock doors as well as service air throughout the reactor, including the Gamma Cave safety interlock system. A 9.3 ft^3 reserve tank stores air at ~ 60 psi; this reservoir is enough to open and close a set of air lock doors several times.

6.3 COMMUNICATIONS SYSTEM

6.3.1 Intercom System

A multioutlet intercom set has stations both inside the reactor containment and outside at various locations in the Radiation Laboratory.

a. Reactor control room

b. Intermediate level floor, (Hot Cell area)

c. Beam level floor, (Thermal Column area)

d. Reactor basement, (Pump Room and Hot Lab)

e. Reactor Supervisor's Office

f. Radiochemistry

g. AC-2 Fan Room

h. Reactor Operations

Intercommunication is possible from any single station to any single station. Communication is unidirectional; the initiating station receives until the initiating station pushes a button which causes the called station to receive. Three station communication is not possible, but communication between station a and station b, for example, does not prohibit concurrent communication between any two other stations.

A second intercom system has been installed to provide communication between the reactor control room and the Gamma Cave/Rabbit 1A area.

6.3.2 Public Address System

The reactor is furnished with a public address system which has speakers of either trumpet or diaphragm type situated in multiple locations on all floors. The intent is to make announcements audible to anyone inside the containment.

6.3.3 Sound Powered Headset System

A sound powered headset system is installed with jack outlet in the control room, and at the reactor bridge, two along the reactor pool, and two outlets in the pump room. This system allows continuous bidirectional communication between stations. As experience dictates, other stations will be added up to a maximum of ten at strategic locations throughout the reactor. All stations in this system send and receive continuously so that this is a multidirectional hookup.

6.3.4 Access System

A two station intercom exists between the ground floor air-lock access doors and the control room. This system is used for communication between persons wishing to enter the reactor and the reactor operator so that the operator can activate the "permit-switch" release when appropriate, as discussed in Paragraph 3.1.2.1.

6.4 LIGHTING SYSTEMS

6.4.1 Normal Lighting

Suspended from the dome or ceiling of the containment structures are fourteen dual fixture 400 W mercury vapor lamps. These concentrate light at the pool top but also serve for general area illumination for the open space from ceiling to beam level floor.

Three hundred watt incandescent lights are distributed throughout the various floor levels and the control room as summarized in Table 6.1.

6.4.2 Emergency Lighting

When normal power is lost, the emergency generator starts and supplies power to the emergency supply panel. Lighting during the delay time is furnished by Emergency Lighting Units manufactured by Electro Powerpacs Corp., Cambridge, Mass. These are six volt wet-cell battery units with a trickle charger fed by normal 120 Vac power which automatically power 6 Vdc lamps upon loss of house power. Return of house power resets the unit. The ratings of these Model No. DR409/50LAT units are given below.

Input: 60 hz: 120V; 1 amp.

Output: 7.5 Vdc: (max.); 32 amps.

Each battery unit feeds two 25 Watt tractor lights. The general location of each unit is given in Table 6.1.

After the emergency generator comes on line, emergency lighting consists of full control room lighting; emergency 30 Watt wall bracket lamps mounted near stairways and other vital areas, exit signs over all air locks, doors, etc. The number of these in each general area is indicated in Table 6.1.

6.5 EMERGENCY GENERATOR SYSTEM

6.5.1 Pinanski Building Emergency Generator

An emergency power generator is situated in the basement of the Pinanski Building as shown in Figure 6.3. This is a 70 kW Kohler Electric Plant, Model 85R78, which supplies 3 phase, 60 hz,

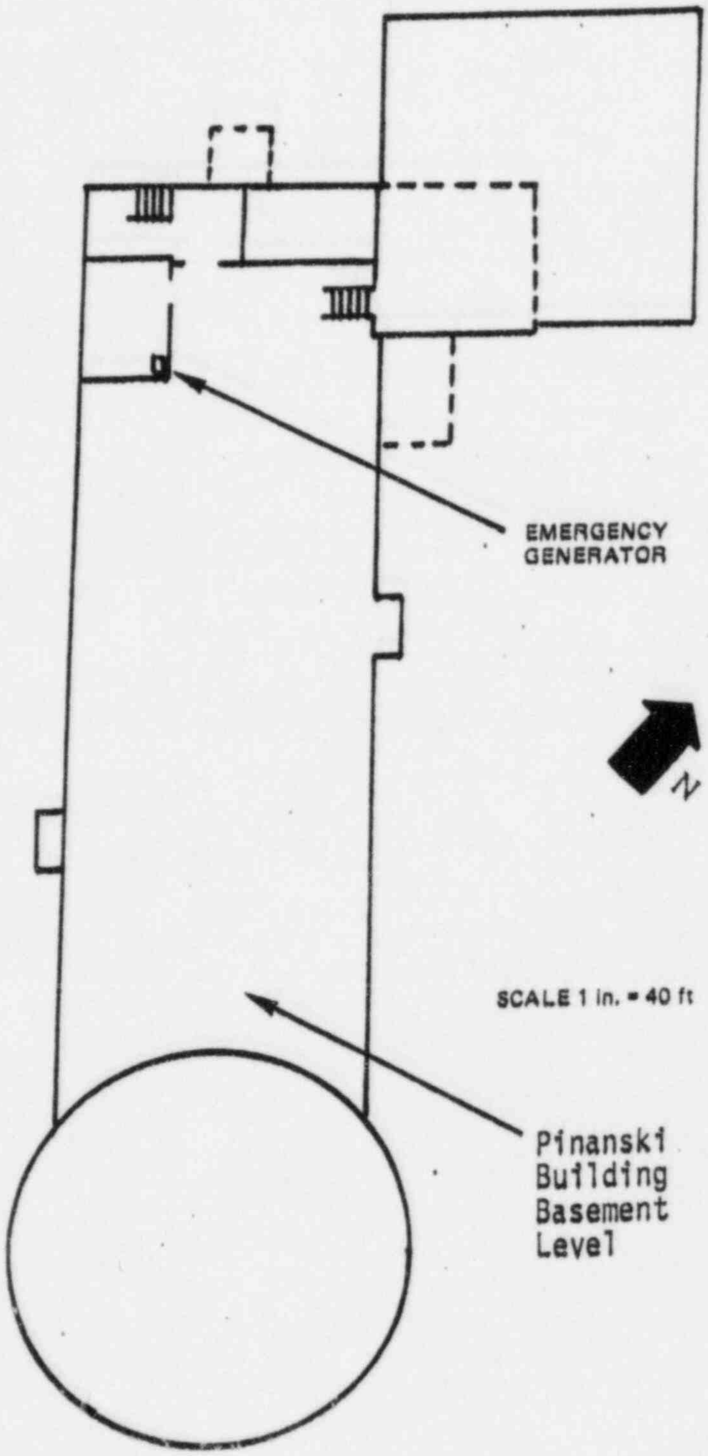


Figure 6.3 Emergency Generator Location

277/480 Volt power using low pressure natural gas as fuel. Starting of the plant motor is automatic upon loss of house power through use of 24 Vdc wet cells which are kept charged by a trickle charger fed by house power. Control is accomplished by a Westinghouse Robonic transfer switch, Catalog No. RON52, Style No. 42-E-2127.

Of the available emergency power, about 12 kW is used for emergency lighting and critical motors, instruments, etc., in non-reactor related emergency use leaving some 58 kW for reactor use. A list of the systems that are powered under emergency conditions is given in Table 6.2.

6.6 ELEVATOR

A six ton capacity freight elevator runs from the basement to the top or pool level in a shaft situated inside the southeast wall of the containment building. This elevator has an emergency alarm bell activated by a push button inside the elevator. In addition, an access trap-door exists in the elevator ceiling.

6.7 POLAR CRANE

A fifteen ton polar crane which is situated above the pool level floor serves the pool level and the open area down to the beam level floor. The crane has an auxiliary hook for loads up to two tons, and all hooks are equipped with safety latches. Both hooks are manipulated with a pendant control.

6.8 REFERENCES

1. See "Application for License for Storage of LTI Reactor Fuel Elements," submitted to USAEC on October 20, 1970.
2. H. C. Paxton, et al., "Critical Dimensions of Systems

Containing ^{235}U, ^{239}Pu, and ^{233}U", pp. 10, 24, 29, 30, ORNL (1965), also TID 7028.

TABLE 6.1

SUMMARY OF LIGHTING IN THE ULR

	Normal Power		Emergency Generator Power		Battery Power	
General Location	Bulb Quantity	Watts Each	Bulb Quantity	Watts Each	Bulb Quantity*	Watts Each
Containment ceiling	14 dual=28 (mercury)	400				
Control room	7	300	7	300	2	25
Pool level (excluding control room)	1 (exit)	67	4	30	2	25
	4	30	1 (exit)	67		
	2	60	2	60		
	2	100	2	100		
Intermediate level	36	300	4	30	2	25
Beam level	37	300	4	30		
	2 (exit)	67	2 (exit)	67	4	25
	4	30	2	100		
	2	100				
Basement pump room	32	300			2	25

TABLE 6.1 (Continued)

General Location	Normal Power: Bulb Quantity	Normal Power: Watts Each	Emergency Generator Power: Bulb Quantity	Emergency Generator Power: Watts Each	Battery Power: Bulb Quantity*	Battery Power: Watts Each
Basement hot laboratory area	31	300	3	30	2	25
Hot Cell	2 (mercury)	1000				
	4	200	4	200		
Elevator shaft	2	100	1	100		

*Each 2 bulbs indicate one battery pack.

TABLE 6.2

EMERGENCY GENERATOR USE

System Powered	Approximate Power Required (kW)
Ventilation Supply Fan AC-2	15
Radiation Monitor System	3.6
Emergency Horns, Flashing Lights, etc.	1.1
Emergency Exhaust System	7.5
Emergency Reactor Lighting	4
Intercom and Public Address Systems	0.5
Console Power	2.3
Nuclear Instrumentation	2.3
Process Control Cabinet	2.3
*Compressor Motor in Reactor	1.5
Air Lock Doors (during operation)	2.3
Fire Alarm	2.3
*Compressor Motors in Fan Room	4.6
Nuclear Center Emergency Requirements	7
Accelerator Emergency Requirements	5

*Delayed automatic reset

7.0 RADIOACTIVE WASTE MANAGEMENT

7.1 SOURCE TERMS

7.1.1 Source Term for Liquid Radioactive Waste

A mixed bed ion exchange column is located on a side stream loop off the primary coolant loop for the purpose of removing radioactive corrosion products and fission products contained in the primary loop. Regeneration of this clean-up demineralizer is the primary source of liquid radioactive waste. The effluent from the regeneration goes to a 3000 gallon sump located in the pump room. This sump also handles the liquid waste from the laboratory sink drains, beam port drains, gamma cave floor drain, hot cell floor drain, unused medical embedment floor drain and the drain for the transfer port between the pool and the hot cell.

The bulk of the liquid radioactive waste is from the regeneration of the clean-up demineralizer. This liquid can be transferred to appropriate storage tanks for analysis and treatment. All the liquid radioactive waste produced in the reactor area is contained and controlled such that there is no contact with or release to the environment until the appropriate analysis is performed.

7.1.2 Source Terms for Gaseous Radioactive Wastes

Gaseous wastes may be generated during normal operations as a result of neutron activation of ^{40}Ar in air, lesser activity levels of other air activation products, release of fission products from the core with subsequent release of fission product gases from the pool, and airborne activities released as a result of experimental procedures.

The activation of ^{40}Ar to produce ^{41}Ar may occur whenever air is in contact with a neutron radiation field. ^{41}Ar production occurs routinely in the beam ports, thermal column, and pneumatic tubes and primary coolant water. Production as a result of irradiation of air in the thermal column case vent, the pneumatic tubes, and the primary coolant is expected to be negligibly small compared to possible production in the beam ports. This aspect is discussed in Paragraph 7.3.5.

Airborne fission product releases are extremely small and are also discussed in Paragraph 7.3.5. Such releases result from the diffusion of fission product gases (resulting from fission product releases from the core to the primary coolant) from the pool water to the containment building air.

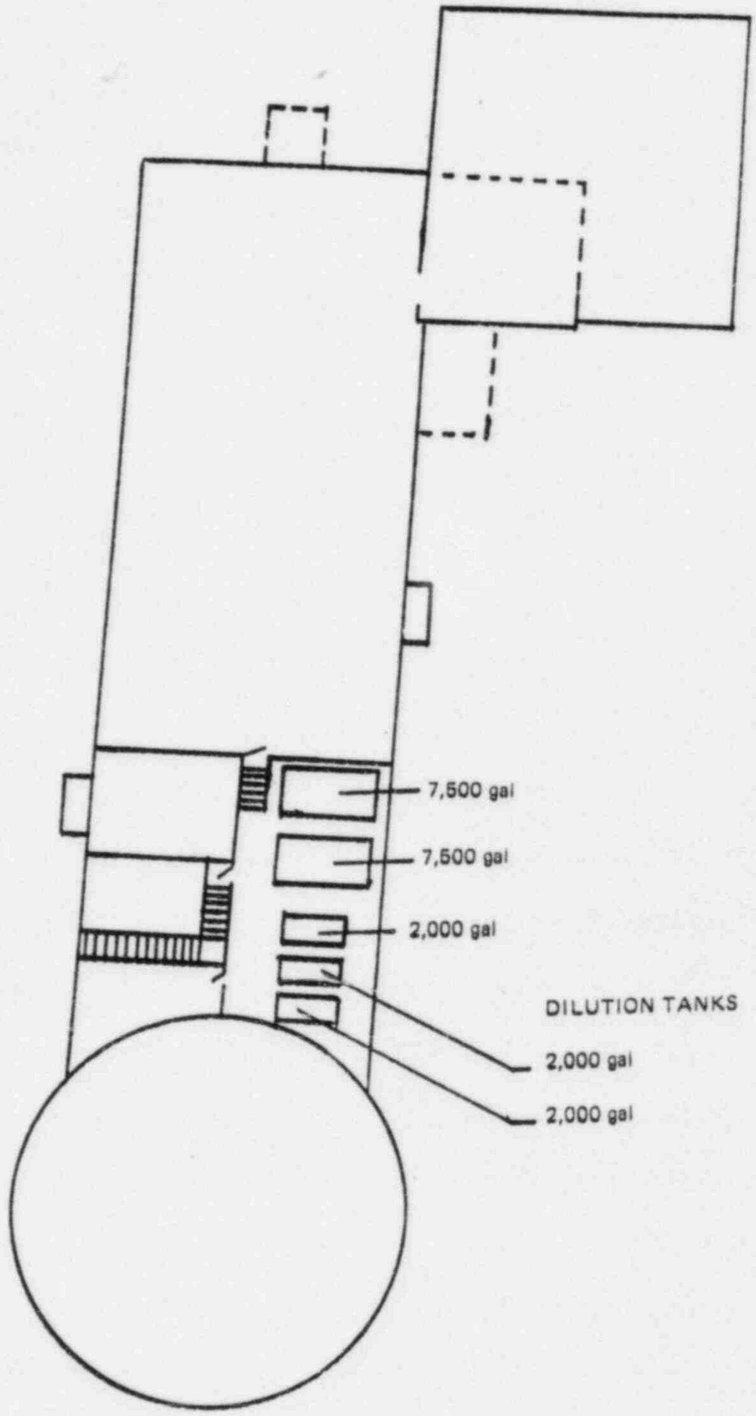
7.2 LIQUID WASTE SYSTEMS

7.2.1 Design Objectives

All liquid waste with a potential of being radioactive is handled in a separate waste system having no physical connection, within the reactor area, to either sanitary drains or the domestic water system. The procedure is to store all liquid radioactive waste, perform an appropriate analysis for radionuclide concentration, and dispose of the liquid waste under controlled conditions in accordance with applicable regulations.

7.2.2 Systems Description

All the liquid waste from the reactor area drains into a 3000 gallon sump. The liquid is transferred to the liquid radioactive waste storage tanks located in the basement of the Pinanski Building as shown in Figure 7.1 where it is analyzed and treated



Pinanski Building Basement

SCALE 1 in. = 40 ft

Figure 7.1 Liquid Radioactive Waste Storage Area accordingly.

A more detailed description of the primary and secondary cooling system is give in Chapter 4, along with the appropriate drawing (Figure 4.11).

7.2.3 Operation Procedure

When the make-up demineralizer needs regeneration (as indicated by water conductivity measurements) the reactor is shut down and a sufficient length of time allowed for the decay of the short-lived radionuclides.

When radiation levels permit, regeneration of the demineralizer can proceed according to the detailed instructions outlined in the manufacturer's manual.

Once the sump in the reactor pump room has sufficient volume to warrant its being emptied (at least once each time the clean-up demineralizer is regenerated) the liquid is transferred to the waste storage tanks where it is analyzed to determine the concentration and quantity of various radionuclides. After the radiochemical analysis is performed, the liquid can be stored to allow decay of certain radioisotopes (if necessary) and then transferred to the sanitary sewer. If warranted, the radioactive material could be transferred to a commercial radioactive waste contractor.

7.2.4 Performance Tests

Continuous monitoring of the radiation levels associated with the cooling water is described in the Emergency Plan. Appropriate measurements of pressure, temperature, flow, conductivity, pH and radiochemical content of the primary and secondary cooling water are performed to determine that the condition of the coolant is within normal expectations.

For the purpose of describing the performance checks for the liquid radwaste system the cooling water is divided into four sections:

POOL: The pool water is sampled periodically to determine water quality and evaluate the presence of radionuclides.

PRIMARY LOOP: An external gamma monitor and delayed neutron detector are used to indicate fission product leaks from the fuel to the primary coolant.

CLEAN-UP LOOP: Continuous water conductivity measurements are made to determine the performance of the clean-up demineralizer.

SECONDARY COOLANT: A sample of water is collected once each operating day to determine the presence or absence of ^{24}Na in significant quantities.

7.2.5 Liquid Release

The release of liquid radioactive waste to the environment is via the sanitary sewer system and therefore is limited to the conditions set forth in 10CFR-20.303. The annual average gross beta release quantity is 0.711 μCi/yr. This is based on actual monthly releases over the past 5 years.

7.2.6 Release Point

The outlet to the sanitary sewer system is located in the liquid radioactive waste storage room. This sewer then ties into the Lowell City sewer main which ultimately empties into the Merrimack River (see Figure 7.2).

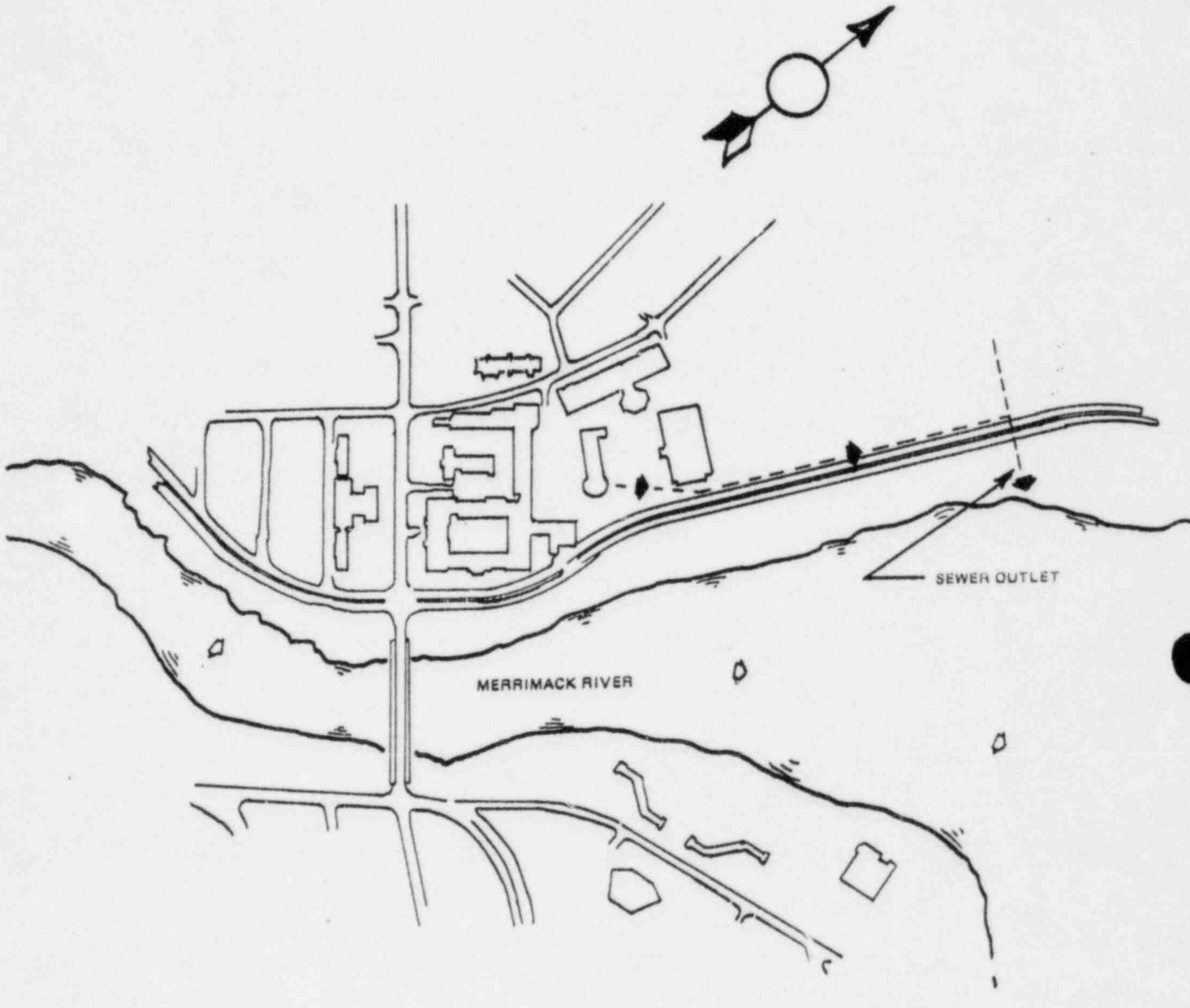


Figure 7.2 UL and Lowell City Sewer Lines

7.2.7 Dilution Factors

Data obtained from the City of Lowell Water Department which have been verified by UL measurements show a UL water use of at least 1×10^{11} cubic centimeters per year. This water drains into the same line as the sanitary sewer outlet of the Pinanski Building.

Further dilution is obtained in the Merrimack River. Data from the Corps of Engineers[2] show the minimum recorded flow rate as 199 cubic feet per second (1.8×10^{14} cm^3/yr). This flow would reduce the sewer effluent concentration by a factor of 1800.

7.2.8 Estimated Doses

The closest point of human consumption of water from the Merrimack River is through the water treatment plant in Lawrence, Massachusetts (see Paragraph 7.6). At an average release of 0.711 millicuries per year, over the past 5 years, this results in an average annual concentration of 7.11×10^{-9} $\mu Ci/cm^3$ in the sewerage outfall. Applying the dilution factor of 1800 results in a downstream concentration of less than 3.95×10^{-12} $\mu Ci/cm^3$. Assuming that a concentration of 10^{-7} $\mu Ci/cm^3$ is equivalent to a whole body dose of 500 mrem/yr, a concentration of 3.95×10^{-12} $\mu Ci/cm^3$ is equivalent to 0.020 mrem/yr. Extrapolating this to a population of 55,000 (population served by Lawrence water system) results in an annual population dose of 1.09 man-rem.

7.3 GASEOUS WASTE SYSTEMS

7.3.1 Design Objectives

The gas waste handling systems are designed to prevent production and/or release of airborne radioactive wastes in amounts, as indicated in 10 CFR parts 20 and 50, which would result in doses to individuals in the reactor containment building and in the external environs in excess of applicable limits. Description of the gas waste systems, operating procedures, and releases as they pertain to these overall design objectives are presented in the following paragraphs.

7.3.2 Systems Descriptions

The ducts, blowers, and vents which direct gaseous flow are indicated in the line diagram, Figure 3.5 of Chapter 3. Also shown are filter systems designed to remove radioactive wastes from the stream. Exhausts from the pneumatic tubes, hot cell, gamma cell, and experimental basement hoods are individually filtered through absolute filters. (Rated at $\leq$ 0.05% penetration for 0.3 micron dioctylphthalate particles and designed to withstand 100% relative humidity at temperatures between 70° and 100°F for at least twelve hours.)

The common exhaust from the thermal column case vent and beamport and unused medical embedment drain line vents is also filtered through an absolute filter. The emergency exhaust is filtered through a two-inch charcoal filter in addition to an absolute filter.

7.3.3 Operating Procedure

Design operation calls for continuous active venting of certain reactor facilities which prevents the unwanted accumulation of possibly high concentrations of radioactive gaseous wastes. This is particularly pertinent with respect to drain line venting of the beamports, thus preventing appreciable buildup of Argon-41 with the attendant possibility of an inadvertant venting of this to the containment building. Similarly, pneumatic tube exhaust prevents the buildup of gaseous waste products in this system.

Under design operating conditions the thermal column case vent, the beamport and unused medical embedment drain line vents, and the pneumatic tubes will be exhausted continuously; the air mover providing common exhaust for the thermal column, beam tubes, and medical embedment is rated at 600 cfm; the pneumatic tube exhaust fan is rated at 200 cfm. The gamma cave and hot cell exhausts, provided by independent 600 cfm blowers, operate when these facilities are in use or as otherwise considered appropriate. All the above facilities exhausts are vented through a common duct to the main building exhaust line outside the reactor building. The experimental hood in the containment building basement is exhausted via a direct conneciton to the main building exhaust line within the building, and, hence, is vented continuously while the reactor is in operation.

In order to reduce ^{41}Ar production, the core end of the beam ports are plugged with hollow, closed end plugs filled with nitrogen.

The emergency exhaust system will operate automatically, under conditions specified in Paragraph 3.4.2.3; it is also subject to manual operation, for instance, in a situation where it might be considered desirable to reduce airborne radioactivity levels by exhaustion, although the containment building internal pressure status does not demand automatic initiation of the system. The emergency exhaust blower is rate at 320 cfm and delivers exhaust via a duct to the main building exhaust line outside the building.

Appropriate lters, as indicated in the above section, are in place when an of the exhaust systems described is in operation.

7.3.4 Performance Tests

Air flow rates in facilities exhaust systems and the emergency exhaust system, as well as stack flow rates are measured at regular intervals as circumstances of facilities operations dictate.

Absolute filters in facilities exhausts and the emergency exhaust system are subjected to appropriate particulate penetration tests (DOP and/or radioactive particulates are normally used) at regular intervals as specified in the Facility Technical Specifications.

The activated charcoal filters in the emergency exhaust system are subjected to iodine penetration tests at regular intervals.

Two continuous air particulate monitors in the reactor building and the stack particulate and gaseous monitor continuously respond to the airborne radioactivity levels in the reactor building and in the stack effluent, thus providing a mechanism for noting possible failure or fault in the performance of certain gas waste removal systems.

7.3.5 Gaseous Releases

Averaged over a five year period, the gaseous releases of ^{41}Ar (the only identifiable radioactivity produced by the reactor) is 0.34 μCi per/sec.

7.3.6 Release Points

The only design vents which open to the atmosphere by routes other than the reactor stack are the acid vent from the basement area and the sanitary system vent from the control room. Both of these are valved to operate automatically in the ventilation containment system, and neither of these vents delivers any significant radioactive gaseous waste to the environment. The double airlock doors which allow normal entrance and egress to and from the containment provide additional pathways to the external atmosphere via the Pinanski Building, but the extent of containment air venting to the atmosphere by this route is negligibly small.

7.3.7 Dilution Factors

Because of the complexity of the reactor site, particularly in terms of the presence of nearby buildings and the modest height of the reactor stack, it is extremely difficult to analytically define and describe the micrometeorological situation in the area of the reactor and, hence, to estimate a realistic dilution factor for purposes of radioactivity release.

Therefore, in order to estimate the consequences of releases of radioactivity from the stack, no credit has been assumed for meteorological dispersion.

7.3.8 Estimated Doses

From a dose point of view the only gaseous waste product released of any note during normal operation is expected to be ^{41}Ar.

Since no atmospheric dispersion is assumed to occur, the release from the stack of ^{41}Ar has been assumed to occur under such adverse physical meterological conditions that the emission is trapped at ground level, and effluent from the stack proceeds, over the duration of ^{41}Ar release, to cause the expansion of a semispherical cloud. Under the assumptions of this very restrictive model the whole body gamma dose rate to an individual standing on the ground and immersed in this cloud is a function of the concentration of ^{41}Ar in the cloud and the radius of the cloud. The ^{41}Ar concentration changes with time as a result of radioactive decay and dilution, which results from expansion of the cloud since the radius of the cloud increases as the stack effluent is fed to the cloud. Instantaneous mixing of ^{41}Ar in the cloud has been assumed. This situation has been analyzed in detail in a September 8, 1970 submission to the Commission from LTI entitled "Evaluation of the Environmental Significance of the Projected 5 MW ^{41}Ar Release Rate of 400 $\mu Ci\ sec^{-1}$ from the Lowell Technological Institute Reactor Facility." The release figure of 400 $\mu Ci\ sec^{-1}$ of ^{41}Ar was founded mainly on data from the Rhode Island Nuclear Science Center's swimming pool reactor whose design is similar to ULR's. The 400 $\mu Ci\ sec^{-1}$ was, however, projected for 5 MW operation and assumed continuous air ventilation of the beam ports.

The areas immediately surrounding the reactor are largely automobile parking areas and footpaths for student traffic between buildings. If an individual were to stand adjacent to the containment building, an exposure of 0.114 mrem would be received by the individual to the whole body. This is based on an average release rate over 5 years of 0.692 μCi/sec.

The conservatism of the model used, in terms of atmospheric stability requirements, can be readily seen; for instance, for a 10 meter radius cloud acted on by a wind speed of 0.1 miles per hour the reduction in activity content would be at the rate of about 20% per minute; for a 20 meter cloud (somewhat less than the radius after one hour) the activity reduciton rate would be at about 4% per minute. Integrated one-hour doses under conditions of even this very low wind speed would be decreased by over a factor of ten with respect to the dose figures estimated above.

7.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

7.4.1 Design Objectives for Liquid Radioactive Effluent

The primary coolant loop is essentially a closed system with no continuous liquid effluent release to the environment. Any liquid effluent produced in the reactor (i.e., pump gland leaks, washings from demineralizers, laboratory sink effluent, etc.) is transferred to the appropriate liquid waste hold-up tanks. (See Paragraph 7.2.) The liquid radioactive waste in the waste tank is then analyzed for its radioactive content and the liquid dispersed in accordance with applicable regulations.

