NATIONAL BUREAU OF STANDARDS REPORT

8998

Final Safety Analysis Report on the National Bureau of Standards Reactor

NBSR 9

NBS Reactor Radiations Division
Institute for Materials Research

U.S. DEPARTMENT OF COMMERCE
NATIONAL BUREAU OF STANDARDS

THE NATIONAL BUREAU OF STANDARDS

The National Bureau of Standards is a principal focal point in the Federal Government for assuring maximum application of the physical and engineering sciences to the advancement of technology in industry and commerce. Its responsibilities include development and maintenance of the national standards of measurement, and the provisions of means for making measurements consistent with those standards; determination of physical constants and properties of materials; development of methods for testing materials, mechanisms, and structures, and making such tests as may be necessary, particularly for government agencies; cooperation in the establishment of standard practices for incorporation in codes and specifications; advisory service to government agencies on scientific and technical problems; invention and development of devices to serve special needs of the Government; assistance to industry, business, and consumers in the development and acceptance of commercial standards and simplified trade practice recommendations; administration of programs in cooperation with United States business groups and standards organizations for the development of international standards of practice; and maintenance of a clearinghouse for the collection and dissemination of scientific, technical, and engineering information. The scope of the Bureau's activities is suggested in the following listing of its four Institutes and their organizational units.

Institute for Basic Standards. Applied Mathematics. Electricity. Metrology. Mechanics. Heat. Atomic Physics. Physical Chemistry. Laboratory Astrophysics.* Radiation Physics. Radio Standards Laboratory:* Radio Standards Physics; Radio Standards Engineering. Office of Standard Reference Data.

Institute for Materials Research. Analytical Chemistry. Polymers. Metallurgy. Inorganic Materials. Reactor Radiations. Cryogenics.* Materials Evaluation Laboratory. Office of Standard Reference Materials.

Institute for Applied Technology. Building Research. Information Technology. Performance Test Development. Electronic Instrumentation. Textile and Apparel Technology Center. Technical Analysis. Office of Weights and Measures. Office of Engineering Standards. Office of Invention and Innovation. Office of Technical Resources. Clearinghouse for Federal Scientific and Technical Information.**

Central Radio Propagation Laboratory.* Ionospheric Telecommunications. Tropospheric Telecommunications. Space Environment Forecasting. Aeronomy.

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** Located at 5285 Port Royal Road, Springfield, Virginia 22171.

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NBS PROJECT 3140100

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Final Safety Analysis Report on the
NBS High Flux Beam Research Reactor

Outline

SECTION 1. INTRODUCTION AND SUMMARY

1.1 GENERAL DESCRIPTION

The NBSR is a reactor-laboratory complex for providing the Bureau with the means of performing research and standards on materials and nuclear processes. Since the largest portion of the work involves the use of thermal neutrons, the reactor type chosen was one which generates a highly degraded and well thermalized neutron spectrum. This combined with the requirement that the flux be competetive at reasonable power dictated the choice of an enriched fueled heavy water moderated and cooled reactor. Since the basic technology of reactors of this type exists, NBS had only to custom design its own version of the Argonne CP-5 class reactor, effecting but minor modifications. This is to say that all of the technology is firmly established either by actual embodiment in previous systems or by extrapolation from those systems via good engineering practice.

The complex is shown in Figure 1.1. The front, or east portion, of the complex consists of offices, cold laboratories, warm laboratories, shops, and other special purpose space, all of which falls outside the nuclear confinement boundary. The north wing of this non-confinement portion of the structure has a cold basement area wherein are located various plant type items such as air-conditioning machinery, power distribution gear, demineralizing equipment, emergency power units, and storage space. This cold basement area is also accessible via a truck ramp on the northwest or rear side of the building.

The reactor, or confinement portion of the structure attaches to the aforementioned office and laboratory building on the latters west or rear side. It is a three level building 90' x 90' in cross section. The control room and general access to fuel elements and in-core thimbles is at the top or above grade level. Radial and through-beam tubes are at grade level. Principal access to this level of the building are through gasketed doors from the non-confinement building and a large gasketed truck door from the yard on the south side. The basement or below grade level of the confinement building contains most of the process systems, fuel element storage pool, and radiochemical laboratories into which the pneumatically operated sample irradiation tubes exit.

The reactor proper is centrally located within the confinement building. Joining the confinement building on its north side are the stack and secondary pump annex. These are outside the confinement boundary.

The location of the reactor facility and other principal buildings on the Bureau's campus are shown in Figure 1.2. The nearest site boundary from the reactor is one-quarter mile to the west. The site itself is in Montgomery County near Gaithersburg, Maryland, and is approximately twenty miles northwest from the center of Washington, D. C.

The reactor under consideration is a heavy water moderated and cooled, enriched fuel, tank type machine to operate at 10 Mw of power. It is of the Argonne CP-5 class. Basically it consists of an aluminum vessel filled with heavy water which also contains the core or grid of enriched plate type fuel elements. Surrounding the vessel is the thermal shield, an iron-lead light water cooled structure to protect the biological shield from excessive radiation heating. Surrounding the thermal shield and penetrated by numerous beam ports, etc., is the high density concrete biological shield. The NBSR differs from the CP-5 only by being upgraded in power and by modification in minor respects.

The core, for example, consists of an array of about twenty-four plate type elements wherein the enriched uranium is alloyed with aluminum within the readily achievable or normal range of composition, and clad with aluminum. The array is such as to generate a well moderated and thermalized reaction, in that the fuel element centers are far apart and allow for a high moderator to fuel ratio. The elements differ from the usual element, however, by being fueled both above and below the midplane of the core. The gap between the upper and lower regions is nearly the same as that between element centers. Neutron transport along the vertical dimension takes place as readily as it does in the radial dimension. The fuel element structure employed to realize this feature is essentially the same as that of a

normal multiplate element, but is a bit longer and is fabricated by a double rather than a single picture frame method. Re-entrant beam tubes welded to the reactor aluminum vessel terminate in the vicinity of the vertical fuel gap, thereby allowing extraction of neutron beams considerably freer of fast neutrons than can be obtained from the usual configuration. The neutron behavior in such a fuel distribution is readily anticipated by standard multi-group calculations, and does not require critical assembly mock-up measurements in advance of actual zero power startup.

Another modification, and one which should prove economical of process water flow, is the double plenum at the bottom of the vessel. By means of two independent concentric plena, flow to the inner and outer array of elements can be separately controlled.

A further significant modification in the NBSR is the method of handling fuel elements. For transferring elements to a different core position or for removing elements from the core, it is not necessary first to withdraw elements into a flask above the reactor top as in the usual CP-5 type system. Pick-up and transfer tools operating through the top lid effect the shuffling and dropout under a blanket of either heavy water or helium. The above core fuel handling system and the bottom receiver constitute the most significant engineering development problem undertaken by NBS in the design, fabrication, and procurement of the NBSR.

From an engineering point of view, the remaining significant modification in the NBSR is the means for providing extensive heavy water emergency cooling. The usual overhead tank can supply water either to the top or to the bottom of the elements. Additionally, two inner tanks within the reactor vessel retain heavy water in the event of a loss of water by the main vessel. One of these tanks supplies coolant flow to the elements, and the other maintains water around the lower half of the core; i.e., keeps the lower half of each element under water. These provisions far exceed those of the usual system. Of special significance is the ability to feed emergency coolant to the lower plenum, thereby forcing water up through the elements rather than depending solely on water being able to trickle down from the top through potentially very hot surfaces. These and more minor features of the system will be further discussed in later sections of the report.

The confinement building is of reinforced concrete and of sufficient tightness that when closed, any initial differential pressure relaxes toward zero in a mean-time of one hour or better. The philosophy of emergency; i.e., under conditions of major fission produce release, is one of controlled effluent. The internal atmosphere of the confinement building is recirculated and filtered. At the same time it is passed out of the building through absolute filters and a stack at such a rate as to maintain a small negative differential pressure across the building walls. Even under the worst hypothetical condition when all of the filtering fails, the dose to a person on the site boundary is negligible.

It is planned to operate the plant and its attendant experimentation twenty-four hours per day for about a three week cycle. Over the weedend, during which time the plant is shutdown, fuel is replenished and other maintenance is performed.

Table 1.1-1 below, lists the more significant nuclear and engineering characteristics of the system and plant.

Table 1.1-1 Principle Plant Characteristics

Power	10 Mw
Core Radius	55 cm
Core Height	74 cm
Inactive Split Height	18 cm
Active Core Volume	541 liters
Mean Power Density (Active Core)	18.5 Kw/liter
Fuel Elements:	
Type	Split MTR Curved Plate
Enrichment	93% U-235
Initial Number	19
Initial Loading	170 gm
Final Number	24
Final Loading	205 gm
Nominal Core Loading	4090 kg
Core Life	24 weeks
Core Cycle	3 weeks
Moderator Coolant	D_2O
Metal-to-Water Ratio	0.085
Reflector Thickness	60 cm
Thermal Neutron Flux (Peak)	$1.7 \times 10^{14}/cm^2 sec$
Fast Neutron Flux (Peak)	$1.2 \times 10^{14}/cm^2 sec$
Neutron Lifetime	700 μ sec
Rod Worths	
One Shim	-25% $\Delta\rho$
All Shims	-46% $\Delta\rho$
Fine Control	0.5% $\Delta\rho$
Void Coefficient	-0.040% $\Delta\rho$/liter
Temperature Coefficient	-.034% $\Delta\rho$/°C
Coolant Water	
Temperature Rise	12°F
Velocity	12 ft/sec
Flow Rate	5100 gpm
Heat Flux	1.0×10^5 Btu/hr ft^2
Shield Materials	
Thermal	2" Pb + 8" Fe
Biological	6' Magnitite

1.2 ORGANIZATION

The Bureau of Standards is organized into three institutes. The NBSR is managed by the Reactor Radiations Division, a division within the Institute for Materials Research. These features are disclosed in Figure 1.3. Shown by underscoring in this figure are those parts of the Bureau which have some significant effect on the operation of the NBSR.

The Reactor Radiations Division is divided into seven sections, three of which are scientific user sections. As with other user staff, both within and without the Bureau, they are not elaborated upon here, other than to indicate that certain of the principle scientists are available for membership on an internal "Hazards Evaluation Committee."

Table 1.2-1 discloses the titles and responsibilities of the seven sections of the Division, the first four of which have some effect on operations.

Table 1.2-1

Section	Responsibility
Administration	- Division administration and management
Scientific Support	- Facility development; applied technology; theory and analysis
Reactor Operations	- Reactor operation and maintenance; AEC compliance
Engineering Services	- Mechanical and electronic design and fabrication
Neutron-Nuclear Physics	- Research and Standards.
Neutron Solid-State Physics	- Research and Standards
Radiation Effects	- Research and Standards

Table 1.2-2

Organizational Unit	Service Rendered
Administrative Section, Reactor Radiations Division	- Guidance policy; funds; establishment of out-section-section relationships; monitoring of overall performance
Scientific Support Section, Reactor Radiations Division	- Precision radiation and plant performance measurement; new systems research and development; analysis
Engineering Services Section, Reactor Radiations Division	- Non-routine mechanical and electronic repair; test, maintenance, and modification; new systems design and procurement; coordination with NBS Shops Division
Hazards Evaluation Committee	- Analysis and approval of proposed experiments; and plant, or operational alterations
NBS Office of Radiation Safety	- Health physics
NBS Plant Division	- Non-nuclear plant maintenance
NBS Administrative Services	- Guard, fire, and ambulance; coordination with Montgomery County Fire Board, and Police.

It will be largely the responsibility of the Reactor Operations Section to sustain the daily relationships with other parts of the Division and the Bureau, insofar as this affects plant performance. The principle services rendered to Reactor Operations by other units are shown in Table 1.2-2.

The Reactor Operations Section will perform regular operation; normal maintenance; scheduling; management of Section funds; logging and reporting of plant performance to AEC or Washington Suburban Sanitary Commission; management of fuel; procurement of materials and supplies; monitoring experimental users and their equipment; installing or supervising the installation of user equipment; and in general assuming proper responsibility for management of the Bureau's Part 50 license. Further details are given in later sections.

1.3 CONTRACTURAL HISTORY

The conceptual plan and design of the plant was undertaken by an original group of NBS scientists and engineers, now incorporated into the Reactor Radiations Division. This group was assisted by the Allis Chalmers Manufacturing Company and certain engineering personnel from Naval Research Laboratory. The detailed engineering design was carried out by Burns and Roe, Inc., of New York, under the direct supervision of the aforementioned NBS-NRL forces. With the assistance of the General Services Administration and particularly of the Public Buildings Service the general construction contract was bid and let to the Blount Brothers Corporation of Montgomery, Alabama, this with the approval of the AEC. In parallel with the general contract the Bureau bid and let approximately fifteen other contracts for major reactor components, these to be installed by the general contractor.

At the time of this writing the Bureau has taken beneficial occupancy of the entire complex. With fuel scheduled for delivery in February 1966, it is anticipated that startup and power could be achieved by the first quarter of 1966.

1.4 SAFETY CONSIDERATIONS

There are no unusual safety problems connected with the NBSR, either by virtue of its basic type or by virtue of its particular design. The basic nuclear reaction in heavy water is slow; i.e., the prompt neutron lifetime is relatively long; and the principle reactivity coefficients such as void and temperature are adequately negative. As was pointed out earlier the technology is firmly established and proven. Beyond this, extensive precautions have been taken to insure multiple redundancy in the instrumentation, the power supply, the emergency cooling systems, and in the effluent filtering systems.

The reactor operates in a low temperature unpressurized condition, and has no large stored energy content. The worst nuclear incident results in a slow melt-down with no pressure rise. Even if all filtering systems fail under the worst conditions, the dose to a person on the site boundary is minor. Lacking any novel or uncertain feature in the system thus requires only that normal care be exercised in the design, construction and operation of the plant. Additionally, of course, the Bureau, being a large experienced technical institution, it possesses all of the peripherial services needed for security and technical support. See for example Figure 1.3.

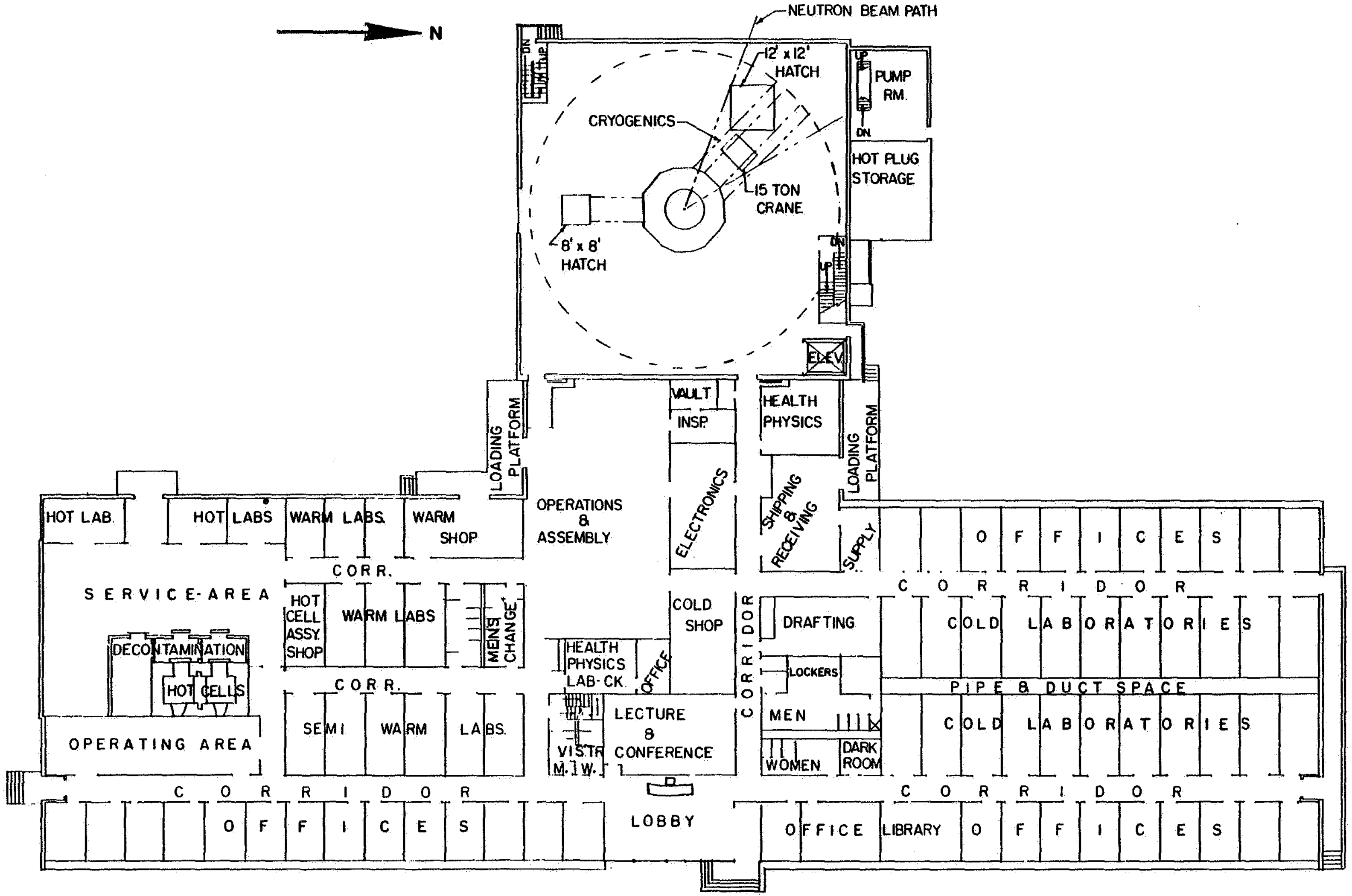

Figure 1.1 Main level floor plan.

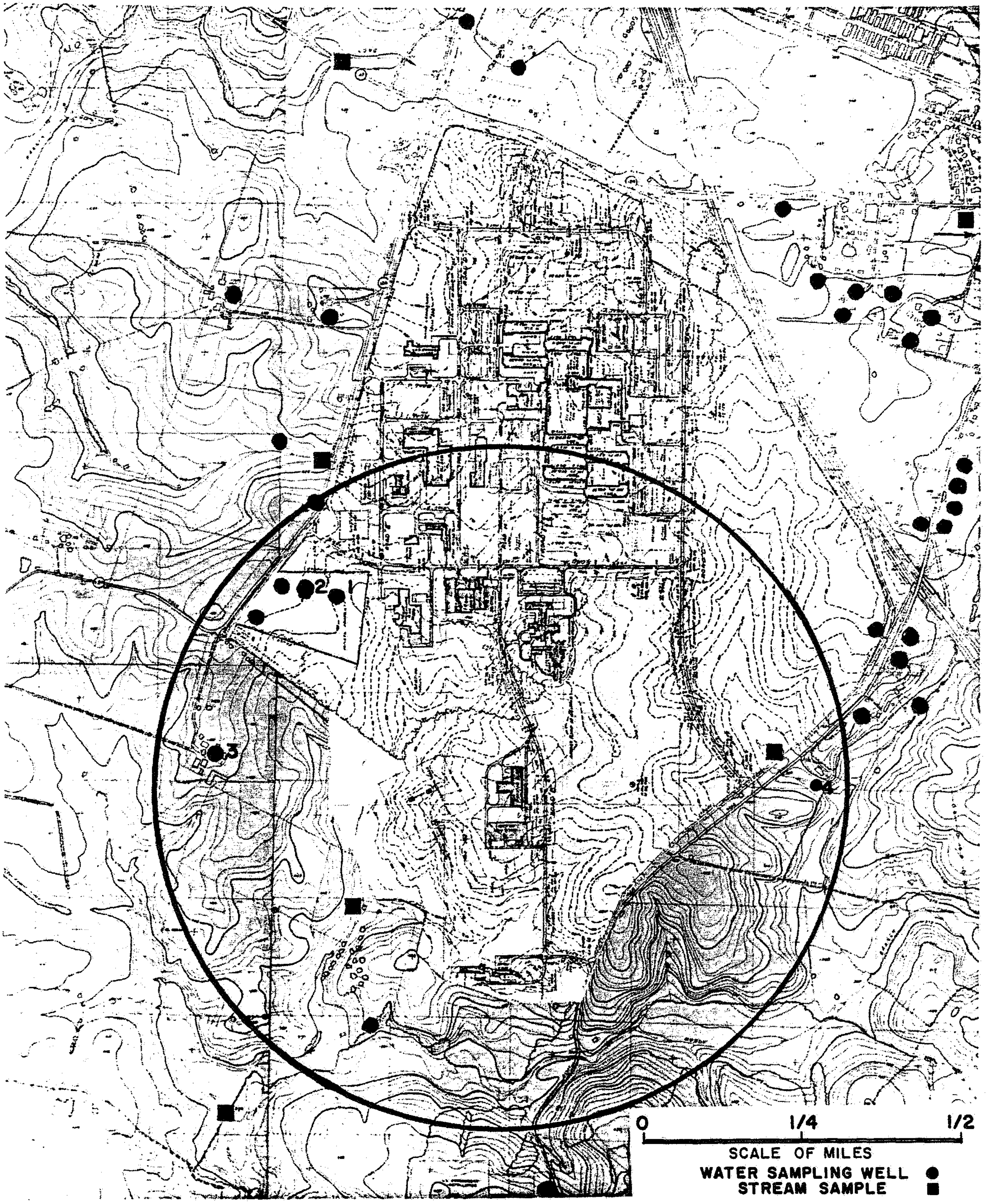

Figure 1.2 Contour map of NBS site and surrounding area.

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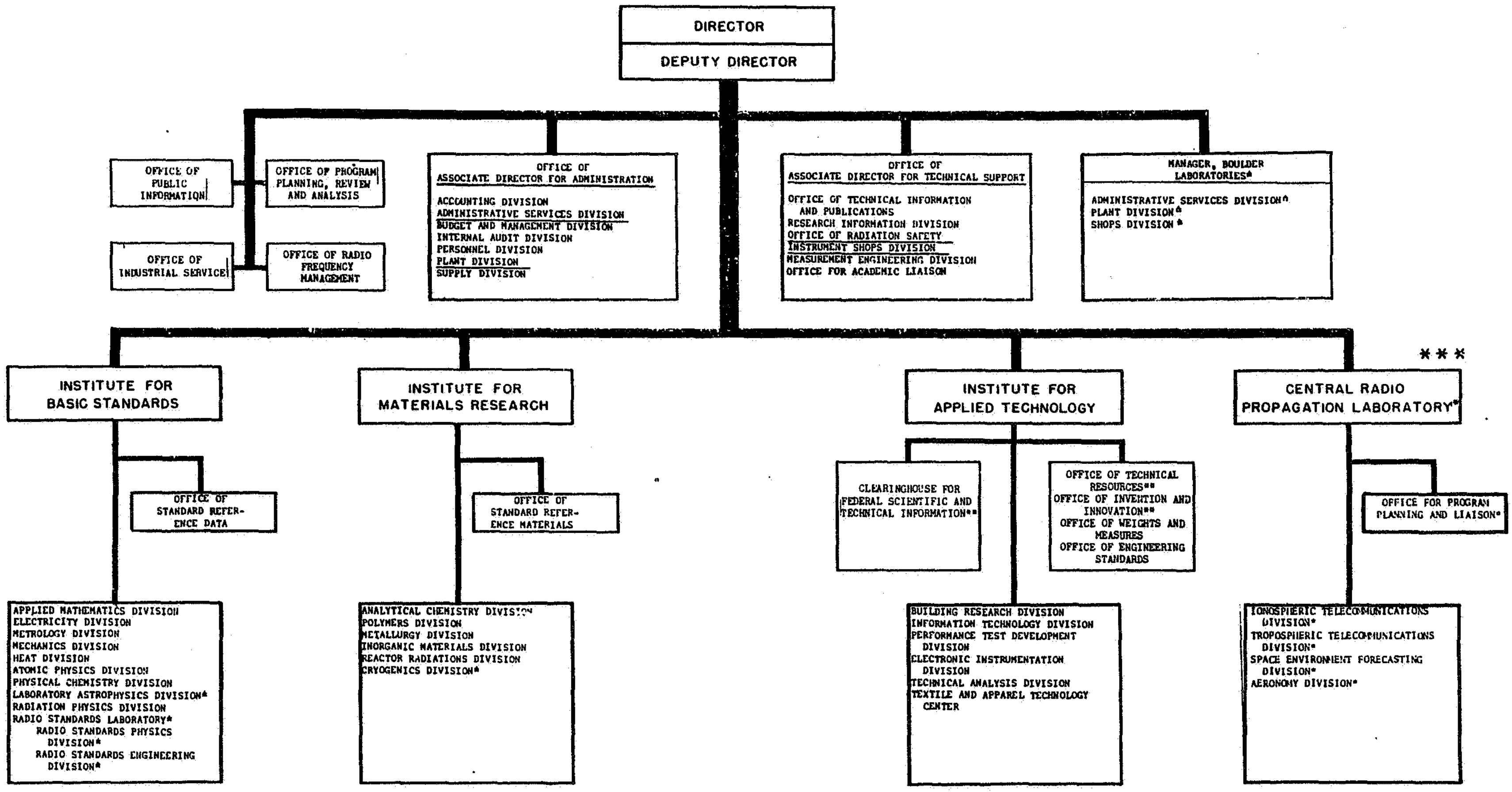

*LOCATED AT BOULDER, COLORADO
**FORMERLY COMPRISED THE OFFICE OF TECHNICAL SERVICES
***TRANSFERED TO ESSA

Herbert W. Klotz
Assistant Secretary for Administration

J. Herbert Hollomon
Assistant Secretary for Science & Technology

Allen V. Astin
Director, National Bureau of Standards

January 15, 1965

Figure 1.3 Organizational chart of the National Bureau of Standards.

SECTION 2. SITE DESCRIPTION

2.1 LOCATION OF FACILITY

The immediate site of the NBSR is as shown in Figure 1.2. The Bureau campus as shown in lighter shade is a 560 acre tract of land bounded on the east by a major highway (70 S), on the north and west Md. 124, and on the southeast by a country road. The south boundaries are private rather than public and presently separate the Bureau from rural residential property. The nearest population centers are Gaithersburg and Rockville located to the east and southeast respectively.

The site, or campus, is located in upper Montgomery County, Maryland, approximately twenty miles northwest of the District of Columbia. The surrounding countryside out to ten miles is shown in Figure 2.1, (located in a pocket on page 3 of the jacket). The drainage, topography, well survey points, and other features shown in Figure 1.2 will be discussed in later portions of this section.

2.2 POPULATION DISTRIBUTION

The immediate daytime population of the campus is 2500 persons, all of whom are under the control of NBS. The 1960 and projected population of the area for the years 1980 and 2000 are given in Table 2.2-1. The data is given as a function of radius and angle from the site. Information in this form is not available for the current and immediate past population distribution. Instead it is given by election district. Table 2.2-2 reveals the 1960 and 1965 census figures for the Gaithersburg election district and the six other election districts that border on Gaithersburg. This group of districts in Montgomery County roughly represents an area around the site of ten miles in radius. A comparison of the figures shows an increase in population from 1960 to 1965 of about 41%, the lower county averaging somewhat higher than the upper county. The angular and radial distributions for 1960 shown in Table 2.2-1 would be scaled upward for 1965 accordingly.

The populations of the District of Columbia and Montgomery County are shown in Table 2.2-3 for the years 1960, 1965, 1980, and 2000.

The closest private housing consists of 5 houses located about 2000 feet or 0.4 mile to the northwest of the reactor. The nearest population center is the town of Gaithersburg (population 6500) located about 1.6 miles from the reactor. The nearest highrise apartment building is currently under construction. It will house 209 apartment units and is 1.8 miles south of the reactor. In a later section it will be shown that even under the worst hypothetical conditions only negligible doses could be given to offsite persons.

2.3 METEOROLOGY

2.3.1 GENERAL INFORMATION. The meteorology of the proposed site shows no unusual characteristics in weather, wind patterns, or atmospheric stability when compared to other locations in this general section of the country. Extensive data are available on the climatology and other meteorological charateristics of the Washington, D. C. area and those aspects pertinent to reactor site selection have been published. (2.1, 2.2) A detailed summary has been prepared for the NBSR site by the U. S. Weather Bureau and has been included in the NBSR 7, Preliminary Hazards Summary Report as Appendix II. (2.3) In this report, it was stated that "there should be no significant differences in the long period averages between the reactor site and Washington in general weather conditions." It was further stated that "since the meteorological frequencies are typical, the population distribution around the site and the character of the reactor design and operation become the controlling factors in evaluating the suitability of the site."

In order to provide direct evidence in support of these statements, a weather station was installed at the NBSR site. The choice of equipment and proper location on the site were made following discussion with the Special Projects Section of the U. S. Weather Bureau. A description of the weather station has been given in NBSR 7C, Preliminary Hazards Summary Report, Supplement C. (2.4)

Table 2.2-1

Radius	Year																	
	1960	200	100	400	--	--	--	--	--	100	--	--	--	--	--	--	--	800
0-1 mi	1980	400	600	1300	500	400	500	500	700	100	--	200	300	200	200	500	400	6800
	2000	600	700	1400	700	500	700	800	900	600	200	200	400	700	500	800	800	10,500
	1960	250	200	1100	1100	1300	100	--	--	100	--	--	--	--	--	100	--	4250
1-2 mi	1980	650	3000	2600	3400	3700	2500	900	1200	1100	100	--	--	--	300	900	900	21,250
	2000	1450	4600	4600	5400	6100	3100	2500	3600	2200	300	100	--	--	400	1600	2400	38,350
	1960	150	100	200	700	100	100	--	200	--	--	100	300	--	--	300	200	2450
2-3 mi	1980	1850	2200	2800	3300	2700	2600	1600	1400	200	200	100	400	100	--	400	1200	23,500
	2000	2950	4200	5100	5500	5200	3200	3100	2000	300	400	300	500	400	200	400	2700	36,450
	1960	100	100	200	300	300	300	700	300	300	300	100	100	--	--	--	100	3200
3-4 mi	1980	500	800	4600	3700	3600	2100	7100	1600	900	500	100	200	200	100	2200	2500	30,700
	2000	1200	1300	6800	6300	5600	3700	11,900	2700	1300	700	300	400	400	400	4900	8400	56,300
	1960	200	100	400	--	--	300	5600	300	500	200	100	400	100	300	400	100	9000
4-5 mi	1980	900	500	400	400	400	800	11,100	4000	1200	200	100	600	200	600	1800	4800	28,000
	2000	2000	1500	900	1500	500	1900	16,800	5300	1500	500	300	700	300	1100	6900	17,100	58,800
	1960	1200	600	900	1200	2900	23,500	44,100	8600	1900	300	100	800	200	700	900		89,300
5-10 mi	1980	5100	3500	3700	4000	9500	17,700	67,400	35,700	4800	300	300	900	200	700	1700		189,300
	2000	15,500	9100	7400	11,400	22,000	64,300	90.500	48,600	11,500	400	300	1100	400	1500	7700		307,000

Total	1960	109,000
	1980	299,550
	2000	507,400

Table 2.2-2

Election District	1960	1965	Ratio
Gaithersburg	8,760	12,048	1.38
Darnestown	3,526	4,336	1.23
Olney	5,320	6,503	1.22
Laytonsville	2,133	2,920	1.37
Damascus	4,488	5,337	1.19
Clarksburg	3,136	3,843	1.23
Rockville	37,896	57,102	1.51
Total	65,259	92,089	1.41

Table 2.2-3

	1960	1965	1980	2000
Montgomery County	340,928	424,535	643,400	995,000
District of Columbia	736,956	803,000	810,000	820,000

The data from the station served two purposes. It permitted a direct comparison to hourly observations taken at the weather station of the Washington National Airport.* The data also permitted direct calculation of atmospheric dilution for the site on a continuous instantaneous basis. A summary of the results of these measurements is presented in Section 2.2.6.

The record high temperature at WNA as recorded through 1957 was 105.6°F which occurred on July 20, 1930. The record low over the same period was -14.9° which occurred on February 11, 1899.

2.3.2 TEMPERATURE. The monthly average maximum, minimum, and mean temperatures for the Washington, D. C. area have been published (2.5) as recorded over a period of years at WNA. The monthly average of the daily mean varies from 37.3°F in January to 78.3°F in July. The extremes as represented by the mean minimum of 29.8°F in January and the mean maximum of 87.3°F in July show the relatively moderate regime of temperature in the Washington area. Summaries from Appendix II of the NBSR 7 report are given in tabular form in Tables 2.3-1 thru 2.3-5 and the "wind rose" shown in Figure 2.2. Data from the weather station at the NBS site and also from a weather station at nearby Germantown, Md. show very high correlation coefficients with the Washington data as discussed in NBSR 7C. (2.4) As stated there, the daily mean temperature at the NBSR site can be predicted within a probable error of 0.6°F from knowledge of the daily mean at WNA.

2.3.3 PRECIPITATION. The mean annual total precipitation as recorded at WNA over a period of years from 1941 to 1957 was 41.12 inches. Yearly mean precipitation data recorded over a ten year period at the nearby Germantown station differed by less than 1/2% from WNA data. Rainfall in excess of 3 inches in 24 hours occurs on the average, once in approximately two years. The average annual snowfall is near 20 inches, and the greatest recorded single fall was 28 inches.

* Hereinafter designated WNA.

2.3.4 WIND PATTERNS. Available wind data is rather extensive. This data indicates a predominance of southwesterly and northwesterly winds. These winds, encompassing a sector of about 130°, account for 50 to 60 percent of all wind directions. East and southeasterly winds are least frequently observed. There is a slight clockwise shift of prevailing winds from summer to winter, resulting in more frequent west and north winds in the colder seasons.

Washington and environs, which subtend an angle of about 60°, would be downwind for winds from about 300° to 360°. Baltimore and vicinity, which subtend an angle of about 30°, are downwind for directions from 230° to 260°. Washington is downwind; i.e., with northwesterly winds, on an average of 25 to 30 percent of the time. These northwesterly winds are more likely to be of a higher speed and associated with relatively unstable lapse rates; consequently, this flow will be favorable for rapid atmospheric dilution.

Wind speeds show the expected seasonal variation with winds in winter and spring somewhat stronger than in the other seasons. Higher speeds can be expected during the late morning and afternoon hours. In the summer, speeds during the afternoon are about twice those of the night hours.

Wind patterns during precipitation differ somewhat from the prevailing flow patterns, particularly the increase in frequency of winds from the north-northeast through the east-northeast, which accounts for 33 percent of all winds during precipitation. See Table 2.3-3.

Winds during inversion show even more preference for certain directions, generally centered on south or south-southwest. Approximately 50 percent of all inversion winds occur within a 70° sector centered at these directions. It should be noted that there is an appreciable frequency of calms associated with the nighttime inversions.

Within 1/4 to 1/2 mile of the site the topography suggests that during calms there will be a tendency for a very light air drift down the slope toward the south and southeast.

Table 2.3-1 Monthly Wind Direction Frequencies (%)

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	CALM
Jan	5	5	4	3	1	1	2	3	11	15	5	3	3	11	14	10	3
Feb	5	3	4	4	3	2	3	4	12	10	4	2	4	11	15	10	3
Mar	4	4	5	5	3	2	4	5	9	8	3	4	5	15	14	9	2
Apr	5	6	5	5	3	3	3	5	11	13	5	4	4	9	11	7	2
May	5	5	8	6	5	3	4	6	10	8	5	4	4	7	9	6	4
Jun	4	4	5	4	3	2	5	8	13	11	7	5	5	7	8	7	3
Jul	4	3	4	3	3	2	4	7	12	14	10	5	5	6	8	7	4
Aug	5	6	5	3	3	3	5	8	11	11	7	4	4	5	9	8	5
Sep	5	5	7	4	3	2	3	6	13	11	7	4	3	4	9	9	5
Oct	8	8	6	3	3	2	3	5	10	12	6	2	3	5	9	10	5
Nov	5	5	5	3	2	2	3	4	10	13	6	4	4	10	11	8	5
Dec	4	4	5	3	2	1	2	3	10	14	7	4	5	12	14	7	4
Ann.	5	5	5	4	3	2	3	5	11	12	6	4	4	9	11	8	4

Table 2.3-2 Monthly Wind Speeds (mph) by Direction

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	MONTHLY AVE.
Jan	12	12	10	9	7	8	8	8	9	11	8	7	10	15	14	13	11
Feb	12	11	10	11	8	8	8	9	10	12	8	7	11	16	13	12	11
Mar	11	11	10	12	10	9	11	11	11	13	9	11	12	17	14	13	12
Apr	12	13	12	11	10	11	9	11	11	13	9	10	10	14	13	13	12
May	10	11	11	10	9	9	9	9	9	9	7	10	8	12	11	11	9
Jun	9	11	10	10	10	8	9	9	9	9	8	8	9	11	10	10	9

Table 2.3-2 (con'd)

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	MONTHLY AVE.
Jul	8	10	10	10	9	9	9	9	9	9	8	8	7	9	9	9	8
Aug	10	10	10	9	8	9	9	9	9	9	7	7	7	8	8	9	8
Sep	9	10	10	10	9	8	8	9	10	10	6	7	7	9	9	9	9
Oct	10	11	11	10	10	10	8	8	9	8	6	7	10	11	11	10	9
Nov	10	10	8	9	9	10	9	9	10	10	7	7	10	15	12	10	10
Dec	11	11	9	9	8	10	9	10	9	10	7	7	10	14	12	11	10
Annual Avg.	10	11	10	10	9	9	9	9	10	10	8	8	9	13	11	11	10

Table 2.3-3 Annual Wind Direction Frequencies (%) During Precipitation

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	CALM
5	13	13	7	4	3	4	6	8	9	4	3	2	4	7	7	4

Table 2.3-4 Annual Wind Direction Frequencies (%) During Inversions

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	CALM
10AM EST	2	4	4	1	2	5	4	24	16	10	7	1	4	5	3	6	4
10PM EST	2	2	3	2	1	1	2	6	14	22	11	4	3	4	6	4	14

Table 2.3-5 Inversion Frequency (%)

Inversion Base	0-499 ft.	500-999 ft.	1000-1999 ft.	Total
10 AM	7	6	17	30
10 PM	40	6	8	54

2.3.5 EXTREME WEATHER CONDITIONS. The probability that the reactor site will be affected by very heavy precipitation, high winds, or tornados is essentially the same as Washington. Sustained high wind speeds are infrequent in this area, and speeds in excess of 60 miles per hour are almost invariably associated with severe thunderstorms or with hurricane centers passing near the area. In the 41 years, June 1905 to 1945, there were only two occurrances of speed in excess of 50 miles per hour which were sustained for 5 minutes or more. A peak wind gust of 100 miles per hour was recorded on June 9, 1928 during a violent thunderstorm.

Tornados are unusual in this portion of the country. In the 30 year period from 1916 to 1945 there have been only 13 reported tornados in the District of Columbia and the three adjacent counties. Thus the occurrances of a tornado at the reactor site is a possibility but has an extremely low probability.

Numerous tropical hurricanes, however, have passed over or near Washington, D.C. More than 60 hurricanes have influenced the area between 1899 and 1958. Eight such storms caused winds in excess of 40 miles per hour; nine resulted in a total rainfall in excess of 3 inches. One such storm deposited 8.67 inches of rain, 7.31 inches of which occurred in 24 hours.

The reactor site is, however, no more vulnerable to these weather extremes than any other location in the Virginia-Maryland area between Appalachian foothills and the Chesapeake Bay.

2.3.6 WEATHER STATION. The weather station has been described in NBSR 7C. Data was recorded for the six weather parameters, wind direction and speed, barometric pressure, temperature, humidity, and rainfall. Data from the instruments was such that in all cases hourly observations could be obtained for comparison with the published data from the Washington National Airport.* The data was treated first by calculating a correlation coefficient between the two sets of data formed by the twenty-four observations of a given day at each of the two stations, NBSR and WNA. Samples of these calculations are shown in NBSR 7C. In addition to treating of the data on an hourly basis a comparison of daily averages for the two stations for the month of April, 1962 was made for each of the parameters recorded except precipitation or rainfall. For precipitation, a source (2.6) of weather data was found to permit comparison of the WNA data with an older station not now operating, located near Germantown, Maryland, a distance of approximately four miles from the NBSR site. Both temperature and rainfall data were available for periods as long as seventeen years from both of these stations.

The results of the correlation coefficient calculations show that very high correlations exist for barometric pressure, temperature, relative humidity, percipitation, wind direction, and wind speed.** The only parameter to show a low correlation was instaneous wind direction. This was not unexpected, however, when it was considered that the average range of wind direction over fifteen minute observation intervals was greater than 80 degrees. Thus, to correlate wind direction changes between the two stations, it would require consideration of the wind direction relative to the geographical location of the two stations and the time lags involved for wind shifts between the two stations depending on wind speeds.

The existance of instantaneous wind speed and wind direction data from the weather station permitted calculation of wind dispersion data directly applicable to the reactor site. The dilution factors calculated in a hazard evaluation are normally based on Sutton's equation. In order to apply this equation, it is necessary to make assumptions concerning the magnitude of certain diffusion coefficients which enter the equation. In general, these coefficients are difficult to establish by experiment; and it is the practice to use more or less general coefficients for any given site provided there are no unusual characteristics in the local weather. The calculations associated with the direct derivation of wind dispersion data from the measured wind parameters are discussed in NBSR 7C, p. 22 et. seq. It was shown that dilution factors in excess of 1200 can be expected for the NBSR site.

The data from the NBSR weather station then served to provide direct evidence in support of the assumption that the weather conditions at the site are not different from those prevailing at WNA. The data has also confirmed the calculations for dilution factors for the site.

2.3.7 DIFFUSION PARAMETERS. The atmospheric diffusion parameters chosen to characterize the wind dispersion at the site have been selected after numerous consultations with the U.S. Weather Bureau, Office of Meteorological Research, Special Projects Section.*** The diffusion parameters which are expected to obtain depend on the stability conditions of the atmosphere.

These conditions can vary through a wide range and must therefore be chosen with consideration of several aspects of a potential accident situation. As discussed in the Hazards Analysis, Section 13 of this report, it is particularly necessary to consider correlation between stability conditions and barometric pressure variation before a choice of parameters is made.

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* These observations are published monthly by the Ashville, North Carolina Records Office of the Weather Bureau, U.S. Dept of Commerce. For the first month of observation, April 1962, data was obtained directly from records supplied by the WNA weather office. The NBS wishes to acknowledge the generous cooperation of Mr. Kenneth Norquest of the airport weather staff.

** Wind speed correlation data was not originally reported in NBSR 7C but subsequent calculations also showed this parameter to yield very significant correlation.

*** The cooperation and assistance of Donald Pack, Isaac VanderHoven, and David Slade are gratefully acknowledged.

In general, however, for calculational purposes, at least three conditions have been considered which are usually called strong inversion, weak inversion, and neutral. The parameters used to characterize these conditions in a Sutton* equation are given in Table 2.3-6.

Table 2.3-6 Sutton Equation Diffusion Parameters

	Strong Inversion	Weak Inversion	Neutral
n	0.5	0.33	0.25
c^2	0.028	0.06	0.09
u (meters/sec)	1.0	2.0	3.0

While the basic equation of Sutton is widely recognized as adequate, there has been considerable concern with methods of determining parameters. In practice, these parameters are determined empirically from actual measurements of wind dispersion of stack plumes. The tabulated values are of the measured type and represent currently accepted values of diffusion parameters and can be considered to apply beyond distances for which the original Sutton data are valid as well as for a wide range of stable and unstable conditions.

As stated in Section 2.3.6, data from the weather station at the NBS site afforded an opportunity to study wind dispersion characteristics at the site itself. A modified form of the Sutton model for calculation of atmospheric dilution could be used with the data that was taken. A discussion of the theoretical basis of the modifications of the model is given in the literature (2.7, 2.8). A discussion of the results of these calculations has been presented in NBSR 7C. In summary, it was shown that the NBSR stack can be expected to give reasonable assurance of dilution factors in the range of one thousand or more even over short time intervals.

2.4 GEOLOGY AND HYDROLOGY

A summary of the geology and geohydrology of the NBSR site was prepared for NBS by the U. S. Geological Survey. The entire text of this report was included as Appendix III of NBSR 7. A second report concerning specifically the rate of movement of ground water in the vicinity of the NBSR site was included as Appendix II of the NBSR 7C. The following discussion is excerpted from the contents of these two reports.

2.4.1 GEOLOGY. The reactor site is in the Piedmont physiographic province. The rocks underlying the site consist entirely of the Wissahickon formation of Precambrian or early Cambrian age. The Wissahickon consists of a great thickness of schist and phyllitic schist. The formation is derived from ancient sedimentary strata which have been greatly compressed, resulting in substantial lithologic alteration. Although the total thickness of the formation is not known in this area, it is believed to be several thousand feet thick.

The Wissahickon has been divided into two facies or divisions on the basis of its mineralogic character; these are the oligoclasemica facies and the albite-chlorite facies. The latter unit underlies the Gaithersburg area and consists of an albite schist or gneiss interbedded with layers of chlorite and/or muscovitemica. The direction of cleavage in the schist is variable, but near Gaithersburg it is commonly north to northeast. Quartz veins and dikes are common, and, as they are more resistant to weathering than the schist, they occur in the soil and subsoil (saprolite) as irregular-shaped pebbles, cobbles or boulders.

Generally, three distinct zones of earth material are present. At the upper surface is the true soil zone which may range from a few inches to as much as one and one-half feet thick. At the NBS reactor site the soil is "grayish-brown to yellowish-brown Chester loam" (Maryland Geological Survey and U. S. Bureau of Soils, 1916). Beneath the true soil lies the subsoil, or the saprolite zone. This is the zone of weathered or decomposed rock. On the basis of data from 21 borings at the site, the thickness of the saprolite zone at the

* For a discussion of this equation and its applicability, refer to Meteorology and Atomic Energy, AECU 3066 (1955).

site ranges from 20 to 65 feet and averages about 36 feet. The saprolite is usually soft, sometimes friable in texture, and importantly, it is commonly more permeable than either the soil or the underlying rock. Its porosity and permeability are such that ground water is stored in and moves through interconnected intergranular pore spaces. It contains substantial amounts of silt and clay-size particles.

The saprolite grades downward into its parent material, the unweathered bedrock, at depths averaging several tens of feet in this area. Although the quantity of ground water moving through the intergrannular spaces of the fresh crystalline rock may be considered nil, ground water does move through interconnected fractures and bedding planes down to depths of a few hundred feet beneath the land surface.

2.4.2 TOPOGRAPHY AND DRAINAGE. The topography in the vicinity of the reactor site is undulating and the relief is moderate. Elevations range from 300 feet above mean sea level in the valley of Muddy Branch to 520 feet at Gaithersburg. On the site itself, the range of elevations is from 365 feet to 465 feet above mean sea level. The reactor is located at an elevation of approximately 420 feet.

The site is generally in the Potomac River watershed. Drainage is to the south and to the west. Drainage to the south is by Muddy Branch and to the west by Long Draught Branch or Seneca Creek. Both streams are tributaries of the Potomac River. Muddy Branch, the easternmost of the tributaries, enters the Potomac near Katie Island at a point about 5.5 miles above Lock 20 at Great Falls.

2.4.3 GEOHYDROLOGY.

2.4.3.1 Hydrology. The source of the ground water in the vicinity of the reactor site, and elsewhere in the Maryland Piedmont, is local precipitation which averages about 41 inches per year. A zone of saturation is maintained in the sub-soil by that precipitation which neither runs off directly, nor evaporates. Generally, the upper surface of the zone of saturation, or water table, is a subdued replica of the topography of the land surface. Hydraulic gradients exist in this zone which result in the general movement of ground water to the streams. The rate of movement of water is variable. In the sub-soil, or saprolite zone, the rate may be on the order of 0.1 to 1 foot per day.

Water-table contours based on measurements of water level made January 20, 1961 in foundation borings in the vicinity of the reactor building are shown in Figure 2.3. Bore holes 7-A, 8-A, 10-A and 3-A are at the corners of the building. Depths to water ranged from 1.67 feet in hole 2-A to 23.0 feet in hole 3-A. All the water levels were in subsoil or decomposed rock. The pronounced difference between the north westward gradient and the south westward gradient appears to be related to structural features of the rock, because the schistosity has a north easterly trend in rock outcrops west and southwest of the reactor site. Ground-water flow parallel to the schistosity would meet with less resistance than perpendicular to the schistosity. Thus it can be inferred with relative confidence that beneath the reactor building the ground water flows in a generally southwestward direction.

Although a stream west and northwest of the reactor site is 500 to 600 feet nearer than one to the southwest, it seems unlikely that the path of easiest movement of ground water would be directly across the schistosity. Ultimately, however, ground water does move to a stream west of the site, because it is the principle drainage for the reactor site area. The presence of a perennial stream between the reactor site and three off-site wells to the west (see Section 2.4.3.2) effectively eliminates the possibility that cones of depression caused by pumping these wells could extend beneath the reactor building.

2.4.3.2 Well and Water Use Survey. The major use of ground water within a one mile radius of the reactor site is for domestic and farm supplies. A survey of existing wells in the vicinity of the site was made for the NBS by E. G. Otten of the U. S. Geological Survey in the course of his preparation of a report "Geohydrology of a Proposed Reactor Site near Gaithersburg, Maryland" which was incorporated as an appendix in NBSR 7. In order to check and up-date the survey for the report, NBSR 7C, Dr. A. Schwebel, Health Physicist at the NBS, undertook an independent survey of the wells adjacent to the site. His records show thirty-one wells on the perimeter of the site within a one mile radius which substantiate and augment the record in NBSR 7. Shown in Figure 1.2 is an

aerial survey map on which all wells within a distance of one-half mile from the reactor site are shown.

Since November 1963, Dr. A. Schwebel has supervised a routine sampling and analysis program for well and ground water radioactivity levels. Following the recommendations of the U. S. Geological Survey, water from six stream locations shown in Figure 1.2 has been checked weekly. Water from a total of thirty-five wells is checked monthly to provide a comprehensive record of the background activity.

Processing is done on 8 gallons of filtered water, counted and automatically tabulated by a General Measurement Gamma/Flow 3910 counter system. The detector is a 3 x 3" NaI (Tl) crystal at the center of a tank shielded by 4" of lead. The gamma spectrum can be grouped into 20 pulse height channels. First the natural gaseous Radon content of the water is determined. After appropriate aeration, a second count indicates any long-lived soluable radioactive constituents. The equipment has been calibrated with standard solutions of isotopes for Radium daughter products and for Cesium-137. Calculations for photons of other energies have been carried out. Concentrations down to about 10 pCi per liter or 1×10^{-8} μCi per ml are detectable. Maximum permissible concentrations of radionuclides released into water as listed in Table II, Appendix B of Title 10 Code of Federal Regulations, Part 20 (10 CFR 20) are greater than 10^{-8} μCi per ml, in most cases by several orders of magnitude.

The records taken to date show relatively high natural radon concentrations in fresh well samples. Data for the four wells shown on the map, correct to pick-up time, average

1) 2100 pCi (Rn-222)/1
2) 2100
3) 1900
4) 1500

Concentrations in other wells range between 400 and 3300 pCi/1. The month-to-month variation of activity from any individual well is about ±20%. The stream water contains lower and more variable concentrations of radon depending on stream flow history. Values for all six stream sampling points average 100 pCi/1 ±200%. Following removal of the radon, natural activity from no other isotope has been recorded. A continuing sampling program will assure that the National Bureau of Standards Reactor is not responsible for radioactive contamination of local wells and ground water.

2.5 SEISMOLOGY*

The reactor site is in the Piedmont physiographic province (refer to Section 2.4). The Maryland Piedmont is a region of comparative crustal stability and most of the quakes recorded during the past 163 years have been of light intensity. A record of the important earthquakes in the middle Atlantic region since 1802 has been published by Freeman (2.9) and was included in the appendix of NBSR 7. These quakes were characterized as to intensity by comparison to the Rossi-Forel scale which was discussed in NBSR 7. It was noted that none of the quakes reported was observed to have an epicenter in Maryland. It is therefore unlikely that all of the shocks were felt in the Gaithersburg area. Although the evidence is not conclusive and any prediction is uncertain at best, Freeman states (p. 136) that only one destructive earthquake may be expected per century in the Atlantic region, comprising 600,000 square miles and extending from Quebec to Florida. The likelihood that the epicenter of a destructive quake would be in or near Montgomery County seems remote.

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* The following discussion is based on the statements of E. G. Otton of the U. S. Geological Survey as they appeared in Appendix III of NBSR 7.

References

(2.1) Environmental Aspects of the Naval Research Laboratory, NRL 4308, February 23, 1954.

(2.2) Meteorology for the Proposed Location of the Army Packaged Power Reactor, Fort Belvoir, U. S. Weather Bureau, June 1954.

(2.3) NBSR 7, Preliminary Hazards Summary Report, USAEC Docket 50-184.

(2.4) NBSR 7C, Preliminary Hazards Summary Report, Supplement C, USAEC Docket 50-184.

(2.5) Local Climatological Data, Washington National Airport, Washington, D. C. 1957, U. S. Department of Commerce, Weather Bureau.

(2.6) Local Climatological Data 1957, Washington National Airport, Washington, D. C., Climatic Summary of the United States-Supplement for 1931-1952, Maryland and Delaware, No. 15, U. S. Department of Commerce, Weather Bureau.

(2.7) F. Pasquill, Meteorological Magazine 90, 33 (1961).

(2.8) F. A. Gifford, Nuclear Safety 1 No. 3 (March 1960); 2 No. 2 (December 1960) 2, No. 4 (June 1961).

(2.9) John R. Freeman, Earthquake Damage and Earthquake Insurance, 1932, McGraw-Hill Book Co., New York, N. Y. pp 126-129.

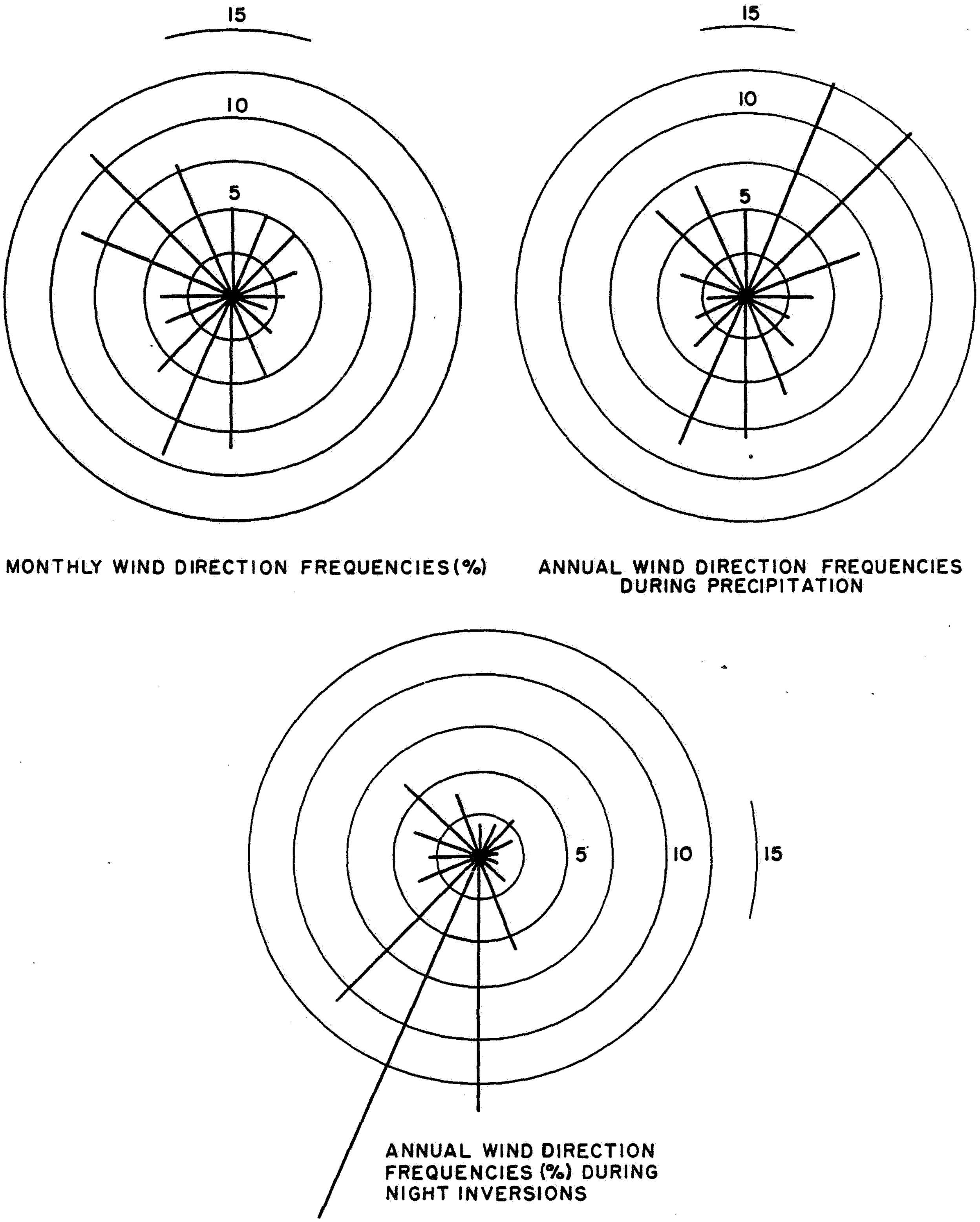


Figure 2.2 Annual wind rose.

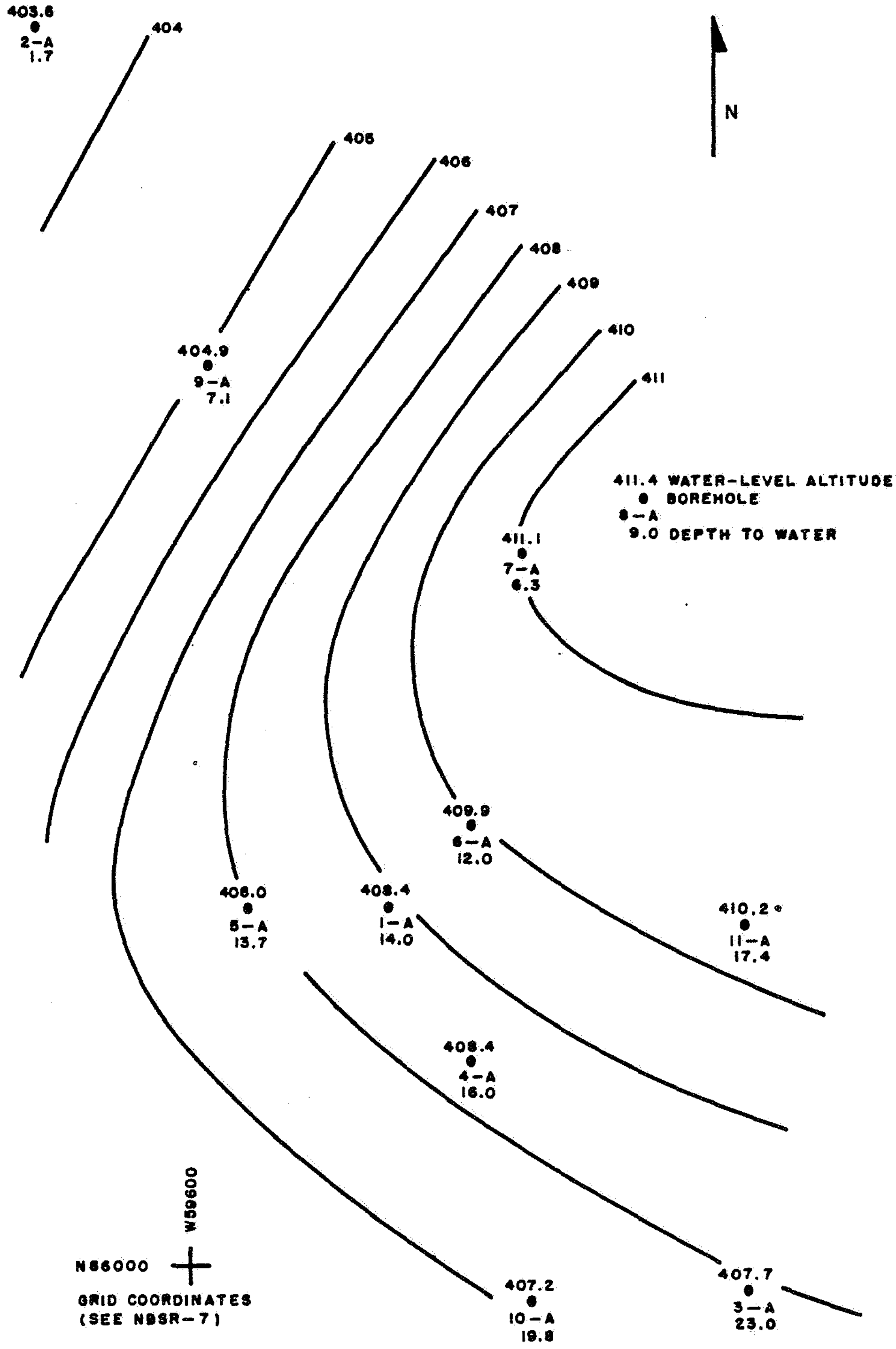


Figure 2.3 Water table contours made at NBS Reactor site, based on water level measurements made in core holes on January 20, 1961. Datum is mean sea level. Scale: 1 inch = 80 feet.

SECTION 3. REACTOR BUILDING AND CONFINEMENT SYSTEM

3.1 GENERAL DESCRIPTION

The Reactor Building is a 90' x 90' square concrete structure designed to house the reactor and confine the results of any creditable accident which might occur. In addition to housing the reactor, it provides space to carry out the scientific programs for which the reactor was designed. Although space is provided within the confinement building for both beam hole and in-core irradiation experiments, the additional laboratories and offices required to support the scientific programs are not located within the confinement building.

3.1.1 DESIGN BASIS

3.1.1.1 General Structural Criteria. It is not considered creditable that any accident would lead to significant over pressure within the building. Thus, the building is not a steel containment vessel, but rather a confinement structure designed to meet the more normal structural requirements of wind and snow loadings. Internally, it is designed to take the large loads imposed by the reactor itself and the heavy shields required for the experiments. As will be discussed in Sections 10 and 13, in several cases, the heavy concrete walls and floors necessary to support the structural loads also serve as radiation shields.

Under normal operating conditions, the reactor building will operate at a pressure slightly below that of the adjacent laboratory building. The two access corridors from the laboratory building to the reactor building are equipped with two sets of doors each to facilitate in the ventilation control of the reactor building. A third door permits access to the reactor building elevator from the basement of the laboratory. In addition to the normal access doors for ventilation control, all three openings are equipped with inflatable gaskets which seal automatically closing doors under emergency conditions. There are two other doors in the confinement building; both opening directly outside. One is a fire exit and is normally never used. It is sealed at all times, and if opened, it shuts and seals again automatically. The other is a large manually operated truck door which is normally sealed with inflatable gaskets, and will not be opened during reactor operation.

Although no overpressure is expected, and the underpressure during normal operation is nominal, it is possible under emergency conditions, when the building is sealed, for rapidly rising atmospheric pressure to create an abnormal external pressure. The structural design has included this effect. In addition, a pressure relief valve has been incorporated which will prevent any detrimental pressure differential from developing across the building walls or roof.

3.1.1.2 Experimental Programs. The prime purpose of the reactor is to provide a source of neutrons and other radiations to further probe the nature of physical phenomena and the basic structure of materials through experimental programs. These programs necessarily cover a broad range of possibilities, and by their very nature of exploration can't be catalogued in any detail in advance. Many of the experimental techniques, however, are well developed and can be briefly outlined.

The beam holes will primarily be used for various types of spectroscopy. Crystal spectrometers, time-of-flight systems, or combinations of both will be the most common equipment used. The beam holes are all located a few feet above the first floor level, and all of this type of research is carrier out at this level which is serviced by an annular crane. Most of these experiments will use beams extracted from the reactor, and very little interaction between these experiments and the reactor core is anticipated.

Four pneumatic tubes have been provided and an extensive program of activation and tracer analysis is being planned. The rabbits are sent and received from laboratories in the basement which are specially equipped for this type of work.

The top floor of the building, which is flush with the top of the reactor shield, provides space for in-pile irradiation experiments as well as being the main area of reactor operations. Holes through the top shielding provide access to several positions in the core and in the reflector. Radiation damage studies and similar programs are planned for these facilities. The programs do not envisage the irradiation of systems or large structures, but rather will limit themselves to the study of small samples which in many cases will be carefully instrumented. The purpose of these experiments is to study the properties of materials under irradiation and not to study whole systems or irradiate structures to destruction.

Each experiment performed in the confinement building will be carefully reviewed with respect to its interaction with the reactor core and all other safety aspects.

3.1.1.3 Emergency Provisions. The worst credible accident at the NBSR is a loss of water accident resulting in a slow meltdown of the core. This would not cause an over-pressure in the building nor lead to a sudden release of fission products. As the fission products leaked out of the reactor shield, they would be sensed by a radiation detector which would initiate emergency conditions. The gasketed doors and shut-off valves would automatically close and seal, the normal ventilation system would be shut off, and the emergency ventilation system started. The emergency ventilation system exhausts air from the Reactor Building through absolute filters and a carbon scrubber, and then up the stack. The operation of these emergency exhaust blowers are controller by a pressure differential sensor across the walls of the confinement building. The system is designed to maintain the internal pressure at 1/4" of water pressure below atmospheric to assure that any leakage through the confinement building walls is from the outside in. Thus, the average exhaust rate of the emergency system is determined by the leakage from the outside through the containment walls to the inside.

Leak rate requirements for the building in principle, are therefore set by the maximum exhaust rate which can be tolerated from an emergency hazard situation. Actually, as will be discussed in a later section, the building was designed to be as tight as practical with a conservative upper limit on the allowed leak rate set at 24 cfm for a pressure differential across the walls of 1" of water. This leak rate is several times smaller than the rate at which the building would have to be exhausted in order to keep up with a rapidly falling barometer. On the other hand, a rising barometer would require very little or no gas to be exhausted from the building. While the emergency exhaust system is maintaining the proper differential across the confinement building walls, a large 5000 cfm internal recirculation system is filtering the confinement building atmosphere through absolute filters and a carbon scrubber. In this way the fission products in the air within the building are rapidly being removed.

Thus, the Reactor Building is not designed to be an absolute containment vessel, but rather to confine the results of an accident and to control the rate and location at which the fission products are released.

3.1.2 BUILDING DESCRIPTION

3.1.2.1 Introduction. The Reactor Building is shown in Figures 3.1 through 3.5. The building has three main levels, the basement, the first floor which serves the reactor beam holes, and the second floor which provides access to the top of the reactor. The ventilation systems in these three areas are separate so that there is no mixing of the air from one floor with that from another. In this way it is hoped to limit the spread of any airborne contamination which might appear in any one area.

Figure 3.1 shows the building elevation. The basement floor elevation is at 403' - 0". The main part of the basement, including the process room, and the pool area, has a 20 foot ceiling. The floor thickness above the process room is 5' of normal concrete in order to shield against the nitrogen 16 radiation which is present in the process room during reactor operations. Similarly, the walls of this room are 4' and 5' thick. Part of the basement has been divided into two levels. A lower level is used for radiological laboratories and a counting room. A mezzanine area above is used for the heating ventilation and air conditioning equipment which serves the confinement building. The main floor, which is at ground level, is at elevation 428' - 0". The reactor beam tubes are on this level and the

area is serviced by a 15 ton annular crane. The hook clearance on the crane is 15' and the height of the room to the underside of the ceiling beams is 19'. The second floor is flush with the top of the reactor at elevation 450' 4-7/8". This area is completely open except for the control room which is located on the east side of the floor. The area is serviced by a 20 ton crane with a hook clearance of 24'. The height of the room to the underside of the roof beams is 33'. Two enclosed stairways on opposite corners of the building lead from the basement to the second floor. In addition to this, all floors are served by a service elevator located in the northeast corner of the building. The roof is flat with a slight slope for drainage and is surrounded by a parapet whose top is at elevation 489' 4-5/8", which is 1' 1-5/8" above the high point of the roof. The stack runs up the side of the building to an elevation of 524' - 0", about 35' above the top of the parapet. It has a double flue, one of which serves the confinement building and the other serves the adjacent laboratories.

3.1.2.2 Basement Level. Figure 3.2 shows the basement plan view. About half of the basement area is devoted to the process room which is surrounded by a thick concrete wall. Access to this room is through a steel shielding door from the pool area. Directly under the reactor and in the process room is the subpile room which is itself a shielded area surrounded by 3' of normal concrete. This area and the heavy water equipment area are surrounded by a 2' high curb. The heavy water systems are confined to this area so if any of them should develop a severe leak, the water would be retained by the curbs and flow to the sump pit where it could be recovered. In the northwest corner of the process room is a pit which is about 13' deep and contains the main heavy water storage tank. This area is covered normally by removable concrete slabs. The process room contains all the process equipment for the reactor, including heat exchangers and pumps, except for the secondary cooling pumps which are located in an adjacent structure outside the reactor building.

On the south side of the basement is the storage pool area. The pool will be used to store spent fuel elements until shipment. A canal leading from the subpile room allows fuel elements to be transferred directly from the reactor vessel into the storage pool without the use of transfer casks. The pool will also be used for irradiation studies, using irradiation from spent fuel elements.

On the east side of the basement there are two radiological laboratories and a counting room. These laboratories are primarily for use in conjunction with the pneumatic tube system. There are four separate tubes, each of which may be operated from either laboratory. A separate small pneumatic system allows samples to be sent from either radiological laboratory directly to the counting room.

Directly above the counting room and radiological laboratory area is a mezzanine area shown in Figure 3.3. This area contains most of the heating, ventilation and air conditioning equipment used in the Reactor Building. This figure also shows a small monitoring room which is part of the confinement building. It is used to monitor certain of the process equipment located in the process room. Adjacent to this room and just outside of the confinement building is a fan room for the stack. Access to this fan room is from the secondary pump room. This figure shows one of the doors penetrating the confinement building leading from the basement to the adjacent laboratories at elevation 412' into the elevator. This door is normally closed and sealed but may be operated locally, closing automatically after being opened.

3.1.2.3 First Floor. A plan view of the first floor of the Reactor Building, elevation 428' - 0", is shown in Figure 3.4. This floor is at the same level as the adjacent laboratories and access from the laboratories to the Reactor Building is through the two doors shown on the east side of the confinement building. The remaining doors penetrating the confinement building are also shown in this figure. The truck door is shown on the south side; it is normally closed and sealed. In the southwest corner is the emergency fire exit which is also normally closed and sealed.

Of the two doors on the east side of the main floor, the northern one opens into a corridor leading to the cold laboratory wing of the building and the other opens into a large room primarily intended for the assembly of experimental and reactor operation equipment. This room is partially shown in Figure 3.4. Access to the unirradiated fuel element storage area is from this assembly area. The storage area is divided into two rooms. One is the actual fuel element storage room and the other is a clean room for final inspection.

The reactor, about 20 feet across, is located in the center of this floor. A biological shield runs up to the ceiling and serves as a support for the inner rail of the annular crane which services the area. Two hatches are shown in the floor; the one to the south is directly over the pool below; the other one is over the process area and provides access for the installation or removal of heavy pieces of equipment.

A neutron window through the west wall of the confinement building is provided for a time-of-flight experiment although there are no immediate plans for its use.

Storage for radioactive plugs is provided on the north wall of the building. Plugs, collimators and similar items may be stored horizontally in this facility.

3.1.2.4 Second Floor. A plan view of the second floor of the Reactor Building is shown in Figure 3.5. This floor is level with the top of the reactor top shielding plug. A trench in the floor around the shielding plug provides access to many of the utilities provided for in-pile experiments. This trench is normally covered with steel plates providing a flush surface. Two large square hatches are provided in the floor for access between the first and second floor. A small round hatch is located on the northwest corner. This hole is located on the outside of the annular crane below and is provided for the eventuality that some particular experiment may require a long vertical shaft. On the north side are provided some vertical plug storage holes for radioactive plugs or experiments which may be taken out of vertical holes in the reactor.

The control room is located on this floor along the east wall. It consists of three sections; the main section where the control panel is located, a small office to the south, and a parts storage room and work area for electronic maintenance to the north. The control room is provided with windows looking out over the reactor itself.

In the southeast corner may be seen a ladder leading up to the emergency cooling storage tank which is located up near the ceiling. Along the wall beneath this tank are located some of the emergency cooling system's valves.

Most of the reactor operations such as fuel element transfer, etc., will be performed on this floor. It is also anticipated that all in-pile experiments will be operated from this floor and all their associated equipment located on this floor. Conduit penetrating the floor at several locations along the walls, make it possible to put some of the electronic equipment for the beam port experiments on this floor if space limitations on the floor below make it desirable.

3.1.2.5 Dimensions and Net Volumes of Confinement Building. The internal dimensions of the Reactor Building are 90' x 90' x 84' high. This is a gross volume of 680,000 cubic feet. The volume of the internal walls, floors, and the reactor biological shield is about 82,000 cubic feet, leaving a net volume of 598,000 cubic feet, including the stairway wells and the elevator shaft. The top floor is 292,000 cubic feet; the main floor is 159,000 cubic feet; and the basement is 147,000 cubic feet, divided up between the process room of 70,000 cubic feet, the laboratory and pool areas of 45,000 cubic feet, and the HVAC equipment room of 32,000 cubic feet.

3.1.3 OCCUPANCY OF BUILDING. In addition to the normal operating staff, numbering between five and ten people per shift, the Reactor Building will be occupied by experimental personnel. Although as many as one hundred people may have business in the Reactor Building sometime during the day, it is not anticipated that the average occupancy will be that high since office and laboratory space is provided in the adjacent building. The average occupancy is more likely to be about 30 people during a normal working day. Since much of the experimental equipment is automated it is expected that the number of people in the building during the night and weekends and holidays will be much smaller than this.

The only normal exclusion area in the Reactor Building will be the process room, which is kept locked at all times during reactor operations. During shutdown, access to this room will be permitted only to properly authorized personnel. Since the fuel elements are transferred completely internally within the reactor shielding, it is not anticipated that any special exclusion area will be required during fuel transfer and reactor maintenance.

3.2 STRUCTURAL DESIGN

3.2.1 LOADINGS

3.2.1.1 Internal Pressure. The maximum internal pressure to be seen by the NBSR confinement building structure is 7 to 8 inches of H_2O. This pressure occurs during the building leak test when the building is purposely sealed and external fans are employed for pressurization. The building is designed to withstand this pressure with the resulting stresses remaining well within the bounds of all applicable codes.

3.2.1.2 External Loads. External loading of the NBSR confinement building results from wind, snow, earthquake, soil pressures, and hydrostatic pressures. In accordance with good design practice and the applicable codes, the design wind load is 25 psf (approximately 100 mph) and the design snow load is 25 psf. Although the structure was not specifically designed to withstand a given severity of earthquake, the professional engineers responsible for the building design have stated that the building is capable of withstanding a force 7 earthquake (3.1). Soil pressures and hydrostatic pressures were calculated on the basis of the data which were obtained from the analysis of the site test borings. A soil pressure of 95 psf was used for design of the structure walls and the storage pool and canal side walls.

3.2.1.3 Internal and Structural Loads. The internal and structural loads in the NBSR confinement building result from the dead loads of the reactor proper, the building and reactor operational equipment and the structural components of the building itself as well as the live loads of personnel, experimental equipment, and operational equipment. The dead load tabulation for the building, exclusive of the reactor is given in Table 3.2-1. The dead load tabulation for the reactor is given in Table 3.2-2. Live loads used in the design are tabulated in Table 3.2-3.

Table 3.2-1 Design Dead Loads (Less Reactor)

	Thousands of Pounds
Roof	
Precase Sections	1215
4" slab	270
Roofing and Insulation	65
3000 gallon D_2O Tank	28
20 Ton Crane	78
2nd Floor (Less Reactor)	
12" Slab	900
Concrete Beams	594
2-Disassembly caves at 20 Tons	80
15 Ton Annular Crane	30
Pump Room Roof	86
1st Floor (Less Reactor)	
5 ft. slab	2100
15 ft. slab	700
Concrete Beams	16
5 Ton Crane	22
Secondary Pump Room Floors	
3-6 in. slabs	138

Table 3.2-1 (cont'd)

	Thousands of Pounds
Mezzanine Floor	
8" slab	233
Beams	64
2"-4" slab	178
Hot Plug Storage Wall	1226
Monitor and Fan Rooms	
6" slab	53
24" Fill	182
Corridor	
6" slab	9
Pump Room	
6" slab	72
Basement	
6" Floating Slab (exclusive of reactor and pool area)	525
3' Fill under Floating Slab	2100
3' Concrete Pile Cap Under Building	3470
Walls	
Above 1st Floor	5140
Below 1st Floor	13,207
Exhaust Stack	615
Internal Columns	96
D_2O Storage Tank Pit	180
Storage Pool	180
Canal	156
Total	34,008

Table 3.2-2 Design Dead Loads, Reactor Only

	Thousands of Pounds
Thermal Shield	193
Aluminum Vessel - Shell only	3.2
Core Support Structure	4.07
Control Arms and Drives	.630
Cold Neutron Facility	2.5
Top Cover Plate	11.3
Top Plugs	90.3
Biological Shield	1530
Total	1835.0

Table 3.2-3 Design Live Loads for Total Confinement Building

	Thousands of Pounds
Roof	
Snow Load	202
Load on 20 Ton Crane	40
2nd Floor	
Movable Concentrated Loads - 2 at 20 Ton	40
Load on 15 Ton Crane	30
Floor Load at 150 psf	1210
Snow Load on Pump Room Roof at 25 psf	29
1st Floor	
Floor Loads	
6800 sq. ft. at 1000 lb/ft^2 620 sq. ft. at 2000 lb/ft^2	8040
Load on 5 Ton Crane	10
Floor Loads on Secondary Pump Room Floors	486
Floor Load on Mezzanine Floor	350
Floor Loads Monitor, Fan Room	105
Floor Load in Corridor	12
Floor Load - Counting Room and Laboratories at 100 lb/ft^2	240
Floor Load - Pool and Process Area at 400 lb/ft^2	1610
Floor Load - Secondary Pump Room, Basement at 300 lb/ft^2	175
14,000 gal D_2O in Storage Tank	130
27,000 gal H_2O in Pool	230
6,000 gal H_2O in Canal	50
4,600 gal D_2O in Vessel	43
Total	13,022

3.2.2 BUILDING STRUCTURE

3.2.2.1 General. The NBSR confinement building is a reinforced concrete structure on a driven steel pile foundation. With the exception of the main roof beams and eight 10" Wf beams in the central column, all interior beams and columns were poured in place. The roof beams are Type IV, prestressed steel reinforced concrete beams as defined by the Joint Committee of ASSHO and PCI.

Above the lower floor, all loads are transmitted by the exterior walls or a large central column of which the biological shield and subpile room walls are an integral part. At the lower floor level, columns and shielding walls, 3 to 5 feet thick in most cases, give additional support which is far in excess of that which is needed for structural integrity.

Both structural and shielding requirements were considered in the design of the exterior walls. The thicknesses which resulted from structural requirements above were, however, far in excess of any shielding requirements and, therefore, the structural requirements were the controlling design parameters.

The roof of the building is designed for shielding from "sky shine" as well as snow loading and the differential pressure loadings which occur during the building leak rate tests.

3.2.2.2 Foundation. The primary components of the foundation for the reactor building are 362 12" BP section steel piles at 74 pounds per linear foot. (Federal Specification QQ-S-741a, Type I or Type II, ASTM A7 or A373) with an aggregate length for 12 of 550'. Each pile is of 95 ton capacity, driven to refusal by a hammer with a minimum of 15,000 foot-pounds per blow. Refusal was defined as a maximum penetration of 0.25" in the last five blows. Control test piles were individually inspected inder the supervision of a registered Professional Engineer and in the presence of the Construction Engineer representing the General Services Administration. The test required that the net settlement at the top of the test pile be not more than 0.005" per ton under twice the design load of 95 tons; and that the increment of settlement for any increment of load shall not exceed 0.01" per ton until twice the design load was applied.

3.2.2.3 Storage Pool and Canal. Sheet piling was used to form the wall of the storage pool and canal. This piling is of the continuous interlock type, steel, conforming to ASTM A-328. The reinforced concrete pool and canal side walls are designed to tolerate a soil pressure of 95 psf and a hydrostatic head of 810 psf. In actuality, the sheet piling was necessary to protect the adjacent foundation areas during construction. When the pool is filled, the internal and external pressures cancel and the pool walls are left essentially unloaded.

3.3 BUILDING LEAKPROOFING

In addition to the normal requirements for the prevention of inward leakage from ground water, rain, etc., the NBSR confinement building was designed and constructed to prevent the uncontrolled outward leakage of radioactive materials.

At the very low, if indeed any, pressure differentials involved, waterproofing of the building is an essential element in leakproofing. A five-ply membrane waterproofing system consisting of five layers of fabric and six layers of pitch covers all exterior basement walls of the NBSR building and extends continuously under all basement foundations to form a complete seal. The waterproofing on outside walls which is exposed to damage by back fill is covered by a 1/2" thick insulating fiber board (Federal Specification LLL-I-535). All horizontal portions of the waterproofing, except those under concrete slabs, are covered with a 3/4" thick (minimum) layer of Portland cement mortar. Horizontal surfaces of the waterproofing under concrete slabs are protected by 1" of concrete.

All construction joints, including the roof slab to wall joints, have in them a 6" x 3/16" bulbed Polyvinyl chloride water stop. Water stops located in expansion joints have a bulb diameter equal to the thickness of the expansion joint. These waterstops have a tensile strength of not less than 1800 psi and an elongation of not less than 350% when tested in accordance with ASTM D-412, and a cold bend brittle temperature of not higher than minus 20°F when tested in accordance with ASTM D-746.

Containment wall surface preparation and coating was a major facet of the design and construction techniques used to assure a leak tight building. All containment walls received the following elastomeric coatings; rust inhibitive primer for ferrous metal surfaces, polychloroprene primer for concrete and concrete-masonry unit surfaces, fabric reinforcing sheet and polychloroprenesheet, for strip and crack sealing, polychloroprene adhesive for polychloroprene sheet, polychloroprene body coats for all surfaces, and chlorosulfonated top

coating for all surfaces. The body coats were applied in alternate black and red coats to aid inspection for complete coverage with each coat. Any point in the surfacing system which showed signs of pin holes, blisters or other discontinuities was removed to the body coats and recoated.

This coating system resulted in a 15 to 20 mil elastometric membrane continuous over all containment surfaces. Any porousity of the masonry structure was thus effectively closed.

3.4 STANDARDS OF MATERIALS

All materials used in the NBSR building were covered by the pertinent parts of the General Services Administration Specification for this project (Project Number 18112). Primary interest, from the standpoint of a safety analysis, is focused on those parts of the general specification which deal with the structural materials.

Reinforcing steel specified for the NBSR building conformed to Federal Specification QQ-S-632, Type II, intermediate grade billet steel with deformation conforming to ASTM Specification A-305, and a design tensile strength of 20,000 psi. All field splices in reinforcing steel were lapped a minimum of 30 bar diameters. All dowels were embedded in concrete for a minimum of 30 bar diameters.

Coarse aggregate for concrete was specified to Federal Specification SS- A-281b, Class 2 or C-33, sized in accordance with ACI-613, Table 2. Portland cement was specified to Federal Specification SS-C-192d or C-150. All structural concrete had a specified 28 day compressure strength of 3000 psi. Determination of proportion of cement, aggregate and water was accomplished according to ACI-318, Method 2 as modified. Measuring, mixing and delivery of ready-mixed concrete with inspection and certification was accomplished in accordance with ASTM C-94 requirements. Slump samples were taken in accordance with ASTM C-172 and tested under direction of the Construction Engineer in accordance with ASTM C-143. Slump was required to be within the recommended limits of ACI-613, Table I. Test samples taken for strength tests showed that in all cases, the specified strengths were met or exceeded.

3.5 PENETRATIONS

3.5.1 SUMMARY TABLE OF PENETRATIONS. The purpose, number, size, and type of the NBSR building penetrations are given in Table 3.5-1 for electrical penetrations and Table 3.5-2 for piping and other mechanical penetrations.

3.5.2 PENETRATION DESIGN DETAILS. Each penetration involves two distinct sealing problems; the sealing of the penetrating member to the concrete walls of the building and the internal sealing of any leak paths through the penetrating member.

3.5.2.1 Seals to the Building. All pipe, conduit and tubing is sealed to the building as shown in Figure 3.6 and noted as "B" in Tables 3.5-1 and 3.5-2. The penetration member and the flange which is welded to it are inserted in the concrete. The outer surface of the flange is flush with the finished inside surface of the concrete. The joint between the flange and the concrete surface is then caulked and sealed with fiber reinforced neoprene. The entire outer surface of the penetration is then coated with the neoprene hypalon system described in Section 3.3.

Door frames and air system plena are sealed to the building as shown in Figures 3.7 and 3.8 respectively and noted in Tables 3.5-1 and 3.5-2 as "D" and "C" respectively. Steel plates with continuously welded joints were invested in the concrete and the frames or plena were in turn fixed to those plates with continuous welds. All exposed welds were then caulked with a hypalon caulking compound, and coated with the neoprene-hypalon system described in Section 3.3. All voids in the door frames were then filled with closed cell urathane which was foamed in place.

3.5.2.2 Internal Seals. All electrical penetrations were sealed as shown in Figure 3.9 and noted as "A" in Table 3.5-1. After all conductors were in place, a sealing compound (CHICOX Fiber A05 manufactured by Crouse-Hinds Company of Syracuse, New York) was poured into the sealing box under sufficient hydrostatic head to force it into all voids between the conductors and between the conductors and the inner walls of the conduit.

Table 3.5-1 Electrical Penetrations of Containment

NAME	NUMBER	SIZE	SEAL TO BUILDING	INTERNAL SEAL
Lights	40	1"	B	A
	9	1-1/4"	B	A
	3	1-1/2"	B	A
	4	2"	B	A
	4	2-1/2"	B	A
	19	3"	B	A
	4	3-1/2"	B	A
	4	4"	B	A
Alarm and Communication	15	1"	B	A
	3	1-1/4"	B	A
	1	2"	B	A
Telephone	2	2"	B	A
Instrumentation	8	1"	B	A
Power to Equipment	6	1"	B	A
	6	1-1/4"	B	A
	3	1-1/2"	B	A
	2	2"	B	A
	5	2-1/2"	B	A
	4	3-1/2"	B	A
	4	4"	B	A

Table 3.5-2 Piping Penetrations of Containment

NAME	NUMBER	SIZE	SEAL TO BUILDING	INTERNAL SEAL
Chilled Water	2	5"	B	Closed System
	2	3/4"	B	Closed System
Steam	1	4"	B	Closed System
Condensate Return	1	2"	B	Closed System
Demineralized H_2O	1	1"	B	Closed System
Domestic Water	2	3"	B	Closed System
	1	2"	B	Closed System
	2	3/4"	B	Closed System
Air	6	1/2"	B	Closed System
	1	3/4"	B	Closed System
	3	1"	B	Closed System
	1	1-1/4"	B	Closed System
	1	2"	B	Closed System
Oxygen	1	1/4"	B	Closed System
Experimental Penetrations	16	2-1/2"	B	Capped
Building Test Instrumentation Sleeves	8	1/4"	B	Capped
Door (Personnel)	1	6' 4-1/2" by 8' 2-1/4"	D	Inflatable Gasket

Table 3.5-2 (cont'd)

NAME	NUMBER	SIZE	SEAL TO BUILDING	INTERNAL SEAL
Door (Truck)	1	12' 4" by 14' 2"	D	Inflatable Gasket
Door (Emergency)	1	3' 5" by 7' 2-1/2"	D	Inflatable Gasket
Window - Time of Flight	1	2' by 2'	B	Compressed Rubber Gasket
View Panel - Emergency Door	1	6" by 6"	D	Compressed Rubber Gasket both Sides
Building Leak Test and Exhaust	1	8"	B	Manual Damper with Rubber Seat
Air Intake and Exhaust	1	7' 6" by 10' 0"	C	36" & 42" Valves
Air Intake	1	7' 5" by 2' 8"	C	36" & 30" Valves
Air Exhaust	1	1' by 1'	C	6" Valve
Air Exhaust	1	6" by 6"	C	4" Valve
Door to Elevator	1	7' 0" by 8' 3"	D	Inflatable Gasket
Door - Personnel	1	5' 4-1/4" by 7' 2-1/4"	D	Inflatable Gasket
Secondary Cooling	2	1/2"	B	Closed System
	2	2"	B	Closed System
	1	3"	B	Closed System
	1	4"	B	Closed System
	1	5"	B	Closed System
	2	20"	B	Closed System
Gas	1	3/4"	B	Closed System
Helium	1	3/4"	B	Closed System
	2	1-1/2"	B	Closed System
CO_2	1	2"	B	Closed System
	1	3/4"	B	Closed System
Drain, Roof	8	4"		Closed System
	2	3"		
	2	6"		
Suspect Waste	1	1-1/2"	B	1-1/2" Valve
Hot Waste	1	4"	B	4" Valve
D_2O	4	1/2"	B	Closed System
	3	1"	B	Closed System
Cryogenic Service Feed Through	1	12"		Closed System

All air ducts and waste lines are sealed internally by rubber seated butterfly valves or dampers. All of these devices were specified and tested to bubble tight specifications.

The cryogenic services penetration is sealed with the same closed cell urathane material which was used in the door frames. This material is foamed in place to assure complete filling of the voids between the pipes which pass through this penetration and to assure intimate contact with the inner walls of the penetration. The ends of the foamed material are coated with an air curing silicon rubber compound (Dow Corning RTV102) which is carried up onto the surface of the pipes to form a minimum radius of 1/4".

All access and exit doors are sealed to their frames by inflatable rubber gaskets as shown in Figure 3.7. Whenever those doors are closed an internal seal pressure of approximately 20 psi minimum causes the gasket to inflate and form a sealing surface approximately 1 inch wide around the entire perimeter of the door. The large truck door is mechanically restrained to prevent it from being pushed away from the seal when the gasket is inflated.

3.6 VENTILATION SYSTEM

A flow diagram of the reactor building ventilation system is shown in Figure 3.10. The diagram shows schematically both the normal and emergency systems as well as the system for building leak rate testing. The reactor building is air conditioned with air that is partially recirculated except for the process equipment area of the reactor basement which is separately heated and ventilated. All ventilation duct work that penetrates the reactor building is provided with automatically sealing closure valves. The reactor confinement building which has a volume of approximately 600,000 cu. ft. has been designed and constructed to achieve a minimum air leakage. Those features of construction pertinent to building tightness are discussed in Section 3.3.

3.6.1 VENTILATION SYSTEM UNDER NORMAL CONDITIONS. Under normal conditions, there are two separate systems supplying ventilation air to the reactor building. Intake air is brought into the mezzanine equipment area through louvres in an areaway in the south wall of the reactor building. Air from the intake is supplied to both the basement heating and ventilating system and the air conditioning fresh air system. The louvres can be sealed by automatic valves ACV-1 and ACV-2. The flow rate of air supplied to the air conditioning fresh air system is 13,680 cfm. The flow rate of air supplied to the basement heating and ventilating system varies from 16,500 cfm in the summer to 2500 cfm in the winter. Incoming air is filtered by filters numbered F-2 and F-11.

The air conditioning fresh air system is a conventional heating and cooling system and supplies air at 53° to the first and second floor air conditioning systems and to the basement laboratory air conditioning system. The first floor system circulates air at a 23,500 cfm rate which is ten times the rate at which fresh air is supplied. The second floor system circulates air at a 20,750 cfm rate which is approximately seven times the rate at which fresh air is supplied. The basement laboratory system uses fresh air only, without recirculation.

The basement heating and ventilating system maintains the process equipment area temperature by heating of fresh air being brought in at a 2500 cfm rate in the winter while recirculating air at 14,000 cfm, or by ventilating with once-through air in the summer at a rate of 16,500 cfm.

The exhaust system for the reactor building consists of three sub-systems for normal operation. The reactor basement exhaust system includes exhaust fan EF-27 which draws air from the process equipment area at a rate of 16,740 cfm. All air drawn by EF-27 is filtered by two banks of filters F-59 and F-60. F-59 acts as a pre-filter for F-60 which consists of so-called absolute filters specifically designed for high efficiency. The latter filters are described in Section 3.6.3. Following the exhaust fan there is a hold-up chamber whose volume is chosen to give approximately 2.5 sec. for the automatic closure valve ACV-3 to operate after a building closure signal, before air, which enters the chamber at the initiation of the closure signal, can exit the chamber. This time it is calculated for the condition of maximum exhaust rate with no recirculation taking place. An adjustable louvre is located at the exhaust of the fan to enable an adjustment of building basement static pressure relative to the first floor reactor area and therefore relative to the outside static pressure.

The normal exhaust system takes air from those areas supplied by conditioned air, namely the first and second floor, pool area, counting room, and radiological laboratory room. The exhaust fan for this system, EF-3, normally discharges at a rate of 7290 cfm. An exhaust fan damper automatically adjusts the rate, however, to maintain a building pressure of -0.10" of water relative to the first floor of the cold lab area. All effluent air drived by EF-3 is filtered by filters F-22 and F-23 in a manner similar to that for EF-26. The filters are described in Section 3.6.3. The exhaust of EF-3 combines with the exhaust from fume hoods in the radiological laboratories which is drawn by fans EF-23 and EF-24. The exhaust from these hoods is filtered in like manner to that of EF-3 by filters F-53, F-54, F-55, and F-56 which are described in Section 3.6.3. The air in the normal exhaust system passes through a hold-up chamber whose volume allows approximately 2.2 sec for the automatic closure of valve ACV-7.

The location of the exhaust ductwork ports are chosen to control air flow within the experimental areas of the reactor building. Most air from the first and second floor areas exhaust at the center of the experimental area through ducts embedded in the biological shield of the reactor. This should serve to control contamination levels within the building, since potential sources of contamination are closest to the reactor.

There is in addition to normal exhaust a supplemental system called the irradiated air exhaust system. This system takes air into the biological shield at all beam port openings to the reactor and is consequently the system most likely to be handling contaminated air. It is designed to assure that all leakage of air is into beam ports rather than the reverse. Any potentially irradiated air is directly exhausted rather than being recirculated. The exhaust fan in this system is EF-4 which draws air at a rate of 650 cfm. All irradiated air passes through filters F-24 and F-25 which are identical to the filters in the normal exhaust system. EF-4 is followed by a hold-up chamber to permit closure time for the duct sealing valve ACV-6.

3.6.2 VENTILATION SYSTEM UNDER ACCIDENT CONDITIONS. All effluent air which is exhausted from the containment building is monitored for radioactivity by the system described in Section 9.6. Each of the absolute filter banks is monitored for particulate activity, air samples are withdrawn for counting of gaseous activity, and a monitor measures activity in the stack at the point of release. In the event that high radiation levels are detected the normal ventilation system will be shutdown, all building closure devices will operate to seal the building and the emergency ventilation system will be activated.

The emergency exhaust system is designed to draw air at such a rate from the building that a pressure differential can be established across the building structure to assure that any leakage is into the building rather than out regardless of likely outside pressure variations due to wind or barometer changes. The system is highly redundant to give maximum assurance of operation and can be controlled, if necessary, from an emergency control station outside the reactor building.

The emergency exhaust system consists of two redundant sub-systems A and B each of which contains an exhaust fan and identical filters and controls. Either sub-system can draw air from the normal exhaust system ductwork at a rate up to a maximum of 100 cfm. Since the reactor basement does not normally exhaust to the normal exhaust system ductwork, a special connection is made to the basement system during emergency conditions by the

automatic operation of ACV-10. The two subsystems are isolated from each other by valves ACV-4, ACV-8, ACV-5 and ACV-9. One of the systems would be set for automatic operation in the event of building closure while the second system was placed in standby. Upon building closure the pre-selected system starts to withdraw air from the building. The standby system remains off during a time delay period to permit operation of pressure switch PS-150. If the fan operates correctly and pressure in the exhaust duct falls then the standby system remains in standby condition while the building pressure is reduced.

The fan will continue to withdraw air until the automatic pressure controller PC-150 indicates a -0.25 " pressure differential between the inside and outside of the building at which time the closure valves operate to seal the building and all exhaust stops. If the initially started fan fails to reduce the duct pressure, then PS-150 will permit the standby fan to start. Both fans EF-5 and EF-6 are driven by either an AC or DC motor. In the event of AC power failure, a corresponding DC control circuit supervises control of the DC motor in a like manner to that for the AC motors.

All air exhausted from the building during emergency conditions passes through separate filters. EF-5 draws air through filters F-26, F-27, F-28 and F-57. EF-6 draws air through filters F-29, F-30, F-31 and F-58. Filters F-26 and F-29 are pre-filters for the "absolute" filters F-27 and F-30. Filters F-28 and F-31 have activated charcoal as their filter medium and are chosen for the purpose of removing non-particulate or gaseous effluent such as iodine. The charcoal filters are each backed by a fourth filter to collect charcoal particulate should the charcoal filter begin to deteriorate during emergency operation. The filters are described more completely in Section 3.6.3.

During emergency operation, the air interior to the reactor building is recirculated and filtered by means of a separate system. Air in this system is drawn from all areas of the reactor building by fan SF-19 at a rate of 5000 cfm and put through filters F-19, F-20, and F-21. F-20 is an "absolute" filter and F-21 is an activated charcoal filter bank similar to those in the emergency exhaust system. This system is designed to remove particulate and gaseous activity such as iodine with an approximate 2 hour time constant for once through cleaning. Since normally the reactor basement and the remainder of the building have separate ventilation, a valve ACV-11 operates whenever the fan SF-19 starts to connect the isolated systems. The emergency ventilation is automatically brought into operation when a "major scram" signal is generated at the reactor control panel is discussed in Section 9.

3.6.3 <u>FILTER DESCRIPTION AND TEST PROCEDURES</u>. The filters installed in the reactor building are of several types. There are two types, however, which are of prime importance and for which special procurement specifications were used. As mentioned in the previous section, so-called "absolute" filters are used in all reactor building exhaust systems to prevent particulate effluent from reaching the reactor building stack and being discharged. The filters were purchased according to the specifications numbered in the Accident and Fire Prevention Information Bulletin #104 of the U. S. Atomic Energy Commission, dated December 11, 1959. All absolute filters were tested at the factory and finally at the Oak Ridge Filter Test Facility following procedures of the Health and Safety Information Bulletin #188 of the U.S.A.E.C. dated July 1, 1964. The filters are of the disposable deep pleated dry type consisting of a continuous sheet of glass media in fire resitant frames of treated plywood. The efficiency of each filter is better than 99.97% as measured at the Oak Ridge Test Facility using dioctyl plitholate (DOP) aerosol of 0.3 micron size. The efficiency of the installation technique for the filters will be verified with in-place testing using the so-called DOP test. A member of the NBSR staff has attended the filter testing instruction courses sponsored by the U.S.A.E.C. and such techniques as learned will be applied to the in-place testing procedures. Equipment identical to that used at the U. S. Naval Research Laboratory for filter testing has been assembled and will be used for these tests.

The activated charcoal filters were also purchased with special specifications. The filter assembly consists of at least two banks of filters in series each of which consists of multiple perforated members, arranged in parallel to form a bed of one inch thickness of granular activated charcoal. The performance of the filters shall be to extract 99.9 percent of iodine from air at approximately 80°F at the rated flow. The filters are of the removable steel cell type in a supporting steel frame, all cadmium plated. The physical arrangement of all filter installations is shown in Figure 3.11.

The charcoal filters in the emergency exhaust ducts are derated by a factor of four to give lower linear flow velocity through the charcoal; i.e., the cell size chosen is adequate for more than 4 times the maximum flow of the emergency exhaust system.

3.6.4 EXHAUST SYSTEM AND STACK. All air which is exhausted from the reactor building is supplied to a dilution exhaust fan EF-2 which is in a plenum at the base of the reactor stack. The stack is actually a dual stack, one side of which exhausts the reactor building and the other side of which exhausts the warm laboratory. The dilution fan exhausts air to the stack at a rate of 30,180 cfm. This rate is kept constant throughout seasonal changes in supply air from the reactor basement ventilation system by means of a velocity controller which opens and closes louvres before a makeup air port at the base of the stack.

The principal parameter of the stack is its height. Since under normal operation, argon-41 activity is produced within the confines of air spaces in the reactor system, it was deemed necessary to elevate the point of release so that appreciable dilution via atmospheric dispersion would occur prior to any possible inhalation of reactor building exhaust air. The height of approximately 100 feet above grade level was chosen to give assurance of meeting the criteria of 10 CFR part 20. It has been calculated and demonstrated by measurements with weather station instrumentation that dilution factors of the order of 1000 are achieved at the site boundary following release from the stack under varying atmospheric conditions.

The other engineering design parameters of the stack were established by the requirements of standard heating and ventilation duct design practice. The exit flue area of 15 sq. ft. creates a linear flow rate of 2000 ft. per min. under normal exhaust conditions. This corresponds to an exit velocity of approximately 25 mph which flow rate gives a pressure head at the dilution fan at the base of the stack of approximately 0.2 inches of water. These values constitute an entirely acceptable set of engineering parameters.

3.6.5 ACCIDENT CONDITION REVIEW. Under emergency conditions the normal intake of fresh air and the recirculation of internal air stops, and the emergency systems start automatically. There are two emergency systems: one exhausts air from the building through activated carbon filters and up the stack; the other recirculates the air internally through activated carbon filters. The first system controls in-leakage to the building by automatically maintaining a negative pressure differential across the building wall of one quarter inch of water. The second system recirculates the air within the building through filters to remove radioactive contaminants. Figure 3.11 shows the locations of the ventilation registers through which the air is exhausted from the various levels of the building by the emergency exhaust system. Air is withdrawn from four different regions of the reactor building at rates proportional to the volume of space occupied by the four separate regions. The following regions are included with the proportion of the total exhaust from each region.

Second Floor (above grade)	40%
First Floor (grade level)	20%
Process Room (below grade)	15%
Pool Area (below grade)	25%

The air from all these regions passes through a common duct, filter and fan system before entering the plenum to the dilution fan at the base of the stack, and is, therefore, thoroughly mixed before leaving the stack.

The recirculation system operates continuously from the instant of building closure and is designed to clean up the reactor building air from all regions with an effective two hour time constant. It would be expected to thoroughly mix the gases from all regions with the same rate.

In general, the delivery and exhaust ducts have been located in appropriate positions to effect two functions: the delivery and exhaust with good mixing of internal air under normal conditions of high flow, and the exhaust of containment air under emergency conditions of slow flow. Also, in general, the exhaust registers were positioned away from any conceivable break in the reactor primary system from which released fission product gases could emanate.

A break in the lower plenum or primary piping in the subpile or process room below the core vessel and thermal shield is the only break which could credibly lead to a slow meltdown and fission product release. As a consequence of such a break and meltdown the fission gases would initially be released in a stagnant or trapped volume, namely the volume of the reactor vessel and its associated piping down to the point of the break where the loss of coolant occurred. The fission gases would have to seep out of this enclosed volume before mixing with the building air. It has been shown in the document entitled, "Answer to AEC Staff's Motion to Reconsider Holding of Hearing" Docket 50-184 dated April 6, 1964 that normal convective and turbulent background air velocities would effect good mixing everywhere within the confines of the process room of the reactor basement except within the immediate vicinity of the exhaust duct where no fission gases can originate. It was also shown that the exhaust duct has little influence on any local air velocity more than a few feet from the duct and certainly from any point beyond the end of the passageway in which the duct is located. It is clear, therefore, that an assumption of uniform mixing of all exhausted confinement building air is valid.

It should be noted in connection with the review of the ventilation system under emergency conditions that even though the system is fully automatic and highly redundant, an emergency control and monitoring station is provided external to the reactor building. An emergency panel is provided in the hot waste vault underground in the front or east side of the reactor laboratory complex. Provided at this station are controls for all fans in the emergency system, namely EF-2, EF-19, EF-5 and EF-6 (both AC and DC controls for the latter two fans). Included also are indicators for valve positions of all valves ACV-1 thru ACV-12. The building differential pressure can be observed at this panel and an indicator shows when exhaust air flow from the emergency exhaust system is achieved. Finally there are two radiation monitors which indicate at this panel to show radiation levels within the reactor building. The latter monitors are discussed fully in Section 9.

3.7 CONFINEMENT SYSTEM LEAKAGE RATE

3.7.1 DESIGN LEAKAGE RATE. Confinement as opposed to containment constitutes a concept suitable to the situation wherein the worst hypothetical reactor incident results in negligible overpressure. Such is the case for the NBSR. Here the attempt is made to provide sufficient confinement or retention of radioactive gasses that they may be filtered and passed up the stack at a reasonably low rate for subsequent dispersion by the atmosphere. The tighter the building, the slower the gas or internal atmosphere need be pumped through the filter and stack.

The actual specification for the NBSR was based on the best state-of-the-art known to the NBS design group. It may be expressed either by relaxation time, which value is 64 minutes for the NBSR; or by the flow rate resulting from an over or underpressure, which value is 24 cfm per inch for the NBSR. At the time of acceptance it was demonstrated that the building exceeded this criterion.

3.7.2 LEAKAGE RATE TESTS. In NBSR-7C, Supplement to the Preliminary Hazards Summary Report, dated August 1, 1962, two types of building leakage rate tests were outlined. One of these involved temperature compensating drums for increased accuracy of measurement over periods of test time during which the temperature might change appreciably. During the actual acceptance tests it was learned that the tests could readily be made during times in which the temperature did not vary sensibly. It was also shown that the building readily exceeded its tightness specifications, and therefore, less time was required to prove its integrity. Accordingly, in the future it is not planned to utilize the temperature compensating drum system even though such a system was fabricated and could be invoked.

Two types of measurements are currently performed. In both tests the intake and exhaust valves are closed and the confinement system tested with the air conditioning systems operating to stabilize the temperature within the building at 70°F. In one test, the test blower flow rates required to establish +7.5", +4.0", and -2.5" are observed and shown to be less than 24 cfm per inch. In another test the building is pumped or exhausted to the same three differential pressures and the relaxation times are observed and shown to exceed 64 minutes.

The equipment required to achieve the above cited tests have been installed in the facility as originally planned and shown in NBSR-7C, p. 3, dated August 1, 1962. It is shown also on Figure 3.10 of this report.

3.7.3 INSPECTION OF PENETRATIONS. Independent of leakage rate tests, building penetrations will be inspected as part of the normal maintenance of the building and systems. It has been found most practical to locate building leaks during an actual test by either direct observation or Freon testing of suspect penetrations. It is planned to continue with such inspections each time the confinement system leakage rate is measured, even though the specification might immediately be met.

3.8 BUILDING UTILITIES AND SERVICES

3.8.1 DOMESTIC WATER SUPPLY. The water for the NBS site is supplied by the Washington Suburban Sanitary Commission. They get their raw water from the Potomac River and process it at the Potomac Filteration Plant. It is then stored in the standpipe located at Shady Grove, Maryland, which supplies the surrounding area including the NBS site.

Water is supplied to the Reactor Building from the NBS distribution system through a 12" pipe at 90 psig. It is anticipated that the Reactor Building and associated laboratory will require about 300 gpm. Of this, 200 gpm is used as make-up for the cooling towers. This water must be given additional treatment at the reactor site. The remaining 100 gpm is used in the hot water system and other normal intermittent building and laboratory uses.

3.8.2 STEAM AND CHILLED WATER. Steam is supplied to the reactor site from the NBS distribution system at 150 psig. Immediately inside the building it is reduced in one step to 15 psig and distributed to the hot water and the heating systems. It is estimated that the reactor and adjacent laboratory building will require about 12,000 lbs of steam per hour.

The reactor air conditioning system does not have its own compressors, but rather uses chilled water supplied from the central NBS plant. The water is supplied at 43°F, circulated through the cooling coils by the air conditioning system and returned to the central plant at 53°F. The total requirements of the Reactor Building and adjacent laboratory will be 1300 gpm. Ninety gpm of this will be used to cool the storage pool in the Reactor Building. The rest of the chilled water will be used in the air conditioning systems.

3.8.3 DRAINS. Since all liquid waste in the Reactor Building will be suspect of radioactive contamination, the drains in the Reactor Building have been carefully limited and controlled. All the systems supplying water to the face of the reactor for experimental use are closed systems with their own return lines. One of the few exceptions are the drains in the radiological laboratories. Here regular sink and cup drains are provided. One additional sink drains into this system from the Health Physics room in the adjacent laboratory. This sink is provided with a deep trap to maintain confinement building integrity. In addition to these laboratory drains a few open drains have been strategically located in certain areas in case of accidental spills. Besides these limited open drains, a valved-off drain is provided at each reactor beam hole.

All these drains join into one system which empties into a sump tank. As this tank fills, the waste water is automatically pumped into the radioactive waste system provided in the adjacent laboratory building.

Two roof drains also penetrate the Reactor Building. These, however, are not part of the Reactor Building drainage system since they simply pass through the building and out again into the storm sewer system. They are completely sealed from the interior of the Reactor Building to form no breach of the confinement system. In a few cases, where a small flow of domestic cooling water is permissable, the water is drained directly into these storm sewer drains. In these cases, each is a closed system from domestic water supply to drain maintaining the integrity of the confinement building.

3.8.4 ELECTRICAL POWER

3.8.4.1 Source. The electric power is supplied by the Potomac Electric Power Company. They maintain a sub-station on the NBS site containing two transformers. This sub-station is fed by two separate 69 Kv tranmission lines. The power company transformers reduce the 69 Kv to 13.8 Kv. This feeds immediately into four large switch-gear stations, which are all connected together by normally closed circuit breakers. An underground 13.8 Kv distribution

system distributes the power from these switch-gear stations to the buildings on the site. Two feeders, each from a separate switch-gear station, bring power into the Reactor Building.

3.8.4.2 Reactor Building Sub-Station. The main one-line diagram for the sub-station in the Reactor Building is shown in Figure 3.12. The two incoming lines feed separate 1500 Kva 13,750-480Y/277V transformers. Each transformer feeds a 2500 amp 480Y/277V, 3 phase, 4 wire bus. The two busses feed 480 volt, 3 phase, 3 wire motor control centers and 480Y/277 volt, 3 phase, 4 wire power and lighting distribution panels directly. They also feed, through transformers, 280Y/120V, 3 phase, 4 wire power distribution panels.

3.8.4.3 Motor Control Centers. All of the motor control centers are 480 volt, 3 phase, 3 wire systems except for one DC center to be discussed later. The building service motor control center consists of two busses, A-1 and A-2, tied together with a circuit breaker. This system is fed through a 1600 amp frame circuit breaker set to trip at 600 amps.

The pump room motor control centers, A-7 and B-8, are fed separately from bus A and bus B respectively. Each is fed through a 1600 amp frame breaker set to trip at 800 amps. They are connected by a normally open circuit breaker, which can be closed in case of power failure on either of the main busses.

The reactor motor control centers, A-3 and B-4, are also fed from separate busses through 1600 amp frame circuit breakers, set to trip at 800 amps, and are again connected by a normally opened circuit breaker. The emergency power motor control centers, A-5 and B-6, connected together by a normally closed circuit breaker are fed from A-3. Should power fail at A-3, a normally closed breaker to A-5 will open and a normally open circuit breaker from motor control center B-4 to the emergency power system will automatically close. Should power fail on both A-3 and B-4 both breakers open and the emergency power system takes over. The details of the emergency power system will be discussed in a later section.

3.8.4.4 Power and Lighting Panels. The two 480Y/277D, 3 phase, 4 wire panels D and E are fed directly from busses A and B respectively. From these panels the power is distributed to the various lighting panels in the Reactor Building and adjacent laboratories.

There are two types of 280Y/120 volt, 3 phase, 4 wire, power panels fed through 480/208 volt transformers. Power panels A, B, and C deliver normal power to the cold labs, warm labs, and the reactor, respectively. Panels A and B are supplied by bus A, and panel C from bus B. Panels F, G, and H supply instrument power to the cold labs, warm labs, and the reactor respectively. Panels F and G are fed from the main bus A, and panel H from bus B. They are tied together with normally open circuit breakers in case of failure of power to one of the panels. A separation of the instrument and normal distribution panels should greatly reduce electrical interference, which is often present on the normal power systems.

3.8.5 FIRE ALARM SYSTEM. The Reactor Building and adjacent laboratories are equipped with automatic detectors. These detectors are divided into zones. A signal from any zone sounds fire bells locally in the building and lights up the appropriate zone on an annunciator panel located near the lobby. The signal is also sent back to the local guard office, where the building is annunciated.

3.8.6 CIVIL DEFENSE SYSTEM. All buildings on the NBS site, including the Reactor Building, will be equipped with Civil Defense horns. These will be activated from a central location in case of national emergency.

Reference

(3.1) U.S.A.E.C. Docket No. 50-184, "Application for Construction Permit, National Bureau of Standards," pages 31 and 32.

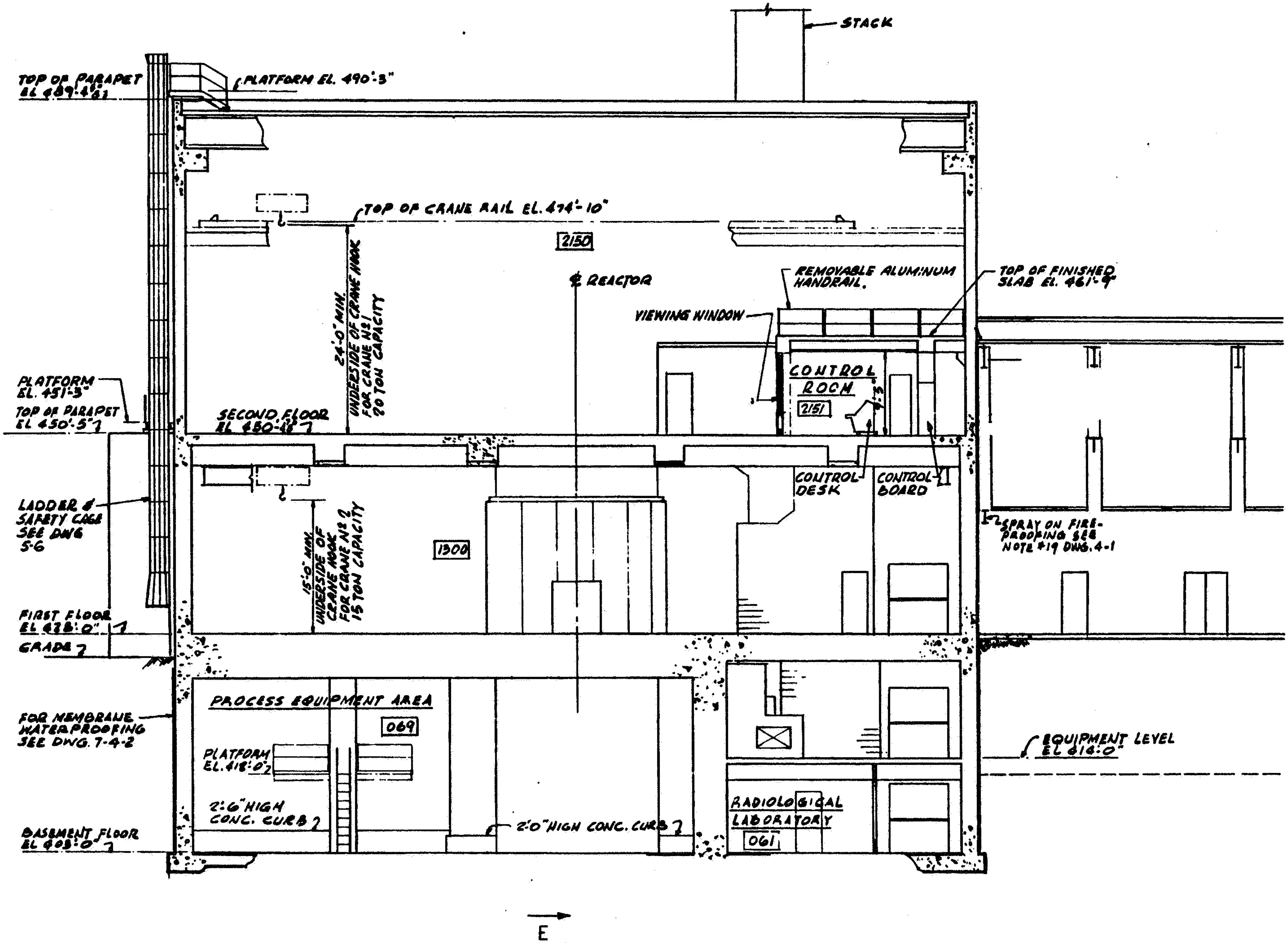


Figure 3.1 Confinement building elevation. East-west section through center.

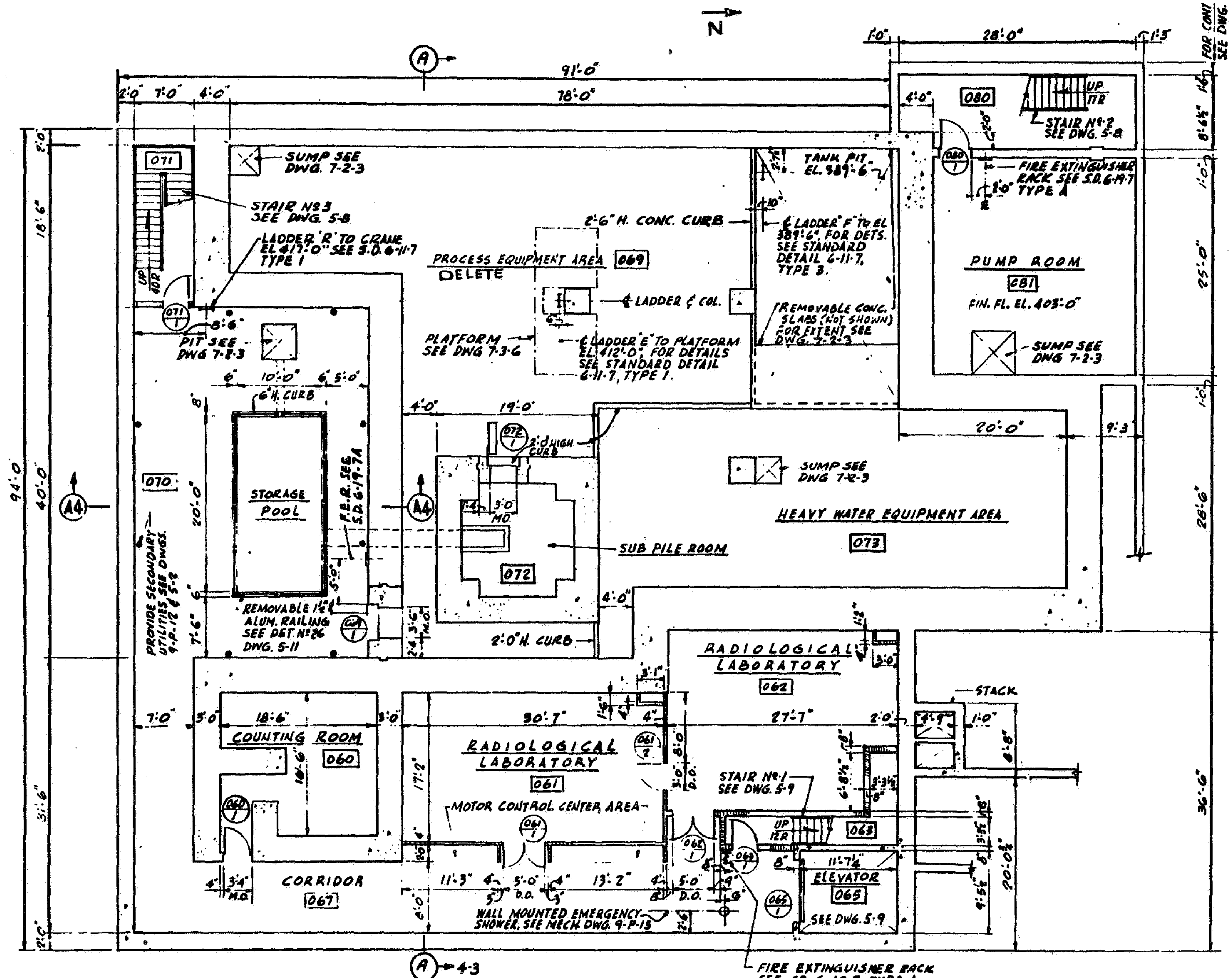


Figure 3.2 Basement plan view. Elevation 403' - 0".

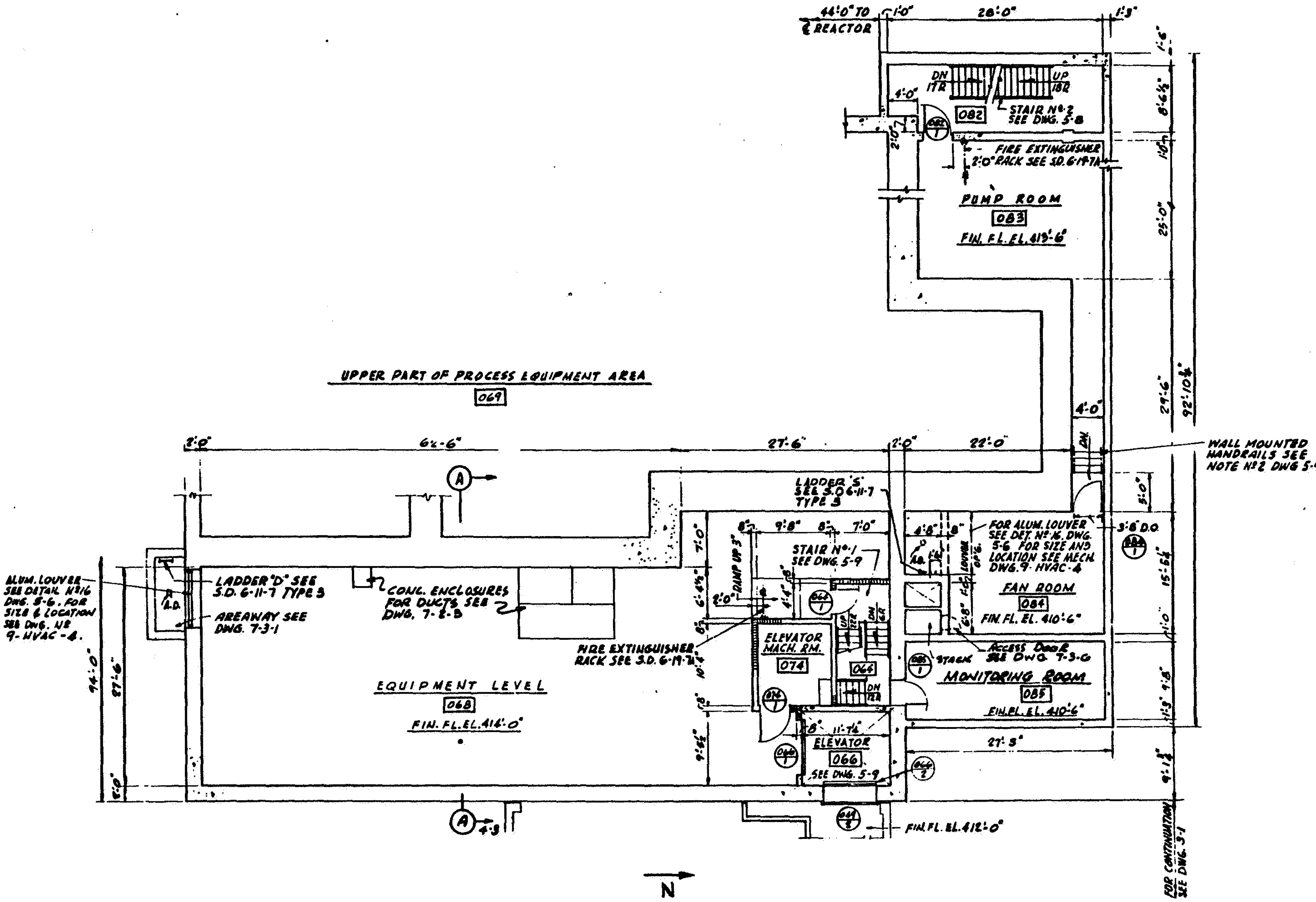


Figure 3.3 Mezzanine plan view.

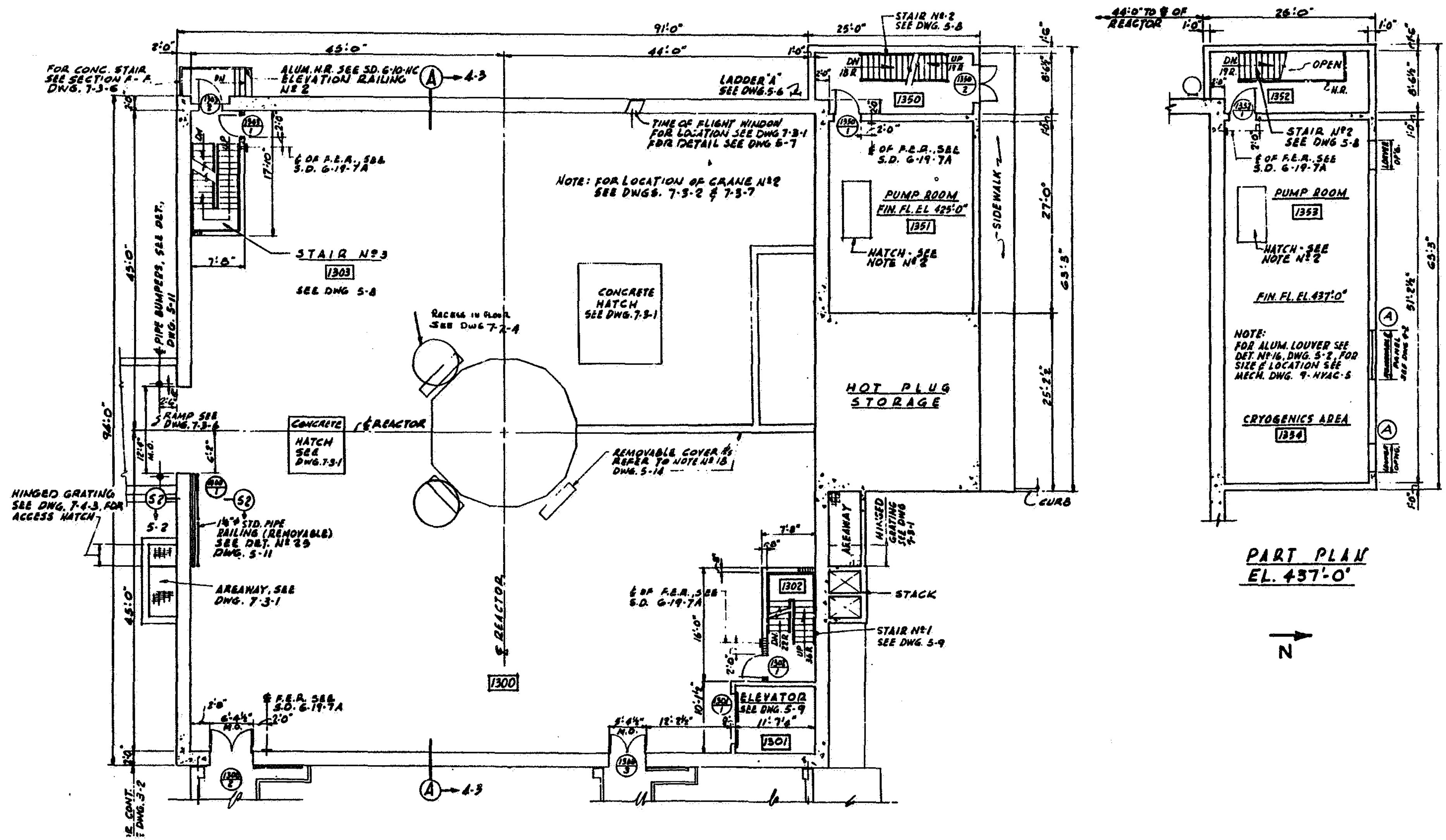


Figure 3.4 First floor plan view. Elevation 428' - 0".

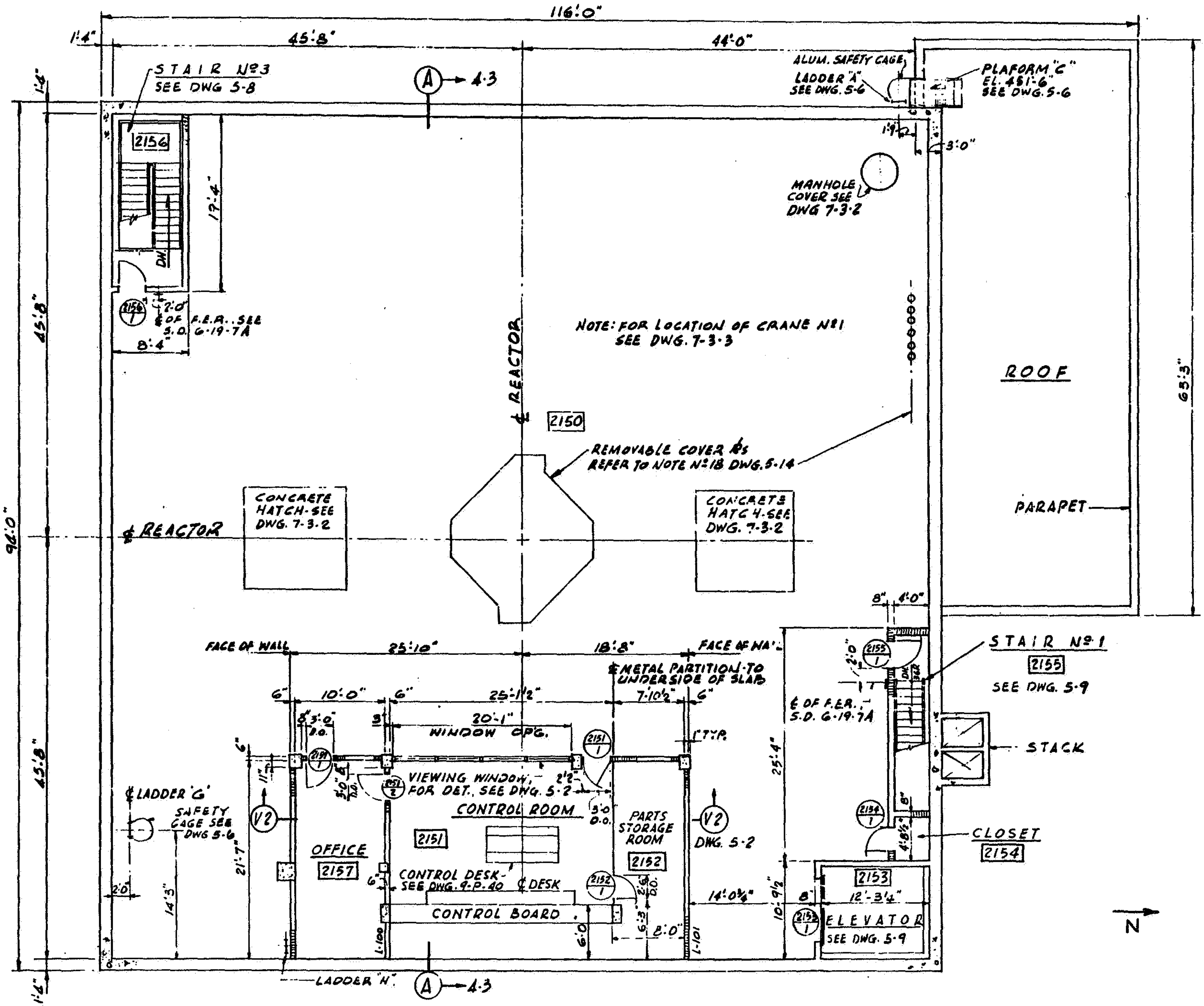
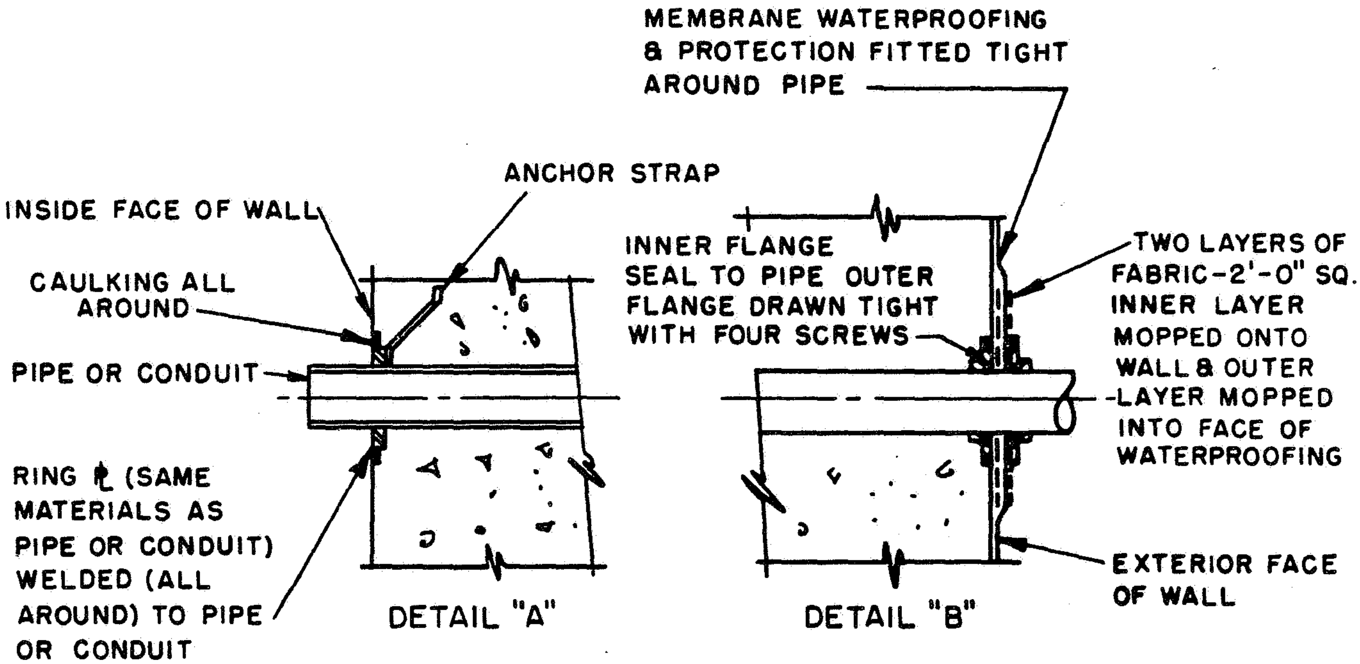


Figure 3.5 Second floor plan view. Elevation 450' 4-7/8".



TYPICAL PENETRATION DETAILS
THRU CONTAINMENT WALLS

NOTE: DETAIL "A" APPLIES TO THE INSIDE FACE OF CONTAINMENT WALLS OF THE REACTOR BUILDING AT ALL PIPE & CONDUIT PENETRATIONS.

DETAIL "B" APPLIES TO THE OUTSIDE FACE OF THE CONTAINMENT WALLS AT ALL PIPE & CONDUIT PENETRATIONS WHERE MEMBRANE WATERPROOFING IS REQUIRED.

Figure 3.6 Typical details of penetrations through containment walls

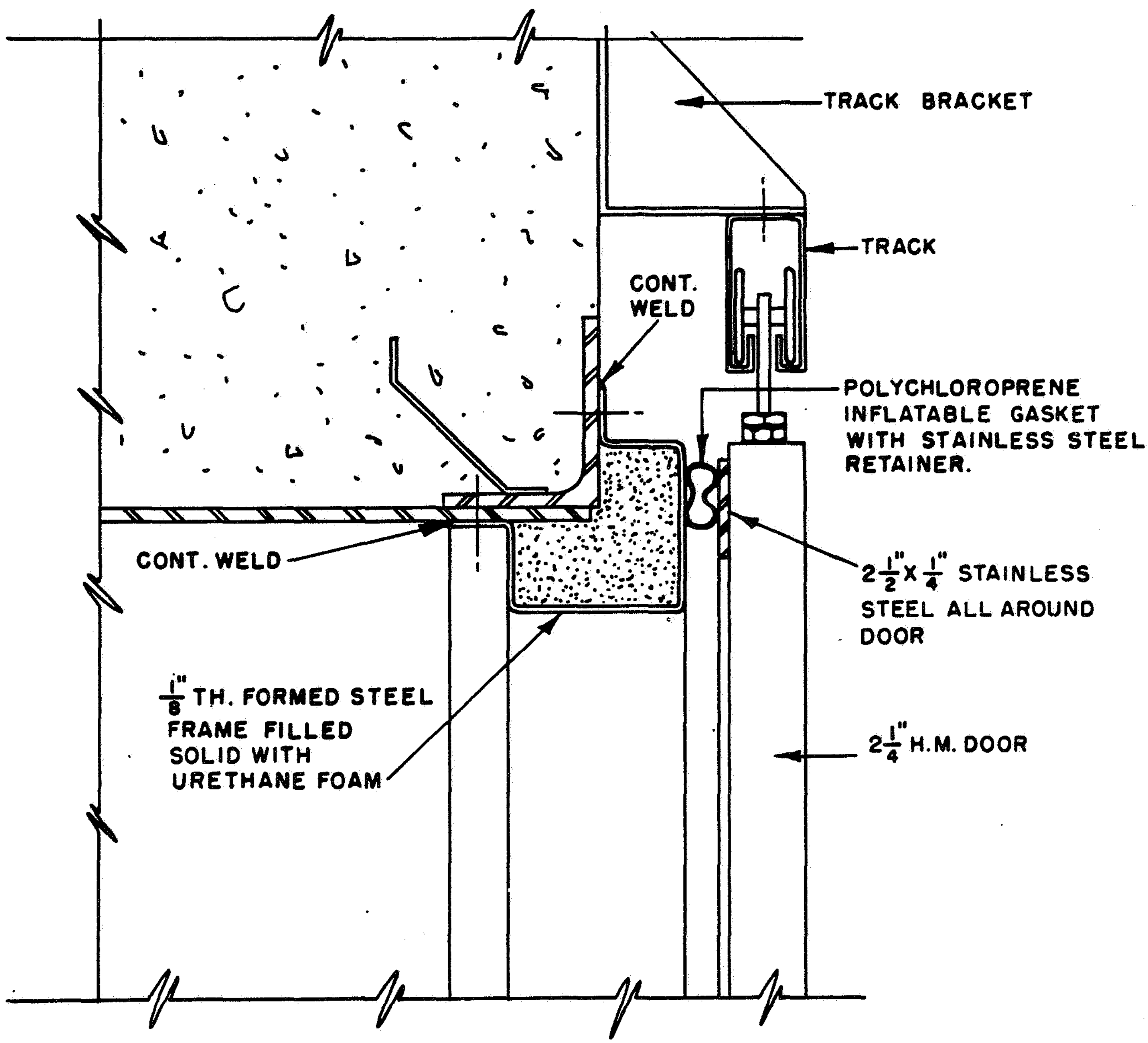


Figure 3.7 Details of door seals.

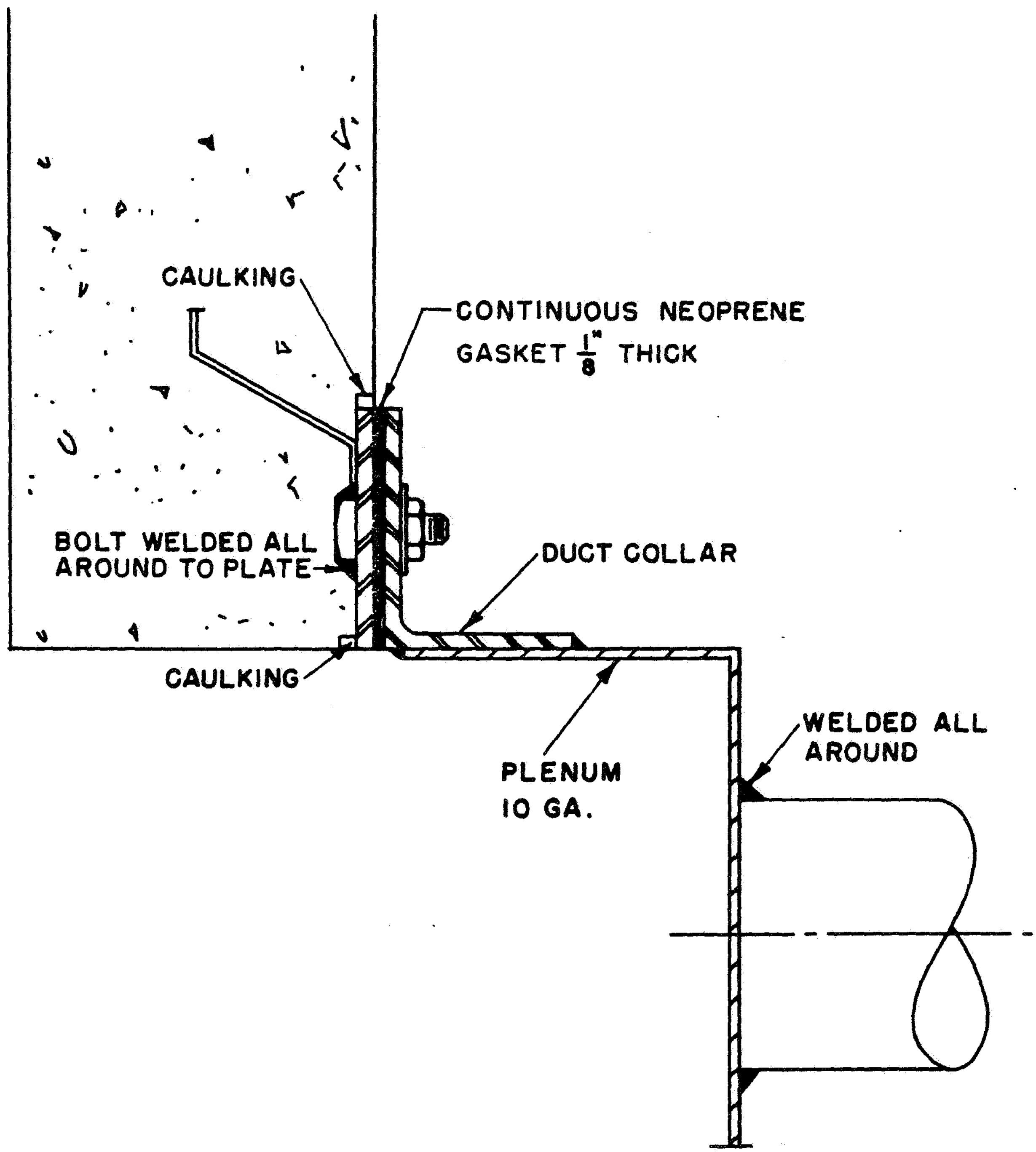
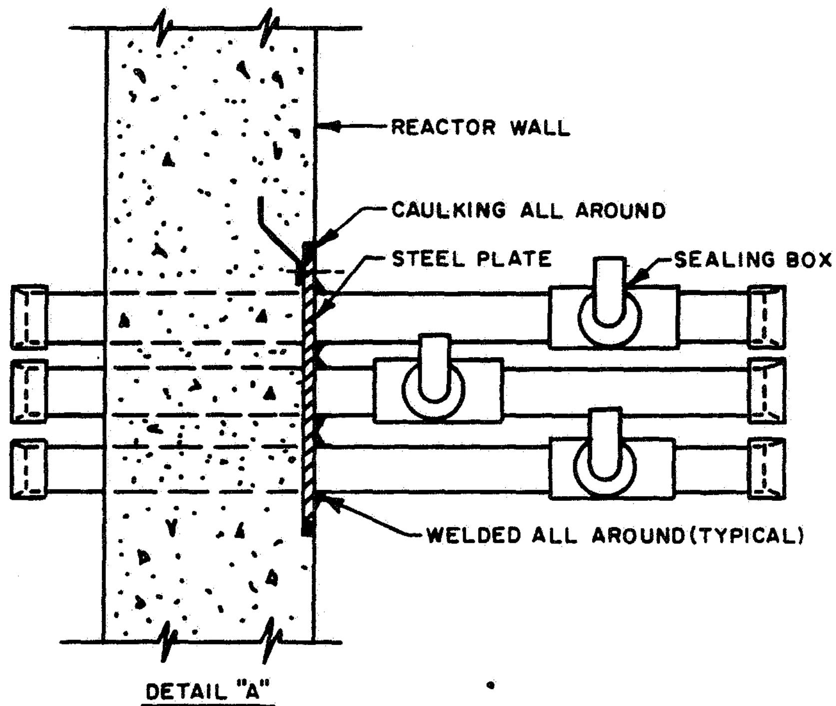


Figure 3.8 Details of air plena seals.