7.4.2 Continuous Monitoring

There is no normal continuous release of liquid radioactive effluent to the environment.

7.4.3 Sampling

(1) Pool Water Sample A sample of pool water is collected once each week and analyzed for gross beta activity. (See also Paragraph 7.2.4.) Some of these samples (or composites of samples) are analyzed by gamma spectroscopy. The purpose of this analysis is to observe a buildup in concentration of long-lived radionuclides. The sensitivity of the gross beta analysis is better than 10^{-7} μCi/ml. (See Paragraph 7.6.)

(2) Secondary Cooling Water Sample A sample of secondary cooling water is collected once daily. This sample is analyzed for ^{24}Na to determine the integrity of the heat exchanger. Assuming a detector efficiency of 0.1 counts/disintegration, a sample size of 10 ml and a 10 minute count time with a counting error of 10% the minimum detectable level with no background or other interference is 5×10^{-5} μCi/ml.

(3) Liquid Radioactive Waste Storage Tank Samples. All the liquid radioactive waste produced in the reactor ultimately is transferred to the storage tanks located in the Pinanski Building. (See Paragraph 7.2.) When the tanks are sufficiently full, a representative sample is collected and analyzed. Gross beta analysis is performed and if warranted, gamma spectroscopy to identify certain radioisotopes.

7.4.4 Calibration and Maintenance

The devices used for analysis of samples described above are a low background proportional counter and a GeLi-MCA gamma detection system. This equipment is used almost daily for analysis and is routinely checked with operational standards.

Calibration is performed as necessary using appropriate standards.

7.4.5 Design Objectives for Gaseous Monitoring Systems

The radiological monitoring system includes several elements intended to give an indication of airborne radioactivity levels in the containment building and in the stack gas effluent.

The continuous monitoring system functions under normal operations to provide data on the routine airborne radioactivity and radiation levels. The constant air particulate monitors, located in the reactor containment building, will normally achieve a relatively constant recorded ouput as the detectors respond to filter paper activities weighted largely by ^{222}Rn and ^{220}Rn progeny. Similarly, the response of the air particulate sampling stage of the stack gas monitor, under normal operations, is due largely to naturally occurring airborne activities; the gas sampling stage of the stack monitoring system will respond to external background radiation and ^{41}Ar; for operations at a constant power level the response should be quite constant in magnitude. A G M type gamma sensitive detector located in the main exhaust plenum responds to external radiation levels and during normal operation does not read above nonoperating background levels. All of the above detection systems will respond according to sensitivities indicated in the following section, to indicate increasing (above normal background) levels of airborne radioactivity and associated radiation levels.

The above mentioned detectors and systems, in addition to several other area radiation monitors, operate continuously; the readout meters at the reactor console are equipped with low and high level mechanisms to activate appropriate alarms when the detector responses exceed the preset levels. The low level trip alarm alerts the operator to a higher than normal radiation or radioactivity level at a given site; the high level trips warn of possible radiation emergency situations and these are tied into the alarm logic system described in the Chapter 10 Appendix.

It is felt that the existing systems are capable of providing ample warning of increasing airborne radioactivity levels significantly above normal operating levels and of providing adequate response for possible accident emergency situations. More quantitative aspects of design of the continuous monitoring instrumentation follow in the next section.

7.4.6 Continuous Monitoring

Following is a brief description of those detectos and monitors in the continuous monitoring system which were mentioned in the preceding section and which are primarily intended to provide an indication of airborne activity levels.

(1) Constant Air Monitors (CAM) - Nuclear Measurements Corp. AM-2D

a. Location: One CAM is located on the third floor in the vicinity of the main ventilation exhaust plenum. Since most of the air leaving the containment does so via this plenum and since the plenum is located such as to draw air across the surface of the pool (source of fission product gases), the probability of seeing airborne activity is, for many situations, greater in this location than in others. A second

CAM operates on the experimental floor of the reactor building; location at this level allows a greater opportunity for detection of possible airborne releases resulting from experimental situations. Both CAMs collect activity on a high efficiency glass fiber or other high efficiency filter paper.

b. Concentration and Measurements: Under normal operations with clean fuel (i.e., no significant tramp ^{235}U) the concentrations of airborne fission product activities are nondetectable; in the absence of any other airborne activities from experimental facilities the response of the CAMs will be due to naturally occurring ^{222}Rn and ^{220}Rn daughters collected on the filter paper. A small part of the normal CAM response is due to external radiation levels.

c. Measurements and Detection Considerations: The CAMs utilize end window, G M type, beta-gamma sensitive gas tubes for detection of filter paper collected activities. Gross beta detection efficiency for the system is about 0.10 counts/disintegration; flow rates are adjustable and, for the filters of interest, are set at about 3 cfm. The sensitivity of the system can be representatively estimated for the case of a small fission product release. If it is assumed that some fission product gases escape from the pool to the containment, the solid decay products most likely to be seen quickly are ^{88}Rb and ^{138}Cs daughters of ^{88}Kr and ^{138}Xe, respectively.

The net rate of accumulation of activity on a filter paper may be expressed as $\frac{d\,A_t}{dt}$:

$$\frac{d\,A_t}{dt} = CFR - \lambda A_t \qquad (7.4.6.1)$$

where:

A_t = activity on paper at time, t

t = elapsed time since collection began

C = airborne activity concentration

F = air flow rate through filter

R = filter activity retention (0.95 assumed)

whence:

$$A_t = \frac{CFR}{\lambda}(1 - e^{-\lambda t}) \qquad (7.4.6.2)$$

If sensitivity is estimated for the condition of activity saturation on the paper and if N is the observed count rate for the detection efficiency, E, then

$$A_{sat} = \frac{N}{E} = \frac{CFR}{\lambda} \quad \text{and} \quad C = \frac{\lambda N}{FRE} \qquad (7.4.6.3)$$

The normal background (due to naturally occuring particulates) is about 500 cpm. If one defines this sensitivity for the above situation as the minimum airborne concentration (of ^{88}Rb - ^{138}Cs mixture) required to produce a net count rate response on the CAM equal to the normal background, then using the appropriate numerical values from above:

$$C = \frac{(0.693)(1000\ cpm)}{(20\ min)(3\ ft^3 min^{-1} x 2.83x10^4 cm^3 ft^{-3})(0.95)(0.10\ cd^{-1})(2.22x10^6 dm^{-1} \mu Ci^{-1})}$$

$$C = 1.9 \times 10^{-9}\ \mu Ci/cm^3$$

This number is quite low compared to what would be an appropriate MPC (about 10^{-6} $\mu Ci/cm^3$) for short-lived ^{88}Rb and ^{138}Cs.

The count rate range covered by the CAMs is over three decades from 50 cpm to 50,000 cpm.

d. Power Sources and Readouts: The power supplies for the CAMs are self-contained in the units and are normally fed via house voltage. A natural gas fueled motor generator is available as an alternate power source in case of electrical power outage.

Meter readouts and recorders are available at the CAM units primarily for on-the-spot estimations of airborne levels by experimenters and others. Meter readouts are also present in the reactor control room and in the Health Physics laboratory. CAM responses are also recorded in the control room.

e. Set Points, Annunciators, and Alarms: The CAM meter readouts in the control room are equipped with low and high level selectable trip points. The low level trip points are set at 5000 cpm to provide the operator with a positive indication of rising air concentration. An annunciator light and an audible alarm will be activated in the control room when this lower set point is exceeded; similar annunciators will be activated in the Health Physics laboratory.

The high level trip point set at a value of 30,000 cpm. For a fission product release and resulting airborne activities (as in Paragraph (c) above) with a half life of 20 minutes and present at a concentration of 10^{-5} $\mu Ci/cm^3$ (i.e., ~ 10 x MPC) the high level trip is activated, according to equation (7.4.6.2) in Paragraph c, above within time, t:

$$t = \frac{\ln (1 - \lambda A_t/CFR)}{\lambda}$$

and substituting the proper values:

$$t = -\frac{\ln\left(1-\frac{0.693 \times 30{,}000}{20 \times 0.10 \times 10^{-5} \times 2.22 \times 10^{6} \times 3 \times 2.83 \times 10^{4} \times 0.95}\right)}{0.693/20}$$

$= 0.168$ min $\simeq 10$ sec

The high level trip points are tied into the logic system as described in the reactor Emergency Plan.

(2) <u>Stack Effluent Monitor - (Tracerlab Map-1B/MCG-1A)</u> The stack gas effluent monitor utilizes a moving filter (HV - 70 type) paper tape monitored by beta scintillation (MD-4B) detectors in conjunction with a shielded gas cylinder containing a twelve-inch G M tube for detecting airborne radioactive particulates and radioactive gases, respectively. The filter tape may be moved either continuously or in a stepwise mode and the buildup and decay of activity may be monitored with the two scintillation detectors, respectively.

a. Location: A sampling probe, designed to allow isokinetic sampling of the moving stack gases connects to a two-inch line (about 25 feet in length) which connects to the input of the stack monitor which is located on the fourth floor level (roof level) of the Pinanski Building; exhaust from the monitor is ducted back to the stack at a higher elevation. The location of the sampling probe in the stack is well above the point where exhaust lines enter the stack and is at a location where provisions have been made for doing stack flow measurements via pitot tube traverses so that isokinetic sampling will be assured.

b. Concentrations and Measurements: During normal operations the concentrations of airborne fission products is nondetectable. Assuming no release of airborne particulate contaminants from experimental facilities, the scintillation detectors will respond to naturally occurring ^{222}Rn and ^{220}Rn progeny collected on the filter tape. The G M tube will respond, above normal background, to releases of ^{41}Ar, the only gaseous reactor produced radioactivity expected to be detectable in normal operations. At a stack flow rate of 7.08 m^3 sec^{-1} the maximum long term (annual average) emission rate does not exceed 80 μCi sec^{-1} producing an average concentration in the stack of 1.14×10^{-5} μCi cm^{-3}. The maximum short term (instantaneous) emission rate could exceed this value by a factor of 100.

c. Measurement and Detection Considerations: The stack monitor utilizes Tracerlab MD-4B beta scintillation detectors to monitor filter tape activities. The flow is set at about 8 cfm.
At the manufacturer's rated flow of 10 cfm and with a detector background of 20 cpm, the manufacturer's quoted specifications for detector sensitivity (net

count rate equal to background) are 1×10^{-9} μCi-cm^{-3} for low energy (^{14}C) betas and 5×10^{-12} μCi cm^{-3} for a medium energy (^{204}Tl). For a likely fission product such as ^{88}Rb, mentioned earlier, the manufacturer's instrument sensitivity would be about 10^{-12} μCi cm^{-3}. In reality the filter tapes would be retaining significant ^{222}Rn and ^{220}Rn progeny activities which would decrease these sensitivity figures; thus, for a detector count rate of 1000 cpm, which would be realistic, and for the expected flow rate of 10 cfm, the ^{88}Rb sensitivity would be reduced to about 10^{-10} μCi cm^{-3} which is still favorable for easy detection of significant releases.

When the tape transport system is in operation, sensitivity for certain radionuclides may be changed somewhat but this is not detrimental; in actuality, with stepwise motion and proper time sequencing real increases in sensitivity can be obtained for certain radioactive species.

The range of detector response, as activity is collected, is indicated by a 4-decade log meter from 20 cpm to 200,000 cpm.

The radioactive gas detector is the Tracerlab MD-12B. Manufacturer's specifications for a background count rate of 40 cpm indicates a sensitivity for ^{41}Ar of

$5.5 \times 10^{-7}\ \mu Ci\ cm^{-3}$.

The range of the G M detector is indicated on a 4-decade long meter from 20 cpm to 200,000 cpm.

d. Power Sources and Readouts: The power supply for the stack air monitor is contained in the unit. Normal house electrical power is backed up by a natural gas fueled motor-generator to be activated in case of loss of power. Meter readouts and a recorder are available at the stack monitor unit; meters and a recorder at the reactor control console also respond to signals from the stack monitor detector, and meter readouts are also available in the Health Physics laboratory.

e. Set Points, Annunciators, and Alarms: The stack monitor meter readouts in the control room are equipped with low and high level selectable trip switches. These will be set to operate in much the same fashion as for the CAMs described above. The radioactive gas detector meter readout low level trip is set at 1000 cpm; the high level trip is set close to 15,000 cpm, this representing the approximate expected resonse for an ^{41}Ar stack concentration close to twenty times the maximum long term stack emission concentration.

Visual and audible annunciating devices are provided for the detector readouts just as described for the

CAMs above. Similarly, the high level trips are tied into the alarm system as described in the ULR reactor Emergency Plan.

(3) External Radiation Detector in Main Exhaust Plenum The detector used in the plenum is a gamma sensitive, lead filtered, halogen-quenched, G M tube, Tracerlab Model TA-61. It responds to all significant increases in external radiation levels and has been located with the intent of providing a response to gamma activity, in addition to general increases in external radiation levels. Concentration of inert gases, such as ^{41}Ar or some fission product gases, present at the occupational MPC would produce exposure rates of 2.5 mR/h at this detector under semi-infinite cloud submersion dose assumptions; in most real cases this detector sees clouds much smaller than "infinite" sizes and detector response at MPC levels may very likely not be detectable above normal backgound; thus, relatively small increases in exposure rate at this detector, if due only to airborne radioactivity, are indicative of relatively high airborne concentration.

The plenum G M detector has a meter readout with range from 0.01 mR/h to 100 mR/h and is equipped with low and high selectable trip points which activate annunciating devices as described above for the CAMs and stack monitor and with high level trips tied into the alarm system as in the reactor Emergency Plan. The set points for these readouts are 2 mr/hr at the low trip and 5 mr/hr at the high trip. The detector is powered by house electrical power via the Tracerlab RMI-110, Model TA-10 power supply located in the reactor control room and alternately by the natural gas fueled emergency generator. Detector response is metered and recorded in the reactor control room and metered in the Health Physics laboratory.

7.4.7 Sampling

Filters in the CAM systems are replaced at intervals of every other day starting on Monday. Filters removed from the CAMs are routinely subjected to gross beta and gamma spectral analysis for possible identifications and quantification of non-naturally occurring activities; sections of filter tape from the stack monitor are also subjected to some routine analysis for similar activities. Further gamma spectral analysis are performed on monthly composites of weekly air particulate samples. Specific isotopic analysis for certain long-lived species, particularly ^{90}Sr, are performed on monthly composited samples on a quarterly basis if gross beta analysis indicated the likely presence of long-lived activities in excess of acceptable limits.

Following any changes in fuel loading, experimental operations, or any other factors which might alter the makeup of the radioactive effluent gases, air sampling techniques and methods will be used as required to characterize the waste gases.

Where possible, gamma spectral analysis using both sodium iodide and germanium detectors are used for isotopic identification and quantification. Beta counts, half life determinations, and beta energy estimations in conjunction with radiochemical separations may also be used for isotope estimation.

For routine operations, with the exception of ^{41}Ar, all radioactive gaseous releases are at concentrations orders of magnitude below appropriate MPC's and most cannot be detected even with sophisticated analytical techniques.

7.4.8 Calibration and Maintenance

All area radiation monitors including the CAMs and the stack monitor are tested daily for proper operability and all monitors are calibrated at intervals not exceeding six months.

The CAMs and particulate detection stages of the stack monitor are checked with appropriate beta-gamma sources during calibration to establish the response of the detectors. Appropriate gamma sources are used to ascertain the G M area monitor (in plenum and stack) responses.

External gamma sources are used for (noncalibration) operational checks on the area monitors.

7.5 SOLID WASTE SYSTEM

7.5.1-2-3-4 Design Objectives, Inputs, Equipment and Volumes

The amounts of solid wastes generated as a result of reactor operations are relatively small.

Operation of the cleanup demineralizer system results in the eventual need to replenish the ion exchange resins. Spent resins are drained of liquid, bagged in plastic bags, and loaded into D.O.T. approved metal drums for pickup by a waste disposal contractor. If short lived activities (e.g., ^{24}Na) are present in significant amounts these will be allowed to decay prior to shipment. A single charge from the ion exchange demineralizer represents about thirteen cubic feet of resin which will nearly fill two fifty-five gallon waste shipping drums. The demineralizer resin may be replaced once or twice per year, ^{51}Cr represents the largest fraction of activity in drums for shipment. Activities are less than one millicurie per drum of spent resin.

Normal operation of the gaseous waste handling system results in the generation of some solid waste in the form of filters (absolute and/or charcoal). On a routine basis, activities on the filters do not exceed microcurie levels. Activity contaminated filters will be bagged and, after decay of any significant short-lived components, transferred to appropriate metal drums for waste pickup by an authorized contractor. Not more than about six filters require replacement on an annual basis, the filters represent a bulk volume of about 110-165 gallons annually (i.e., two to three 55-gallon drums).

Additional solid wastes will be produced as a result of experimental and maintenance operations. Such items as disposable clothing, sample transfer rabbits, contaminated paper and laboratory items, miscellaneous hardware, and certain cleanup and "housekeeping" items are disposed of routinely by a disposal contractor. Waste of this sort accounts for approximately 220 gallons annually in volume (i.e., four 55-gallon drums). Average activity per drum is less than one millicurie, most of which will likely be represented by metal ion activation products.

7.5.5-6-7 <u>Packaging, Storage, Shipment</u>

All of the above solid wastes are packaged, as stated above, in D.O.T. approved metal drums for pickup by an authorized waste disposal contractor. Wastes being held for decay and/or pickup will normally be stored in the liquid waste tank basement area of the Pinanski Building (shown in Figure 7.1) and/or in the subbasement area of the reactor building. Both of these locations are remote from frequent personnel operations and are subject to controlled access. In the case of storage of any high level solid wastes appropriate shielding will be provided as required. Waste pickups will be made by truck provided by the waste disposal contractor.

Spent fuel elements will be stored in the wall rack locations provided in the reactor pool. Pick up of such elements for shipment and fuel reprocessing will be carried out by an authorized contractor (e.g., NL Industries, Inc.); the contractor will perform any operations necessary for getting the fuel elements into appropriate shipping casks. All such operations are controlled by the UL staff. To date, no such spent fuel movement has been performed.

7.6 OFFSITE RADIOLOGICAL MONITORING PROGRAM

The operational environmental surveillance program at UL consists primarily of measuring long-lived gross beta activity in air and water and direct measurement of external gamma doses. The type and frequency of sampling is described below, where the specific values give nominal sampling frequency and volumes, but should be interpreted as typical rather than absolute.

7.6.1 Expected Background

External dose measurements by the use of film badges over a three year period of time showed less than minimum detectable exposures (20 mR per month). Measurements of the external gamma dose by the use of thermoluminescent dosimeters began in 1972 and the range of such measurements is from 10-20 mR/4 weeks.

7.6.2 Critical Pathways

Based upon our release rates, as described in Paragraphs 7.2 and 7.3, the principle mode of exposure is from Argon-41.

7.6.3 Sampling Media, Locations and Frequency

During the period of preoperational environmental surveillance external gamma dose measurements and samples of water, soil, vegetation and air were collected at various locations. Based upon the experiences to date it was decided to collect weekly samples of airborne dust and 2-week composite samples of water from the Merrimack River and external dose measurements for the operational program.

Film badge dosimeters are placed at selected site locations around the UL campus. These dosimeters will be collected at nominal four-week intervals for measurement of the integral external gamma dose.

The raw water samples are collected at the Lawrence Municiple water treatment plant (11 miles downstream from the UL campus) and at the Lowell Municiple water treatment plant (4 miles upstream from the UL campus). A 200 ml aliquot of untreated water is sampled each working day and transferred to a large polyethylene container. This composite sample is collected every two weeks for analysis.

Airborne dust is sampled continuously by pumping air at a rate of 0.5 to 1.0 cfm through a 2-inch filter. These filters are brought to the laboratory weekly for analysis. The location of the air sampling site is on the roof of Ball Hall. (See Figure 2.3 in Chapter 2.)

7.6.4 Analytical Sensitivity

Typically, a gas flow proportional counter with an efficiency of 0.39 counts/beta and a background of 5.5 cpm is used to determine the gross beta radioactive content of the airborne dust and water samples. The individual errors (based upon counting errors) range from approximately 3 to 20%. Setting the error at some select value ($\pm$ 10%) the minimum detectable activity (MDA) and associated concentration level (sensitivity) can be calculated.

$$E = \frac{S_{n_x}}{n_x} = \frac{1}{n_x} \left[\frac{n_s}{t_s} + \frac{n_b}{t_b} \right]^{1/2} \qquad (7.6.4.1)$$

where:

E = fractional error (set at 0.10 to define MDA)

S_{n_x} = standard deviation of net count rate

n_x = net count rate = $n_s - n_b$

n_s = gross count rate

n_b = background count rate

n_s = sample counting time

n_b = background counting time

Solving for the MDA:

$$n_x = \frac{1}{2\ E^2\ t_s} \left[1 + \left(1 + \frac{4\ E^2 n_b t_s}{t_b} (t_s + t_b) \right)^{1/2} \right] \qquad (7.6.4.2)$$

Setting $t_s = t_b = 50$ minutes with $E = 0.10$

$$n_x = 1 + (1 + 4\ n_b)^{\frac{1}{2}}$$

with n_b = 5.5 cpm, then n_x (min) = 5.8 cpm

$$\text{MDA} = \frac{n_x \text{ (min)}}{\text{efficiency}} = \frac{5.8 \text{ cpm}}{0.39 \frac{c}{\beta}\left(2.22\text{x}10^6 \frac{\beta\text{pm}}{\mu\text{Ci}}\right)} = 6.7 \times 10^{-6}\ \mu\text{Ci}$$

7.6.5 Data Analysis and Presentation

The gross beta count is converted to activity concentration units (microcuries per cubic centimeter) and the data tabulated for internal use. Table 7.1 describes the types of samples and the collection frequency.

7.6.6 Program Statistical Sensitivity

The objective of the environmental surveillance program is to analyze accurately the current levels of gross beta concentration in air and water and to measure the external gamma dose directly. Comparison and observation of the data is primarily to determine if there is an unusual difference between similar samples collected over the similar time periods from the different sites. An arbitrary decision is then made to determine if a particular sample(s) needs further investigation and analysis to determine the concentration and origin of certain radionuclides.

TABLE 7.1

ANALYTICAL SENSITIVITY FOR ENVIRONMENTAL SAMPLES

Sample Media	Location	Sample Size	Minimum Detectable Concentration (μCi/cm)
Airborne dust	2 sites on campus	$1.3\text{x}10^8$ cm^3	$5.2\text{x}10^{-14}$
Water	(2) Lowell & Lawrence	2000 cm^3	$3.4\text{x}10^{-9}$
External gamma dose	Various site locations on campus	--	

7.7 REFERENCES

(1) Letter dated Jan. 3, 1972 from Rhode Island Nuclear Science Center to USAEC, Assistant Director for Reactor Operations.

(2) Surface Water Supply of the United States 1961-65, Part 1, North Atlantic Slope Basins, Vol. 1, Basins from Maine to Connecticut, U.S. Department of Interior, p. 320.

(3) Skrable, K. W., "Identification, Source, and Levels of Airborne Fission Product Radioactivities Associated with the Operation of a 5 MW Research Reactor," Master of Science Thesis, Vanderbilt University (May 1964).

8.0 RADIATION PROTECTION

8.1 SHIELDING

8.1.1-2 Design Objectives and Descriptions

The design of the swimming pool reactor and its containment building are such as to provide more than sufficient shielding to abide by the requirements of 10 CFR Part 20. Bulk shielding against core radiations is provided by the pool water and the normal and high density concrete shield around the pool; the structural details of the pool and biological shield are presented in Paragraph 4.1.10, and Figure 4.8. Additional shielding between the building and the external environs is provided by the two-foot thick concrete shadow shield which lines the containment building walls. This is as described in Paragraph 3.1.1.

Actual radiation levels above the pool depend largely on the purity of the water. Radiation surveys conducted at 1 megawatt indicate a normal level less than 10 mR/hr to be expected above the pool under the reactor bridge.

Following the loss-of-flow scram, it is calculated that the radiation level at the pool surface would be approximately 8 mrem an hour. This is based on the assumption that a significant proportion of the cooling water rises from the reactor to the pool surface without mixing. The 15-hour Na-24 in the cooling water is the primary contributor to the dose rate.

Normal external dose rates in the containment building in most nonexperimental areas of the first floor, second floor, and third floor are less than 1 mrem/h. Dose rates in certain locations near ongoing experiments (e.g., certain beam tube experiments) may be greater than this at times; however, in such cases local shieldings (e.g., cement blocks, lead bricks, etc.) are utilized as required to reduce dose rate to acceptable levels. Gamma radiation levels in the holdup tank area of the pump room in the containment building basement are normally high (several R/h) during forced flow at 1MW. Access to the pump room is restricted and appropriate supervision is required for all personnel operations in this area.

While dose rates may vary considerably at specific locations, depending on the types of experiments being conducted, individual workers, in accordance with ULR policies and 10 CFR 20 regulations, are not allowed to accumulate more than currently established values.

Radiation doses to visitors and other nonstaff personnel who are not directly involved with the operation and/or use of the reactor are carefully controlled. Access to high radiation areas (as defined in 10 CFR 20) is not normally allowed, and personnel monitoring is provided as required in 10 CFR 20. For the great majority of people in the above category integrated exposures for a normal visit average much less than 10 mrem; in all cases normal policy is to limit integrated quarterly doses in accordance with values in 10 CFR 20. All restricted areas are posted and controlled in accordance with 10 CFR 20.

For normal operations, dose rates outside the containment building were originally expected to be very close to normal external background levels. Environmental radiation surveys conducted around the perimeter of the containment building during

1MW operation of the ULR have confirmed this expectation. Normal levels at 1MW are less than 0.05 mrem/h.

The liquid and gaseous waste handling systems are presented in Sections 7.2 and 7.3. In these systems, the primary coolant loop and its included components are the only potential sources of external radiations. Local shielding is utilized as required for personnel operations involving the primary coolant system. Exposure to external radiation from ^{41}Ar released from the stack has been discussed in Paragraph 7.3.8.

8.1.3 Source Terms

Nitrogen-16 is the greatest source of external gamma radiation in the primary coolant system. For sustained operations ^{16}N levels in the area of the holdup tanks have been discussed in Paragraph 8.1.3.1. Local concrete block shielding is provided to lower pump room radiation levels in the holdup tank area, to acceptable levels for personnel entry and operations. The cleanup demineralizer as discussed in Paragraphs 7.2 and 7.5, collects activities from the primary coolant; however, the external radiation levels from these activities while the reactor is in operation are smal compared to levels from the ^{16}N. After shutdown and allowance for decay of ^{24}Na, the resin held activity has been found not to be an external radiation hazard of significance. Spent cleanup system resins during resin changes have been found to have exposure rates less than 1 mr/h on contact.

8.1.3.1 Unshielded Exposure Rates at the Surface and at 1 Meter from the Holdup Tank Area

Nitrogen-16 is produced via the threshold reaction O-16(n,p)N-16, by the interactions of neutrons with energy in excess of 9.6 MeV with O-16 in the core coolant water.

The holdup tank is approximately a cylindrical volume of 260 cm diameter and 180 cm high. Surveys performed in the pump room during steady 1MW operation have concluded that the unshielded exposure rates in the holdup tank area are <2R/hr in the vicinity of the core exit lines, <400 mR/hr on contact with the surface of the holdup tank.

8.1.4 Area Monitoring

The continuous monitoring system includes several detectors and readouts capable of responding to and indicating external radiation levels. The monitoring system is as presented in the appendix to Chapter 10. Those monitors which function to respond to gaseous and liquid waste activities have been described in Paragraphs 7.2, 7.3, and 7.4.

Of the remaining external radiation detectors located at several points throughout the building (approximate locations given in Emergency Plan), nine are GM tubes (Tracerlab TA-61) with a response range of 0.01-mR/h to 100 mR/h; three are GM tubes (Tracerlab TA-62) with a response range of 1 mR/h to 10 R/h; and two are ion chambers (Tracerlab TA-64) with a response range of 0.1 mR/h to 10R/h. All of these detector outputs are metered at the reactor console and the meters are equipped with high and low level selectable trip points. Low level trips are set at a small fraction of full scale to warn the operators of rising radiation levels; high level points, dependent somewhat on operational conditions, are set at about 100 mR/h for many of the area monitors. Tripping at either level will result in activating an annunciating light in the reactor radiation monitoring cabinet. Certain detectors and readouts are tied in with the radiation emergency alarm systems as described in the Emergency Plan. Meter readouts and annunciating alarms for the above detectors (with the exceptions of the TA-62s) are also present in the Reactor Supervisor's office. Recording of the detector output is effected in the control room.

Power for these continuous monitors is provided via a Tracerlab TA-10 power supply in the reactor control room and normally fed by house voltage. A backup natural gas fueled motor generator supplies emergency power in the case of power outage as described in Chapter 6.

8.1.5 Operating Procedures

Under normal circumstances the potential for significant external personnel exposures is greatest for certain experimental setups and for certain maintenance and repairs, particularly in the pump room area.

All previously unevaluated experiments are reviewed by the Reactor Safety Subcommittee to judge their overall implications from a safety point of view before they are approved. The Health Physics Group provides the necessary support to assist in establishing and maintaining radiation safety in a particular experiment. Area surveys are made to evaluate radiation levels, and appropriate signs and barriers are erected to inform personnel of potential levels and to restrict access to certain areas.

Access to the pump room is limited. Keys to the pump room area are issued only to selected personnel. Entrance to the pump room while the reactor is in operation requires the knowledge and consent of the operator in charge. Operations conducted in this area are carried out only with appropriate health physics support. Maintenance and repair work is normally scheduled for reactor shutdown periods, and appropriate health physics support is required. Direct reading dosimeters are required of personnel for entrance into the pump room area.

Doses to visitors and other nonworkers is established by the mandatory use of appropriate personnel dosimeters, particularly pocket dosimeters and film badges.

8.1.6 Estimates of Exposure

Doses to personnel directly involved with the operation and use of the reactor are maintained below the quarterly values specified in 10 CFR 20.101. The actual quarterly doses from external sources are appreciably below these 10 CFR 20 values. Even though at locations in the pump room and around particular experimental setups external radiation levels may be several rem per hour, integrated exposures in these fields are kept small.

Direct external radiations resulting from reactor operations are expected to contribute virtually nothing to the man-rem dose to the population outside the reactor building.

8.2 VENTILATION

The design and description of the ULR containment ventilation system is given in Chapter 3 of this USAR. The source terms, airborne radioactivity monitoring, and operating procedures are considered and discussed in Paragraphs 8.1 and 8.3. Inhalation doses resulting from airborne radioactivity in the containment building are insignificant. The only airborne constituent of any note which is likely to be present in the building is ^{41}Ar produced in experimental facilities and discussed in Paragraphs 7.1 and 7.3. ^{41}Ar, an inert gas, does not represent an inhalation dose problem. Containment building ^{41}Ar concentrations are not great enough to result in an external dose problem of any consequence.

8.3 HEALTH PHYSICS PROGRAM

8.3.1 Program Objectives

The present health physics staff consists of two health physicists, one of whom is the Radiation Safety Officer, and one health physic technician. In general, the Health Physics Group functions to ensure that the radiation safety program, at a minimum, is consistent with applicable Federal and State regulations. More specifically, the group is responsible for carrying out policies approved and promulgated by the Radiation Safety Committee at the University as indicated in Paragraph 10.5. Health Physics service operations are supported by budget allocations to the program through the Radiation Laboratory operating budget. More descriptive detail as to administrative organization is provided in Chapter 10.

Services provided by the Health Physics Group include personnel monitoring, radiation monitoring, radiation instrument calibration, waste pick up and disposal, consultant services, transportation and shipping assistance, emergency assistance, and radiation safety training.

The radiation safety officer serves as a permanent member of the Reactor Safety Subcommittee and situations likely to require health physics support can be recognized and evaluated in advance.

8.3.2 Facilities and Equipment

8.3.2.1 Instrumentation The Health Physics Group maintains a large number of portable instruments for detection and measurement of radiation. Among these are the following types and models:

GM type (Eberline E-120, 120 G, Teletector)

Ionization Chamber (Eberline PIC-6, Victoreen 444, Victoreen Condenser-R Chambers, NUCOR CS-40A, Jordan)

Neutron Detectors (Tracerlab "Snoopy" NP-2, Victoreen 448A)

Scintillation Detectors (Eberline Gadora-1, Ludlum NaI(Tl), Eberline-ZnS alpha probe)

The above instruments represent reliable gamma radiation measuring capability over the range from natural background levels to >1000 R/h. The portable neutron instruments can accurately provide direct neutron dose measurements over the range from 0.1 mrem/h to 2 rem/h.

All portable instruments in use are calibrated at six month intervals.

In addition, as noted in the reactor Emergency Plan, continuous monitors in the reactor building have readouts in the health physics laboratory. Some of these monitors are tied in with the emergency alarm system.

Other laboratory type and specialized radiation detection and measuring instrumentation is also operated and maintained by the Health Physics Group. Included in this group are hand and foot monitors, a TLD reader system, proportional gas counting systems, a 512 channel gamma analyzer with a NaI(Tl) detector (3-inch solid crystal) and flow-through gas ionization chambers and associated electronics. Additional instrumentation, including solid state detectors, multiparameter analyzing systems, gas counting systems, and liquid scintillation systems are available through the Radiation Laboratory staff counting room for health physics use.

Laboratory instruments used daily are subject to daily calibration checks. Instruments not used routinely are calibrated at the time of use or at intervals consistent with overall instrument reliability. For the gas flow proportional counting system, which is expected to be used for much routine counting, the minimum detectable activity (net count rate of two times the standard deviation in the background count rate) for most expected beta emitters is approximately 2×10^{-5} microcuries. Similarly, the minimum detectable activity for the liquid scintillation system presently available is about 10^{-4} microcuries. Available flow-through gas ionization chambers have sensitivities of about 10^{-6} Ci cm^{-3} for likely gases (e.g., ^{41}Ar) of interest in air.

8.3.2.2 <u>Sampling Equipment</u> Several pumping systems and grab sampling (vacuum bottles, charcoal cartridges, etc.) devices are on hand for purposes of obtaining air particulate, gaseous, and liquid samples which may be required. Two-stage respirable fraction particulate air samplers and particle sizing devices are available for airborne particulate activity characterization. Several lead-lined (one-inch and two-inch thick walls) storage containers are available for handling and transporting gamma emitting sources. Remote handling tools are also available for handling sources at a distance. Lead bricks and cement blocks are available for providing local shielding when required.

8.3.2.3 Protective Clothing and Respiratory Equipment

Disposable and nondisposable laboratory coats and overalls, of paper and plastic impregnated fibers are available. Foot wear of disposable plastic and nondisposable synthetic rubber is available as booties, rubbers, and boots. Gloves of plastic, cloth, and rubber are also on hand. Soft and hard protective headwear is also available. Such clothing will be provided as required for all work in contaminated or contaminating environments and will be readily available as necessary for routine operation. Six full face gas masks equipped with canisters designed for efficient collection of radioactive particulates and iodine are maintained for use in areas of airborne contamination. Two one-half hour, self-contained compressed air units are available for use in situations of high airborne radioactivity levels.

8.3.2.4 Other Facilities and Equipment. Various signs, placards, labels , barrier devices, decontamination supplies, and miscellaneous hardware are on hand for routine and emergency use.

The room directly opposite the health physics laboratory contains shower facilities and will serve as a decontamination area in the event of personnel contamination.

Routine first aid supplies and equipment are also stocked and maintained by the health physics group.

8.3.3 Personnel Dosimetry

External exposures are monitored on a monthly basis via the use of beta-gamma and neutron sensitive film. (Landauer film badge service.) Pocket dosimeters are on hand for making on-the-spot estimation of gamma exposures. For operations in high radiation areas, the use of such dosimeters will be mandatory. A TLD system is also available to the health physics operations and some use of thermoluminescent dosimeters may be made for personnel monitoring. Operation in high radiation areas will also normally require the use of appropriate portable survey instruments for proper evaluation of radiation levels.

Film badges are normally processed on a monthly basis. With the proper control gamma exposure of <5 mR can be accurately evaluated with the use of TLD devices. Pocket dosimeters, whose leakage histories are established, with a range of 0-200 mR can yield reliable integrated exposures as low as about 5 mR over a period of a few days or less.

Internal dosimetry is not conducted on a routine basis. Analysis of biological excreta, in conjunction with whole body counting techniques where desirable, will be carried out in cases where internal contamination is suspected.

Exposure records will be maintained in accordance with approved NRC formats.

9.0 SAFETY ANALYSIS

9.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

9.1.1 Safety Limits

9.1.1.1 Basis for Technical Specifications The Technical Specifications for the ULR list the limiting reactor power, coolant flow rate, poolwater temperature and pool height. Details of the analysis leading to the specifications are given in this Chapter.

9.1.1.2 Thermal Analysis of the ULR In order to establish the safety limits of the reactor, an analysis of thermal behavior under varying flow conditions was done. In the forced flow mode, the onset of nucleate boiling was used to establish these limits.

Essential to the safe operation of the ULR is the preservation of stable, single phase (i.e., liquid) flow of primary coolant water through each of the core parallel flow channels, and this is assured once it is established that net steam formation in any channel is an unlikely event. Consequently, adverse combinations of fluid and thermal mechanisms leading to hydraulic resistance which could reduce channel flow are excluded by specifying the onset of nucleate boiling as a limiting condition. An analytic procedure was developed to predict nucleate boiling, and subsequently the hot channel factor approach was used to arrive at reasonable Safety Limits for the ULR.

The analysis was done for a core containing 26 elements, the actual core size. It was assumed that coolant flow through control blade sheaths was unrestricted by control blades, which is the conservative assumption since restricted flow here would force a higher percentage flow through the fuel channels and thus allow higher powers.

A limit of 110°F was placed on the pool water temperature and was used throughout the calculations. This limit is based on the fact that reactor operations under natural convective cooling at 100 kW for 1000 kWh results in 5-6°F temperature rise of the pool water, and since the pool water after extended operation is normally between 80 and 90°F, the added 10 degrees results in an upper limit of 110°F under worst conditions. The same limit was chosen to apply to forced convection cooling also.

The water height was stipulated to be 24 feet above core center line. A reference primary coolant flow rate of 1400 gpm was used throughout the calculations. This is a conservative value since the actual operating coolant flow rate is 1600 gpm; it is, however, the flow rate used by the reactor manufacturer.

The thermal analysis consisted of calculations of the nucleate boiling heat fluxes as a function of coolant flow rate. The equation of Rohsenow and Bergles[1] was used to predict the onset of nucleate boiling.

The wall temperatures were calculated as a function of axial distance assuming a cosine power distribution. The calculation was done for the hot channel of the ULR; this is adjacent to the water gap formed by removal of the control blades. A flux peaking factor for the hot channel was calculated using 2-D multigroup diffusion theory calculations by Hackney for the ULR cores.[2]

The axial heat flux in the hot channel was assumed to be given by

$$q''(Z) = q''_c \cos \frac{\pi Z}{H_e} \qquad (9.1.1.1)$$

where:

q''_c = heat flux at midpoint of hot channel

Z = axial distance along hot channel; origin at core center, ft

H_e = effective height, i.e., height of fuel bearing region plus twice the reflector savings calculated by Hackney (2.57 ft)

For the ULR the heat flux at the center of the hot channel is

$$q''_c = 1.82 \times 10^4 \text{ (P) Btu/h ft}^2 \qquad (9.1.1.2)$$

where:

P = reactor power in MW.

The constant in Equation 9.1.1.2 contains peaking factor equal to 2.00 which was calculated from the neutron flux distributions. It is the ratio of peak flux in the hot channel to the core average flux, excluding hot spot factors discussed in Paragraph 9.1.1.6.

The following sections discuss the empirical correlations and methods used to calculate the thermal performance of the ULR.

9.1.1.3 Forced Convection Heat Transfer Coefficient Local heat transfer coefficients were predicted by the Hausen equation[3]

$$h(Z_t) = 0.116\left(\frac{k}{D_e}\right)\left[Re^{2/3} - 125\right]Pr^{1/3}\left[1+1/3\left(\frac{D_e}{Z_t}\right)^{2/3}\right]\left(\frac{\mu_b}{\mu_w}\right)^{0.14} \qquad (9.1.1.3)$$

where:

$h(Z_t)$ = local heat transfer coefficient, Btu/h ft^2 ^oF

k = Thermal conductivity of fluid, Btu/h ft ^oF

D_e = equivalent diameter, ft

Re = Reynolds number

Pr = Prandtl number

Z_t = axial distance along the channel; origin at core top, ft

μ_b = viscosity at bulk fluid temperature, lb/sec ft

μ_w = viscosity at wall temperature, lb/sec ft

The Hausen equation has been verified by Geidt[4] and is applicable in the transition region of Re = 2300 - 6000 as well as for higher values of Re. In the analysis, total core flow was varied from 400 gal/min to 2200 gal/min with corresponding Reynolds numbers 2747 to 16,760. The equation is also applicable in regions along the channel where entrance effects exist and is applicable for channels having length-to-diameter ratios of 40 to 180. The ULR fuel elements have ℓ/d = 120.

9.1.1.4 Peak Clad Temperature In order to predict the onset of nucleate boiling, the wall temperature at the hot spot in the hot channel must be calculated. Assuming symmetric heat generation about the center of the hot channel, the location of the peak clad temperature is displaced from core center. The analysis of the location and value of the peak clad temperature was done according to the methods outlined by El Wakil.[5] The wall temperature is given by

$$t_w = t_{f_1} + q_c''' A_c \left[\frac{H_e}{\pi \dot{m} c_p} \left(\sin \frac{\pi Z}{H_e} + \sin \frac{\pi H}{2H_e} \right) + \frac{1}{h(Z)C} \cos \frac{\pi Z}{H_e} \right] \quad (9.1.1.4)$$

where:

t_w = wall temperature at Z, °F

t_{f_1} = bulk fluid inlet temperture, °F

q_c''' = volumetric heat source at center of fuel plate in the hot channel, Btu/h ft^3

A_c = cross sectional area of fuel plate, ft^2

H_e = extrapolated height of core, ft

c_p = specific heat of fluid, Btu/lb °F

$\dot{m}$ = coolant mass flow rate, lb/h

$h(Z)$ = heat transfer coefficient, Btu/h ft^2 °F

C = circumferential length of the clad fuel plate, ft

H = height of core, ft

From Equation 9.1.1.4 it is seen that wall temperature is a function of position. Since the pressure does not change appreciably over the channel, boiling will begin at the position in the channel where the wall temperature is the highest, which, in the forced convection mode, will be displaced beyond core center in the flow direction. This position is given by

$$Z_{max} = \frac{H_e}{\pi} \tan^{-1} \left(\frac{hCH_e}{\pi c_p \dot{m}} \right) \quad (9.1.1.5)$$

and the bulk fluid temperature, t_f, at Z_{max} is then

$$t_f = t_{f_1} + \frac{q_c''' A_c H_e}{\pi c_p \dot{m}} \left(\sin \frac{\pi Z_{max}}{H_e} + \sin \frac{\pi H}{2H_e} \right) \quad (9.1.1.6)$$

In the evaluation, the inlet bulk fluid temperature was taken to be 110°F, as mentioned previously, and the coolant properties were evaluated at the temperature given by Equation 9.1.1.6. Because of the interdependence of the heat transfer coefficient, the position of highest wall temperature, the bulk fluid temperature, and the wall temperture, an iterative procedure was used to arrive at a wall temperature for specified flow conditions, and the wall temperature, in turn, was used to predict nucleate boiling heat fluxes as discussed in Paragraph 9.1.1.5.

9.1.1.5 Prediction of Nucleate Boiling The Bergles and Rohsenow[1] correlation was used to predict the onset of nucleate boiling. Their correlation depends on pressure and wall temperature and is applicable over the pressure range of 15 to 2000 psia. The pressure at the core center of the ULR under 24 ft of water is 25.09 psia.

The Bergles and Rohsenow correlation is

$$q''_{NB} = 15.6\ p^{1.156}\ (t_w - t_{sat})^{2.3/p^{0.0234}} \quad (9.1.1.7)$$

where:

q''_{NB} = nucleate boiling heat flux, Btu/h-ft^2

p = pressure, 25.09 lb/ft^2 absolute

t_w = wall temperature, °F

t_{sat} = saturation temperature at pressure p, 240.1 °F

Because of the absence of large cavities on fuel element surfaces which initiate and support nucleation, Equation 9.1.1.7 predicts nucleate boiling at lower wall temperatures than actual.[1] Consequently, the analysis is conservative.

Table 9.1 lists q''_{NB} and resultant reactor power as well as Z_{max}, t_f, and t_w, for various coolant flow velocities.

The wall temperature, t_w, listed in Table 9.1 is that of the plate location at which nucleate boiling would initiate, and is thus the result of the stipulation of the onset of nucleate boiling criterion at the different coolant velocities. It is important to note that the wall temperature varies from 246°F to 254°F under this criterion, which means that the onset of nucleate boiling criterion in the forced convection case leads to plate temperatures not largely different from the criterion of a maximum plate temperature of 250°F used for natural convection case in Paragraph 9.1.1.7 below.

9.1.1.6 Allowance for Error Under ideal conditions, no corrections to the predicted heat fluxes would be necessary. However, due to uncertainties in fuel distributions, variations in channel dimensions, flow maldistribution, etc., hot channel factors were used to account for the effects of these uncertainties on the predicted heat fluxes.

The hot channel factors used were:

f_1 = 1.2, Increased in temperature of primary coolant due to dimensional tolerances in coolant flow passage.

f_2 = 1.04, Change in single phase (i.e., liquid) heat transfer coefficient due to dimensional tolerances in coolant flow passage.

$f_3 = 1.03$, Inhomogenities in fuel distribution with a clad fuel plate.

$f_4 = 1.02$, Variation in fuel mass per element.

$f_5 = 1.12$, Coolant bypassing active part of core.

$f_6 = 2.00$, Flux peaking factor - already included in the analysis.

The product method is used to form the total hot channel factor. Thus, $F_1 = f_1 \times f_2 \ldots\ldots f_5 = 1.47$ and $F = F_1 \times 2.00 = 2.94$ the overall hot channel factor.

It should be noted that the hot channel factor computed by the product method implies that all the "hot spots" occur simultaneously in space and time at one point in the core. Hence, heat fluxes and resultant powers corrected by the product hot channel factor are clearly conservative. Table 9.1 includes corrected powers for the ULR as a function of primary flow.

9.1.1.7 <u>Limiting Power During Operation Under Conditions of Natural Convention</u> The upper limit in power for operation with natural convective flow is stipulated to be the power which results in clad temperatures of 250°F. The limit of 250°F is conservative and well below the point of 1200°F where clad damage would begin. Further conservatism is built into the analysis by neglecting chimney effects and also by using lower bound heat transfer coefficients.

The analysis is done by a method described by Glasstone and Sesonske[6] where the total amount of heat removed under conditions of natural convective cooling is given by

TABLE 9.1

RESULTS OF LTIR THERMAL ANALYSIS FOR FORCED FLOW OPERATION

Primary Flow (gpm)	Z_{max} (ft from center)	t_f (°F)	t_w (°F)	q''_{NB} (Btu/h ft^2 x 10^{-4})	P (uncorrected for hot channel factors) (MW)	P (corrected for hot channel factors) (MW)
400	0.278	153.9	245.7	2.60	1.52	1.03
500	0.338	163.4	246.7	3.66	2.20	1.50
600	0.362	167.4	247.5	4.58	2.78	1.89
700	0.374	169.4	248.1	5.43	3.33	2.26
800	0.379	170.5	248.6	6.29	3.86	2.62
900	0.381	171.0	249.2	7.11	4.37	2.97
1000	0.381	171.3	249.7	7.97	4.88	3.32
1100	0.380	171.3	250.1	8.75	5.37	3.65
1200	0.378	171.1	250.4	9.54	5.86	3.99
1300	0.376	171.0	250.9	10.34	6.34	4.31
1400	0.374	170.7	251.3	11.13	6.81	4.63
1500	0.371	170.5	251.6	11.92	7.28	4.95
1600	0.368	170.2	252.0	12.70	7.75	5.27
1700	0.365	169.9	252.3	13.49	8.21	5.58
1800	0.362	169.6	252.6	14.26	8.67	5.90
1900	0.359	169.2	253.0	15.02	9.12	6.20
2000	0.356	168.9	253.3	15.81	9.57	6.50
2100	0.354	168.6	253.6	16.55	10.01	6.87
2200	0.351	168.3	253.8	17.29	10.46	7.11

$$q_{tot} = (\rho_1 + \rho_2)(\rho_1 - \rho_2) \frac{A_f\ g_c\ D_e^2\ c_p \left(t_{f_2} - t_{f_1}\right)}{192\ \mu_{ave}} \qquad (9.1.1.8)$$

where:

$\rho_{1\ \&\ 2}$ = density of fluid at inlet and outlet, respectively, lb/ft^3

A_f = total flow area, ft^2

g_c = gravitational constant ft/h^2

D_e = equivalent diameter, ft

c_p = specific heat, Btu/lb°F

μ_{ave} = average viscosity, lb/h ft

$t_f\ \&t_f$ = inlet and outlet temperatures, °F

Heat transfer coefficients were calculated from[7]

$$N_u = 0.55\ Ra^{1/4} = \frac{hZ_b}{k} \qquad (9.1.1.9)$$

where:

Ra = the Rayleigh number and is the producer of the Grashof and Prandtl numbers

N_u = Nusselt Number

Z_b = axial distance along the channel; origin at core bottom, ft

The local Grashof number depends on the temperature rise of the fluid which depends on the total heat removed from the core. The total heat was related to maximum heat flux by Equation 9.1.1.2 and the wall temperature was calculated by

$$q''_{max} = h\left(t_w - t_{f_{ave}}\right) \qquad (9.1.1.10)$$

The analysis was done by varying the coolant temperature rise to calculate q_{tot} and h and from these a wall temperature was calculated. The results indicated that for natural convective cooling a wall temperature of 250°F would be reached at a power level of 0.97 MW. This was further reduced by the hot channel factor of Paragraph 9.1.1.6 to 0.66 MW. Based on these conservative assumption, this is the maximum power at which the ULR could be operated when cooled by natural convection.

9.1.1.8 Safety Limits The curve shown in Figure 9.1 defines the safety limits for the ULR. The upper limit on coolant inlet temperatures is 110°F. The minimum height of water above the core is included as a condition in the analysis and is limited to not less than 24 ft above the core center line.* The safety limits for the forced convection mode are established by the combination of coolant flow rate and reactor power which results in nucleate boiling in the hot channel. Disruption of fuel element integrity is assumed to occur at the onset of nucleate boiling. Plotted is the total core power in MW versus the fraction of full core flow normalized to the reference core flow of 1400 gpm. The curve above ~34% of full core flow represents the forced convection mode and the limiting criterion on core power is the onset of nucleate boiling in the hot channel. Above and to the left of this portion of the curve is the boiling region and below and to the right is

*See exception for low power natural convection in Paragraphs 9.1.2.3.3 and 9.1.2.4.

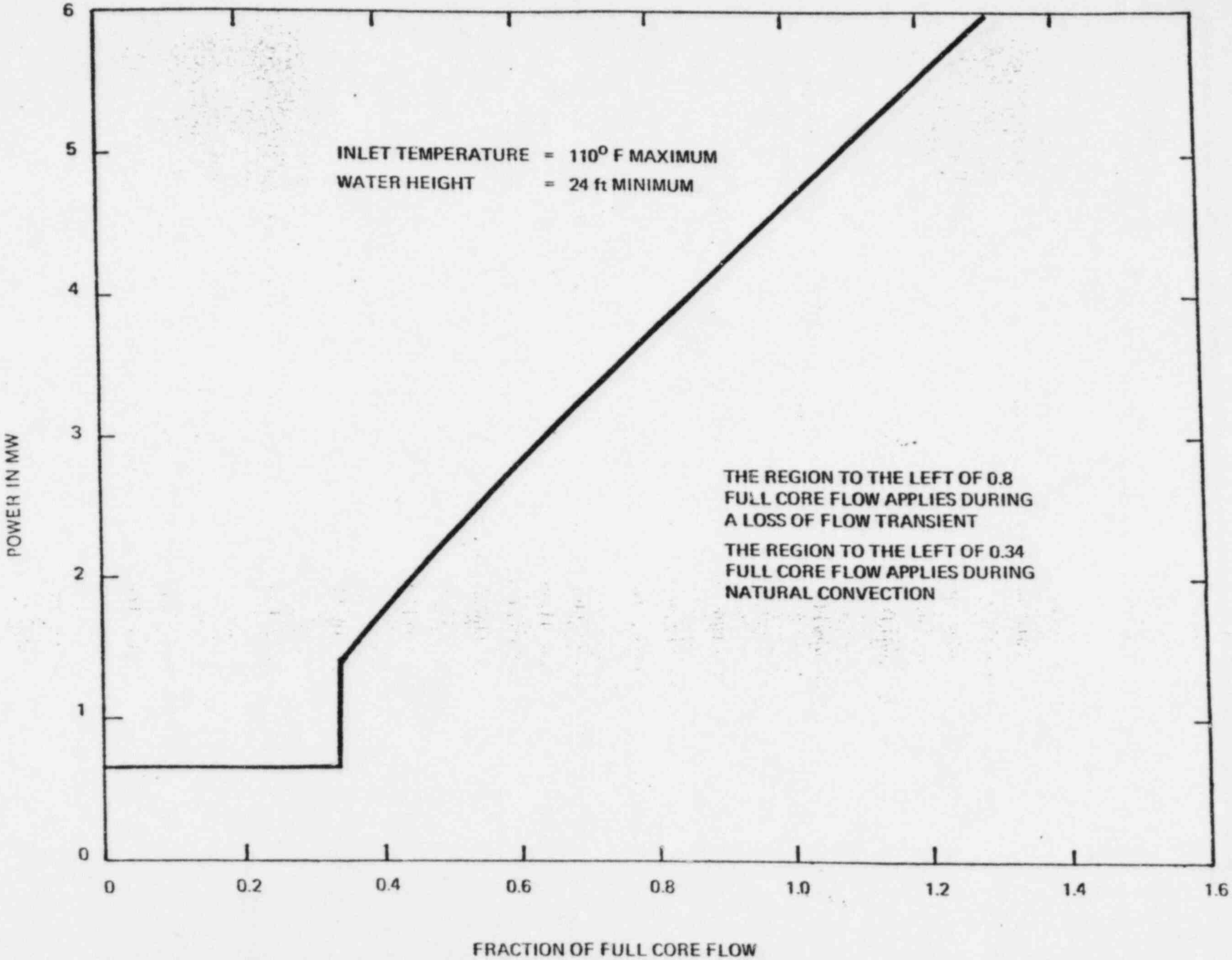


Figure 9.1 Power-Flow Safety Limit Curve

the region of no boiling.

In the region below 34% of full core flow the limiting criterion is the fuel element clad temperature. Operation above 0.66 MW under natural convection would result in clad temperatures exceeding 250°F.

The data shown represent minimum heat fluxes at which fuel damage could possibly occur under worst conditions.

9.1.1.9 Burnout Heat Flux The burnout heat flux was estimated from the Mirshak-Durant equation. This empirical equation was used to predict burnout heat fluxes for the Omega West Reactor.[8] The only real difference between the ULR and OWR fuel is that OWR used curved MTR fuel plates having an ℓ/d ratio of 104 versus 120 for the flat fuel plates in the ULR.

The Mirshak-Durant equation is

$$q''_{BO} = 479000(1 + 0.0365v)(1 + 0.00507\ \Delta t_{sub})(1 + 0.0131p) \pm 16\% \quad (9.1.1.11)$$

where:

Δt_{sub} = saturation temperature minus the bulk fluid temperature, °F

v = fluid velocity, ft/sec

p = pressure, lb/in^2

For purposes of setting the Safety Limits it was assumed that the onset of nucleate boiling results in disruption of fuel element integrity, while in actuality fuel element failure will not occur until burnout heat fluxes are attained. Consequently, a measure of the built-in factor of safety that is attained by the assumption used to specify Safety Limits is given by the ratio of the burnout heat flux to the nucleate boiling heat flux. This ratio is shown in Figure 9.2 as function of total core flow. The burnout heat flux has been reduced by 16% to reflect the uncertainty in the Mirshak-Durant equation and by the factor $F_1 = 1.47$.

Design operation of the ULR was at 1 MW with a core flow of 1400 gpm. The burnout ratio is defined as the ratio of the burnout heat flux to the local heat flux in the hot channel; the burnout ratio was calculated for the operating conditions specified by reducing the q''_{BO} by the factor $F_1 = 1.47$ and increasing the local heat flux, q''_{max}, by the same factor. Under these conditions the burnout ratio is 29.

9.1.2 Limiting Safety System Settings

9.1.2.1 Coolant Inlet Temperature The Limiting Safety System Setting (LSSS) on the coolant inlet temperature is 108°F. Normal operation is at about 80-90°F. The manufacturer's specified probable error on the measuring instrument is less than ± 0.5°F and the standard deviation is ± 0.75°F. Using 2 standard deviations for 95% confidence factor the temperature can be determined within ± 1.5°F. For an LSSS of 108°F the limiting true value of the inlet temperature is less than 110°F. All calculations are based on an inlet temperature of 110°F.

For operation in the natural convection mode, the LSSS is imposed on the pool temperature monitor rather than the inlet

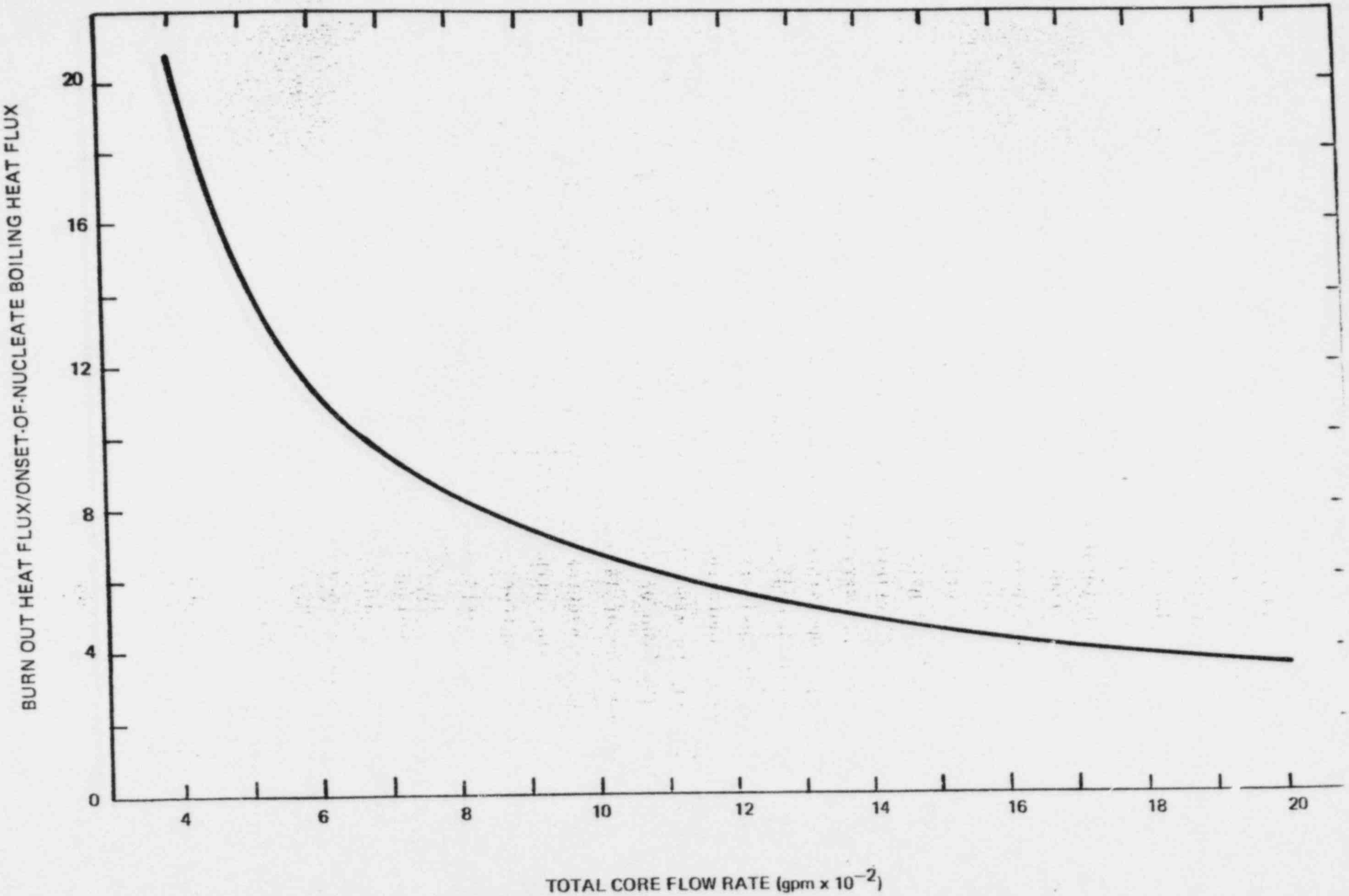


Figure 9.2 Ratio of Burnout [illegible] Nucleate Boiling Heat Fluxes Versus Flow Rate

temperature monitor, since the latter is downstream in the primary system piping and therefore not functional in this mode. The pool is effectively open in this mode of operation and the pool temperature monitor is effectively the coolant inlet temperature. In fact, when the monitor is located near the core suspension structure, the temperature read is more nearly characteristic of the outlet temperature; taking this as the defining monitor is very conservative since the outlet temperature is always higher than the inlet temperature. The LSSS on the pool temperature monitor for the natural convection mode of operation is specified as 108°F.