TYPICAL PENETRATION DETAIL THROUGH CONTAINMENT WALLS

Figure 3.9 Details of electrical penetrations.

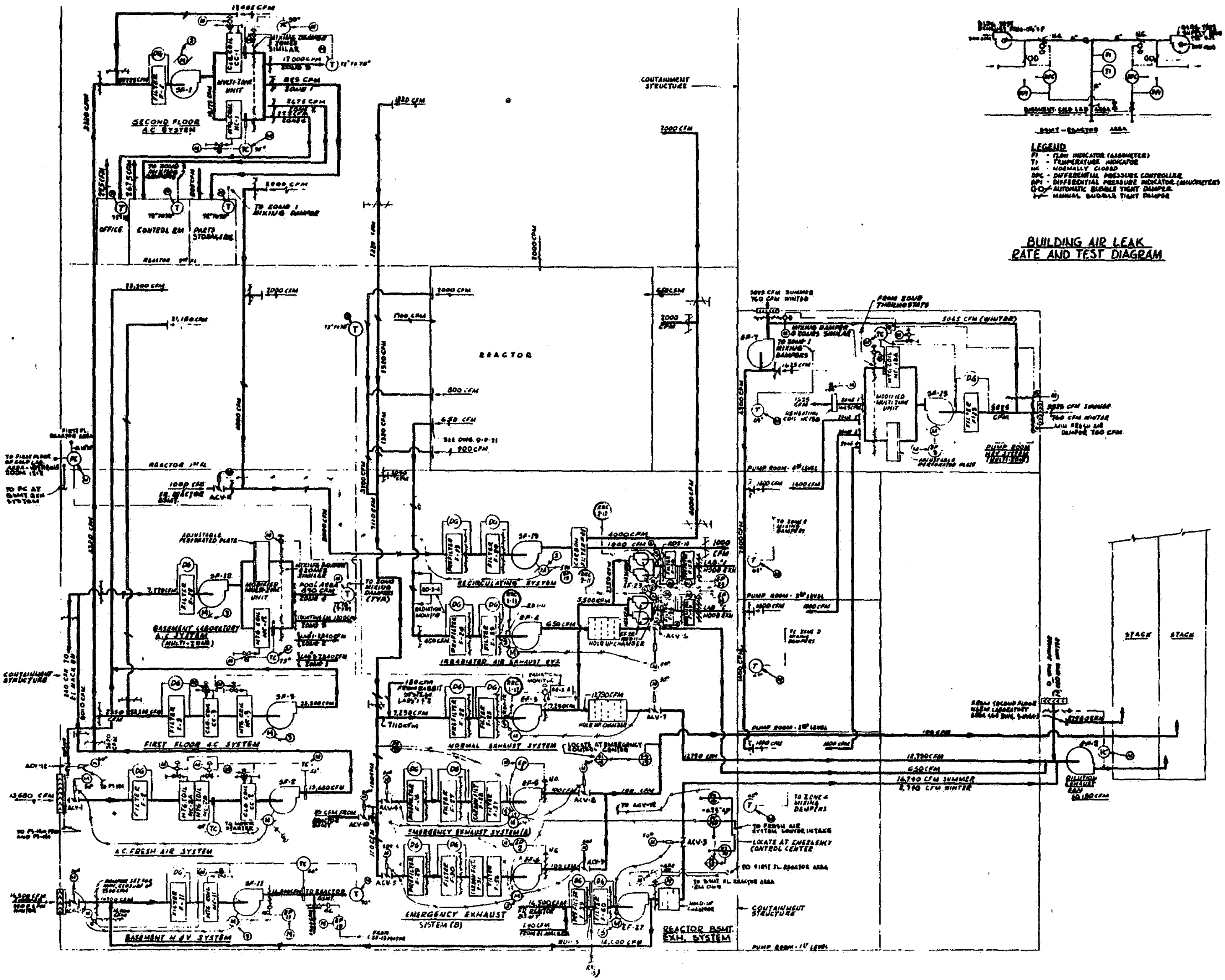


Figure 3.10 Ventilation systems flow diagram.

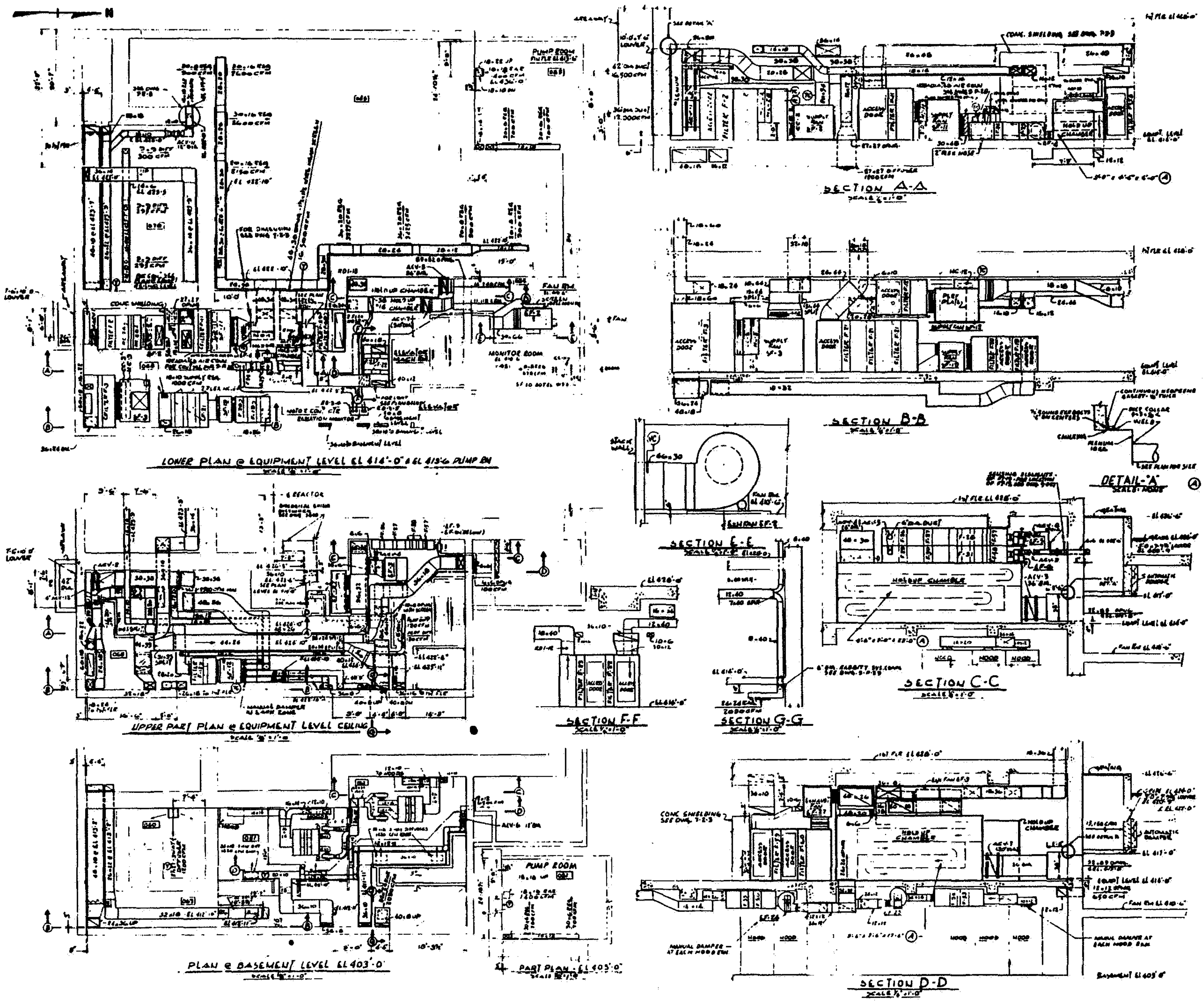


Figure 3.11 Ventilation system - mechanical arrangement.

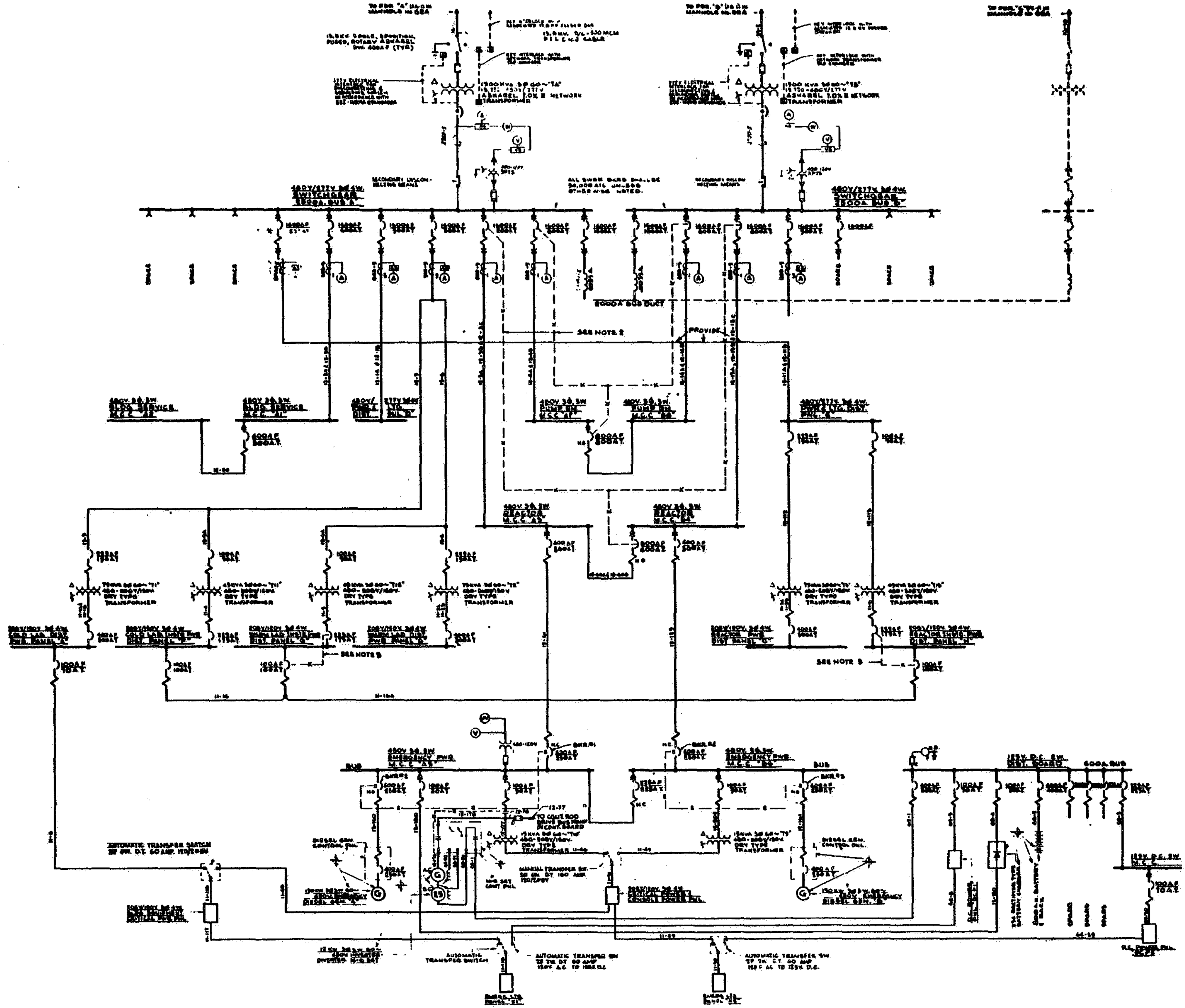


Figure 3.12 Main one line electrical diagram.

SECTION 4. REACTOR CORE

4.1 GENERAL DESCRIPTION

4.1.1 BASIC REACTOR TYPE. The NBSR is a D_2O cooled, moderated, and reflected reactor designed to operate at 10 MW thermal capacity. The fuel elements are located on 7" centers in an hexagonal array. This relatively large spacing makes the NBSR a well thermalized reactor, and at the same time makes it possible to introduce many in-core experimental facilities in addition to the beam holes and thimbles located in the reflector.

Cooling water enters through a plenum at the bottom of the fuel, passes up through the fuel and into the reactor vessel, and then out through two outlet tubes in the bottom of the vessel. A certain amount of freedom in the arrangement of the fuel elements within the core is made possible by the use of a double plenum system. The inner 7 fuel positions are fed by 1 plenum and the remainder by a second concentric plenum.

The reactor is unpressurized except for a small pressure of about 4" of water maintained by a helium blanket. The inlet temperature of the D_2O coolant is 100°F and its out-temperature is 112°F. Since most of the neutron moderation is done in the D_2O surrounding the fuel, rather than within the fuel element itself, the average moderating temperature is approximately the same as the coolant outlet temperature.

4.1.2 REACTOR SIZE. Figure 4.1 shows an elevation drawing of the reactor. The core is contained in an aluminum tank 7' in diameter and 16' high. By the use of fuel elements with an unfueled center section, the core is split into an upper and lower section. Each section is 44" in diameter and 11" thick. The unfueled section between the two fueled sections is 7", thus the overall dimension of the core is 44" diameter by 29" high.

The fuel elements are supported by two grid plates 62" apart. The top of the lower grid plate is 9" below the bottom of the core and the bottom of the top grid plate is 24" above the top of the core. The fuel elements are held in place against the upward water flow by a locking mechanism held down by the upper grid plate.

The side reflector is 20" thick and the top reflector thickness is determined by a 3" overflow pipe which maintains a water level at 117" above the top of the core. This large space above the core allows transfer of fuel elements to the fuel element transfer chute.

An emergency dump line concentric with the fuel element transfer chute can be used to drop the water level to 1" above the core. Not shown in Figure 4.1 is another low level overflow concentric with the 3" overflow pipe located at the upper grid plate elevation. This overflow is to be used whenever fuel elements are being transfered in a helium atmosphere (low water level) rather than under water.

4.1.3 CORE ARRAY. The grid plates provide for 37 fuel element positions and four, 2-1/2", semi-permanent irradiation thimbles. The normal core configuration is shown in Figure 4.2. Seven of the fuel positions are especially adapted for 3-1/2" experimental thimbles and 6 other positions on the outside corners of the hexagon are normally loaded with dummy fuel elements, leaving 24 positions for normal fuel element use. With the normal loading of 24 fuel elements, each of the seven 3-1/2" experimental thimbles is surrounded by 6 fuel elements. Only six of the seven 3-1/2" experimental positions are available for in-pile irradiation, since the seventh position is used for a regulation rod.

4.1.4 EXPERIMENTAL FACILITIES IN THE REFLECTOR. The experimental facilities will be described in detail in Section 8. Here we will just briefly point out the location of the main facilities. Radial beam ports are all located in the central plane and look at the unfueled section of the core. The large beam hole is provided for the future installation of a low temperature moderator for use in the production of high intensity, low energy, neutron beams. Two through tubes pass below the radial beam tubes at the bottom of the lower core. The reflector also contains four rabbit tubes and positions for the installation of up to 7 vertical thimbles.

4.1.5 REACTOR CONTROL. The reactor is controlled by 4 shim safety arms and 1 regulating rod. The shim safety arms are the semaphore type similar to those used at CP-5. As shown in Figure 4.1, they are located just below the upper grid plates. Their poison segment is .040" thick cadmium clad with aluminum.

The regulating rod consists of a 3" diameter aluminum cylinder filled with helium. The poison is supplemented by the helium void. It is mounted on a vertical rod located in one of the 4" experimental facilities in the core.

4.1.6 EMERGENCY SHUTDOWN MECHANISM. There are 3 emergency shutdown mechanisms. The primary one is the shim safety arm system. This can be backed up by the second mechanism which dumps the top reflector to a level 1" above the core. Since our top reflector is so thick, it takes about 40 seconds to dump, so it appeared desirable to supplement this mechanism with a third system. The third system is the void shutdown system which can introduce helium gas into the region below the core, thus forming helium voids within the core. The helium bubbler works rapidly and within a second or two introduces enough void to cause a negative reactivity sufficient to initiate shutdown. The void shutdown system is so interlocked that it will always initiate an emergency dump of the top reflector.

4.1.7 D_2O HOLDUP TANKS. In the event of a major rupture of the sub-pile piping that would drain the reactor vessel, D_2O will be held up in two places within the reactor vessel itself. These can be seen in Figure 4.1. An annular shaped tank located in the top reflector can only be drained through two pipes at the bottom. These pipes feed a distribution pan which routes emergency cooling water to the individual elements in the core. Water can only be drained from this pan through fuel element seats in the lower grid plate. The end fittings of the fuel elements and any other tubes inserted into the lower grid plate are conically shaped so as to minimize leakage of water down through the fuel element seats. The pan then tends to keep the lower core submerged in the water and also serves to collect any of the water from the upper emergency tank which splashes over the top of the distribution pan or runs down the outside of the fuel elements.

4.2 FUEL ELEMENTS

4.2.1 GENERAL DESCRIPTION. The NBSR fuel element is an MTR plate type element as shown in Figures 4.3 and 4.4, which give all pertinent dimensions and tolerances. The fueled section consists of 17, two-core fuel plates. Two unfueled, curved, outside plates and two side plates form a box section structure which surrounds the fuel plates.

The bottom adapter is both an inlet nozzle and a valve. Coolant enters the internal passage of the bottom adapter, flows up through an internal conical transition section to the lower portion of the box section, through the channels between the fuel plates and between the fuel plates and the outside unfueled plates, and then through the upper box section and out the top adapter. A small amount of coolant is also bypassed around the external surface of the lower nozzle to prevent any possibility of bulk stagnation. This bypass flow is possible only when the exterior conical section of the lower adapter is lifted off a mating conical seat in the lower grid plate. A 12 mil gap then exists between the nozzle and the hole in the grid plate. The element is spring loaded down by the latching mechanism which will be discussed in a following paragraph. The lifting force necessary to achieve the bypass flow results from the hydraulic drag of the coolant on the fuel assembly. If, however, flow should cease for any reason, the elements will drop down on the seats and so hold a portion of the bulk coolant in a pan-like structure which surrounds the core to mid-fuel height. The application of this system is discussed in Section 7.1.

The upper adapter contains the mechanism which locks the fuel element into the grid plate structure. The details of the locking mechanism are shown in Figure 4.5. This is a spring loaded cross bar type of lock. When the fuel element has been fully inserted through the upper grid plate into the lower grid plate, further downward pressure on the handling head will cause the spring to compress and will bring the cross bar down, inside the upper adapter, to a position just under the upper grid plate. Counter clockwise rotation of the handling head with continuing downward pressure will cause the cross bar to rotate. The ends of the cross bar will then project out through the side windows of the upper adaptor and pass under the bottom surface of the upper grid plate. Release of the downward pressure will then allow the spring to pull the cross bar up into small notches in the bottom surface

of the upper grid plate. The fuel element is thus locked between the grid plates. The notches in which the cross bar is resting prevent the bar from turning out until a downward force is again applied by a handling tool. The spring is used only to assure that the cross bar engages the grid plate properly when there is no coolant flow. During operation, the water flow forces the element directly against the cross bar forcing it firmly into the notch in the grid plate.

The fuel elements are located on 6.928" centers in the NBSR core. Since each element fits into a unit cell of 3-1/8" x 3-13/15", there is a good deal of space between elements. The tolerances of the outer configuration are not, then, of primary importance.

4.2.2 MATERIALS. All materials used in the NBSR fuel element contain less than 5 ppm of boron and less than 15 ppm of cadmium. The fuel alloy is an uranium aluminum alloy containing 18 weight percent enriched uranium. The uranium is enriched to a minimum of 93% U-235. The U-235 content of each fuel element is 170 ± 3.4 grams. The U-235 content of each fuel core (two cores per plate) is 5 ± .15 gram. The aluminum melting stock for the fuel core alloy is 1100 series (ASTM specification B 179-62, Alloy 995A or a higher purity grade).

The fuel plate core frames and cladding are 1100 series aluminum (ASTM specification B 209-62).

The side plates and the unfueled outer plates are aluminum, alloy 6061 Temper T-6 (ASTM specification B 209-62).

End adapter castings are of type 356 aluminum tempered to the T-6 condition (ASTM specification B 26-60T Alloy SG70A).

4.2.3 FABRICATION METHODS. The NBSR elements are fabricated by what have become standard industry techniques for the manufacture of MTR plate type fuel elements. The cladding and core materials are metallurgically bonded by hot rolling processes. After blister testing (see below) the plates receive a final cold reduction of not less than 15% or more than 25%. Final curving is accomplished over a warm die.

Several methods of fastening side plates to fuel plates are available. The first group of NBSR elements will be assembled by the roll-swaging process. The element specification requires that all the side plate to fuel plate strength must exceed 150 lbs per linear inch of joint.

The upper and lower end castings are welded to the box section formed by the two unfueled outside curved plates and the two side plates. The lower adapter is attached with a full seam weld to form a watertight joint. The upper adapter is attached with 4 plug welds on each of 4 sides. All welding is done by the inert gas shielded air process.

Each fuel core is assigned a serial number which is recorded along with the melt serial number, the U-235 enrichment fraction for that melt and the U-235 content of the core based on melt chemical analysis and core weight. Each fuel plate is then assigned a serial number which is recorded along with its core numbers. These fuel plate numbers are then recorded along with their position in a serial numbered fuel element. The element serial number is engraved on the side plates on both sides. One serial number adjacent to each half of the split fuel core. Thus, if for disposal purposes, a cut is made through the gap in the fuel plates both ends will carry identifing numbers.

4.2.4 TESTING AND INSPECTION. Samples for chemical analysis are taken from the top, center, and bottom of each fuel alloy ingot for chemical analysis. The fuel cores stamped from the rolled ingot are numbered so that they can be related to that specific melt.

After each fuel plate is hot roll-bonded it is blister tested by being heated to 900°F and held at that temperature for one hour. Any plate which shows evidence of blistering or lamination is rejected.

After blister testing, each plate is ultrasonically tested for voids, inclusions, or other discontinuities; any such irregularities are cause for rejection.

Prior to assembly, the surfaces of the aluminum cladding on the fuel bearing section is examined for pits, scratches and dents. Pits or scratches greater than 0.003" deep over fuel or 0.006" on any other surface will result in rejection of the plate. Dents greater than 1/8" in diameter and/or greater than 0.006" deep will also result in rejection of the plate.

Welds are usually inspected for evidence of cracks, inclusions, and inadequate penetration into the bare metal. Unsatisfactory welds must be repaired to meet specifications.

Each fuel plate is fluoroscoped to establish the outline of the fuel alloy material for final shearing to size. All fuel plates are then radiographed and the radiographs examined for segregation and variation of uranium in the fuel alloy. Those plates which show any variation in density detectable on the radiographs, by eye, are rejected.

One randomly selected fuel plate from each rolling lot (or a minimum of 1 out of 40) is destructively tested for cladding thickness. A small longitudinal sample from each end of the two fueled sections plus a transverse section through the center of each fuel core (a total of 6 samples) are examined microscopically to establish the actual thickness of cladding covering the fuel filler and to determine the amount of local thickening (dog-boning) at the fuel core ends. Cladding thickness less than .011" will result in rejection of the entire rolling or sampling lot. Fuel core thickness of .020"(+0.000"/-0.002") (0.23" maximum 1" from the ends of each core) is specified and will also be criteria for lot rejection. For the purpose of this inspection, a rolling lot is defined as all plates rolled during a given rolling operation on a given piece of equipment at given equipment settings by a given operator at given room temperature and environmental conditions.

The efficiency of the roll swagging assembly technique is tested by determining the force which is necessary to fail test sections. Minimum joint strength of 150 lbs per linear inch of roll swagged joint are required by NBSR specifications. The assembly of elements in the production run is done in the same manner and by the same operations used in the fabrication of satisfactory pull test specimens.

After assembly of the element the water channel spacing is measured along the element centerline in each water channel of each element. These measurements, together with the other dimensional measurements, are submitted as part of the inspection data on each fuel element. Certified copies of reports identifying all materials used in the fabrication of the fuel assembly are also required.

After fuel assemblies are received at NBSR, the inspection procedure calls for another full dimensional check, including water channels, as well as a complete visual check for surface defects on fuel plates. The first group of fuel elements received at NBSR will all be hydraulically tested at flow rates exceeding the maximum flow to be achieved in operation of the reactor. Sampling techniques for hydraulic testing of future elements will be developed using the experience gained during the initial testing program.

4.2.5 PROTOTYPE TESTS. The outer shell of the NBSR fuel element represents the only major variance from the classic MTR plate type fuel element. Since this outer shell controls the establishment of the proper hydraulic regime, for heat transfer purposes, confirmation of the structural and hydraulic design objectives was accomplished on a hydraulic stand, using a fuel element assembly fitted with dummy plates. Flow rates over twice those seen in normal operation (20 to 30 ft/sec) were employed to measure flow conditions in each channel and across typical channels as well as total pressure drop, drag forces, bypass flow around the lower nozzle and the vibration characteristics of the spring loaded element lock. The results of those portions of the test results which effect heat transfer are discussed in Section 4.7. In general, the predicated performance of the NBSR element design was confirmed. The primary features of uniform flow delivered to all channels, lack of structural deformation, and absence of vibration were all proven.

4.2.6 PAST EXPERIENCE WITH SIMILAR ELEMENTS. The enriched uranium fueled plate-type element with aluminum or a structural cladding material has a long and trouble-free history in research and test reactor technology. All of the variations in the basic plate-type element derive from the MTR design and development work done about 1950. The MTR itself commenced operation in 1952. Since that time a variety of reactors using the same general type of element have been built and operated in this country and abroad. The basic

Table 4.2-1 Fuel Element Operating Conditions in High Power Density Research and Test Reactors.

Reactor	NBSR	ORR	MTR	ETR	HFBR
Core Inlet Water Temperature, °F	100	120	115	120	120
Water Velocity in Fuel Element Channels, ft/sec	12	30	33	35	35
Core Pressure Drop, psi	12	25	40	45 nominal 55 maximum	31
Nominal Water Channel Thickness, in.	.116	.104	.116	4 at .119 2 at .115 12 at .105	2 at .129 2 at .116 2 at .108 12 at .102
Fuel Plate Thickness, in.	.050	.050 inside .065 outside	.050 inside .065 outside	.050	.050 inside .140 outside
Fuel Meat Thickness, in.	.020	.020	.020	.020	.020 inside .010 outside
Weight % U in Fuel Alloy	18	18	18	22	30
Width of Fuel Plates Between Side Plates, in.	2.415	2.512	2.622	2.624	2.446
Curved or Straight Plates	Curved	Curved	Curved	Straight	Curved
Radius of Curvature, in.	5.5	5	5.5	- -	6
Maximum Heat Flux, Btu/hr-ft	3.82×10^5, first core 2.68×10^5, equilibrium	7.5×10^5	9×10^5	1.35×10^6	1.60×10^6, first core 1.48×10^6, equilibrium
Hot Spot Surface Temperature, °F	240	240	312	400	359 first core 344 equilibrium
Average U-235 Burn-up, %	40	35-40	20-25	17	20.4

plate-type fuel element, operating at coolant conditions and power densities far more severe than those of the NBSR, has by this time many hundreds of megawatt years of successful operating experience behind it.

4.2.7 OPERATING CONDITION, LIFETIME, AND CORROSION. The life cycle and burnup of the NBSR fuel element is discussed in Section 4.6.4.

Temperature, pressure, and water flow conditions are summarized in Table 4.2-1 and discussed in detail in Section 4.7.

The corrosion history of aluminum MTR type fuel elements has been studied extensively. Fuel plates of the same basic construction and the same material as those used in the NBSR element have been operated at higher flows, higher temperatures, and at much higher heat fluxes (Table 4.2-1) than will be achieved in the NBSR. All of these factors generally increase the corrosion rate and yet corrosion of the fuel elements during lifetimes comparable to those in NBSR has not been a problem from the standpoint of structural integrity. The buildup of the low conductivity oxide layer on the heat transfer surfaces is, of course, a matter for consideration. This effect is discussed in Section 4.7.

4.3 CONTROL RODS

The NBSR has two types of control rods. Primary control of the reactor is accomplished by use of 4 semaphore type shim safety arms. Fine control is accomplished by the use of a single vertical type regulating rod.

The 4 shim safety arms are each 1" thick by 5" wide by 52" poisoned length. The hollow interior is filled with helium and the .040" thick cadmium poison is clad with aluminum on both the outside and inside. They are mounted on hanger brackets just under the grid plate. The drive shafts penetrate the reactor vessel below water level and drive the shim arms directly. The vessel penetrations are sealed with rotating seals and the drive mechanisms are mounted in recesses in the biological shield. The total reactivity worth of the four shim safety arms will be about 46% and the worth of a single arm about 14%.

The regulating rod consists of a helium filled aluminum cylinder, 3" in diameter by 29" long. It is located in one of the 3-1/2" vertical thimbles directly in the core as shown in Figure 4.2. The vertical drive mechanism is mounted in the top plug and is a standard commercial design. The volume of the regulating rod void combined with its aluminum structure as a poison is designed to make the rod worth 1/2 percent reactivity.

4.3.1 SHIM SAFETY ARMS. Figure 4.6 shows the detailed design of the NBSR shim safety arms and the location of the various materials of construction. These arms are identical to the arms used in the CP-5 reactor with the exception of a slight increase in length.

4.3.1.1 Materials. The neutron absorbing element in the shim safety arms is cadmium, in sheet form 99.9% pure. This cadmium is clad on both sides with 1100 series aluminum. The hub portion of the assembly is aluminum alloy 6061-T6.

4.3.1.2 Fabrication Methods. The blade portion of the NBSR shim safety blades is rough formed by assembling three concentric tubes to form an aluminum, cadmium, aluminum sandwich. The aluminum tubes are seamless extrusions; the cadmium tube is roll formed to the diameter needed. This sandwich section is then collapsed to the approximate configuration of the completed arm. The ends of the section are then seal welded to prevent the entry of lubricants or other foreign matter and the section is mounted on a draw bench. Roughing and finish dies are then drawn through the inside of the section. Small inelastic strains occur which bring the section to its final dimension and, more importantly, assure intimate contact between the cadmium and the aluminum at all points.

The hub section is machined from plate stock. This section contains a drilled hole which leads from the outer surface of the hub to the void in the center of the blade when the assembly is complete.

Assembly of the two sections is accomplished by welding. After all traces of lubricant from the drawing operation have been removed, the ends of the sandwich section which

were previously seal welded are cut back to expose a clean edge. The hub is then forced into the blade section and the joining welds made by the inert gas shielded tungsten electrode method. The end of the blade section is capped with a fitted aluminum end piece which is welded in place by the same technique.

The void in the blade section is then repeatedly evacuated and purged with helium. Helium at just slightly above atmospheric pressure is left in the void. The opening through which the void was purged and filled is then seal welded.

4.3.1.3 Inspection and Tests. The components of the shim safety blades and the final assembly are thoroughly inspected for adherence to the design drawing requirements and tolerances. The blade section receives additional inspection to assure intimate contact between the cadmium and the aluminum.

Radiography of sample blade sections followed by destructive testing of these sample sections provided the information necessary to establish radiographic standards for the inspection of completed blades. All parts of the completed blades were then radiographed and the radiographs compared to the standards for the necessary extrusion of the cadmium into voids which existed before the drawing operation and for the absence of new voids or inclusions. Any such irregularities visible by eye are cause for rejection.

After complete assembly of each unit, all welds were radiographed and visually examined for cracks, checks, absence of penetration or undercut. Unsatisfactory welds are cut out, repaired, and reinspected.

The leak tightness of the assembly is checked by filling the void in the blade section with helium at approximately 15 psig and then searched for leaks over the entire surface and at all joints with a mass spectrometer helium leak detector. No helium leakage is allowed on acceptable units.

4.3.1.4 Operating Conditions. Each NBSR shim safety arm has an operational travel of 41°. When in the full-in position, the blade centerline is 41° below the horizontal. Full retraction brings the blade to the horizontal position, just below the upper grid plate.

The blades are supported by a hub-unit which rides on two ball bearings. These bearings, in turn, are mounted in a hanger bracket. The hanger brackets are bolted to the reinforcing ring welded to the vessel.

The drive systems and shock absorbers are mounted on the biological shield. A stainless steel, splined shaft connects the drive units to the arm assemblies. The arrangement of the system, including the position transmitters and over drive limit switches, is shown in Figures 4.7 through 4.9.

4.3.1.5 Drive Package Design. The shim arm drive package is identical in all major respects to the system which has been in use on the CP-5 reactor for some years. Essentially the drive consists of a large compression spring which is compressed by a ball nut and screw jack when the shim arm is raised. The shim arm shaft is connected to the housing which holds the ball nut and so the arm is raised or lowered as the nut rides up and down the screw.

The ball screw jack is driven, in turn, by an electric motor, through a high ratio gear case and finally through an electromagnetic clutch. The arrangement of the parts is shown in Figure 4.9. Whenever the arm is raised, the compressed spring is pushing on the very low friction ball nut, attempting to force it back down to its rest position. This, of course, would require the screw to turn, but it cannot turn because it is connected through the clutch to the output shaft of the high ratio gear box. Should the clutch be disengaged, however, the screw is free to turn and the spring will ram the nut, and so the shim arm, back to the "in" position.

The energy of a rapid return or "scram" is absorbed by a hydraulic shock absorber. This shock absorber is mounted on the biological shield, adjacent to the drive package. A mechanical stop, to prevent over travel of the arm should the shock absorber bottom out, is located on the linkage which connects the shim arm shaft lever to the shock absorber.

All impact loads are, therefore, born by the biological shield.

4.3.1.6 Shaft Seal. The shaft connecting the drive package and the shim arm must be sealed where it passes through the vessel wall. This seal is accomplished by a Kopper Company face type seal. These seal units were tested at 50 psi water pressure both by the manufacturer before shipment and when installed at the NBSR. No leakage was found at 50 psi, normal operating pressure is 3-1/2 psi maximum.

4.3.1.7 Shim Safety Arm Lifetime. The lifetime of the shim arms is affected by poison burnup, corrosion, and radiation damage. As will be discussed below the latter two effects are not important relative to the poison burnup rate.

The poison in each shim arm consists of a total thickness of .080" of cadmium. The burnup rate during shutdown, when the rods are fully inserted, is negligible compared to the burnout rate during operation when the arms will normally be withdrawn above the core at an angle of about 10°. In the presence of the shim arms, the flux will fall rapidly with distance above the core so the shim arms will burnout much more rapidly at the bottom. Assuming that, due to the effect of the shim arms themselves, the flux is signigicant only over the bottom 2" of the arm, it will take slightly over two years of 290 day per year operation at 10 Mw to burn up enough cadmium to reduce the minimum four arm shutdown margin from 32% to 29%. Similarly the 2 arm shutdown margin would be reduced from 13% to 10%. Since even a very thin section of cadmium is just as black to thermal neutrons as a thick sheet, the shutdown margin is only changed by complete cadmium burnout. Thus if the high flux is spread out over a larger area of the shim arms than assumed above, complete burnout would be delayed and the arms would last somewhat longer. So the shim arms should be able to operate for at least two years without replacement before any significant change in the minimum shutdown margin occurs.

That corrosion does not limit the shim arm lifetime is demonstrated by the fact that similar arms have remained in the CP-5 reactor for about 8 years until poison burnout required their removal. This is a much longer period of time than anticipated for the NBSR shim arms.

The radiation damage is not great because during reactor operation, the shim arms are in the top reflector above the core where the fast neutron flux is relatively low. The shim arms of the CP-5 reactor are located in about the same position, but due to the smaller core size and other geometrical differences they are exposed to a slightly higher fast flux per unit power than are the NBSR arms. Thus, the CP-5 arms would have been exposed to at least as much radiation damage for the same Mw-hours of operation as the NBSR arms. The first arms were removed from CP-5 after 121,000 Mw-hours of operation (4.1) - an exposure comparable to that discussed above for the NBSR arms. Since the CP-5 arms did not exhibit any serious radiation damage effects, the NBSR arms should not be affected by radiation damage for the exposures realized during its poison burnout limited life.

4.3.2 REGULATING ROD

4.3.2.1 Control Philosophy. The regulating rod in the NBSR is a hollow aluminum cylinder 29" long by 3" O.D. The interior is sealed and filled with helium. The relatively large volume of the rod allows the use of aluminum as the poison which is supplemented by the void worth of the rod. The low absorption cross section of the poison (aluminum) insures the long life of the absorbing atoms; and the spreading out of the poison over a large volume minimizes the local thermal flux depression. Furthermore, the total neutron absorption required of the rod is reduced because a portion of the reactivity control is achieved by the rod void. The void works primarily by changing the fast neutron leakage and so has very little effect on the thermal neutron flux.

These features combined with the location of the rod near the center of the reactor, see Figures 4.1 and 4.2, is expected to cause very little perturbation of the thermal flux at the beam holes as the rod is moved.

4.3.2.2 Drive Package. The NBSR regulating rod (fine control) is driven by a standard drive package built by the Diamond Power Specialty Corporation of Lancaster, Ohio.

The drive train consists of a two phase electric servo motor which drives an extremely accurate lead screw-nut combination. An extension shaft mounts on the nut at one end, and carries the regulating rod at the other. As the screw revolves, the nut, and thus the regulating rod moves up or down at a fixed rate. The rate of movement, 30" per minute, is determined by the motor speed and the pitch of the lead screw.

Position indication transmitters are connected to the motor shaft through servo gear trains. Regulating rod position indicators mounted on the main control panel show the rod position to .02". Operational tests have shown the maximum error between true and indicated position to be less than .01".

The total rod travel is 29". Limit switches mounted in the drive package are used to indicate and limit the extremes of travel as well as to signal the operator when the rod is 7" from each extreme end. Knowing this, the operator may re-shim to keep within the limits of the regulating rod operational range.

4.3.2.3 Regulating Rod Lifetime. Since the regulating rod poison is simply the aluminum out of which the rod is fabricated, poison burnout presents no problem. (The half life of an aluminum atom in a flux of 10^{14} n/cm^2-sec is about 1000 years). Since the rod is made of the same material as the rest of the core structure, it will suffer no more corrosion than the other core components.

In a 20 year life, it would experience an integrated fast neutron ($>$.3 ev) exposure of about 5×10^{23} nvt. This is less than anticipated for the exposure of similar alloys in the HFBR (4.2), and in view of the low stress non-structural nature of the rod should present no problem.

4.3.2.4 Regulating Rod Heating. The regulating rod will operate in a shroud of approximately 3-1/2" I.D. The shroud has the same general configuration as the 3-1/2" experimental thimbles. A fixed orifice in the nozzle of the shroud will deliver a coolant water flow of 8 gpm from the inner plenum. This flow will pass up around the regulation rod and then out into the bulk coolant.

At the calculated heating rate of 2.9 watts per square centimeter, at 10 MW power, this coolant flow will result in a maximum regulation rod wall temperature of 140°F.

4.4 OTHER COMPONENTS RELATED TO THE REACTOR CORE

In addition to the fuel elements and control rods, there are three other types of structures in the reactor core. When the core contains its normal arrangement of 24 fuel elements, there are, in addition, seven 4" experimental thimbles, four 2-1/2" experimental thimbles, and six dummy elements without plates. With the exception of the four 2-1/2" experimental thimbles, all these structures have a lower end fitting whose external dimensions are the same as the fuel elements. It is possible to replace either the 3-1/2" experimental thimbles or the dummy elements with normal fuel elements. The only modification needed for the fuel elements to be placed in the positions normally occupied by the 3-1/2" experimental thimbles is a modified top fitting.

Each of the three types of non-fuel structures has a hole of appropriate size in its lower end fitting to allow water from the plenum to flow through it to provide adequate cooling for itself and any experiments within it.

4.4.1 3-1/2" EXPERIMENTAL THIMBLES. The major features of the core structure are shown in Figure 4.1. For convenience of display, each type of in-core structure is shown without overlap so their relative positions differ slightly from the true structure. The true locations are shown in Figure 4.2.

The 3-1/2" experimental thimbles are held down against the upward thrust of the water by tubes fixed to the top plug. To remove these thimbles, the hold down tubes must be removed and a simple adapter inserted in the top of the thimble which allows it to be handled just like a fuel element by the transfer system.

Experiments are inserted through the top plug and hold-down tube into the thimble. If more cooling is required than can be supplied by the water flowing through the thimble, it can be supplied from the top along with any special instrumentation that may be desired. These experiments do not interfere with the transfer or removal of fuel elements.

4.4.2 POISONED HOLD-DOWN TUBES. The tubes which hold down the 3-1/2" experimental thimbles pass through certain of the transfer tools and are fastened to them. In addition to holding down the 3-1/2" experimental thimbles, they serve another purpose. They contain poison in the region between the dotted lines as shown in Figure 4.1. The poison section is a .040" thick cadmium tube sandwiched between two aluminum tubes. This region is far enough from the core so it has no effect on core reactivity or shim safety arm control. The maximum neutron flux in the vicinity of the poisoned region is about 2×10^{12}. It is, of course, much lower in the cadmium itself, and the lifetime of the bottom 10 cm of the cadmium is estimated to be 30 years. The poison greatly reduces the neutron flux at the bottom of the top plug and minimizes the activation of the fuel element transfer mechanism and the bottom of the top plug. This will greatly simplify the maintainance of the system should it be required. The poison will also greatly reduce the activation of supports and connections for experiments in the thimbles.

4.4.3 2-1/2" EXPERIMENTAL THIMBLES. As seen in Figure 4.2, each of these smaller thimbles is located between two fuel elements. This will provide a higher fast flux than is normally available in the larger thimbles.

These thimbles are semi-permanent, being held down by the top grid plate as shown in Figure 4.1. Cooling water from the plenum flows through a hole in the bottom of the thimble in the same way as it does for the larger thimbles.

4.4.4 REGULATING ROD STRUCTURE. The regulating rod is permanently located in one of the 3-1/2" experimental locations. With the exception of the details of the attachment of the hold-down tube to the top plug and the mounting of the regulating rod drive mechanism, the hold-down tube and thimble are the same as those for the other 3-1/2" thimble positions. The rod is cooled by water coming up through the 3-1/2" thimble from the plenum.

4.4.5 TOP AND BOTTOM GRID PLATE INSERTS. Two inserts which are bolted into the center of each grid plate to fill 7-1/8" diameter holes can be seen in Figure 4.1. The openings in the grid plates are as large as possible without interfering with other core structures. They are included in the design in order not to preclude the possibility of installing a large central thimble or tube at some future date without replacing both grid plates. Normally, a 3-1/2" experimental thimble occupies this center location as shown.

4.5 CORE SUPPORT STRUCTURE

The internal structure of the reactor vessel supports the core, the shim safety arms, and the inner emergency cooling tank.

4.5.1 CORE STRUCTURE. The fuel elements, experimental thimbles, dummy fuel elements, and regulating rod shroud are located between an upper and lower grid plate as described in Section 4.1. The grid plates are detailed in Figures 4.10 and 4.11 and their relationship to other reactor components is shown in Figure 4.1. The grid plates determine the fuel element and 3-1/2" experimental thimbles spacing at 6.928" center-to-center. The fueled portion of the elements extends from 9" above the bottom grid plate to 24" below the upper grid plate with a 7" unfueled section in the central plane. Coolant flow through the core structure comes from 2 concentric plena located just below the lower grid plate. The D_2O coolant passes up through the fuel elements, experimental thimbles, dummy elements, and regulating rod shroud. In addition, a small amount flows around the base of the fuel elements directly into the bulk coolant.

With the exception of the 3-1/2" experimental thimbles, which are held down by tubes from the top plug (see Section 4.4.2), all the core components are held down against the upward thrust of the water by the upper grid plate. The fuel elements are locked under the upper grid plate as described in Section 4.2.1, and the 2-1/2" experimental thimbles are semi-permanently held down directly by the upper grid plate. Thus, the upper grid plate must take the upward thrust of the core components pushed by the water. The lower grid

plate is loaded by the hydraulic pressure in the two plena, the weight of the core components when water is not flowing, and thermal stresses arising from the radiation heating. The grid plate loadings under operating conditions are given in Table 4.5-1. Both grid plates are made of aluminum alloy (ASTM B209-62, 6061-T6).

Table 4.5-1 Grid Plate Operating Conditions

	Upper Grid Plate	Lower Grid Plate
Plenum pressure load - lbs		27,600
Fuel element drag - lbs	540	
Weight of core components - lbs		1,100
Maximum stress (mechanical plus thermal) - psi	1,570	1,690
Design allowable working stress - psi	6,300	6,300
Maximum deflection - inches	0.01	0.05

The lower grid plate is completely supported at the edges by a 1" plate welded to the outer plenum. The plate also has sections extending to the vessel wall where they are welded for further support (see Figures 5.13 and 5.14 in Section 5). The grid plate is fastened to this supporting plate by eighteen 1" diameter stainless steel bolts. The large number of bolts is employed to give a relatively watertight seal between the lower grid plate and its mounting surface, and their loading capacity far exceeds that required to handle the grid plate loading.

The upper grid plate is mounted on four mounting brackets welded to the vessel wall (see Figures 5.9 and 5.10 in Section 5). These brackets are further reinforced by quarter rings welded to them and to the vessel wall. The grid plate is bolted to the mounting brackets by ten 3/4" diameter stainless steel bolts.

4.5.2 SHIM SAFETY ARMS AND INNER EMERGENCY COOLING TANK SUPPORT. The 24" space between the top of the fueled region and the bottom of the upper grid plate is provided to allow the complete withdrawal of the shim safety arms into a region of negligible reactivity worth without passing through the grid plate. The shim safety arms operate through a range of 41° from the fully withdrawn, horizontal, position just below the upper grid plate to the fully inserted position (41° to the horizontal). Each shim arm is individually supported through a bearing by a hanger bracket. Each hanger bracket is then inserted into one of two mounting brackets and bolted in place. These mounting brackets are the same ones used to support the grid plate. The shim arm drive shafts are inserted into the hub of the shim arms from the side through the vessel wall.

The upper grid plate mounting brackets also serve to support the inner emergency cooling tank. It stands on four legs, each resting on one bracket, and is bolted in place by one bolt passing through each leg into the mounting bracket.

4.6 CORE NUCLEAR CHARACTERISTICS

4.6.1 SUMMARY. The main properties of the reactor core are summarized in Table 4.6-1. Except where otherwise stated, the parameters listed refer to the equilibrium core. The equilibrium core is the core loading achieved after enough refueling cycles have taken place so each element has the same history in the core.

4.6.2 FLUX AND POWER DISTRIBUTION. The nuclear characteristics of the core listed in Table 4.6-1 were determined from two group diffusion theory using the two dimensional, multi-regional Equipoise 3A code (4.3). The material parameters were modified in certain regions in order to better estimate the effect of several conditions which did not lend themselves to direct inclusion in the calculations.

Table 4.6-1 Summary of Core Nuclear Characteristics

Moderator and Coolant	D_2O
Reactor Power	10 Mwt
Core Volume	
Fueled Region	541 liters
Gap	174 liters
U-235 Operating Mass	
Equilibrium Core - 24 Elements - Start of Cycle	4090 grams
Start-up Core - 19 Elements	3230 grams
U-235 Critical Mass	
Equilibrium Core - 24 Elements	3340 grams
Start-up Core - 19 Elements	2790 grams
Average Metal Volume Fraction	0.085
Core Dimensions	
Diameter	111 cm
Top Fueled Height	28 cm
Gap Height	18 cm
Bottom Fueled Height	28 cm
Reactivity, Clean Cold-All Shim Safety Arms Withdrawn	9.7%
Reactivity after Xe and Sm have built up - All Shim Safety Arms Withdrawn	
Start of Cycle	5.4%
End of Cycle	3.1%
Total Reactivity Effect, All Shim Arms	-46%
Minimum Shut Down Margin - All Shim Arms	-32%
Reactivity Effect, Two Shim Arms	-25%
Minimum Shut Down Margin - Two Shim Arms	-13%
Void Coefficient of Reactivity	
Average Fueled Region	-.040%/liter
Gap Region	-.020%/liter
Temperature Coefficient of Reactivity	-.018%/°F
Neutron Lifetime	700μsec
Effective Delayed Neutron Fraction	.0080
Peak-to-Average Power Density, Equilibrium Core	
Inner Plenum	1.62
Outer Plenum	1.70
Whole Core	1.71
Peak-to-Average Power Density, Start-up Core	
Inner Plenum	1.53
Outer Plenum	1.69
Whole Core	1.82

In the core region, to account for the fact that the slowing down kernel in D_2O is not quite that represented by two group theory, a modified age was used. The modification follows that outlined by Hummel (4.4). The hardening of the thermal neutron spectrum due to neutron absorbtion in the core region was also taken into account by calculating the effective neutron temperature according to the prescription given in ANL 5800 (4.5).

The effect of empty beam tubes in the reflector was taken into account by dividing the reflector into several regions. The material properties of each region was modified to include, to a first approximation, the effect of neutron streaming out the beam tubes. The method used to account for the streaming is that described by D. J. Behrens (4.6).

4.6.2.1 Two Group Flux Plots. The flux distribution in the core was studied for a wide range of conditions, including loading at start and end of cycle, and ranging from more heavily loaded in the center, to more heavily loaded in the outer region of the core. Also the effect of shim arm position was studied. It was found that the flux distributions showed very little dependence on the radial fuel distribution within the core for the range of loadings reasonably applicable to the NBSR, including the fewer element start-up core (See Section 4.6.2.3). The vertical flux distribution, however, was found to be very sensitive to the position of the shim arms.

Figure 4.12 shows the radial flux distribution for the equilibrium core at the start of the cycle. The curves shown are taken through the central plane and through a plane 17 cm below the center in the fueled region. These curves are typical of all the core loading configurations considered.

The vertical flux is sensitive to the shim arm position. The thermal flux distributions (equilibrium core) for two shim arm positions are shown in Figure 4.13 and the fast flux distributions in Figure 4.14. The flux distributions are plotted along the central axis (R=0). The 10° shim arm position is estimated to be the just critical position when the Xe and Sm have approached equilibrium near the start of a cycle and the 16° position is the just critical position for the same core before the Xe and Sm poison have built up. The compression of the flux toward the lower half of the core is apparent and shows that the movement of the shim arms to compensate for xenon build up at the begining of a cycle will have appreciable effect on the flux distributions. Once the xenon poison has approached equilibrium, the shim arms will be largely out of the core and the further motion required to compensate for fuel burn-up will have much less effect on the flux distribution.

4.6.2.2 Effect of Gap on Reactor Control. Figure 4.15 shows the comparison between the curves of the statistical worths ($(\phi\phi^{\dagger})_s$) of the NBSR and the same core with the gap removed. The curves are vertical plots in the vicinity of the regulating rod. The similarity of the curves indicates that the presence of a gap in the NBSR does not introduce any unusual control features. This is further borne out by the calculated shim arm and regulating rod curves presented in Section 4.6.6.

4.6.2.3 Power Distribution. The energy distribution in U-235 fission is given in Table 4.6-2.

Table 4.6-2 Energy Distribution in U-235 Fission

Energy deposited at site of fission (fission fragments and β decay)	174 Mev
Total gamma ray energy	15
Fast neutron kinetic energy	5
Neutrino energy	10
Total	204

It is assumed that all of the neutrino energy escapes, so only a total of 194 Mev per fission appears as thermal energy. Of this about 2% escapes the primary cooling system and shows up in the thermal shield. Neutron caputre gamma rays in the core, however, contribute about an additional 2% and so just about compensate for the loss to the thermal shield.

The fraction of total thermal power deposited directly by fission in the fuel is then

$$F_d = \frac{174}{194} = 0.897$$

In addition, the gamma ray heating has been calculated to vary from 0.5 watts/gm at the core edge to 1.2 watts/gm in the core center. This gamma heating rate when applied to the fuel element, including unfueled side plates, contributes an additional 1-1/2 percent of the total power. Although this is more uniformly distributed throughout the element, it is assumed, conservatively, that it has the same distribution as the direct fission power. The energy lost by fast neutrons in the fuel is negligible. Thus, the ratio of power deposited in the fuel elements to the total power in the primary system is F_d = .897 + .015 = .91.

It is necessary to consider several different power distributions that arise in the NBSR. In addition to the equilibrium core, the startup core must be considered and the two shim arm positions of 10° and 16° must be included for each.

The power is proportional to the product of the thermal flux and fuel concentration. To find the power in each fuel element, vertical flux plots were made through each fuel element position. The product $m_i \int \phi_i(z)dz$ was computed where m_i is the U-235 mass in the ith element and the integral is taken over the fueled portion of the element. This product was then summed over all elements and normalized to 10 Mw and modified by the factor F_d discussed above. The peak-to-average power density ratio for each element is determined from the vertical flux plots of which Figure 4.13 is typical, but it must be further modified to include the fact that in all the calculations described above, the fuel was assumed to be uniformly and homogeneously distributed through the various core regions. Therefore, a unit cell calculation was made in which the fuel was concentrated in the same volume and geometrical distribution that it actually occupies in the real fuel element. Figures 4.16 and 4.17 show typical vertical and radial flux plots respectively. These data were used to calculate a hot plate factor of 1.10 for the outside fuel plate. The peak-to-average power density ratio from the unit cell calculation was compared to that from a homogeneous fuel distribution in an otherwise similar geometry. The comparison showed that the peak-to-average power density computed from a homogeneous fuel distribution should be increased by a factor of 1.15 to account for the heavier fuel loading actually required (see Section 4.6.3), and for the flux peaking at the edge of the fuel element. Table 4.6-3 summarizes the power distribution in the core modified by F_d = 0.91 and the peak-to-average power densities in an element corrected by the unit cell factor of 1.15.

Table 4.6-3 Power Distribution by Fuel Element Ring

Ring	No. of Elements Per Ring	P/ring Mw	P_{max}/el	R_n/el	R_s/el
Equilibrium Core					
Inner	6	2.26	398	1.53	1.63
Middle	6	2.20	384	1.55	1.67
Outer	12	4.64	431	1.50	1.62
		9.10			
Startup Core					
Inner	6	3.43	571	1.53	1.63
Middle	6	2.86	477	1.55	1.67
Outer	7	2.81	401	1.50	1.62

The fuel elements are arranged in three rings in the core. The inner and middle rings contain 6 elements each and the outer ring normally contains 12 except for the start-up core. Because, as discussed in Section 4.6.4, the elements progress in groups of three through the core during their lifetime, each ring contains at least two different fuel element loadings. Therefore, the table includes the power in the most heavily loaded element in each ring under the heading "Pmax/el", and its peak-to-average power density ratio.

Table 4.6-4 gives the same tabulation for each of the two plena except that the peak-to-average ratio is now for the whole plena instead of an individual element. This breakdown of data is given because the inner ring is fed by the inner plenum and the outer two rings by the outer plenum. Each of the two plena have separate water flow control and so the two parts of the core can be treated separately for heat transfer purposes. Finally, the peak-to-average ratio for the whole core is also included in Table 4.6-4.

Table 4.6-4 Power Distribution by Plenum

Plenum	No. of Elements per plenum	P/plenum Mw	P_{av}/el Kw	R_n	R_s
Equilibrium Core					
Inner	6	2.26	377	1.62	1.73
Outer	18	6.84	380	1.70	1.84
Whole Core	24	9.10	379	1.71	1.85
Start-up Core					
Inner	6	3.43	571	1.53	1.63
Outer	13	5.67	436	1.69	1.82
Whole Core	19	9.10	480	1.82	1.94

All data in Tables 4.6-3 and 4.6-4 are for conditions near the start of a cycle when peak-to-average ratios will be most severe. In the tables, R_n stands for the normal peak-to-average power density ratio after xenon and samarium have built-up, and R_s stands for the ratio shortly after start-up before the fission poisons have built-up.

4.6.2.4 Fast Neutron Energy Distributions. The fast neutron energy distributions in the central plane at several radii in the reflector were computed with the aid of the CORNPONE Code (4.7). This 20 group, P-1 code was used to compute the energy distributions for a single fuel element in simple cylindrical geometry and then the final distribution was obtained by the appropriate superposition of the contribution from each fuel element. The distribution of the fast flux in the central plane as a function of lethargy ($u \equiv \ln \frac{2}{E}$, E in Mev) is shown for three different values of the radius in Figure 4.18.

4.6.3 CRITICAL MASS AND FUEL ELEMENT LOADING. The critical mass has been computed for the equilibrium core and for the start-up core. As discussed in the Section 4.6.4, the equilibrium core is more heavily loaded on the outside than in the center. The radial distribution of U-235 was determined for the normal fuel cycle described in Section 4.6.4, and the core divided into four concentric annular regions. The U-235 in each region was then considered to be uniformly distributed throughout the region and the corresponding homogeneous parameters were calculated. This calculation gave a value of 2630 gm of U-235 for the critical mass at T = 115°F including the effects of xenon and samarium.

In the actual core, the fuel is concentrated in a much smaller volume than the core regions used in the above calculation. Because of the higher concentration of fuel, there is a local flux depression which causes the fuel to be worth less than in the homogenized case. The amount that the actual fuel loading must be increased over that used in the homogenized calculations to give the same multiplication factor for the real core is called the "disadvantage factor." The unit cell calculations described in the previous section were used to determine this factor. The "disadvantage factor" for the equilibrium core was found to be 1.27 so the calculated homogenized mass of 2630 gm must be multiplied by 1.27 to give

3340 gm for the actual true critical mass.

The start-up core refers to the initial start-up of the reactor when all fuel elements are fresh. As discussed in the next section, the start-up core consists of 19 uniformly loaded elements. Similar calculations to those for the equilibrium core lead to a critical mass of 2790 gm including a disadvantage factor of 1.23.

The mass of U-235 in the operating core must, of course, be greater than the just critical mass to give sufficient reactivity to operate. The reactivity requirements are discussed in Section 4.6.5, so here we will discuss only the choice of fuel element loading to give the necessary reactivity.

At the start of an equilibrium core cycle, but after fission poisons have built-up, the reactivity available should be 5.4%. The calculations show that a mass of 4090 gm of U-235, including a disadvantage factor of 1.32, is required to give this reactivity. Assuming the cycling procedure described in the next section, the fresh fuel element loading is found to be 205 gm per element. This is based on the thermal flux distribution and the burn-up that an element experiences in each position that it occupies as it moves through the reactor during its life cycle.

It is anticipated that the experimental load in the start-up core will be much lighter than in the equilibrium core. If .8% reactivity is allowed for experiments, the available reactivity after fission poisons have built-up should be 3.7% for the start-up core. This is achieved by a loading of 19, 170 gm elements giving a total mass of 3230 gm of U-235 including a disadvantage factor of 1.26.

The critical masses and fuel loadings are summarized in Table 4.6-5. The column "available reactivity" refers to the reactivity near the start of a cycle, but after fission products have built-up. It is that reactivity available for experiments and reactor operation. It is not necessarily all available for insertion since some is tied up in semi-permanent experiments.

Table 4.6-5 U-235 Masses and Fuel Element Loading

Core	Number of Elements	Critical Mass	Operating Mass	Available Reactivity	Fresh Fuel Element Loading
Equilibrium	24	3340	4090	5.4	205
Start-up	19	2790	3230	3.7	170

4.6.4 <u>FUEL CYCLING AND BURNUP</u>. At 10 Mw, the U-235 fission rate will be 10.9 gm per day and the additional U-235 loss by neutron capture will be 1.9 gm per day to give a total U-235 burnout rate of 12.8 gm per day. For the NBSR 24-element core, this corresponds to an average burnout rate of 0.533 gm per element per day. As shown below, each element will spend 148 operational days in the reactor burning up 79 gm. The initial fuel element loading for the equilibrium core is 205 gm so the percent burnup is 39%. This percent burnup is similar to that achieved at CP-5 for similarly loaded elements.

The normal fuel cycle will add new fuel elements in the outside ring of the core and remove elements from the center ring. The normal loading of 24 elements are located in 3 rings. The inner and middle rings contain 6 elements each and the outer ring contains 12. The following is typical of the type of fuel cycle anticipated. At the end of every three weeks of 10 Mw operation, the three most spent fuel elements will be removed from the center ring and replaced by three from the middle ring. Three from the outside ring will be moved to the middle ring and fresh elements will be placed in the outside ring. In this way the most heavily loaded elements are placed in the lowest flux, minimizing hot spot problems.

In order to approximate the average fuel loading of the equilibrium cycle, it will be necessary to use more lightly loaded elements for the initial 10 Mw loading since all the elements will be fresh. Advantage will be taken of the versatility of fuel element

configurations allowed by the double plenum design to initially load fewer than 24 elements. Loading with fewer elements concentrates more power in the center 6 elements, but these are fed by a separate plenum so the water velocity can be increased through them without requiring increased capacity to increase total flow in all parts of the core. The burnup of U-235 will then be compensated for by the addition of fresh elements at each cycle. Calculations of burnup using the flux shape calculated from the previous core loading configuration were made for each successive loading until the equilibrium cycle was reached. These indicated that an initial loading of nineteen 170 gm elements with the addition of two 170 gm elements on each of two successive cycles would lead to the equilibrium loading with only minor perturbations (< 1% reactivity).

4.6.5 REACTIVITY REQUIREMENTS FOR OPERATION. The core loading at the start of a normal cycle is designed to have sufficient reactivity to provide for temperature changes, xenon and long lived fission product buildup, effects of experiments, and fuel burnup during the cycle. The core loading chosen, on the basis of calculations, has an effective multiplication factor, k_{eff}, of 1.107 at the start of a cycle at room temperature with all rods withdrawn. When temperature equilibrium is reached, k_{eff} has decreased to 1.097, and after xenon and samarium have reached equilibrium it has decreased to 1.057 neglecting U-235 burnup during this time. This remaining reactivity covers the requirements for semi-permanent experiments, small transient experimental changes, and U-235 burnup. The multiplication factor at the end of the three week cycle is 1.032 of which no more than 2% is available, the remainder being tied up in semi-permanent experiments.

Although the reactivities allocated to various purposes do not add strictly, it is useful to summarize them as shown in Table 4.6-6.

Table 4.6-6 Summary of Reactivity Requirements

Source	Reactivity Effect
Temperature change from shutdown to full power	0.8%
Xenon	2.9
Samarium and long lived fission products	0.6
U-235 burnup	2.3
Reactor control at end of cycle	0.5
Experiments	2.6
Total	9.7

The reactivity allowance for experiments includes 0.2% for the pneumatic rabbit system, not more than 1.3% for removable experiments, and the remainder for semi-permanent experiments that can be removed only during reactor shutdown. No single rabbit tube sample will be worth more than 0.1% and the total worth of samples being inserted into all 4 rabbit tubes simultaneously will not exceed 1/4%. The 1.3% allowed for removable experiments is to allow for the possibility of the slow insertion or removal of low reactivity worth experiments in the vicinity of the core during reactor operation. No individual experiment would be worth more than could be controlled by the regulating rod ($\sim$ 1/2%). The remainder of the experiments cannot be removed during operation and so the reactivity required to compensate for their presence is never available for insertion during operation. Thus, once xenon and samarium have built up, the maximum reactivity available is only that for transient experiments, U-235 burnup and reactor control at the end of the cycle totaling no more than 4.3%. As is shown in Section 4.6.6.1, this is less than the worth of one shim safety arm.

Table 4.6-6 is based on the anticipated normal fuel cycle. During the initial loadings approaching the normal cycle, the reactivities may vary slightly from those tabulated, but it is not anticipated that the total reactivity will exceed that given in the table by more than 1%.

4.6.6 CONTROL ROD REACTIVITY WORTHS

4.6.6.1 Shim Safety Arms. The NBSR shim safety arms are of the semaphore type similar to those used in CP-5. Because of the awkward geometry it is difficult to calculate their worth with great confidence. The worth of the CP-5 shim safety arms have of course been measured and have proved completely adequate for the control of CP-5 covering a wide range of core loadings over a period of more than 10 years. The similarity of the NBSR system to CP-5, therefore, assures us that the NBSR system will prove to be quite satisfactory. It is desirable, however, to make the best estimate practicable of the NBSR shim safety arm worths and to this end a method of calculation was developed and checked against the known CP-5 data.

The method consisted of dividing the core and top reflector into relatively small regions and noting what fraction of each shim safety arm was in each region as a function of shim arm angular position. Since the shim arms are black to thermal neutrons, their effect was estimated by dividing their projected area in each region by the volume of the region to find an equivalent Σ_{poison}. This was then put into the two group, two dimensional, multi-region code, Equipoise-3A, used for the other core calculations. It should be noted here that, as in the other calculations, all the regions had cylindrical symmetry so the code could be used in the r-z mode to cover three dimensions.

This calculational technique was checked by applying it to the CP-5 reactor. The calculated results are compared to the measured values (4.8) in Figure 4.19. This shows the change in reactivity, $\Delta\rho$, as a function of angular position, θ, of the bank of 4 shim arms. The agreement is seen to be quite good. The $\Delta\rho$ for complete insertion from full out (θ from 50° to 0°) is .304 for the measured value and .330 for the calculated. Thus, the calculational method appears to have an accuracy of about 10%.

The results of this same method applied to the NBSR are shown in Figure 4.20. In the NBSR case, the angular position of the shim arm, θ, is measured from the full out position instead of the full in position as for CP-5. $\Delta\rho$ is the change in the reactivity when the shim arms are moved from θ to the full in position here defined as $\theta = 40°$. From this data it is seen that the maximum reactivity worth of the shim arm bank is about 46%. A similar calculation for a single shim arm based on the assumption that all shim arms have equal worth, showed it to be worth about 14-1/2% and 2 arms worth about 25%.

The maximum withdrawal rate for one shim arm or the whole bank of 4 is the same, and is 4×10^{-2} degrees per second. From Figure 4.20 the maximum change in reactivity with change in angle is 1.5% per degree for the bank of 4, so the maximum rate of reactivity insertion by shim arm withdrawal is .06% per second.

4.6.6.2 Regulating Rod Worth. The curve of the regulating rod worth as a function of position has been calculated taking into consideration both its poison and void worth. The Equipoise code was used to determine the void coefficients and poison worths along the path of the regulating rod. The result is shown in Figure 4.21, where the reactivity change is shown in percent as a function of rod position. Z = 0 corresponds to the fully inserted regulating rod position and $\Delta\rho$ is determined relative to Z = 0. The rod stroke is the full 29" shown in the figure, controlling a .47% reactivity. Although the worth curve is quite linear, it may be desirable to increase the linearity by operating over a reduced range. Any necessary adjustments in the regulating rod geometry will be made when the actual worth curve has been experimentally determined.

4.6.7 BEAM TUBE REACTIVITY WORTHS.

The beam tube reactivity worths were calculated by modifying the parameters in the code to account for the filling of the beam tubes with D_2O. The two through tubes were done as a group. The large cryogenic beam tube was not calculated directly, but its worth was estimated by comparing its statistical weight based on the product of the thermal flux and its adjoint with the statistical weight of the radial tubes. The results are summarized in Table 4.6-7.

Table 4.6-7 Beam Tube Worths

			% Reactivity
9	Radial Beam Tubes	(Total)	.76
2	Through Beam Tubes	(Total	.29
1	Cryogenic Beam Tube		1.80
		Total	2.85

Although the full void of the cryogenic beam tube is worth 1.8% $\Delta\rho$, it should be noted that the reactor will not be run with the tube fully voided. It will always be at least partially filled with a bismuth shield or a water tank which will reduce the void that could be flooded to a volume no greater than three beam tubes. So the maximum reactivity change that could result from flooding this hole during operation would not exceed .3%.

4.6.8 TEMPERATURE AND VOID COEFFICIENTS. The temperature and void coefficients were determined by varying the parameters that were used in the Equipoise code in an appropriate way.

The results showed the temperature coefficient to be -0.018% $\Delta\rho$/°F.

The void coefficient was determined for several parts of the core. The lower, upper, and gap region of the core were calculated separately. In addition, the void coefficient in the regulating rod region of the upper and lower core was also calculated. Table 4.6-8 summarizes the results.

Table 4.6-8 Core Void Coefficients

Upper Core	0.038% $\Delta\rho$/liter
Gap	0.020
Lower Core	0.043
Regulating Rod Region	
Upper Core	0.045
Lower Core	0.050

4.6.9 TOP REFLECTOR WORTH. The worth of the top reflector is not independent of the shim arm position. Since the shim arms are such a strong poison, they markedly reduce the worth of any reflector above them. Calculations were performed for the unperturbed reflector worth as a function of thickness and for the change in reactivity resulting from dumping with shim arms in the reflector.

The unperturbed reflector worth was calculated by moving the top black boundary and computing the multiplication factor accordingly. The results are shown in Figure 4.22. Zero thickness refers to a water level equal to the elevation of the top of the fueled region. The change in reactivity, $\Delta\rho$, is measured from this level. The worth is not increasing very fast above a thickness of 16" and appears to be approaching a value not much greater than about the 16% calculated for a 2' thickness. On the other hand, the reactivity is changing more rapidly near the core at a rate of about 1-1/2% $\Delta\rho$/in. over the first 4".

The main purpose in investigating the top reflector worth is, of course, to estimate the change in reactivity to be expected when the emergency dump is activated. Therefore, the presence of the shim arms has been included and the change in ρ calculated for the emergency dump as a function of shim arm position in the region of normal operation. The emergency dump is to a level 1-1/6" above the fueled region of the core. Figure 4.23 shows the absolute value of $\Delta\rho$ ($\Delta\rho$ is, of course, negative) plotted against shim arm position. With the shim arms fully withdrawn ($\theta = 0°$), ρ decreases by about 13% when the top reflector

is dumped. As expected, this value decreases as the shim arms are lowered towards the core. The normal shim arm position during operation will be in the region between 0° and 15°, so the value of the emergency dump will vary between -13% and -6%.

4.6.10 H_2O CONTAMINATION OF D_2O. The effect of adding H_2O to the NBSR at critical was calculated. Figure 4.24 shows the resulting effective multiplication factor, k, plotted against the volume percent of H_2O. The curve shows that the addition of any amount of H_2O represents a negative reactivity insertion. The slope of k versus % H_2O is steepest for small additions of H_2O.

4.6.11 XENON TRANSIENTS. The growth of Xe and Sm poison after shutdown of the reactor has been studied because of the importance of this question for reactor restart. If it is assumed that the reactor has been operating at 10 MW for a sufficient time to saturate the Xe and Sm poison, and it is further assumed that the reactor is shutdown instantaneously*, the growth of Xe and Sm poison can be calculated. Figure 4.25 shows the behavior in terms of percent poison as a function of time in hours after shutdown. It can be seen that the maximum total poison is reached 11 hours after shutdown and is equivalent to 24.4% poison**. It can also be seen that the total poison that must be overcome before startup at the end of a 48 hour shutdown is 5.5%, assuming none of the poison is removed as elements are changed. Approximately one-eighth of the core will be unloaded each shutdown so that the poison will be reduced by approximately this factor. To examine the restart problem more fully the question of the amount of time available before growth of xenon prevents restart was studied. The growth of poison for times immediately after shutdown is shown plotted in Figure 4.26 as percent poison versus time in minutes. It has been calculated that approximately 51 minutes is available for restart if an initial excess reactivity of 2.8% required for burnup is utilizable. If inadvertent shutdown should occur halfway through a 19 day running cycle, only 24 minutes are available. It is because of this rapid growth of poison that a maximum effort has been made to assure reliable power operation. Little or no excess reactivity will be carried to extend the time during which xenon poison can be overridden in the event of failure to maintain full power.