9.1.2.2 Flow Rate The primary coolant flow is measured by an orifice meter of the head type having a manufacturer's specified standards error of ± 2%. For a 95% confidence factor the error is then ± 4%. The LSSS (minimum) on the flow rate is 1170 gpm. The true value would then be no less than 1120 gpm which is 80% of the full core design flow of 1400 gpm.

9.1.2.3 Reactor Power Reactor power is monitored by the linear and log N neutron channels, and an effluent primary coolant radiation monitor (N-16). Additional confidence in these calibrations may be gained by applying calorimetry techniques in which the power is determined by the product of the flow rate x ΔT, the coolant temperature rise across the core. Safety level scram points are set by adjusting the neutron monitors to read the same power level as the value measured with the calibration devices.

Each of the above methods of calibration are comparable in terms of accuracy attainable. To establish a limiting true value, the latter method is analyzed.

The standard errors in determining the power level settings are:

(1) Flow rate $\pm$ 2%

(2) Temperature $\pm$ 1%

(3) Scram system settings $\pm$ 2%

Since the errors are independent, they add in quadrature so that the overall standard error is 3%, resulting in an uncertainty of 6% (for 95% confidence).

9.1.2.3.1 Case of Forced Convection and 24 Feet (Minimum) of Pool Water The LSSS on reactor power will be such that the true value does not exceed the Safety Limit even under the loss of flow condition described in Paragraph 9.1.3. Under this condition, flow decreases until 80% of rated flow is reached, at which point the reactor scrams. The control blades for the ULR drop into the core in one second; the negative reactivity with all blades in is conservatively 7% $\Delta k/k$ so that the power has dropped by a factor of approximately 10 one second after the flow reached 80% of it rated value. For the ULR operating in the region of 1 MW steady power, the power after the prompt drop from blade insertion is natural convection (see Figure 9.1), so that there is no possibility of exceeding a Safety Limit from this time onward, and it is only necessary to establish that Safety Limits are not exceeding during the flow coast down.

From Figure 9.3, it can be seen that the flow at 1 second after the time corresponding to 80% of full flow is 38% of full core flow. Reduction by a further 10% for uncertainty in Figure 9.3 leads to 34% of full core flow which corresponds to a power of 1.4 MW from Figure 9.1.

The LSSS on power is stipulated to be 1.25 MW which is 125% of the steady-state operating power of 1 MW. Taking the uncertainty in power to be 6%, the true value of the power (95% confidence) would then be no more than 1.33 MW, which is below the 1.4 MW derived above.

It should be recognized that the analysis presented is very conservative in that it takes (and requires) no credit for the power drop as the control blades are being inserted, but rather considers the time of the full prompt jump. In reality, as the blades start down, the reactor is immediately subcritical and decreasing in power; only about 1/10 of the full 0.07 $\Delta k/k$ available is needed to reduce reactor power to 0.66 MW, the Safety Limit specified for natural convection.

9.1.2.3.2 <u>Case of Natural Convection and 24 Feet (Minimum) of Pool Water</u> Under natural convection, 24 feet of pool water, and 110°F, the maximum measured value of power, i.e., the LSSS based on a Safety Limit power variable, is 620 kW. Using an uncertainty of 6% (for 95% confidence) as above in the power measurement, the true value of power would then be no more than 0.66 MW, the Safety Limit for natural convection as shown in Figure 9.1.

While the LSSS, determined from consideration of the Safety Limit which is based on plate temperature, is found above, a limit of 133 kW will be imposed for operation under natural convection.

In this mode, the scram setting will be 125% of 100 kW, which means that the true value of power, assuming an uncertainty of 6% for 95% confidence, will be no more than 133 kW. This limit is imposed to ensure a low rate of production and escape of ^{16}N.

9.1.2.3.3 <u>Case of Natural Convection and 2 Feet (Minimum) of Pool Water</u> At less than 24 feet (but $\geq$ 2 feet) of water above core center line, the maximum true value of power is 1.33 kW. At this low power ^{16}N is not a problem, nor is there any chance of boiling the core water. Using an uncertainty of 6% in the power measurement as above, the maximum power at the safety system scram setting (LSSS) of 1.25 kW for this mode of operation is 1.33 kW.

9.1.2.4 <u>Pool Water Height</u> The LSSS on pool water height is 24 feet 3 inches above core centerline for all operation >1.33 kW. This is the minimum scram setting on a float switch attached to the reactor bridge. The error in height is taken to be $\pm$1.5 inch, which includes allowances for positioning the pool ripple. Taking the uncertainty as twice this value leads to the conclusion that the true value of pool water height will be no less than 24 feet above core center line.

The LSSS on pool water height for the special allowed case of operation up to 1.33 kW is 2 feet 3 inches, where the 3 inches, as above, assures that the true water height will be no less than 2 feet above core center line.

9.1.3 <u>Loss of Primary Flow</u>

In the event of a loss of primary flow, the reactor is set to scram at 80% of the full core flow. Also, the gates in the flow channels above the core fall open and permit natural convection cooling with pool water. The operation of the gates is described in Chapter 4.

The flow coast down curve for the ULR is given in Figure 9.3. For comparison, also shown are the curves used in the original FSAR analysis. These are from the Rhode Island Nuclear Science Center Reactor and from a calculation using a nodal technique by Temple[9]. It should be noted that the ULR has better flow retention during coast down in comparison to both the calculated and the Rhode Island Nuclear Science Center Reactor Curve.

Figure 9.4 shows the behavior of the flow calculated for an extended time interval, and predicts that the flow coasts down and reverses because of natural convective effects after about 85 seconds.

The predicted flow behavior over relatively long time intervals is strongly dependent on the input parameter values because of opposing tendencies in the system as a result of the chimney effect of the plenum beneath the core and extending up the outside of the core to the riser coolant gate. In the calculation, friction and acceleration terms tend to slow down the flow but the coolant density terms have a much more complex effect. Of great importance are the coolant densities in the modes which describe that part of the system from the core exit (core bottom), bottom plenum, and return plenum up to the riser coolant gate. On the one hand, these density terms which are a function of flow rate and total heat being removed can be such

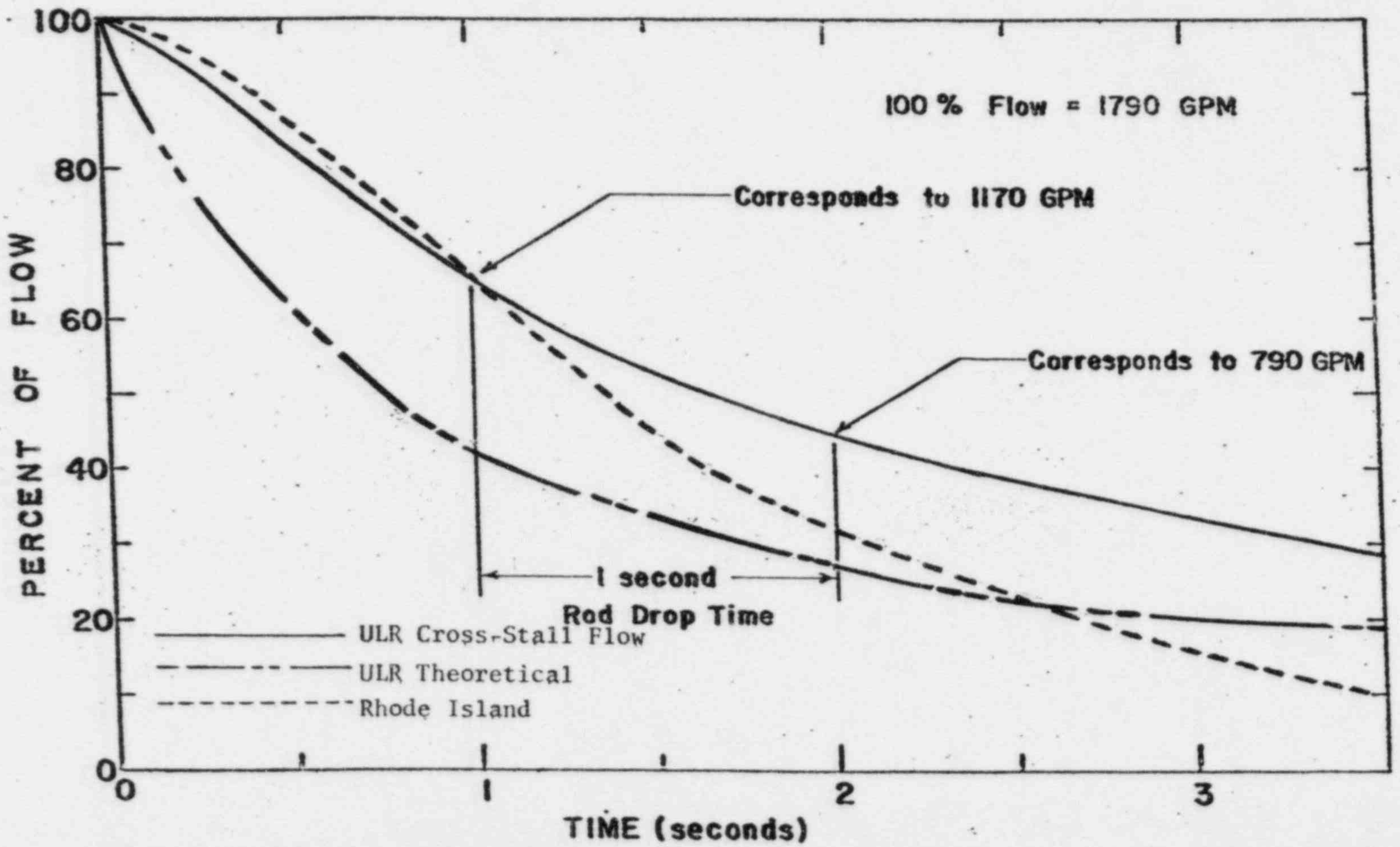


Figure 9.3 Normalized Primary Flow Coast Down

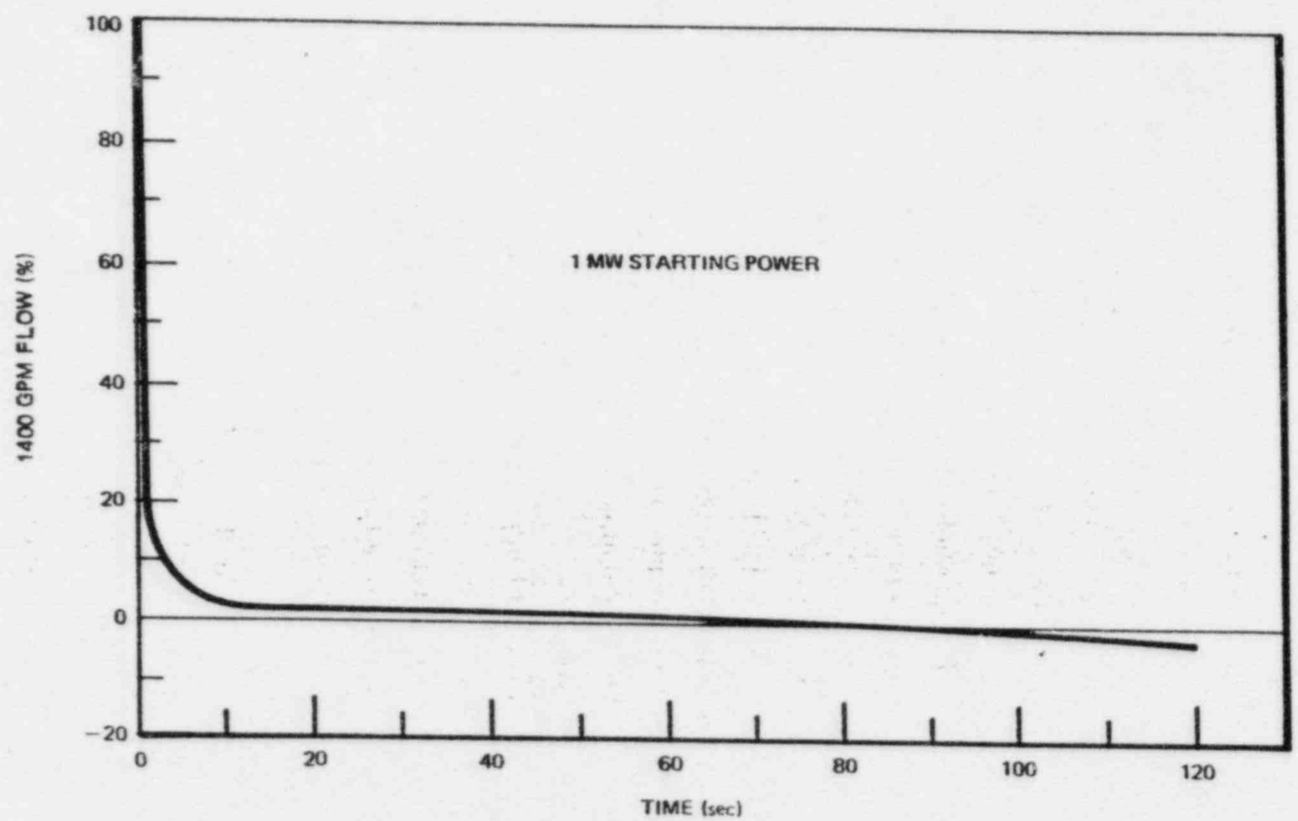


Figure 9.4 Coast Down and Reversal of Primary Flow After Loss of Pump

that flow reversal is predicted, because the natural tendency for hot water in the core to rise dominates the chimney effect of warm water moving up the plenum; on the other hand, the density of the exit water and flow condition can be such that the chimney effect (i.e., warm water exiting from the core bottom, rising through the exit plenum, and flowing into the pool through the riser gate above the core) predicts non-flow reversal, since once this mode is established, warm water continues to flow up the chimney, thus drawing colder water down through the core. An example of the latter situation was given in an earlier submittal,[10] in which the calculation, described in detail, was applied to the original design power of 5 MW and flow of 2800 gpm before the loss of flow, and predicted no flow reversal.

While the above phenomenon is academically interesting, it has no real bearing on safety considerations because, in the time region being considered, the residual core power is about equivalent to 50 kW for the 1 MW steady state operating case presently being considered. If flow reverses, the reversal being discussed is a theoretical bulk reversal, whereas, in fact, in the ULR core system the water flows through parallel channels, some of which are hotter than others; thus, not all channels would behave identically and the instantaneous stoppage of <u>all</u> core flow just prior to reversal is not implied. If the flow does not reverse, then a continual cooling effect is realized since the driving mechanism for this case is still natural convection, even though the water is going through the core in a direction not normally associated with natural convection. For either case, at any given point in time, some core flow would occur, so that there are no reasonable grounds to consider a heat buildup problem at the low residual power level under consideration, whether flow reverses or not. Indeed natural convection cooling is effective up to powers of some 660 kW (under very conservative assumptions) as shown in Paragraph 9.1.1.7.

9.1.4 Partial Loss of Water

9.1.4.1 Design Considerations The pool is specifically designed to preclude the probability of drainage. It is constructed of reinforced concrete with approximately six foot thick walls and a heavy aluminum liner to resist the most severed earthquake that might reasonably be expected in the area. There is no penetration of the reactor pool below the top of the core that is open to pool water.

All penetrations of the pool are provided with multiple barriers against the possibility of leakage of pool water. Those penetrations at or near core level are described below.

(1) The 4x4 foot square thermal column and 2x2 foot square gamma cave penetrations of the pool walls are provided with a heavy aluminum reinforcing plate welded behind the liner for structural strength. Secondary barriers in the form of outer doors which are normally locked are also provided.

(2) The 3x3 foot square unused medical facility is a nozzle which is welded to the liner, but is closed off by two heavy bolted cover plates.

(3) Four 6-inch and two 8-inch beam ports penetrate the liner at core centerline. Each beam port is closed by a bolted

cap at the pool end, by a heavy lead shutter located within the pool wall, and by a bolted shield plug at the outer end.

It is highly unlikely that the beam ports could be severely damaged while the reactor core is in the stall pool because of the restricted space and the protection afforded by both the reactor and the bridge which cover a major part of the area.

While the reactor core is in either pool, severe damage to a penetration in the adjacent pool could be prevented from uncovering the core by closing the gate between the two pools, an operation that can be done in a matter of a few minutes.

A guillotine pipe break in the ten-inch diameter primary coolant pipe between the reactor and heat exchanger would not empty the pool. Such a double ended pipe break would allow flow from both ends of the pipe, i.e., from both inlet and outlet pipes in the pool, but both inlet and outlet pipes are fitted with antisiphon devices which would prevent siphon-draining of the pool below a level of 11 or 12 feet above core center line. This is substantially less severe than the assumption of a beam port break used in the description of the design Basis Accident below.

9.1.4.2 Design Basis Accident In view of the inherent intergrity of the design features, a loss of pool water to the point of uncovering some of the core is unlikely; nevertheless, an analysis was carried out to determine the consequences of a partial loss of water from the pool. The power condition imposed was considerably more severe than anticipated power in order to impart a large measure of conservatism to the conclusions.

The basic assumptions used were:

(1) An 8-inch diameter beam tube is ruptured following an infinite period of operation at a 5 megawatt power level.

(2) The reactor scrams concurrent with the beam tube rupture. No single failure renders inoperable the automatic and manual scram system, so this assumption is well founded.

Emptying of the reactor pool to the lower edge of a suddenly opened 8-inch diameter port will require 27 minutes. The lower active portion of a fuel plate will then be standing in slightly more than 8 inches of water. Time for emptying is derived as follows:

Consider a certain reservoir of liquid with uniform vertical cross section, A_1, filled to a height y. At the bottom of the reservoir is a closed orifice of cross-sectional area, A_2, which can be instantly opened. Sudden opening of the orifice is accompanied by a diffential decrease in volume equal to $A_1 dy$ after the incremental lapse in time, dt. Conservation of mass requires that the same differential decrease in volume be equal to the quantity discharged through A_2. Thus

$$A_1 dy = A_2(2gy)^{\frac{1}{2}} dt \qquad (9.1.4.1)$$

where g is the acceleration due to gravity (32.2 $ft\text{-}sec^{-2}$).

Following separation of variables, and integrating between the limits

$y = y_1$, at $t = 0$

and

$$y = 0, \text{ at } t = t$$

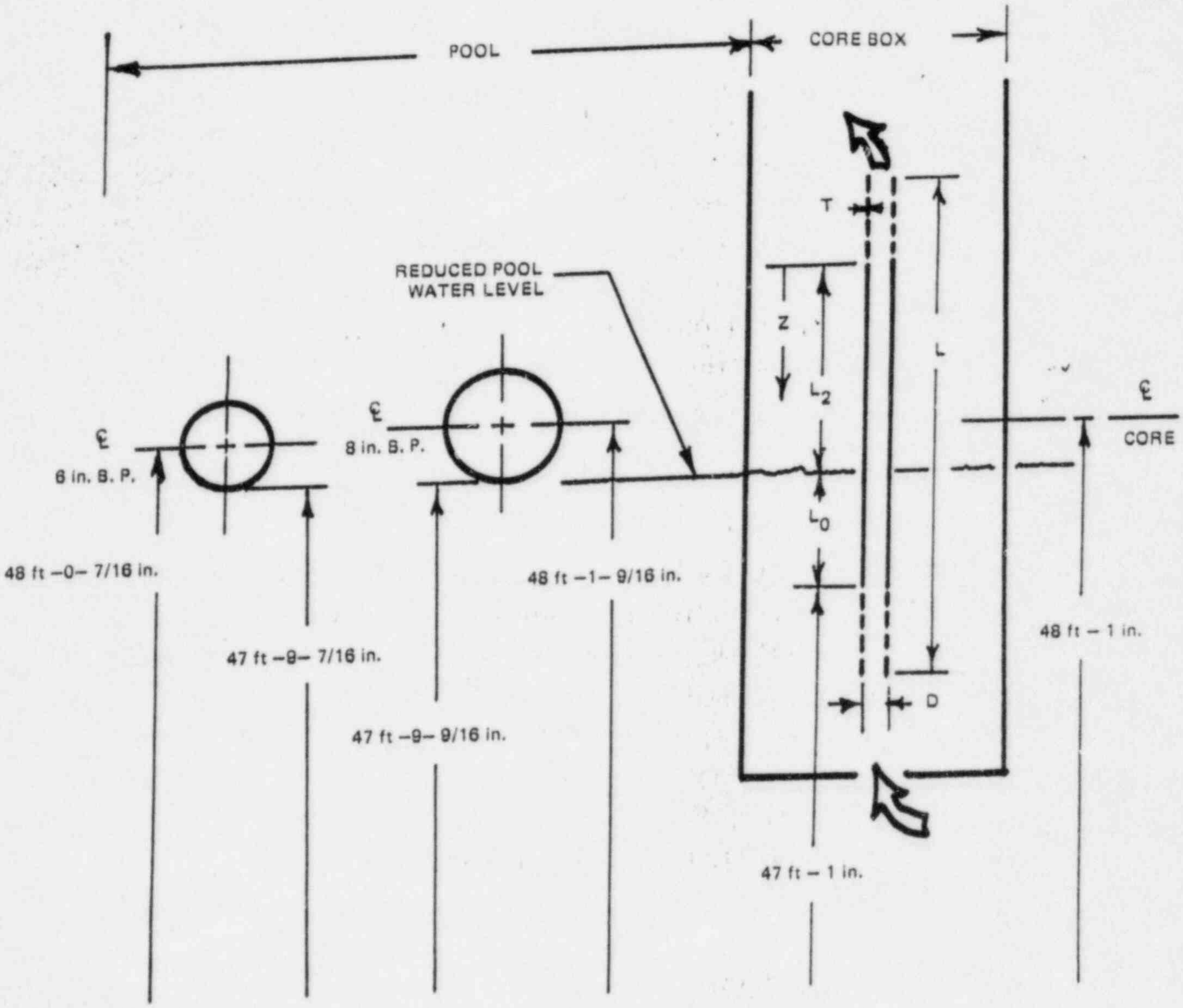
we find that the time for emptying is given by

$$t = \frac{A_1}{A_2}\left[\frac{2y_1}{g}\right]^{1/2}, \text{ seconds} \qquad (9.1.4.2)$$

The pool will thereby empty in 1640 seconds (27 minutes) after discharging through an 8-inch diameter orifice initially submerged at a location 24 feet below a constant surface of 372 square feet for all pool water depths. The orifice coefficient was taken as 0.9.

If the nuclear reactor experienced a scram concurrent with the rupture of a beam tube, according to the Way-Wigner formula[7] fission product decay heat at the end of 27 minutes would evolve at a rate of 2.9×10^5 Btu h^{-1} following an infinite period of operation at a 5 megawatt power level.

The lower fuel bearing portion of the reactor core would now be standing in at least 8 inches of water heated to saturation (212°F) at atmospheric pressure. (See Figure 9.5.) In the meantime, a highly saturated steam-water mixture would by natural convection be circulating upward through the rectangular coolant flow passages. Natural convection heat transfer by boiling would thereby maintain the surface temperature of the exposed fuel



Z ; DISTANCE FROM ORIGIN, ft
L_2; EXPOSED LENGTH OF ACTIVE FUEL PLATE, ft
L_0; SUBMERGED LENGTH OF ACTIVE FUEL PLATE, ft
L ; TOTAL LENGTH OF COOLANT PASSAGE, ft
D ; HYDRAULIC RADIUS OF COOLANT PASSAGE, ft
T ; THICKNESS OF FUEL PLATE, ft

Figure 9.5 Idealization of Reactor Core Conditions Following Partial Loss-of-Water Accident slightly above (6 to 8°F) the saturated temperature of the steam-water mixture. Thus, the aluminum clad fuel plates would experience no damage whatsoever since 220°F is much below the 1200°F temperature required for melting.

Consider a single coolant flow passage as depicted in Figure 9.5. Equilibrium conditions for natural convection flow require that the net available difference in head, H_a, between the steam-water mixture inside a coolant flow passage, and the more dense water surrounding the partially submerged core exactly balances the various system losses, H_s, encountered by the upward flowing mixture percolating through the core. Once equilibrium is reached, both H_a and H_s will have simultaneously equal values for the identical weight fraction of steam contained within the flowing mixture. Hereinafter, the weight fraction of steam, x, will be designated as the <u>top dryness fraction.</u>

Intersection of the respective plots for both $H_a(x)$ and $H_s(x)$ versus the top dryness fraction, x, will by successive trial yield the desired simultaneous solution for each of the above.

The net available head, $H_a(x)$, as a function of the top dryness fraction, x, can be calculated by means of

$$H_a(x) = \frac{d_o}{d_{sat}} L_o - \frac{d_m(x)}{d_{sat}} L \qquad (9.1.4.3)$$

where:

$H_a(x)$ = net available head, ft

d_o = density of 100°F water in partially emptied pool, lb-ft^{-3}

d_{sat} = density of saturated water (212°F), lb-ft^{-3}

$d_m(x)$ = true mean density of steam-water mixture, lb-ft^{-3}

L_o = submerged length of active portion of fuel plate, ft

L = total length of coolant passage, ft

The true density, d_m (where the x has been dropped for brevity), for a saturated steam-water mixture being generated in a uniformly heated, vertical flow passage can be derived as follows:

Let d_m be defined as the quotient obtained by dividing the fractional change in weight between the entering saturated water and the exiting steam-water mixture (over the volume of interest) by the difference in specific volumes of the same. Thus,

$$d_m = \frac{\int_{v_f}^{v_e} \frac{1}{v}\, dv}{v_e - v_f} \qquad (9.1.4.4)$$

where v is the specific volume, and the subscripts, f and e, refer to the entrance and exit conditions, respectively.

Upon integrating, we get

$$d_m = \frac{\ln \frac{v_e}{v_f}}{v_e - v_f} \qquad (9.1.4.5)$$

for the average, or true mean density. But, the specific volume of the exiting steam-water mixture, v_e, is equal to the specific volume of the saturated water, v_f, plus the volume fraction of steam, xv_{fg}, added thereto during the process of steam generation over the length and volume of the coolant flow passage.

Hence,

$$d_m(x) = \frac{1}{xv_{fg}} \ln \left[\frac{v_f + xv_{fg}}{v_f}\right], \ (lb\text{-}ft^{-3}) \qquad (9.1.4.6)$$

In the limit, as the top dryness fraction, x, approaches zero, $d_m(x)$ approaches $\frac{1}{v_f}$, i.e., the density of saturated water. Obviously, no steam would be present under the stated condition.

In a coolant passage wherein steam is being uniformly generated throughout its length, the system loss, $H_s(x)$, as a function of the top dryness fraction, x, is equal to the sum of the following losses, namely,

entrance loss + friction loss + acceleration loss + exit loss

or,

$$H_s(x) = \frac{G^2(x)}{2g} \left[\frac{{v_f}^2}{2} + f \frac{L}{D} {v_a}^2(x) + 2\ v_a\ (x) \left[v_e(x) - v_f\right] + v_e^2(x)\right] \qquad (9.1.4.7)$$

where:

x = top dryness fraction, lb steam per lb water

$G(x)$ = mass flow, $lb\text{-}ft^{-2}\text{-}sec^{-2}$

g = acceleration due to gravity, 32.2 $ft\text{-}sec^{-2}$

v_f = specific volume of saturated water, $ft^3\text{-}lb^{-1}$

$v_e(x) = v_f + xv_{fg}$; specific volume of exit mixture, $ft^3\text{-}lb^{-1}$

v_{fg} = difference in specific volume between saturated steam and saturated water, $ft^{-3}\text{-}lb^{-1}$

$v_a(x) = \frac{v_e(x)+v_f}{2}$; average specific volume of steam water-mixture, $ft^3\text{-}lb^{-1}$

f = hydraulic friction factor for coolant passage

L = total length of coolant passage, ft

D = hydraulic radius of coolant passage, ft

Intersection of the trial curves of $H_a(x)$ and $H_{s(x)}$ occurs when the top dryness fraction, x, is equal to 0.025 for a submergence depth of 8 inches (or, 0.67 ft). The total flow rate of saturated water passing through the entire core is 11,900 lb per hour. This is readily verified by inspection of Figure 9.6. Also, the same plot shows that the corresponding mass flow rate, G(x), is equal to 3.56 $lb\text{-}sec^{-1}\text{-}ft^{-2}$.

Low concentration by weight of steam in saturated steam-water mixtures gives rise to evidence that large heat transfer coefficients realized through the mechanism of boiling are certainly present. For the average thermal heat flux (840 $Btu\text{-}h^{-1}\,{}^{o}ft^{-2}$) present in our situation (i.e., the decay

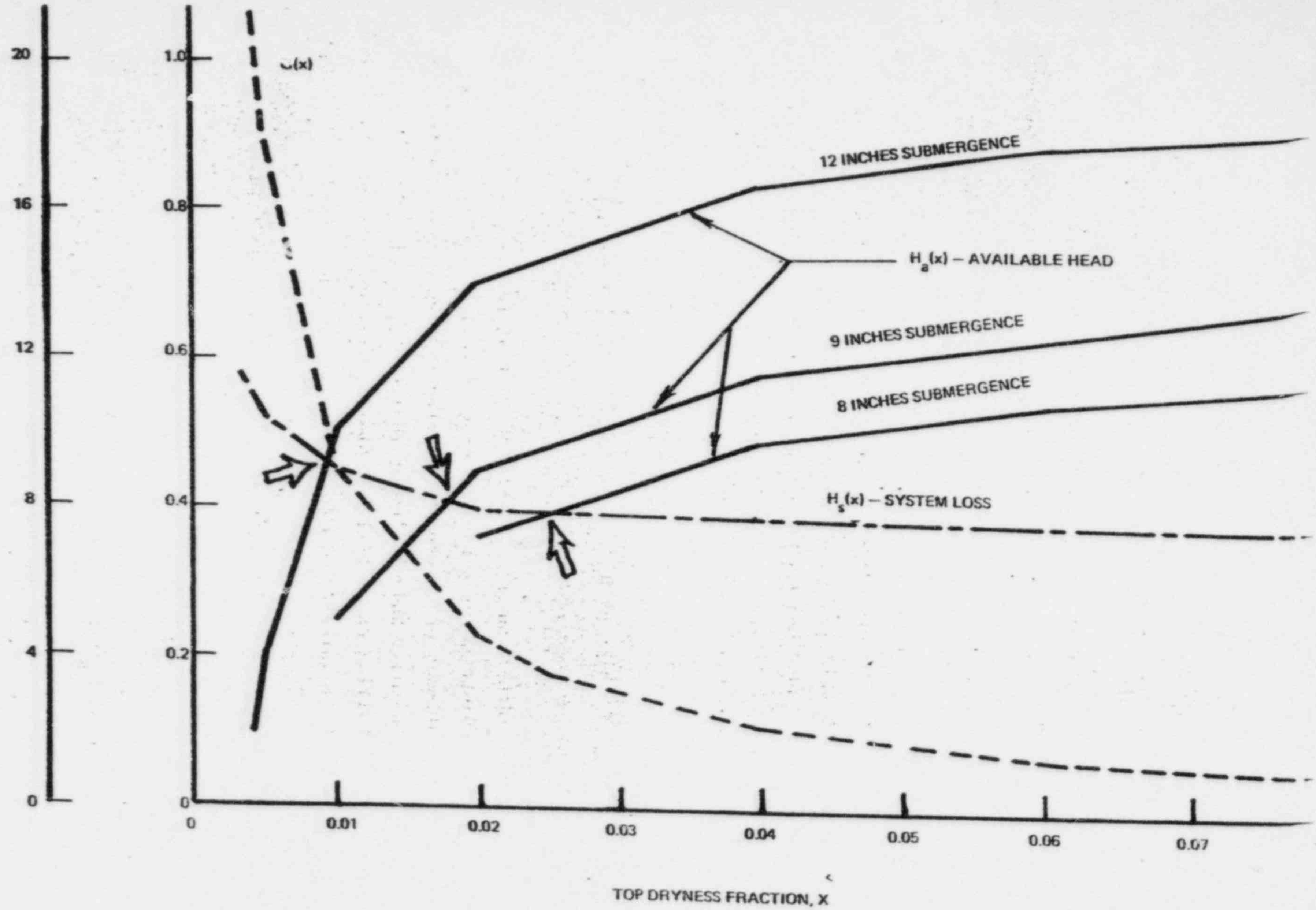


Figure 9.6 Trial Curves for the Functions $H_a(x)$ and $H_s(x)$

heat release rate 27 minutes after shutdown following a period of "infinite" operation at 5 megawatts) the associated boiling water heat transfer coefficient[11] would be no less than 250 Btu h^{-1} ft^{-2} $^oF^{-1}$.

Heat removal from a fuel plate under the stated conditions is in general accomplished through two modes of heat transfer: namely, conduction downward along the fuel plate to what water remains in the pool, and convection to the stream of boiling water flowing upward through the core coolant passages.

Radiant heat transfer can be ignored since the prevailing temperature difference between the fuel plates and immediate surroundings is insignificant insofar as this temperature differtial could affect any appreciable amount of energy transfer by this latter method.

When due allowance has been made for heat transfer by both conduction and connection around an element of volume, w t dz, of fuel plate material, at a distance z from the top active portion of the plate, a simple heat balance will give the following:

$$wtk \frac{dT(z + dz)}{dz} + wt\ dz\ q_o = w\ dz\ h \left[T(z) - T_o\right] + wtk \frac{dT(z)}{dz} \quad (9.1.4.8)$$

where:

w = width of active portion of fuel plate, ft

t = thickness of active portion of fuel plate, ft

dz = element of length, at z, along active portion of fuel plate, ft

k = thermal conductivity of fuel plate, Btu h^{-1} ft^{-1} $^oF^{-1}$

h = boiling water heat transfer coefficient,

Btu h^{-1} ft^{-2} $^oF^{-1}$

q_o = heat release rate per volume of fuel plate material ,

Btu ft^{-3} h^{-1} (assume constant throughout volume)

T_o = saturation temperature of steam-water mixture, and

saturated water surrounding fuel plate material, ^oF

$T(z)$ = temperature of fuel plate at z, ^oF

also,

$\frac{dT(z)}{dz}$, and $\frac{dT(z + dz)}{dz}$ are the temperature gradients at z, and z + dz, respectively.

In the limit, as dz approaches zero, we get the following nonhomogeneous differential equation. Thus

$$\frac{d^2T(z)}{dz^2} - \frac{h}{t\ k} T(z) = -\left[\frac{h}{t\ k} T_o + \frac{1}{k} q_o\right] \qquad (9.1.4.9)$$

after rearranging terms.

The general solution to the above differential equation is of the form

$$T(z) = A \exp (a\ z) + B \exp (-a\ z) + C \qquad (9.1.4.10)$$

where:

$$a^2 = \frac{h}{t\ k}.$$

Solving for C, we get

$$T(z) = A \exp(a\ z) + B \exp(-a\ z) + T_o + \frac{tq_o}{h} \qquad (9.1.4.11)$$

At the origin, z equal to zero (i.e., the topmost edge of the active fuel bearing portion of the fuel plate), the leakage term, $\frac{d^2T(z)}{dz^2}$ does not exist because of the discontinuity in the gradient $\frac{dT}{dz}$ brought about by the lack of heat sources for all z less than zero. Therefore,

$$A + B = 0$$

and

$$T(z) = A \sinh(a\ z) + T_o + \frac{t\ q_o}{h}\ . \qquad (9.1.4.12)$$

At the point of submergence, $T(L_2)$, where L_2 is the length of exposed fuel shown in Figure 9.5, equals T_o (i.e., the saturation temperature of the surrounding water) since T(z) is a continuous function. Thus

$$A = -\frac{t\ q_o}{h}\ \frac{1}{\sinh(a\ L_2)}\ . \qquad (9.1.4.13)$$

The complete solution for T(z) is now

$$T(z) = \frac{t\ q_o}{h}\left[1 - \frac{\sinh(a\ z)}{\sinh(a\ L_2)}\right] + T_o \qquad (9.1.4.14)$$

(9.1.4.14)

which is valid for all z ranging between the limits of zero, and L_2.

Substitution of the following list of appropriate numerical values, e.g.:

$t = 0.005$ ft

$L_2 = 1.33$ ft

$h = 250$ Btu-h^{-1}-ft^{-2}- F^{-1}

$k = 115$ Btu-h^{-1}-ft^{-1}- F^{-1}

$a = 20.5\ ft^{-1}$

$q_o = 2.98 \times 10^5$ Btu-h^{-1}-ft^{-3}

into the previous equation for T(z), finally gives

$$T(z) = 6.00 \left[1 - \frac{\sinh(20.5\ z)}{\sinh(27.3)}\right] + 212;\ \text{deg F.} \qquad (9.1.4.15)$$

Inspection of the immediately preceding two expressions for T(z) reveals that the dominant mode for heat transfer over the greater part of the exposed length of clad fuel plate is principally governed by the mechanism of convection. Heat transfer by conduction does not become too important when z is much less than L_2. Hence, for practical purposes we can safely assume that the maximum temperature of the fuel bearing portion of the clad plate will never exceed 220°F, provided, of course, that a few inches of water from the partially emptied pool be sufficient to submerge at least the lower quarter, or lower third of the otherwise uncovered fuel plates.

9.1.5 Total Loss of Water

Section 9.1.4 deals with the case of a partial loss of water from the reactor pool and shows that no fuel melting would result even if the reactor were operating at 5 MW. Although an event or sequence of events leading to a total loss of water from the reactor pool is difficult to postulate, it is informative to consider the consequences if all water were lost as the result of, for example, and earthquake of magnitude severly higher than any experienced in the recorded history of this area.

Work was done by Wett[12] in investigating the surface temperature of Oak Ridge Research Reactor (ORR) fuel elements under natural convection air cooling. This work, along with experimental results from the Low Intensity Testing Reactor (LITR) and the Livermore Pool-Type Reactor (LPTR) was correlated by Webster,[13] whose conclusions are pertinent because the ORR, LITR, and LPTR are all light-water moderated research reactors with solid plate-type fuel (commmonly called MTR or BSR type elements) which are very similar to the ULR design. Webster states, "This analysis indicates that the LITR could be operated continuously at 3 MW and could lose the cooling water through a rupture in the reactor tank without danger of melting the fuel, even without core spray." The indication that a total loss of water from the pool of the ULR operating at 1 MW would lead to no melting of the fuel is clear.

Despite the evidence cited in the preceding paragraph, safe practice demands that all reasonable precautionary measures be taken to preclude the possibility of inadvertant draining of the pool. Consequently, a "through-tube" facility present in the original design of the ULR was deleted because it passed beneath the core, and thus a failure in intergrity of this tube could lead to a drainage of the pool. For similar reasons, the prohibition of open feed pipes which extend below the core level for such services as make-up water, etc., precludes the possibility of siphoning the pool water to levels below the core.

9.1.6 Fuel Element Failure Under Operating Conditions

Fuel element failure can be caused by excessive hydraulic forces or excessive fuel temperature. At 1400 gpm flow rate, the pressure drop through the core is about 0.2 psi. Failure caused by hydraulic pressure imbalance under operating conditions is therefore beyond the realm of credibility.

The burnout ratio calculated in Paragraph 9.1.1.9 was 29, and this steam blanketing from excessive temperatures during normal operation is not indicated.

9.1.7 Binding of Control Blades

The time required for the safety blades to fall into the core has been measured to be ~650 milliseconds. If the blade is treated as a freely falling body then the drop time (for 2.5 ft) is 395 milliseconds which illustrates the time effect of water displacement by the falling blade. A somewhat more conservative blade drop time curve is given Figure 9.7,[15] according to which about 720 msec are required for a 2.5 ft drop. This curve is used in analyses in Paragraph 9.1.10.

Calculations indicate that distortion due to unequal heating is negligible and the design clearance between blade and shroud is 0.36 inches with a clearance of 0.1 inch between blade and blade

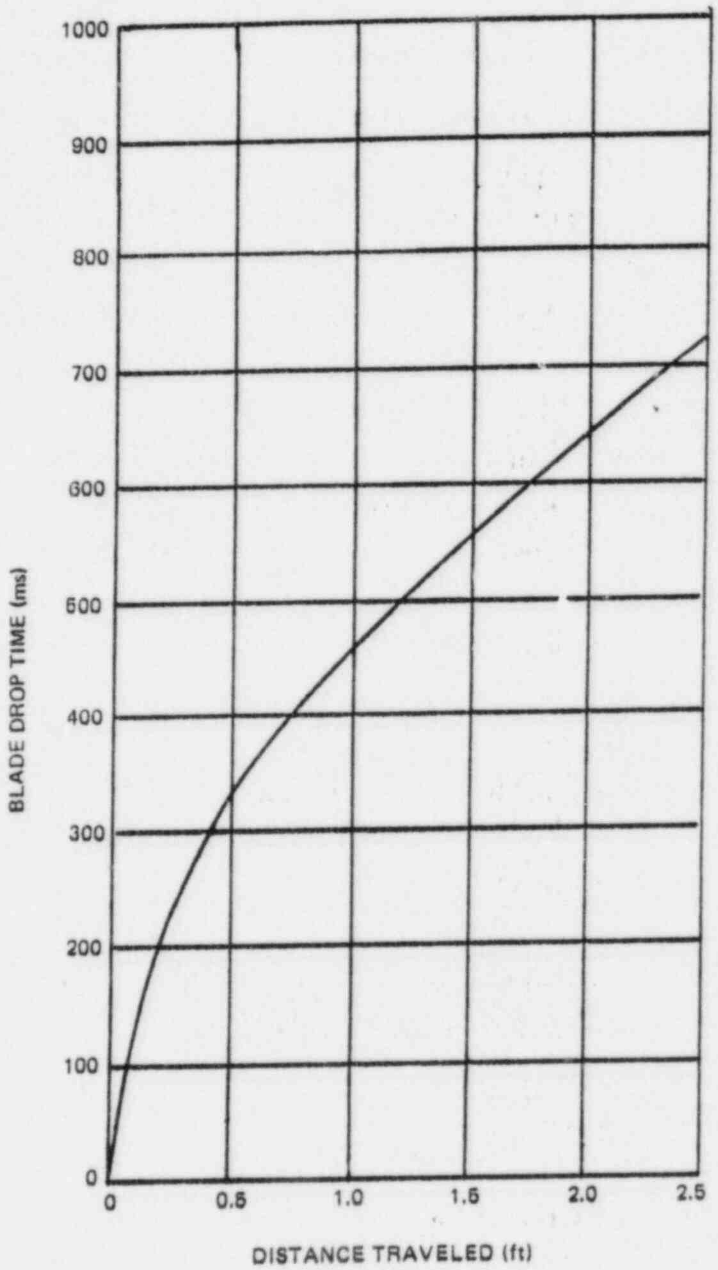


Figure 9.7 Drop Time for Control Blades

guide. Since any points of contact between blade and guide are lubricated by a water film, the effect of friction is negligible. A prototype control element was tested through approximately 100 scrams with no apparent wear.

On the basis of the calculations and tests, the possibility of scramming the control blades is judged to be positive under all conditions.

9.1.8 Release of Coolant Header Gates During Operation

A coolant header gate is provided in the primary coolant downcomer plenum and another one in the riser plenum, each of which will open in the event of reduced forced circulation flow during reacor operation. The various causes and effects of opening either or both of these gates during reactor operation were evaluated, with the results described below.

Case I Loss of Forced Circulation Pump

Coolant flow will drop and both header gates will open, while signals from each of the following will independently initiate reactor scram:

a. 20% loss-of-coolant flow (2 sensors)

b. coolant downcomer gate opens

c. coolant riser gate opens

Case II Downcomer Coolant Gate Opens*

It is unlikely that the gate would open under forced circulation flow, since the higher pressure within

*In the downcomer forced convection cooling mode the plenum relative to the pool will keep the gate closed. Nevertheless, if it is postulated that the gate does open, the flow through the core will not decrease. Since all flow going to the pump must pass through the core to the lower plenum and into the suction line, an open downcomer gate simply presents an alternate flow path to the core. In addition, the reactor would be scrammed by the "open-gate" switch.

Case III <u>Riser Coolant Gate Opens</u>

It is unlikely that the riser coolant gate would open even if the paddle broke off, so long as forced circulation flow continued, because of the pressure differential between pool and plenum. If it is postulated that the gate opens, the reactor would be scrammed by the "open-gate" switch and low flow.

9.1.9 Refueling Accident

The fuel element grapple can be used not only to replace fuel elements but also reflectors and radiation baskets. This feature creates the possibility of a refueling error. Erroneous removal of a reflector element with subsequent reloading of a fuel element is not an easy mistake because of the peripheral location of the reflector elements, the small core size and clearly recognizable reference points, the normal refueling sequence, and the fact that the refueling operation is supervised during the process. Even if it occurred, the difference in reactivity between a fuel element and a graphite element near the core periphery is about 0.5% or less, and this is not enough to jeopardize the shutdown capability of the rods which is at least 7% $\Delta k/k$. Dropping an element on top of a fully loaded core would add less than 0.5% $\Delta k/k$.

It is not possible to go critical with the control blades fully inserted in the core even if the entire grid were filled with fuel elements. If a loading error occurred, critically would be achieved with the safety blades in a lower position than normal, which would cause the operator to shut the reactor down before going to power.

Some experiments may need the high flux available inside the core; access would require replacement of a centralized fuel element with a radiation basket, and perhaps some more elements on the periphery of the core to make up reactivity. Experiments of this type require careful consideration before the fact (see Facility Technical Specifications) and are done, if approved, according to approved procedures in a way to ensure that there is no violation of the technical speifications which require minimum shutdown margins.

9.1.10 Step Increase in Reactivity

Each experiment must be analyzed (see Paragraph 10.5.3) to assure that no more than 0.5% reactivity can be added under any condition.

A digital computer analysis of a near instantaneous insertion of 0.5% $\Delta k/k$ at 1 MW power was done using TER-6,[16] a transient computer program which consists of a single mode flux model, a six group kinetic model and a multiple axial and radial node thermal model. The program incorporated the effect of density changes in the moderator, and was modified to include the effect of voids.

The TER-6 program calculated the power versus time transient throught the prompt jump and the subsequent peel of the power increase caused by delayed neutron transients to a point of power equilibrium from temperature and void coefficients effects. This curve was modified to include the effect on power of the control blades dropping into the core using the following assumptions.

(1) A scram is initiated during the first 100 msec of the transient by any or all of, (a) the log N safety channel (7 sec period) or, (b) one of the two redundant hight flux safety channels (120% of power). Since the safety channels involved are separate, and two are redundant, the single failure criterion is met in the protective action of the protective system level will be initiated.

(2) The control blades start to drop 180 msec after scram initiation (280 msec from the start of the transient) from an initial position of 30 inches withdrawn, i.e., the flattest part of the blade-worth versus position curve.

(3) The blade position versus time is as shown in Figure 9.8. The worth versus positions values used were taken from the more conservative experimental curve renormalized to a total curve worth of 7% $\Delta k/k$ as shown on the right side coordinate of Figure 9.8.

(4) As the blades are inserted, the power curve is affected only by the prompt drop, i.e., no credit is taken for the decrease in delayed neutron precursors, and temperature and void effects

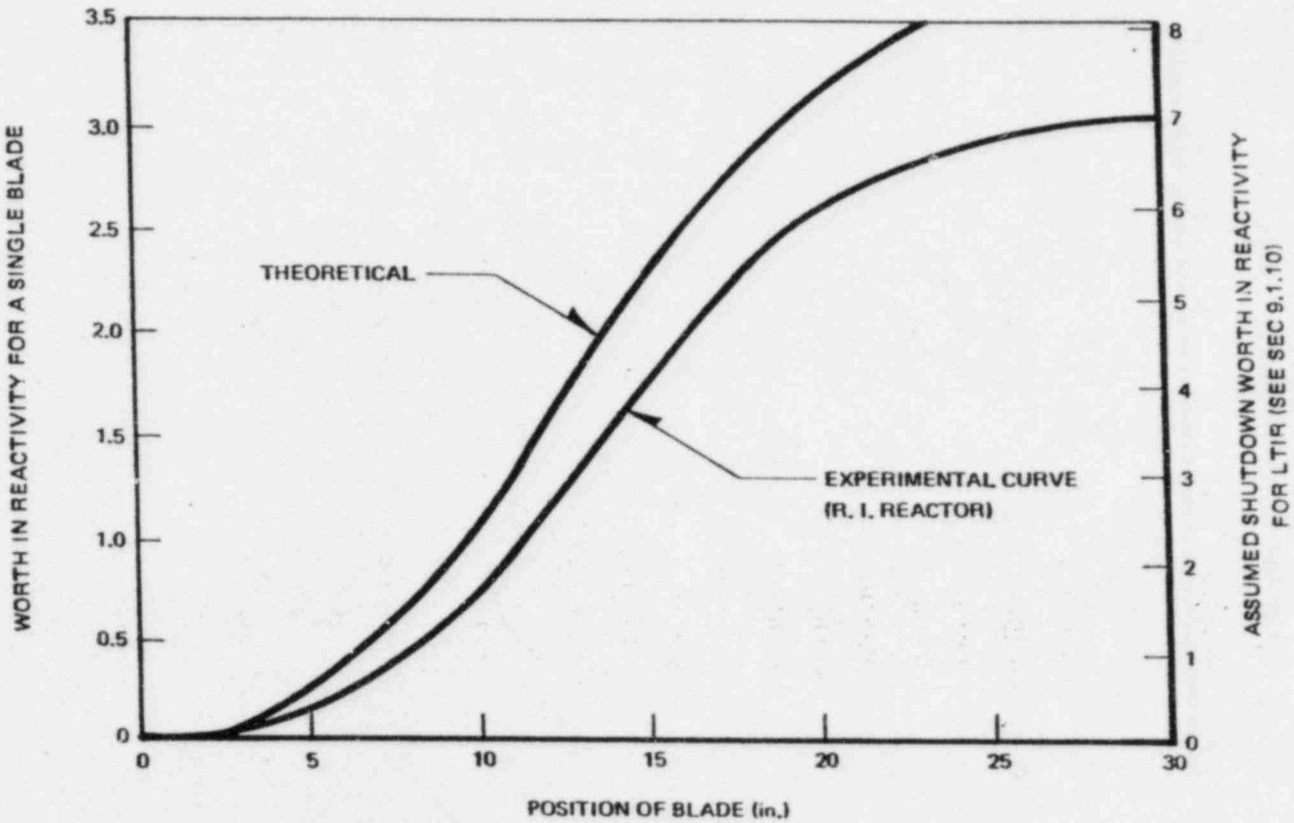


Figure 9.8 Blade Worth Versus Position

are the same as for the transient calculated by TER-6.

The results, based on the above conservative assumptions, are displayed in Figure 9.9 which shows that the power trace during the total transient after 0.5% Δ k/k insertion stays below 4 MW; this is below the Safety Limit as shown in Figure 9.1.

9.1.11 Withdrawal of a Control Blade

The consequences of the continuous withdrawal of a control blade starting while the reactor is being operated at steady-state power were investigated. It was assumed that reactivity was added at the continuous rate of 0.035% $\Delta k/k$/sec. This is a reasonably high rate of reactivity addition, being about 75% greater than the maximum rate predicted with the regulating rod (which has a maximum worth of $\rho < \beta$). The calculation was done using TER-6[16] and Figure 9.10 shows the power trace as a function of time.

If it is postulated that the withdrawal started at 1 MW, a scram would be initiated from the high flux channels 120% of power after 3 seconds and the reactor would be shut down by the control blades dropping and terminating the transient 1 second later. It is not necessary to follow the control blade insertion transient because the maximum power is less than 1.4 MW (power from Figure 9.10 at 4 seconds) which is well below the Safety Limit shown in Figure 9.1.

Analysis of the scram system shows that the single failure criterion is met of this postulated transient. There are two redundant channels for high flux indication and scram initiation. These channels are physically separated; the detectors are

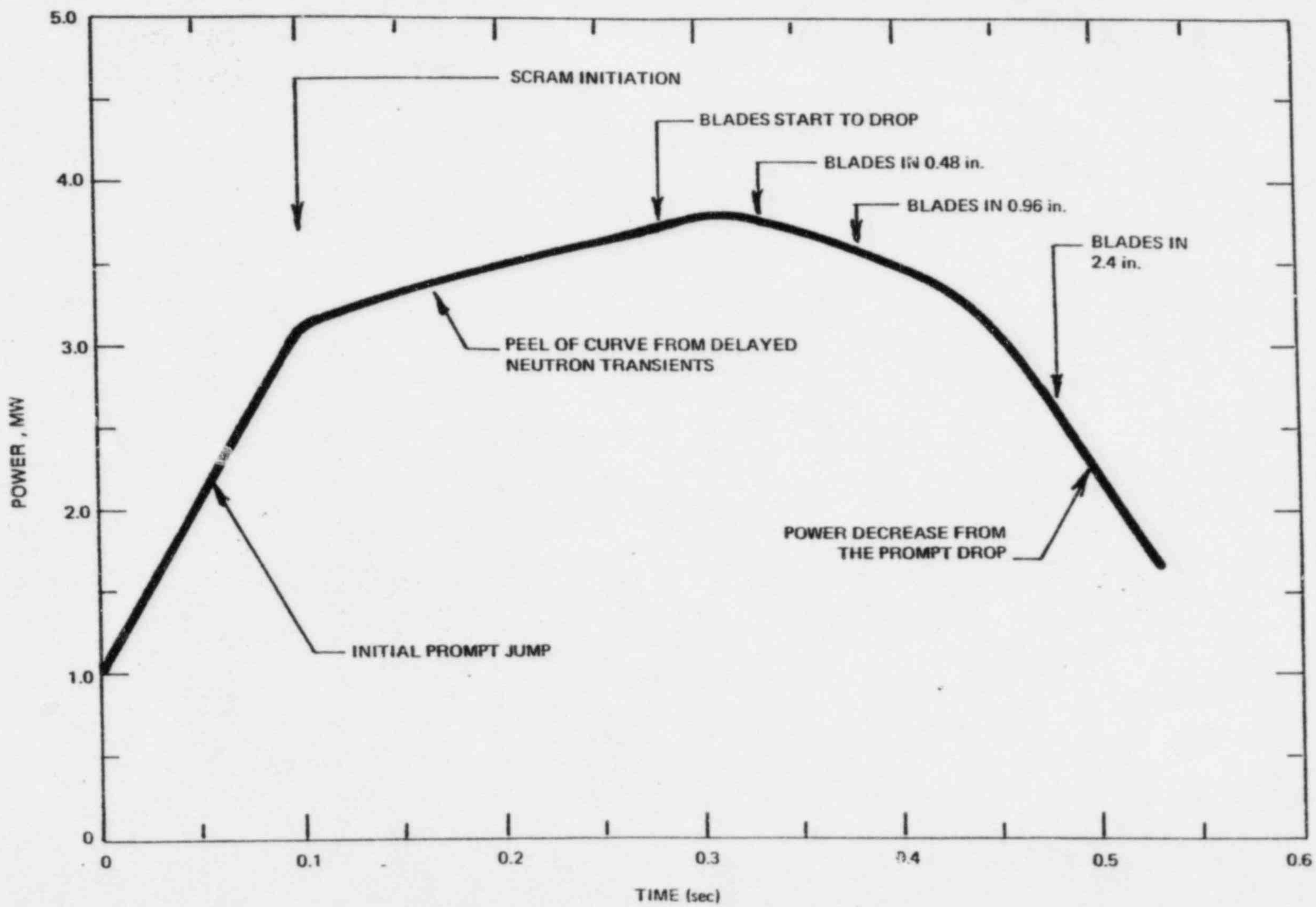


Figure 9.9 Power Versus Time After 0.5% Reactivity Step Followed By Scram

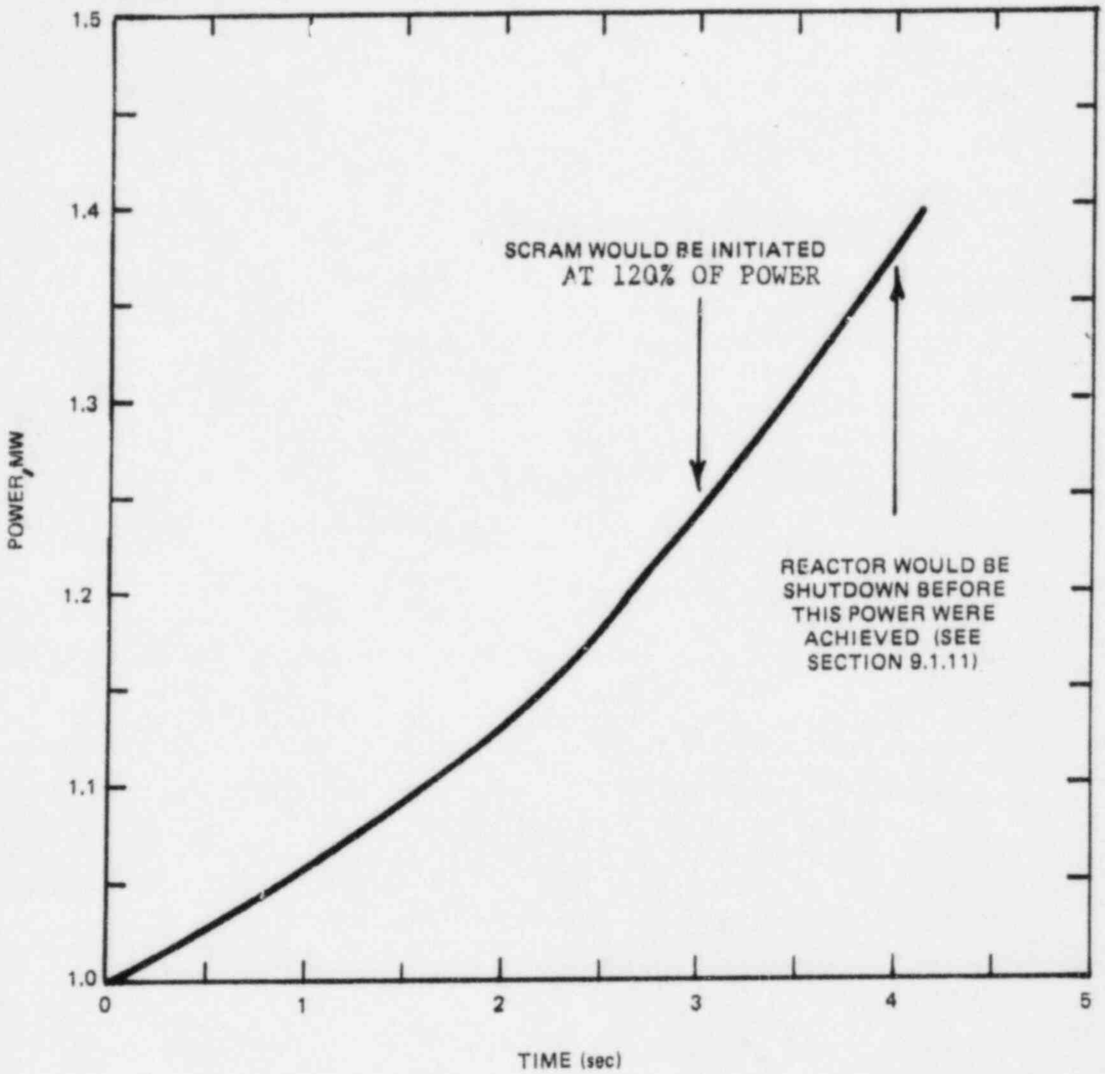


Figure 9.10 Power Versus Time for a 0.035%/sec Reactivity Ramp

independent and the channel wiring is separated through use of armored cable which covers the leads from the bridge up the bridge standpipe, over to the floor standpipe, down to the cable tray system, and up to the console. Furthermore, the period scram and manual scram are functional.