Analog computer studies were made of Xe transients to examine reactor shutdown characteristics. Both linear ramp and step reductions of reactor power were examined to study the growth of Xe poison with decreasing power level. Slow linear shutdown can achieve a minimum Xe poison buildup but shutdown rates sufficiently slow to minimize Xe poison buildup were so slow as to be uninteresting as a practical shutdown regime. The problem created by rapid shutdown to reduced power level was studied and results of computer runs show that for power reduction greater than a factor of two Xe poison buildup will be sufficient to prevent continued operation without increased reactivity margins.

4.6.12 INHERENT TRANSIENT RESPONSE OF THE REACTOR. An extensive series of analog computer studies have been made of the kinetic behavior of the NBSR under various normal and emergency conditions. The asymptotic behavior of a transient can be calculated from the so-called in-hour equation which gives the final stable reactor period for a transient following a step reactivity insertion. A series of curves was run for reactor power or flux versus time following a number of step insertions of reactivity. In the initial stages of this work the in-hour equation was used to test results of the analog computer. In using the in-hour equation it is necessary to establish the delayed neutron group structure and the neutron lifetime of the reactor. The following discussion is related to the nature of these parameters for the NBSR.

4.6.12.1 Neutron Lifetime. Calculations of the neutron lifetime of the reactor have been carried out as a prerequisite to reactor kinetic studies. A D_2O moderated reactor is expected to have a relatively long lifetime and this is a distinct advantage from the point

*The long period delayed neutrons, of course, prevent this, but for times of the order of hours after shutdown this is a good approximation.

** The reactivity effect of poison depends on absorption in the core other than fuel. The reactivity equivalent of 24.4% poison is approximately 20% under the core assumptions which makes $\Sigma_o = 0.2\Sigma_u$.

of view of reactor control and safety. A long lifetime not only simplifies reactor control but leads to greater reactor periods for a specific reactivity insertion. The short period transients are harder to achieve and for a given reactivity insertion which is greater than the delayed neutron fraction more time is available for heat transfer from the fuel elements and for bubble formation within the moderator.

An equation for calculating the mean thermal lifetime of prompt neutrons has been derived by T. H. Pigford, et. al. (4.9). The derivation assumes two group diffusion and first order perturbation theory to hold. In addition, the core is assumed homogeneous and spherical with a spherical reflector. All absorption is assumed to take place at thermal energies. The thermal lifetime

$$\ell = \frac{\int_V \frac{\phi^{\dagger}\phi_s}{v_s} dV}{\int_V k_{\infty} \sum_s \phi_f^{\dagger}\phi_s dV}$$

where the volume integrals are taken over the entire reactor volume. The subscript s and f refer to the thermal and fast groups, respectively. The superscript refers to adjoint flux properties. The numerator of the equation for ℓ gives essentially the statistically weighted density of neutrons in the reactor since the density of neutrons in the fast group can be considered negligible. The denominator gives the neutron production rate weighted statistically. The regeneration time or lifetime is, therefore, the ratio of the two. The calculation of fluxes and adjoint fluxes was carried out using the so-called WANDA code (4.10). All material parameters were derived in the same manner as for the reactor core flux calculations previously discussed in Section 4.6.2.

Calculations were made for two radii after evaluating an equivalent spherical core to replace the actual cylindrical reactor. The effect of increased absorption in the reflector was also studied. The results of the calculations were given in NBSR 7 (2.3). The values of lifetime range from approximately 850 μ sec to 1030 μ sec, being longest for the largest reactor with unpoisoned reflector.

The lifetime was recalculated with the latest core parameters using the code Equipoise 3A (4.3). A slightly lower value of 700 μ sec was calculated which is to be expected with the somewhat heavier loading. It is to be expected that the lifetime will be a function of core loading and thus variable over narrow limits. It is clear that the correct order of magnitude of lifetime has been calculated. These numbers can be compared with the numbers reported for the MITR (4.11) and HFBR (4.2) which are 610 μ sec and 672 μ sec respectively.

The lifetime of a D_2O moderated and cooled reactor is seen to be an order of magnitude longer than that for the light water cooled reactors.

4.6.12.2 In-Hour Equation. The actual final stable period for any reactivity insertion can be calculated from the so-called "in hour" equation once the lifetime has been calculated. The delayed neutron fraction, β, must be introduced in a way which considers the numerous delayed neutron groups. The D_2O moderated reactor differs from other reactors in that the number of effective neutron groups is augmented by those delayed neutrons caused by photoneutron production in the D_2O. This photo production is caused by gamma rays from the decay of fission products. For this report the neutron groups reported by Johns and Sargent (4.12) have been used. Their analysis of the fission neutrons into delayed groups differs to some degree from that of Keepin (4.13) but for the purpose of this report the Johns and Sargent data have been chosen. These data have been derived from D_2O moderated reactor studies and include delayed neutron groups due to photo-disintegration of D_2O.

The "in-hour" equation for reactivity versus stable period is the following:

$$\rho = \frac{\ell}{k_{eff}T} + \sum_{i=1}^{n} \frac{\beta_i}{1 + \lambda_i T}$$

where ρ is the reactivity, ℓ the neutron lifetime, k_{eff} the effective multiplication factor, T the stable reactor period, β_i the fraction of delayed neutrons in the i^{th} group and λ_i the decay constant for the i^{th} neutron group.

A plot of the results of applying the delayed neutron group data of Table 4.6-9 is shown in Figure 4.27. Results are plotted for three separate choices of lifetime ℓ. Interpolation permits the use of these curves for prediction of stable period for a range of different lifetime.

The lifetime and delayed neutron group data permitted the writing of an analog computer program to simulate the complete transient behavior of the reactor under various reactivity insertions. For the computer program the delayed neutron group data were rearranged to yield only 5 equivalent groups for convenience in applying the kinetic equation to the computer. A typical result of this aspect of the computer studies is shown in Figure 4.28. These results display the complete divergent behavior of the reactor including the so-called "prompt jump" while the "in-hour" equation gives only the resulting final stable period. The differential equations describing the kinetic behavior, set of typical analog computer machine equations, and a block diagram of the machine equations is shown is appendix IV of NBSR 7.

Table 4.6-9 Delayed Neutron Groups (4.12)

Mean Life sec	Fraction β_i %
80.2	0.0260
31.8	0.1700
6.50	0.2176
2.19	0.2465
0.62	0.0867
0.07	0.0255
3.6	0.02073
59.0	0.00650
204	0.00223
660	0.00107
2340	0.000659
8640	0.000743
22680	0.000102
273600	0.0000327

Total Fraction $= \sum_{i=1}^{14} = 0.8044$

4.6.13 VOID SHUTDOWN AND DUMP SYSTEM. As was pointed out in Section 4.6.9 the worth of the heavy water top reflector; i.e. down to the dump line is 6 to 12 percent depending on the shim arm positions. Since it requires approximately 40 seconds for the top reflector to drain there may exist occasions wherein it is desirable to introduce more rapid auxiliary shutdown as a backup to the dump. This is provided by the void shutdown. By manual switch the operator may initiate the flow of helium gas through nozzles located in the core. Such action will also initiate the dump and command the scram.

The helium system is designed to flow gas at the rate of ~ 180 CFM into the core moderator. Since the minimum accumulation time of core bubbles is ~ 2.5 sec, approximately 7-1/2 ft^3

of void may be sustained at this flow rate. Such a void volume is worth ~ 8 $\Delta k/k$, a value comparable to the worth of the dump. This should be sufficient to hold the reactor down until the top reflector is drained.

At 10 psig, the helium flow rates cited will prevail. They are sufficient to exhaust the contents of one helium bottle in about one minute. A bank of six is provided. If additional flow is required for some continuing or other emergency condition, the operator may also release instrument air into the system as described more fully in Section 9.5.4.2 and 9.5.4.3.

4.7 THERMAL AND HYDRAULIC CHARACTERISTICS

4.7.1 CORE THERMAL ANALYSIS AND DESIGN. The criteria for the thermal analysis was verification of the safe and efficient operation of the NBSR at a power level of 10 Mw thermal, using MTR plate type fuel elements. The first analysis was performed under the restriction that the surface temperature at any part of the fuel plates would not be allowed to exceed the saturation temperature at that point. The minimum allowable coolant flow through the fuel elements were thus calculated. The thermal conditions under the anticipated flow conditions, based on the design capabilities of the NBSR primary system, were then calculated to show that an additional wide margin of safety exists. Table 4.7-1 gives a summary list of the NBSR thermal characteristics.

Since the flow to the fuel elements is channeled through two plena, there are two distinct thermal conditions for any given core. The fuel elements in the inner ring, fed by the inner plenum, have a much higher power density than the remainder of the core at startup. However, the double plenum arrangement allows these "hot" elements to be given the higher flow rates necessary during the startup phase without supplying a large surplus flow to the remainder of the core. The necessity of supplying excess pumping capacity, which would be inefficiently used during normal operation, is thereby avoided. Additionally, a great deal of flexibility for future core arrangements is provided.

The thermal and hydraulic requirements of the NBSR treated in this section are calculated for normal reactor operation after the Samarium and Xenon buildup. The most severe operating condition (inner plenum at startup) is then examined for the pre-Samarium, Xenon buildup period to show a safe operating condition.

4.7.2 COOLANT FLOW CHARACTERISTICS

4.7.2.1 Reactor Vessel Flow Rates. The minimum acceptable fuel element flow rate was established by applying the same hot spot-temperature = saturation-temperature limitation mentioned in Section 4.7.1. The actual fuel element flow rates were determined by comparing the hydraulic characteristics of the NBSR fuel elements and primary system to the capability of the primary pumps. Bypass occurs in paths parallel to the core. Some flow is bypassed around the inlet nozzle of the fuel elements; the remainder is sent through the experimental and regulating rod positions. The bypass flow rate around the lower nozzle of the fuel element was determined by measurements made on a hydraulic test stand with a dummy fuel assembly in place. Bypass flow rates for the remainder of the flow paths are easily calculated since they are simple orifices. A summary of the vessel flow rates and flow areas is given in Table 4.7-2. The fuel element flow listed in the minimum column represent the results of the "hot channel" calculations, which were made under the saturation temperature limitation. Those flows shown in the actual column represent the design flows of the NBSR primary cooling system.

4.7.2.2 Velocity Distribution. The velocity distribution of the NBSR fuel element assembly has been measured on a hydraulic test stand. Measurements were made of the relative velocity in each channel of the fuel element assembly as well as the velocity distribution across the fuel core section of a typical channel. There was no evidence of systematic velocity changes due to the location of the flow channels with respect to the end adapters. Examination of the data concerning the distribution across the channel showed the classic flow pattern found in rectangular and near rectangular flow channels. The numerical results and uncertainties are summarized in Section 4.7.4.

Table 4.7-1 NBSR Thermal Characteristics

Reactor power, MW thermal	10	
Average heat flux Btu/hr-ft^2	1.024×10^5	
Peak heat flux (hot spot) Btu/hr-ft^2		
Inner Plenum - Startup		
Nominal	2.48×10^5	
With maximum heat flux factor	3.64×10^5	
Outer Plenum - Startup		
Nominal	1.99×10^5	
With maximum heat flux factor	3.06×10^5	
Inner Plenum - Equilibrium		
Nominal	1.64×10^5	
With maximum heat flux factor	2.52×10^5	
Outer Plenum - Equilibrium		
Nominal	1.74×10^5	
With maximum heat flux factor	2.68×10^5	
Overall peak to average power density ratio		
Startup		
Inner plenum	1.53	
Outer plenum	1.69	
Whole core	1.82	
Equilibrium		
Inner plenum	1.62	
Outer plenum	1.70	
Whole core	1.71	
Hot channel factors:		
Bulk water factor	1.59	
Heat flux factor	1.35	1.54*
Heat transfer coefficient factor	1.48	
Maximum fuel plate - water interface temperature, °F		
Inner plenum		
Startup	225	
Equilibrium	217	
Outer plenum		
Startup	221	
Equilibrium	217	
Saturation temperature at hot spot, °F		
Startup	244	
Equilibrium	235.5	
Minimum burnout ratio		
Startup		
Inner plenum	2.24	
Outer plenum	2.48	
Equilibrium		
Inner plenum	2.80	
Outer plenum	2.73	

* At Dogbone

Table 4.7-2 (cont'd)

	Minimum Acceptable	Actual
Flow areas, square inches:		
Nominal channel	.307	
One element	5.53	
24 elements	132.6	
Equivalent diameter, inches:		
Nominal channel	.224	
Fuel element flow rates, gpm		
Startup		
Inner plenum		
One element	249	290
Total (6 elements)	1494	1745
Outer plenum		
One element	209	244
Total (13 elements)	2717	3165
Total startup flow through fuel elements	4211	4910
Normal Operation		
Inner plenum		
One element	171	194
Total (6 elements)	1026	1165
Outer plenum		
One element	183	208
Total (18 elements)	3294	3745
Total normal flow through fuel elements	4320	4910
Bypass flow rates, gpm		
Regulating rod cooling (1) at 8 gpm		8
2" experimental (4) at 10 gpm		40
4" experimental (6) at 8 gpm		48
Around fuel elements and other core components		94
Total primary system design flow		5100

4.7.3 HOT SPOT TEMPERATURE CALCULATIONS. The various temperature conditions of interest are calculated using the power for the various fuel element positions developed in Section 4.6.2, the physical and hydraulic characteristics of the coolant flow channels, and the thermodynamic properties of the coolant. The relationships used in the following section were then employed to calculate a minimum velocity for each plenum for both startup and equilibrium cores. This minimum velocity would be that necessary to prevent boiling at any point. Each factor affecting the hot spot temperature calculation was then calculated on the basis of both this minimum velocity and the actual velocity which would exist during normal operation.

The method for calculating each factor of interest is demonstrated in the following subsections. The results of these calculations for the area of interest are summarized in Table 4.7-3 in Subsection 4.7.3.7.

4.7.3.1 Average Coolant Temperature Rise. The average temperature rise of the cooling water in the NBSR is calculated as follows:

$$\overline{\Delta T}_C^b = K_1 P/Q\rho C_p$$

where

K_1 = a conversion factor 4.26×10^5 Btu-gal/Mw-ft^3-min

P = total power 10 Mw thermal

Q = total volumetric flow rate through reactor, 5100 gpm

ρ = average water density, 68.5 lb/ft^3

C_p = specific heat, 1.0 Btu/lb-°F

The resulting average cooling water temperature rise is 12.2°F for the normal reactor operating conditions.

4.7.3.2 Coolant Temperature Rise in Hot Channel. In this analysis the peak power in the hot channels is used and mixing of the coolant across the width of the channel is assumed negligible. The cooling water temperature rise in the hot channel is calculated as follows:

$$\Delta T^b_{HC} = K_2 F_b \frac{\overline{P}_C F_{HC}}{VA\rho C_p}$$

where

K_2 = a conversion factor .9786 Btu/Kw-sec

F_b = hot channel factor for bulk rise, 1.49 (Section 4.7.4)

$\overline{P}_C$ = average power per channel for the hottest element in a ring, Kw

F_{HC} = ratio of power in the hot channel to the average power per channel, 1.10 (Section 4.6.2)

V = nominal coolant velocity

A = flow area of a single channel .0384 ft^2

ρ = average water density 68.5 lb/ft^3

C_p = specific heat 1.0 Btu/lb-°F

4.7.3.3 Heat Flux. The maximum heat flux at the hot spot is:

$$\varphi_{HS} = K_3 F_q \frac{R_n \overline{P}_C}{A_{ht}} \text{ Btu/hr-ft}^2$$

where

K_3 = a conversion factor 3.42×10^3 Btu/Kw-hr

F_q = hot channel heat flux factor, 1.54 (Section 4.7.4)

R_n = ratio of peak to average power in an element (Section 4.6.2

$\overline{P}_C$ = average power per channel for the hottest element in a ring

A_{ht} = total heat transfer area of a single fuel plate, .745 ft^2

The maximum heat flux occurs just below the midplane at the top of the fueled section of the hot fuel plate. It is at this location that both peak power density and local thickening of the split fuel cores ("dogboning") will occur. All the calculations include the effect of "dogboning."

4.7.3.4 Heat Transfer Coefficient. The heat transfer coefficient is calculated by the modified Colburn correlation (4.14).

$$\frac{hDe}{k_f} = 0.023 \left\{\frac{D_e V\rho}{\mu}\right\}^{0.8}_f \left\{\frac{C_p\mu}{k}\right\}^{0.3}_f$$

where

h = heat transfer coefficient, Btu/hr-ft^2-°F

D_e = Equivalent diameter, .0186 ft

k = thermal conductivity of coolant, Btu/hr-ft-°F

V = coolant velocity, ft/hr

ρ = coolant density, lb/ft^3

μ = coolant viscosity, lb/ft-hr

C_ρ = coolant specific heat, Btu/lb-°F

The subscript "f" means that the properties are evaluated at the film temperature. The film temperature is considered to be the average of the heat transfer surface temperature and the bulk water temperature.

4.7.3.5 Film Temperature Rise. The temperature difference between the fuel plate surface and the bulk water in the hot channel at the hot spot is given by:

$$\Delta T^f_{HS} = \frac{F_h\ \varphi_{HS}(Z)}{h}\ °F$$

where

F_h = hot channel heat transfer coefficient factor, 1.48 (Section 4.7.4)

φ_{HS} = maximum heat flux at hot spot

h = heat transfer coefficient, calculated from the modified Colburn correlation

4.7.3.6 Fuel Plate Surface Temperature in Hot Channel. The fuel plate surface temperature in the hot channel at the hot spot is calculated as follows:

$$T^s_{HS} = T^b_o + \Delta T^b_{HC} + \Delta T^f_{HS} + \Delta T_{tm}$$

where

T^b_o = nominal bulk inlet coolant temperature, 100°F

ΔT_{tm} = allowance for temperature measurement error, 5°F

ΔT^b_{HC} = bulk water temperature rise

ΔT^f_{HS} = film temperature rise

4.7.3.7 Summary. The results are summarized in Table 4.7-3. The data are presented for both plena for both the startup and the equilibrium cores. The minimum safe condition is presented for comparison to the nominal operating condition. With the exception of the last section of the table, all data are based on the flux distribution calculated after Xe and Sm have approached equilibrium. For comparison, the last section of the table shows the situation for normal operating conditions obtaining from the calculated transient flux distribution at startup before Xe and Sm have built up. It can be seen from this table that the subcooling (i.e., the difference between the saturation temperature and the actual fuel plate surface temperature) provides a comfortable margin of safety for all nominal operating conditions.

Table 4.7-3 Summary of Hot Spot Temperature Conditions

	Startup		Equilibrium	
	Inner Plenum	Outer Plenum	Inner Plenum	Outer Plenum
Peak Heat Flux (Hot Channel) Btu/hr-ft^2	3.64×10^5	3.06×10^5	2.52×10^5	2.68×10^5
Peak Heat Flux (Normal Channel) Btu/hr-ft^2	2.48×10^5	1.99×10^5	1.638×10^5	1.74×10^5
Minimum Acceptable Conditions				
Velocity (ft/sec)	14.4	12.1	9.9	10.6
Flow rate (gpm)	249	209	167	178
Saturation Temperature °F (Hot Spot)	239	236	230	232.7
Fuel Plate Surface Temperature at Hot Spot, °F	239	236	230	232.7
Hot Channel Bulk Temperature Rise °F	24.8	24.6	25	25.4
Hot Channel Film Temperature Drop °F	121.6	118.7	113.2	115
Subcooling °F	0	0	0	0
Hot Channel Coolant Outlet Temperature °F (with 5F° added for Temp. Meas. error)	129.8	129.6	130	130.4
Nominal Operating Conditions				
Velocity (ft/sec)	16.8	14.2	11.6	12.4
Flow Rate (gpm)	290	244	194	208
Saturation Temperature °F (Hot Spot)	243	238	234.5	235.5
Fuel Plate Surface Temperature at Hot Spot, °F	224	221	217	217
Hot Channel Bulk Temperature Rise °F	21.2	21	21.3	21.7
Hot Channel Film Temperature Drop °F	108	105.5	100.9	101
Subcooling °F	19	17	17.5	16.5
Hot Channel Coolant Outlet Temperature °F (with 5F° added for Temp. Meas. error)	126.2	126	126.3	126.7
Pre Sm and Xe Build Up				
Hot Channel Film Temperature Drop °F	113.4	114	107.5	109
Fuel Plate Surface Temperature at Hot Spot °F	229.0	229.5	223.2	224.9
Subcooling °F	14	8.5	11.4	10.6

4.7.4 HOT CHANNEL FACTORS. Uncertainties in the operating conditions of the reactor and in the heat transfer correlation coefficients must be allowed for in determining the proper water flow through the elements. Many of the factors discussed below are statistically independent so it is very unlikely that they will all be present at any one time; but, since during the life of the reactor there will be many different combinations of conditions, it must be assumed that at some time all adverse effects to create a hot channel will be present together. These uncertainties are expressed below as hot channel factors.

4.7.4.1 Reactor Power Measurement. It is hoped to determine power removed from the reactor by the primary coolant with an accuracy between 1 and 2%. The power deposited directly in the fuel, however, has additional uncertainties which include deviations in

power between calibration times of the neutron flux level detectors, small variations in the automatic controller, and error in the determination of the fraction of total power actually deposited in the fuel. These effects combine to give an uncertainty in the reactor power measurement as applied to the fuel elements of 5%. The bulk temperature rise and heat flux terms are increased by 1.05 to account for these uncertainties.

4.7.4.2 Power Density Variations. The power density distribution is calculated by the multi-region, two-group code. It includes the effects of flux squeezing by the shim arms and the variation throughout the core of U-235 concentration due to fuel burnup and cycling. Since the calculated peak-to-average power densities for the whole core varied by less than 15% from each other over the whole range of core loadings and shim arm positions studied, it is reasonable to expect that the calculations reflect the true power density distribution to better than 15%. Therefore, a factor of 1.15 will be conservatively applied to the bulk and heat flux terms.

4.7.4.3 Variations in U-235 Concentration in Fuel Core. The fuel element specifications state that the U-235 content of any fuel core shall not vary from the specified amount by more than 3%. So the contribution to the bulk temperature rise in a channel is 1.03.

The local heat flux is directly proportional to fuel core thickness. These variations are limited by the specifications to no more than a 5% increase in thickness in the main body of the fuel core. It is more difficult, however, to control the variations at the ends of the fuel core. Here the specifications allow an increase in thickness of 15%. Since the area at the inner end of the fuel cores are located in the highest flux, the local variations, or "dogboning," are important in the hot spot analysis. Therefore, although it is anticipated that the manufacturer will do somewhat better than required by the specifications, we will conservatively use a factor of 1.20 in the dogbone region. A factor of 1.05 applies to the remainder of the fuel core region.

Another factor affecting the magnitude of the heat flux is the fuel core area (heat transfer area). The dimensional tolerances on the length and width of the fuel core allow a variation of ± 7% in this area. This factor, combined with the local thickness variation, contribute 1.28 and 1.12 to the heat flux factor for the dogbone regions and the central regions respectively.

4.7.4.4 Channel Width Variations. Assuming that the pressure drop across any channel is independent of the channel width for small variations thereof, and neglecting any variations in entrance and exit losses, the flow through a channel varies as the 5/3 (4.15) power of the channel width. The channel width of the NBSR elements may vary by no more than ± 6% to be acceptable under the specifications. Thus, the bulk temperature rise factor must include a contribution of $(1.06)^{5/3} = 1.10$ for this effect.

In addition to the decrease in velocity caused by a narrow channel, a local increase in channel width will cause a further decrease in velocity. The average velocity which is decreased by $(1.06)^{2/3}$ in the narrow channel may be further decreased locally by the ratio of the maximum allowed width to the minimum. It can be shown that this effect causes the heat transfer coefficient to be reduced by a factor of $(d\ max/d\ min)\ (d\ nom/d\ min)^{1/3}$. Thus a factor of $(1.06/.94)(1/.94)^{1/3} = 1.15$ must be included in the heat transfer coefficient factor for the effect of local variation in channel widths.

4.7.4.5 Velocity Variation with Channel Position. The lower end fitting of the fuel element is designed to maintain a uniform flow in each channel independent of location. To do this it tapers gradually from the 2" entrance diameter to the approximately 3" x 3" cross section of the fuel plates. The success of this design was checked by flow measurement in a dummy fuel element. The results showed the flow to be independent of channel position within the accuracy of the measurements.

The relative flow was determined from pressure measurements. The relative accuracy of these measurements was estimated to be about 5%. Since the velocity varies as the square root of pressure, the uncertainty in velocity was about 2-1/2% or conservatively a factor of 1.03 should be used in the bulk hot channel factor and $(1.03)^{0.8} = 1.024$ in the heat transfer coefficient factor.

4.7.4.6 Velocity Variation Across Channel. The velocity distribution across a single channel was measured and showed the minimum velocity to be 6% less than the average. If no mixing is assumed across the width of a channel this introduces a factor of 1.06 in the bulk term and $(1.06)^{0.8} = 1.05$ in the heat transfer coefficient.

4.7.4.7 Heat Transfer Coefficient Deviation. The Phillips Reactor Safeguard Committee recommends the use of the modified Colburn correlation with a maximum negative deviation of 20% (4.14). The deviation results in increasing the heat transfer coefficient term by 1.20.

4.7.4.8 Summary of Hot Channel Factors. The hot channel factors discussed above are summarized in Table 4.7-4.

Table 4.7-4 Hot Channel Factors

Bulk Factors, F_b		
Reactor power measurement	1.05	
Power density distribution uncertainty	1.15	
U-235 variation in fuel core	1.03	
Channel widths variation	1.10	
Velocity distribution uncertainties	1.03	
Velocity variation across channel	1.06	
Total F_b	1.49	
Heat Flux Factors, F_q		
Reactor power measurement	1.05	
Power density error	1.15	
Fuel core thickness variations	1.12	(1.28)*
Total, F_q	1.35	(1.54)*
*Dogbone region		
Heat Transfer Coefficient Factors, F_h		
Channel widths variation	1.15	
Velocity distribution uncertainties	1.024	
Velocity variation across channels	1.05	
Correlation equation	1.20	
Total, F_h	1.48	

4.7.5 CLADDING AND FUEL TEMPERATURE. The formation of a layer of boehmite on the fuel plate cladding will increase the temperature of the cladding and fuel. The rate of formation of the oxide layer was measured by Griess, et. al. (4.16) under conditions similar to those in the NBSR. The oxide layer thickness was found to be given by the following relationship:

$$X = 443\,(t)^{.778}\exp(-8290/T), \text{ mils}$$

where

X = oxide layer thickness in mils

t = exposure time, hours

T = oxide-water interface temperature °R

The maximum interface temperature in the NBSR core, 224°F, occurs in the inner plenum of the startup core. If, for sake of simplicity, it is assumed that this temperature and the element power persist throughout the entire 6 month life of an element, the maximum oxide layer thickness achieved would be .0017", and the corresponding oxide temperature drop and maximum material temperatures would be 40.8°F and 265.8°F respectively.

The application of the same technique to the hottest element in the equilibrium core yields an oxide layer thickness of .0012" and an oxide temperature drop and maximum surface temperature of 24.4°F and 241.4°F, respectively.

These temperatures clearly present no hazard to the integrity of the cladding or structure.

4.7.6 STEADY STATE BURNOUT ANALYSIS. The burnout prediction correlation developed by Mirshak, Durant and Towall (4.17) is based on data which fits the precise operating conditions of the NBSR. Since this correlation is well known and widely used, its use in the steady state burnout analysis of the NBSR is indicated.

Using this correlation the heat flux at burnout is given by:

$$\varphi_{bo} = 266{,}000\ (1 + 0.0365V)(1 + 0.00914T_s)(1 + 0.0131P)$$

where

φ_{bo} = heat flux at burnout, Btu/hr-ft^2

V = coolant velocity, ft/sec

T_s = subcooling °C

P = pressure, psia

For the purposes of this analysis, the coolant velocities and, by interrelation, the amount of subcooling and the pressure at the hot spot were reduced by the channel dimensional tolerance and velocity distribution measurement error hot channel factors, 1.10 and 1.03. A factor of safety of 1.2 was then applied to the result to allow for uncertainties in the correlation. The operating and burnout heat fluxes and their ratios are listed in Table 4.7-5.

Table 4.7-5 Burnout Heat Fluxes and Burnout Ratios

	Peak Heat Flux Btu/hr-ft^2	Burnout Heat Flux Btu/hr-ft^2	Burnout Ratio
Normal Operation			
Startup Core			
Inner Plenum	3.64×10^5	9.8×10^5	2.24
Outer Plenum	3.06×10^5	9.37×10^5	2.48
Equilibrium Core			
Inner Plenum	2.52×10^5	8.47×10^5	2.80
Outer Plenum	2.68×10^5	$9{,}04 \times 10^5$	2.73
Pre Xm, Sm Buildup			
Startup Core			
Inner Plenum	3.82×10^5	9.81×10^5	2.41
Outer Plenum	3.28×10^5	9.37×10^5	2.38
Equilibrium Core			
Inner Plenum	2.68×10^5	8.47×10^5	2.64
Outer Plenum	2.90×10^5	9.03×10^5	2.60

4.7.7 GAMMA RAY HEATING. Extensive calculations were made of the gamma ray intensity and spectrum in the NBSR. The fission spectrum of U-235 was determined from the work of Gamble or prompt gamma rays as described by Maienschein in 8/670 of the Second Geneva Conference Report (4.18), and from the data tabulated in the Nuclear Engineering Handbook (4.19). The contribution of the decay fission products to the operating gamma spectrum was obtained from analysis of the data in P.1670 and the results of Perkins and King (4.20). Capture gamma rays from U-235 based on the work of Groshev (4.21) and from aluminum based on the work of Kinsey as summarized in the Reviews of Modern Physics (4.22) was included to complete a gamma ray source spectrum.

The gamma ray calculations followed the procedures developed in NYO-3075 (4.23). Using the source spectrum described above, the gamma ray intensities and spectra at several radii were calculated by summing the contributions from the fuel elements and other gamma ray sources. The gamma ray spectrum in the center of the reactor is shown in Figure 4.29. I_o represents the direct contribution from unscattered gamma rays, I_s represents the contribution from scattered gamma rays, and I represents the sum. At points further from the core, the scattered radiation becomes more predominate and the spectrum softens.

In addition to the center of the core, R = 0, the gamma ray spectra and intensities were calculated at the surface of the core, R = 55 cm, and at the tank wall, R = 107 cm. The calculated intensity at the wall includes a contribution of about 25% from capture gamma ray produced in the aluminum tank wall.

The spectrum may be integrated to give the total gamma ray flux. The core gamma rays at the core surface have an intensity of 16.5 w/cm^2 when the reactor is operating at 10 megawatts, and they decrease approximately exponentially to a value of .65 w/cm^2 at the reactor vessel wall. The gamma flux at the core surface in the region of the unfueled gap is about one half the unperturbed value of 16.5 w/cm^2. The reactor vessel wall makes an additional contribution in its immediate vicinity of .25 w/cm^2. Table 4.7-6 summarized the gamma flux intensities.

Table 4.7-6 Gamma Flux Intensities

Radius	Intensity
0 cm	45 w/cm^2
55	16.5
107	.9

The energy deposited in a gram of material can be determined from the mass absorption coefficients and the gamma ray spectrum. This was done for aluminum and water for the three radii given in Table 4.7-6. In the case of aluminum, in addition to the gamma ray heating, the energy of the 2.9 Mev beta particle accompanying every neutron capture must be considered. We have assumed all of the beta energy to be deposited in the aluminum. Table 4.7-7 summarizes these results.

Table 4.7-7 Energy Deposition Rates

Material / Radius	Aluminum γ	Aluminum β	Aluminum Total	Water
0 cm	1.2	0.5	1.7 w/g	1.4 w/cm^3
55	.45	.2	.65	.5
107	.025	.025	.05	.03

The heating of various structural members was investigated. Of particular interest are the beam tube tips and the reactor vessel wall. The total energy deposited in a normal beam tube tip is about 200 watts which represents about 1 watt per square centimeter of surface which must be removed by the reflector water. The large cryogenic port has 5000 watts deposited in its tip which corresponds to about 2 w/cm^2 of water cooled surface. In the central plane, the rate of energy deposition in the reactor vessel wall is about .3 watts per square centimeter of inside tank wall. These relatively low heating rates present no problem in cooling since the beam tubes and tank wall are immersed in the reflector which is directly part of the primary cooling loop. A conservative calculation shows the temperature of the cryogenic port tip doesn't exceed 180°F.

The radiation heating of the thermal shield is not discussed in this section, but is included in the shielding section where the thermal shield design is covered.

4.7.8 PRIMARY FLOW COASTDOWN AND SHUTDOWN COOLING. Normally the main primary pumps are started before the reactor is brought to power and not shutdown until sometime after the reactor has been shutdown. In the case of a loss of power, however, the two main pumps will coast-down and one of the two shutdown pumps will come on automatically. The shutdown pumps are supplied with emergency power, including separate DC motors capable of operating directly off of the emergency battery bank, and each pump supplies an 800 gpm flow.

Figure 4.30 shows the primary flow as a function of time after power failure. The upper curve is the flow resulting from the automatic startup of the shutdown pump upon power failure to the main pumps. The lower curve is the coast-down of the two main pumps when no shutdown pump goes on. The curves in Figure 4.30 are based on direct measurements made on the NBSR system. They were not exactly those of the final operating system, but differ only in that H_2O was used instead of D_2O, and the flow resistance of the core was simulated by the use of the flow control valves in the system. It is not expected that these differences will greatly effect the measured curves, although the greater density of D_2O might increase the coast-down time slightly.

Conservatively, the NBSR should scram to shutdown power in less than 0.3 sec after a power failure. During this time the flow will drop from 4400 gpm to about 4000 gpm. Assuming zero specific heat for the fuel element, the surface temperature of the hottest fuel plate will rise only 8°F during this interval. The rapid drop in reactor power following reactor shutdown then allows a sharp drop in surface temperature caused by the relatively high flow during the early seconds of coast-down. After about 30 seconds, the flow as shown by the upper curve in Figure 4.30 approaches its steady state shutdown value of about 800 gpm. This results in a peak film temperature drop of only 13°F 35 seconds after power failure. This is, of course, the maximum film temperature drop that would occur during normal shutdown cooling.

Since the flow in the NBSR is upward, any perturbation in flow caused by local heating should not be such as to impede the forced flow.

4.7.9 FAILURE OF SHUTDOWN PUMPS. Great care has been taken to minimize the possibility of failure of the shutdown pumps. There are two pumps, one of which is normally standby, and each is equipped with both an AC and DC motor. In addition to normal power, the AC motor is supplied by the diesel generator emergency system, and the DC motor can operate directly off of an emergency battery bank.

In the highly unlikely event that all normal and emergency power should fail, the primary flow will follow the lower curve in Figure 4.30. The rather abrupt drop in primary flow shown in the curve at about 30 seconds after power failure results from the closing of the spring loaded check valves in the primary system. Thirty seconds after shutdown the flow is still sufficient so that the maximum film temperature drop is only 46°F. A few seconds later, zero flow through the elements must be assumed and the only cooling available is from natural convection to the water in which the elements are immersed.

The elements are widely spaced and so are cooled by the water around them as well as the water in them. It is very difficult to anticipate the effectiveness of the direct cool-of the plates. Boiling may initially occur and the rate at which water can flow into the interior of the element from above against the escaping vapor cannot be estimated well. This

interior cooling is surely not negligible and may be quite effective. Fortunately, this is not the only cooling available. Each element is surrounded by a large volume of water whose natural convection is not impeded by steam in constricted spaces. If no internal cooling whatsoever is assumed so that all cooling is external, the peak temperature at the fuel plate hot spot will not be more than 340°F above the temperature of the side plate because of thermal conduction to the side plate. This external cooling means that the no flow situation is less severe in the NBSR core than in the close-packed cores such as the ORR. Measurements made at the ORR (4.24) showed that the no-flow condition after shutdown was safe up to at least 15 MW of operating power. In the ORR tests they shut off all pumps and let the low flow signal scram the reactor. For operating power up to 15 Mw boiling was not even observed. Therefore, it is reasonable to expect that complete loss of primary flow after shutdown of the NBSR does not constitute a hazardous condition.

4.7.10 FUEL ELEMENT AFTER HEAT. The fission product decay heat was determined from the curves of Shure (4.25). Each NBSR fuel element will normally remain in the reactor for 148 operating days. The decay heat curves for the NBSR shown in Figure 4.31 are based on this operating life. The figure shows the decay power in watts per watt fission power for total decay power as well as the two components, gamma and beta power.

The full reactor power as a function of time after shutdown is primarily due to the fission product decay. The photoneutron production by the core decay gamma rays on deuterium produces some neutrons which are multiplied by the sub-critical reactor giving a small shutdown fission power in the range of 10 to 100 watts. While this fission source is very convenient for reactor startup instrumentation and control, it is negligible as a heat source compared to the fission product decay heat rate.

References

(4.1) H. Berger, J. H. Talboy, and J. P. Tylka, Nuc. Sci. & Engr., 18, 235 (1964).

(4.2) J. M. Hendrie, "Final Safety Analysis Report on the Brookhaven High Flux Beam Research Reactor," BNL 7661, 1964.

(4.3) T. B. Fowler, M. L. Tobias - ORNL 3199. C. W. Nester, Jr. - Addendum to ORNL 3199.

(4.4) H. H. Hummel, "Selection of Calculation Method and Parameters for Specific Reactor Types." Unpublished.

(4.5) Reactor Physics Constants, ANL 5800, 2nd Ed. (1963).

(4.6) D. J. Behrens, AERE T/R 103 (1958).

(4.7) W. E. Kinney, et. al., ORNL 2081, p 133 (1956).

(4.8) J. P. Tylka, "CP-5 Shim-Safety Rod Calibration Made Sept. 1-3, 1964". Unpublished.

(4.9) Pigford, Troost, Powell, and Benedict, A. I. Ch. E. Journal 2, 219 (1956).

(4.10) O. J. Marlowe, et. al., WAPD-TM 28 (1956).

(4.11) Thompson, Benedict, Cantwell, and Axford, Final Hazard Summary Report, MIT 5007.

(4.12) M. W. Johns, and B. W. Sargent, Can. Jour. Phys., 32, 136 (1954).

(4.13) G. R. Keepin, et. al., Phys. Rev. 107, 1044 (1957).

(4.14) R. J. Nertney, "Calculated Surface Temperatures for Nuclear Systems and Analysis of Their Uncertainties", IDO-16343 (June 1, 1957).

(4.15) B. W. LeTounneau and R. E. Grimble, Nuc. Sci. & Eng., 1, 359 (1956).

(4.16) J. C. Griess, et. al., "Effect of Heat Flux on the Corrosion of Aluminum by Water", Part III, ORNL 3230 (Dec., 1961) and Part IV, ORNL 3541 (Feb., 1964).

(4.17) S. Mirshak, W. S. Durant, and R. H. Towell, "Heat Flux at Burnout", DP-355 (Feb., 1959).

(4.18) F. C. Maienschein, et. at., P/670, Vol. 15, 368, Second Geneva Conference Report (1958).

(4.19) H. Etherington, Editor, Nuclear Engineering Handbook, McGraw-Hill Book Co., Inc. (1958).

(4.20) Perkins and King, Nuc. Sci. & Eng., 3, 726 (1958).

(4.21) L. V. Groshev, et. at., P/2029, Vol. 15, 138, Second Geneva Conference Report (1958).

(4.22) P. M. Endt, and J. F. Kluyrev, RMP 26, 95, (1954).

(4.23) H. Goldstein, and J. E. Wilkins, "Calculations of the Penetration of Gamma Rays", NYO-3075.

(4.24) T. E. Cole, and J. A. Cox, "Design and Operation of the ORR", Vol. 10, 86, Second Geneva Conference Report (1958).

(4.25) K. Shure, "Fission Product Decay Energy", WAPD-BT-24 (1961).

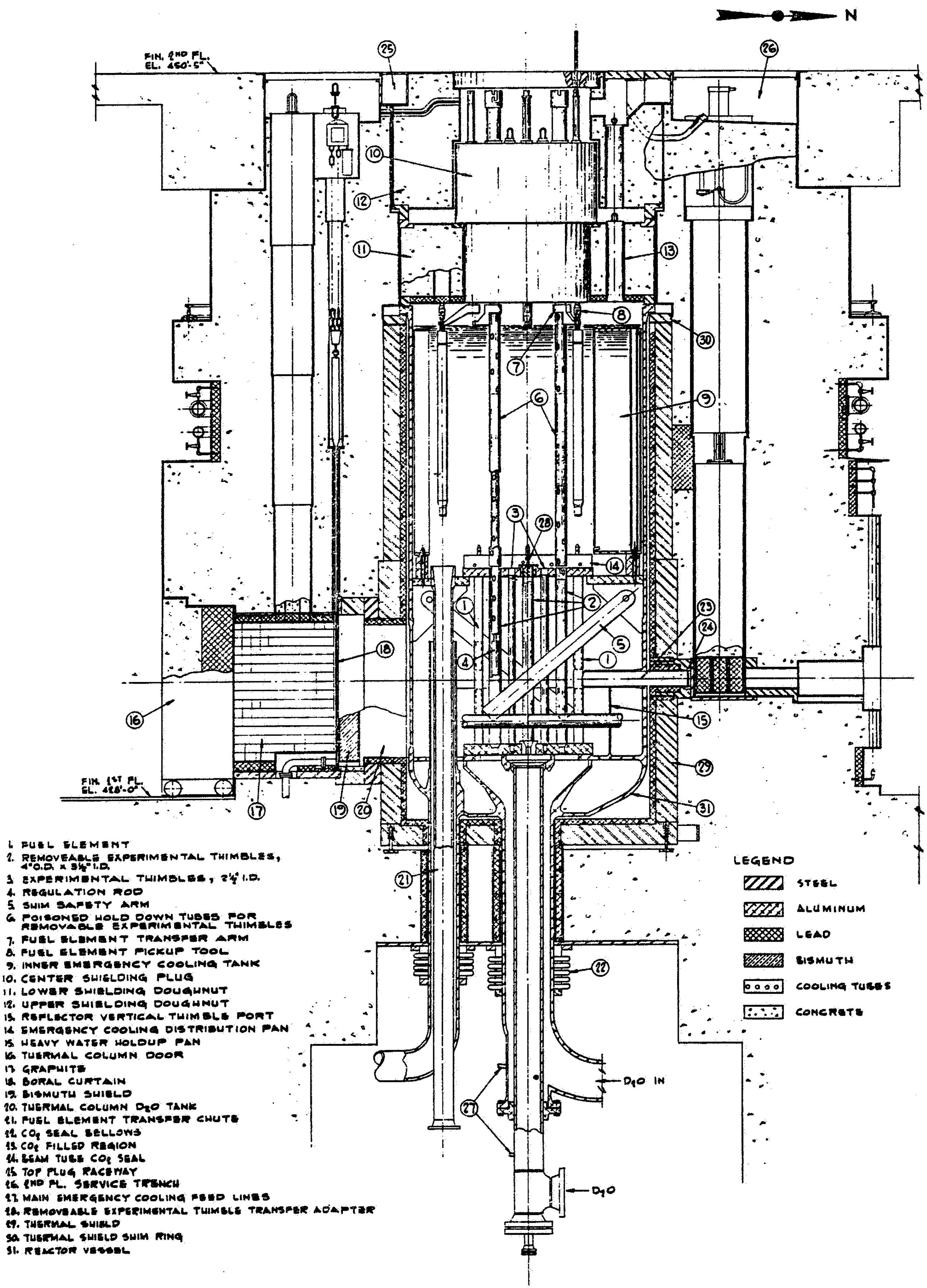


Figure 4.1 Reactor elevation.

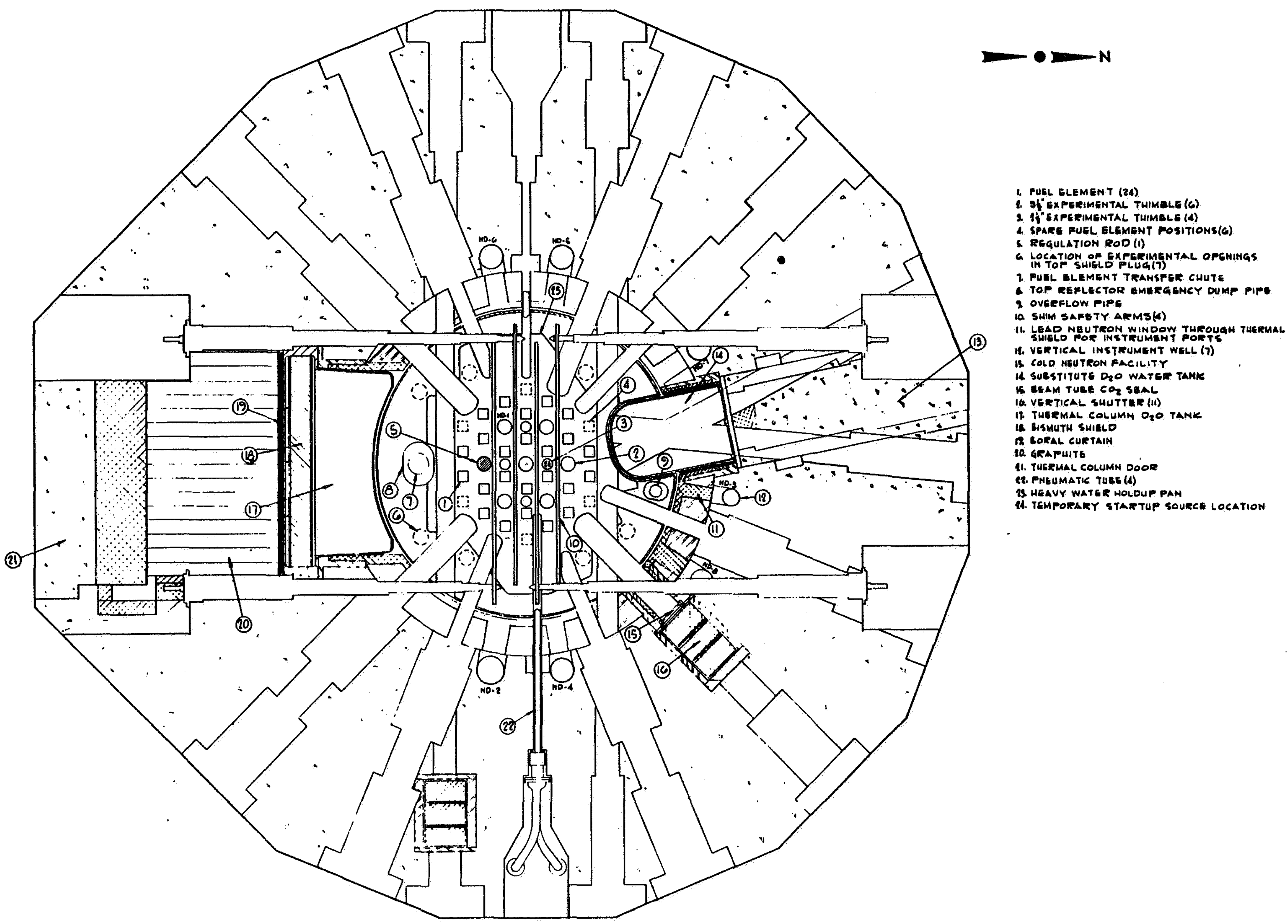


Figure 4.2 Reactor plan view.

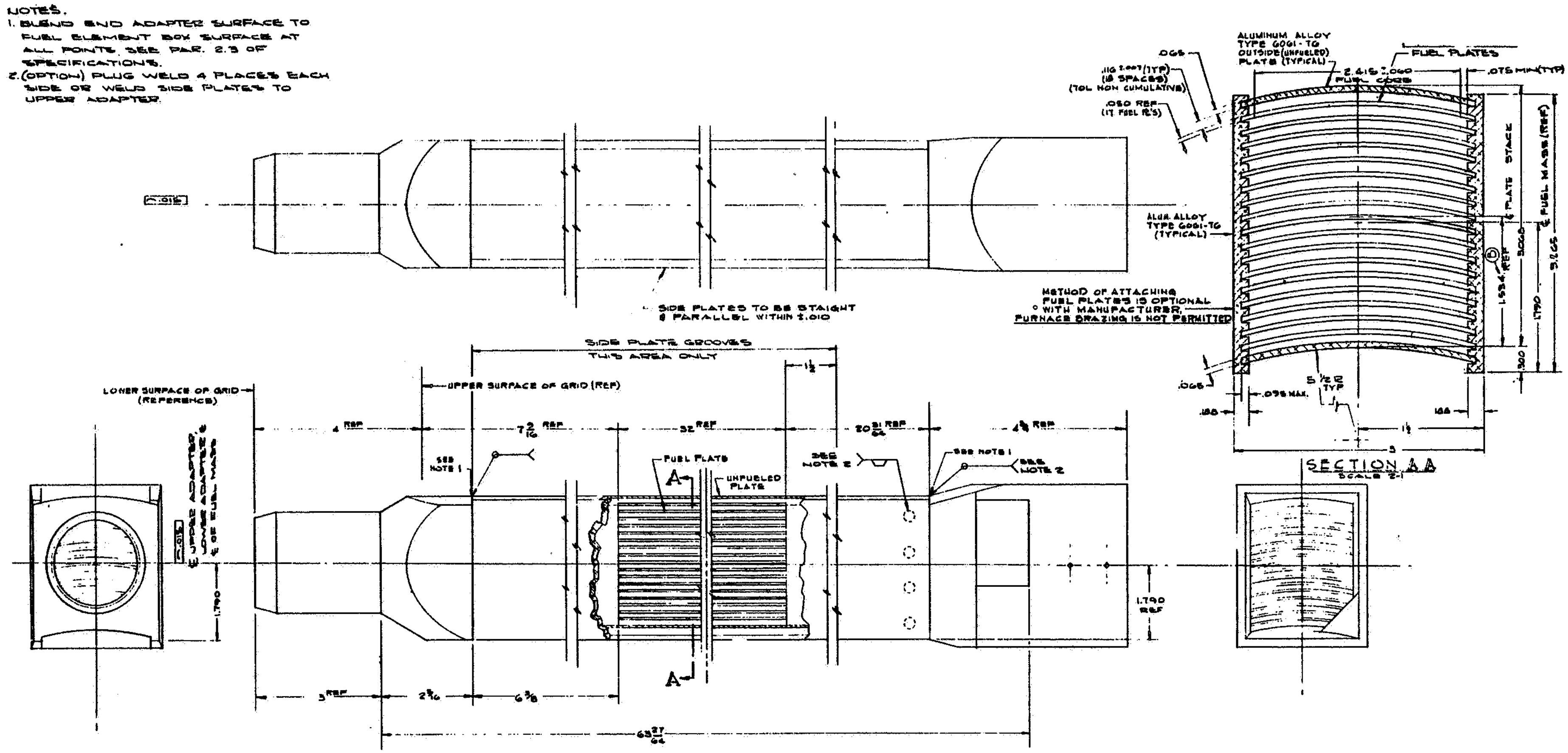


Figure 4.3 Fuel element assembly.

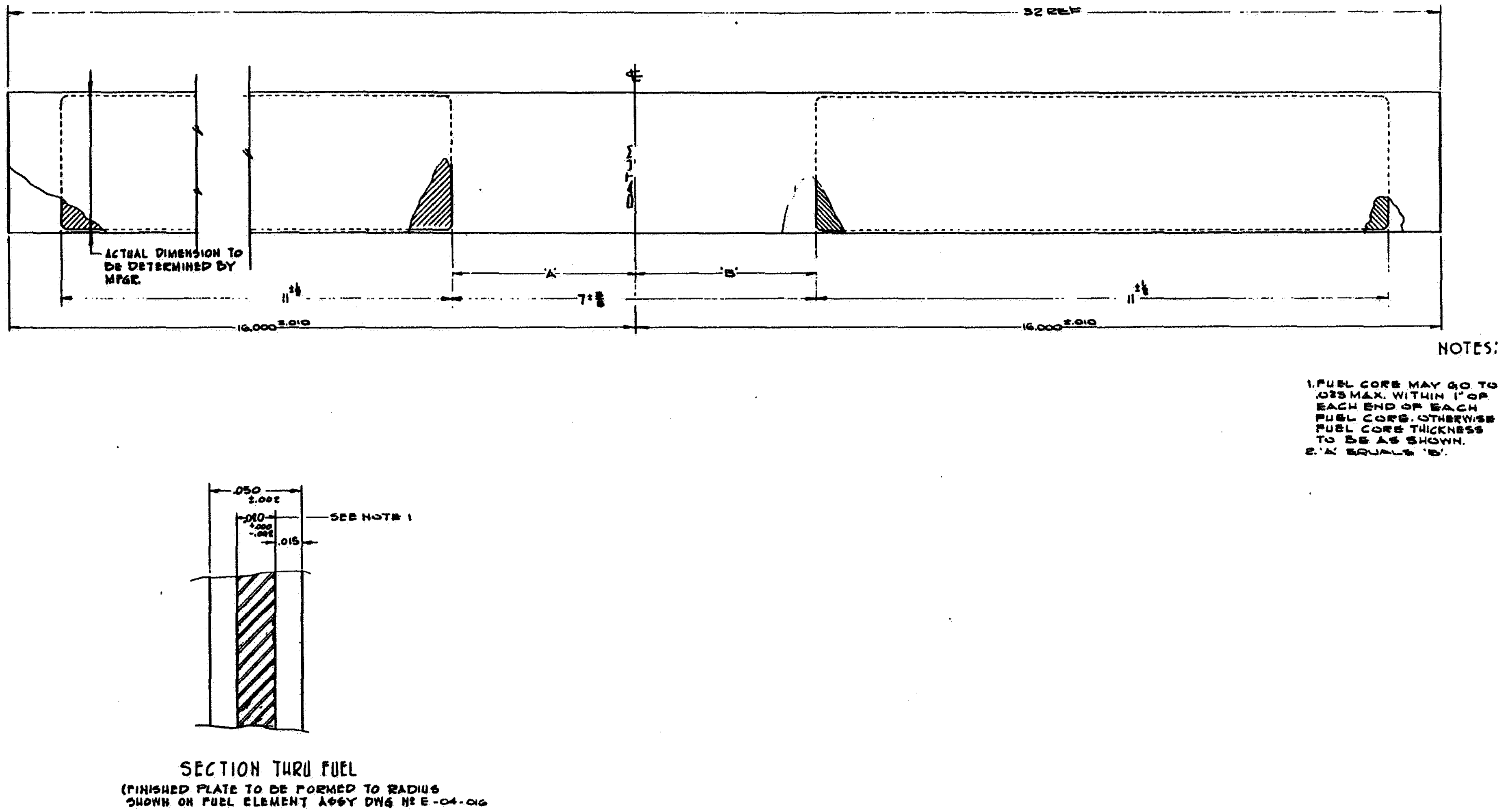


Figure 4.4 Fuel plate details.

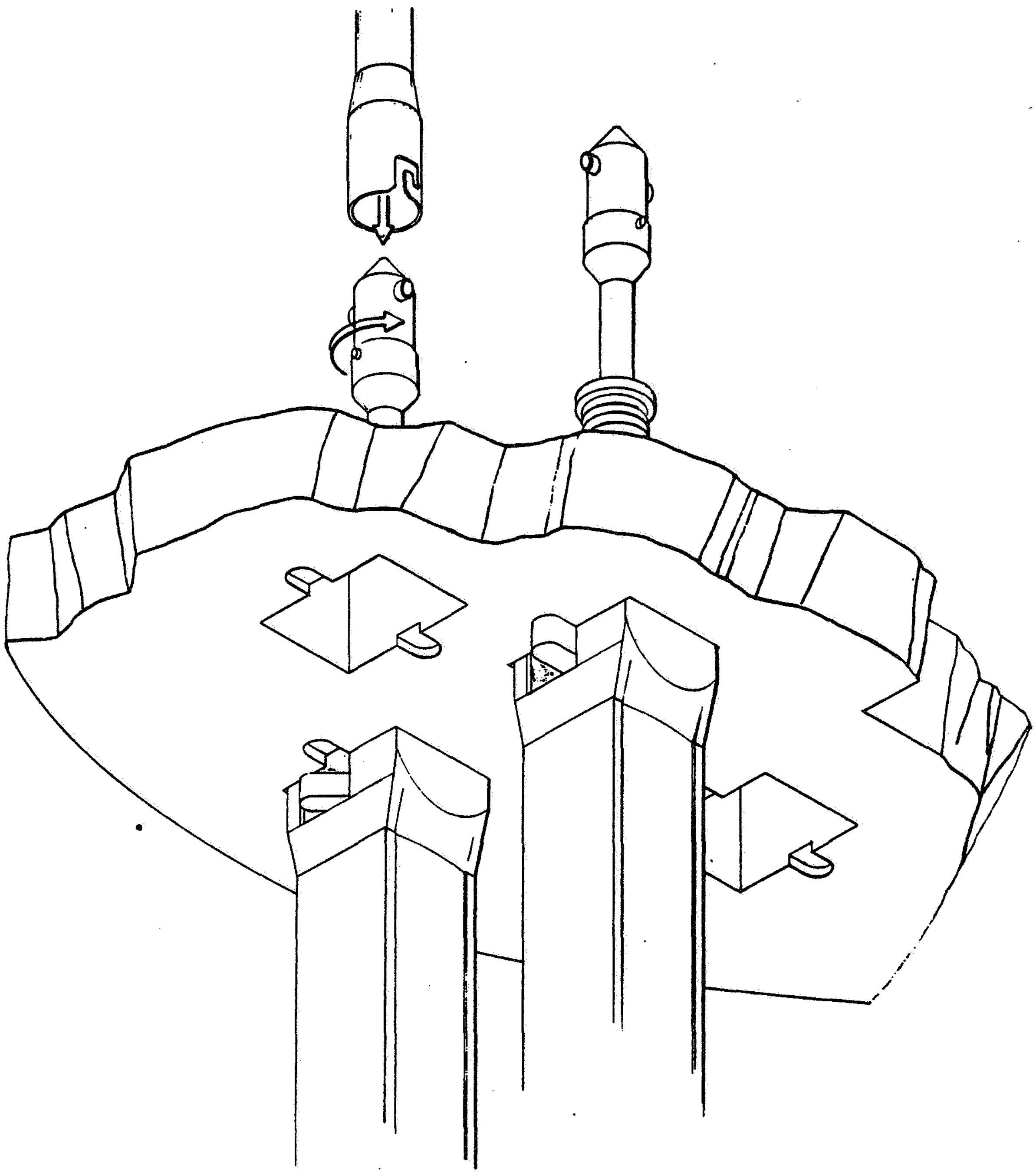


Figure 4.5 Isometric of fuel element locking mechanism.

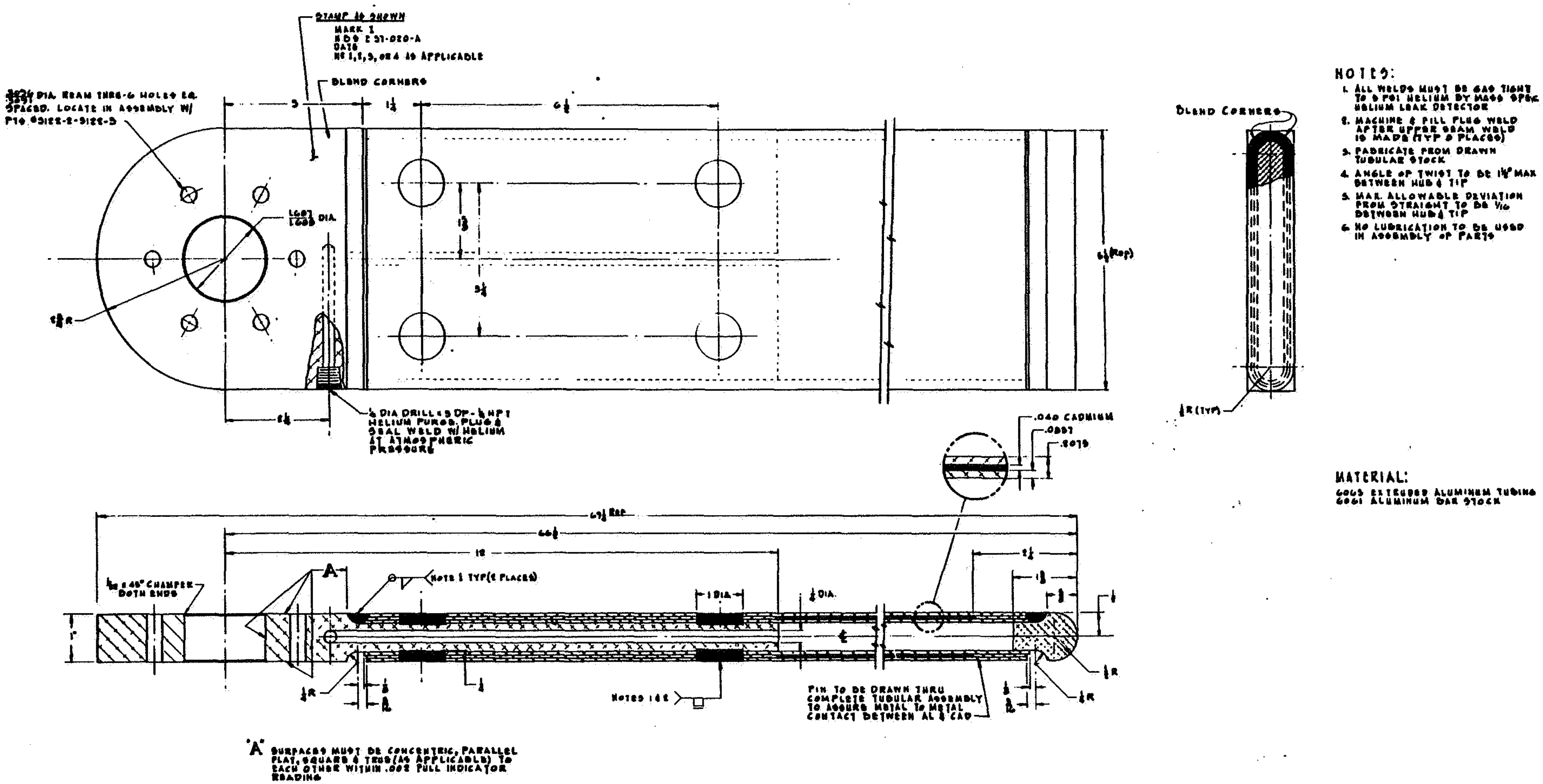


Figure 4.6 Shim safety arm - detailed design.

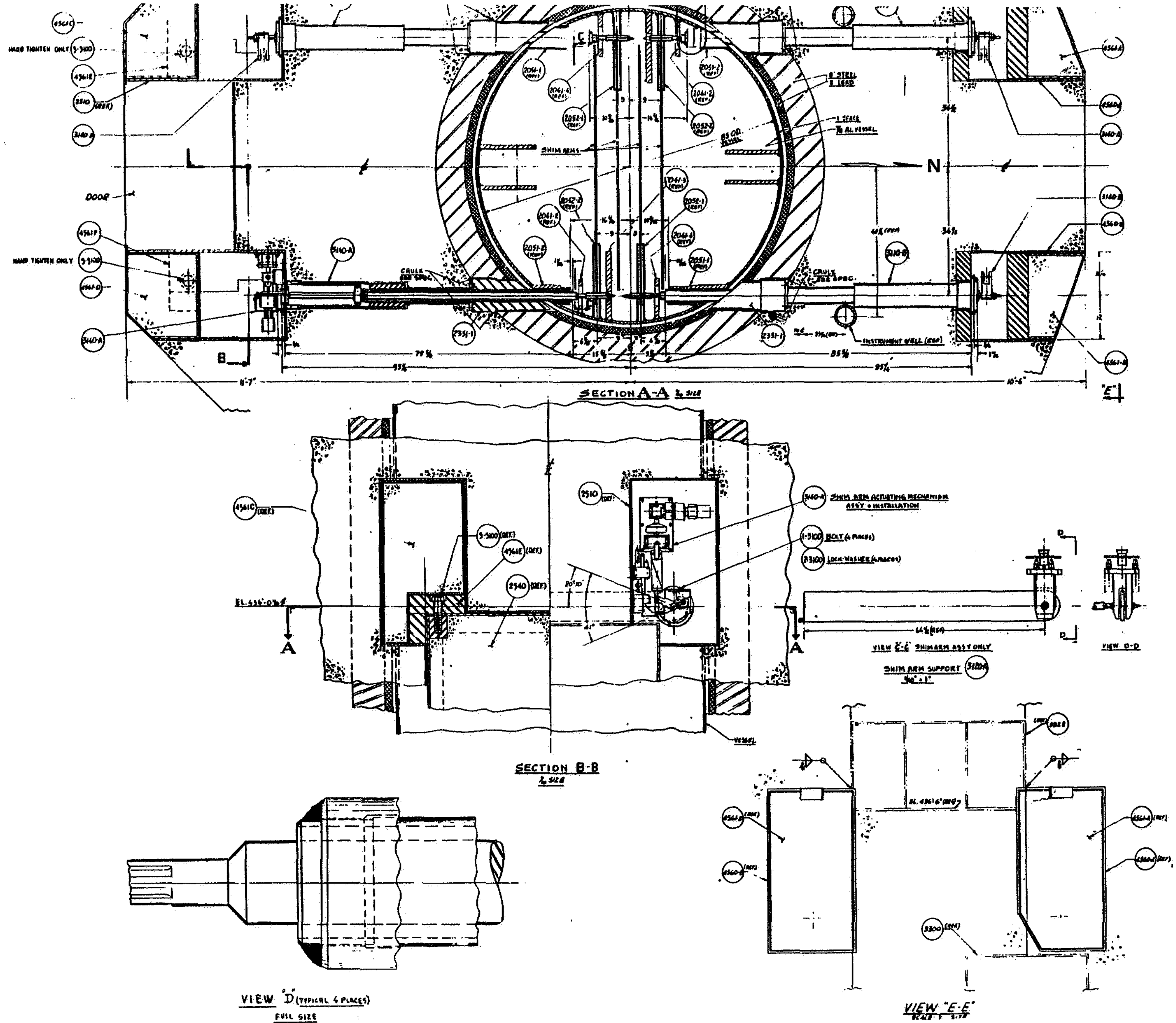


Figure 4.7 Shim safety arm system assembly.

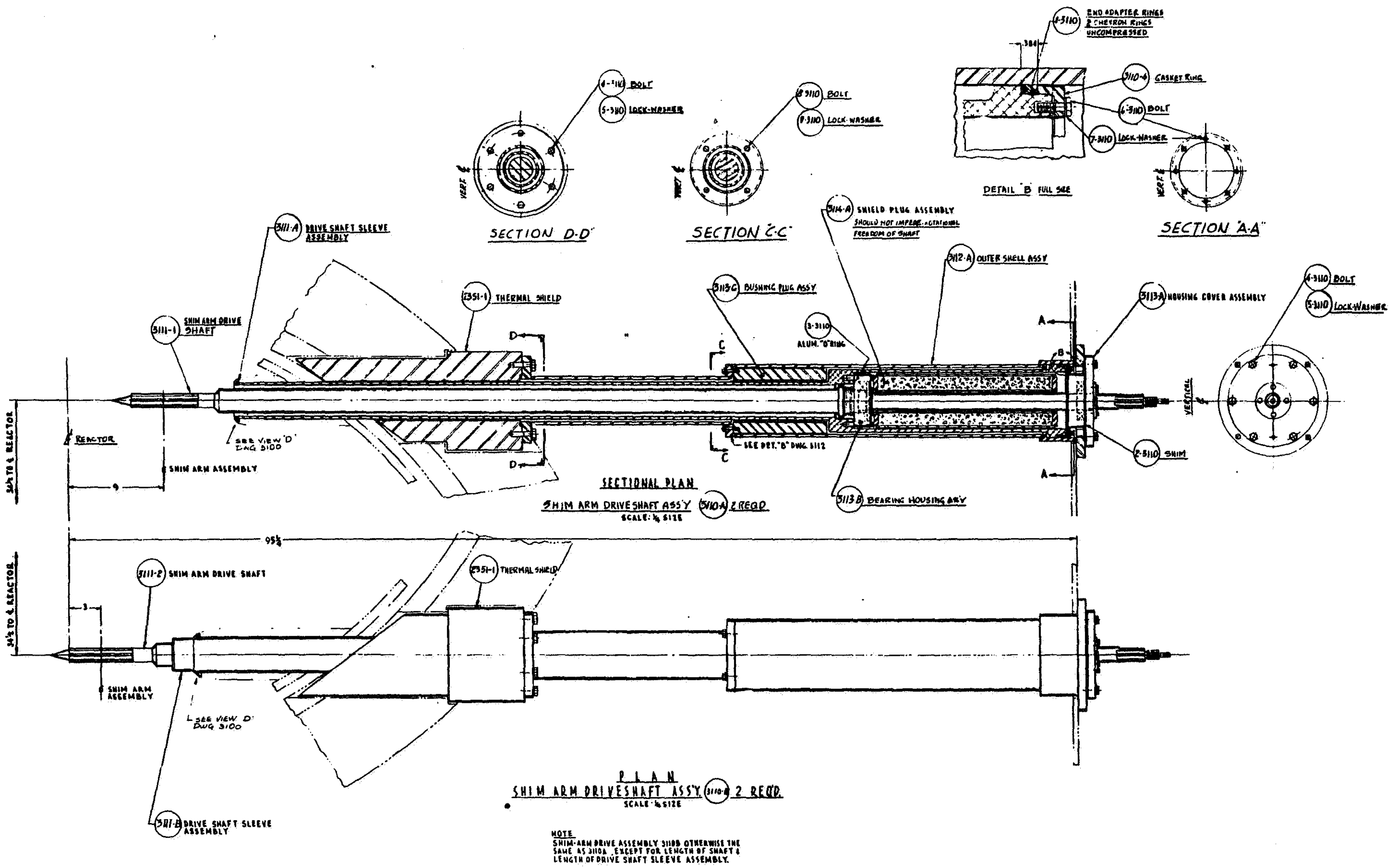


Figure 4.8 Shim safety arm drive shaft assembly.

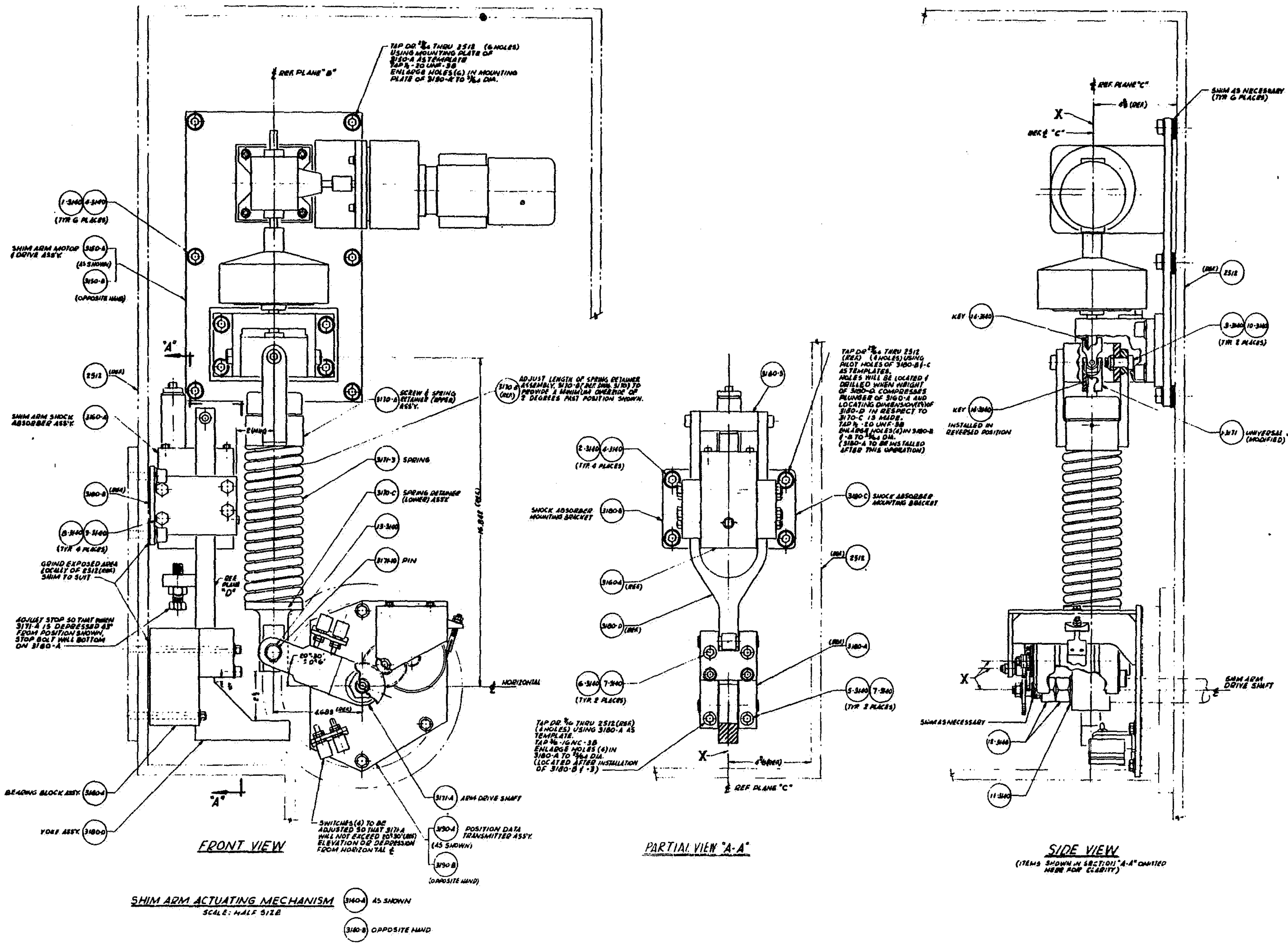


Figure 4.9 Shim safety arm drive mechanism.