9.1.12 Cold Water Insertion

The secondary coolant loop is provided with an in-line throttle valve which may be opened or closed in response to a temperature sensor located in the primary system. The function of this valve is to regulate secondary flow to maintain constant core inlet temperature.

If it is postulated that this valve fails from a partly closed to an open position during winter operation when the secondary water is very cold in such a manner as to release a large slug of cold water into the secondary side of the heat exchanger, the primary water will be rapidly cooled. When the cold water enters the core, reactivity will be added by virtue of the negative temperature coefficient.

Even in the extreme case of postulation of a primary temperature change from 100°F to 32°F while the reactor is being operated at 1 MW steady power, which is far more severe than any credible situation, the reactivity added to the core is about 0.3% $\Delta k/k$.[14]

This is less reactivity than the 0.5% $\Delta k/k$ already considered in Section 9.1.10; the consequences of this situation could not be more severe than those of Paragraph 9.1.10, which lead to no violation of a Safety Limit.

9.1.13 <u>Safety Analysis of Cross Pool Flow During Forced Convection Operation</u> The forced convection flow path described in Section 4.2.4 of the FSAR involves the supply of water through the inlet flow or downcomer channel, from which it is forced down through the core and into the outlet flow or riser channel, and thence to the holdup tank, pump, and heat exhanger before return to the inlet. This flow path results in both a forcing and a suction type action above the core, and causes oscillation in the control blades which results in reactivity noise when operating with the blades partly inserted in the core to hold down the excess reactivity. Another flow mode, which involves a cross pool flow pattern, was investigated and found to eliminate the control blade oscillation.

The cross flow mode eliminates the water feed in the downcomer channel, and substitutes pool water as the supply by routing the return water from the heat exchanger into the cross pool inlet pipe. As explained in Section 4.2.4, a pair of pipes for supply and return of primary water is located at each end of the pool so that the reactor can be operated in the forced convection mode at either pool end. The selection of the proper pair of pipes is made by opening or closing gate valves in the pump room which are electrically interlocked with the reactor position for correct operating in the forced convection mode above 0.1 MW. (See Figure 4.11).

Trials were made at low power (so that the requirements on primary piping alignment in Technical Specification 3.3 was not violated) in which the piping alignment was changed to allow the return flow to enter the opposite end of the pool from the reactor location, and thus to allow the pool itself to be the feedwater source. This change resulted in operation with no detectable oscillation in the control blades.

The safety consideration items that are affected by the cross flow mode include, 1) any effect on the Loss of Primary Flow Transient described in Section 9.1.3 and the consequences on Limiting Safety System Settings discussed in Section 9.1.2.3.1, and, 2) the effect on the release of ^{41}Ar into the pool and into the air which was calculated in 7.3.5 of the FSAR as a consequence of ^{40}Ar being irradiated in the primary coolant.

Measurements have been made of the primary flow after loss of pump for the ULR Reactor with the reference 26 element core in place. These data were taken by connecting a fast recorder to the flow signal from the orifice flow sensor located in the primary return line after the heat exchanger (see Figure 4.11), and recording the flow as a function of time after the pump power was turned off. The results are shown in Fig. 9.11 and Fig. 9.12 appended for the original design flow pattern and for the cross flow pattern, respectively. There is no real difference in the two curves. The error bars shown in Fig. 9.11 represent the standard deviation in the data read from the recorder curves, and reflect the uncertainty in initial chart paper zero calibration. The uncertainties in Fig. 9.12 are within the circle boundaries shown. Neither of these curves reflect any systematic error.

Figure 9.3 shows the cross flow data normalized to percent of flow compares with the curves from the FSAR. Inspection shows that the measured ULR flow curve is similar in shape to the measured Rhode Island curve of Fig. 9.3, but that flow retention is better in the ULR case. Both of these curves show some flow retention from the pump coast down.

The Limiting Safety System Setting for forced convection operation is 1170 gpm as given in Technical Specification 2.2.1; taking this as the scram point, the control blades would be inserted 1 second later corresponding to 44% of the 1790 gpm or 790 gpm flow from Fig. 9.3. Thus the original analysis FSAR is extremely conservative, since the equivalent values from Fig. 9.3 are 38% of 1400 gpm or 535 gpm. No re-analysis is required, because the measured ULR case is more safe than the acceptable case presented in the FSAR. No change in Limiting Safety System Settings is required or desirable. This is true for either the design flow mode or cross flow mode of forced convection operation.

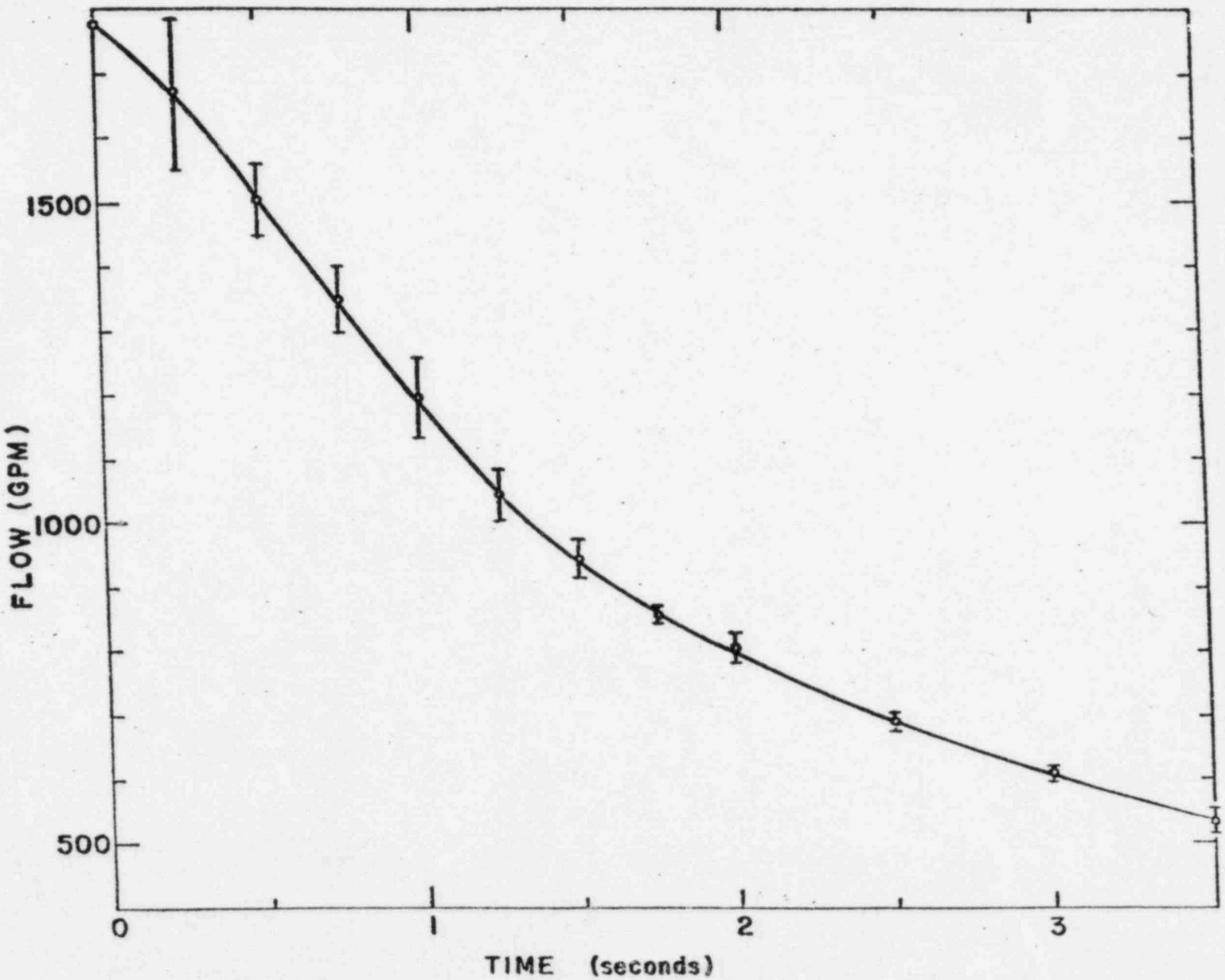


Figure 9.11 Primary Flow After Loss of Pump; Downcomer Flow Mode

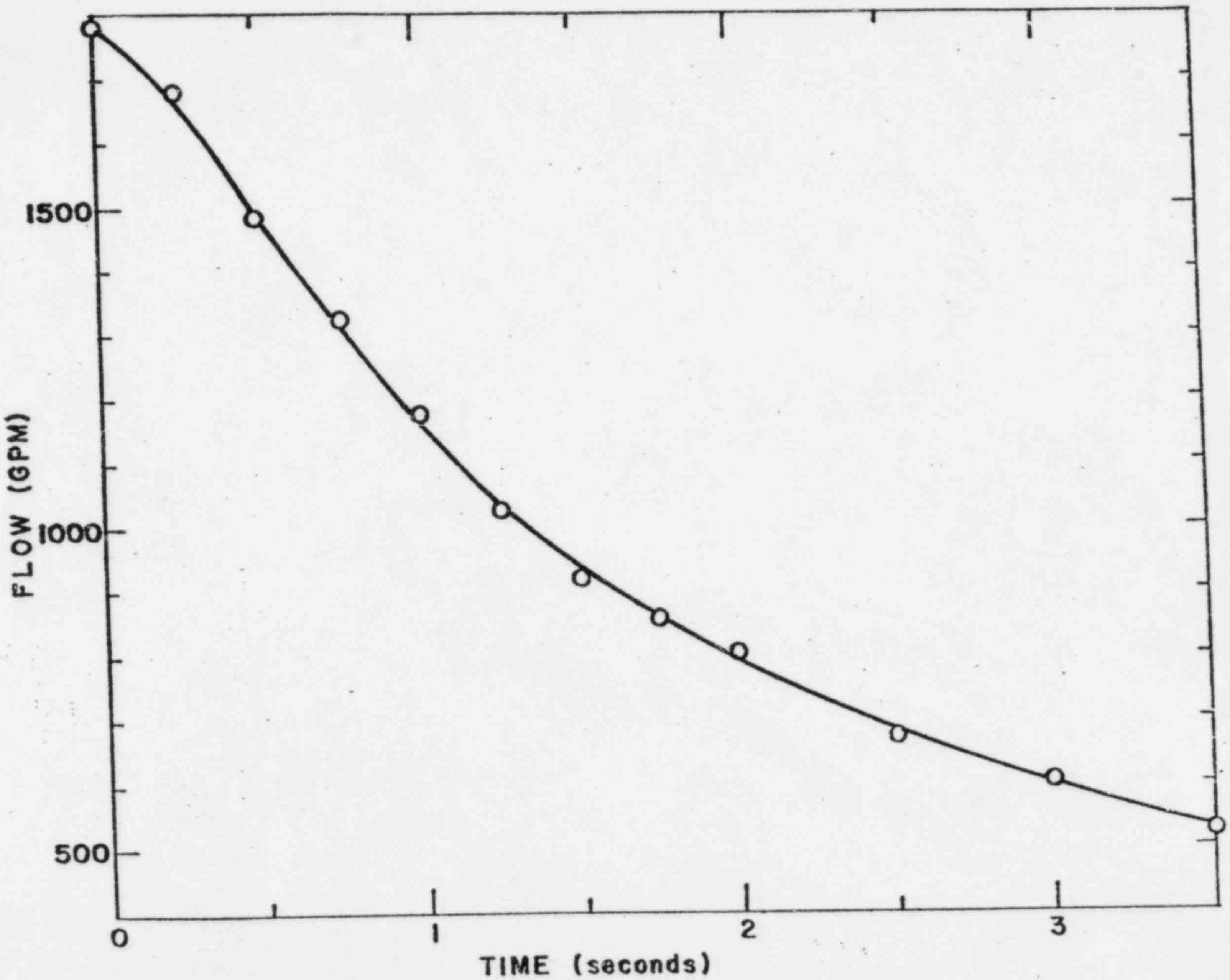


Figure 9.12 Primary Flow After Loss of Pump; Cross Pool Flow Mode

9.1.14 Power Failure

The reactor will scram in the event of power failure.

After a short delay which is required for the emergency generator to start and get to operating power, crucial information and process functions and H.P. monitors are supplied power (see Paragraphs 3.4.2, 4.4.2.6, etc.) so that the status of the reactor and associated equipment will be known. Thus, the reactor can be secured in an orderly fashion even if the external power failure persists.

9.1.15 Fire

The reactor building and supporting laboratories are designed to be fireproof, although a fire of limited magnitude is possible since it is not feasible to exclude every combustible material. Portable fire fighting equipment is available at every level, as well as manual alarm stations which activate Klaxon buzzers and send a signal to the Lowell Fire Department.

The Safety Committee of ULR has issued a guiding document covering the treatment of fires. In essence, if a fire can be controlled by local personnel and portable extinguishers, then direct action is to be taken to put out the fire, and no evacuation is necessary, although a complete report to the Lowell Fire Department is required. Common examples are small chemical fires, waste basket fires, and small electrical fires.

Judgment is generally the rule concerning response to fire in the reactor building; if the fire is saml1 enough to be handled by local treatment is effective, no reactor shutdown is necessary. Indeed, in some cases involving experiments and attendent personnel on levels other than the operations personnel become aware of it. If the fire appears to be other than small and local, the reactor must be shut down and secured, and fire alarm signal to the Lowell Fire Department activated. Subsequent action by operations personnel is again a matter of judgment, although a man must be detailed to meet Fire Department personnel.

The fire alarm system is common to the Pinanski Building and the reactor building. Thus, an alarm can be initiated outside the reactor which will set off the Klaxons inside the containment; therefore, the reactor must be shut down and secured if the Klaxon alarm sounds (except from known tests). If the fire is apparently outside the reactor building, evacuation of the reactor should be by a considered route, probably the truck door, in which case an operator must stand guard to prevent unauthorized entrance via the unlocked door. Regardless the Lowell Fire Department must be notified for record purposes.

9.1.16 Earthquakes

The reactor is fitted with a seismic device described in Chapter 4, which scrams the reactor upon detection of an earth tremor of significant magnitude (IV on the Modified Mercalli scale). Since the reactor pool is constructed of a large mass of reinforced concrete, the structure will vibrate as a unit. The frequency of vibration of the pool unit could be different from that of the building which might cause dislocation between the reactor and immediate surroundings. A resultant break in the primary coolant pipes would allow the pool to drain no lower than 12 feet above the core, due to the pipe location in the pool concrete and anti-siphon provision.

An earthquake of magnitude sufficient to dislocate the pool surroundings would be clearly recognized, and of course the reactor, if operating, would scram at the first significant tremor. The reactor would be secured, and not started up again until an examination has been made to assure that no damage had occurred to the reactor, or damage conditions, if any, were properly rectified.

9.1.17 Storm, Flood, Strike, or Riot

In the event of a sev. -e storm, flood, strike, or riot, the reactor will be secured and the air lock doors locked if there appears to be even a remote chance of danger in operating the reactor at the time. Weather reports are obtained and logged as a matter of record each operating day.

9.2 STANDBY SAFEGUARDS ANALYSIS

9.2.1 Estimation of Consequences of Fission Product Release

The probability of an accident taking place which might result in a significant release of fission products from the ULR core is extremely remote. The complete loss of water from the reactor pool, leaving the core cooled only by air, although certainly a serious situation from a protection point of view, was shown in Paragraph 9.1.5 to result in no loss of fuel cladding integrity for operation at one megawatt and, hence, no significant release of fission products.

For the plate type fuel utilized in the ULR, physical damage to an element is possible via pitting of plate cladding or inadvertant mechanical shock; for instance, scrathing of a plate surface in handling; such damage could lead to the escape of fission products. While the real magnitude of such damage is indeterminate, it is difficult to conceive of any situations or processes within the scope of reactor operations which could result in the release of a large fraction of the core fission product inventory.

In order to evaluate the consequences of fission product releases, it is assumed that one plate in an element (from the aspect of vulnerability to mechanical damage, an outside plate would be most likely involved) is damaged to such an extent that total cladding integrity is lost and that volatile fission products are completely available for release to the primary coolant.

The consequences of release of gaseous fission products to the primary coolant and from there to the pool from which escape to the containment air occurs are reviewed below. Release of air from the containment is assumed to occur at the fractional rate of 0.10 of the building volume per day.

9.2.1.1 <u>Dose Consequences of Release of Radioiodines from a Single Fuel Plate</u> For the purposes of estimation, the following assumes release of all fission product iodines in the form of elemental iodine which, in the amounts present, would readily dissolve in the primary coolant water and be transferred by diffusion to the pool volume from which they would escape to the containment air. In the first section of following calculation, complete mixing of the iodines in the pool volume is assumed, although no correction is applied for radioactive decay during mixing.

A conservative flux peaking factor of three is assumed to apply to the plate of the failing element. Operation at one megawatt with twenty-six elements and eighteen plates per element is assumed, and iodine activites are assumed present in saturation quantities.

The radioactive production and decay rate of the i^{th} radioiodine in the whole core is given by $\lambda_i N_i$ where:

$$\lambda_i N_i = (3.1 \times 10^{10} f \text{ sec}^{-1} \text{ watt}^{-1})(10^6 \text{ watts})(\gamma_i)$$

and where:

λ_i = radioactive decay constant for the i^{th} radioiodine

N_i = saturation number of atoms of the i^{th} radioiodine present

γ_i = fractional fission yield for the i^{th} radioiodine

Table 9.2 gives values of γ_i, $\lambda_i N_i$, λ_i, and N_i for the appropriate iodines. The total number of iodine atoms present is $\sum_i N_i = 1.22 \times 10^{21}$, whence the number of molecules of I_2 present is 6.1×10^{20}. The number of radioiodine molecules released from the single plate to the coolant and eventually to the pool is:

$$\text{Radioiodine molecules released} = 6.1\text{x}10^{20} \times \frac{1}{26} \times 3 \times \frac{1}{18} = 3.9\text{x}10^{18}$$

For a water volume of about 3×10^4 gallons in the stall end of the pool, the mole fraction, X_{H_2O}, of radioiodines in the pool water is:

$$X_{H_2O} = \left(\frac{3.9 \times 10^{18} \text{ molecules}}{6.02 \times 10^{23} \text{ molecules mole}^{-1}}\right) \Big/$$

$$(3 \times 10^4 \text{ gal})(3.8 \times 10^3 \text{ cm}^3 \text{ gal}^{-1})(1 \text{ gcm}^{-3}) \left(\frac{1 \text{ mole}}{18 \text{ g}}\right) = 1.0 \times 10^{-12}$$

The partial pressure of iodine in the air, assuming equilibrium conditions, P_{I_2} is approximately given by:

$$P_{I_2} = P_o X_{H_2O}$$

where P_o is the vapor pressure of pure iodine. (P_o = 1.58 mm of Hg has been calculated for an assumed pool temperature of 110°F from the equation, $\log_{10}Po = (-0.2185A/T) + B$, where Po is the vapor pressure in mm of mercury at temperature, T(°K), A is the molar heat of vaporization in cal $mole^{-1}$ (13,057 for I_2) and B is a constant (9.24 for I_2).) Substituting appropriate values yielded:

$$P_{I_2} = (1.58 \text{ mm})(1.0 \times 10^{-12}) = 1.58 \times 10^{-12} \text{ mm Hg}$$

whence the molar fraction of radioiodine in air, at equilibrium, X_{air}, is:

$$X_{air} = \frac{1.58 \times 10^{-12}}{0.76 \times 10^{3}} = 2.1 \times 10^{-15}$$

For a total building volume of 335,000 ft^3 the total number of moles of radioiodine present would then be M:

$$M = \frac{(2.1 \times 10^{-15})(3.35 \times 10^{5} \times 28.31)}{24.51 \text{ moles}^{-1}}$$

The number of moles of the i^{th} radioiodine, M_i, can be readily calculated from the relative atom populations determinable from the values of N_i, in Table 9.2. The activities, A_i, of the i^{th} isotope are then $A_i = \lambda_i(2\ M_i)(6.02 \times 10^2)\ sec^{-1}$, and these are given in Table 9.3.

The quantity S_i present at any time, t, after initial release of radioiodines to the building is related by the expression:

$$S_i = S_i^o\, e^{-\lambda_i t}$$

where λ_i is the radioactive constant for isotope, i, and where, for convenience, S_i, is expressed in units of rads to the thyroid (i.e., dose which would ensue if a person were breathing the radioactive iodine present) and S_i^o is the quantity present at time, zero. It is to be noted that although radioactivity is removed from the building by leakage, it is assumed that iodines are continuously released from the pool to maintain equilibrium between the water and containment air, and the amount of radioactivity in the building is decreased only by radioactive decay. Under this assumption it is also clear that concentrations of iodines in a leakage cloud outside the building (allowing for no natural dissipation processes) would be the same as those in the building. The concentration, C_i, in units of rads to the thyroid per cm^3 of air is given by:

$$C_i = \frac{S_i}{V_B}$$

where V_B is the building volume ($9.5 \times 10^9\ cm^3$).

The integral dose to the thyroid of an individual in the

TABLE 9.2

RADIOIODINE ATOM POPULATION PRESENT AT SATURATION IN THE CORE

Iodine Isotope	Y_i	$\lambda_i N_i$	$\lambda_i (sec^{-1})$	N_i (atoms)
131	0.029	9.0×10^{14}	9.96×10^{-7}	9.04×10^{20}
132	0.043	1.33×10^{15}	8.26×10^{-5}	0.16×10^{20}
133	0.065	2.02×10^{15}	9.20×10^{-6}	2.20×10^{20}
134	0.080	2.48×10^{15}	2.20×10^{-4}	0.11×10^{20}
135	0.064	1.98×10^{15}	2.86×10^{-5}	0.69×10^{20}
				$\sum_i N_i = 1.22 \times 10^{21}$

TABLE 9.3

ACTIVITIES OF RADIOIODINE ISOTOPES IN CONTAINMENT AT TIME OF RELEASE FROM POOL

Iodine Isotope	$M_i = \frac{N_i}{\Sigma N_i} (8.1 \times 10^{-10})$	$A_i = \frac{\lambda_i (2M_i)(6 \times 10^{23})}{3.7 \times 10^4} (\mu C_i)$
131	6.0×10^{-10}	1.94×10^4
132	1.06×10^{-11}	2.85×10^4
133	1.46×10^{-10}	4.37×10^4
134	7.3×10^{-12}	5.22×10^4
135	4.54×10^{-11}	4.24×10^4

containment received as a consequence of breathing this concentration over time period, T, is D_i

$$D_i = \int_0^T BC_i dt = \frac{BS_i^0}{V_B} \int_0^T e^{-\lambda_i t} dt \qquad (9.2.1.1)$$

Integrating and evaluating between the limits yields

$$D_i = \frac{BS_i^0}{V_B \lambda_i} \left(1 - e^{-\lambda_i T}\right) \qquad (9.2.1.2)$$

where B is the breathing rate, taken as 2×10^7 cm^3 day^{-1}.

Values of S_i^o are calulated on the assumption of release from a single fuel plate as described above and are given in Table 9.4. Integrated thyroid dosed calculated according to Equation 9.2.1.2 are presented in Table 9.5 , along with the total thyroid doses for various exposure times.

It can be seen from the values in Table 9.5 that nearly one week of continuous exposure to released radioiodine is required, under this model, to commit the thyroid to a 300 Rem dose. For an individual outside the containment and breathing the air in an expanding leakage cloud at ground level, commitment of such a dose would require unrealistically severe restrictions on cloud dispersion.

It is difficult to conceive of any combination of meteorological conditions and personnel occupancy which could result in a real dose of this magnitude even over a much longer period. Any measurable wind speed would produce mixing and

TABLE 9.4

SOURCE DATA FOR THYROID DOSE CALCULATION

Iodine Isotope	Curies Available	D_i/A_i (rads Ci^{-1})	S_i^0 (rads)	λ_i (day^{-1})
131	1.94×10^{-2}	1.48×10^6	2.88×10^4	0.0861
132	2.85×10^{-2}	5.35×10^4	1.53×10^3	7.14
133	4.37×10^{-2}	4.00×10^5	1.75×10^4	0.795
134	5.22×10^{-2}	2.50×10^4	1.31×10^3	19.0
135	4.24×10^{-2}	1.24×10^5	5.27×10^3	2.47

TABLE 9.5

THYROID DOSE COMMITMENT FROM BREATHING RELEASED RADIOIODINES (RADS)

Iodine Isotope	Exposure Time 2 h	1 day	2 days	4 days	7 days	30 days	∞
131	5.04	58.09	111.40	205.17	318.77	651.00	704.2
132	0.20	0.45	0.45	0.45	0.45	0.45	0.45
133	2.97	25.41	36.89	44.41	46.16	46.34	46.34
134	0.12	0.15	0.15	0.15	0.15	0.15	0.15
135	0.08	0.41	0.45	0.45	0.45	0.45	0.45
Total	8.4	84.5	149.3	250.6	366.0	698.4	751.6

dissipation thus drastically reducing the iodine exposure. For instance, if the effluent air at 7 m^3sec^{-1} from the stack were to mix continuously with the cloud volume, a dilution by about a factor of 650 would result and reduce the infinite thyroid dose to about one rad. Similarly, a wind speed of 0.1 miles per hour (2.7 meter min^{-1}) acting on a semispherical cloud of 10 meter radius (this would be the size after approximately two days of leakage), aside form dissipating the cloud's contents, would present an effective activity removal rate and consequent dose reduction equilvalent to ~22% per minute.

Similarly, natural convective air currents and natural gaseous diffusion would dramatically decrease the effective cloud concentration and hence the integrated thyroid dose.

The earlier assumptions relative to release of iodines are very conservative and likely yield considerably higher iodine release than would actually occur in any accident situation, in particular:

(1) that no decay is assumed in the initial mixing and release from the pool;

(2) that all iodines released for the core are in molecular form;

(3) that no plateout of iodines in the containment building is assumed;

(4) that instantaneous equilbirium of iodines in the containment with respect to pool concentrations is assumed to prevail as containment air leaks from the building.

9.2.1.2 Estimation of Whole Body Gamma Ray Exposure Consequences of Release of Gaseous Radioactivity from a Single Fuel Plate In the following calculations the same assumptions of release relative to the radioiodines apply as in the previous paragraph and, in addition, it is assumed that all of the noble gases (those assumed in TID 14844) escape from the failed plate to the pool and from the pool to the containment. No allowance for decay during diffusion from the pool is made.

Estimations of whole body gamma doses are made on the basis of the submersion dose principle; i.e., the concentration of a particular nuclide at a particular time is compared to the calculated submersion dose MPC for that nuclide, and the resultant dose is adjusted for the fact that the semispherical cloud in which the activity is contained is not infinite in size. MPCs for infinite cloud gamma submersion doses are cal´ ˈated according to ICRP recommendations:

$$(MPC)_i = \frac{2.6 \times 10^{-6}}{\Sigma E_i} \ (\mu Ci\ cm^{-3})$$

where $(MPC)_i$ is for an occupational situation (i.e., exposure to $(MPC)_i$ results in 2.5 mrem h^{-1} whole body dose) and where $\sum E_i$ is taken as the total gamma energy emitted per disintegration of radionuclide, i. Table 9.6 contains pertinent data for the nuclide of major concern used in the calculations. Activities, λ_{iNi}, are for releases of the noble gases from a single plate as in the previous paragraph. Values of A_i for radioiodines are as given in Table 9.3.

TABLE 9.´

DATA USED IN ESTIMATION OF SUBMERSION CLOUD GAMMA DOSES

Nuclide	λ_i (h^{-1})	γ_i (Fission Yield)	E_i* (MeV d^{-1})	$(MPC)_i$ ($\mu Ci\ cm^{-3}$)	$\lambda_i N_i = (3.1x10^{10})(10^6)(\gamma_i)\left(\frac{3}{18x26}\right)(sec^{-1})$	$A_i^0 = \frac{\lambda_i N_i}{3.7x10^4}(\mu Ci)$
I-131	0.0035	0.029	0.4	6.5×10^{-6}		1.94×10^4
I-132	0.2888	0.043	2.12	1.2×10^{-6}		2.85×10^4
I-133	0.0333	0.065	0.55	4.7×10^{-6}		4.37×10^4
I-134	0.7920	0.080	1.25	2.1×10^{-6}		5.22×10^4
I-135	0.1037	0.064	1.5	1.7×10^{-6}		4.24×10^4
Kr-85m	0.1589	0.013	0.19	1.4×10^{-5}	2.59×10^{12}	7.0×10^7
Kr-87	0.5331	0.025	0.63	4.1×10^{-6}	4.97×10^{12}	1.34×10^8
Kr-88	0.2502	0.036	2.18	1.2×10^{-6}	7.18×10^{12}	1.94×10^8
Xe-131m	0.0024	0.029	0.002	1.3×10^{-3}	5.77×10^{12}	1.56×10^8
Xe-133m	0.0126	0.065	0.006	4.3×10^{-4}	1.29×10^{13}	3.48×10^8
Xe-133	0.0055	0.065	0.08	3.2×10^{-5}	1.29×10^{13}	3.48×10^8
Xe-135m	2.6637	0.064	0.15	1.7×10^{-5}	1.27×10^{13}	3.43×10^8
Xe-135	0.0760	0.064	0.24	1.1×10^{-5}	1.27×10^{13}	3.43×10^8

*This is the total gamma energy emitted per disintegration of the radionuclide as distinct from the average gamma energy per disintegration. Values of E were obtained by dividing the core source strength at 1 MW (MeV per sec as given in Table IV of TID-14844) by the disintegration rate of the particular radionuclide at saturation activity in the core.