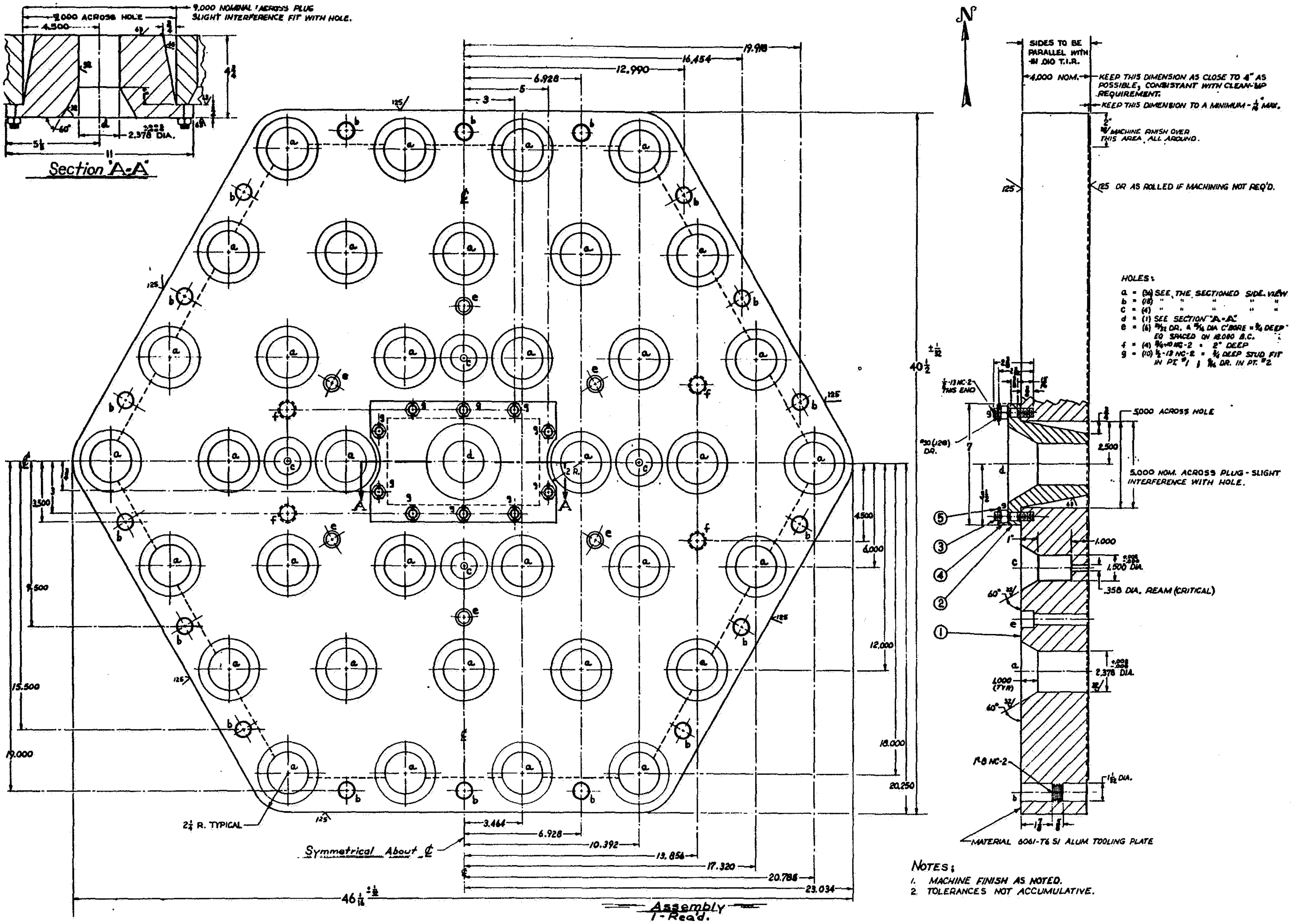


Figure 4.10 Lower grid plate.

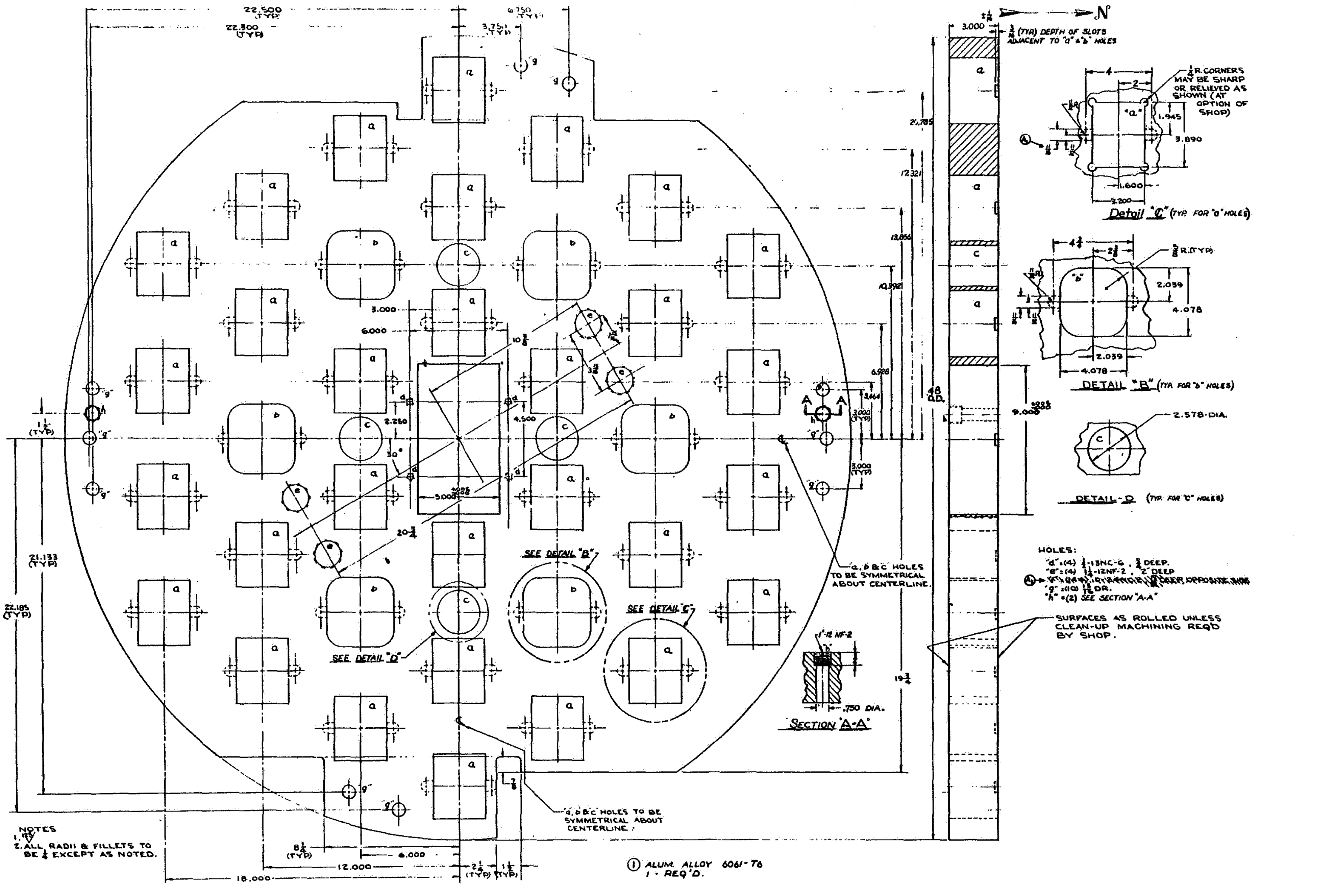


Figure 4.11 Upper grid plate.

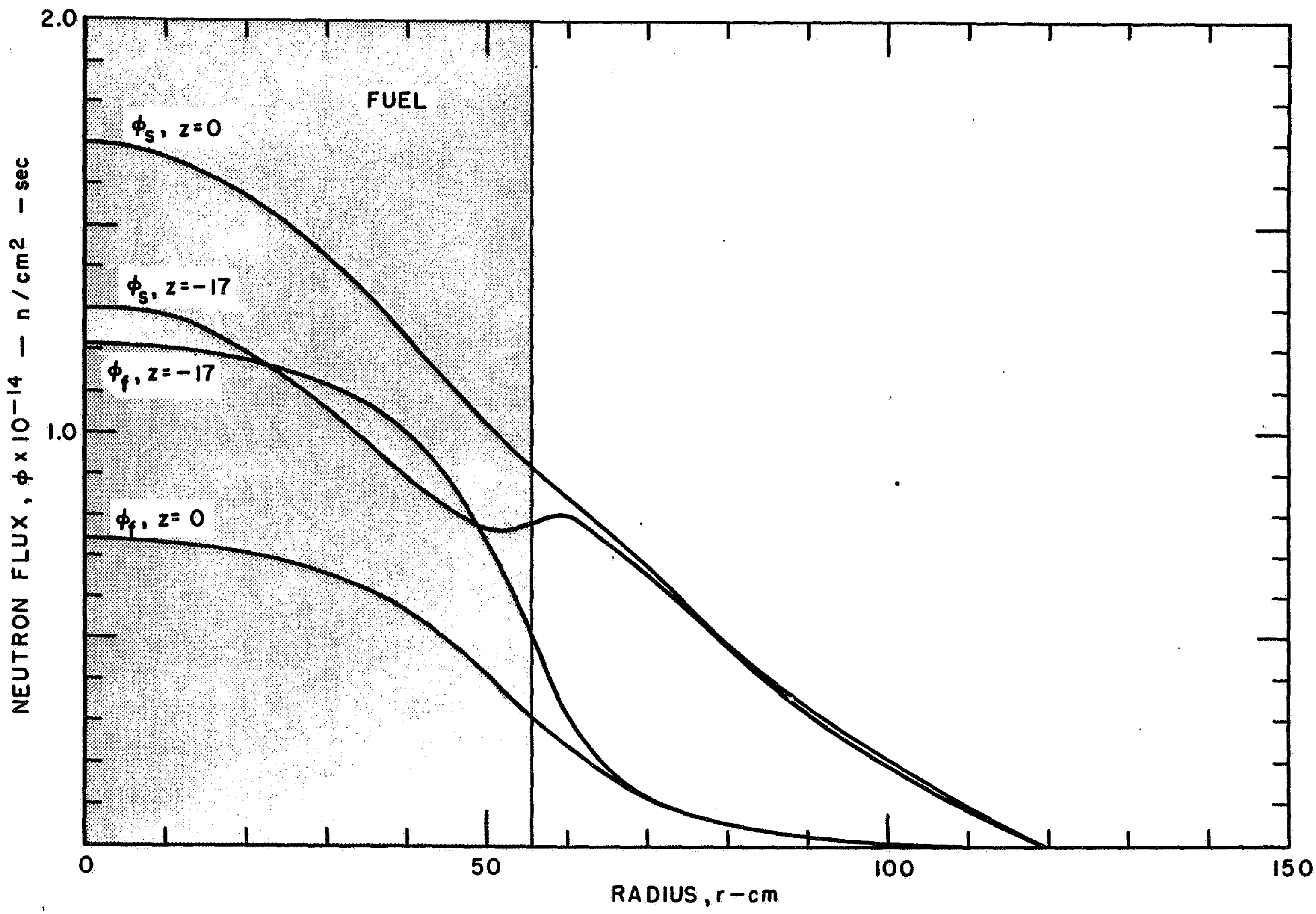


Figure 4.12 Radial flux plot for equilibrium core. Z is the distance in centimeters from the central plane.

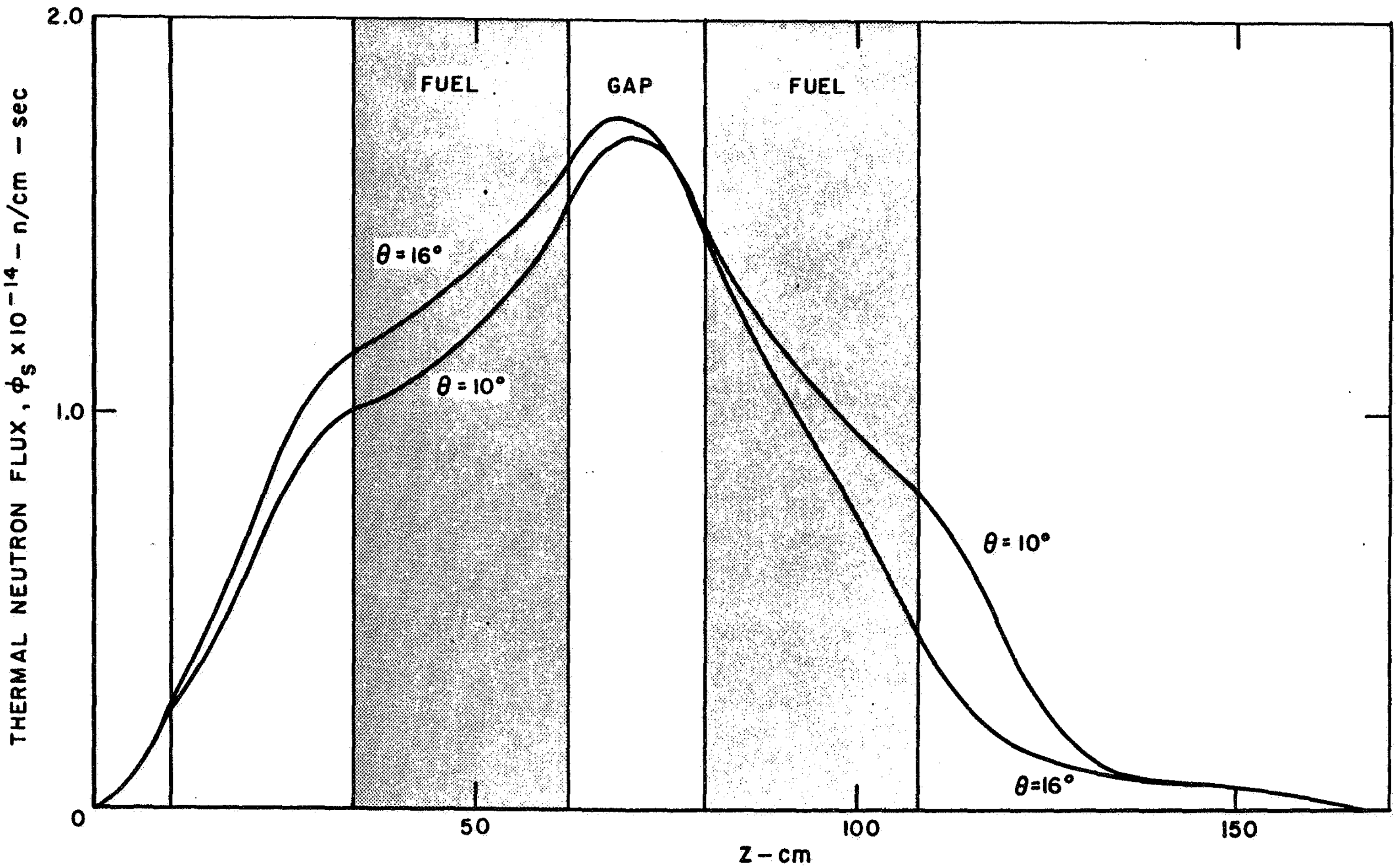


Figure 4.13 Vertical distribution of thermal neutron flux along core centerline for two shim arm positions (θ = 10° and θ = 16°).

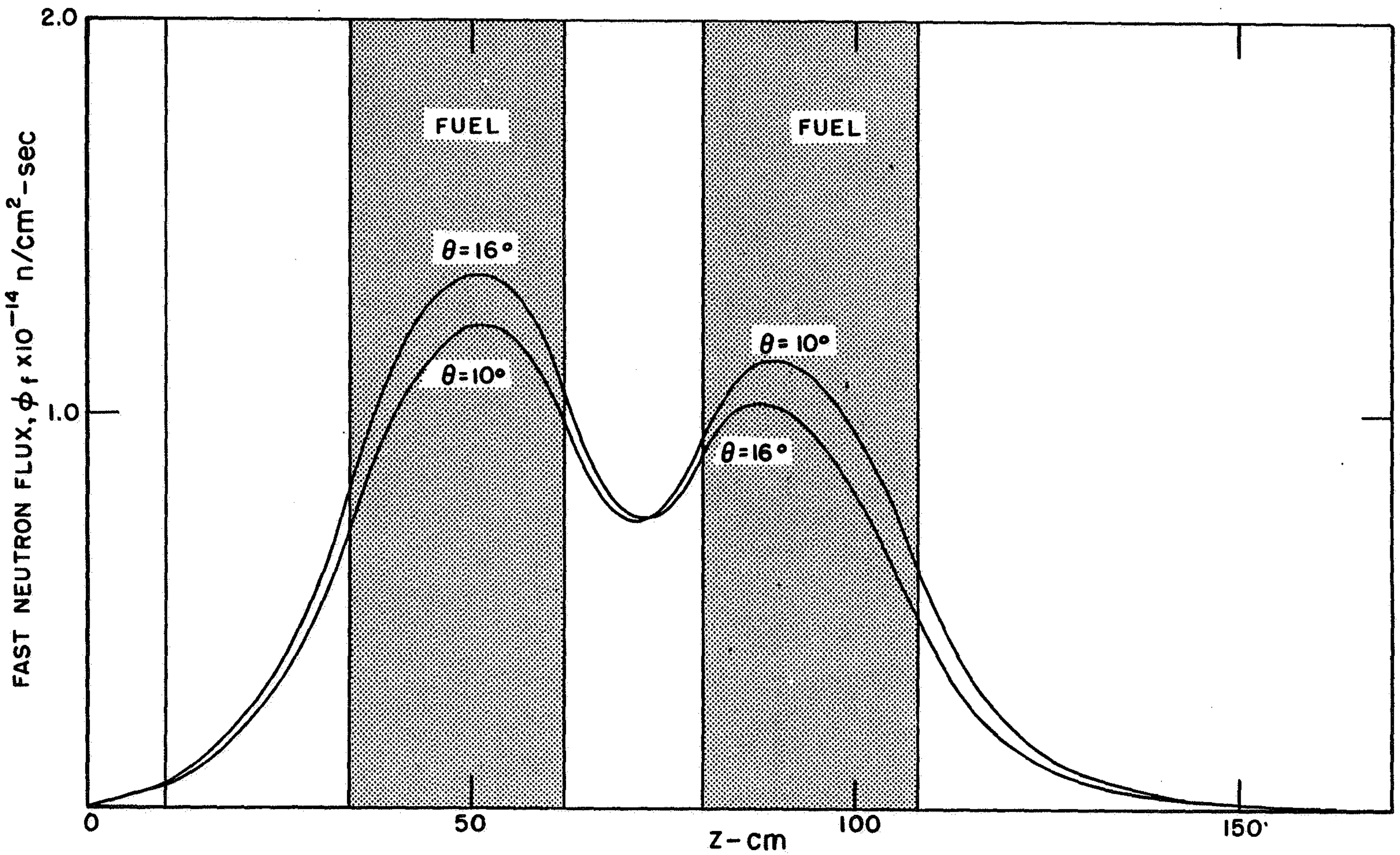


Figure 4.14 Vertical distribution of fast neutron flux along core centerline for two shim arm positions (θ = 10° and θ = 16°).

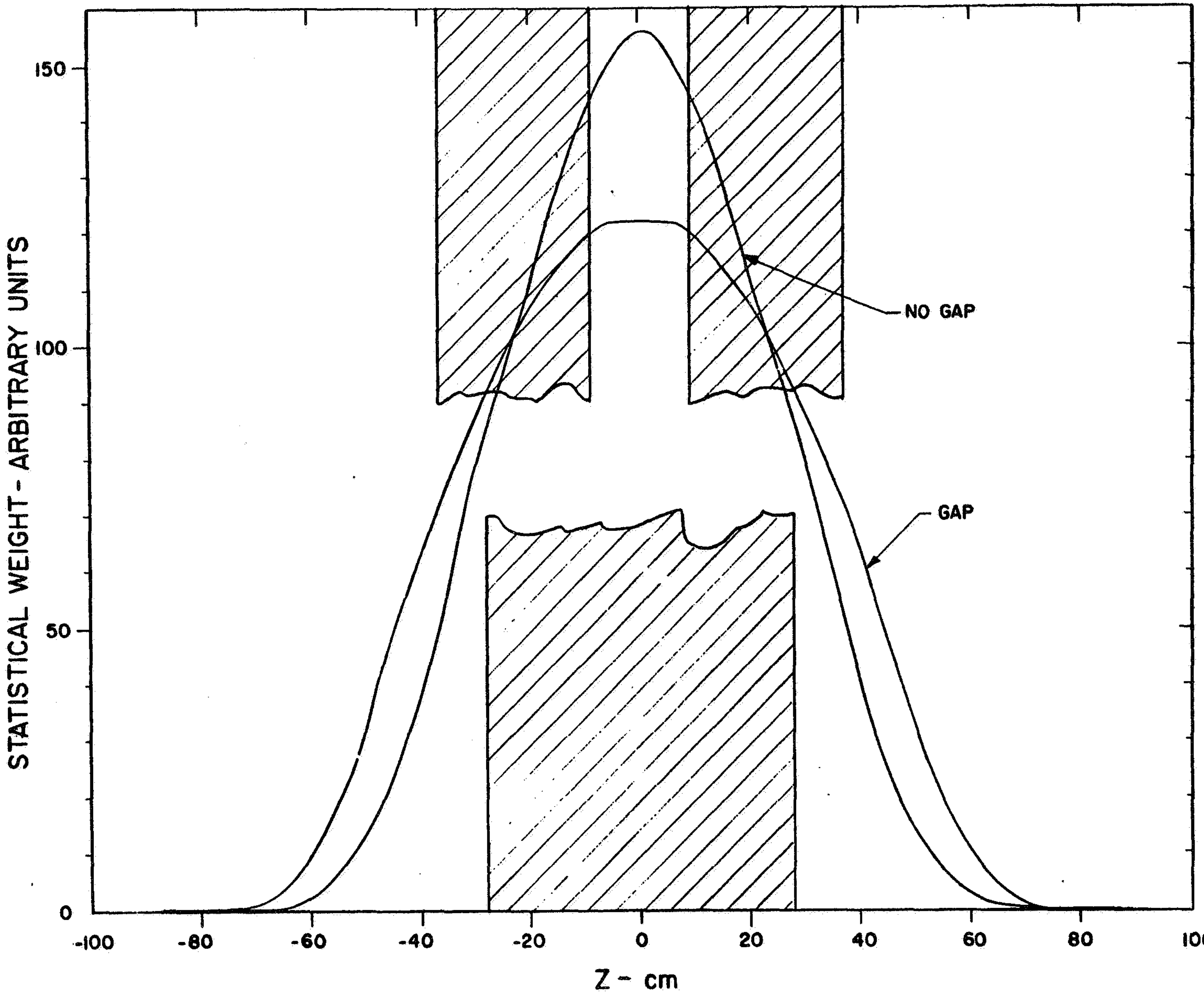


Figure 4.15 Comparison of the vertical statistical weight distribution for thermal absorption $(\phi\phi^{\dagger})_s$, for the NBSR core with and without a gap.

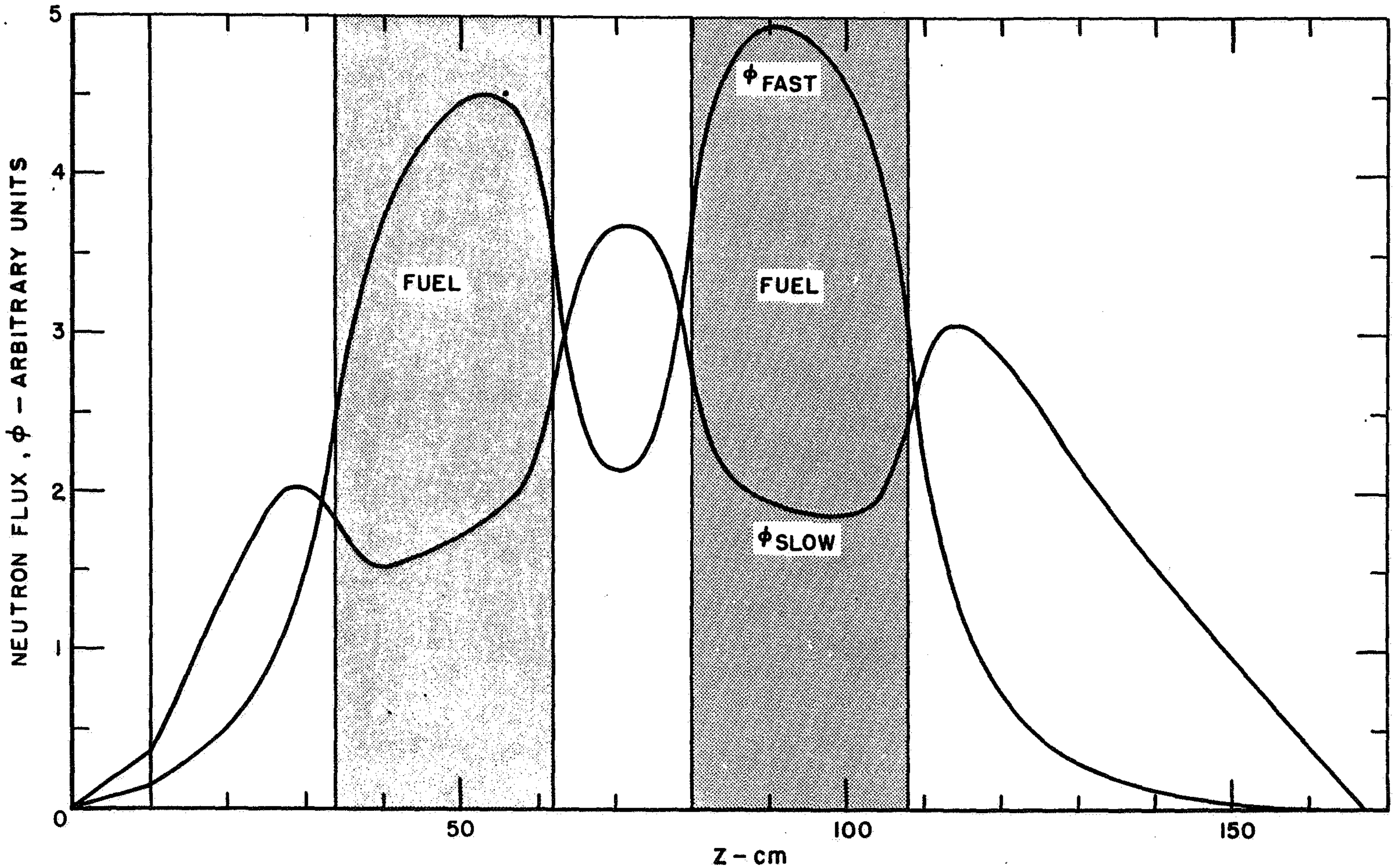


Figure 4.16 Unit cell neutron flux distribution along the central axis of the fuel element.

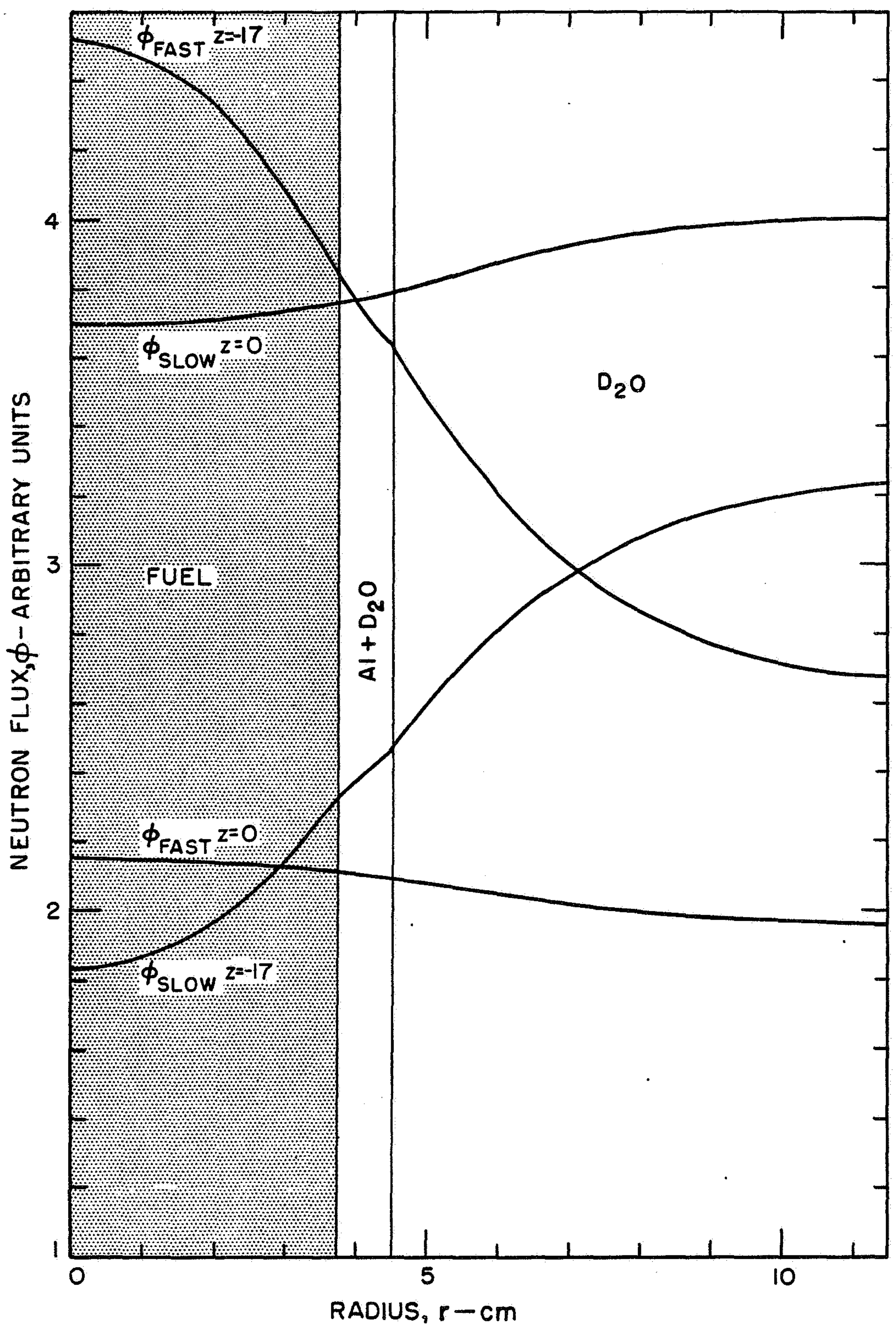


Figure 4.17 Unit cell radial flux distribution, $\phi(r)$. Z is the distance of the plane of the distribution from the central plane in centimeters. (Note: the ordinate does not start at 0).

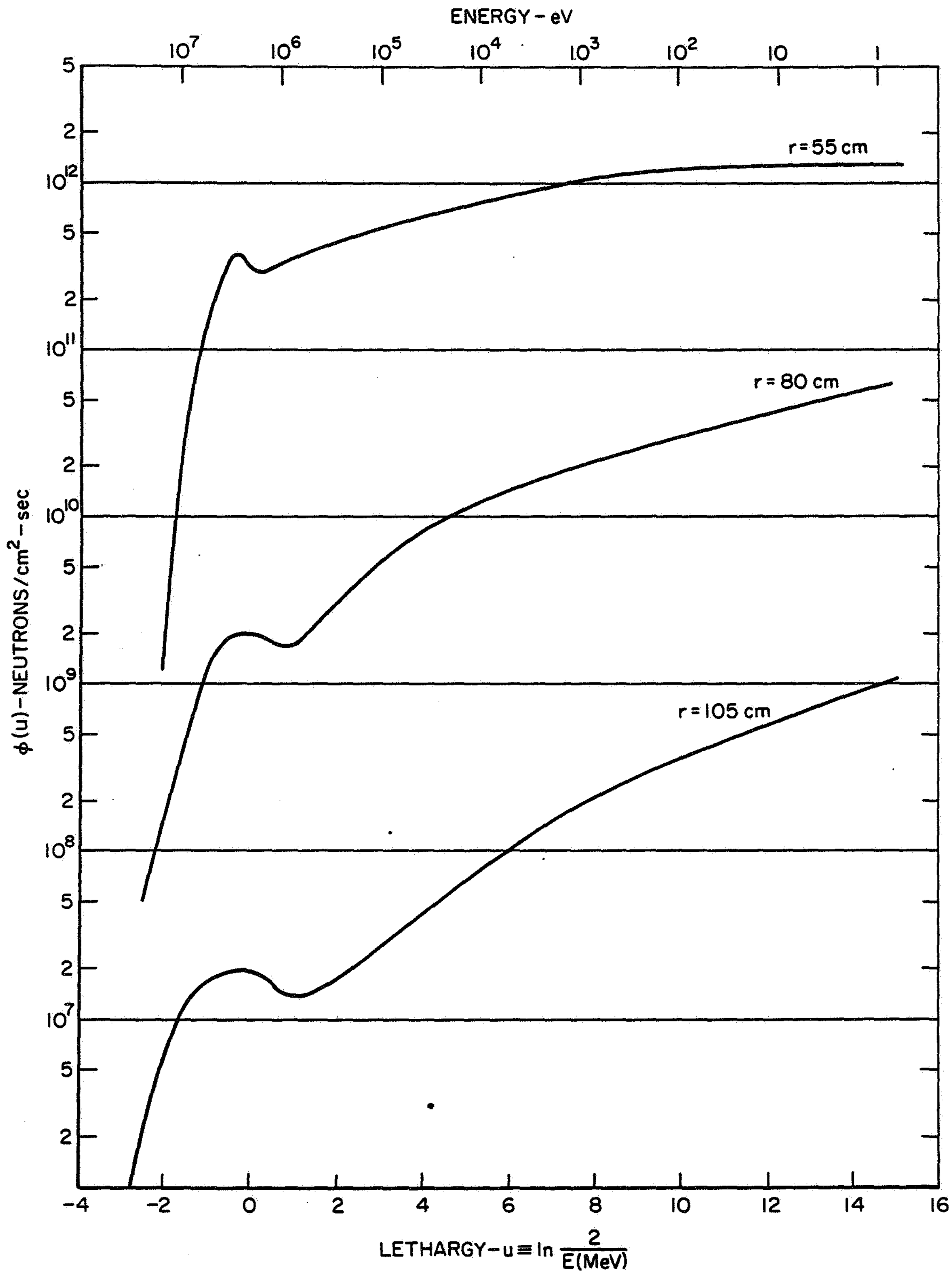


Figure 4.18 Fast flux energy distribution as a function of lethargy $u \equiv \ln 2/E(\text{Mev})$ for three positions in the reflector.

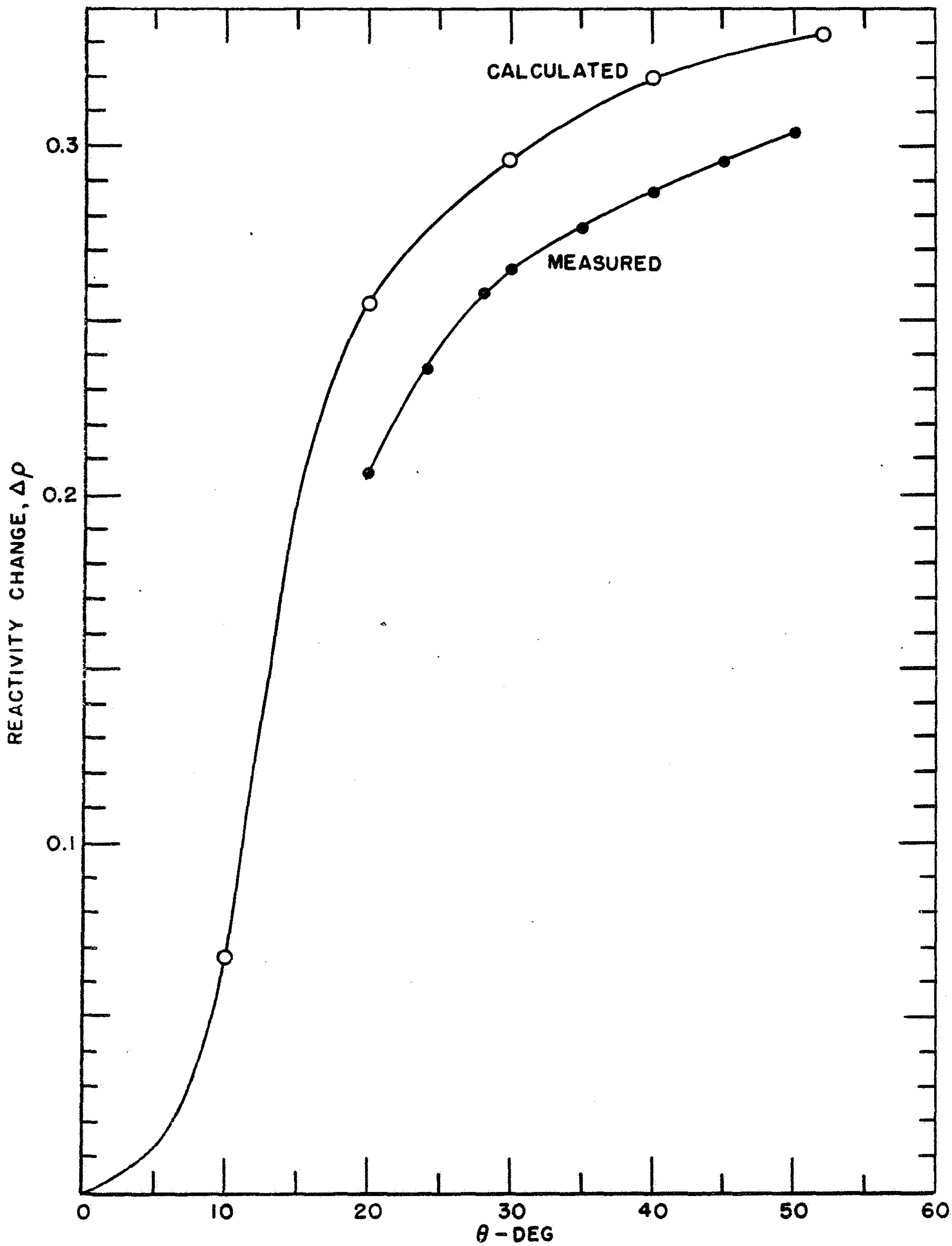


Figure 4.19 Comparison of calculated and measured rod worth curves for the Argonne CP-5 reactor.

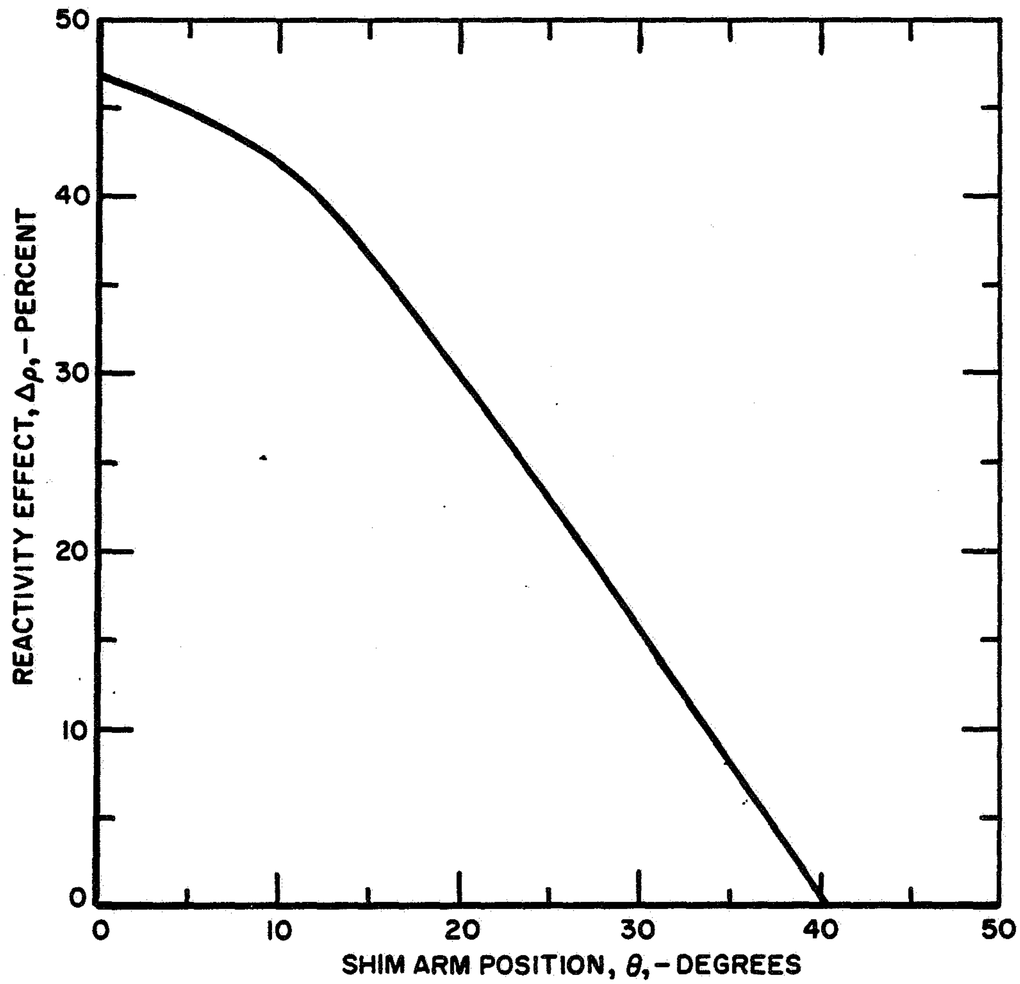


Figure 4.20 NBSR shim arm worth curve.

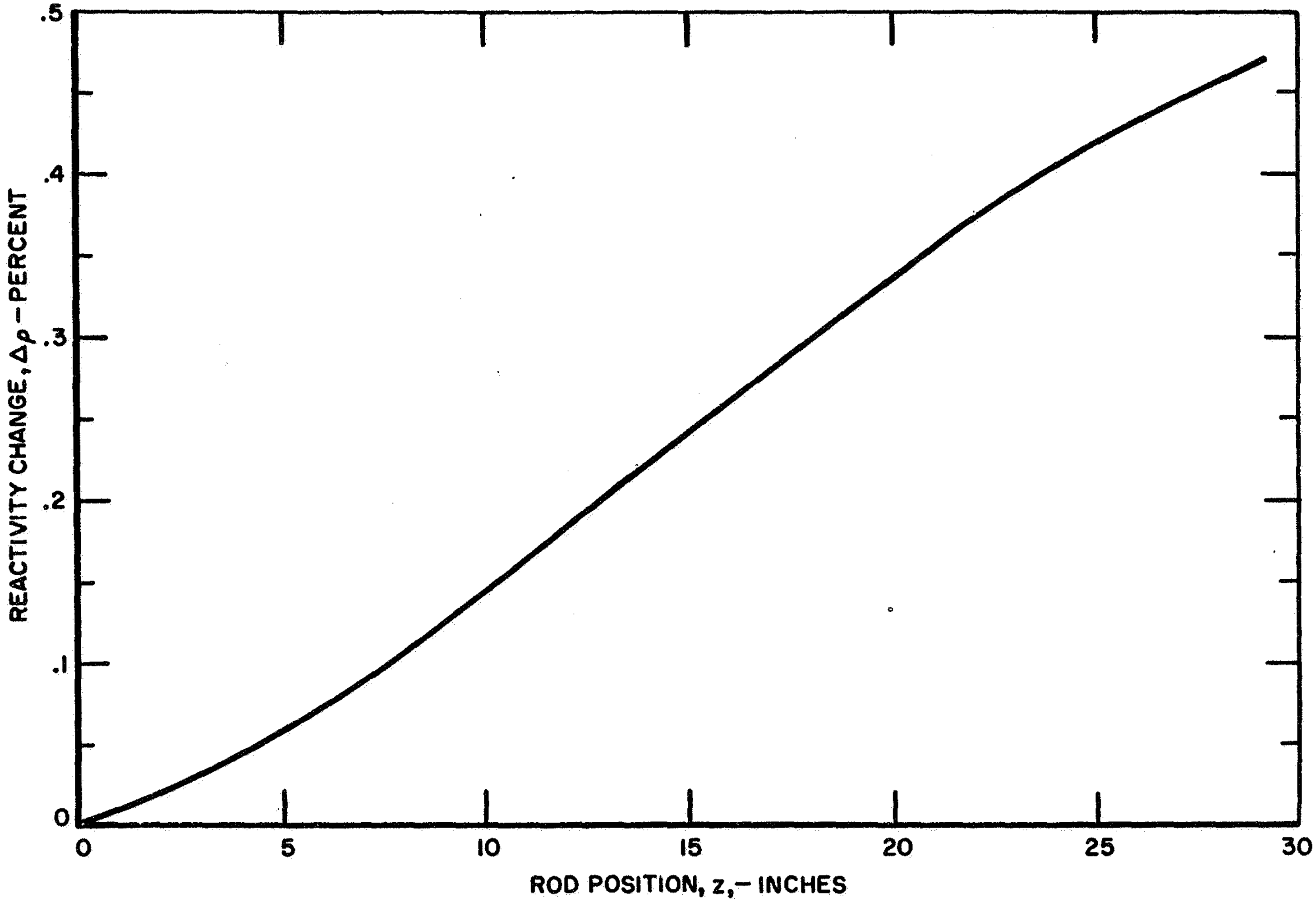


Figure 4.21 Regulation rod worth curve.

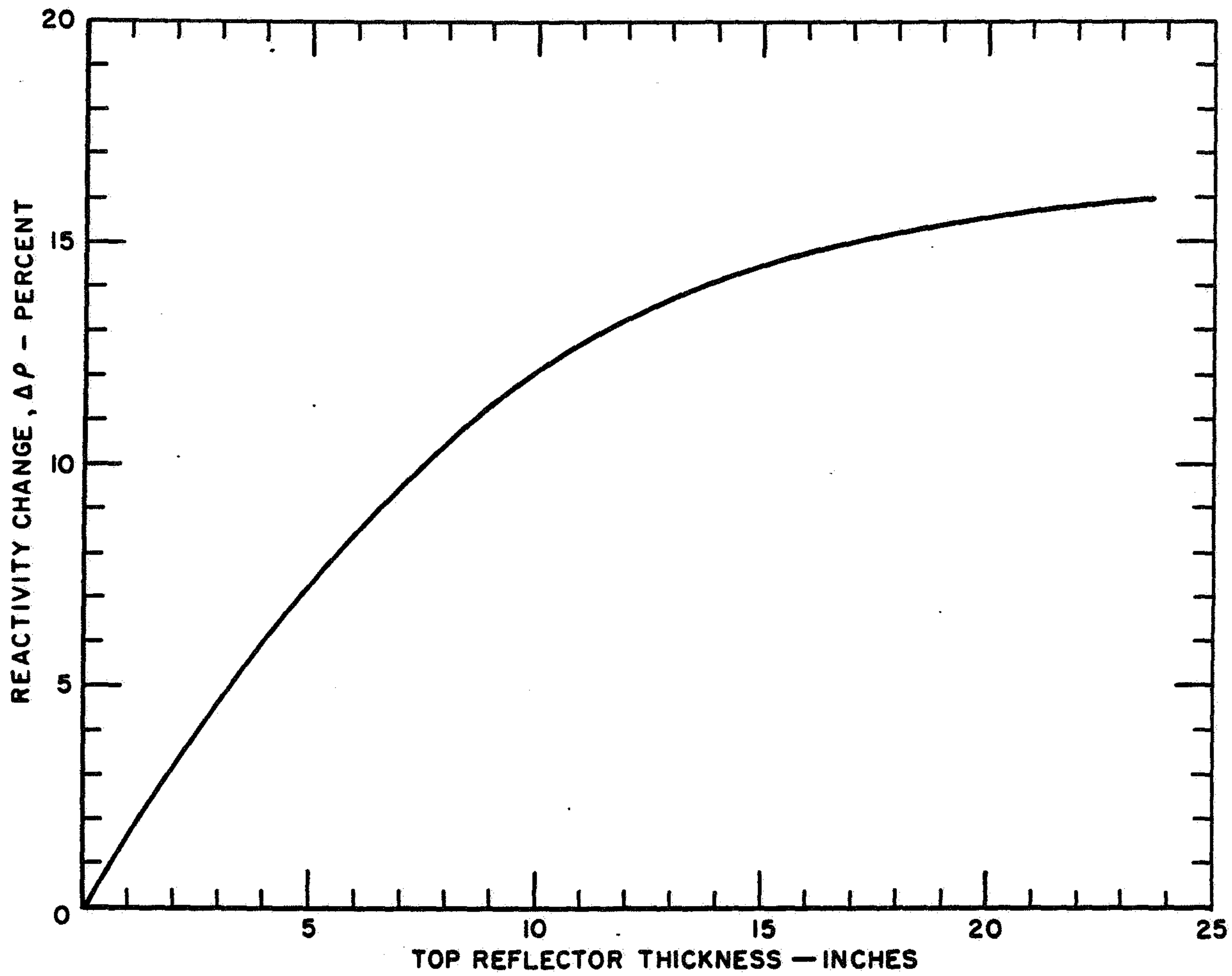


Figure 4.22 Unperturbed top reflector worth as a function of thickness.

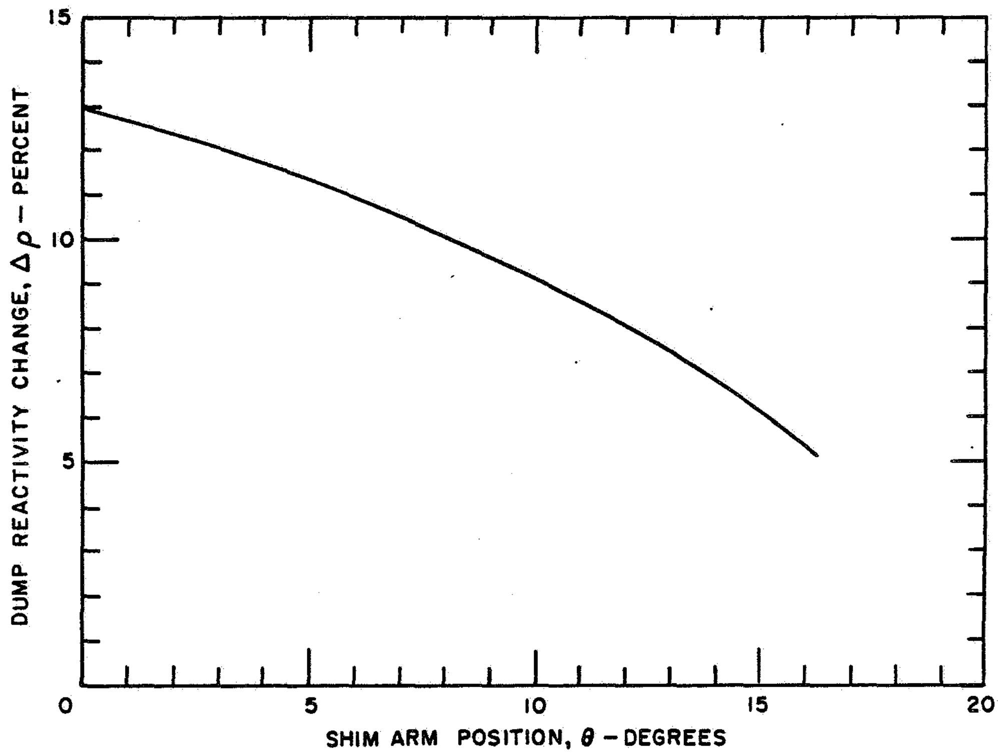


Figure 4.23 Reactivity change for top reflector dump as a function of shim arm position.

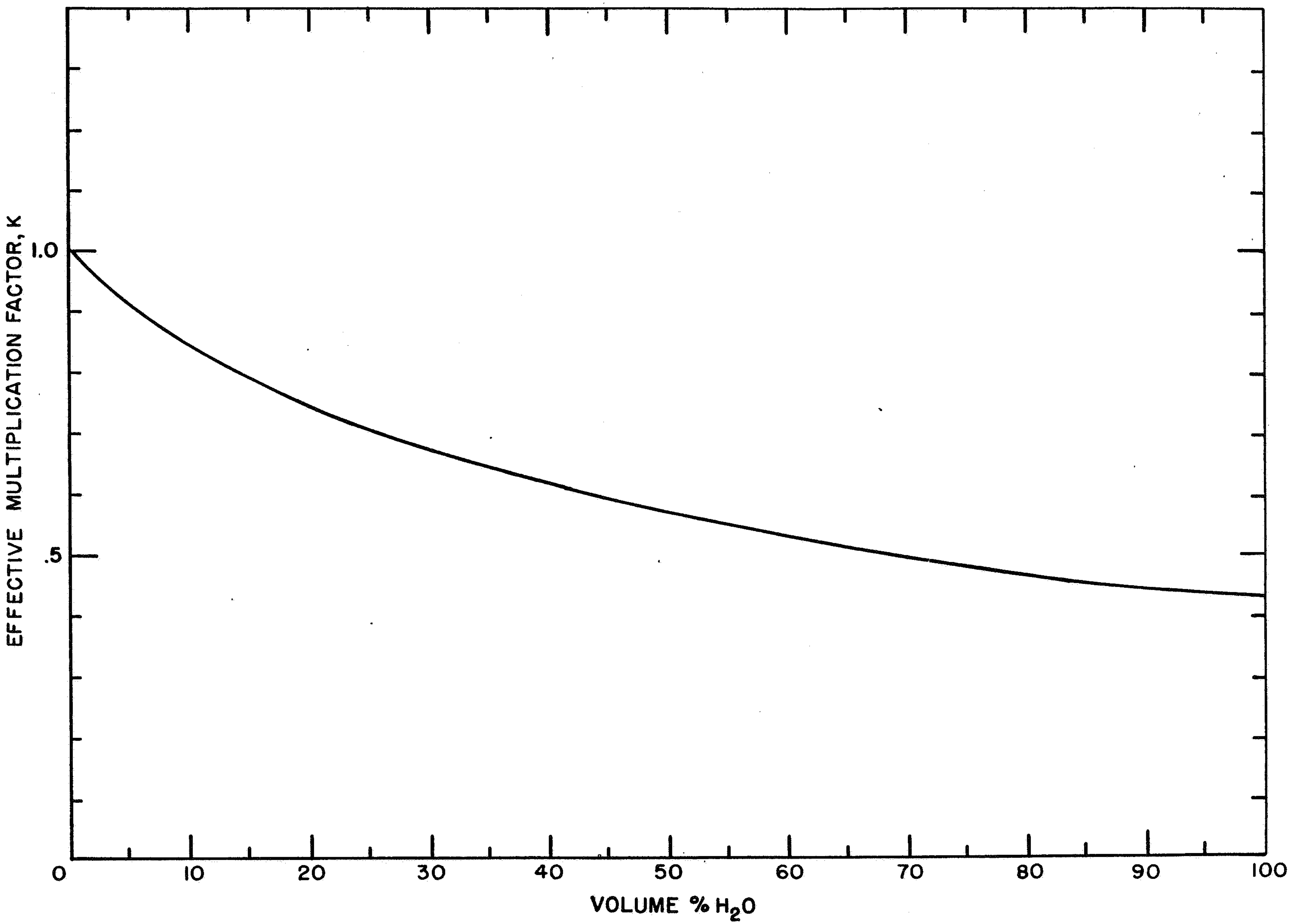


Figure 4.24 Effective multiplication factor as a function of the volume percent of H_2O in the primary system water.

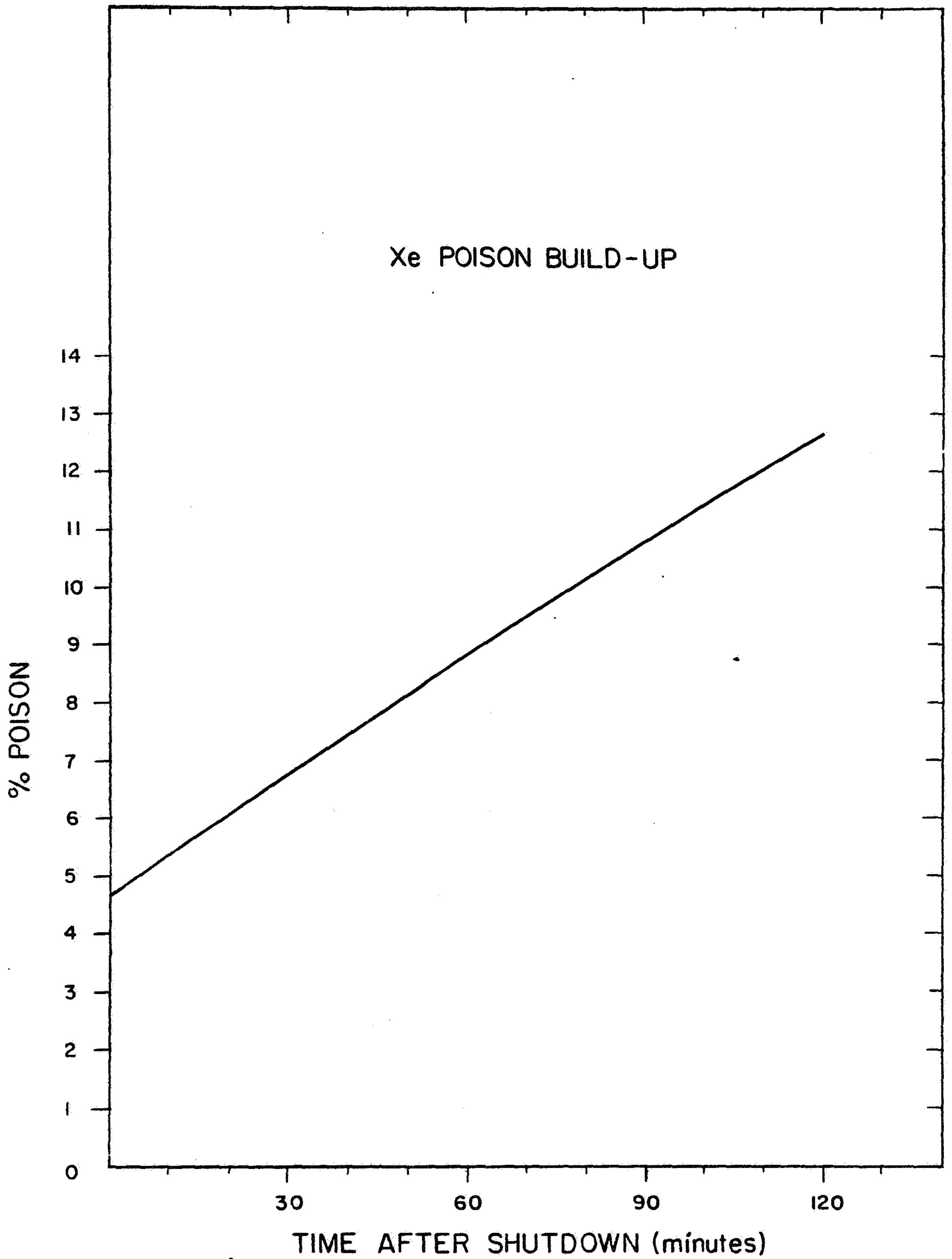


Figure 4.25 Xe buildup as a function of time in hours after shutdown.

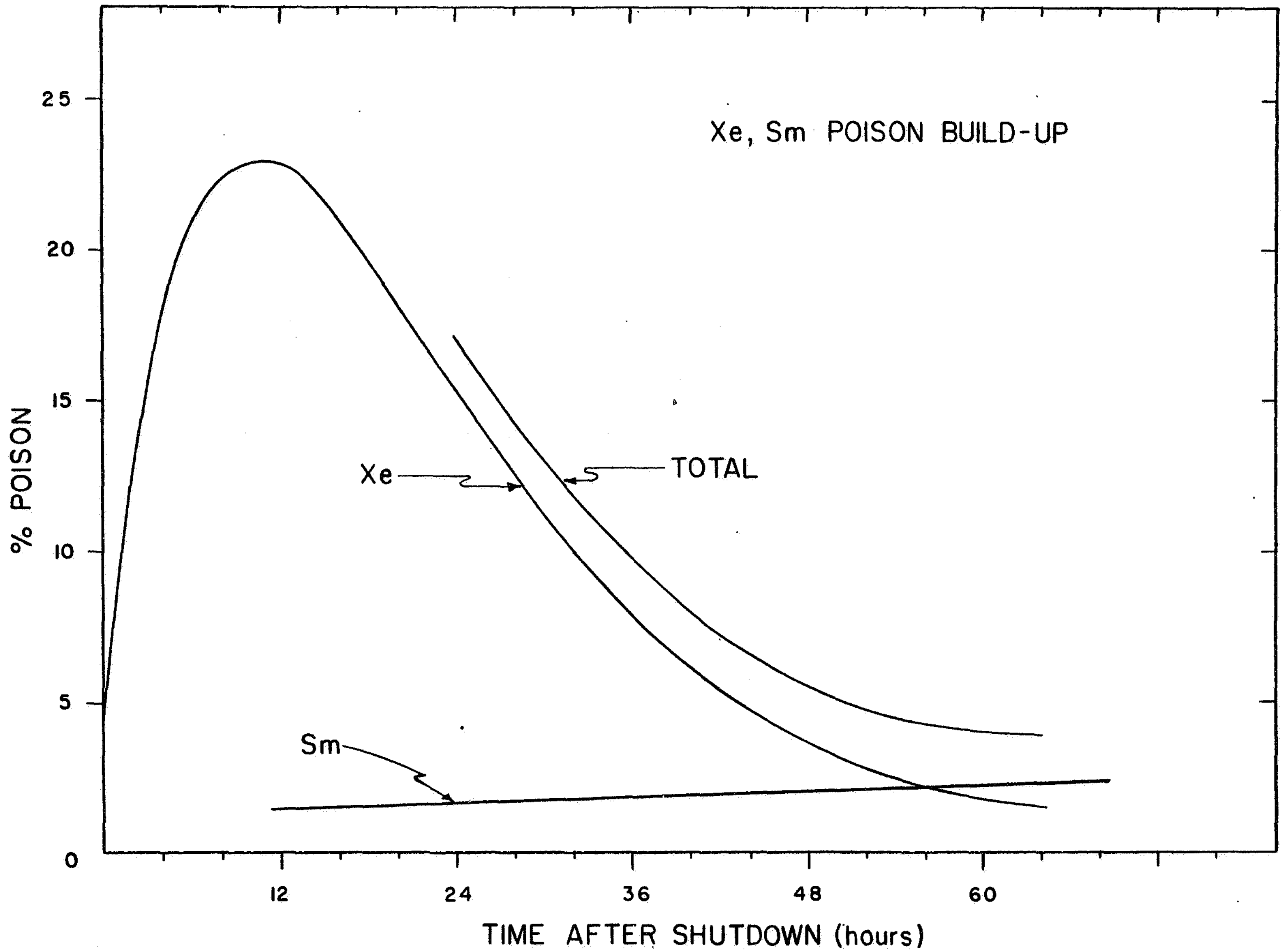


Figure 4.26 Xe and Sm buildup as a function of time in minutes after shutdown for times immediately after shutdown.

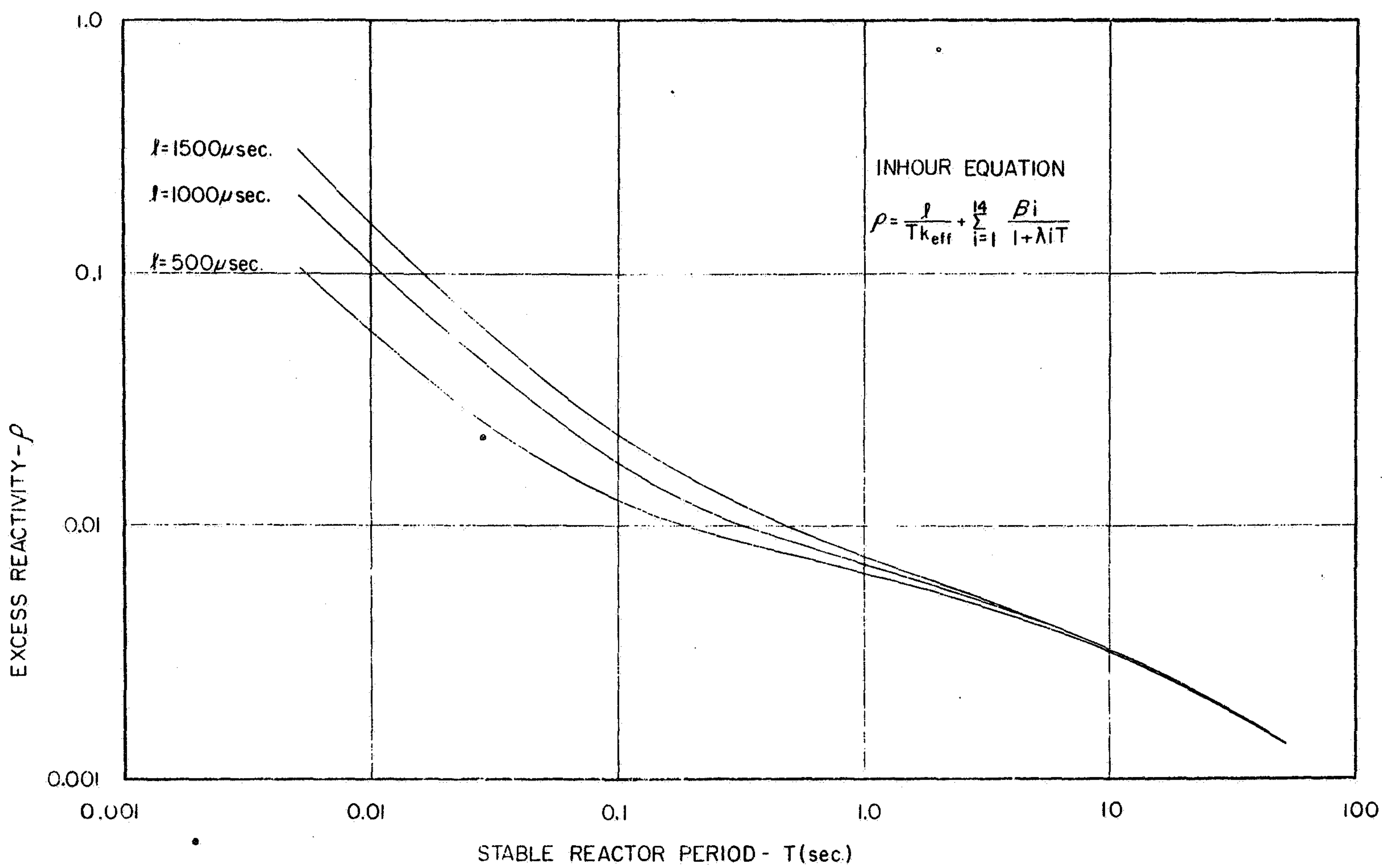


Figure 4.27 Stable period as a function of step reactivity insertion for three neutron lifetimes.

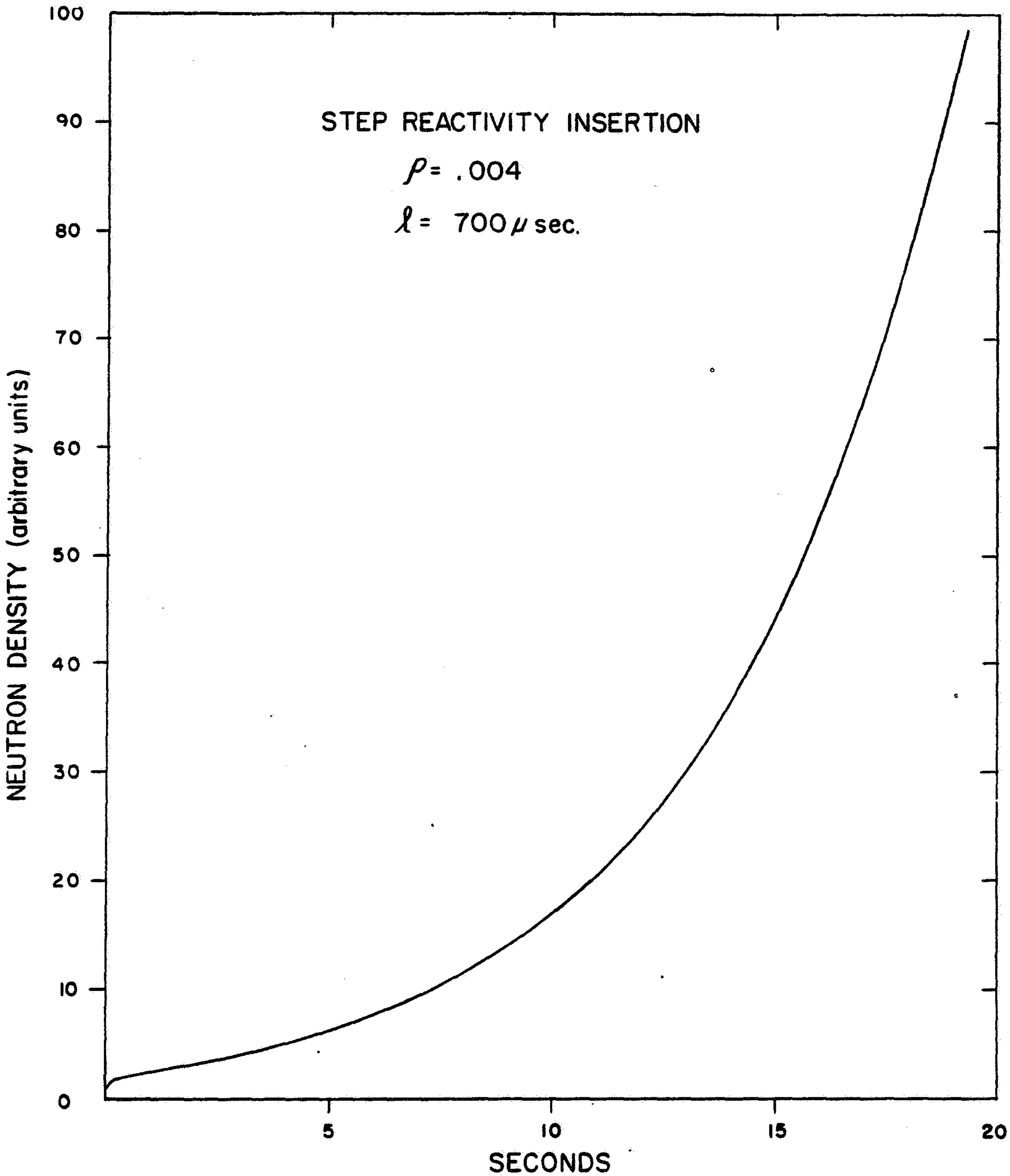


Figure 4.28 Analog computer calculation of the relative neutron density as a function of time resulting from a step reactivity insertion of .004.

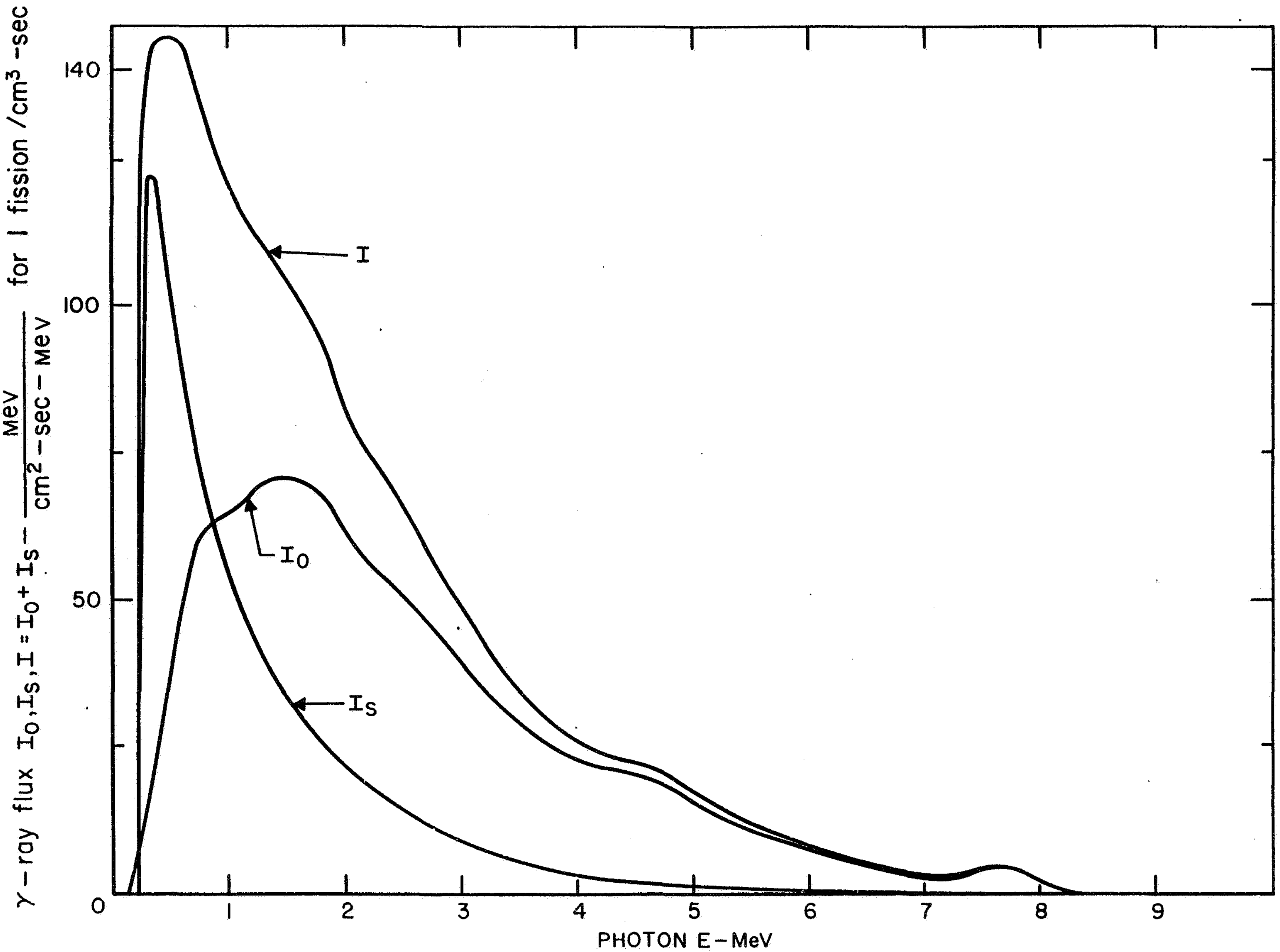


Figure 4.29 Normalized gamma flux spectrum in the center of the NBSR. Io is the direct contribution, Is is the scattered contribution, and I = Io + Is the total gamma spectrum.

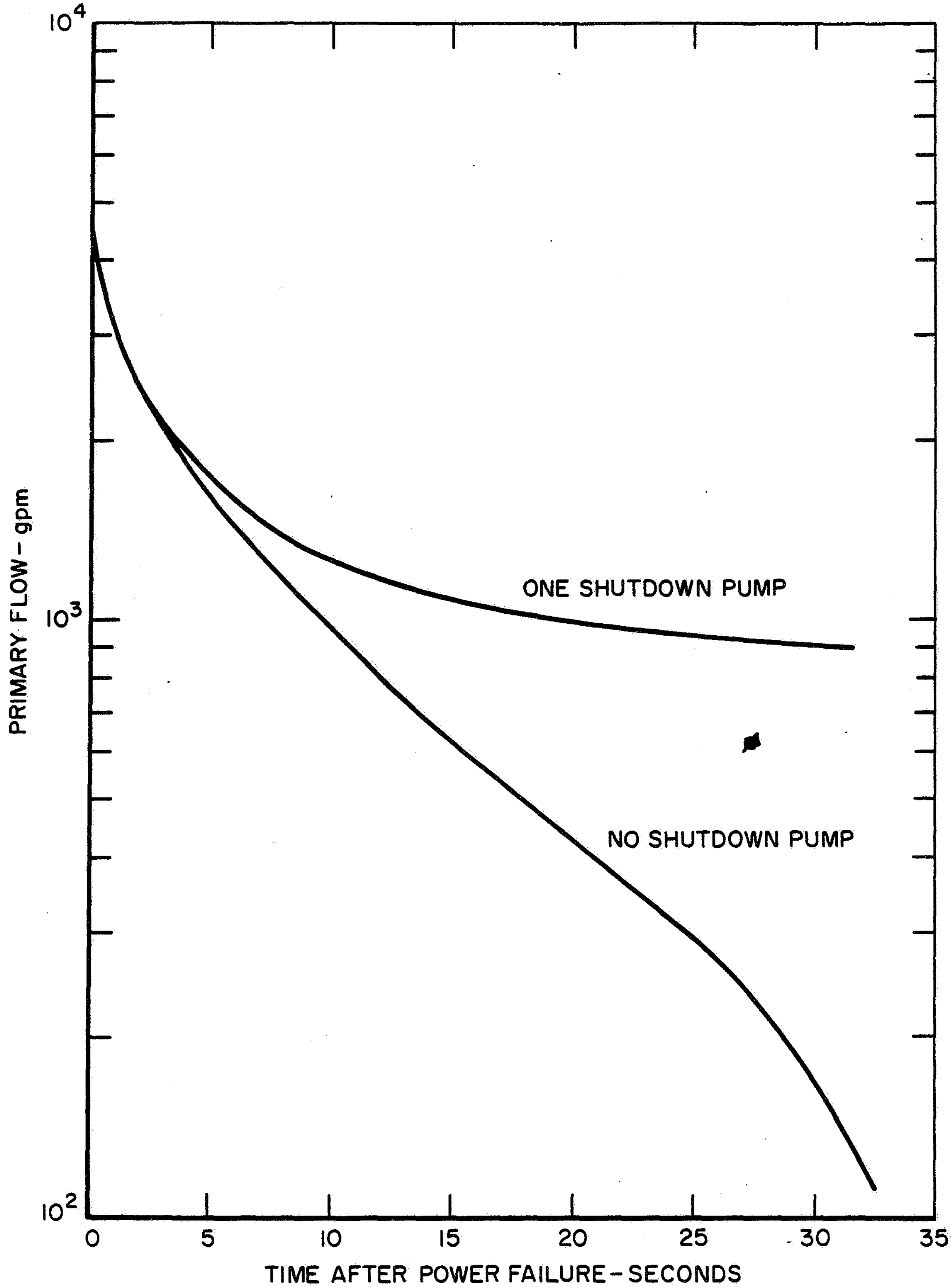


Figure 4.30 Measured primary flow coastdown the lower curve results when the two primary pumps are stopped, and the upper curve shows the effect of the atomatic start-up of one shutdown pump.

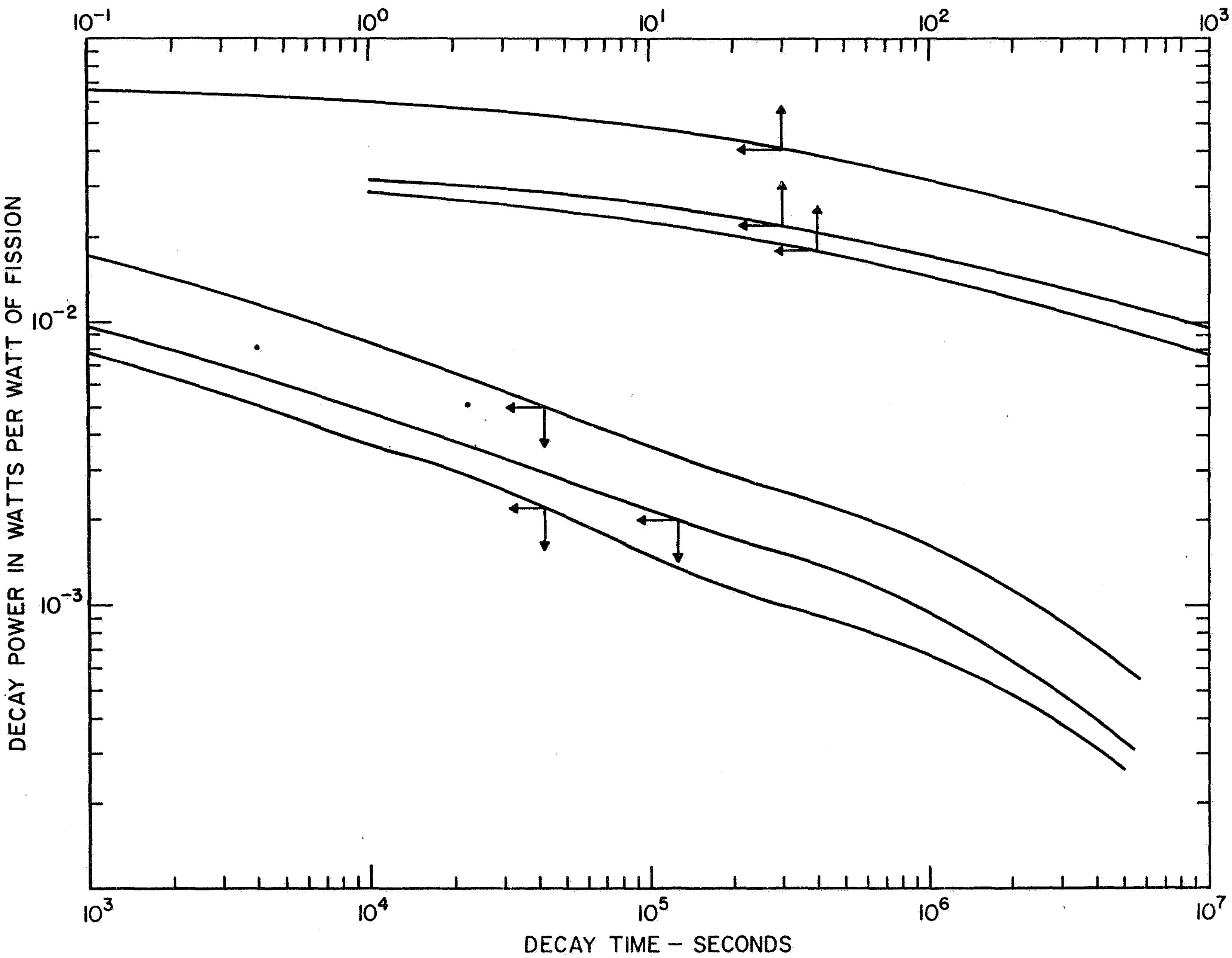


Figure 4.31 Fission product decay heat in watts per watt of operating power as a function of time after shutdown.

SECTION 5. PRIMARY COOLANT SYSTEM

5.1 GENERAL DESCRIPTION

The primary coolant system, as shown in Figure 5.1, circulates heavy water through the reactor at a flow of approximately 5100 gpm. Figure 5.2 shows the system integrated with other heavy water systems. Symbols used in these and subsequent drawings are listed in Figure 5.3.

For the equilibrium core at 10 Mw, approximately 1200 gpm of heavy water, supplied by the primary coolant pumps, enters the inner plenum to cool the central 6 fuel elements and the remaining 3900 gpm is directed to the outer 18 fuel elements via the outer plenum. About 4% of the total flow in each plenum bypasses the fuel elements and cools the various in-core thimbles. The water passes up through the fuel element and down the outside; it leaves the reactor vessel through two 12" pipes which join outside the sub-pile room. After passing through a flow orifice and a strainer, water enters the pump suction header. Three main coolant and two shutdown coolant pumps are arranged in parallel. During normal operation two of the primary pumps are required.

The coolant then flows through the tube side of a horizontal U-tube heat exchanger. Secondary light water transfers the heat to the atmosphere via pumps and a cooling tower. Primary water then is returned to the reactor, having completed the flow cycle.

The purification system is supplied by a 3" overflow line from the return to the primary system via the emergency cooling tank.

Thermal expansion and contraction of the D_2O is absorbed by a 14,650 gallon storage tank which also acts as a reservoir for D_2O dumped from the core tank.

5.2 SYSTEM COMPONENTS

5.2.1 MAIN HEAT EXCHANGER, HE-1. The main heat exchanger, HE-1, shown in Figure 5.4, is an all aluminum, horizontal, cylindrical unit of the shell and tube type, with primary coolant circulating inside the tubes and secondary light water flowing on the shell side.

Table 5.2-1 lists pertinent data on the heat exchanger. The tube bundle consists of 1285, 3/4" O.D. Alclad aluminum "U" tubes. The two inner rows are of 16 BWG (0.065 inch average wall thickness) with the remaining of 18 BWG (0.049 inch average wall thickness).

There are double tube sheets of Alclad aluminum having a 3 inch space between sheets with a drain and a vent to detect any leakage into the space. The tubes are rolled into the inner tube sheet and are rolled and welded to the outer tube sheet. The primary head inlet and outlet chambers are separated by a horizontal partition plate. Both vertical and horizontal baffles are provided to distribute the secondary cooling water.

The dimensions of the heat exchanger are 52-3/4 inches shell outside diameter by 23 feet 8-1/8 inches overall length. The secondary inlet and outlet nozzles extend above and below the shell to give an overall height of 8 feet 10-3/4 inches.

Table 5.2-1 Heat Exchanger, HE-1

Manufacturer:	Whitlock Manufacturing Company
Tubeside Fluid:	Heavy Water
Shellside Fluid:	Secondary Light Water

Table 5.2-1 (con'd)

Design Flow, (gpm):	
Tubeside:	5100
Shellside:	4950
Design Pressure, (psig):	
Tubeside:	100
Shellside:	75
Design Temperature, (°F):	
Tubeside:	200
Shellside:	150
Operating Pressure, (psig) (max.):	
Tubeside:	75
Shellside:	45
Allowable Pressure Drop, (psi):	
Tubeside:	7.5
Shellside:	7.5
Operating Temperature at Design Flow (°F):	
D_2O In:	112
D_2O Out:	100
Secondary Cooling In:	84
Secondary Cooling Out:	98
Fouling Factor:	
Tubeside:	.0005
Shellside:	.001
Duty, (BTU/hr):	34.6×10^6
Materials of Construction:	
Tubes: 3/4 inch O.D. x 18 BWG	Alclad 6061 Aluminum
3/4 inch O.D. x 16 BWG (2 inner rows)	Alclad 6061 Aluminum
Shell: 1/2 inch thick	Alclad Aluminum
Tube Sheets: 3-1/2 inch thick	Alclad Aluminum
Channel Head: 7/16 inch thick	Alclad Aluminum
Number of U-Tubes:	1285
Closure Design:	
Channel-to-Tube Sheet:	Welded
Tube-to-Tube Sheet:	Welded and Rolled
Shell-to-Tube Sheet:	Welded
Inspection Covers:	Bolted

5.2.2 MAIN COOLANT PUMPS. The three primary coolant pumps are single stage centrifugal units operating in parallel. They are of the shaft-sealed type.

Each pump motor is a single speed, 480 volt, 3 phase, 60 cps unit having a rating of 100 hp. Table 5.2-2 lists pertinent data on these pumps.

During normal power operation two pumps are required to maintain the required flow, with the third pump serving as a spare. Figure 5.5 shows the flow vs. head and other information for one pump operating. Pumps are brought up to speed before opening the discharge valve to prevent excessive current on startup.

Table 5.2-2 Main D_2O Coolant Pumps

Manufacturer:	Allis-Chalmers Mfg. Company
Design Flow Capacity at 139' D_2O (gpm):	1955
Number of Stages:	One
Motor Size (hp):	100
Speed (rpm):	1750
Design Head of D_2O (ft):	139
Materials of Construction:	316 Stainless Steel
Type of Shaft Seal:	John Crane Type I Bellows Shaft Seal
Motor Power Supply:	
DP-1:	MCC A-3
DP-2:	MCC B-4
DP-3:	MCC A-3
Control Points:	Lock-Out-Stop Switch (Local) Hand-Off-Auto Switch (Control Room)

5.2.3 SHUTDOWN COOLING PUMPS. Two 800 gpm centrifugal pumps are installed in parallel with the primary coolant pumps to provide forced cooling to the reactor during shutdown periods and in the event of a power failure to the main pumps. Each pump has a 7-1/2 hp AC motor and a like size DC motor connected to a single shaft. Table 5.2-3 lists pertinent data on these pumps.

Table 5.2-3 Primary D_2O Shutdown Pumps

Manufacturer:	Allis-Chalmers Mfg. Company
Design Flow Capacity at 24 ft. D_2O (gpm):	800
Number of Stages:	One
Motor Size (hp):	
AC Motor:	7-1/2
DC Motor:	7-1/2
Speed (rpm)	1160

Table 5.2-3 (con'd)

Design Head of D_2O (ft.):	24
Materials of Construction:	316 Stainless Steel
Type of Shaft Seal:	John Crane Type I Bellows Shaft Seal
Motor Power Supply:	
DP-5:	AC Motor - MCC A-5 DC Motor - 125v DC Motor Control Center
DP-6:	AC Motor - MCC B-6 DC Motor - 125v DC Motor Control Center
Control Points:	
AC Motor:	Lock-out-Stop Switch (Local)
AC Motor:	Hand-Off-Auto Switch (Control Room)
DC Motor:	Hand-Off-Auto Switch (Control Room)

5.2.4 VALVES

5.2.4.1 Control Valves. There are two types of remotely positioned valves installed in the primary system, an air operated type and a motor driven type.

DWV-1 and 2 are respectively 12" and 8" motor operated diaphragm valves which are positioned from the control room to provide the flow distribution required for the two reactor inlet plena. DWV-1 supplies the necessary flow for the outer plenum, while DWV-2 distributes flow to the inner plenum.

All materials in contact with D_2O are aluminum, stainless steel or government rubber styrene. Table 5.2-4 lists the pertinent data on each valve. Both valves are equipped with hand-wheel operators so they can be positioned manually. Each valve is equipped with a leak detector in the valve body which will give control room indication in the event of a diaphragm leak.

Table 5.2-4 Control Valves DWV-1 and DWV-2

Manufacturer:	Automotive Rubber Company
Type:	Motor Driven, Diaphragm
Design Pressure, (psig):	125
Design Temperature, (°F):	150
Hydrostatic Test Pressure, (psig):	185
Working Pressure, (psig):	65
End Preparation:	Two 150 psi ASA Flanges
Materials of Construction:	
Body:	Aluminum

Table 5.2-4 (con'd)

Compressor:	Cast Iron
Bonnet:	Cast Iron
Diaphragm:	G.R.S. Rubber
Stem:	Steel
Motor:	
Size DWV-1, (hp):	1.6
DWV-2, (hp):	0.65
Volts, (AC):	480
Cycles:	60
Phase:	3

Additional remote, air-operated, diaphragm, control valves are provided on the discharge of each primary pump and shutdown cooling pump. These valves are operated from the control room and are installed to allow the pump to be brought up to speed before opening, thus reducing the starting current required for the pump.

Other air-operated diaphragm valves are provided in the two overflow lines for the reactor vessel and in the 8 inch dump line. All valves are equipped with leak detectors to detect a diaphragm rupture. Figure 5.6 shows a typical installation.

Pertinent data is given for these valves in Table 5.2-5. These valves are not equipped with positioners and can therefore be used only in the full "open" or "closed" position.

Table 5.2-5 Primary System Air-Operated Control Valves

Valve Tag No.	Size (in.)	Diaphragm Operation	Pilot Tag No.	Pilot Solenoid Operation
DWV-3	10	ATO*	SV101	ETC*
DWV-4	10	ATO	SV102	ETC
DWV-5	10	ATO	SV103	ETC
DWV-7	6	ATC	SV105	ETC
DWV-8	6	ATC	SV106	ETC
DWV-9	8	ATC	SV117	ETC
DWV-10	3	ATC	SV116	ETC
DWV-37	3	ATO	SV118	ETO

*ATO - Air-To-Open
ATC - Air-To-Close
ETO - Energized-To-Open
ETC - Energized-to-Close

5.2.4.2 Manual Valves. The majority of manual valves in the primary system are Hills-McCanna valves equipped with G.R.S. or hypalon-nylon reinforced diaphragm. They are rated for 125 psig and have 150 pound ASA flanged ends.

All valves which are 3 inches and larger are equipped with leak detectors to detect any diaphragm failure.

5.2.4.3 Check Valves. Aluminum check valves are installed on the discharges of the primary coolant pumps, and the shutdown coolant pumps. Two additional check valves are provided on the inner and outer inlet plenum to the reactor vessel. Back leakage on these two check valves is limited to a maximum of $10cc^3$ per minute with a ΔP of 10 psi. All check valves are seal-free and completely enclosed to avoid any leakage of the contained D_2O.

5.2.5 PIPING. Type 6061-T6 aluminum is used as the piping material. Eighteen inch pipe with a nominal 1/2 inch wall thickness is used for full flow throughout most of the system. Where the coolant flow is divided on the inlet, then fourteen inch pipe is used. On the outlet of the reactor, twelve inch diameter pipe is installed. Twelve inch pipe and smaller is ASA schedule 40. All fittings are forged 6061-T6 aluminum. Flanges are also forged 6061-T6 aluminum.

Flanged connections have 1/8" thick full face gaskets made of G.R.S. or hypalon-nylon material. Leak detectors are installed at major flanges to locate any leakage of heavy water.

All welded joints are examined by radiograph and accepted on the basis of ASME standards.

5.2.6 STRAINERS. Two 18" aluminum strainers are provided. One located on the inlet of the reactor, the other on the outlet. Both strainers have removable bolted covers for easy removal of stainless steel #4 wire mesh baskets. Vents and drains are also provided on each strainer.

5.2.7 INSTRUMENTATION

5.2.7.1 General. Necessary instrumentation is installed to provide remote read-out on: reactor inlet flow to each plenum, reactor outlet flow, reactor ΔT, reactor vessel level, and reactor overflow. Pressure measurements of various points in the system are by local gauges which do not give control room read-out.

5.2.7.2 Flow. Reactor inlet plenum flow is sensed by two flow elements, FE-3 and FE-4, installed in the line to each plenum. Their function is to measure, monitor and record the flow. An alarm unit provides four "on-off" control signals; one to the annunciator, two to scram networks, and one to a scram interlock relay. FRC-3, the outer plenum flow channel has a range of 0-4000 gpm, while FRC-4 is ranged at 0-1600 gpm.

Outlet flow from the reactor is sensed by FE-1. The flow signal from the D/P cell is converted from a square-root function to a linear function, and is transmitted to the BTU measuring channel KWH-1, to the flow recorder, and to an alarm unit. The alarm unit has four independent "on-off" outputs. These outputs feed two scram signals, the scram interlock relay and an annunciator. The range is 0-6000 gpm.

Flow measurement is provided in the 3" over-flow line from the top of the reactor vessel, thus assuring the vessel is filled to normal operating level. The flow signal from FCA-2 supplies an alarm unit. This alarm unit feeds two independent "on-off" signals which supply an annunciator and the reactor startup interlock relay. The range is 0-30 gpm.

5.2.7.3 Temperature. TRC-1 measures the temperature of the primary coolant at the inlet and outlet of the reactor vessel, to provide continuous recording and monitoring of the ΔT. The differential temperature signal is transmitted to the BTU measuring channel, KWH-1, to recorder TR-1 and to an alarm unit. This alarm unit supplies four signals, two to the scram circuitry, one to the scram interlock relay and the fourth to an annunciator. Range is 0-20°F ΔT.

A resistance type temperature detector installed in the reactor outlet line feeds channel TRC-2. An R to I converter then supplies signals to an alarm unit and a recorder. Range of this channel is set at 50-150°F.

A similar channel TRC-3 is provided on the reactor inlet to sense and control the temperature by regulating the secondary coolant by-pass flow around the cooling tower, through control of SCV-1 and SCV-2, thus maintaining a constant primary coolant inlet temperature. An alarm unit provides annunciator action on low temperature. Temperature range of the channel is 50-130°F.

TIA-4 indicating temperature channel, senses primary coolant temperature on the outlet of HE-1, heat exchanger. High temperature annunciation is provided. The range of this channel is 50-130°F.

5.2.7.4 Level. Moderator level in the reactor vessel is sensed, monitored and recorded by LRC-1. Alarm units provide two signals at various levels of moderator, a shim rod insertion signal and a scram signal. In addition, on extremely low level, the primary coolant pumps are automatically turned off and the shutdown pumps placed in service. The range of the channel is 0 to 200 inches D_2O.

5.2.7.5 Pressure. Local Bourdon tube, pressure gauges are provided on the inlet and outlet of each primary and shutdown coolant pump and across the heat exchanger, HE-1.

5.3 COOLANT

5.3.1 PROPERTIES. The primary coolant for the NBSR is heavy water. The freezing point of D_2O is 38.8°F and the boiling point is 214.5°F at atmospheric pressure. The pD of neutral heavy water is 7.35 as compared to a pH of 7.0 for neutral light water.

Impurities in the D_2O will be kept at low concentrations by filters and ion exchange beds of the purification system. Isotopic purity of the D_2O as received will be ≥99.75%. The principal sources of degradation will stem from refueling operations where the D_2O will be in contact with a light water vapor atmosphere, and during change of IX resin. System pressures are so adjusted that any rupture of heat exchanger tubes will result in D_2O out leakage into the light water system.

5.3.2 INVENTORY. Table 5.3-1 gives the calculated heavy water inventory for the D_2O systems.

Table 5.3-1 Heavy Water Inventory

PRIMARY SYSTEM	
Reactor Vessel	4043
Heat Exchanger, HE-1	1224
Piping, Valves, Strainers, and Pumps	2272
Sub Total	7539 Gallons
D_2O PURIFICATION SYSTEM	
Ion Exchangers	106
Heat Exchanger, HE-2	36
D_2O Storage Tank Sump	326
Piping, Valves and Filters	175
Sub Total	643 Gallons
THERMAL COLUMN TANK COOLING SYSTEM	
Tank	240
Piping and Valves	40
Sub Total	280 Gallons

Table 5.3-1 (con'd)

D_2O EXPERIMENTAL COOLING SYSTEM	
Piping, Pumps and Valves	200 Gallons
EMERGENCY COOLING SYSTEM	
Tank	3000
Piping and Valves	20
Sub Total	3020 Gallons
TOTAL PLANT INVENTORY	11,682 Gallons
	or 107,700 lbs. at 70°F

5.3.3 LEAKAGE RATES. Leakage rates of D_2O are not expected to be measurable because of the low temperature and pressure of heavy water systems. Maximum primary system temperature is expected to be approximately 112°F on the reactor outlet with a maximum pressure of 64 psig occuring at the discharge of the primary pumps.

The most probable sources of leakage would be at flanged connections, pump shaft seals, and shim blade shaft seals. Valves are of the diaphragm type, thus eliminating stem packing leakage.

Figure 5.7 is a typical arrangement of the shaft seals employed in all D_2O circulating pumps. A neoprene bellows with carbon wear ring is employed as an inner seal with the outer seal of teflon. The cavity between each seal is monitored by a control room indicator for possible leakage.

All major flanges have leak detectors which indicate in the control room. This leak detection system consists of a grid-type printed circuit sensing unit mounted at locations throughout the systems. If a leak should occur, the water will complete the electrical circuit and energize a relay. This relay will sound an annunciator. Then by means of selector switches the exact location of the leak may be determined.

All control valves and all manual valves 3" and greater have standard gasoline engine stainless steel spark plugs tapped into the valve bonnet. Refer to Figure 5.6 for a typical installation. In the event of a diaphragm rupture, the valve body fills with D_2O to complete the electrical circuit between contacts of the spark plug.

Other installations of leak detectors are employed in the cavity between heat exchanger tube sheets, in each shim arm shaft seal, and in the CO_2 system header beneath the reactor.

Shim arm shaft seals are discussed in Section 4.3 of this report.

5.3.4 RADIOACTIVE CONTAMINANTS. The calculated tritium concentration for the first 14 years operation is given in Figure 5.8 Since D_2O losses will be negligible, no makeup of heavy water was assumed in arriving at expected concentrations. Saturation activity was calculated to be 4230 μc/ml. Because of the low energy beta emission by tritium no shielding is provided. Tritium presents a biological hazard only when ingested or absorbed into the body.

Small quantities of fission products from fuel surface contamination and cladding transmission may also be found in the primary coolant, however, concentrations will be kept low by the purification system and through use of stringent fuel element cleaning procedures. Fuel element fabrication specifications limit surface uranium to less than 1 microgram/ft^2.

Na^{24} will also be found in the primary coolant resulting from corrosion and neutron interaction with aluminum. Concentrations are expected to be approximately 0.0317 $\mu c/cc^3$

using a purification flow rate of 25 gpm. It will not present a significant problem for personnel since concentrations will be reduced by a factor of approximately 16, using a 25 gpm flow rate, 12 hours after shutdown.

5.4 DESIGN CONSIDERATIONS

5.4.1 GENERAL. The primary coolant system is designed to transfer 10 Mw of heat from the core to the secondary cooling system while operating at a nominal flow of 5100 gpm with a reactor inlet temperature of 100°F and an outlet temperature of 112°F. A maximum pressure of approximately 64 psig will occur at the pump discharge.

5.4.2 PIPING AND FITTINGS. The piping and components of the system are designed for an internal pressure of 125 psig and a temperature of 150°F except that the heat exchanger tubes are designed for 100 psig at 200°F.

Piping 12" in diameter and smaller, is of 6061-T6 aluminum alloy conforming to ASTM Specification B241 and ASA Schedule 40. Piping larger than 12" may be seamless, conforming to ASTM Specification B235, Alloy 6061-T6 or from rolled plate with an electric fusion welded longitudinal seam.

Fittings are of 6061-T6 aluminum forgings conforming to ASTM Specifications B247. All fittings are butt welded type conforming to ASA Standard B16.9.

Flanges are forged, conforming to ASTM Specification B247, Alloy 6061-T6 and to Standard B16.5.

On straight runs of pipe, spacing of hangers does not exceed 10' on centers. Hangers are placed within 1' of each horizontal elbow. On piping 1-1/2" and smaller hangers are not more than 5' apart. Vertical runs of pipe not over 15' are supported by hangers not more than 12" from the elbows on connecting horizontal runs. All other vertical piping is supported by clamps which rest securely on the building structure.

5.4.3 INSPECTION AND TESTING

5.4.3.1 Heat Exchanger, HE-1. A mass-spectrometer leak test of the head welds, and gasket joints was performed using helium at 18 psig with a probe sensitivity of 0.0012 microns-ft^3/hr. No leaks were detected. Another test using 15 psig helium showed no leakage of the 1285 U-tubes.

5.4.3.2 Pumps. Each pump was hydrostatically tested at 200 psig for 30 minutes with no leakage observed. Pump casing and covers were tested at 263 psig for 30 minutes.

5.4.3.3 Piping. All welded joints were subjected to examination by radiograph and each length of pipe tested by the manufacturer to a hydrostatic test which produced in the pipe wall a stress of 4000 psi for the 14" and 16" sizes, and 5400 psi for the 18" and 20" sizes.

Upon ionclusion of installation, the entire primary system from FE-1, Flow Element 1, to the reactor inlet check valves was tested at 100 psig for 2 hours using light water without a loss in pressure. A 50 psig test of the system, including the reactor vessel, which was equipped with a special test head, was then conducted. After draining a helium leak test of 4 psig was conducted, with no leaks detectable.

5.4.3.4 Valves. All valves were shop tested to 150 percent of their design pressure or to a minimum of 185 psig for 30 minutes. In addition, major motor operated and air operated valves were radiographed and helium leak tested.

5.4.4 CORROSION ALLOWANCES. Since corrosion of 6061-T6 aluminum at NBSR operating conditions is not expected to exceed 0.032 mil/yr. and all piping and tanks have wall thicknesses in excess of 0.100 inches, corrosion allowance was not a design criteria. HE-1 tubes are Alclad to reduce corrosion attack.

5.5 REACTOR VESSEL

5.5.1 GENERAL DESCRIPTION. The reactor vessel contains the active core and core support

structure, the heavy water coolant tank, the control rods, and the experimental thimbles. The overall configuration of the vessel proper and the location of the various penetrations through the vessel walls are shown in Figure 4.1 and Figures 5.9 through 5.14. The size, number and purpose of each penetration are given in Table 5.5-1.

The NBSR vessel is an aluminum alloy vessel designed in accordance with the ASME Boiler and Pressure Code for Unfired Pressure Vessels, Section VIII, 1959 Edition, including all revisions, addenda, and applicable case rulings in effect at the time of the design.

The basic configuration of the vessel is a vertical cylinder with an eliptical cap at the bottom and a flange at the top. The vessel is supported from this top flange which rests on the shim ring of the cylindrical thermal shield surrounding the vessel. A nominal gap of one-inch is maintained between the reactor vessel and the thermal shield. The outer top plug assembly rests on the vessel flange, and thus clamps the vessel flange between itself and the thermal shield shim-ring. The outer plug assembly and the vessel flange are independently fastened to the thermal shield shim-ring.

A stainless steel "0" ring gasket forms a seal against helium or heavy water at the interface of the reactor top plug assembly and the reactor vessel flange (upperface). A second stainless steel "0" ring gasket forms a seal against carbon dioxide at the interface of the reactor vessel flange (lowerface) and the shim ring of the thermal shield.

Table 5.5-1 Reactor Vessel Nozzles and Penetrations

Purpose	No.	Size	Location
Shim Arm Nozzles	4	3-1/2" Diam.	Section C-C, Fig. 5.11
Cryogenic Facility	1	22"	Section D-D, Fig. 5.12
Horizontal Beam Ports 1A, 2A, 3A, 4A	4	6-1/2"	Section D-D, Fig. 5.12
Horizontal Beam Ports 1B, 1C, 1D, 2B, 2D	5	5-1/2"	Section D-D, Fig. 5.12
Rabbit Tubes	2	2-1/2"	Section D-D, Fig. 5.12
Rabbit Tubes	2	2-1/2"	Section E-E, Fig. 5.13
Through Tubes	4	6"	Section E-E, Fig. 5.13
Coolant Inlet	1	18"	Bottom Head
Coolant Outlet	2	12"	Bottom Head
Moderator Emergency Dump Line	1	10"	Bottom Head
Low Level Overflow	1	6"	Bottom Head

Table 5.5-2 NBSR Vessel Design Conditions

	Design	Actual
Pressure	50 psi	2" - 3" H_2O
Temperature	250°F	130°F (maximum wall temperature)

Table 5.5-2 compares the basic design conditions for the vessel with the anticipated operating conditions.

Coolant flow into the vessel is directed up through the bottom in two concentric inlet lines; through the fuel elements, and through the various flow paths in parallel with the fuel elements. The coolant leaves the vessel through two 12 inch outlet pipes located on the vessel bottom. The flows through the inner and outer inlet pipes yield nominal velocities of 6-1/4 ft./sec. and 8-1/2 ft/sec. respectively. The nominal velocity in each of the 12 inch outlet pipes is 8-1/2 ft./sec.

5.5.2 STRUCTURAL DESIGN. The vessel was designed in accordance with the ASME Boiler and Pressure Code for Unfired Pressure Vessels, Section VIII, 1959 Edition. In all cases the code allowable working stress for the materials and conditions considered was used as the design criteria and at no point in the vessel design were these stresses exceeded.

5.5.3 MECHANICAL AND STRUCTURAL LOADING. The mechanical and structural loadings considered in the design of the vessel consisted of the static loads resulting from component and coolant dead weights, the external and internal loads resulting from pressure differentials, the loads resulting from flow conditions in the primary system and the operation of the safety shim blade mechanisms.

5.5.3.1 Static Loads. The static loads imposed on the vessel are given in Table 5.5-3.

Table 5.5-3 Vessel Static Loads

Vessel Dead Weight (with inner D_2O reserve tank)	7000 lbs
Grid Plates	1080
Fuel Elements (24 at 23 pounds each)	552
Experimental Thimbles	465
Piping	580
Shim Safety Arms	220
D_2O Inventory at 70°F (540 ft.^{3} at 69 lb/ft.^{3})	37,260
Total	74,157 lbs

Design static load - 75,000 pounds.

5.5.3.2 Pressure. The reactor will be operated with a vessel cover gas pressure of 3 to 5 inches of water. The gas supply consists of standard high pressure bottles which feed through suitable pressure regulating manifolds to a closed recirculating system. This system, which includes the vessel and a helium gasholder, as well as all D_2O storage tanks, is protected by three safety valves set at levels of 6" H_2O, and 12.5 psi respectively, as well as a gasholder high level valve which will blow off the system whenever the gasholder diaphragm reaches the top of its travel. In the unlikely event that all regulating and relief systems failed concurrently, there is sufficient volume in the system to allow the bottled gas to expand with a regulating system pressure increase of less than 2 psi.

5.5.3.3 Piping Reactions. The vessel and its associated piping are free to move under the influence of thermal expansion. Only the reactions from the bellows type CO_2 seals are transmitted to the vessel. The major portion of all reactions resulting from primary system flow in the external piping are absorbed by sliding pad type pipe supports. The resulting loads on the vessel are small and in conjunction with all other loadings do not cause any

stress levels above the maximum allowable working stress for the various reactor sections.

5.5.3.4 Impact Loads. There are no impact loads transmitted to the vessel. The shim safety arm drive and shock absorbing systems are mounted on the biological shield so that only the extremely small reaction between the outer races and the balls of the safety arm bearings are transmitted to the vessel.

The pressure surges which might be generated in the NBSR by reactor power transients are small, and would not cause the vessel to exceed the 50 psi design pressure.

5.5.3.5 Seismic Loads. The vessel design was checked for a capability of withstanding forces resulting from horizontal accelerations of 0.1g. The combined stress levels resulting from this loading plus all other design loads were well within the allowable limits for the various vessel sections. This horizontal acceleration is in the range of an intensity VII to VIII earthquake on the modified Mercalli scale.

5.5.4 THERMAL LOADING CONDITIONS. During the course of the vessel design consideration was given to the loadings resulting from constraining forces or members, and from steady state and transient thermal conditions, including emergency conditions.

5.5.4.1 Steady-State Heating. The low heating rates experienced by this vessel and the excellent thermal conductivity of the aluminum combine to yield negligible stresses from internal temperature gradients.

Radiation Heating, Section 4.7.8, will result in a calculated vessel wall temperature of approximately 130°F at full power operation. All allowable stresses were chosen at 250°F.

Areas of distinct interest from the standpoint of thermal expansion are the grazing tube to shell joints and the grazing tube column reactions resulting from end restraint. Both these areas have been investigated and the resulting stresses considered in the vessel design. These loadings do not cause the code allowable working stresses to be exceeded at any point.

The NBSR vessel is fabricated entirely of aluminum alloys. Stresses resulting from differential expansion between dissimilar materials are therefore negligible.

5.5.4.2 Thermal Transient Loading Conditions. The very small temperature differentials between the coolant and vessel components result in thermal transient loadings which are insignificant.

5.5.5 NEUTRON IRRADIATION EXPOSURE. The beam tube tips experience the maximum radiation exposure. The neutron exposure of the tips presented in Table 5.5-4 is based on the calculations discussed in Section 4.6.2. A 90% load factor was assumed and the time-integrated-flux (nvt) for 1 year and a 20 year vessel life was computed for two neutron energy groups, 0-1 Mev and >1 Mev. The bulk of the neutrons in the 0-1 Mev group are of thermal energy and the remainder have an average energy of only a few hundred Kev. Thus, the fast group, although containing fewer neutrons, represents the greater radiation damage potential.

Table 5.5-4 Reactor Vessel Maximum Neutron Exposure

Energy Group	Flux, n/cm^2-sec	nvt/yr	nvt/20 yrs
0-1 Mev	1.3×10^{14}	3.7×10^{21}	7.4×10^{22}
> 1 Mev	6×10^{11}	1.7×10^{19}	3.4×10^{20}

The results of irradiation damage studies run to date on various aluminum alloys show that the changes in the engineering properties of these materials are not significant from

the standpoint of the NBSR vessel design. A program of aluminum alloy irradiation being conducted at the ETR (5.1) is employing irradiation levels and temperature levels that fit the NBSR conditions extremely well. To date the program has exposed the alloy samples to a fast neutron exposure of 1.17×10^{21} nvt (> 1 Mev) at a temperature of 150°F, close to the calculated maximum temperature of the NBSR beam tube tips. The results of this program indicate a 10 to 20% increase in the yield and ultimate strength of these alloys and very small changes in the ductility (a slight decrease in the case of 5052-0 aluminum) when exposures comparable to the NBSR conditions have heen reached.

5.5.6 CORROSION. Since this is an all aluminum system, and since the flow rates in the vessel and associated piping are small, only aluminum corrosion was considered in the vessel design. A 1/8 inch minimum (125 mil) corrosion allowance was allowed on all pressure containment surfaces of the vessel. The maximum predicted corrosion for the design temperature of 250°F and coolant pH of 5 would be $.87 \times 10^{-4}$ mils/day or approximately .7 mils for a twenty year design life. The corrosion rate is extremely temperature dependant. This corrosion rate is quite conservative then, since the maximum predicted wall temperature is 130°F. Furthermore, this 1/8" allowance was applied to the calculated skin thickness required to keep stresses below the allowable working stress and then the commercial plate thickness equal to or greater than the sum of these two was specified. This procedure leads to at least 50% increase in the resulting actual corrosion allowance.

5.5.7 MATERIALS OF CONSTRUCTION. The materials used in the various parts of the vessel are given below in Table 5.5-5.

Table 5.5-5

Component	Form	Type	ASTM Specification
Vessel Shell	Plate	5052-0	B209-59T
Vessel Bottom Head	Plate	5052-0	B209-59T
Supporting Flange and Hub	Forging	6061-T6	B247-59T
Nozzles-Bottom Head	Forged	6061-T6	B247-59T
18 Inch Outer Funnel	Plate	5052-0	B209-59T
Beam Port Penetration Thimbles	Forged	6061-T6	B209-59T
Cryogenic Port Cylindrical Section	Forged Billet	6061-T6	B247-59T
Head End Section	Plate	5052-0	B209-59T
Grazing Tubes	Tube	6061-T6	B210-59T
Shim Safety Balde Nozzles	Forged Billet	6061-T6	B247-59T

5.5.8 FABRICATION, INSPECTION AND TESTING. The vessel was fabricated to meet the requirements of National Bureau of Standards Reactor Specifications, Section 92, Reactor Vessel Unit, by Dixie Manufacturing Company, Baltimore, Maryland. In-process inspection was carried out by the NBS reactor staff, as well as by the manufacturer's inspectors. Prior approval of fabrication and assembly procedures were required at all stages of the job.

All personnel and equipment to be used on the welding of the vessel were qualified in accordance with the ASME Code, Section IX.

All welding on the vessel was done by the inert gas consumable electrode metal arc welding process and the inert gas shielded metal arc welding process with tungsten electrode.

All completed pressure containing welds received a 100 percent radiographic inspection. Any weld areas showing any discontinuities were removed and rewelded to meet specifications. Visual or radiographic evidence of undercutting, icicles, lack of penetration and slugging were also cause for rejection of the weld area involved. All welds were brought to specifications before acceptance of the vessel.

Before the shop hydrostatic test took place, the vessel was instrumented with SR-4 type strain gages. These gages were mounted at points where critical values of strain would be expected to occur. The fifty points chosen included all shell and penetration welds, all areas of variation in shell thickness or configuration and samples of each point on the shell which was reinforced by internal rings or brackets.

Analysis of the strain gage data showed that the very small local inelastic strains which occurred were typical of the stress relief process seen in a ductile material. The maximum tensile and compressive stresses recorded were .0013 in/in. and .0006 in/in. respectively.

The vessel was hydrostatically tested in accordance with the ASME Code. The fully assembled vessel was tested in the horizontal position with water at 68°F temperature and 75 psig pressure. This maximum pressure was developed incrementally for purposes of the strain gage readings and then held for one hour during which all welds and surfaces were checked for leaks. Two small "weeps" were located during the examination. The vessel was depressurized, the "weeps" repaired and the repairs radiographed. The entire hydrostatic test was then repeated. No leaks were found during the second test.

After installation of the vessel, including all connecting welds, a 50 psig hydrostatic pressure test was run. The vessel was capped, vented, filled to a zero void condition and then pressurized to 50 psig along with the rest of the primary system. This pressure was held for several hours while inspection teams searched for leaks. No evidence of leaks was found at any point on the vessel.

Reference

(5.1) M. J. Graber and J. H. Ronsick, "ETR Radiation Damage Surveillance Program, Progress Report I," IDO - 16628 (Jan. 1961).

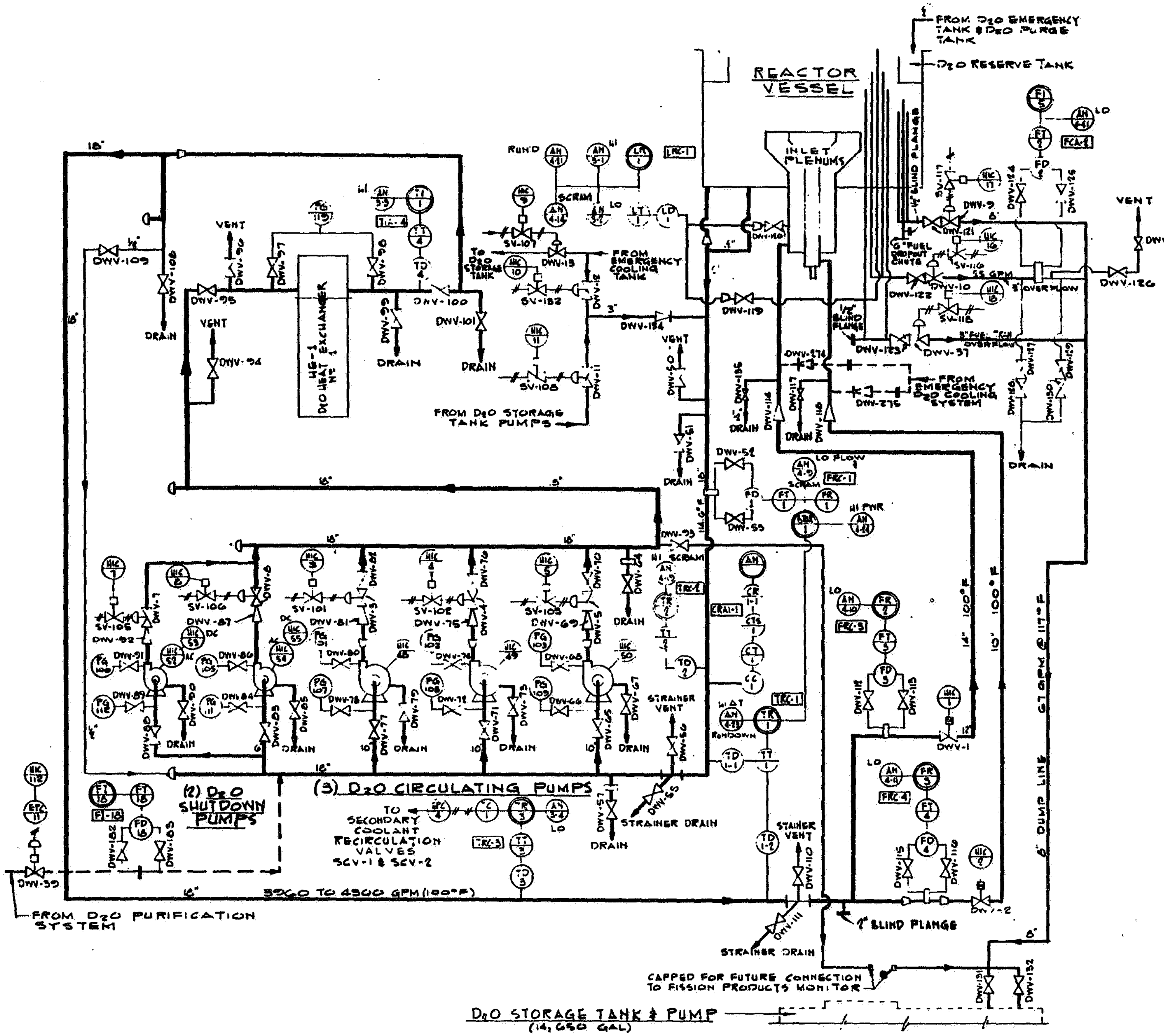


Figure 5.1 Primary D_2O cooling system.

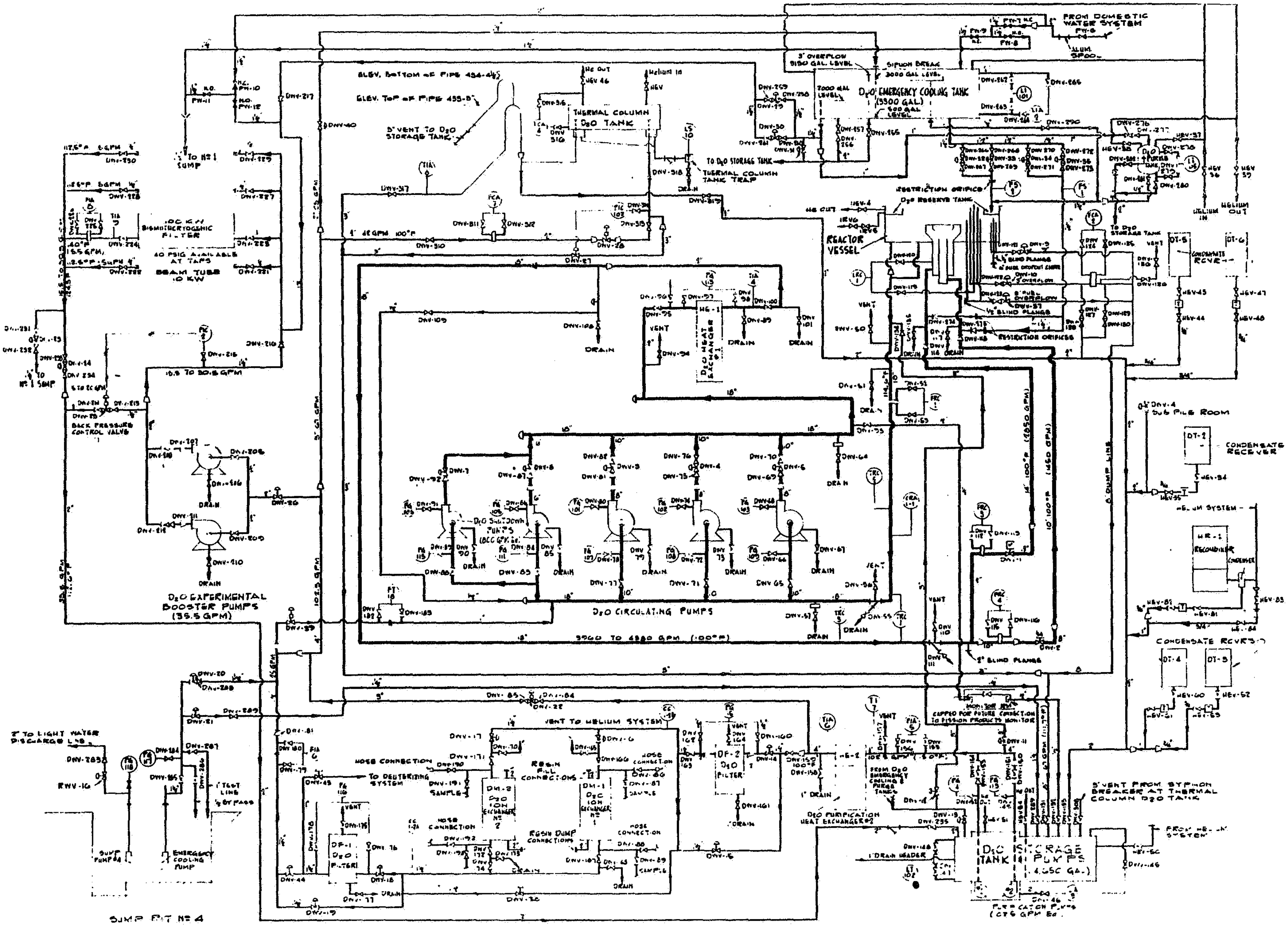
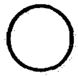
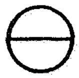
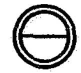


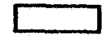
Figure 5.2 D_2O system flow diagram.

 Indicator or Detector - Local

 Indicator, Alarm Switch, or Recorder - Remote

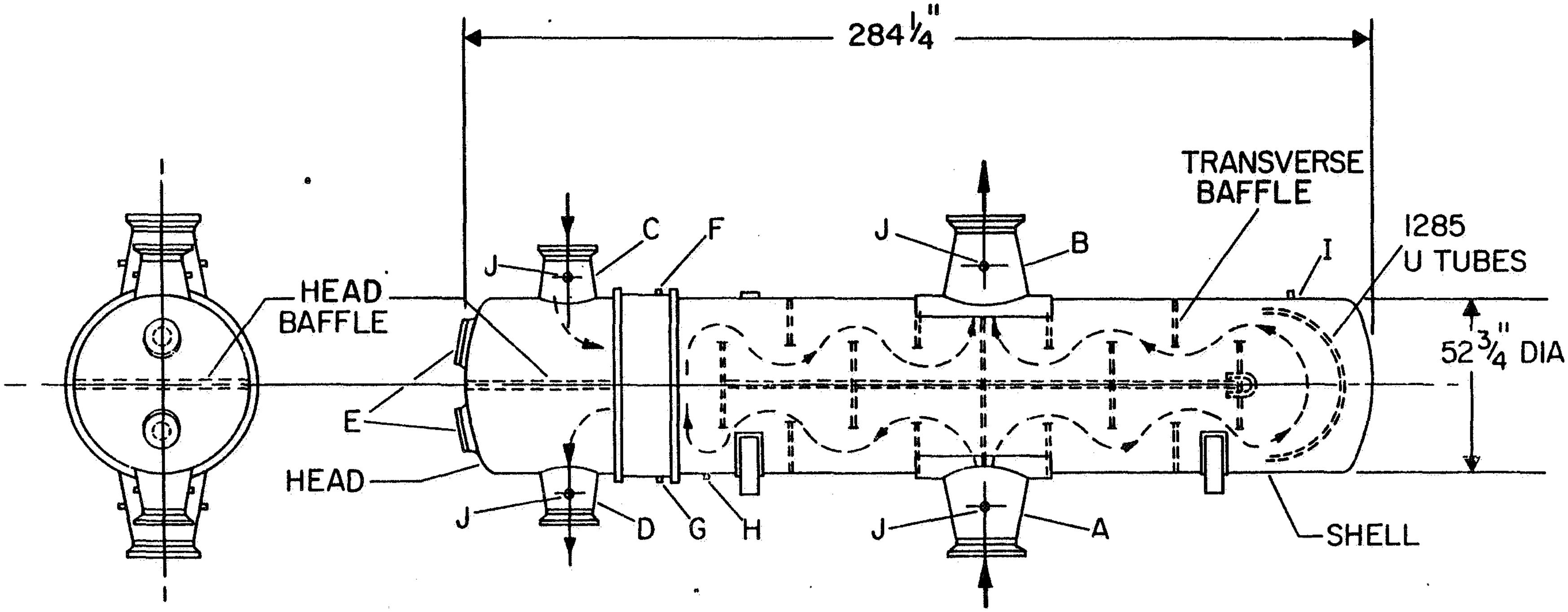
TRC	Temperature Recorder Controller
TIA	Temperature Indicator Alarm
LIA	Level Indicator Alarm
TD	Temperature Detector
CRA	Conductivity Recorder, Alarm
CA	Conductivity Alarm
CC	Conductivity Cell
CI	Conductivity Indicator
CR	Conductivity Recorder
CS	Conductivity Switch
FA	Flow Alarm
FD	Flow Detector
FC	Flow Controller
FI	Flow Indicator
FR	Flow Recorder
FS	Flow Switch
FT	Flow Transmitter
SFA	Scram Flow Alarm
LA	Level Alarm
LC	Level Controller
LI	Level Indicator
LR	Level Recorder
LS	Level Switch
LT	Level Transmitter
SLA	Scram Level Alarm
PA	Pressure Alarm
PC	Pressure Controller

 Indicator or Recorder - Panel Mounted

 Channel Number

PG	Pressure Gauge
PIC	Pressure Indicator Controller
PR	Pressure Recorder
PS	Pressure Switch
PT	Pressure Transmitter
SPA	Scram Pressure Alarm
TA	Temperature Alarm
TC	Temperature Control
TI	Temperature Indicator
TIC	Temperature Indicator Controller
TR	Temperature Recorder
TS	Temperature Switch
TT	Temperature Transmitter
STA	Scram Temperature Alarm
BTUR	BTU Recorder
FRC	Flow Recorder Controller
LIC	Level Indicator Controller
FIT	Flow Indicator Transmitter
HIC	Hand Control Indicator
EPC	Electro Pneumatic Converter
FIC	Flow Indicator Controller
FCA	Flow Control Alarm
FIA	Flow Indicator Alarm
CIA	Conductivity Indicator Alarm
PRC	Radiation Recorder & Controller
PI	Pressure Indicator
AN	Annunciator

Figure 5.3 Instrument symbols.



SCHEDULE OF CONNECTIONS

A SHELL SIDE INLET - SECONDARY H_2O COOLING
B SHELL SIDE OUTLET - SECONDARY H_2O COOLING
C TUBE SIDE INLET - PRIMARY D_2O COOLING
D TUBE SIDE OUTLET - PRIMARY D_2O COOLING
E HAND HOLE INSPECTION PORTS
F BETWEEN TUBE SHEETS VENT
G BETWEEN TUBE SHEETS DRAIN
H SHELL SIDE DRAIN
I SHELL SIDE VENT
J TEMPERATURE & PRESSURE TAPS

Figure 5.4 NBSR heat exchanger.

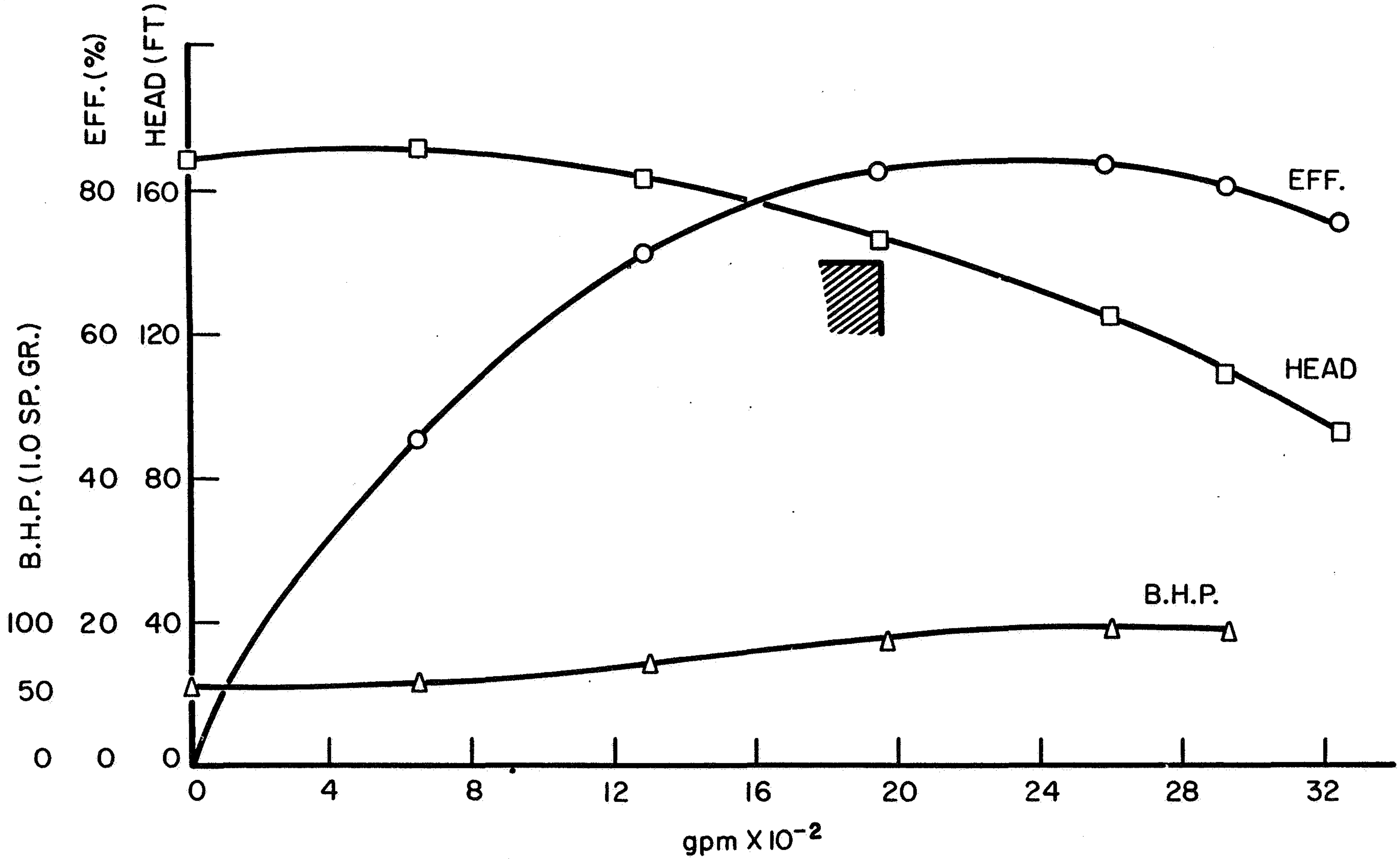


Figure 5.5 Primary pump performance curves.

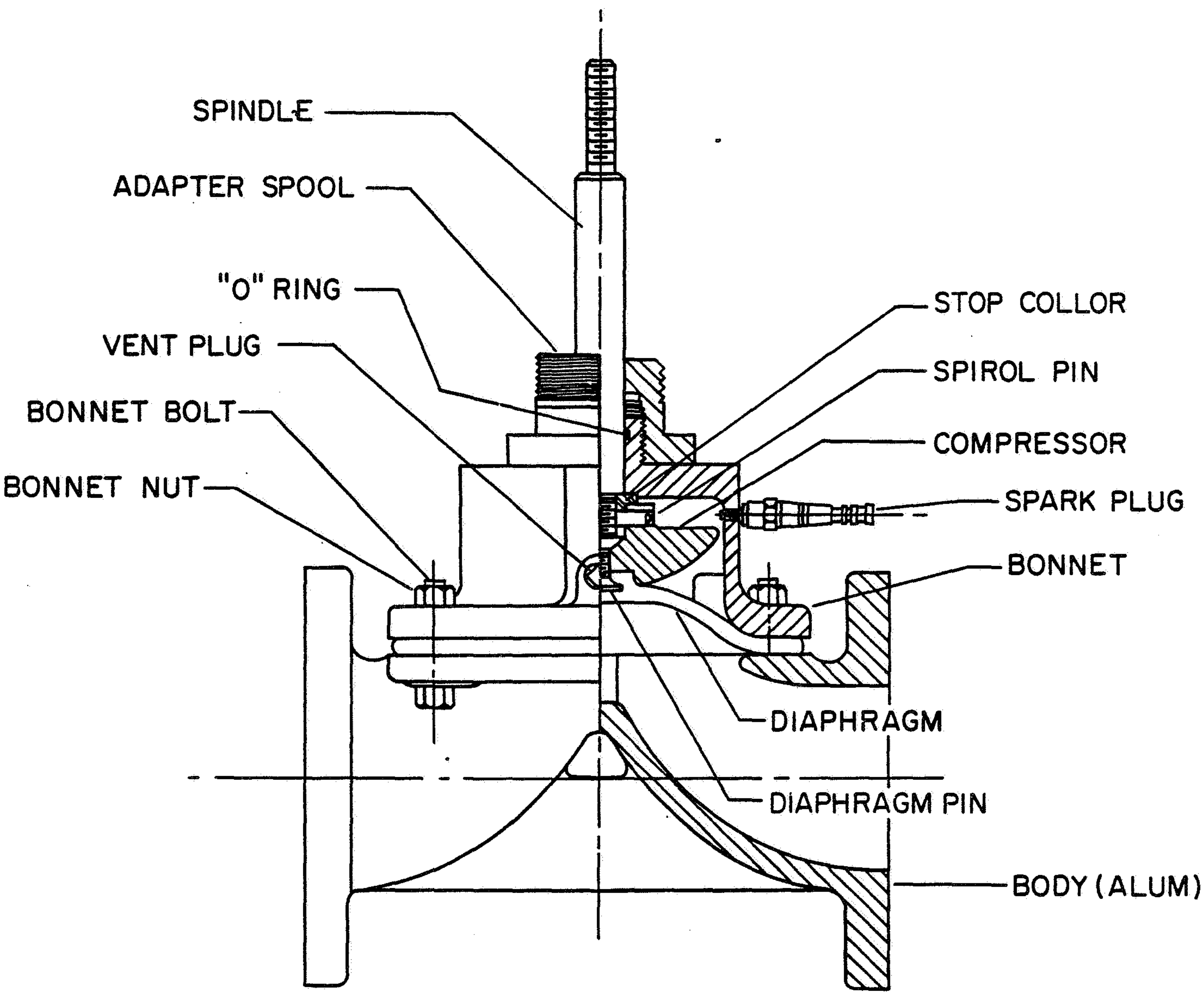


Figure 5.6 Valve cross section.

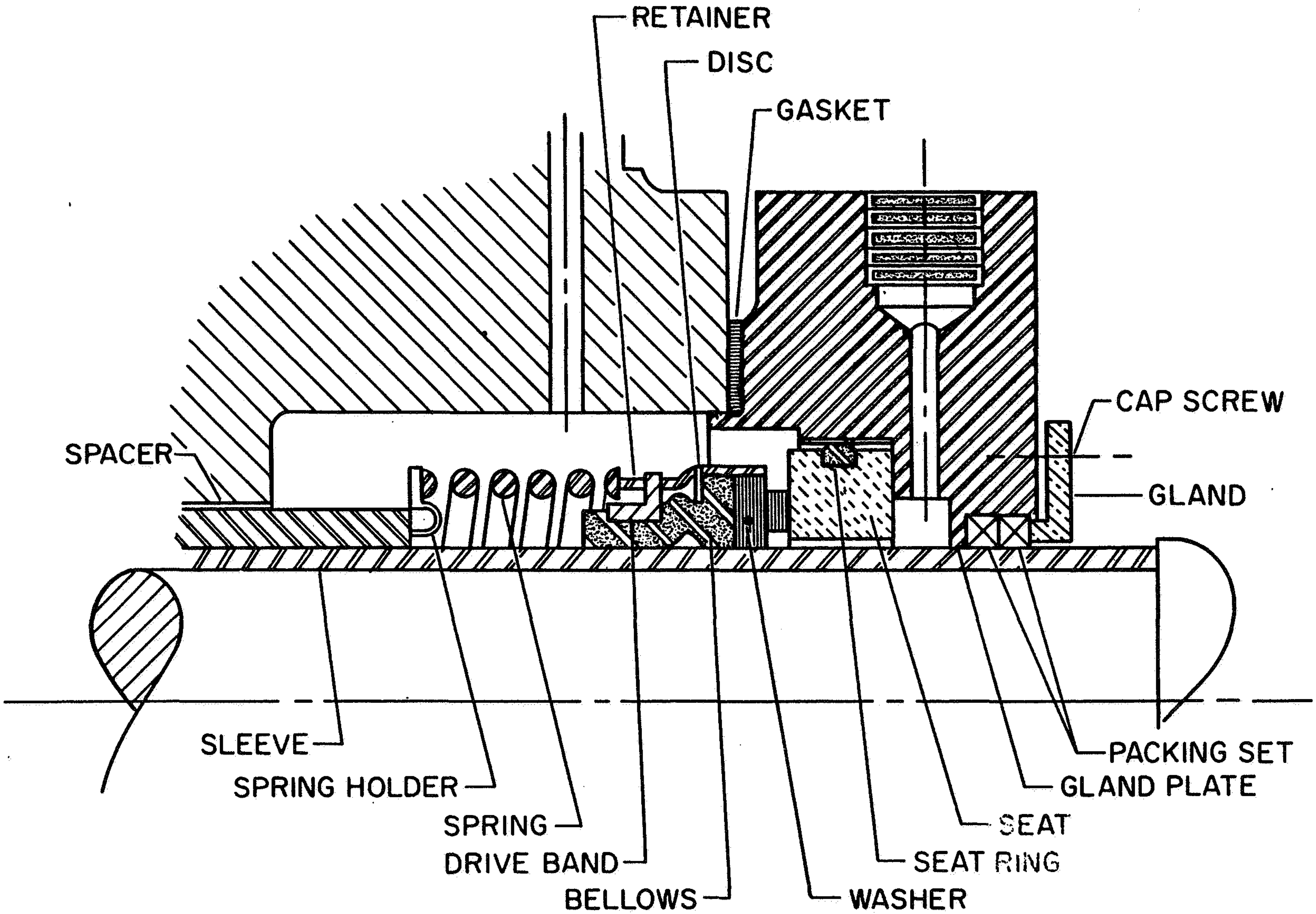


Figure 5.7 Primary pump backing.