The instantaneous whole body gamma dose rate, X_i, resulting from exposure to nuclide, i, is given by:

$$X_i = f_i \frac{C_i}{(MPC)_i} (2.5 \times 10^{-3})(\text{rem h}^{-1}) \qquad (9.2.1.3)$$

where f_i, the fraction of the infinite semispherical cloud dose rate, is a weighting factor to account for the finite size of the cloud and has been found to be well represented by $f_i = \mu_{en(air)i}R$* for values of f_i less than 0.4), where $\mu_{en(air)i}$ is the linear energy absorption coefficient for the gamma ray of interest, and R is the radius of the cloud; C_i is the instantaneous concentration of nuclide, i, in the cloud volume, V. The approximation for f is a conservative value for gamma energies greater than a few tenths of an MeV in that larger values of $\mu_{en(air)i}$ R would overestimate the fraction of the semispherical infinite cloud doses. For low energies, the value $\mu_{en(air)i}R$ is an underestimate of f, but for the small cloud sizes considered here, the effect will not be significant.

Integral whole body gamma doses have been calculated for exposure in the containment building according to the following expressions:

$$D_i = \int_0^T \frac{2.5\text{x}10^{-3}\mu_{en}R_B A_i}{V_B(MPC)_i}\,dt = \frac{2.5\text{x}10^{-3}\mu_{en}R_B A_i^0}{V_B(MPC)_i}\int_0^T e^{-(\lambda_i+k)t}\,dt \qquad (9.2.1.4)$$

*See Table 9.9 and description thereon.

where D_i is the integral dose (rems) over the exposure period, T; V_B is the building volume (9.5×10^9 cm^3); R_B is the effective radius of the containment building (assuming a semispherical shape for purposes of calculation), taken as 16.56 meters; λ_i is the radioactive decay constant (h^{-1}) for nuclide, i; and k is the building volume leakage constant (0.00417 h^{-1}).

Integration of Equation 9.2.1.4 between 0 and T yields:

$$D_i = \frac{2.5\times10^{-3}\mu_{en}R_B A_i^0}{(\lambda_i+k)V_B(MPC)_i}\left(1-e^{-(\lambda_i+k)T}\right) \qquad (9.2.1.5)$$

Table 9.7 gives values of average gamma energy per disintegration of nuclide, i, and values of $\mu_{en(air)i}$ for this energy. Table 9.8 shows integral whole body gamma doses for various exposure times.

It is interesting to note the whole body gamma dose for a two-hour exposure is only 5.3 rems and the maximum dose which is reached, for all practical purposes within one month, is about 29 rems.

Integral dose for exposures occurring to an individual submersed in an expanding cloud outside of the containment have not been calculated, since it can readily be seen that these will not exceed doses shown in Table 9.8. For instance, after a two-hour period the cloud external to the building would be 0.0083

TABLE 9.7

VALUES OF AVERAGE GAMMA ENERGY AND ENERGY ABSORPTION COEFFICIENTS

Radionuclide	E_{avg} (MeV)	$\mu_{(en)(air)i}$ ($cm^{-1} \times 10^5$)
I-131	0.4	3.9
I-132	0.8	3.7
I-133	0.55	3.9
I-134	1.3	3.4
I-135	1.5	3.3
Kr-85m	0.20	3.5
Kr-87	2.0	3.0
Kr-88	2.0	3.0
Xe-131m	0.16	3.3
Xe-133m	0.23	3.6
Xe-133	0.08	3.2
Xe-135m	0.52	3.9
Xe-135	0.25	3.6

TABLE 9.8

INTEGRAL WHOLE BODY GAMMA RAY DOSES IN THE CONTAINMENT (REMS)

	Time				
Radionuclide	2 h	24 h	48 h	168 h	720 h (∞)
I-131	0.0001	0.0011	0.0020	0.0049	0.0065
I-132	0.0006	0.0013	0.0013	0.0013	0.0013
I-133	0.0000	0.0025	0.0035	0.0042	0.0042
I-134	0.0004	0.0005	0.0005	0.0005	0.0005
I-135	0.0000	0.0031	0.0033	0.0033	0.0033
Kr-85m	0.1300	0.4582	0.4675	0.4675	0.4675
Kr-87	0.524	0.756	0.795	0.795	0.795
Kr-88	3.315	8.292	8.309	8.309	8.309
Xe-131m	0.0035	0.0383	0.0711	0.1756	0.2600
Xe-133m	0.0250	0.2513	0.4192	0.7138	0.7138
Xe-133	0.3002	3.2533	5.8309	12.6044	12.6044
Xe-135m	0.1279	0.1280	0.1280	0.1280	0.1280
Xe-135	0.9030	5.2103	6.1011	6.1011	6.1011
Total	5.330	18.397	22.133	29.309	29.395

TABLE 9.9*

COMPARISON OF THE QUANTITY $\mu_{en(air)}R$ WITH f, THE FRACTION

OF THE SEMISPHERICAL INFINITE CLOUD DOSE

Energy (MeV) / Cloud Radius	10 m f	10 m $\mu_{en(air)}R$	20 m f	20 m $\mu_{en(air)}R$	50 m f	50 m $\mu_{en(air)}R$	100 m f	100 m $\mu_{en(air)}R$
0.1	0.033	0.030	0.071	0.061	0.199	0.151	0.416	0.303
0.2	0.035	0.035	0.072	0.069	0.186	0.173	0.370	0.347
0.5	0.038	0.038	0.076	0.077	0.188	0.191	0.358	0.383
1.0	0.036	0.036	0.071	0.072	0.174	0.180	0.328	0.360
2.0	0.030	0.031	0.060	0.061	0.146	0.153	0.278	0.304

*Values of f, the fraction of the infinite semispherical cloud dose, indicative of finite clouds of gamma emitting radioactivity, have been calculated as outlined in the September 8, 1970 submission to the Commission entitled, "Evaluation of the Environmental Significance of the Projected 5 MW ^{41}AR Release Rate of 400 μCi sec^{-1} from the Lowell Technological Institute Reactor Facility."

It was there shown that f can be given by:

$$f = \frac{1-e^{-\mu R} + \dfrac{a}{(b-1)^2}\left\{1 - \left[1 + (1-b)\mu R\right] e^{-(1-b)\mu R}\right\}}{1 + \dfrac{a}{(b-1)^2}}$$

where R is the radius of the cloud and a and b are constants which appear in Berger's formulation for the buildup factor, $B = 1 + a\mu r\, e^{b\mu r}$, and have been determined as in the above cited document. Values of B, μ and μ_{en} used were obtained, also as in the document cited, from Jaeger's, "Engineering Compendium on Radiation Shielding," Vol. 1 (Springer - Verlag, New York, 1968).

of the building volume and would have a radius of less than 3.5 m and would contain ~0.0083 of the activity present in the building at that time; the net result is that after two hours the dose rate in the cloud outside the building would be lower by about a factor of five than that inside the containment, the radius of the cloud being the controlling factor. Integral doses would be reduced in a similar manner. Similarly, after two days the radius of the cloud outside the building is ~9.7 m, the cloud contains ~18% of the total activity present at that time, and the dose rate is approximately half of that in the building. After ten days, the cloud outside the building has achieved the same volume as represented by the containment but contains only 63% of the total activity present at the time. The integral dose at the end of ten days for exposure outside the building would be considerably less than 0.63 of the ten-day integral dose in the building because of the smaller cloud volume during the exposure interval. At the end of a month's exposure outside the containment, the cloud's radius would have increased to about 24 meters about 50% greater than the effective radius of the containment, but still insufficient to result in doses greater than those calculated in Table 9.8. Beyond this time, the major activities have decayed to such an extent that virtually no increase in integral dose ensues from further exposure.

Again, it should be noted that meteorological stability requirements for stationary cloud to remain intact over a period of even a few days are probably impossible to sustain, and real doses resultant from submersion in a leakage cloud from the containment would be drastically reduced.

9.2.2 Effects of Cross Flow on ^{41}Ar Release The ten inch primary coolant line which, in the cross-flow mode, delivers primary water to the pool volume enters the pool approximately 12 feet below the surface. There is considerable mixing of water from the line with the pool water volume, allowing the ^{41}Ar entering the pool from the ten inch line to experience a significant delay in terms of release to the air above the pool.

If it is assumed, in the most favorable case, that no appreciable fraction of the ^{41}Ar entering the pool via the ten inch line is released to the air then only the return line from the 40 gpm cleanup deminerlaizer represents a source of ^{41}Ar. Under these assumptions ^{41}Ar would build up in the pool and primary line volumes and be removed to the air at the rate it is transferred to the pool surface from the demineralizer loop.

The number of ^{40}Ar atoms available in the core volume for irradiation is $N_T = 4.7 \times 10^{20}$ (FSAR p. 7-10). The number of ^{41}Ar atoms, N_p in the total pool plus primary line volume at time, t post startup can be estimated by solving the differential equation,

$$\frac{dN_p}{dt} = N_T \, \sigma\phi - \lambda_{eff} \, t$$

where $\lambda_{eff} = \lambda_{rad} + k$ where λ_{rad} is the radioactive constant for ^{41}Ar and where k is the instantaneous fractional removal rate, via the demineralizer, of ^{41}Ar from the total pool plus primary line volume.

$$\lambda_{eff} = 1.1 \times 10^{-4} \text{ sec}^{-1} + \frac{(40 \text{ gpm}/60 \text{ sec m}^{-1})}{80{,}000 \text{ g}} \cong 1.1 \times 10^{-4} \text{ sec}^{-1}$$

$$N_p = \frac{N_T \sigma\phi}{\lambda_{eff}} (1-e^{-\lambda_{eff}t})$$

At saturation $N_{p(sat)} = \dfrac{N_T \sigma\phi}{\lambda_{eff}}$

and substituting appropriate values yields

$$N_{p(sat)} = \frac{(4.7 \times 10^{20})(0.62 \times 10^{-24})(10^{13})}{1.1 \times 10^{-4}} = 2.65 \times 10^{13} \text{ atoms}$$

The release rate via the demineralizer to the pool and the air is kN_p:

$$kN_p = (2.65 \times 10^{13})(8.3 \times 10^{-6}) \text{ atoms sec}^{-1} = 2.2 \times 10^{8} \text{ atoms sec}^{-1}$$

and the activity release rate is $\lambda_{rad}\ kN_p/3.7 \times 10^4$ dps μC_i^{-1}.

$$\frac{\lambda_{rad}\ kN_p}{3.7 \times 10^4 \text{ dps}/\mu C_i} = \frac{(2.2 \times 10^8 \text{ atoms sec}^{-1})(1.1 \times 10^{-4} \text{ sec}^{-1})}{3.7 \times 10^4 \text{ dps } \mu C_i^{-1}} = 0.65\ \mu C_i/\text{sec}$$

In the worst case assuming all of the ^{41}Ar entering the pool via both the demineralizer and the ten inch line was released to the air, the release rate would be

$$\frac{\lambda_{rad}\ N_T\ \sigma\phi}{3.7 \times 10^4 \text{ dps } \mu C_i^{-1}} = 8.7\ \mu C_i \text{ sec}^{-1}$$

In reality the ^{41}Ar release rate might lie between the two extreme values calculated. A factor of two reduction in the release to the air of ^{41}Ar entering the pool 12 feet below the surface is likely a conservative estimate. In this case the ^{41}Ar release rate as a result of the input from the ten inch line and the demineralizer would be approximately $\frac{8.7}{2} = 4.4$ $\mu C_i\ sec^{-1}$ at one megawatt.

It is likely, then, based on the above estimates that operation in the cross flow coolant mode will produce slightly lower argon release rates than would be realized form the "closed" primary loop operational mode.

The FSAR makes mention of the essentially closed nature of the primary system notably on pages 4-18, 4-19, and 7-19; this description is not appropriate for the cross pool mode of forced convection operation.

On page 9-15 of the FSAR, in Section 9.1.3 under the discussion of loss of flow, it is stated that the "...gates in the flow channels above the core fall open and permit natural convection cooling...". In the cross flow mode the downcomer gate will be open, so only the riser gate will open with loss of primary flow.

On page 9-30 of the FSAR, in Section 9.1.8, Case II the discussion of events occurring as a result of the downcomer gate opening are not appropriate for the cross flow model.

The case of cold water insertion considered in Section 9.1.12 of the FSAR dose not apply for the cross flow mode of operation, since any cold water generated in the heat exchanger would be warmed by the pool.

The cross flow mode of forced convection operation of the ULR dose not pose any safety consideration more severe than has been already covered in the FSAR. In some aspects, this mode appears to be more desirable. No change in Safety Limits of Limiting

Safety system setting is required.

9.2.3 Containment Integrity with Overpressure

9.2.3.1 Intentional Overexposure The containment building is a welded steel shell designed to be adequate for an internal gage pressure of 2.0 psi. (It was tested at an internal test pressure of 2.5 psig.) It is, of course, possible intentionally to achieve this magnitude of overpressure by sealing the building and pumping air in from external compressors, a technique that is used when leak-testing the building under pressure. It is a much more difficult task to postulate an accidental situation which would lead to a 2 psi overpressure.

9.2.3.2 Accidental Overpressure A malfunction of ventilation system interlocks can be assumed which would result in the closure of the exhaust fan while leaving the supply fan and valve open. Characteristic curves for the supply fan (i.e., AC-2 of Figure 3.5) indicate a static head pressure capability of 0.19 psi. Such an overpressure is a small fraction of the design specification, and poses no problem with containment integrity.

It is informative to consider the building overpressure achievable from vaporization of the pool water as a result of a nuclear excursion. The most conservative point of view is to assume that the entire thermal output results in the formation of steam, with all the heat thus assigned to latent heat and none to sensible heat, so that building temperature is constant. The air volume of the building is 335,000 ft^3; if the temperature is taken as 78°F, then 1.32×10^4 moles of water must be vaporized to achieve an overpressure of 0.5 psi, and this requires an energy equivalent to an excursion of 538 MW sec. This is nearly four times the 135 MW sec achieved in the Borax experiments,[16, 17, 18, 19] is which some 4% excess k was intentially added mearly as a step function by rapid control rod withdrawal, resulting in a minimum period of 2.6 msec, and a maximum power of 1.9 x 10^{10}w. An excursion of four times this magnitude with the ULR Reactor is extremely improbable, so that unacceptable overpressure by this mechanism is not credible.

9.2.3.3 <u>Effects of Low Atmospheric Pressure</u> Low barometric pressures are associated with storm conditions, and are extreme in the centers of tropical hurricanes. Pressures of the order of 28.0 inches of mercury are commonly found in the eyes of hurricanes, and 27.0 inches has been recorded.

Consider the following conservative case, in which the

diameter of eye	= 30 miles,
pressure at edge of eye	= 29.0 inches Hg,
pressure at center	= 27.0 inches Hg,
drift velocity	= 15 mile h^{-1},

If the eye passed directly over the reactor, one hour would be required for the barometric pressure to drop by two inches Hg, corresponding to a linear $\Delta P/\Delta T$ = 0.983 lb-in.$^{-2}$-h^{-1}. If the further assumption is made that before the approach of the hurricane, the reactor building went into a condition of containment with internal pressure at approximately 1 atmosphere. then the internal overpressure would be 3 inches of Hg, or 1.47 psi. Even if the containment condition were associated with 0.5 psi water vapor overpressure considered in paragraph 9.2.2.2, the total overpressure is 1.47 + 0.5 = 1.97 psi, which is within the 2.0 psi design pressure.

These considerations apply only when the containment building is sealed. With the containment valves open, the excess internal pressure would be relieved through the stack. The stack is designed for a pressure drop of less than 0.25 inches water column when exhausting at 15,000 ft^3-min^{-1}. Since this low pressure driving force is capable of exhausting such a large volume of air, the application of the much larger pressure drop (on the order of inches of Hg) postulated for the hurricane would lead to a rapid equalization with the valves open.

9.2.4 Containment Integrity with Underpressure

9.2.4.1 Intentional Underpressure The containment ventilation exhaust blower (EF-12 of Figure 3.5) is set to exhaust air at 15,000 ft^3-min^{-1} while the supply blower (AC-2 of Figure 3.5) is set to supply air at 14,500 ft^3-min^{-1}. The imbalance between exhaust and supply results in a slight negative pressure (about 0.1 inch water column) in the containment building, compared to external pressure. The object here is to have "in" leakage rather than "out" leakage through the containment during normal operation so that all building exhaust is through the stack.

9.2.4.2 Accidental Underpressure It is possible to conceive of a malfunction of the ventilation system interlocks that would result in the closure of the supply air valve while leaving the exhaust fan and exhaust valves open. Characteristic curves for the exhaust fan (EF-12 of Figure 3.5) indicate a static head pressure capability of 0.13 psi. The six-inch diameter vacuum breaker opens at about 0.5 inches water column, and thus this intake of air would tend to reduce the underpressure of the building, so that the containment design for 0.2 psi external pressure load is adequate.

9.2.4.3. Effects of Atmospheric Pressure Changes The severe hurricane considered in paragraph 9.2.3.3 moved at such a velocity that the pressure decrease rate as the eye approached was 2 in. Hg h^{-1}. If an accident resulting in no (or slight) overpressure but requiring containment is postulated just before the hurricane, as the hurricane approaches and diminished external pressure the building is effectively becoming pressurized. At 0.25 inches water column building overpressure, the emergency exhaust system starts to remove air from the building (see Paragraph 3.4.2). If the emergency exhaust blower is able to remove the overpressure in the containment building as the hurricane approaches, the rate of exhaust must be

$$\left(\frac{2\ \text{in. Hg h}^{-1}}{29.9\ \text{in. Hg atm}^{-1}}\right)\left(\frac{335000\ \text{ft}^3\ \text{atm}^{-1}}{60\ \text{min h}^{-1}}\right) = 373\ \text{ft}^3\ \text{min}^{-1}$$

of air through the blower (rated at 320 $ft^3\ min^{-1}$) to decrease the internal pressure at the same rate the external pressure is being lowered by the hurricane. No building overpressure results, but as the hurricane recedes and barometric pressure increases, the containment building will become underpressurized, and this must be relieved by the vacuum breaker.

If the hurricane is symmetrical and of constant linear velocity, the barometric pressure will increase at 2 in. Hg h^{-1}, and air must be supplied at this rate throughf the 6-inch diameter vacuum breaker pipe to equilibrate. This corresponds to an air velocity through the pipes of

$$\frac{(373\ \mathrm{ft}^3\ \mathrm{min}^{-1})(144\ \mathrm{in.}^2\ \mathrm{ft}^{-2})}{\pi\ (3)^2\ \mathrm{in.}^2} = 1895\ \mathrm{ft\ min}^{-1},$$

or about 22 miles per hour. This is not an excessive requirement.

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10.0 CONDUCT OF OPERATIONS

10.1 ORGANIZATIONAL STRUCTURE

10.1.1 Corporate Organization

University of Lowell is an educational institution organized by the Commonwealth of Massachusetts and administered through a Board of Trustees appointed by the Governor of the Commonwealth. The University structure is shown in Chart 10.1.

10.1.1.1 Contractors The ULR reactor was designed, manufactured, and installed by the General Electric Company. The building which houses the reactor was designed by the architectural firm of W. Chester Brown Associates and built by the Harvey Construction Company, Inc. The welded steel containment shell was furnished and erected by the Chicago Bridge & Iron Company.

10.1.2 Operating Organization

The reactor is operated under a U.S. Nuclear Regulatory Commission license and is controlled administratively by the Radiation Laboratory which, under the Director, is organized as shown in Chart 10.2.

10.1.2.1 Reactor Supervisor The Reactor Supervisor is responsible to the Director of the Radiation Laboratory for the operation of the reactor. The supervisor has direct responsibility for all activities in or about the reactor facility which may affect reactor operations, or, in conjunction with the Radiation Safety Officer, involve radiation hazards. This includes the control of reactor fuel, maintenance of the reactor and equipment, refueling the reactor, compliance with USNRC or other applicable regulations, the status of auxiliary support and safety equipment, the training and retraining of operations personnel, and the safe conduct of all phases of reactor operations.

10.1.2.2 Chief Reactor Operator The Reactor Supervisor is in charge of the Reactor Operations Sections, which, at the time of writing, consists of a Chief Reactor Operator (CRO) two Senior Reactor Operators and two Reactor Operators. The CRO assists the Reactor Supervisor particularly in the scheduling and supervision of experiments utilizing the reactor or any supporting facilities and equipment, the maintenance of logs and records, the maintenance of the physical conditions of the reactor and supporting equipment, and the training and retraining of personnel.

10.1.2.3 Senior Reactor Operator The Senior Reactor Operator (SRO), under the direction of the CRO, is responsible for the safe operation of the reactor. The SRO will supervise all experiments, maintenance, and any other activities capable of affecting the operation of the reactor.

10.1.2.4 Reactor Operators The Reactor Operators under the direction of a Senior Reactor Operator are responsible for the safe operation of the reactor and all its appurtenances including the reactor control system, reactor cooling system, reactor bridge, reactor pool, all experimental systems associated with the reactor, overhead crane, reactor building ventilation system, makeup and cleanup demineralizer systems, retention tanks, and all other equipment capable of affecting the operation of the reactor.

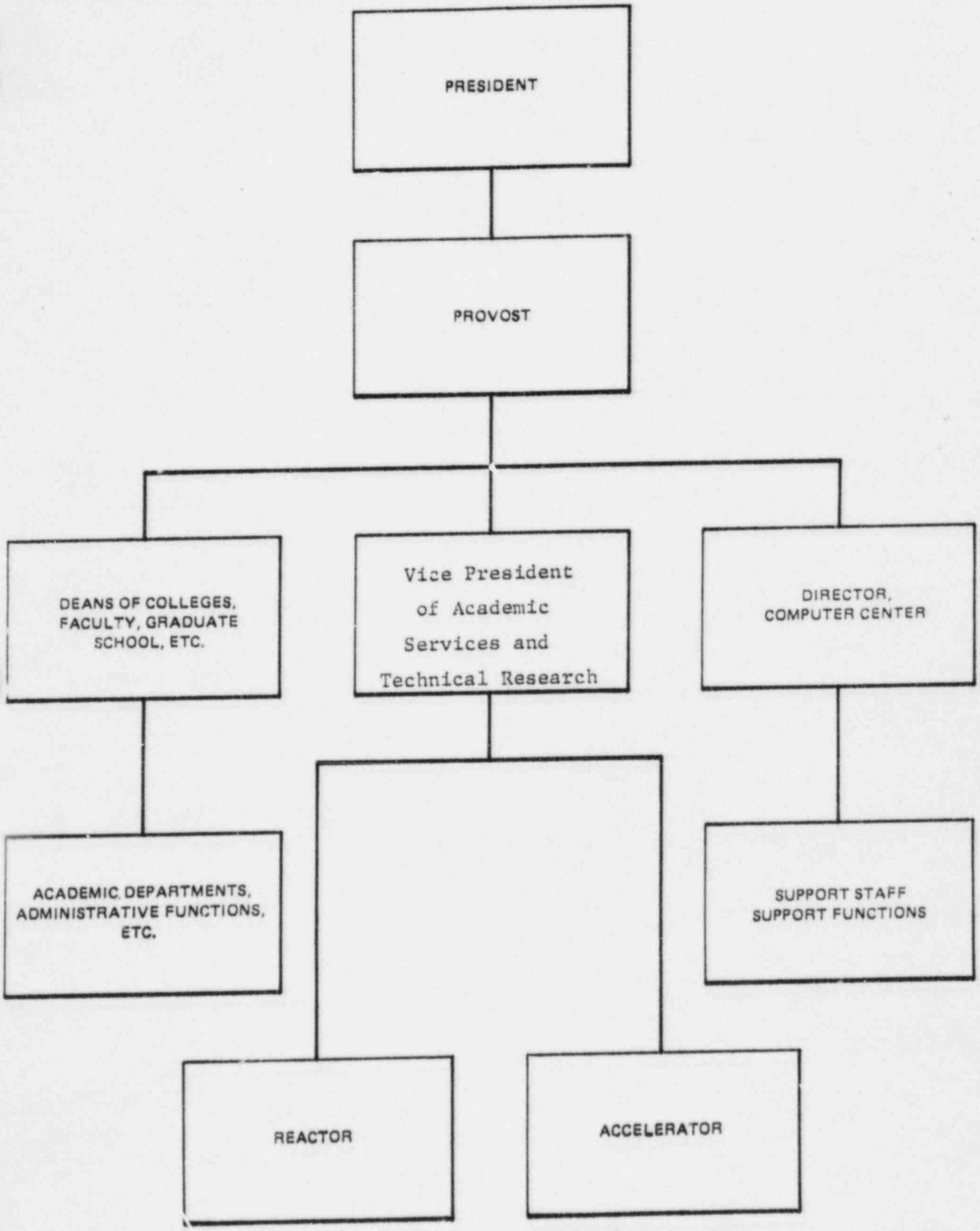
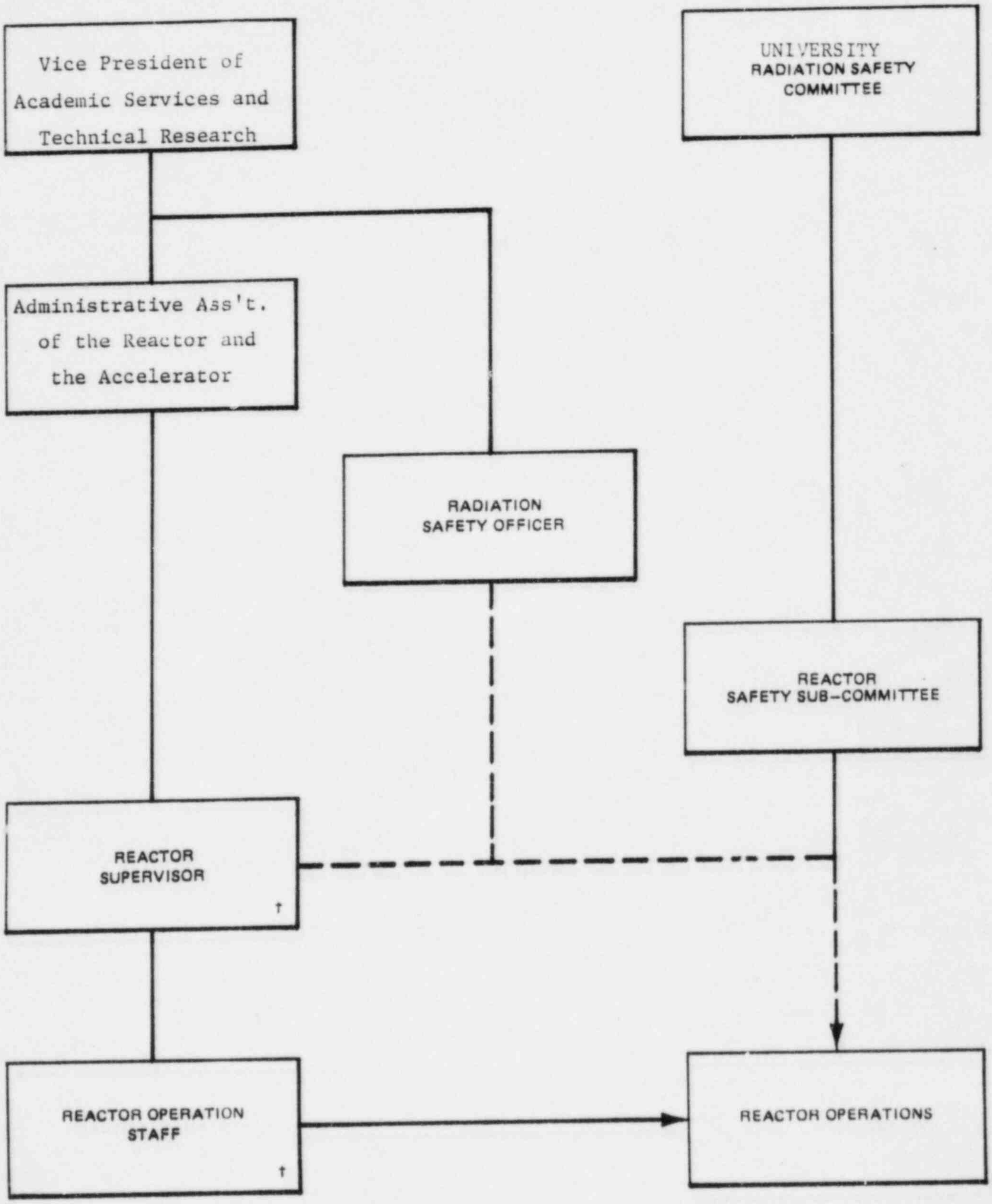


Chart 10.1 UL Administrative Structure



† USNRC Operator License Required

Chart 10.2 ULR Organizational Structure

10.1.2.5 Shift Crew Composition For a given shift the minimum crew will consist of two operators designated by the CRO. The CRO will insure that license requirements as specified in appropriate University of Lowell procedures are met.

Of the two shift Operators, one must be at the console at all times during which the control system is unlocked. The operator at the console has the primary responsibility under the Senior Reactor Operator (he may be the Senior Reactor Operator) for the operation of the reactor and all associated control and safety devices. The other Reactor Operator (who may be a Senior Reactor Operator) is responsible for required tasks in or about the reactor such as logging remote meter readings, inspection of equipment remote from the control room, supervising experiments, contacting meteorology, etc.

10.1.2.6 Support Groups Much routine maintenance is performed by the reactor operations staff. When necessary, work of a nature that requires help from the other than the operating staff is ordered and supervised by the operations staff, normally through the CRO or the Reactor Supervisor.

The Health Physics group is an integral component of the radiation Lab. In addition, an electronics shop and a model/machine shop are in the same building and are routinely used by the Lab for various activities. The University maintenance group, security department, etc., are available on an "as needed" basis.

10.1.3 Qualifications Requirements for Personnel

10.1.3.1 Reactor Supervisor The Reactor Supervisor is expected to have a thorough knowledge of nuclear engineering, particularly in the field of core physics and nuclear kinetics, as well as a thorough knowledge of all rules, regulations, practices, and procedures required by the USNRC, the Massachusetts Department of Public Health, and UL Radiation Safety Committee. He must have the ability to deal effectively with others and to train and direct personnel. He must hold a USNRC Senior Reactor Operator License, and should have had considerable supervisory experience at a similar reactor facility. The Reactor Supervisor is normally expected to hold an academic degree in one of the physical sciences or engineering disciplines.