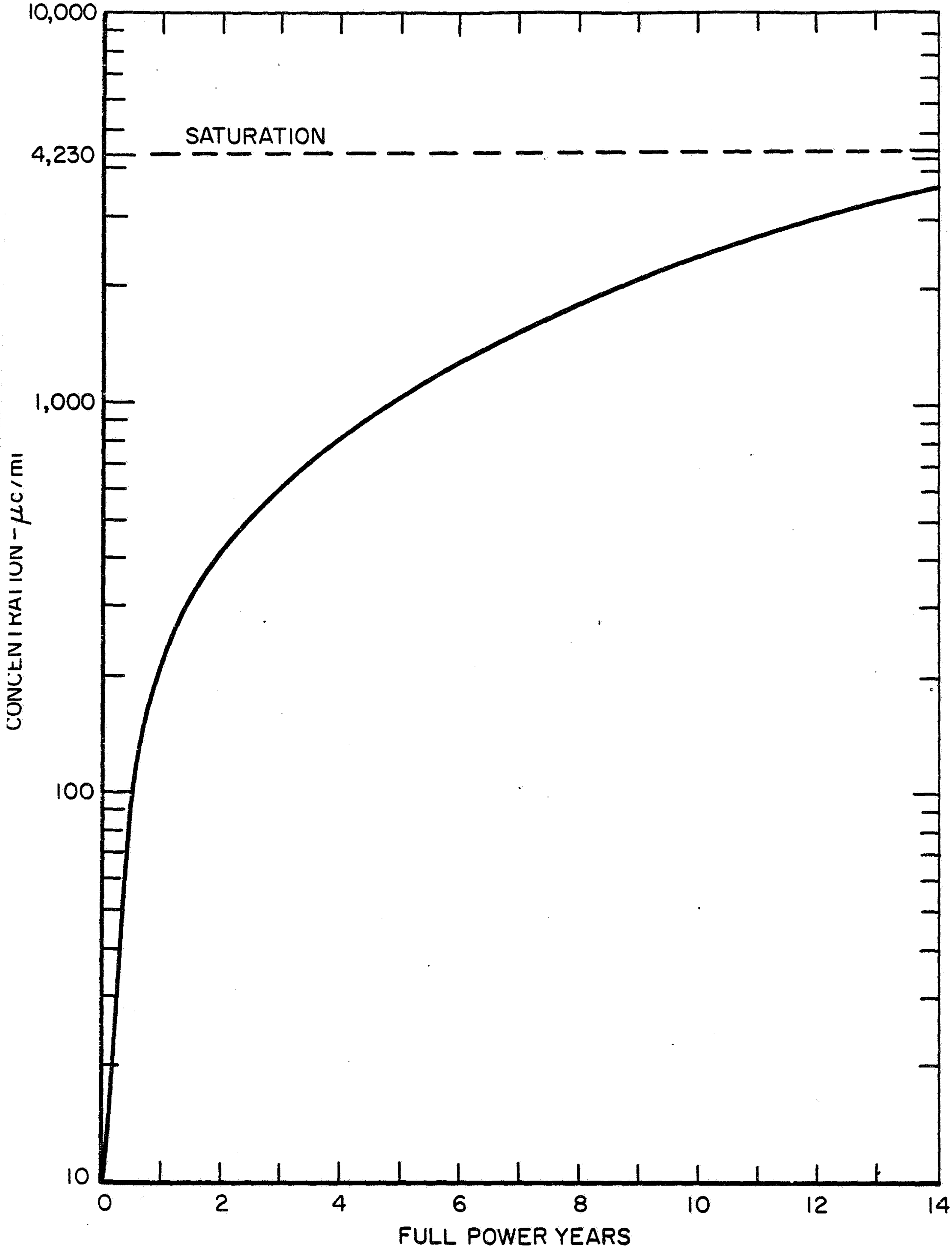


Figure 5.8 Expected tritium concentration in D_2O systems versus full power years.

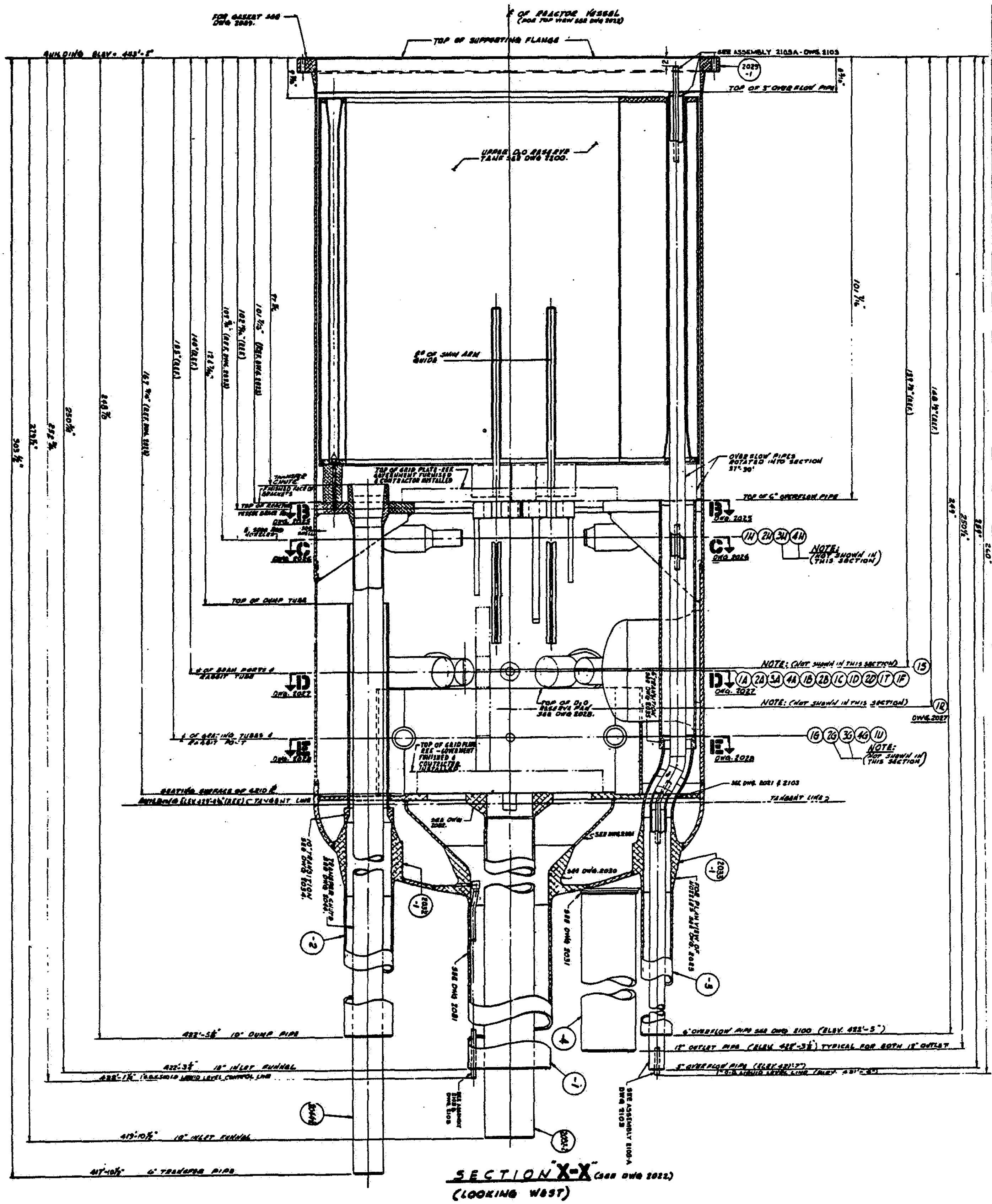


Figure 5.9 Reactor vessel section (B & R 2020).

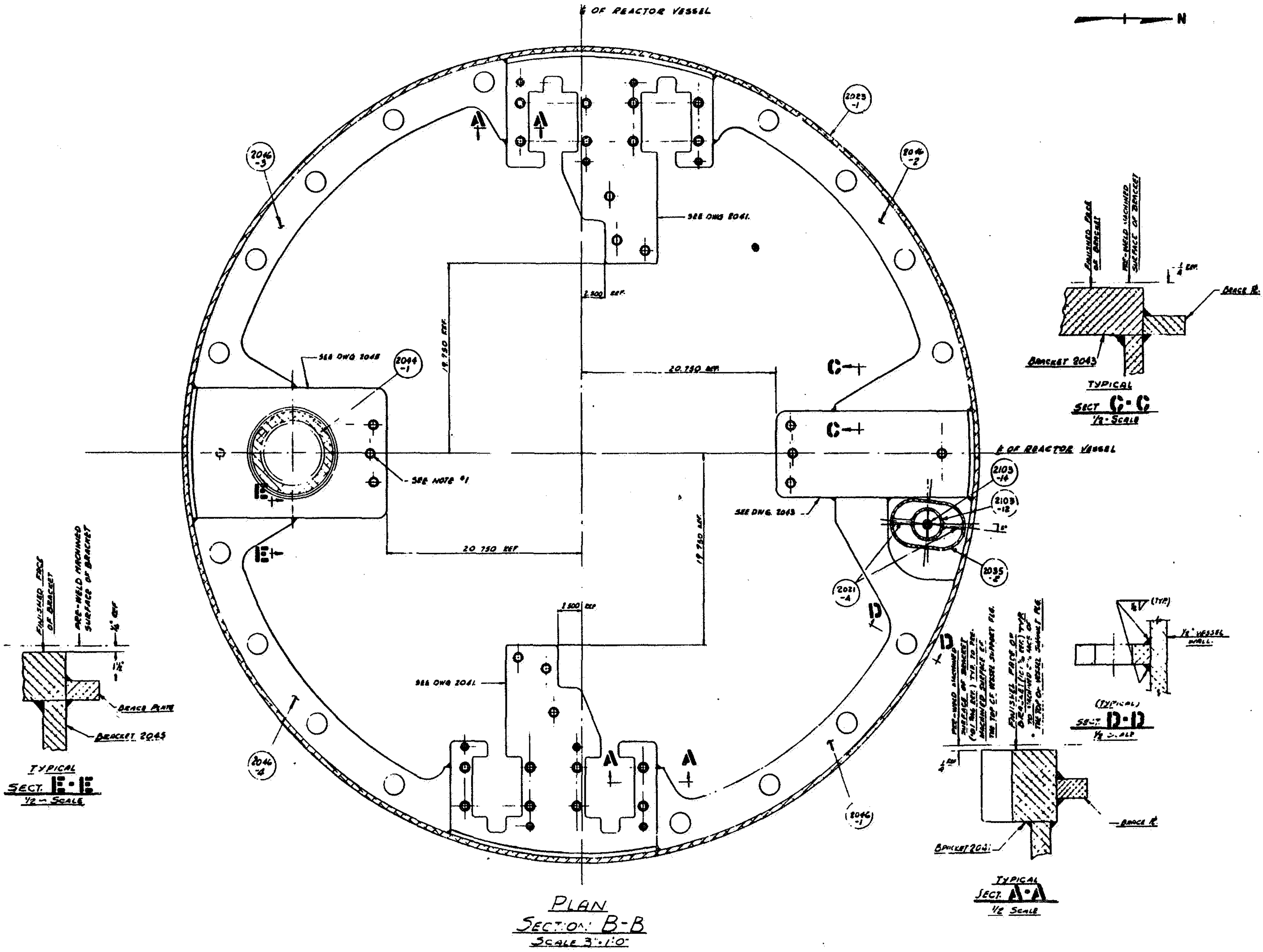


Figure 5.10 Reactor vessel section BB (B & R 2025).

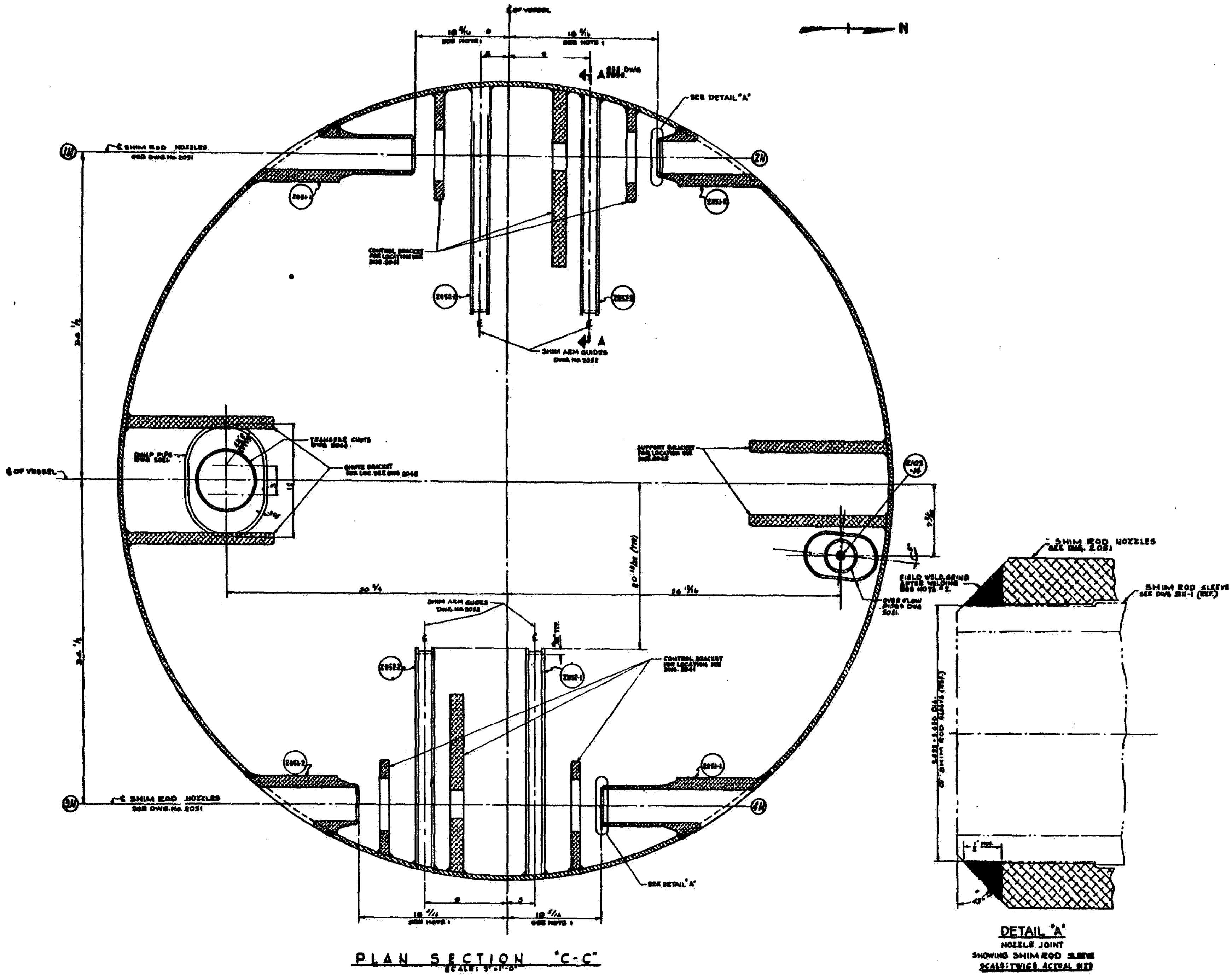


Figure 5.11 Reactor vessel section CC (B & R 2026).

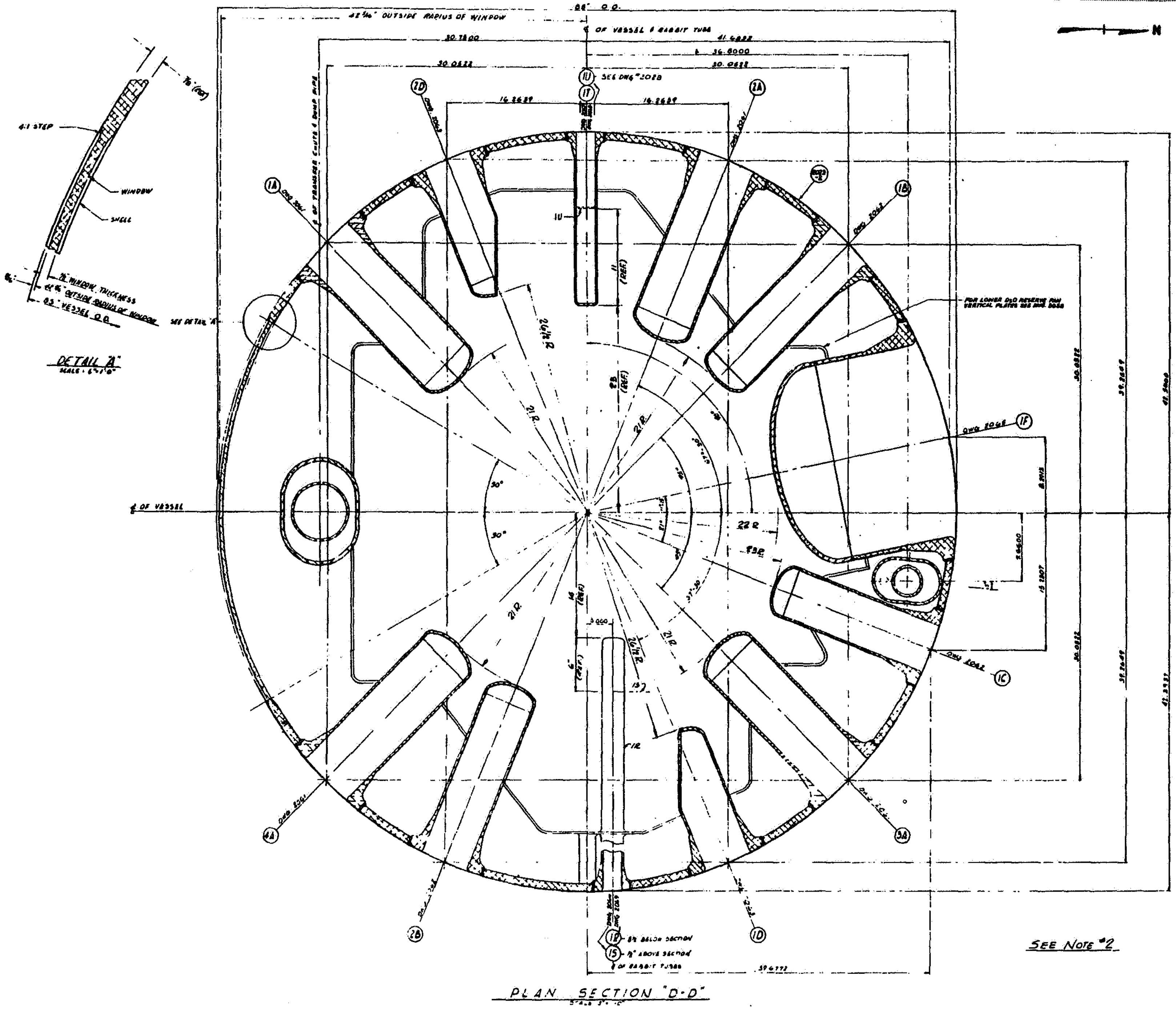


Figure 5.12 Reactor vessel section DD (B & R 2027).

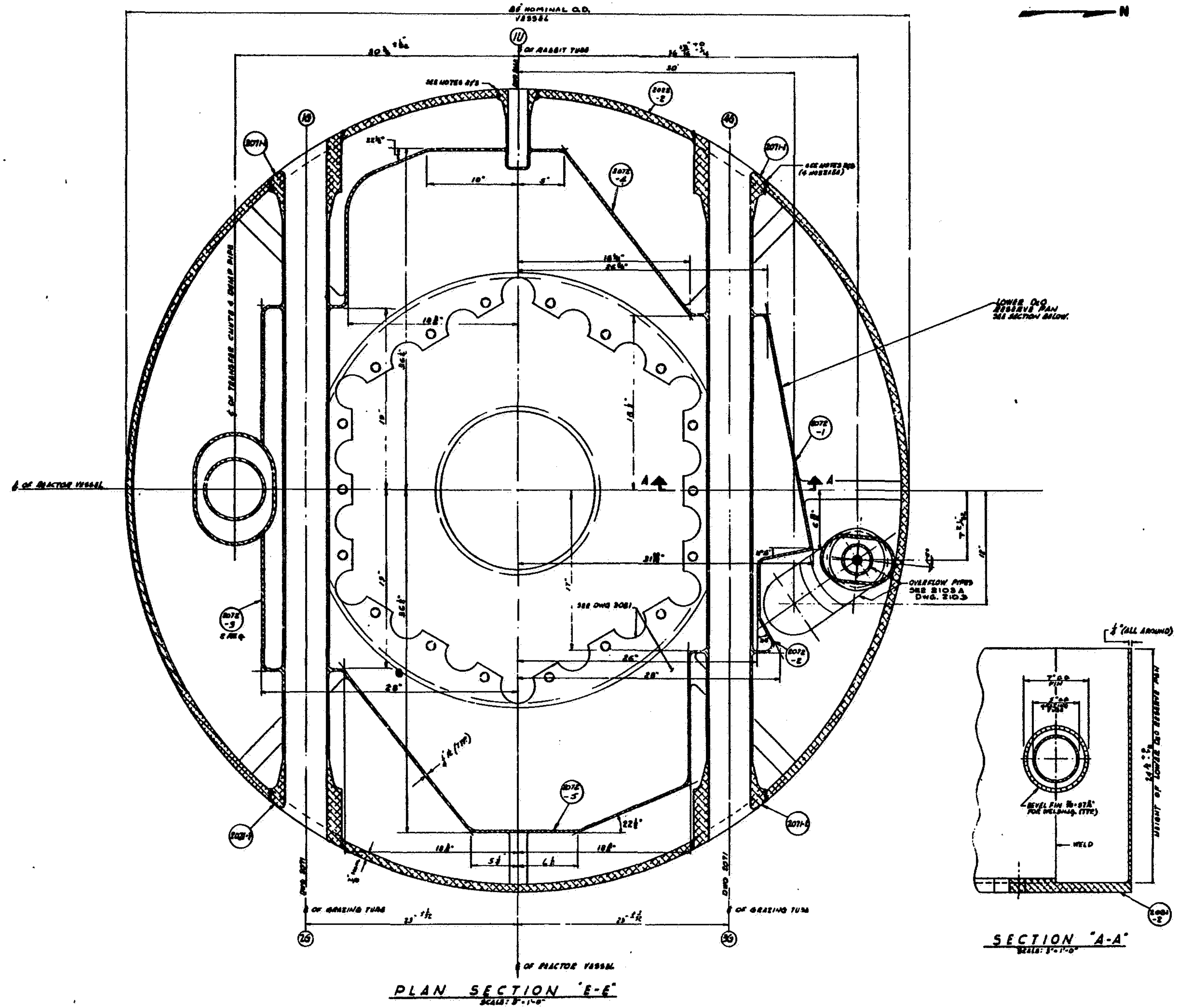


Figure 5.13 Reactor vessel section EE (B & R 2028).

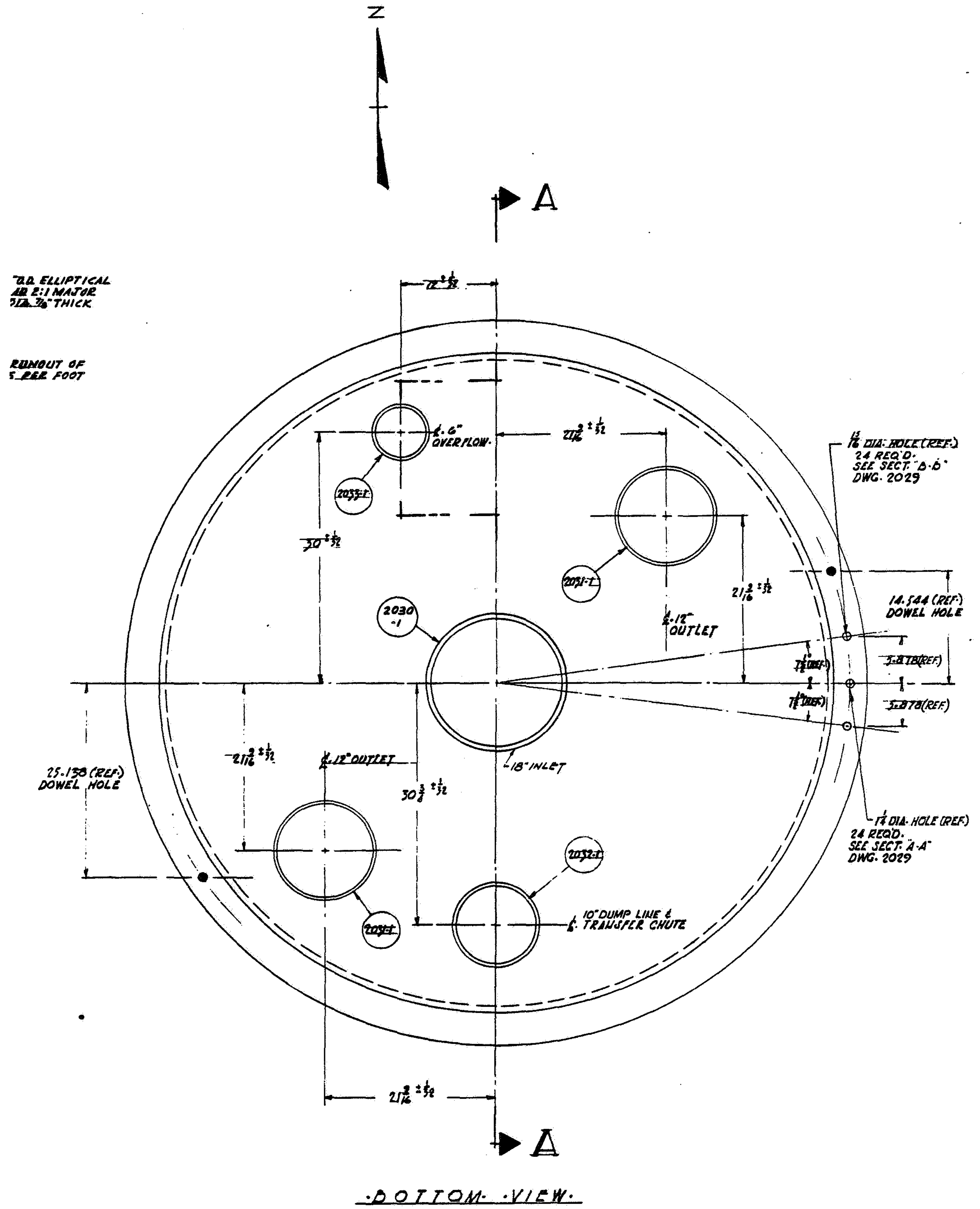


Figure 5.14 Bottom view (B & R 2023).

SECTION 6. SECONDARY COOLING SYSTEM

6.1 GENERAL DESCRIPTION

Secondary light water coolant is circulated by three main secondary coolant pumps arranged in parallel. These pumps supply coolant to the shell side of the primary heat exchanger, the D_2O purification heat exchanger, the thermal shield heat exchanger, and to the tube side of the demineralized water experimental heat exchanger. Upon leaving the heat exchangers a sample of the water passes through a radiation detector. Figure 6.1 illustrates the location of this monitor and other equipment.

Water then enters the suction of three booster pumps and a shutdown cooling pump arranged in parallel. Pump discharge flow is measured by an installed flow element. The flow is then directed to the cooling tower or partially bypassed through valves SCV-1 and SCV-2, depending on cooling requirements as established by primary heat exchanger outlet temperature as sensed by TIC-1.

The cooling tower is an induced draft type which cools the water by evaporation. As the water cascades down the cooling tower, air is drawn through the tower by the fans. The cooled water collects in the basin and provides a head of water to the suction of the main coolant pumps, thus completing the circuit.

The secondary cooling system losses, due to evaporation and leakage, are automatically made up from the domestic water system by valve SCV-5. Cooling tower sump level controls SCV-5, so that an adequate head of water is maintained for the coolant pumps.

There is a separate, automatically-controlled, chemical addition system to control the pH of the secondary water. The pH control treatment used is sulfuric acid.

Fifty gpm is circulated through an experimental cooling loop which provides additional cooling to the cryogenic facility, grazing tubes, vertical thimbles, and the beam ports. This system utilizes the cooling tower to remove its heat, but otherwise operates independently of the secondary system. A detailed discussion is given in Section 7.2.4 of this report.

Operation of the number of main circulating and booster pumps depends on plant heat removal requirements and air temperature at the cooling tower. The shutdown cooling pump is operated to remove decay heat from the primary system.

6.2 SYSTEM COMPONENTS

6.2.1 MAIN SECONDARY COOLING PUMPS. There are three identical 2828 gpm (28 psig head) single-stage, centrifugal pumps which circulate the coolant from the cooling tower to the heat exchanger. These pumps are controlled from the control room and may be run in any combination to achieve the flow required for a particular power level, or to maintain a specified rate of cooling dependent on the air temperature at the cooling tower.

The pumps are driven by 60 hp, 480 volt, 3 phase 6 cps motors. Pumps No. 1 and No. 3 are supplied from motor control center MCC-A7 and Pump No. 2 from MCC-B8.

6.2.2 SECONDARY COOLING BOOSTER PUMPS. Three 2828 gpm (28 psig head) single-stage, centrifugal pumps circulate the coolant from the heat exchanger to the cooling tower. These pumps are also controlled from the control room and may be run in any combination. These pumps are identical to the main pumps, except they are driven by 50 hp, 480 volt, 3 phase 60 cps motors. Pumps No. 1 and No. 3 are supplied from motor control center MCC-A7 and pump No. 2 from MCC-B8.

6.2.3 SECONDARY COOLING SHUTDOWN PUMP. One shutdown pump provides cooling to remove the decay heat from the primary system during periods when the reactor is shut down. It is controlled from the control room and is interlocked with the main pumps so that it will

start automatically upon loss of the main pumps if its control switch is in the "AUTO" position. It is also interlocked with SCV-6, main secondary coolant pump bypass valve, so that this valve will open when the pump is started. This pump is driven by a 3 hp motor and is rated at 125 gpm with a discharge pressure of 22 psig. Power is supplied to it from emergency power motor control center, MCC-A5.

6.2.4 COOLING TOWER. The cooling tower is a Foster-Wheeler design, induced-draft two-cell unit. The water is cooled by pumping it to the top of the tower and allowing it to spill down louvers. As the water cascades down, air is drawn across it by the fans which accelerate the evaporation process. Each cell has a wetted surface of 98,250 sq. ft. and a capacity of 5707 gpm. The fans are driven by 40 hp motors at 254 rpm, and have a capacity of 337,000 cfm each. The motor, drive gear, and fan assembly are protected by a vibration cut-out switch to prevent excessive wear from an unbalanced load due to icing of the blades in cold weather. The gear train is provided with a heater to warm the lubricant before starting in cold weather. In the event that the louvers freeze up, the fans can be reversed to discharge warm air through the tower for de-icing. The fans are powered from motor control centers MCC-A7 and MCC-B8.

The tower sits on a concrete basin which is 22 feet by 49.3 feet by 5 feet deep and has a capacity of 32,500 gallons. The basin is equipped with heating steam to prevent ice from forming in the basin during cold weather. The basin overflows into a sump which provides a suction for the coolant pumps. Each cell is rated for 5 Mw and the manufacturer recommends operation in the range of 80 to 112 percent for peak efficiency.

6.2.5 ASSOCIATED VALVES AND PIPING. Piping throughout the system is carbon-steel, except in the chemical addition system. Main piping is 20-inch, auxiliary piping to HE-2, HE-6, and HE-7 is 5 inches, and experimental cooling piping is 2 inches in diameter. The cooling tower basin has a 4 inch drain to grade. Table 6.2-1 lists information on the operation of control valves in the system.

Table 6.2-1 Secondary Cooling System Air-Operated Control Valves

Valve No.	Size (Inch)	Diaphram Operation	Operator Tag No.
SCV-1	6	ATO	EPC-4
SCV-2	6	ATO	EPC-4
SCV-3	2	ATO	PIC-9
SCV-4	2	ATO	TIC-19
SCV-5	2	ATO	EPC-5
SCV-6	4	ATO	EPC-23

6.2.6 INSTRUMENTATION

6.2.6.1 General. Instrumentation necessary to monitor secondary cooling water flow temperatures of the cooling water at various locations in the loop, pH of the circulating water, radiation level of the water, level of the cooling tower basin and pressures at various points in the system are provided.

6.2.6.2 Flow. FIA-12 flow channel, senses, monitors, and indicates flow in the secondary coolant system from the discharge of the booster pumps. The channel has a range of 0 to 6000 gpm.

6.2.6.3 Temperature. TD-14 senses secondary cooling water temperature at the outlet of the main heat exchanger. Local temperature indicators are also provided on the other three heat exchangers served by the system.

TD-13 measures the temperature of the cooling tower basin. Indication is on control board mounted, TI-8. The range is set at 25-125°F with a low temperature alarm at 40°F.

6.2.6.4 pH. pHT-1 measures the pH of the secondary coolant. Local recording is provided by a recorder-controller mounted in the secondary pump room. Based on a manual set point this controller regulates the operation of the acid metering pump to inject dilute sulfuric acid as needed to maintain a slightly acidic system with a pH of 6.0 to 6.5.

6.2.6.5 Pressure. All pressure measurements are made with local Bourdon-tube type pressure gages. Suction and discharge pressure of all pumps are sensed as well as the inlet pressures of the three small heat exchangers.

6.2.6.6 Radiation Monitor. RD-3-1 measures any radioactivity in the secondary coolant that may be present from a primary to secondary leak in the heat exchangers. The radiation monitor records and alarms in the control room.

6.2.6.7 Level. LT-9 measures the level of the cooling tower and controls valves, SCV-5, to maintain an adequate head of water for the main secondary coolant pumps. In addition, it provides a high-level alarm and a low level alarm in the control room. Its range is fixed at 0 to 50" H_2O, with alarm set points to be determined.

6.3 COOLANT

6.3.1 MAKE-UP WATER. Water used as make-up to the secondary cooling system is supplied by the domestic water system of the National Bureau of Standards from the Washington Suburban Sanitary Commission. A typical analysis of this water is given in Table 6.3-1.

Evaporation in the cooling tower tends to concentrate solids in the secondary water. In order to permit a continuous blowdown, the rate of make-up to the tower must exceed the evaporation rate. The blowdown water is directed to the pumphouse sump, then to the sewer, thus preventing excessive build-up of dissolved solids.

Table 6.3-1 Domestic Water Analysis

Component	PPM	pH
Ca^{++}	32.2	
Mg^{++}	6.9	
Na^{+}	8.8	
HCO_3	78	
$SO_4^{=}$	40.9	
Chlorides	13.2	
NO_3	0.46	
F^-	0.98	
CO_2	2.0	
$Si\ O_2$	6.3	
Fe	0.03	
		7.6

6.3.2 CORROSION CONTROL. A chemical addition system is provided so that sulfuric acid may be added to control the pH of the water. The system consists of a 75 gallon acid storage tank, a 1000 gallon acid mixing tank, an injection pump, one-air ejector, and interconnecting poly-vinyl-chloride piping.

The sulfuric acid is received in 50 gallon drums in a concentrated solution of 66° Be (98%). The acid is siphoned from the shipping drum to the storage tank by the air ejector. It is held in the storage tank until needed, at which time it is measured into the water in the mixing tank for addition to the system. The diluted solution is held in the mixing tank until it is metered into the coolant stream by the injection pump. The injection pump is controlled by the in-line pH sensor; thus, the pH of the system is automatically controlled. Make-up water to the mixing tank is supplied from the domestic water system.

A commercial, low chromate treatment is also added to the secondary water for corrosion control.

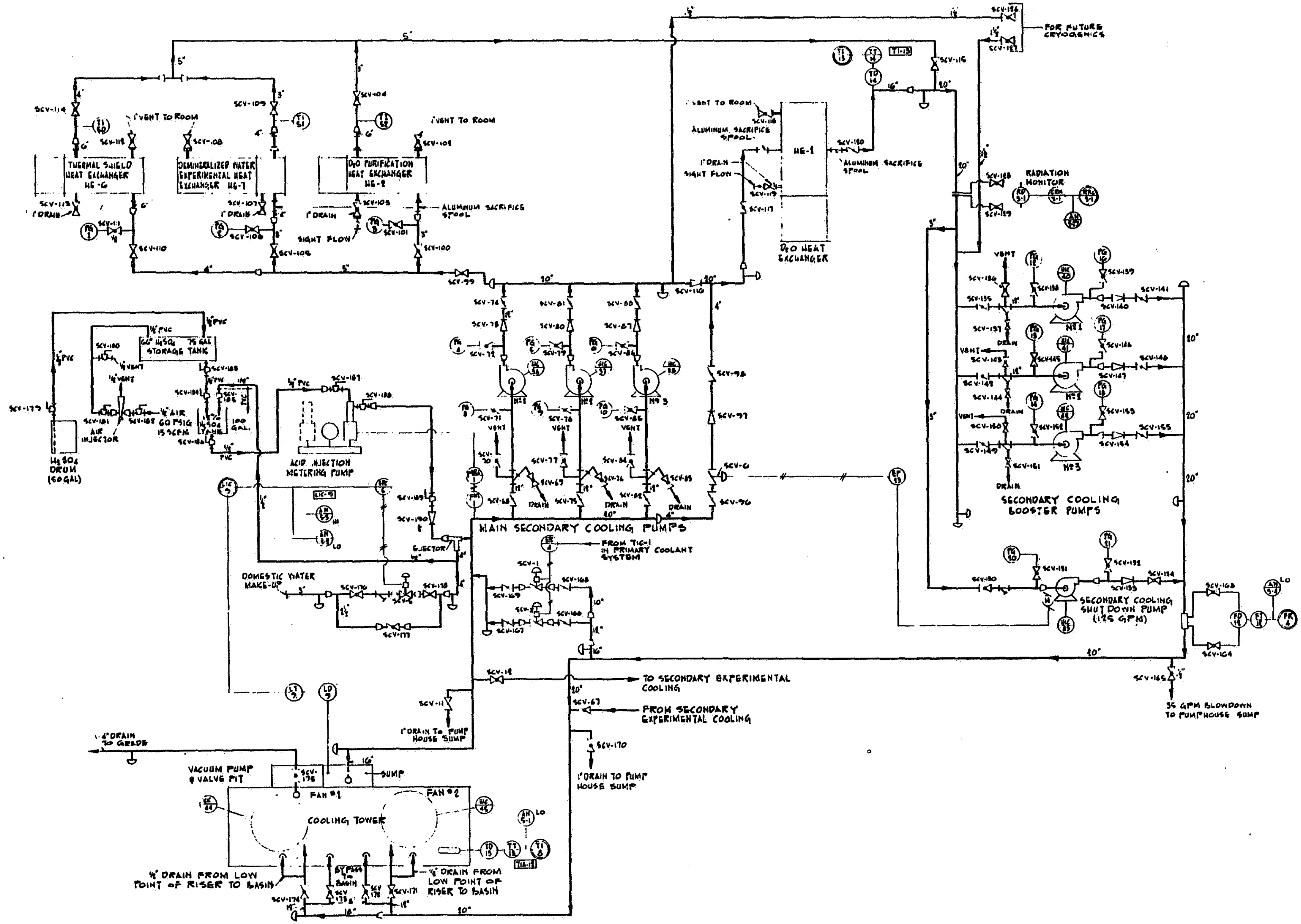


Figure 6.1 Secondary cooling system.

7.1 EMERGENCY SYSTEMS

7.1.1 EMERGENCY COOLING SYSTEM

7.1.1.1 General Description. This system as shown in Figure 7.1 provides cooling for the reactor core and experiments in the event of a loss of normal coolant through a pipe rupture.

The D_2O emergency cooling tank, acting as an overhead reservoir, is located approximately 37' above the reactor core. It receives a continuous flow of approximately 25 gpm from the heavy water purification system. Water overflows from an internal standpipe, located to hold up 3,000 gallons, to a reactor inner emergency cooling tank. This inner tank is located within the reactor vessel immediately above the reactor core, but below the normal D_2O level in the vessel. The inner tank has a holdup volume of approximately 800 gallons. During normal operation water from this tank returns to the primary system within the reactor vessel. In the event of a loss of normal reactor coolant, control valves can be operated manually or remotely from the control room which will drain D_2O from the bottom of the emergency tank to the inner emergency cooling tank or to each inlet plenum. The point of injection is based on evaluation of the location of the pipe rupture. Should the break occur downstream of the inlet plenum check valves, water would drain from the inner tank to the core; the inner tank being replenished from the emergency tank. The coolant passes over the fuel, removing decay heat, then drains out the pipe rupture into the sub-pile room. The water returns to a sump where it can be pumped back to the emergency cooling tank or to the main storage tank. If the break occurs elsewhere in the system, D_2O is drained from the emergency cooling tank to each reactor inlet plenum. With this mode the coolant passes upward through the core and exits through the normal reactor outlet lines. It then drains out the pipe break and is directed to a sump by a concrete curb and floor drains located in the D_2O process piping area. From this sump the heavy water can be pumped back to the emergency cooling tank.

This system also provides 1000 gallons of D_2O reserve for cooling of experiments. Control valves can be operated locally or remotely from the control room to drain D_2O from this tank to the experimental cooling system.

Domestic light water can be added to the emergency tank and to the experimental cooling system in extreme emergencies through double manual isolation valves with an open drain between.

7.1.1.2 Components Description. The emergency cooling tank is an aluminum reservoir approximately 20' in length and having a diameter of 4'6". It has a capacity of 3300 gallons but normally contains only 3000 gallons. It is equipped with two internal standpipes; one to hold up a volume of 3000 gallons and the other at the 2000 gallon level. Since the reactor core is of primary concern, the amount of emergency coolant available for experiments is limited by this second standpipe to 1000 gallons. Decomposed gases are swept from this tank by the helium sweep gas system.

The inner emergency cooling tank is made of concentric cylinders of approximately 82" diameter and 48" diameter. Construction material is 5052 aluminum with walls 1/4" and 3/8" nominal thickness and 1" thick top and bottom. There are six 5" diameter vertical thimble port penetrations, five 5" diameter top vent holes which go go the reactor vessel, and one 5" diameter penetration for the reactor vessel overflow pipe, one 2-1/2" diameter D_2O inlet penetration. Two 23/32" I.D. nozzles, which penetrate the inner cylindrical wall will drain the 8000 gallons of D_2O directing the flow into the emergency cooling distribution pan.

The emergency cooling sump pump is a single stage, centrifugal unit having a capacity of 40 gpm. It is driven by a 5 hp motor supplied from motor control center MCC-B6. The pump is controlled from the control room.

Remote control valves are installed to drain the emergency tank to the reactor vessel, to the experimental cooling system, and to the D_2O storage tank. D_2O can be returned to the

emergency tank from the storage tank through DWV-40 and from the sump through DWV-20.

Instrumentation is provided to sense emergency tank water level and temperature, emergency cooling flow to the reactor core, sump level, and emergency cooling sump pump discharge pressure.

7.1.1.3 Design Considerations. The inner emergency cooling tank is always full during reactor operation. Should the water level in the reactor fall for any reason, the inner tank will start draining through the two unvalved nozzles near its bottom into the emergency cooling distribution pan. This flow requires no mechanical action of any kind but is simply a direct physical result of the loss of water in the reactor vessel. Initially the flow from the inner tank is 40 gpm, but decreases as the water level in the inner tank falls. The total time to drain the tank is about 28 minutes. This provides time to asses the situation and determine which of the alternative procedures for use of the main emergency coolant supply should be employed.

The main emergency cooling tank is designed for 15 psig at 110°F. It was fabricated to ASME code requirements. The elliptical heads are 3/8", 5052-H112 aluminum with a 3/16" thick shell of 5052 aluminum.

Piping and valve design requirements are the same as required for the primary system. Refer to Section 5.4.

7.1.2 VOID SHUTDOWN SYSTEM

7.1.2.1 General Description. An additional mechanism for providing significant amounts of negative reactivity is provided in the form of an in-core, helium, gas bubble system as shown in Figure 7.2. On an appropriate signal this system will blow helium gas into the reactor core to create voids in the moderator. Six helium bottles supply the system with a flow rate of 180 SCFM at approximately 10 psig. This amount of helium would be exhausted in approximately 6 minutes; however, if sustained flow is required, the operator using a manual system may release air at 10 psig backed up by instrument air into the system.

If the void system is placed into operation the reactor scrams and the moderator dump valve will open.

7.1.2.2 Components Description. The system consists of six standard 220 SCF helium cylinders arranged in two banks of three bottles each. Each bank is equipped with its own 2200 to 25 psig reducer. A control regulating valve HEV-2 further reduces the supply pressure to that required for a flow of 180 SCFM. Immediately downstream of the regulating station is a remotely operated isolation valve, HEV-1. Back-up air can be injected into the system through HEV-8.

Helium flow is sensed by an installed orifice giving control room indication and annunciation. A pressure switch on the bottle header alarms in the control room on low pressure. For information on the design consideration of this system refer to Section 4.6.13.

7.2 REACTOR AUXILIARY SYSTEMS

7.2.1 FUEL HANDLING SYSTEM

7.2.1.1 Introduction. The NBSR fuel handling system provides for the rearrangement of the fuel within the core and the removal of spent elements completely within the shield. As shown in Figure 4.1 (Section 4), there is space above the reactor to raise a fuel element out of the core where it may be transferred to another position or to a transfer chute where it can be lowered into a receiving mechanism in the H_2O canal leading to the storage pool.

There are three phases of the fuel handling procedure. First the spent fuel elements are removed. Next the remaining elements are rearranged as desired in the core, and finally fresh fuel elements are loaded in the last phase.

The most critical phase is the removal of the spent elements since overheating must be prevented. The handling system offers two procedures. In one, the water is lowered below

the top of the transfer chute and the element is raised into the helium filled region above the core, transferred above the open transfer chute, and lowered directly into the receiving mechanism in the canal. In the alternate procedure, the water level is maintained at normal operational level so the region above the core and transfer chute is filled with D_2O. As before, the element is transferred to the transfer chute, but now it is lowered into a lock in the chute where it is held briefly by a special tool until the D_2O is drained from the lock and the element is picked up by the receiving mechanism. In this latter procedure, the element is only out of water about 3 minutes while it is in the lock compared to about 10 or 15 minutes for the all dry procedure. In the case of normal operation at 10 Mw, the decay heat production rate in an NBSR fuel element 8 hours after shutdown will be sufficiently low as discussed below, so that the dry transfer procedure will be the standard procedure used for the NBSR elements.

After the spent elements have been removed, the valves in the transfer chute will be closed and the water raised to its normal level. The elements will then be rearranged as desired.

Finally, the plug over the transfer chute is removed, and fresh elements are inserted through this opening into the transfer mechanism for insertion into the core.

7.2.1.2 Transfer Mechanism. The transfer mechanism consists of a set of pickup tools and transfer arms which penetrate the top shielding plug. There is a pickup tool over each fuel element, and transfer arms are located so that every possible fuel element position can be reached by at least one transfer arm. A simplified isometric presentation of the system is shown in Figure 7.3. For clarity, certain of the transfer arms and the details of the pickup heads are not shown. Figure 7.4 shows the actual pickup tool, fuel element head, and transfer arm fitting. Wherever a transfer arm location might interfere with an experimental thimble, it was designed to rotate on a cylinder around the experiment. In this way all fuel element transfers can be made without interfering with any of the in-core experimental facilities.

A typical transfer procedure is as follows. A pickup tool is manually lowered until it engages the head of a fuel element. It is then rotated under slight pressure until it slips over the pins in the fuel element head. Then it is pushed down and rotated clockwise. This disengages the fuel element locking bar from the top grid plate and allows the pickup tool to be withdrawn lifting the element. As the end of the pickup tool nears the top plug, it enters a mechanical maze which allows it to be accurately located and supported while a transfer arm is rotated into place. The position of the transfer arm is shown by a mark on the rotating cylinder outside the plug and is further located in the proper position to receive the element by a mechanical detent. When the arm is in place, the element is lowered into it. The small pins in the fuel element pickup head engage the grooves in the transfer arm preventing the element from rotating as the pickup tool is rotated counter clockwise to disengate it from the element. The transfer arm can then be rotated to another position where another pickup tool can engage the element and lift it from the transfer arm, allowing the transfer arm to be rotated away. If it is at a transfer position as shown in Figure 7.3, it can be placed in another transfer arm and moved to another spot. In this way, the element can be moved to any desired location or placed over the transfer chute for removal. Once in the desired location, the element is lowered on the pickup tool into place. The mechanical maze mentioned earlier assures the proper orientation of the fuel element as the pickup tool leaves the maze so that the rectangular element enters the rectangular opening in the grid plate properly. No distinction is made by the system between the two possible orientations 180° apart. The element is locked in the top grid plate by pressing down and rotating counter clockwise. A mechanical interlock at the top of the shielding plug makes it impossible to disengage the pickup tool from a fully inserted element unless the locking bar has been rotated to the full lock position.

7.2.1.3 Transfer Chute and Receiving Mechanism. A schematic drawing of the transfer chute and receiving mechanism is shown in Figure 7.5. In addition to the piping, it consists of 3 hydraulically operated valves, a fuel element holding tool and a hydraulic telescoping cylinder. The telescoping cylinder is the receiving mechanism and is pivoted so it can swing from the horizontal to the vertical position. This motion is controlled by a small hydraulic cylinder.

In the normal transfer procedure the D_2O water level is lowered below the top of the transfer chute. This lower level is guaranteed by opening a dump line whose top is at the

desired level. Next, the D_2O is drained from the transfer chute and the two upper valves are opened. After sufficient time has been allowed for thorough draining, the bottom valve is opened and the hydraulic cylinder raised so its receiving end is above the water level in the canal. The fuel element is then lowered until the bottom engages the mating receiver atop the hydraulic cylinder and released in the same manner used in releasing an element to a transfer arm. Since the element may be at a temperature above the boiling point of water, it will be necessary to close valve number 2 to prevent the H_2O steam generated by the hot element entering the canal from contaminating the main D_2O system. As soon as the valve is closed the element is lowered into the canal. When the cylinder is completely withdrawn it is tipped to the horizontal and extended so the element can be reached by conventional handling tools from the pool and placed in storage.

It is also possible to return a partially spent element to the reactor by reversing the above procedure. Using the hydraulic cylinder, the element is inserted into the lock between valves 2 and 3 with the top valve closed. The element is left there for a few minutes to evaporate all the H_2O off it; then the top valve is opened and a pickup tool is lowered to lift the element into the transfer mechanism. Although it is not anticipated that it will be desirable to do this often the system is designed so as not to preclude this possibility.

If, for any reason, it should be desirable to transfer an element out of the reactor while keeping it submerged during the transfer operation above the reactor core, the following procedure can be used. The top valves are opened and the fuel element holding tool is inserted. Then the element is lowered and released in the holding tool in the same manner as in a transfer arm. The top valve is then closed and the D_2O drained. The D_2O draining will take about 30 seconds. Then the lower valve is opened and the hydraulic cylinder is raised to engage the fuel element and lift it off the holding tool. The holding tool is withdrawn and the element lowered into the canal. The total time the element is not covered by water would be about 3 minutes or less.

7.2.1.4 <u>Fresh Fuel Insertion</u>. Fresh fuel elements will be inserted through the top plug of the reactor. The plug over the transfer chute will be removed and a fresh element inserted through the hole into the transfer arm. The insertion tool will pass through a short simple plug. Once in the transfer arm the element will be transferred to any desired location and placed in the core in the usual fashion.

In order to minimize the amount of tritium diffusing up through the 5" diameter hole during the brief time it is open, a cylinder will project below the D_2O water surface from the bottom of the plug so only a very small surface area will be exposed to the hole. In this way only a few cubic inches of D_2O saturated helium will be available to the hole instead of the whole gas volume over the water. This cylinder will also practically eliminate contamination of the helium cover gas by air.

7.2.1.5 <u>Safety Provisions of Transfer Chute</u>. Although great care has been taken to prevent a fuel element from becoming stuck in the transfer chute and it is considered highly unlikely, provisions have been made for auxiliary cooling. If the element is anywhere above the bottom valve, the valve can be closed and the chute flooded with water. Normally D_2O would be used but H_2O is also available. In addition, D_2O, helium, or H_2O may be sprayed on the element in the transfer chamber if in some fashion the bottom valve were blocked from closing. The helium would be used first, and then if the situation were prolonged, D_2O or H_2O could be used as appropriate.

7.2.1.6 <u>Fuel Element After Heating</u>. The hottest element in the 10 Mw equilibrium core will have been operating at a fission power rate of 475 Kw (431/.91, see Table 4.6-3, Section 4). Of the decay heat rate, less than one-half of the gamma heat will remain in the element although most of the beta energy will be converted to heat in the element. So, using the curves in Figure 4.31 (Section 4), and assuming all the beta energy and one-half the gamma energy is deposited in the fuel element, the heating power as a function of time after shutdown can be calculated. The results for the 475 Kw element are summarized in Table 7.2-1.

At the heating rates listed, 25 minutes is required after 8 hours decay to bring the element to its melting point (>1100°F) if no cooling of any kind is allowed. This assumes that all the fuel plates and the side plates in the vicinity of the fuel plates are sharing in the heating, but that the end pieces and connecting plates are not.

Table 7.2-1 Decay Heating Rate for 475 Kw Element

Time After Shutdown	Heating Rate
1 Min.	12,400 Watts
1 Hr.	3,980
8 Hrs.	2,020
24 Hrs.	1,270

A fuel element suspended in the helium atmosphere above the core would, of course, be cooled by free convection. The convection cooling of spent fuel elements has been studied by J. F. Wett, Jr., at ORNL (7.1). In this study they pulled elements into a hot cell over the pool (at various times after shutdown) and measured the surface temperatures of the interior plates of several elements. One element studied had a decay heating rate of about 2300 watts. The interior fuel plate temperatures reached equilibrium after about 40 minutes. The maximum equilibrium temperature was 640°F, well below the melting point. This temperature does not present a safety problem.

The NBSR element is very similar to the ORR except for different end fittings and overall length. Although the overall NBSR length is greater than the ORR, the fuel plates are about the same. Therefore, the Oak Ridge measurements should apply very closely to the NBSR elements of equal decay power cooled in air. In the NBSR case, however, the elements are in a helium atmosphere. A study of convection cooling in gases shows that helium is considerably more efficient as a convective cooler than air. The hottest NBSR element (475 Kw during operation) has a heat production rate of 2020 watts 8 hours after shutdown. This is somewhat smaller than the ORR element discussed above. Therefore, this comparison combined with the more effective cooling capability of helium assures us that the maximum temperature that an NBSR equilibrium core fuel element will reach in the transfer mechanism above the core after 8 hours of decay will be less than 640°F.

These measurements do not completely answer the question of fuel element temperature during the passage of the fuel element through the more confined region of the transfer chute. It should be noted, however, that the transfer chute is appreciably larger than the element allowing appreciable convection to take place, and that the element only spends about 2 minutes in the chute before it enters the water.

Although no problems are anticipated in the dry transfer of the element described above, a study will be made at lower power to confirm our expectations, and these results will dictate the minimum decay time required before an element will be transferred.

7.2.1.7 New Fuel Storage Facility. New fuel shipping containers will be received at the unloading platform of the Operations and Assembly Room. Refer to Figure 3.4, Section 3, for the location of this area. After checking for any shipping damage and a radiation survey, a single container will be opened and a fuel element removed from the container and placed in a numbered storage rack position in the fuel storage vault. This position is recorded with the fuel element serial number and kept for inventory purposes. This method is repeated until all elements are stored in the vault.

Elements are carried to the upper floor of the reactor for loading into the top of the reactor.

The fuel vault is located adjacent to the Operations and Assembly area. It provides storage space for 61 elements. Figure 7.6 shows the layout and construction of the vault and storage rack. Minimum centerline spacing between elements is 9". The elements are supported on end in polychloroprene lined slots held in place by a latch bar about mid plane of the element. The vault has three walls made of 8" thick hollow core cinder blocks with the fourth side of 24" solid concrete. The floor is of 8" concrete slab construction. No floor drain has been provided. Above the suspended ceiling of the vault is approximately a 4' high crawl space. The roof is precast concrete slabs and built-up design. There is a 4" diameter cast iron roof drain which runs through this crawl space to the storm sewer. A 3" diameter galvanized vent pipe from drains in an adjacent room runs through this crawl space.

Neither of these pipes are normally filled with water. No other sources of water enter the vault. No combustible materials are stored in the area, thus eliminating the hazard of fire.

The laboratory and office building is kept locked during all non-working hours. The vault can only be entered through a locked door, the keys to which are in the possession of the Chief Nuclear Engineer and the Reactor Radiations Division Chief. Entry to the vault cannot be gained through the overhead crawl space.

A radiation monitor is mounted on the exterior wall of the fuel vault in the operations and assembly area. It is of the scintillation type having a Na I, thallium activated crystal. Its range is 0.1 mr/hr to 100 mr/hr with a response time of 0.63 seconds for full scale deflection. It is supplied with a check source causing the instrument to read slightly above the low scale level. If power or component failure occurs, the lower contact of a dual contact indicating meter is closed, causing an amber light to be lighted and an alarm to sound. On high radiation the upper contact is closed causing a red light to be lighted and an alarm to sound. The instrument will not jam in a high radiation field. It must read full scale until the field subsides to a readable level. Power is fed to the instrument from the critical power supply.

The storage facility is designed to assure that no critical configuration can be achieved even under the highly improbable event that the (non-water-tight) room should be flooded with H_2O. Only 18 elements are stored in the center island and the others are stored in a single row along the room walls at least 3' away. Adjacent element storage positions are 9" on center. To demonstrate the impossibility of the existance of a critical configuration a calculation of the multiplication factor was made under the following very conservative assumptions.

(1) Room completely flooded with H_2O.
(2) Each element contained 250 gm of U-235.
(3) Room completely filled with elements on 9" centers (approximately 200 elements).
(4) Perfect neutron reflection at the top and bottom of fueled section of element.
(5) Fuel loading continuous through gap. (This implies a total loading of 330 gm for the hypothetical no gap element)

The resultant effective neutron multiplication factor, k_∞ was calculated to be 0.85. Clearly, even with the room flooded, the multiplication factor of the designed configuration is less than this conservative calculation and therefore there is no possibility of criticality.

7.2.1.8 Spent Fuel Storage Facilities. Facilities are provided in the fuel storage pool to receive and store irradiated fuel assemblies, load irradiated fuel assemblies in casks for shipment, cut aluminum end castings from irradiated fuel elements, and to store these castings prior to disposal. Storage racks and miscellaneous tools are provided.

As fuel is discharged from the reactor to the pool, the elements are placed in a storage rack designed to hold full length fuel elements. The storage rack is placed along one side of the pool wall. The elements are hung by their pickup heads from brackets along the back wall of the rack as shown in Figure 7.7. In the fueled region of the elements, the back wall of the rack is lined with boral and a boral plate extends out from the wall 7" between each fuel element. The spacing between partitions is 4-1/8". A bar runs across the front of the rack near the bottom of the fueled region, and a latching bar is located near the top of each fuel element partition. Besides providing a secondary means of preventing the elements from falling out of the rack, the latches and bar prevent the accidental approach of a second fuel element to closer than 3" minimum separation from one in the rack. The rack is designed to hang 30 elements in a single row along the pool wall.

The boral partitions between elements are added to preclude any possibility of criticality. Even without the boral, calculations indicate that a single row of an infinite number of 250 gm elements in an infinite H_2O pool would yield a substantially subcritical configuration (effective multiplication factor of 2 to 5).

For shipping, the fueled sections are cut out of the full fuel element to yield two sections per fuel element of about 13" length each. Figure 7.8 shows the design of a rack to hold 60 cut sections (30 full elements). They are stored on sloping shelves along the

pool wall. The are separated by 9" so the minimum space between elements in a vertical column is 6". The vertical columns of shelves are defined by boral partitions and each column holds 4 elements. The spacing between partitions is 4-1/8". Since the minimum distance (6") between fuel elements in the vertical column is more than twice the neutron migration length, any interaction between fuel elements along the vertical column is very small and the boral prevents significant interaction between adjacent columns.

An area at the east end of the canal is reserved for the loading of cut elements into a shipping cask. The shipping cask is lowered into the canal in this area, opened for loading, and the fuel elements placed in the cask. The cask cover is then set in place, the cask lifted from the pool, washed and removed through an overhead hatchway from the building via truck and trailer.

The transportation of spent fuel elements to the reprocessing plant will be consistent with health and safety requirements of pertinent AEC and ICC regulations. The shipping cask will be approved and a Bureau of Explosives permit will be issued for each shipment.

7.2.2 PRIMARY PURIFICATION SYSTEM

7.2.2.1 General Description. The primary purification system is designed to remove both soluble and insoluble corrosion and other foreign materials from all heavy water systems. This task is accomplished by 2 installed, mixed bed, resin columns and 2 cartridge type filters.

D_2O supply to the system comes from 2 pumps located within the D_2O storage sump tank, replenished with bleed flows from the reactor vessel, the thermal column tank cooling system, and the D_2O experimental cooling system. One pump acts as a standby with the other supplying approximately 102.5 gpm to the purification heat exchanger. Here the D_2O is cooled from approximately 115°F to 100°F before entering the ion exchangers. Outlet flow from this heat exchanger is divided with 77.5 gpm bypassing the IX columns and 25 gpm passing through the purification system. The purification system consists of a prefilter, two IX columns in parallel (one on standby), and an afterfilter. After passing through the purification system, the 25 gpm demineralized flow joins the bypass flow to supply the D_2O experimental cooling system, the emergency cooling system, and the thermal column tank cooling system. Water can be returned to the primary system directly through control valve DWV-39. A complete flow diagram of the system is shown in Figure 7.9 with an integrated flow scheme shown in Figure 5.2.

7.2.2.2 System Components. The main components of the systems are: the D_2O storage tank, 2 supply pumps, a heat exchanger, a prefilter, 2 ion exchangers, an afterfilter, associated piping, valves, and instrumentation.

The D_2O storage tank, located in a pit below the reactor basement floor, is sized to receive the entire plant D_2O inventory. Total capacity is 14,650 gallons, with a sump capacity of 326 gallons. During normal operation the sump is filled with D_2O. The remaining tank capacity can receive D_2O dumped from the reactor vessel via the 8" dump line and valve DWV-9.

Storage tank pumps, DP-7 and 8 are canned motor, submersible, single stage, centrifugal units. All parts in contact with heavy water are of 316 stainless steel. Electrical supply for DP-7 is from emergency power motor control center MCC A-5 while DP-8 receives power from MCC B-6. The control station for both pumps is on the control room console.

Heat exchanger, HE-2, is of the shell and U-tube type with D_2O in the tubes and secondary cooling water in the shell. It is constructed entirely of 6061-T6 aluminum with Alcad tubes and double tube sheet.

The IX prefilter and afterfilter are identical units containing 6 replacable, cellulose, acetate cartridges which will remove particles 5 microns and larger from the system.

Two IX columns each having approximately a 6.5 ft^3 volume are provided. A mixed DOD resin bed is used to remove all dissolved materials from the system. Normally one column will be used with the other acting as an installed spare. Construction of the vessels is 304 stainless steel. The method of resin removal is discussed in Section 7.2.2.3.

Air operated control valves are installed in the system to return D_2O to the deuterization system, and to return D_2O to the primary system at the pump suction.

Instrumentation is installed to provide remote readout of D_2O storage tank level, purification flow rate, heat exchanger inlet and outlet temperature, ion exchanger influent and effluent conductivity, ion exchanger outlet flow, and flow return to the primary system.

7.2.2.3 Ion Exchanger Resin Change Procedure. Both primary ion exchangers are capable of having their resin changed in place. When an ion exchanger is exhausted, it is isolated from the system. The largest radiation problems arise from Na^{24} and tritium (see Section 5.3.4). The column activity is allowed to decay before resin recharging is attempted, thus reducing the direct radiation problem from Na^{24}. The tritium bearing D_2O is then removed by a de-deuterization process wherein the D_2O is forced out a 1/2" diameter pipe at the bottom of the IX vessel by light water which enters at the top through another 1/2" diameter line. After the resin has be de-deuterized the spent resin is removed in controlled amounts from the bottom of the vessel through a 2" diameter line. The vessel is then rinsed thoroughly with light water and allowed to dry. New resin which has been deuterized is then installed through a 2" diameter connection at the vessel top. The ion exchanger is then ready for service.

7.2.2.4 Design Considerations. The D_2O storage tank pumps are designed to deliver 102.5 gpm at a discharge pressure of approximately 75 psig. Motors are sized at 15 hp and are supplied from emergency AC power.

IX vessels are designed for 110 psig at 110°F. They are sized to operate continuously with a flow of 25 gpm. Additional high density concrete block 8" thick is used for shielding.

The design pressure for the filters is 125 psig at 120°F. They are constructed to facilitate easy removal of cartridges.

Heat exchanger, HE-2, has a tubeside design of 200°F at 100 psig and a shellside design of 150°F at 75 psig and will transfer a heat load of 1.18×10^6 to the secondary coolant system. It is designed and fabricated in accordance with the ASME code, Section VIII, including nuclear case rulings 1270N and 1273N. At rated flows of 102.5 gpm in the tube side and 170 gpm in the shell side, D_2O is cooled from 119.5°F to 100°F while secondary water enters at 84°F and exits at 98°F. Total allowable pressure drop in both the tube and shell side is 7.5 psi. Design shell fouling factor is established at 0.001 while tube side is 0.0005.

7.2.3 SPENT FUEL POOL COOLING SYSTEM

7.2.3.1 General Description. This system removes the decay heat from the spent fuel in the storage pool, provides a means of maintaining the water purity, and removes any particulate activity that may be present in the system. A flow diagram of the system is shown in Figure 7.10.

Demineralized water is circulated through the system by two storage pool circulating pumps. One pump is normally operated with the other on standby. The running pump takes its suction from the collection basin and provides 75 gpm to HE-8, the storage pool heat exchanger. The inlet temperature to HE-8 is approximately 72°F and the outlet about 60°F, depending on the injection temperatures of the chilled cooling water on the shell side of HE-8. From the outlet of HE-8, 65 gpm goes directly back to the storage pool and 10 gpm through a prefilter, an ion exchanger, and an afterfilter. From the afterfilter the water is returned to the storage pool.

Fuel elements are transferred from the reactor vessel to the storage pool canal via the reactor vessel drop-out chute. (See Section 7.2.1) In the canal the fuel element is handled underwater as it is moved to the storage rack in the bottom of the pool. The spent fuel remains in the pool until it is shipped off site for reprocessing.

The storage pool is filled with demineralized water from the water treatment system. System losses are automatically made up by a storage pool level control instrument. The level instrument senses the storage pool level and sends an appropriate signal to valve WTV-1, which controls the makeup flow of demineralized water from the water treatment system.

7.2.3.3 Components Description. The storage pool is 18'2" deep, 20' long, and 10' wide. When filled to its operating level it contains approximately 30,000 gallons of water. A canal extends from one end of the pool to the sub-pile room so that spent fuel can be passed from the reactor vessel drop-out chute to the canal.

At the other end of the storage pool is a collection basin into which the storage pool overflows. The storage pool is normally filled to the point that the collection basin is partially filled so that a head of water is provided for the circulating pumps. A 6" curb is provided around the top of the storage pool.

Two identical 75 gpm centrifugal pumps driven by 2 hp motors provide a means of circulating the storage pool water. Both pumps are controlled from the reactor control panel by means of hand-operated switches. One pump will normally be running and the other on standby. Should the running pump trip, the standby pump will start automatically. Pump #1 gets its power from motor control center MCC A-3, and pump #2 is supplied from MCC B-4.

HE-8 is a U-tube type heat exchanger which has an aluminum tube side and a carbon steel shell. The tubes and tube sheet are carbon steel clad with aluminum. Chilled water circulates on the shell side at 90 gpm with storage pool water on the tube side at 75 gpm. The heat exchanger transfers 450,000 BTU/HR from the storage pool system to the chilled water system.

One 10 gpm centrifugal pump driven by 1 hp motor circulates the storage pool water through the purification section of the system. It is hand controlled from the control board and gets its source of power from MCC B-4.

Two 5 micron, cellulose fiber, replaceable type filters filter the water at the inlet and outlet of the ion exchanger. The filter housing and all metallic internals are of 304 SS and designed to withstand a pressure of 125 psig.

One 6.5 cubic foot ion exchanger maintains the purity of the storage pool water. The ion exchanger is filled with a mixed bed HOH resin. The ion exchanger vessel is made of type 304 stainless steel and is designed for 110 psig and 110°F.

The storage pool circulating system consists of 2" aluminum piping and valves except for the purification section which is 1-1/2" aluminum piping and valves. All components are capable of being fully isolated, and in addition, the purification system filters are capable of being bypassed.

There are 3 air operated control valves in the system, WTV-1, SPV-1, and SPV-2. SPV-1 and SPV-2 are identical in construction and function. They are the discharge valves from the storage pool circulating pumps and are hand controlled from the control panel. The discharge valve on the running pump should be opened only after the pump has reached full speed so that the pump motor is started in a no-load condition.

Instrumentation is provided to detect storage pool water level, system flow, purification flow, pressure drop across filters and conductivity of system water.

7.2.3.3 Design Considerations. The system is designed to remove heat generated in the spent fuel pit while storing two full core loadings. The temperature of the water is maintained at approximately 70°F.

Impurities are removed from the water to assure the optical clarity required for manipulating spent fuel or for refueling operations and cleanliness of fuel assembly surfaces. The filter removes all solids which are 5 microns or larger in size. The system will be operated to maintain a turbidity not to exceed 2 PPM.

Water conditions require a pH of 6 to 8. Principle metals in contact with the water are stainless steel and aluminum.

The heat exchanger was fabricated in accordance with Section VIII of the ASME code, and Standards of Tubular Exchange Manufacturer's Association, Class C, and is code stamped.

7.2.4 EXPERIMENTAL COOLING SYSTEMS

7.2.4.1 Secondary Water Experimental Cooling. This system circulates 50 gpm of secondary cooling water for experimental use in which water conditions are not critical. This water is distributed to the service boxes at experimental stations by either of 2 centrifugal pumps.

There are 19 connections in all, 2 for the cryogenic facility, 4 for the grazing tubes, 4 for the vertical thimbles, and 9 for the beam ports. An automatic bypass (SCV-3), controlled by system pressure is located between the supply and return headers to the experimental facilities. This will recirculate the water when the system is not in use and provides reduced recirculation when cooling is desired. In addition, there is a cooling tower bypass valve, SCV-4, which is controlled by the temperature of the supply header. This valve will maintain a constant inlet temperature and during cold weather if the system is not in use the coolant can be recirculated to prevent freezing. Figure 7.11 illustrates this system.

7.2.4.2 Demineralized Water Experimental Cooling. This system will circulate demineralized water through a closed loop to provide a supply of cooling water at each experimental facility. One circulating pump will normally be running to circulate the coolant through heat exchanger HE-7, where the heat from the system is given up to the secondary cooling system. From HE-7 the coolant goes to a header where water is made available for cooling experiments. A pressure-controlled, automatic bypass valve controls the recirculation flow to prevent overpressurizing the system when cooling is cut off to a given experiment and to prevent underpressurization of the system when flow is established to a given experiment. The automatic bypass regulates the experimental cooling by maintaining system pressure under varying load conditions. From the supply header 5 gpm is diverted to the purification section of the system where the water undergoes filtration and ion exchange. This 5 gpm is returned to the storage tank as is the water from the experiments and the bypass. In addition, there is a 1" supply and return line to the sub-pile room for the installation of cooling to future experiments.

The storage tank has a capacity of 2400 gallons. It acts as a surge tank to accomodate the expansion and contraction of the coolant and provides a suction head for the two circulating pumps. The tank is vented to the irradiated air system. Make-up water is supplied from the water treatment system. Figure 7.12 shows the flow diagram for this system.

7.2.4.3 D_2O Experimental Cooling System. This system, as shown by Figure 7.13, supplies a maximum of 35 gpm of D_2O to the reactor experimental facilities. The system is supplied from the D_2O purification system through DWV-26. One of two booster pumps then circulates D_2O to connections in the reactor upper trench and at the reactor face on the first floor level.

A back pressure regulating valve DWV-25 is available to control experimental supply pressure, thus preventing an over-pressure or under-pressure of other experiments as flow is established or shut off from any facility. Heated D_2O is then returned to the storage tank where it is cooled via the purification system heat exchanger.

Emergency back-up cooling is available from the emergency D_2O tank and from the domestic water system. The largest heat load on the system initially will be the bismuth plug for the cryogenic facility, requiring approximately 16 gpm with an exit temperature of 140°F.

Instrumentation is provided to sense facility supply pressure, cryogenics plug outlet temperature, and cryogenics plug cooling water flow.

The circulating pumps are supplied electrically from emergency AC motor control centers MCC-A5 and MCC-B6. An electrical interlock prevents operation of either experimental cooling pump unless one of the D_2O purification pumps is in operation.

7.2.4.4 Thermal Column Tank Cooling System. This system is provided to cool the bismuth shield and to further assist in the slowing down of neutrons for use in the graphite thermal column. Figure 7.14 outlines the system.

The tank receives D_2O from the purification system supplied by the storage tank pumps. Flow is sensed by an installed orifice and is controlled by an automatic regulating valve. Water supply pressure is controlled by a regulating valve which returns D_2O to the storage tank. Heavy water enters the thermal column tank and flows upward and over the top of the bismuth shield serving as a coolant. It leaves the tank at the bottom of this shield. Although the tank remains full of D_2O, the small cavity on the other face of the bismuth shield has the water level maintained by an elevated "U" bend in the outlet line. Another "U" bend vented to the gas space of the D_2O storage tank serves as an over-pressure relief.

Normally there is a 1" gas space between the top of the bismuth shield and the top of the thermal column tank. This area is swept by helium to remove any disassociated D_2O which may have collected.