10.1.3.2 Chief Reactor Operator The Chief Reactor Operator is expected to be familiar with the physics as well as the engineering aspects of reactors similar to the ULR. He will normally be cognizant of other systems but to a lesser degree. He must have a thorough knowledge of all rules, regulations, practices and procedures required by the USNRC, the Massachusetts Department of Public Health, and the UL Radiation Safety Committee. He must be able to deal effectively with others and to instruct reactor personnel. He must hold a USNRC Senior Reactor Operator License and should have had supervisory experience. The Chief Reactor Operator is normally expected to have a Baccalaureate's degree or equivalent.

10.1.3.3 Senior Reactor Operator A Senior Reactor Operator is expected to have a knowledge of the engineering aspects and reactor physics of the ULR, as well as of the rules, regulations, practices and procedures required by the USNRC, the Massachusetts

Department of Public Health, and the UL Radiation Safety Committee. He must have considerable knowledge of the safety precautions peculiar to nuclear operations and the ability to supervise, instruct, and train employees in reactor operations as well as to operate reactor equipment himself. He must possess a USNRC Senior Reactor Operator License.

10.1.3.4 Reactor Operator A Reactor Operator is expected to have working knowledge of the basic science and engineering underlying the ULR as well as of the rules, regulations, practices and procedures required by the USNRC, the Massachusetts Department of Public Health and the UL Radiation Safety Committee. He must understand the safety precautions peculiar to nuclear operations and be able to perform the various manipulations and exercise the various judgments required in the regular operation of the reactor. He must possess a USNRC Reactor Operator License.

10.2 TRAINING PROGRAM

10.2.1 Subject Matter

The training program is intended to impart to operator candidates the knowledge requisite for the position sought, including the definitions of 10CFR55. The program is tailored to the requirements of the individual with self-study being the preferred method. Formal lectures are given in areas where candidates are inexperienced. A qualification card encompassing all physical systems, procedures, instrumentation, as well as operating experience must be completed prior to recommendation for a license exam.

10.2.2. Scheduling

Normally the entire training program is expected to be carried out in about one year for a group of candidates who have no prior experience. With individual candidates, the time can vary by wide margins, depending on prior experience, ability, etc. For example, a man who has completed the U.S. Navy Nuclear Power Training Program would normally need only to familiarize himself with the ULR and the applicable rules, regulations, and procedures, all of which might be done in several months.

10.2.3 Authority

The Reactor Supervisor is in charge of all training and retraining of reactor operators.

10.2.4 Retraining Program

The retraining program is delineated on our approved Requalification Program. This requalification program is presently inforce and being fully implemented.

10.2.5 Replacement Training

Replacement training follows the general pattern of the original training program, with appropriate modifications due to the accessability of the plant, prior experience of the candidate, etc.

10.3 RECORDS

Records of the qualification, training, experience, and retraining as appropriate is kept for operating personnel. Personnel exposure records are maintained in accordance with 10CFR20.

10.4 EMERGENCY PLANNING

The University maintains a seperate Emergency Plan, approved by the USNRC. This Emergency Plan is written according to 10CFR50.54 and 10CFR50 Appendix E. The Reactor Area Radiation Monitoring System and Emergency Alarm System is enclosed in Appendix 10 as a reference for this Updated Safety Analysis Report.

10.5 REVIEW AND AUDIT

10.5.1 UL Radiation Safety Committee

10.5.1.1 Membership The Radiation Safety Committee is appointed by the President of UL and approved by the Board of Trustees. The Committee includes, in addition to certain persons required by Federal Regulations, members who represent broad areas or divisions of UL which are likely to use radiation sources and is thus a mechanism for dissemination of information to the various possible users.

10.5.1.2 Committee Responsibilities, Delegation of Authority, and Subcommittees The Radiation Safety Committee promulgates policy, regulations, and procedures relative to radiation safety. The Committee has the ultimate responsibility for all aspects of safety in the use of any device or source capable of emitting hazardous radiation. The Committee promotes the safe use of radiation sources and insures the health and safety of personnel and property both within the University and the public at large.

The Committee is responsible for assuring that an adequate safety program is developed and implemented. This is accomplished through delegation of authority to various persons, ad hoc subcommittees, and standing subcommittees with specific expertise in areas under their purview. In order to ensure good communication and to ensure that all activities are carried out according to established policies and procedures, the Radiation Safety Officer is a permanent member of the Radiation Safety Committee and all subcommittees. The various delegated authorities are outlined below.

(1) The (Standing) Accelerator Safety Subcommittee reviews all aspects of safety in and around the accelerator facility. Subcommittee members are appointed by the Radiation Safety Officer and the Accelerator Supervisor, persons with training experience relative to accelerator operations. Decisions of the Subcommittee are binding subject to the ultimate approval by the Committee.

(2) The (Standing) Reactor Safety Subcommittee considers all safety aspects of the reactor. Subcommittee members are appointed by the Radiation Safety Committee and include, in addition to the radiation Safety Officer and Reactor Supervisor, persons with training and experience relative to reactor operations and persons with specific expertise in various scientific and engineering disciplines. Decisions of the Subcommittee are binding subject to the ultimate approval by the Committee. (See Section 10.5.2 below).

(3) The Radiation Safety Officer, in addition to administering the Radiation Safety program, reviews all applications to use radiation sources. the Radiation Safety Officer in agreement with the appropriate Supervisor approves applications for the use of the accelerator or reactor of a routine nature already considered by the appropriate Subcommittee, and ultimately the Radiation Safety Committee.

Additional requests of a routine nature relating to the use of other radiation sources are approved by the Radiation Safety Officer. This approval is considered to be in effect subject to ultimate review by the Committee. Requests to use such radiation sources, which, in the judgment of the Radiation Safety Officer, involve a safety question not yet considered by the Committee, are placed before the Committee. The Committee resolves the question either by direct action, or, where indicated, by appointment of an Ad Hoc Subcommittee composed of persons with expertise in the area under question whose decision is binding subject to approval by the Committee.

The Radiation Safety Office under the direction of the Radiation Safety Officer, with the ulitmate approval of the Committee, formulates the safety program, which must be consistent with applicable Federal and State regulations governing the use of radioactive material and radiation producing devices. The approved program containing the overall policies and procedures relating to the safe use of radiation sources in the University is distributed to actual or potential users in the form of the Radiation Safety Guide for the University.

10.5.1.3 Committee Meetings The Committee and all Subcommittees meet as frequently as required in order to fulfill their responsibilities. The Committee or any of the Subcommittees shall also meet when a respective member formally requests a meeting. Binding committee decisions require a majority of the members to be present including the Radiation Safety Officer or his designee. Minutes of the meetings will be recorded and kept on file for review.

10.5.2 UL Reactor Safety Subcommittee

The Reactor Safety Subcommittee is composed of at least five members, one of whom is the Radiation Safety Officer and another of whom is the Reactor Supervisor. The other members are appointed (by the Radiation Safety Committee) with the aim of achieving a proficiency in all areas of reactor operation and reactor safety. Normally the members are senior scientific or engineering staff members or faculty. The Chairman is normally chosen from the senior members and normally does not have line responsibility for operation of the reactor.

The authority of the Subcommittee is the authority of the Radiation Safety Committee. The responsibilities of the Subcommittee include:

(1) Review and approval of normal, abnormal, and emergency operating and maintenance procedures and records. The Facility Technical Specifications lists specific procedural categories covered.

(2) Review and approval of proposed changes to the facility systems or equipment, procecures, and operations.

(3) Review and approval of proposed tests and experiments utilizing the reactor facilities.

(4) Determination of whether a proposed change, test, or experiment would constitute and unreviewed safety question requiring a change of the Technical Specifications or facility license.

(5) Review of all violations of the Technical Specifications

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

and NRC Regulations, and significant violations of internal rules or procedures, with recommendations for corrective action to prevent recurrence.

(6) Review of the qualifications and competency of the operating organization to assure retention of staff quality.

The Subcommittee meets at the request of any member, and at least quarterly. Minutes of all meetings are kept. A quorum of the Subcommittee is an absolute majority of the full Subcommittee and must include the Radiation Safety Officer or his designee and the Chairman or his designee.

10.5.3 Experimental Approval Mechanism

All proposed experiments using the reactor are evaluated by the experimenter and a staff member approved by the Reactor Safety Subcommittee. The evaluation includes consideration of 1) the reactivity worth of the experiment which must be no more than 0.5% $\Delta k/k$, 2) the integrity of the experiment, including effects of changes in temperature, pressure, or chemical composition, 3) any physical or chemical reaction or interaction that could occur with reactor components, and 4) any radiation hazard that may result from the activation of materials from external beams.

The initial evaluation is reviewed by the Reactor Supervisor and the Radiation Safety Officer; if the experiment meets their approval and complies with the provision of the utilization license, the Technical Specifications, and 10CFR20, it is

a. submmitted by the Reacor Supervisor to the Reactor Safety Subcommittee for approval if it is a new experiment or involves a safety question not yet reviewed by the subcommittee, or,

b. scheduled with the Reactor Supervisor's approval if it is a routine experiment.

If the experiment is submitted to the Subcommittee for evaluation, the following aspects are considered.

a. The purpose of the experiment.

b. The effect of the experiment on reactor operation and the possibility and consequences of failure of some aspect of the experiment, including, where indicated or significant, chemical reactions, physical integrity, design life, proper cooling interaction with core components, radiation and reactivity effects.

c. Whether or not the experiment, by virtue of its nature and/or design constitutes a significant threat to the integrity of the core, the integrity of the reactor, or to the safety of personnel.

d. A procedure for the performance of the experiment.

A favorable Subcommittee evaluation must conclude that failure of the experiments will not lead to direct failure of any reactor component or of other experiments.

No experiment may be conducted until a favcrable evaluation indicated in writing is rendered by the Reactor Safety Subcommittee.

If an experiment has had prior Subcommittee approval it becomes a routine experiment and approval of a routine experiment or a minor variation with no significantly different safety questions may be done for the subcommittee by agreement of the Reactor Supervisor and the Radiation Safety Officer.

10.6 SECURITY

All aspects of security as it relates to the UL reactor are delineated in the approved Physical Security Plan.

10.7 QUALITY ASSURANCE PROGRAM

The QA program is largely based on the surveillance requirements for important systems and sub-systems specified in the ULR Technical Specifications. The QA program is not static, and is modified as indicated by operating experience to assure an adequate program. The areas included in the QA program are:

a. Reactor Control and Safety Systems
b. Radiation Monitoring Equipment
c. Containment Building and Filter Checks
d. Pool Water Level Channel
e. Emergency Power System
f. Maintenance

The QA program is intended to ensure that:

a. Proper authority for scheduling, carrying out, and approving surveillance tests or maintenance is documented,
b. Personnel involved in the various aspects of surveillance or maintenance are qualified in a prescribed manner. Training and qualifications will be documented.
c. The equipment and test components used to perform the various checks are calibrated, certified, and properly documented.

d. Performance verification in the various surveillance and maintenance tasks is documented. Repair and/or replacement of components will be controlled from inception through installation and performance tests.

e. Surveillance and maintenance schedules are maintained.

f. Periodic audits are made by appropriate personnel independent of the reactor organization.

APPENDIX 10

APPENDIX 10

Reactor Area Radiation Monitoring System and Emergency Alarm System

The reactor radiation monitoring system, in its operations on both a routine basis and in radiation emergency situations, is intended to provide concerned personnel with reliable information for estimating the extent of any radiation problem which might exist. In particular, the area monitoring system and the emergency alarm system provide an indication, to appropriate personnel within the reactor and in the Pinanski Building, of radiation hazards resulting from the release of airborne radioactivity, and/or the presence of significant ambient gamma radiation levels, and/or the release of fission product activity.

Instrumentation in these systems has been assigned to one (or more) of three categories according to the type of radiation hazard to which the instrumentation is intended to respond. These categories are (I) Airborne Radioactivity, (II) Gross Radiation and (III) Fission Product Release. the detectors and associated instrumentation are categorized as follows, locations being shown where appropriate:

(I) Airborne Radioactivity

(A,B) Stack Monitors* GM and scintillator types.

(C,D) Reactor Constant Air Monitors, 2 in number, GM type.

(II) Gross Radiation

(F) Facilities Exhaust Filter-GM 2**

(G) Rabbit Exhaust Filter-GM 2

(H) Hot Cell Area Monitor-GM 2

(I) Building Exhaust Plenum-GM 1**

* Stack monitor has two types of detectors (particulate and gaseous radioactivity)

** GM 1 = Tracerlab TA-61	GM tube; range 0.01-100 mR/h
GM 2 = Tracerlab TA-62	GM tube; range 1 mR/h-10 R/h
I.C. = Tracerlab TA-64	Ion Chamber; range 0.1 mR/h-10 R/h

(K) Bridge - I.C.**

(L) Opposite thermal column (wall) - GM 1

(M) Rabbit Tube No. 1 - GM 1

(N) Rabbit Tube No. 2 - GM 1

(O) Experimental Floor near stairs - GM 1

(P) Pump Room - GM 1

(Q) Opposite Gamma Cave (wall) GM 1

(R) Control room - GM 1

(III) <u>Fission Product Release</u>

(E) F.P. Monitor

(S) Core Exit Line - I.C.**

The simultaneous "tripping", at present alarm points, of two different detectors, each from a different category (I, II, III) is intended to be indicative of a potentially hazardous situation which would warrant initiation of the General Radiation Emergency Alarm (GREA). In the absence of positive knowledge to contraindicate such an emergency situation the reactor operator would be expected to initiate the GREA. Of the above listed detectors those which will be used to indicate such an emergency situation are as follows:

All in Category I
Detectors I, K, L, R in Category II
All in Category III

**	GM 1 = Tracerlab TA-61	GM tube; range 0.01-100 mR/h
	GM 2 = Tracerlab TA-62	GM tube; range mR/h-10 R/h
	I.C. = Tracerlab TA-64	Ion Chamber; range 0.1 mR/h-10 R/h

Thus, those detector combinations which would be indicative of an emergency situation warranting a GREA are:

A + 1
or A + K
or A + L
or A + R
or A + E
or A + S

or

B + 1
or B + K
or B + L
or B + R
or B + E
or B + S

or

C + 1
or C + K
or C + L
or C + R
or C + E
or C + S

or

D + I
or D + K
or D + L
or D + R
or D + E
or D + S

or

E + I
or E + K
or E + R

or

S + I
or S + K
or S + R

Those detectors of A, B, C, and D which have the requisite sensitivity for the radioisotopes which are capable of being detected by the system have preset "trip" levels above 10 times MPC; the "trip" levels on those systems that are capable of rapidly detecting a fission product release are set at a level which has a potential to result in about 10 times MPC of airborne radioactivity or 100 mR/h, of gross radiation; levels for detectors, I, K, L and R are set at about 100 mR/h. The GREA is sounded by pulsating horns located in the reactor building and the laboratory-classroom building.

In addition to the GREA system, there exists a Limited Radiation Emergency Alarm (LREA) system. This alarm would be activated by the operator in particular situations wherein certain combinations of the above mentioned detectors simultaneously reach a present "trip" level, unless such action is contraindicated by positive knowledge on the part of the operator. The LREA will utilize only those alarm horns of the GREA system which are located within the reactor building. The detector combinations which would warrant initiation of the LREA are the following:

$$\boxed{E + S} \xrightarrow{\text{or}} \boxed{C + D} \xrightarrow{\text{or}} \boxed{K + R} \xrightarrow{\text{or}} \boxed{L + M + O}$$

The preset "trip" levels for detectors C, D, E and S are determined in the field; detector I, in the building exhaust plenum is set to trip at about 5 mR/h; detectors K, L, M, O and R are set at about 100 mR/h. Sounding of the LREA in the reactor building will be accompanied by simultaneous sounding of individual audible alarms in preselected offices in the Radiation Laboratory.

All detectors listed above (A through S) have meter readouts available at the reactor control room in the Radiation Laboratory; detector outputs will also be recorded at the console. All readout meters located at the control console are equipped with high and low selectable "trip" level switches; simultaneous tripping of high level switches on certain detector combinations which could indicate a GREA or LREA situation as noted above, will activate an annuciator light and an audible alarm in the reactor control room. "Tripping" at the low preset level for any detector readout activate an annuciator light. Similar annunciating devices are present in the H.P. laboratory; however, meter readouts in the H.P. laboratory do not include "trip" switches.

The reactor operator may also initiate the GREA or LREA in any situation which he feels warrants such action although the specific combination of detector responses discussed above may not be in effect.

Financial Qualification Statement

The University of Lowell Radiation Laboratory functions as a service and research organization within the University. It is a separate organization, having a Director who is responsible to the Associate Vice President for Research. The Radiation Laboratory is made up of the Accelerator group, the Health Physics group and the Reactor group. The budget for the reactor is an integral part of the Radiation Laboratory budget and no attempt has been made to separate it.

Funding for the Radiation Laboratory comes from two sources: directly from the University/State budget and from direct and overhead charges to research and service contracts. The University has made a committment to fund the Laboratory at the rate of $300,000 per year. Any funding needs above that must come from research service contracts. The University supplies additional support in the form of utiliites, space, maintenance, police/security, machine ships, electronic repair shop, etc., for which no specific charges are made.

Figure 1 gives the estimated budget for Fiscal 1985 and gives an estimate for 1986 to 1989. Estimated future budgets are based on 5%/yr inflation of salaries and 10% cost increased/yr in other expenses. Projected income rates are based on increases of 10 to 20% per annum.

In fiscal 1985, it is projected that grants and contracts valued over $900,000, will be performed by the Radiation Laboratory. Of this amount, most will go to the actual performance of the contract or to the Research Foundation for administrative support. However, $112,000 (est.) will be returned to the Radiation Laboratory, either directly or through the Principle Investigator, for use in the Laboratory programs.

Also included is a balance sheet of the total expenditures of Commonwealth of Massachusetts funds. Most of the operating budget of the Univerity comes from the Commonwealth. An accounting of other sources of funds, i.e., student fees, continuing education, etc., are not available at this time.

Decommissioning

At the present time, the University does not plan to shut down the reactor facility in the foreseeable future. No specific plan has been devised for the decommisioning of the reactor facility. However, a reasonable estimation of the cost would be $250,000 to $1,000,000 depending on the degree of dismantling and decontamination of the facility required. The lower figure is based on the removal of all fuel and loose structural material. The building would have to be maintained in a secured condition with periodic surveillance by health physics personnel. It is anticipated that initial shutdown would be this case. All funding for such a shutdown would have to come from the University budget or through special appropriation of the state legislature.

FIGURE-1 SOURCES OF FUNDING FOR RADIATION LABORATORY

SOURCE	1985	1986	1987	1988	1989
UNIVERSITY	300000.00	300000.00	300000.00	300000.00	300000.00
UNIV. UNACC*	150000.00	162000.00	174960.00	188956.80	204073.34
CONTRACTS					
OVERHEAD[1]	112000.00	123200.00	135520.00	149072.00	163979.20
OTHER[2]	50000.00	60000.00	66000.00	72600.00	79860.00
TOTAL	612000.00	645200.00	676480.00	710628.80	747912.54

ESTIMATED EXPENDITURES FOR RADIATION LABORATORY

	1985	1986	1987	1988	1989
SALARIES	319236.00	335197.80	351957.69	369555.57	388033.35
UNIV. UNACC*	150000.00	162000.00	174960.00	188956.80	204073.34
EQUIPMENT	50000.00	55000.00	60500.00	66550.00	73205.00
SUPPLIES	50000.00	55000.00	60500.00	66550.00	73205.00
TRAVEL	10000.00	11000.00	12100.00	13310.00	14641.00
TOTAL EXPEN.	579236.00	618197.80	660017.69	704922.37	753157.70
MISC.[3]	32764.00	27002.20	16462.31	5706.43	-5245.15

1 - Indirect costs, recovered from outside contracts, returned by the University to support the operation of the Radiation Laboratory.

2 - Income from direct service contracts.

3 - This figure is used to balance estimated costs vs. income. It is considered discretionary funding and is typically used for student support, etc.

* Some costs, i.e., maintenance, utilities, security, fringe benefits, etc., are paid for directly by the University. These costs are only estimated here for illustrative purposes.

ACCOUNT NO ACCOUNT NAME
7220-6001 UNIVERSITY OF LOWELL

UNIVERSITY OF LOWELL-MAINTENANCE
ALLOCATION OF 7100-0100

EXECUTIVE OFFICE FOR ADMINISTRATION AND FINANCE
COMPTROLLERS DIVISION

FINAL STATEMENT OF BALANCES - 1984
ACCOUNT EXPIRATION DATE 06/30/84
ROUTING CODE 0366 PAGE 1

	TOTAL AVAILABLE	UNALLOTTED BALANCE	ALLOTTED	EXPENDITURES	ADVANCES	ENCUMBRANCES	TOTAL CHARGES	BALANCE-INCLUDING-UNALLOTTED REVERTED	RESERVED
SUB 01 END BAL	23,367,669.00		23,367,669.00	23,303,944.90		16,000.00	23,319,944.90	47,724.10	
SUB 02 END BAL	126,624.00		126,624.00	120,268.62		6,000.00	126,268.62	355.38	
SUB 03 END BAL	1,188,520.00		1,188,520.00	1,172,312.45	4,599.31	11,128.22	1,188,039.98	480.02	
SUB 05 END BAL	6,000.00		6,000.00	5,839.20		158.99	5,998.19	1.81	
SUB 06 END BAL	98,820.00		98,820.00	91,634.09		7,164.30	98,798.39	21.61	
SUB 07 END BAL	4,120.00		4,120.00	3,294.20		801.95	4,096.15	23.85	
SUB 08 END BAL	2,811,500.00		2,811,500.00	2,659,261.72		152,238.28	2,811,500.00		
SUB 09 END BAL	8,000.00		8,000.00	7,469.13		522.66	7,991.79	8.21	
SUB 10 END BAL	46,350.00		46,350.00	45,581.32	430.00	300.79	46,312.11	37.89	
SUB 11 END BAL	73,500.00		73,500.00	72,753.17		746.83	73,500.00		
SUB 12 END BAL	990,727.00		990,727.00	882,540.97		108,072.98	990,613.95	113.05	
SUB 13 END BAL	247,161.00		247,161.00	221,180.96		25,963.06	247,144.02	16.98	
SUB 14 END BAL	450,056.00		450,056.00	391,921.66	10,142.05	47,026.14	449,129.85	666.95	
SUB 15 END BAL	2,333,231.00		2,333,231.00	1,949,865.36		283,710.34	2,233,575.70	99,655.30	
SUB 16 END BAL	84,053.00		84,053.00	72,659.25		11,393.75	84,053.00		
TOTAL END BAL	31,836,331.00		31,836,331.00	31,000,527.26	15,271.36	671,227.31	31,687,025.65	149,305.35	

The attached is a copy of the University's FiscalYear 1984 operating budget. Below are descriptive titles of the subsidiary accounts funded by the state maintenance appropriations:

Subsidiary	01	Salaries, Permanent Positions
	02	Salaries, Other
	03	Services - Non-employees
	05	Clothing
	06	Housekeeping Supplies & Expenses
	07	Laboratory and Medical Supplies, and Expenses and General Care
	08	Heat and Other Plant Operation
	09	Farm and Grounds
	10	Travel and Automotive Expenses
	11	Advertising and Printing
	12	Maintenance-Repairs, Replacements and Alterations
	13	Special Supplies & Expenses
	14	Office and Administrative Expenses
	15	Equipment
	16	Rentals

III Environmental Impact Report

Introduction

The University of Lowell one megawatt research reactor, during its first ten years of operation, has produced no significant environmental impact. Future operation is expected to continue as in the past, and no discernible effects on the environment are anticipated.

Facility

The University reactor is housed in a steel containment building attached to the Radiation Laboratory building located on the University North Campus. The reactor gas effluent stack and cooling tower are located on the site immediately adjacent to the reactor building. No additional exterior conduits, pipelines, electrical or mechanical structures or transmission lines, other than utiltity service facilities, are attached to or adjacent to the facility.

The 1.22 x 10^7 BTU/hr (design capacity) cooling tower allows for heat dissipation when the reactor is operated at power levels in excess of 100 kw.

Make up for this cooling system is readily available and is obtained from the city of Lowell water supply.

Radioactive gaseous effluents are limited to Ar-41, and the releases of radioactive liquid effluents are carefully monitored and controlled. These liquid wastes are collected in storage tanks located in the basement of the Radiation Laboratory

Building, to allow for decay and monitoring prior to possible dilution and release to the sanitary sewer system. Solid radioactive wastes are packaged according to NRC-DOT requirements and shipped off-site; the solid waste is transported by a licensed contractor to an existing NRC approved site (presently in the state of Washington).

Environmental Effects of Facility Operation

At power levels below 100 kw the reactor is typically cooled by natural convection in the reactor pool water. At levels greater than 100 kw forced primary coolant flow is directed through a heat exchanger in the pump room of the reactor building; secondary water flow through the heat exchanger removes heat from the primary coolant. The secondary water is then cooled through heat dissipation via the cooling tower located within a fenced area external to and adjacent to the reactor building. The small temperature rise of primary coolant through the reactor core and the relatively small amount of waste heat transmitted to the atmosphere produce little fog from the cooling tower and no significant environmental effects.

Argon-41 is the only reactor produced radioactive contaminant which is observed in the gaseous effluent from the reactor stack. Extremely conservative calculations, taking no account of atmospheric dispersion, were carried out in the FSAR submitted to the NRC in September of 1973. At that time a projected Ar-41 continuous release rate of 8 μCi sec^{-1} yielded a projected maximum whole body annual dose to an individual (security guard) in the adjacent parking area of 4.2 mrem. Actual Ar-41 release rates, based on measured emissions over the past five years, yield an average continuous emission rate of 0.34 μCi sec^{-1}. This would cranslate to a maximum whole body annual dose of <0.2 mrem to the same indiv:dual. Thus, the dose significance of the Ar-41 release is felt to be insignificant. The annual releases of Ar-41 for the past five years are given in Table 1.

Radioactive liquid wastes generated in the reactor are transferred to storage tanks in the basement of the Radiation Laboratory building. Short-lived activity is allowed to decay and, after monitoring levels of radioactivity in the tanks to assure that concentrations are acceptably low, the liquid waste is drained to the sanitary sewer. The total quantity of gross beta activity released over the past five years of operation has been 3.655×10^{-3} Ci. Annual releases for the past five years are summarized in Table 2. The sanitary sewer system eventually empties into the Merrimac River. Daily aliquots (200 ml) of river water are collected at sites upstream and downstream of the sewerage outfall. The aliquots from each site are combined to comprise biweekly samples which are analyzed for radioativity content. No radioactivity resulting from discharges from liquid wastes has been identified in any of the samples collected over the past five years of reactor operation. The impact of the environmental release of liquid radioactive wastes has been negligible and is expected to remain so.

No significant releases of nonradioactive toxic substances has attended operation of the facility, and such are not anticipated.

Table 1

Ar-41 Releases in Gaseous Effluent

during the Past Five Years

Year (July 1 - June 30)	Curies Released
1979 - 1980	23.56
1980 - 1981	4.09
1981 - 1982	14.03
1982 - 1983	5.19
1983 - 1984	6.31
Total over five years	53.18
Average release rate over five years	0.34 microcuries/sec.

Table 2

Gross Beta Activity Releases to Sanitary Sewer during Past Five Years

Year (July 1 - June 30)	Curies Released
1979 - 1980	2.183×10^{-3}
1980 - 1981	2.344×10^{-4}
1981 - 1982	3.520×10^{-4}
1982 - 1983	6.977×10^{-4}
1983 - 1984	1.879×10^{-4}
Total over five years	3.655×10^{-3}
Average diluted concentration over five years	7.42×10^{-9} microcuries/ml

Environmental Effects of Accidents

The most severe accident resulting in releases of radioactivity to the environment, postulated for the University of Lowell Reactor involves loss of cladding of one plate in a fuel element with resultant transfer of all volatile fission products to the primary coolant. The consequences of such an accident have been presented in the FSAR of September 1973 submitted to the NRC. The analysis assumes direct exposure of an individual to activity released from the building. If allowance is made for any atmospheric dispersion of the released activity, projected doses represent a very small fraction of the 10CFR Part 100 guidelines. Allowance for even a 0.1 mile per hour wind acting to dilute the activity released from the building would reduce total thyroid doses from released radioiodines to less than two rads and whole body gamma dose to less than 60 mrads.

Effects of Facility Operation

No adverse effects on the environment are expected from operation of the reactor.

Alternative to Operation of the Facility

The reactor facility is used for education, training, and research. While a few of the research applications could be carried out using the University's Van-de-Graaff accelerator, there are no suitable alternatives which can achieve the large majority of objectives of operating the reactor.

Long-Term Effects of Facility Operation

No significant long-term effects are expected in association with operation of the reactor. As noted above, the thermal load on the air environment and the dose impact of the small quantities of radioactivity routinely released in gaseous and liquid forms are insignificant and far below allowed limits. Solid wastes generated by reactor operations typically fill less than five 55 gallon drums per year; this number is extremely small compared to the waste generated at other nuclear facilities and will have a minimal impact on the longevity of low level waste burial grounds.

The reactor facility has operated on campus for approximately ten years, and has been accepted with no difficulties or complaints by the University and by the public. The facility has no more adverse impact from the point of view physical aesthetics, noise generation, or impact on the local topography than do other campus laboratory type buildings; it is expected that this will remain so in the future.

Costs and Benefits of Facility and Alternatives

The annual operating costs of the reactor facility are discussed in the Financial Qualifications Section of this application. The facility directly benefits at least one hundred students per year who make use of the facility in training in reactor operations, training in radiation safety and protection, and research activities at both the undergraduate and graduate levels. Indirect benefits accrue to many additional students through the ongoing research of faculty and staff whose research involvement and accomplishments impact directly on the course material offered to students. The reactor facility is open to tours by the public or specific groups with an interest in the facility, and has served a continuing function in providing information relative to many aspects of reactor operations, the role of nuclear power, and the environmental effects of nuclear related operations.

Fifteen students have received M.S. degrees on the basis of research conducted directly in association with the reactor. The Radiological Sciences program and the Nuclear Engineering program rely on operation of the reactor in the conduct of required courses and laboratories. The chemistry, physics, plastics, and mechanical engineering departments have all made use of the reactor in a variety of student laboratories and projects.

No reasonable alternatives to this facility are available for the great majority of work carried out using the reactor. If the reactor were not available it would have a large negative impact on a significant part of the student body and on the programs which use it.