The thermal column tank is constructed to fit the outside curvature of the reactor vessel. An inner can is filled with approximately 7,600 pounds of bismuth. Construction is of 1/4" 5052 aluminum conforming to ASME codes. It has a design pressure of 1 psig, plus a full tank of D_2O, at 150°F. Its measured water volume is 32.2 ft^3.

7.2.5 THERMAL SHIELD COOLING SYSTEM

7.2.5.1 General Description. The thermal shield cooling system removes the gamma heat generated in the thermal shield, experimental facilities, and the lower shield plug assemblies.

During normal operation one pump will circulate 670 gpm of demineralized water through the system to cool the lead thermal shield, experimental facility shutters and the lower shield plug assemblies. The control switch for the second pump will be placed in the stand-by position so that it will start automatically in the event the running pump should trip off. Both pumps and their respective discharge valves are controlled from the reactor console.

Figure 7.15 shows the thermal shield cooling system. The coolant passes through the heat exchanger HE-6 where it gives up heat to the secondary cooling system. Shield cooling water circulates on the tube side of HE-6 and at 10 Mwt the inlet temperature is expected to be approximately 100°F with the outlet at 97°F.

From the main cooling header 10 gpm is diverted to a purification system which consists of a prefilter, 2 ion exchanger columns and an afterfilter. The purification system may be placed in service from the reactor console by control of the automatic inlet valve TSV-5. Either ion exchanger may be placed in service from the console by positioning TSV-3 and TSV-4. The purified water is returned to the storage tank from the purification system. This flow is accomplished by the differential pressure across the circulating pumps.

In the main header 660 gpm goes to the ring header, the first floor trench and the second floor trench which are the distribution points for the cooling water. From the first floor trench 77 gpm is provided for cooling 5 lower piping shielding sleeves and the bottom shield bedplate. The ring header provides 550 gpm to 189 cooling tubes imbedded in the 2" lead thermal shield. The second floor trench provides 43 gpm for cooling the lower outer shield plug assembly and the experimental facility shutters plugs. From the various return headers the water is collected in the 1,450 gallon storage tank which provides a suction head for the circulating pumps and acts as a surge tank to accomodate any expansion or contraction of the coolant. In the return header to the storage tank there is an automatic valve (TSV-6) which opens and closes as pumps are started and stopped. This is to keep the tubes filled with water when the pumps are shutdown and to prevent water hammer when the pumps are started.

7.2.5.2 Components Description. There are 2 identical 570 gpm single stage centrifugal pumps which circulate the coolant in the system. They have a bronze casing and impeller with a stainless steel shaft. These are a heavy duty type designed for continuous operation. The pumps are controlled from the reactor console and are interlocked so that if one should trip off the other will start automatically if its control switch is in the stand-by position. Both pumps are also interlocked with valve TSV-6 (Storage Tank Return Valve) so that the valve will open when either pump is started and shut when both pumps are off unless its control switch is in the "open" position, in which case the valve will remain open. The pumps also have automatic discharge valves but they are not interlocked with them and must be opened from the console. The pumps should be started with their discharge valves closed and the

valves opened after the pump is up to speed. This is to prevent the motor from starting under a full load condition. Circulating pump #1 receives its power from MCC-A5 and #2 from MCC-B6.

HE-6 is a 2.08 x 10^6 Btu/hr heat exchanger which transfers heat from the thermal shield cooling water to the secondary cooling water. Secondary cooling water circulates on the shell side at the rate of 350 gpm and shield cooling water on the tube side at the rate of 670 gpm. The shield cooling water enters the heat exchanger at 103°F and leaves at 97°F at 10 Mwt. Design tube side pressure is 125 psig at 250°F and shell side is 75 psig at 200°F. The tube side is all copper or copper-nickel and the shell side is all steel except the tubes themselves which are 70-30 copper-nickel inside and steel outside.

The shield cooling storage tank acts as a reservoir in the system and provides for expansion and contraction of the liquid at various power levels. It has a capacity of 1,450 gallons and provides a suction head to the circulating pumps. The tank is all copper and has a design pressure rating of 15 psig. A sight glass allows for local observation of the tank level and a level transmitter gives a remote readout in the control room. Make up water is taken from the water treatment system; however, this valve must be operated manually from the process room.

Two identical filters at the inlet and outlet of the purification system filter the water. The filters are a replaceable 5 micron, white cellulose fiber type. There are 6 cartridges to a unit. The filter housing and all metallic internals are of 304 stainless steel and designed to withstand an internal pressure of 125 psig.

There are two 6.5 ft^3 mixed bed ion exchange columns in parallel that maintain the purity of the shield cooling water. Either bed can be placed in operation from the control room by opening or closing automatic valves TSV-3 and TSV-4. The ion exchanger vessel is made of type 304 stainless steel and is designed for 110 psig and 110°F. The ion exchanger is filled with mixed bed HOH resin.

Primary concern in this system is given to temperature, flow, and conductivity instrumentation.

FIA-16 measures the flow through the purification section of the system. It indicates and provides a low flow alarm on the console in the control room. It is an orifice type element.

FIA-15 measures the flow in the main supply line. It is a venturi type element supplying a D/P cell which in turn provides indication, a low flow alarm, and a low flow shim rod rundown.

System temperature indication is provided by 5 installed sensors; TI-24 in the return ring header; TI-23 in the return trench header; TI-22 at the storage tank inlet; TIA-17 at the heat exchanger inlet; and TIA-16 on the outlet of the heat exchanger. All channels have indicators and alarms, except TI-22 which does not have an alarm.

Conductivity is sensed at three locations in the system; CC-6 at the outlet of the heat exchanger, and CC-7 A and B on the inlet and outlet of the ion exchangers. Control room indicators and alarms are provided.

LIA-8 measures the level of the storage tank. It indicates and alarms in the control room. In addition, there is a sight glass for local indication.

There are 6 control valves in the system; TSV-1 and TSV-2 circulating pump discharge valves; TSV-3 and TSV-4 ion exchanger inlet valves; TSV-5 purification system flow control valve; and TSV-6 storage tank return valve. All valves are pneumatically powered diaphragm motors with spring returns and operate on 35 psig instrument air. All valves except TSV-5 have limit switches indicating their open or closed positions. TSV-5 is an electro-pneumatic throttle valve utilizing a feedback control to eliminate hunting of the valve. The properties of these valves are summarized in Table 7.2-2.

Table 7.2-2 Valve Characteristics

Tag #	Size	Motor Operation	Service	Solenoid Operation
TSV-1	6"	ATC	Open/Close	ETO
TSV-2	6"	ATC	Open/Close	ETO
TSV-3	1-1/2"	ATC	Open/Close	ETC
TSV-4	1-1/2"	ATC	Open/Close	ETC
TSV-5	1-1/2"	ATO	Throttle	---
TSV-6	6"	ATO	Open/Close	ETO

All piping throughout the system is seamless copper. The purification leg is 1-1/2" pipe, the main piping is 6", and the cooling coils are 1/2".

7.2.5.3 Design Considerations. Control valve materials conform to ASTM specifications B-61 and have a minimum working pressure rating of 125 psig. They have cast iron bonnets with hypalon of GRS nylon reinforced diaphragms.

Manual valves in the process room are of the diaphragm type with bronze bodies conforming to ASTM specification B-61. Shut-off valves on the top of the reactor or in the trench are angle or globe types with disc seats and graphite asbestos packing. Valves are rated for 125 psig.

Pipe is threadless cold-drawn seamless copper, conforming to ASTM specification B-302 "Threadless Copper Pipe, type DHP drawn temper (except where bending is required, in which case it was annealed for bending). Tubing is seamless, annealed, type DHP conforming to ASTM specification B-88, type K.

Piping was fabricated to conform to the requirements of Sections 1 and 6 of ASA Code for Pressure Piping.

The storage tank was fabricated from 3/16" thick copper conforming to ASTM Specification B-11. Design was in accordance with the ASME Unfired Pressure Vessel Code. A 1/16" corrosion allowance was required. Design pressure was established at 15 psig.

The heat exchanger was designed in accordance with ASME Code, Section VIII, TEMA Class R.

Design fouling factor for the tube side is 0.0005 and 0.001 for the shell side.

7.2.6 HELIUM SWEEP GAS SYSTEM

7.2.6.1 General. This system provides an inert helium atmosphere over all vessels and tanks which normally contain heavy water. Among such tanks are the reactor vessel, the storage tank, the emergency cooling tank, the thermal column tank, and the purge tank. The system also maintains the proper oxygen concentration in the sweep gas, and recombines any disassociated D_2O which might be present. Refer to Figure 7.16 for the flow diagram.

One of 2 helium blowers circulates helium at approximately 20 cfm through a recombiner preheater where the gas is heated to a fixed temperature. The helium enters a recombiner. Water vapor formed in the recombiner is condensed by an aftercooler. Helium is then distributed to various components of the D_2O systems. A helium accumulator is provided to maintain a constant pressure on the system. Makeup is provided by a bank of 4 cylinders located in the cold lab basement. A cold trap condenser is installed using liquid nitrogen as a heat sink for purging operations thus recovering any D_2O vapors.

7.2.6.2 Components Description. The helium heater is constructed of 3/16" aluminum plate, with a removable head capable of withstanding an internal pressure of 2.5 psig. The heating element is of the strip type designed to heat moving gas. The heater is rated for a net output of 750 watts operating on 110 volts, 60 cycle single phase AC power supply.

The helium blowers are of the rotary, positive displacement type, capable of delivering 20 cfm of helium gas while developing a total dynamic head of 8" of water with inlet conditions of 14.7 psig. Each blower has a shaft seal to minimize shaft leakage and each blower is powered through a double-grooved V-belt drive with a 1 hp motor. Each blower and motor is enclosed in an aluminum tank to retain any helium which may leak from the shaft seals. Each tank is capable of withstanding an internal pressure of 2.5 psig.

The helium recombiner is a cylindrical vessel made of 1/4" thick aluminum plate capable of withstanding an internal pressure of 15 psig and temperature of 200°F. The vessel has a removable flanged head to facilitate changing the 15 lbs of 1/8 diameter alumina-palladium pellets. These pellets are held in place with a .031 diameter #7 mesh wire cloth basket. Pressure drop through the recombiner is 0.726" H_2O at 20 scfm helium flow rate.

The recombiner condenser is a cylindrical vessel made of 304 SS SCH 10S capable of withstanding an internal pressure of 15 psig with cooling coils wrapped external and an internal providing a continuous path for chilled water.

The cold trap is a cylindrical vessel made of 304 SS SCH 10, and will withstand an internal pressure of 15 psig. Thirteen 1" diameter tubes filled with liquid nitrogen are suspended from the upper flange. Sweep gas enters from the side, passes across the tubes, and exits from the bottom.

The gas holder is a 6061 aluminum cylinder 5'11" inside diameter and 6'8" high. Inside the holder is a traveling piston which moves up and down to accommodate volume changes on the system. The piston has a rubber fabric seal which travels with the piston and prevents helium leakage. The piston is weighted to obtain the desired pressure on the system. Mechanical linkage connects the piston to a pointer and micro-switch arrangement for level indication, high and low volume alarm, and operation of a relief valve.

The carbon filter is activated charcoal enclosed in an aluminum tank made of 1/4" plate.

The helium seal filter is activated charcoal cartridges contained in an aluminum tank made of 1/4 " plate.

Instrumentation is provided to indicate the volume of contained gas in the accumulator, D_2O condensate levels in traps, helium flow in the system and flow rates to the reactor vessel and emergency cooling tank. Also provided is temperature controls and indicators for heater control and recombiner outlet temperature. Remote indication of recombiner outlet pressure is installed with local pressure indicators on gas cylinder pressure and inlet-outlet pressure of helium blowers.

7.2.7 CO_2 SYSTEM

7.2.7.1 General. The CO_2 system provides a means of purging all air from the cavity between the reactor vessel and the thermal shield. The pneumatic irradiation system is also purged and supplied with makeup gas from the CO_2 system. The purpose of these purges is to remove all air, thus preventing argon activation in the areas of high neutron flux. A flow diagram of this system is shown in Figure 7.17.

The CO_2 gas is supplied in standard commercial gas cylinders at a pressure of 2200 psi. There are two high pressure manifolds, and each manifold is connected to three bottles of gas. Each manifold has a reducer which supplies a common header with gas at 10 psig. The 10 psi header supplies two regulating valves, one for the CO_2 gas holder and the other for the pneumatic rabbit system. They are capable of being cross-connected so that either one can supply both systems and the gas holder can supply either or both systems.

The regulating valves reduce the system pressure from 10 psi to 2" of water pressure. After each purge, the gas holder is recharged and used to maintain a slight positive pressure on the system and take up any volume changes due to expansion and contraction of the gas. Two relief valves on the supply line protect the system from over-pressure, COV-1 senses main line pressure and relieves to the irradiated air system, while COV-2 is interlocked with the gas holder diaphragm to protect it from over-pressure. COV-2 also relieves to the irradiated air system. The CO_2 purge fan sweeps the reactor vessel cavity of air and

gasses at the rate of 150 cfm: the purge fan discharges to the irradiated air system which ultimately discharges the gasses to the atmosphere via the stack.

7.2.7.2 Components Description. The gas holder is a carbon steel cylinder 5'11.2" inside diameter and 6'8" high. Inside the holder is a traveling piston which moves up and down to accommodate volume changes in the system. The piston has a rubber fabric seal which travels with the piston and prevents CO_2 leakage. The piston is weighted to obtain the desired pressure on the system. Mechanical linkage connected to the piston and a pointer give volume indication on the side of the gas holder. Microswitches located on the side of the holder and actuated by the pointer give remote level alarms both high and low, as well as operating a relief valve to prevent over-pressurizing the gas holder.

The CO_2 purge fan is a Westinghouse, series 500, industrial fan rated at 150 cfm. The fan is belt driven by a 1/3 hp, 115 volt, 5.3 amp motor. The motor is powered from MCC-A-3 and controlled from the reactor control room.

All piping in the low pressure section of the system is 2" carbon steel. The low pressure regulators are designed to reduce the upstream pressure, which varies between 20 - 40 psi, to a downstream pressure of 2" H_2O. Two relief valves, one downstream of the high pressure regulator and the other downstream of the low pressure regulator are designed to relieve the full capacity of the regulators. They relieve at 100 psig and 1/2 psig respectively. Two automatic control valves, COV-1 and -2, are single-seated, diaphragm-operated, air to open valves.

Instrumentation is provided to sense system pressure and gas holder diaphragm levels. Pressure indicator controller PIC-101 senses the system pressure and controls relief valve COV-1 which will open on a high pressure signal to prevent over-pressurization of the system.

Each manifold consists of one high pressure reducer, one upstream isolation valve and associated piping to three gas cylinders. All piping upstream of the H.P. reducing valves is seamless brass pipe with forged bronze fittings and is rated at 3000 psig. The H.P. reducing valves are designed to reduce the inlet pressure, which will vary from 2200 to 500 psi, down to 40 psi. They are two stage, forged bronze regulators with pressure gages to indicate upstream and downstream pressure. Piping downstream of the H.P. reducers is schedule 40, black seamless steel, rated at 2000 psi.

7.3 FACILITY SERVICE SYSTEM.

7.3.1 EMERGENCY POWER SYSTEM

7.3.1.1 General. This system is installed to provide emergency electrical power to the reactor in the event of a complete loss of outside power. The system consists of two 150 Kw diesel driven generators, a DC to AC inverter-diverter, a battery bank, and a standby battery charger. Figure 7.18 shows the fuel oil and cooling flow diagram for the diesel generators. Refer to Section 3.8 and Figure 3.12 for details and a schematic diagram of the normal and emergency electrical supplies.

Power to the facility is supplied by two 69 KV to 13.8 KV feeders at the NBS substation by a 1500 Kva 13,750/480 transformer through a 2500 amp network protector. Substation B is similar to A.

The emergency power motor control center consists of two 480 volt, 3 phase sections MCC-A5 and MCC-B6 tied together with a normally closed manually operated bus tie circuit breaker.

Under normal conditions the emergency motor control centers are supplied by power from the motor control center MCC-A3, with a standby feed from motor control center MCC-B4

In the event of loss of power on one feed, the other will pickup the load through a reactor tie. If the loss is due to an overload, the reactor tie breaker will open. If the overload is on B substation it would not affect the emergency motor control centers. In the event power failed on 480 V Bus "A" or MCC-A3, electrically operated breaker #1 on bus station A5 will open and the electrically operated breaker #4 closes. This will feed power through MCC-B4 from substation B.

In the event that power from all sources has failed, breaker #4 will also trip automatically and call for power to be supplied from the diesel generator. This is accomplished by an undervoltage relay with a time delay to prevent momentary surges from causing unnecessary operation of the breaker.

If the power loss is due to an overload on breaker #1 or #4 the critical power will be supplied by the station battery.

7.3.1.2 Diesel Generators. Each diesel engine set has a single bearing AC generator rated at 150 Kw, 85% power factor, 480/277 volts, 3 phase, and 60 cycles.

Each control panel is equipped with: automatic voltage regulator, voltage regulator switch, manual field rheostat control, generator voltmeter, generator ammeter, frequency meter, and synchronizing lights. A loss of power on the emergency motor control center 480 volt bus will close a contact to open the fuel valves, and initiate the cranking cycle. This cycle is 15 seconds "crank" and 15 seconds "rest."

After the fourth unseccessful attempt it will actuate the diesel engine troble alarm on the annunciator. At that time it will call for set "B" to start.

After the diesel driven generator reaches normal voltage and frequency it will automatically close the associated circuit breaker on the emergency motor control center to transfer to MCC load to the generator.

The generators may be paralleled for a greater load carrying capacity.

7.3.1.3 DC to AC Inverter-Diverter. The DC to AC inverter-diverter motor generator is a two unit four bearing flexible coupled bedplate mounted set with a regulator mounted on the end of a DC generator.

Controls are provided on a separate panel. This panel contains all necessary controls for proper operation of the motor generator set as a battery charger, and the necessary automatic transfer and synchronization for operation in inversion. Table 7.3-1 lists pertinent information.

Table 7.3-1 Inverter-Diverter Design Data

AC Synchronous Motor		**AC Synchronous Generator**	
Horsepower	40	Kva output	18.75
RPM	1200	Power factor	1.0
Volts, AC	480	Volts, AC	440
Phase	3	Phase	3
Cycle	60	Cycle	60
Power factor	0.8		
DC Motor		**DC Generator**	
Horsepower	25	KW	25
RPM	1200	RPM	1200
Volts, DC	125/120	Volts, DC	140

Power failure is caused by either the under voltage meter which is set at 85% full load or the reverse current meter. This will drop out the breaker at the emergency motor control center MCC-B6. The M.G. set will then act as an inverter supplying power to the vital service bus. The control circuitry and instruments receive all its power from this bus.

When power is returned to the emergency motor control centers either by the substations or diesel generators, the under voltage relay will energize a time delay relay, thus protecting against false power return. After 60 seconds the time delay relay will energize the synchronizing circuit. When it is equal to and in phase with the voltage on emergency motor control center MCC-B6 the breaker will reclose.

7.3.1.4 Station Batteries. There are sixty 2 volt lead-acid type batteries with an output of 830 ampere hours at an 8 hour, 641 ampere hours at 3 hours, or 481 ampere hours at a 1 hour rate of discharge to 1.75 volts per cell.

Normally they will be "floating" on the DC bus with the generator carrying the equipment load. There is a silicon rectifier constant voltage battery charger unit for standby service.

The system is isolated from ground with a ground detector set to alarm at less than 60,000 ohms.

7.3.2 DEMINERALIZED WATER TREATMENT SYSTEM. This system, consisting of a "packaged" water treatment plant and a storage tank, supplies demineralized water to the thermal shield cooling system, the demineralized water experimental cooling system, the fuel storage pool, and a reservoir for laboratory service. Figure 7.19 is a flow diagram of the system.

Domestic water from NBS service facilities is supplied to a Hungerford and Terry treatment plant. The incoming water first enters an anthracite filter, then passes through an activated carbon filter to a mixed bed ion exchanger. Acid storage and mixing tanks, a sodium hydroxide mixing tank, a salt storage tank, and an installed water softner are provided for regeneration of the resin bed.

Effluent from the treatment plant is distributed to various systems or to an overhead 500 gallon storage tank. Instrumentation is provided to automatically control the level of the demineralized water storage tank and the spent fuel canal. Treatment plant instrumentation monitors effluent conductivity and totalizes flow for regeneration records and times.

7.3.3 COMPRESSED AIR SYSTEMS

7.3.3.1 General. This system consists of 3 inter-connecting sources of compressed air to supply the reactor and laboratory buildings, as well as emergency supply to the helium void shutdown system. Figure 7.20 illustrates the flow diagram of the integrated systems. Instrumentation is provided to annunciate on low pressure at various locations in the system.

7.3.3.2 Building Service Air. A source of 90 psig air is supplied from NBS plant services. This air line enters the cold lab basement of the building and is distributed to supply the laboratories with compressed air. This source also serves as a backup to the normal facility instrument air. This 90 psig air also is reduced to 50, 60, and 17 psig to supply ventilation control systems, various air ejectors and air conditioning butterfly valves.

7.3.3.3 Instrument Air. 150 psig dry instrument air is supplied by a single stage, 32.9 scfm Worthington air compressor. It is located in the cold lab basement of the building with local suction from the basement atmosphere. The discharge air is cooled by an after-cooler and passes through a moisture separator to a 150 psig air receiver. A backup supply to the 10 psig emergency air is provided from the compressor also. Air passes from the receiver through a dual Anders-Driline, reverse purge, air dryer.

It is distributed to various pneumatically operated valves, instruments, beam port shutter operators, and reactor building door seals.

7.3.3.4 Emergency Void Shutdown Air. A single stage, 10 psig, 106 scfm Worthington compressor supplies backup air to the helium void shutdown system. This supply is in turn backed up by the 150 psig air compressor as discussed above.

The 10 psig system also has an after-cooler, moisture separator, receiver, and air dryers similar to the 150 psig system; however, the dryers are sized to match the higher capacity of the compressor.

7.4 RADIOACTIVE WASTE HANDLING

Under normal operations the reactor will generate radioactive sources, irradiate specimens or samples, and in general both purposefully and incidentally activate many materials. It will ultimately be necessary to control the disposition of all such radioactivity according to the prescribed laws of handling and disposal. It is the responsibility of the reactor management to insure that proper disposal is continuously achieved. The assistance of the health physics staff is required and their responsibilities clearly specified to confirm that this achievement is realized.

The release of radioactive wastes to the environment will conform with Title 10, Chapter I, Part 20, Code of Federal Regulations, entitled "Standards for Protection Against Radiation." No radioactive materials will be released to the sewer which exceed in amount or concentration the applicable limits set forth in Part 20. The release of gaseous or airborne activity will likewise be in conformance with these protection standards. No solid wastes containing significant activity will be released except as a formal shipment of radioactive material to an authorized consignee. Packaging of such shipments will be in accondance with the ICC regulations. In general, it will be the policy of the reactor management to minimize to the extent of reasonable procedures the release of radioactive materials in all cases, considering the permitted release always as an upper limit to the desired release.

It will be the responsibility of each individual experimenter using the reactor facility to consider the possible radioactive waste disposal problem that he creates with his experimentation. It will be his responsibility to keep to a minimum the volume of waste generated by reasonable design. He will be instructed and assisted in this matter by operations and health physics personnel to the extent necessary. In addition, his experiment and its design must be approved from this point of view by the reactor safety committee.

7.4.1 PHILOSOPHY OF WASTE CONTROL. As indicated in the facility description, all areas other than the cold laboratory and the general administrative and office space will be treated as potentially contaminated and a source of radioactive wastes. The general philosophy of the management of these suspect areas will be to confine and concentrate all high level activities. Dilution and dispersion will only be permitted in limited cases where reasonable volumes of dilutant are indicated and concentration or confinement is impractical.

The laboratories in the warm area will be the principle sources of wastes. The reactor operations will generate large volumes of solid wastes but the confinement of these is more obvious and easily managed. Each occupant of a warm area laboratory will be expected to confine his radioactivity first to his immediate experimental arrangement, second to his laboratory, and last to the general warm area. Each laboratory will be provided with survey equipment during the course of an experiment so that all exits of personnel or material from the laboratory can be monitored to limit contamination spread and prevent loss of control of wastes. No disposal system or drains will be provided in the separate laboratories for radioactive waste disposal. The drains which are provided for the warm area will be treated as cold drains but all liquid waste in the drains will be monitored and held in a system as described in Section 7.4.3 for dilution or treatment in the event of contamination.

Facilities will be provided in one hot laboratory, room B-154, for concentration of high level liquid waste prior to formal shipment to an authorized consignee. These facilities will probably include evaporators, condensers, filters, ion exchangers, and other means for reducing the liquid waste to a suitable solid form.

High level solid waste such as in-pile experimental apparatus, ion exchange resins, filters, solidified liquid waste, etc., will be stored in shielded containers prior to ultimate disposal.

Low level wastes, other than liquids, will be segregated and reduced in volume by baling or incineration whenever possible. Low level liquid wastes will be diluted and dispersed as indicated.

7.4.2 SOURCES AND PHYSICAL FORMS OF WASTE. Under normal operations the reactor will produce a variety of radioactivities, all of which are contained except for argon 41. Estimates have been made of the production rates of A^{41}, N^{16}, H^3, O^{17}, O^{19}, and C^{14}.

It has been shown that none of these activities created credibly dangerous conditions for on-site or off-site personnel. The following describes the calculated estimates and the nature of the problem associated with each activity.

The only possible radioactive airborne effluent from the reactor facility under normal operating procedures should be the argon (A^{41}) produced by neutron activation of air in the voids within the biological shield of the reactor. The warm laboratory area can be a source of contaminated airborne effluent but this is of a type which is filterable. Adequate filtering and filter monitoring are included, together with automatic shut-off of effluent in the event of excessive levels. In the normal steady state only the non-filterable A^{41} activity will be released. The quantity of argon produced and released was estimated assuming all beam tubes to be filled with air exterior to the thermal shield. This is a conservative assumption since this volume is normally at least partially filled with beam collimators and shielding. The argon produced in the air in the thermal column was also calculated. The biological shield design is made with provision for a system to exhaust all air from the ports and the thermal column through the main stack. The beam tube volume interior to the thermal shield and main reactor tank is filled with CO_2 and is not a source of argon.

The number of curies/sec discharged by the stack will be 9×10^{-6}. When this number is used to calculate an upper limit to the concentration at the top of the stack taking into consideration the dilution afforded by the fan EF-2, it has been shown that the concentration is less than 10 times the limit set forth in 10 CFR, Part 20 for off-site unrestricted areas. A calculation was made of the further dilution that can be expected when this effluent undergoes turbulent mixing in the atmosphere. Sutton's equation* was used to indicate the magnitude of this dilution. A steady turbulent diffusion condition was assumed; i.e., transient conditions, such as "looping" or "Fumigating," which can lead to higher ground level concentrations than average were not considered. These conditions would not be expected to persist and thus would not lead to smaller time-averaged dilution factors. The calculations are conservative in that there was no averaging over all wind directions. The micrometeorological constants needed for this calculation have been discussed in Section 2.2.7.

The results indicate that the concentration of A^{41} at ground level directly downwind from the 100 ft. stack ** is 2×10^{-4} curies/m^3 or 2×10^{-10} curies/cm^3, at the nearest site boundary, a distance of 400 meters from the stack. This is a factor of 250 below the "maximum permissible concentration" for unrestricted areas without taking into consideration the additional dilution to be expected when averaged over all wind directions.

Since there is a varying concentration in the diffusing cloud, it is of interest to calculate the gamma ray dosage at ground level due to passage of the effluent directly overhead. For this calculation the case of inversion with a minimum of diffusion may be considered. The example actually calculated is that of an infinitely long line source at the 100 ft stack height directly over a ground level point. Assuming a wind velocity of 2 meters/sec carrying the source of 9×10^{-6} curies/sec into a line source, this calculation yields a dosage of 3×10^{-7} r/hr. A buildup factor for multiple air scattering is included. This is conservative in that it should also have been time averaged over all wind directions.

An estimate has been made of the activity to be expected in the D_2O coolant flow exterior to the main tank. The main source of activity under normal conditions is from the decay of N^{16}. N^{16} is produced in the main tank from the reaction of O^{16} (n,p) N^{16}. The neutron threshold energy for this reaction is above 10 Mev. It was assumed, therefore, that all collisions other than those leading to an (n,p) reaction lead to removal of sufficient neutron energy to bring a neutron below the threshold. The cross section for the production of N^{16}, when averaged over the fission spectrum above threshold, was taken to be 4×10^{-29} cm^2 per O^{16} atom. The production rate of N^{16} atoms was then taken to be the product of the

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* For a discussion of this equation and its applicability see Sections 2.2.7 and 13.6.

** The stack height was taken as the physical height; i.e., no credit was taken for an increased effective height which might result from effluent flow up the stack. Sutton's equation applies to a point source of effluent and no credit was taken for mixing of air within the stack or for enhanced dilution which might result from immediate expansion of the plume at the top of the stack.

fission rate and the ratio of the mean (n,p) cross section to the total cross section in D_2O averaged over the fission spectrum. This yields for 10 Mw operation a production rate of 10^{13} atoms/sec. The mean life of a N^{16} atom in the core is determined by the decay rate and the mean time to leave the core due to coolant flow. The latter, to a certain degree, is uncertain without test but the approximate mean time of a D_2O molecule in the tank can be calculated from the coolant flow. The resultant disintegration rate of N^{16} in the coolant piping is calculated to be 50 curies for every 2.5 ft of piping.

Calculations were made for activities due to other reactions such as $O^{17}(n,p)N^{17}$ and $O^{18}(n,\gamma)O^{19}$. The D_2O activity resulting from these activities is negligible compared to that due to N^{16}.

Other possible activities to be found in the D_2O due to the corrosive products are less amenable to calculations, especially long period activities such as Na^{24} which results from the (n,α) reaction on aluminum. Experience at ORR has been reported from which the order of magnitude of the activities with which the NBSR must contend can be judged. The problem is not one of providing adequate shielding since this is determined by the N^{16}. The activities to be expected in the demineralizers are dependent upon these longer period activities. Provisions for shielding the ion exchanger column resins has been included.

Conservative calculations of the activities to be produced in the CO_2 blanket within the thermal shield show that $O^{16}(n,p)N^{16}$, $O^{17}(n,p)N^{17}$, $O^{18}(n,\gamma)O^{19}$, and $C^{13}(n,\gamma)C^{14}$ produce activity levels which are small compared to argon. The only long half life activity is that due to C^{14}. If it is assumed that 10% of the irradiated CO_2 leaks per day and enters the irradiated air system before dilution in the stack exhaust system, the calculated concentration at the top of the stack is less than 10^{-13} μc per cm^3. This is at least a factor of one million below the levels set by 10 CFR, Part 20.

Calculations of the concentrations of tritium which are produced by the reaction $D(n,\gamma)T$ in the heavy water have been made to determine the magnitude of the tritium hazard associated with the NBSR. The latest calculations using the most recent neutron flux calculations and including the production of tritium in the moderator as well as the core show that at saturation after many years of operation the total inventory of tritium will be in the range of 190,000 curies. If the total inventory should be spilled, the fraction which would be evaporated and be available for release to the stack is of the order of 0.1% of the total; i.e., approximately 200 curies. A calculation shows the concentration of this tritium in the form of tritiated heavy water vapor to be less than 5 times the maximum permissible concentration of 10 CFR, Part 20, at the top of the stack before atmospheric dispersion. If the latter dilution is also considered, the concentration at ground level would be reduced to the order of 2% of the limits of 10 CFR, Part 20.

As far as normal or routine operations are concerned, experience has shown that for heavy water research reactors, primary coolant leakage can be maintained at extremely low levels. Construction specifications for the NBSR have called for stringent verification by helium leak testing of the integrity of the primary system. An extensive system of leak detection is provided which is capable of detecting leaks of the order of one cubic centimeter per hour. A steady leak of this rate evaporating into the building air would lead to concentrations less than one percent of the levels of 10 CFR, Part 20, in the process room.

A calculation has been made of what is probably the most pessimistic non-accident situation, namely when the reactor is open for refueling or experiment change. If one assumes that a 4" diameter port is open and that the effective free heavy water surface is at the floor level of the operations area (the heavy water level is actually at least 5' below the floor level at the bottom of what would be effectively a 4" diameter tube), one can estimate an equilibrium average concentration of tritiated heavy water vapor at that level. This concentration is at least a factor of two less than the levels of 10 CFR, Part 20, under these assumptions. In any case, tritium monitors and careful special ventilation control will be effected during such operations to assure that maximum permissible concentration levels are not exceeded.

7.4.3 ROUTINE WASTE HANDLING PROCEDURES. Since solid, liquid, and gaseous wastes are generated during the operation of the NBSR, these wastes must be removed and disposed of according to the State of Maryland and U.S.A.E.C. regulations. The following describes the

routine methods of handling such wastes and is excerpted from the Radiation Safety Manual for the NBSR.

Solid radioactive waste will be packaged according to I.C.C. regulations and will be routinely picked up for disposal by the U.S. Army Chemical Corps, Edgewood Arsenal. Waste cans marked, "Radioactive Waste" and having plastic liners will be placed throughout the reactor and laboratory work areas. These cans are for low level (less than 25 mr/hr) waste only. The liners are to be closed and taped when the liner is filled. They are then to be placed in 40 gallon fiber drums with further liners which are heat sealed by Health Physics. The drums are then sealed and stored for shipment. High level (greater than 25 mr/hr) solid waste must be stored for radioactive decay or shipped in shielded containers.

Gaseous wastes have been discussed in describing the ventilation system, Section 3.6. Before being released to the atmosphere from the 100 ft NBSR stack, all gaseous wastes are passed through roughing and high efficiency (99.97%) filters for the removal of airborne particulates. Continuous monitoring of the effluent stream and particulate filters is designed to permit building closure whenever gaseous effluent exceeds prescribed radioactivity levels. An internal recirculation system containing both high efficiency particulate and activated charcoal filters can be started to clean up any gaseous activities detected.

Liquid waste will be maintained as concentrated contained liquids and will be stored in capped polyethylene bottles inside fume hoods until a volume of about 2-1/2 gallons is reached. The wastes then can be treated in numerous ways depending on the radiation levels and chemical forms. Low level liquid waste will be absorbed in a material such as vermiculite which will then be encased in plastic containers and treated as solid waste. It is expected that a treatment facility will be established in room B-154 in the hot laboratory section of the warm laboratory wing. This facility will include provision for evaporators, filters, condenser, ion exchange resins, etc., to effect change of high level liquid waste to suitable solid form before shipment by an appropriate consignee.

A design to effect the liquid waste control philosophy stated above has been developed. As previously stated, the purpose of the system is to protect the drainage system from the reactor and warm laboratory areas from accidentally discharging to the main laboratory sewer significant levels of radioactive liquid waste. All drains from this area are treated as suspect of radioactivity although none of these drains are permitted to be used for waste disposal purposes. There should be no reason to expect other than small accidental spills to any of this system.

All suspect waste drains lead to a monitor and holdup volume as shown in Figure 7.18. Flow rate in this system will be monitored by a variable area flow meter. Valve RWV-2 will be normally open while valve RWV-3 will be normally closed. A holdup tank which permits a 3 minute delay before discharge at the maximum estimated flow rate of approximately 60 gpm is included before valve RWV-2. If the counting rate of monitor RD4-3 is integrated over the delay time associated with the holdup tank, a decision as to the radioactive content of the effluent can be made with reasonably good statistics to give assurance that maximum concentration levels are not being exceeded. If pre-set levels are exceeded, RWV-2 will close and RWV-3 will open to cause all effluent to be diverted to the 1000 gallon retention tank. The retention tank is sized to give approximately 15 minutes for operations personnel to effect a stoppage of all effluent to the system even under the unlikely condition of maximum flow of approximately 60 gpm. Normally it would be expected that less than 1000 gallons of effluent will have been accumulated that is contaminated.

The contents of the retention tank can be sampled by means of an ejector system located at the hot waste control panel in the cold lab basement where the radiation monitor and flow meter indicators are located. Controls of pumps and valves are located at this panel to permit transfer of the contents of the retention tank to either a 5000 gallon holdup tank for storage, and/or dilution, and subsequent radioactive decay, or to one of either of two batching tanks. The contents of the batching tanks can be sampled before pumping to room B-154, the hot waste treatment area in the hot laboratory. Tank levels and tank transfer controls can be effected from panels in either area 057 or B-154. Finally, the contents of the 5000 gallon holdup tank can be sampled after which its contents can be returned to the effluent control system and discharged to the sewer. This latter procedure would be accomplished only after safe radiation levels have been achieved; but in any event, the discharge of this tank would be treated like any suspect effluent and counted by detector RD4-3.

All valves and pumps in this system are stainless steel, and all piping is stainless steel acid resistant glass or unplasticized polyvinyl chloride, schedule 80.

7.4.4 ACCIDENT CONDITIONS. In general, an analysis of the activities to be undertaken at the NBSR show that there is no reason to expect a major radioactive liquid spill. The most serious condition would be a rupture in the main coolant system which accompanied a major core melt and release of fission products to the coolant. Such a spill would be confined to the basement of the reactor building. The basement, itself, is sealed and the entire building is designed to minimize leakage, particulate, gaseous, vapor or liquid. Provision for collecting and pumping from a sump to a storage tank is provided for any spilled coolant. The ultimate clean-up and disposal from such a spill would be a major undertaking. The off-site consequences of this accident are discussed in Section 13.

As previously discussed in Section 7.4.2 a calculation has been carried out to estimate in a conservative manner the consequences of a primary coolant spill which released only tritium or tritiated D_2O to the reactor basement area. The tritium hazard to the public from the operation of the NBSR is negligible.

The following discussion is meant to apply to the question concerning an accidental spill of soluble radionuclides upon the ground exterior to the reactor building. At the outset it should be kept in mind that the circumstances of a release of radioactivity to the ground is most likely delimited in the following way. Because of the nature of the reactor containment and the "worst possible accident" there appears to be no way in which large quantities of fission products can be "spilled" on the ground in soluble form except in the remote event of the core meltdown, the consequences of which are discussed in Section 13. The fuel elements which are alloys of aluminum are unchanged in physical form during their entire history at the reactor. There seems to be no reason to suppose that an irradiated fuel element could be dissolved and spilled on the ground at the site. The nature of a radioactive spill, if it occurs, must be the result of loss of control of some soluble compound being studied in the hot laboratory. By the very nature of such experiments it would, therefore, be limited to a spill of magnitude no greater than a few tens of curies, the level of activities that can be processed in the hot laboratory. No experiments involving liquids having higher activity levels than this are planned nor are they possible to handle under the present design.

Should such a spill take place, there is the possibility that some of the radioactivity could reach the ground water. Appendix II of NBSR 7C (2.4) consisted of an analysis of the nature of such an event. In summary it was stated that (a) the sorptive properties of the rock and soil at the NBS site would tend to retard the flow of some types of radioactive contaminants that might accidentally reach the ground-water system, perhaps reducing the velocity to a few percent of the ground water velocity, (b) hydraulic dispersion should produce a dilution effect thus providing an additional factor of safety, (c) the movement of ground water itself is probably slow. It should be noted, furthermore, that the earth cover above the ground water plane is an average of ten to fifteen feet thick and constitutes an additional retarding barrier to such an event. Adequate time should be afforded for any number of emergency procedures including the boring of a well as is suggested in the above noted report or the removal of sufficient earth to recover the spill.

The recommendations of the report of the U. S. Geological Survey cited above as Appendix II of NBSR 7C (2.4) have been and are being followed concerning ground water sampling at strategic locations. There is every reason to be confident that ground water contamination can be monitored to provide assurance that no person's drinking water ever exceeds maximum permissible concentrations of radioactivity due to operations undertaken at the NBSR.

The following program for surveilling a possible increase in radioactivity in the ambient air and vegetation surrounding the NBSR site has been undertaken by the Health Physics Section of the NBS.

A Nuclear Measurement Company AM-3A Air Monitor has been operating since January 1965 at a location about 1000 yards remote from the reactor stack. The monitor records airborne particulate contamination deposited on a moving filter and gaseous radionuclides flowing through a shielded hold-up tank. The concentration of natural radon and thoron and their progeny in the air (as determined by the moving filter monitor) varies from 1.2×10^{-9}

$\mu Ci/cc$ to 2.4×10^{-8} $\mu Ci/cc$, depending on weather conditions. The gaseous monitor, with 3" of Pb shielding, has a sensitivity of 2.5×10^{-7} $\mu Ci/cc$ for 0.5 Mev β emitters. Several types of portable air samplers and associated counting equipment are also available in case of a suspecded release of radioactivity to a location away from a permanent monitor. A program for monitoring of the integrated gamma dose with Radiation Detection Company XBG film badges around the complete perimeter of the Bureau grounds was initiated in April 1965.

Radiochemical analysis for Strontium-90 in ashed grass samples from the National Bureau of Standards grounds was made in the autumn of 1964. Such studies will be conducted periodically by the Health Physics Section to test local foilage, soil, milk, etc. for assimilation of various nuclides.

Section 2.3.3.2 gives further details of environmental monitoring studies undertaken by the Health Physics Section.

Reference

(7.1) J.F. Wett, Jr., "Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air," ORNL 2892 (1959).

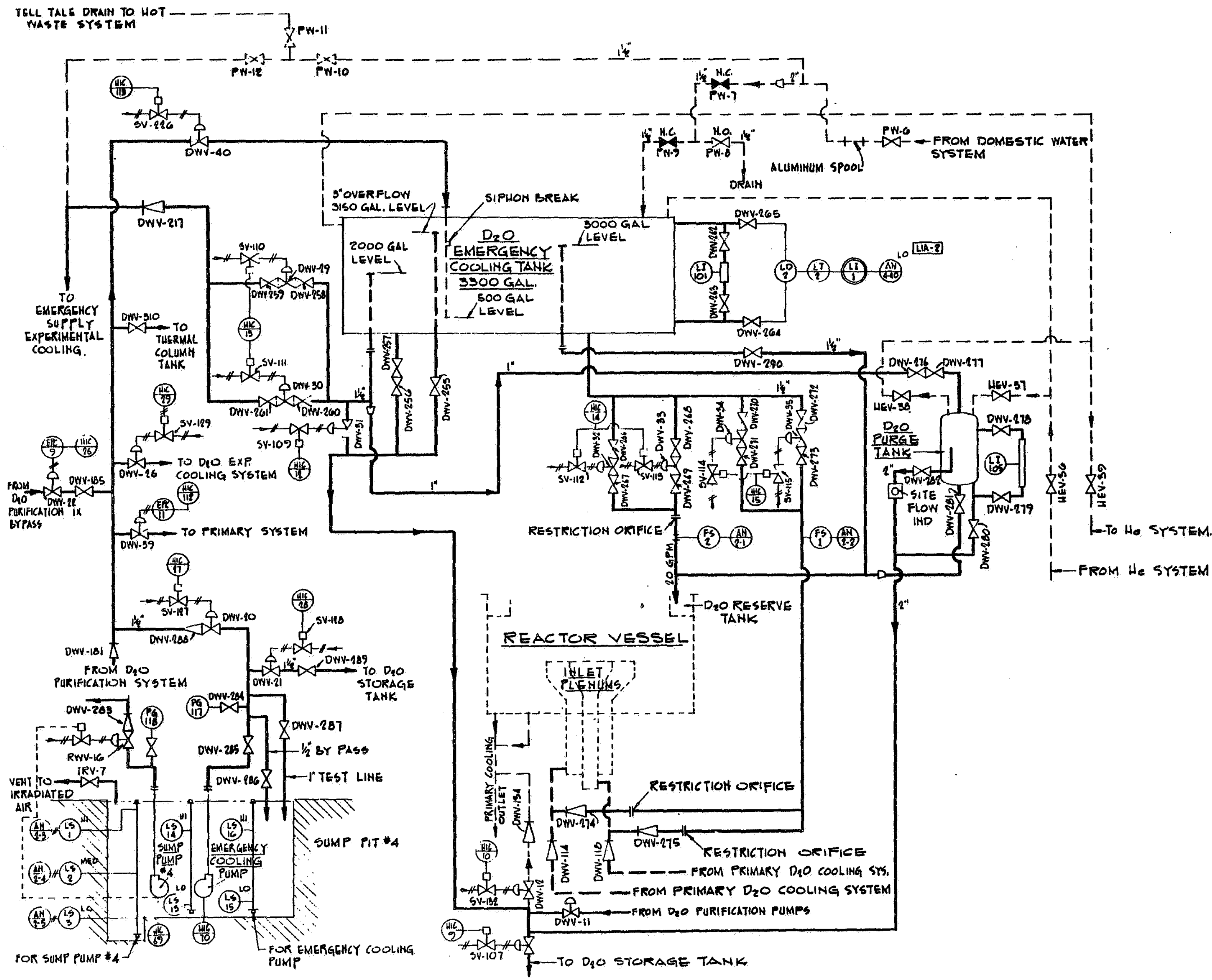

Figure 7.1 Emergency cooling flow diagram.

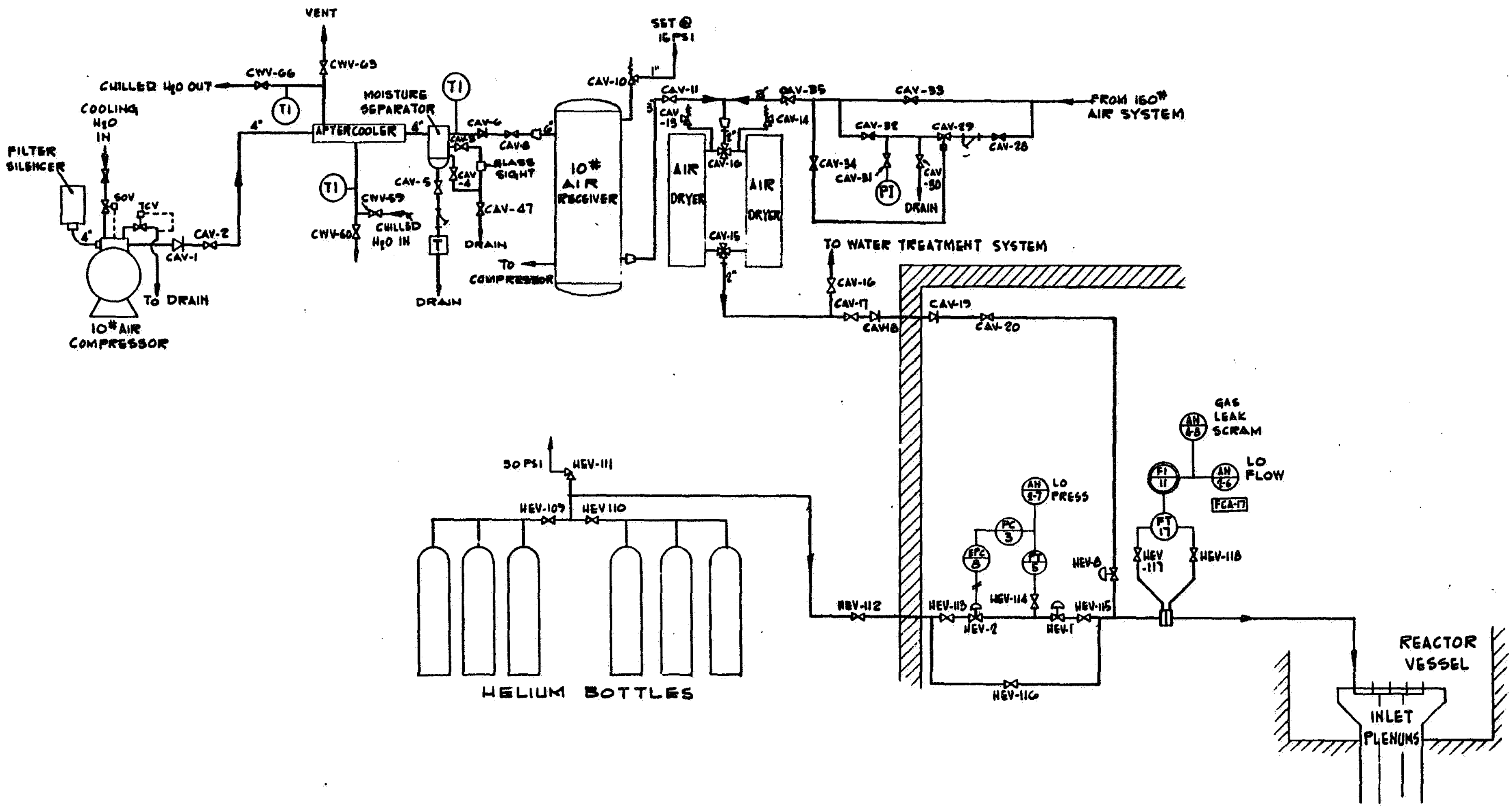

Figure 7.2 Void shutdown flow diagram.

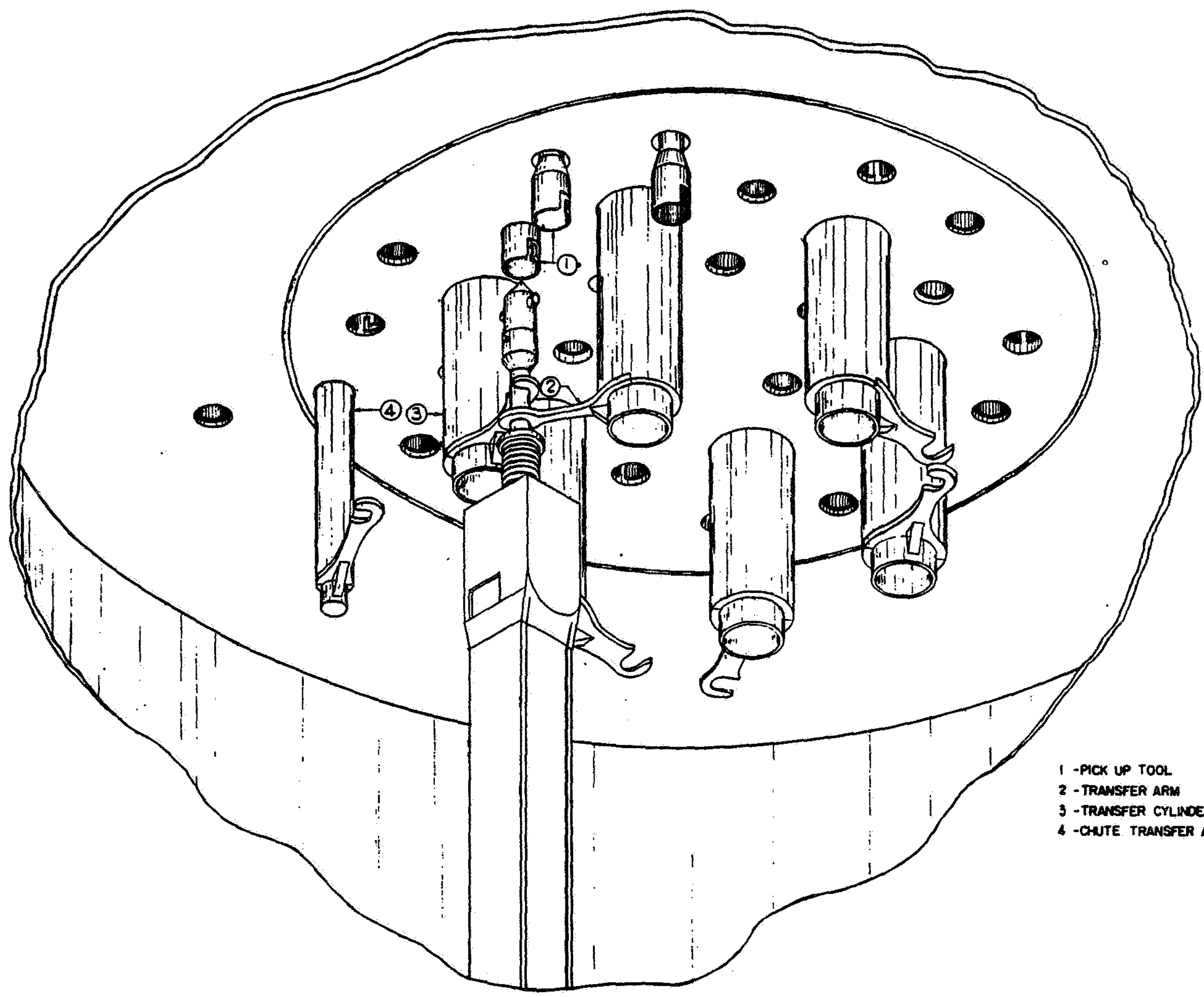

Figure 7.3 Isometric of fuel transfer system.

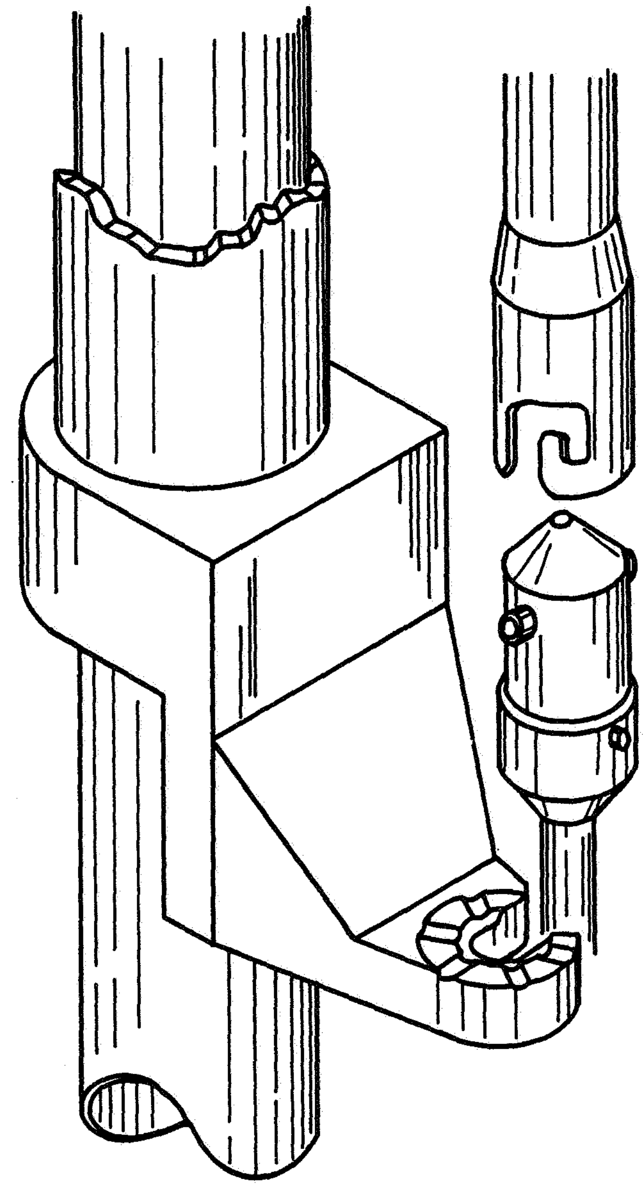

Figure 7.4 Details of transfer arm, pickup tool, and fuel element pickup head.

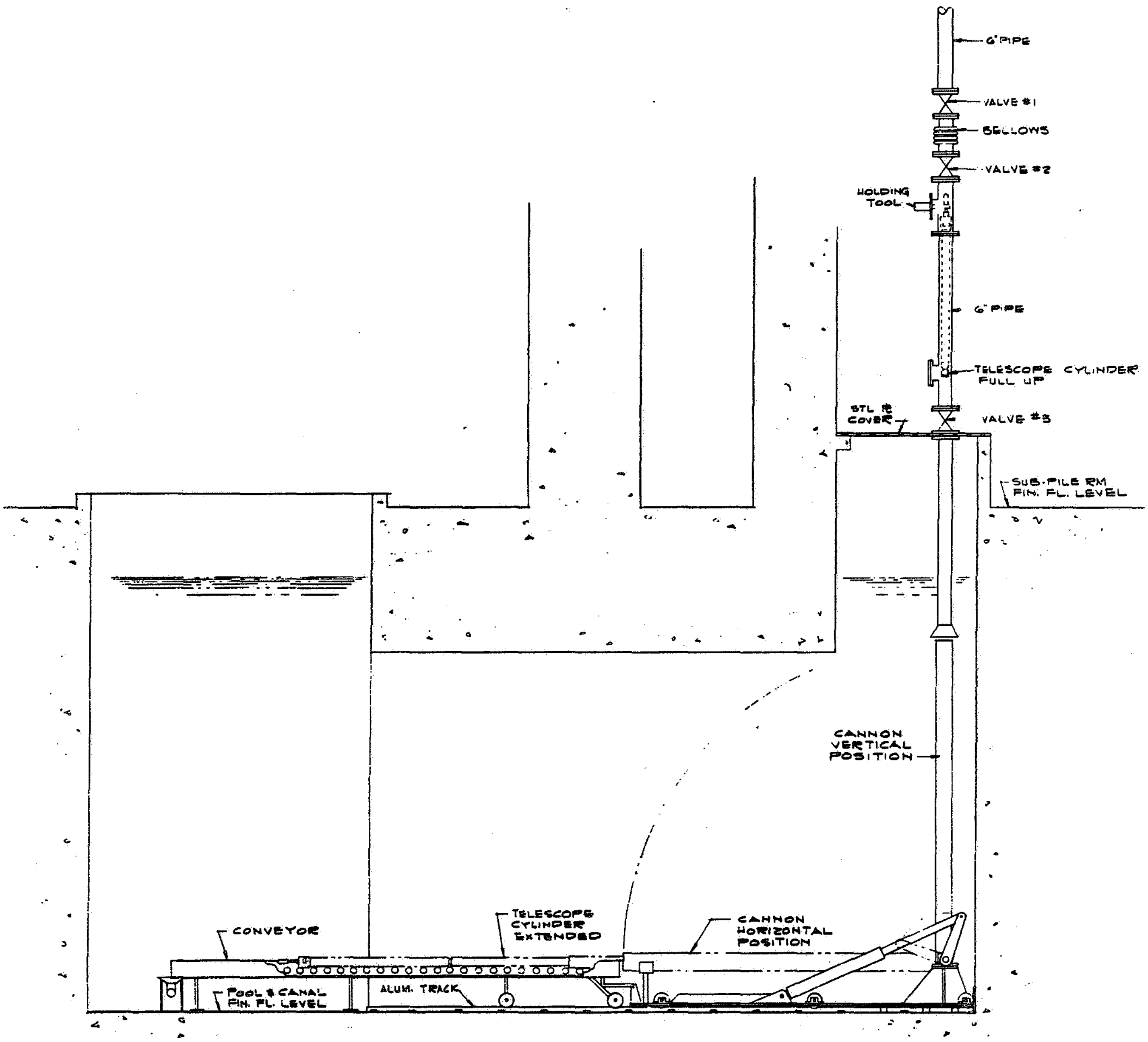

Figure 7.5 Fuel transfer receiving mechanism.

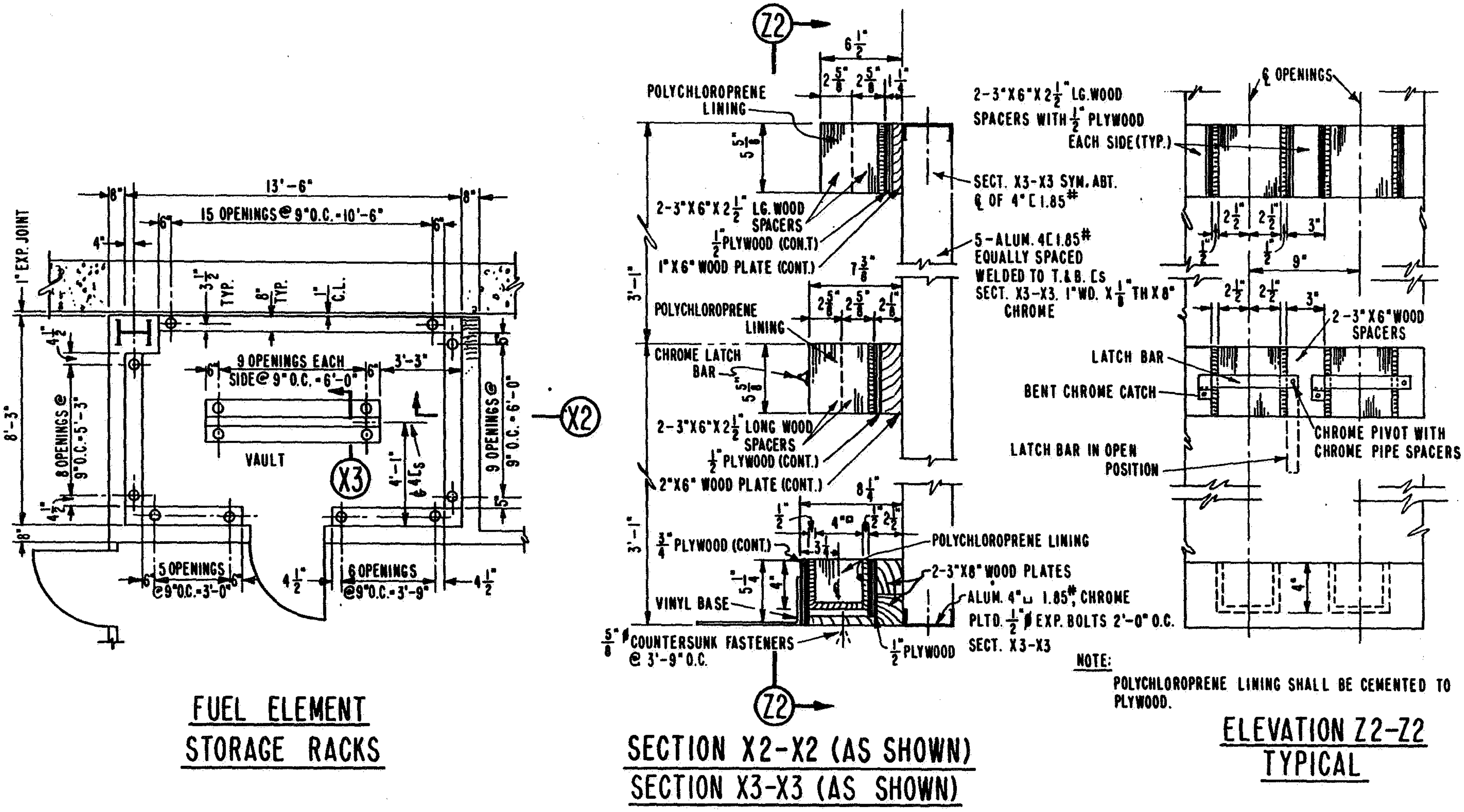

Figure 7.6 Unirradiated fuel element storage facility.

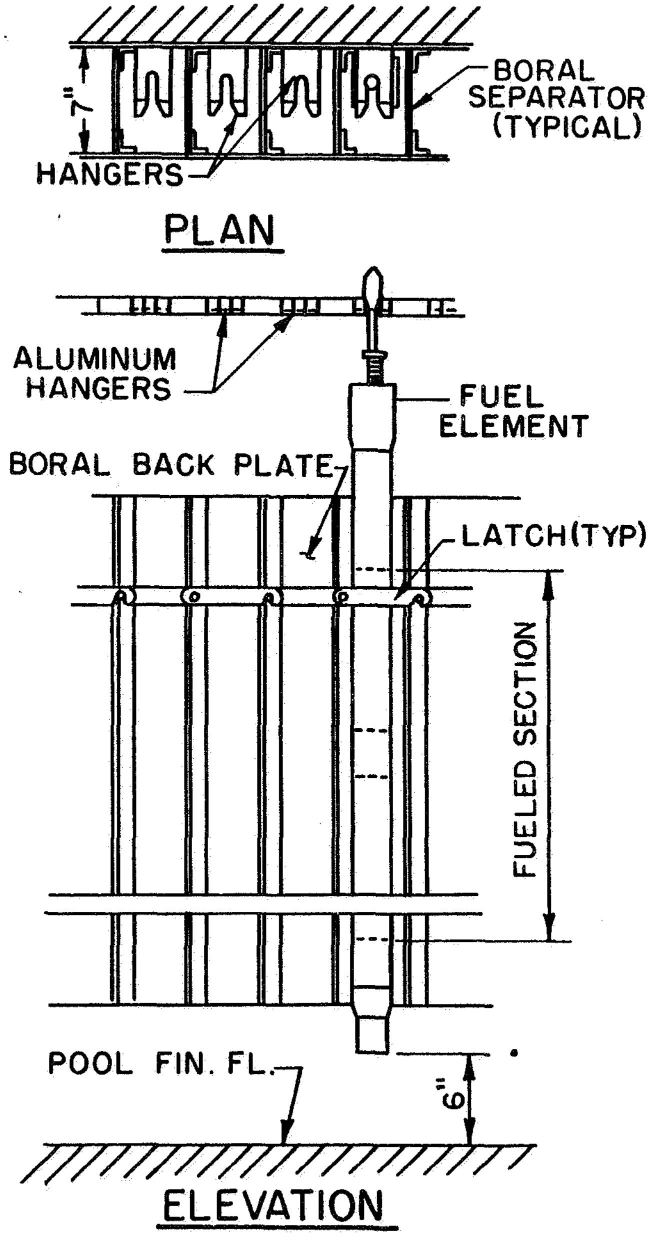

Figure 7.7 Uncut spent fuel element storage rack.

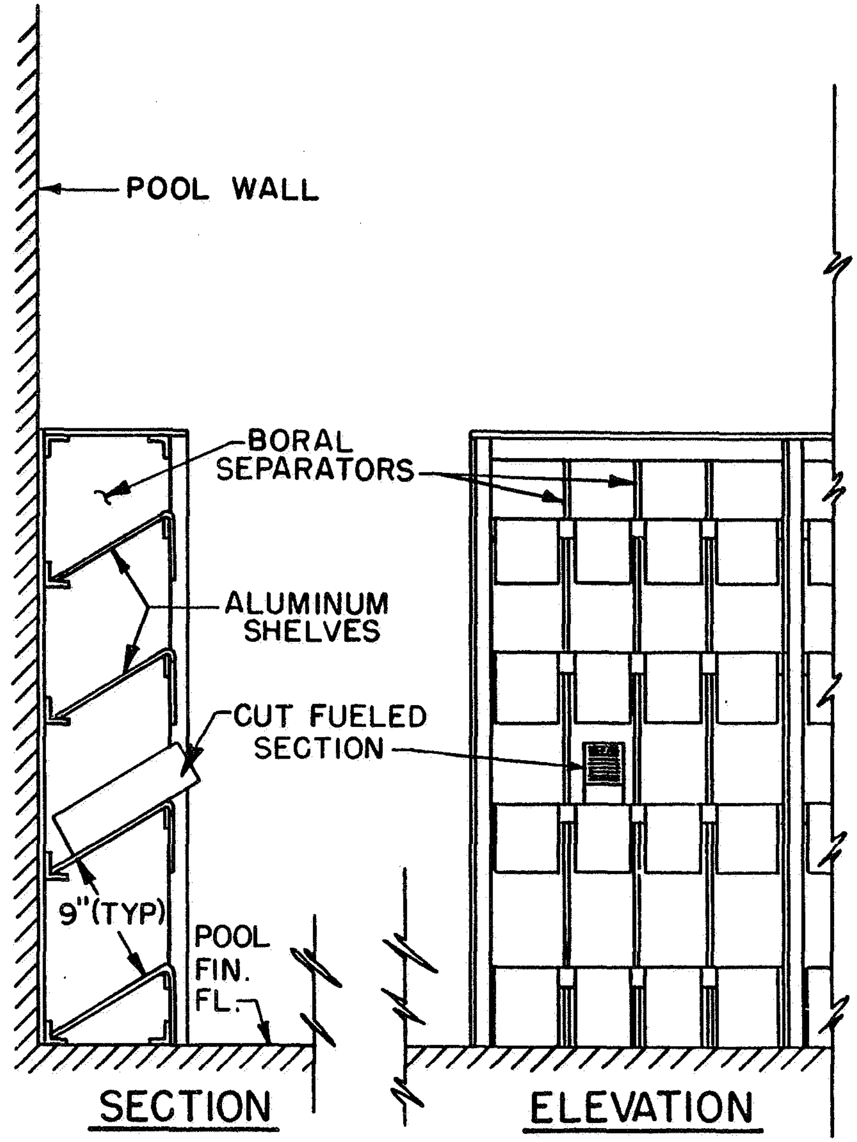

Figure 7.8 Spent fuel element storage baskets for cut fueled sections.

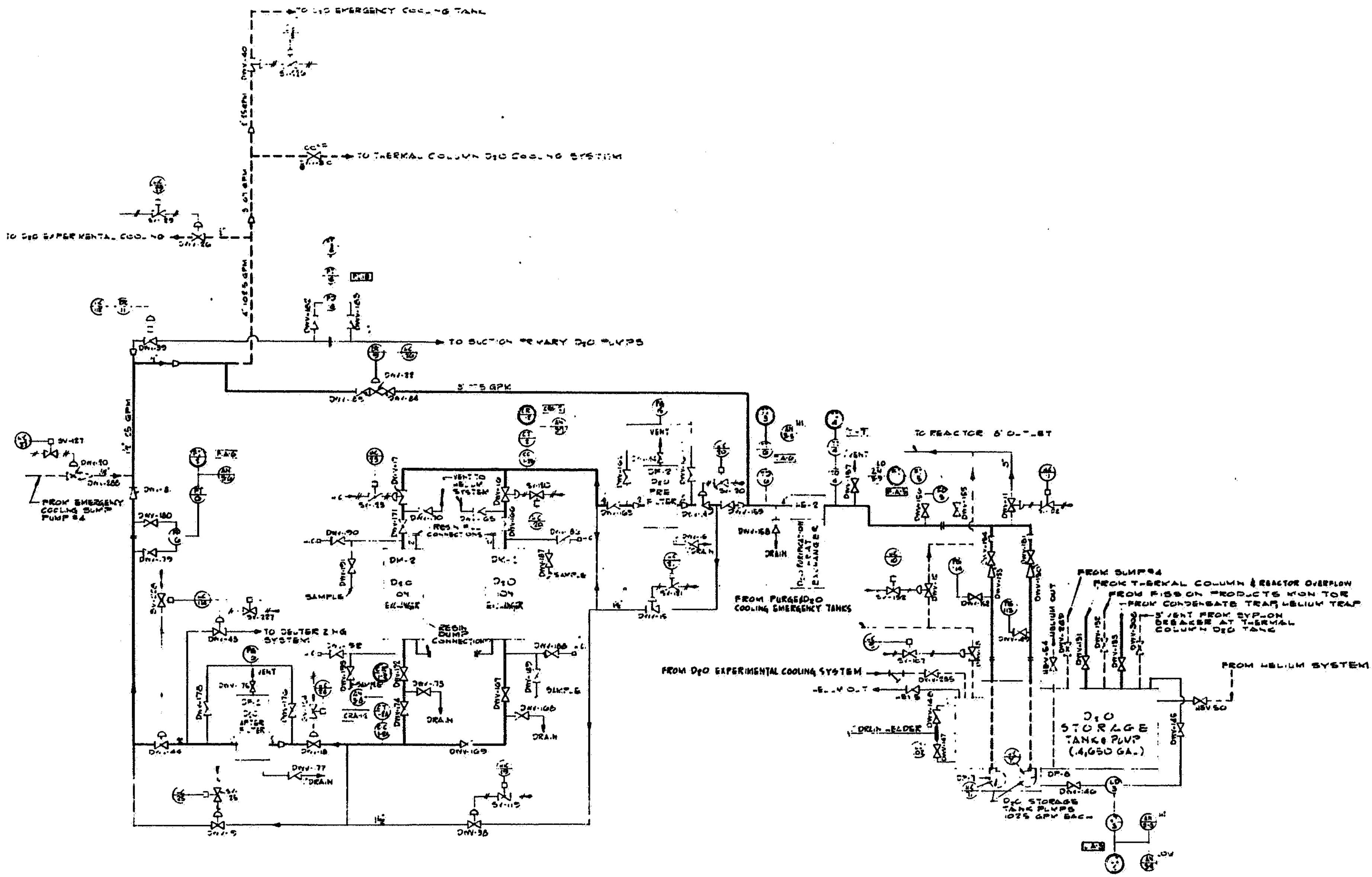

Figure 7.9 Primary purification system.

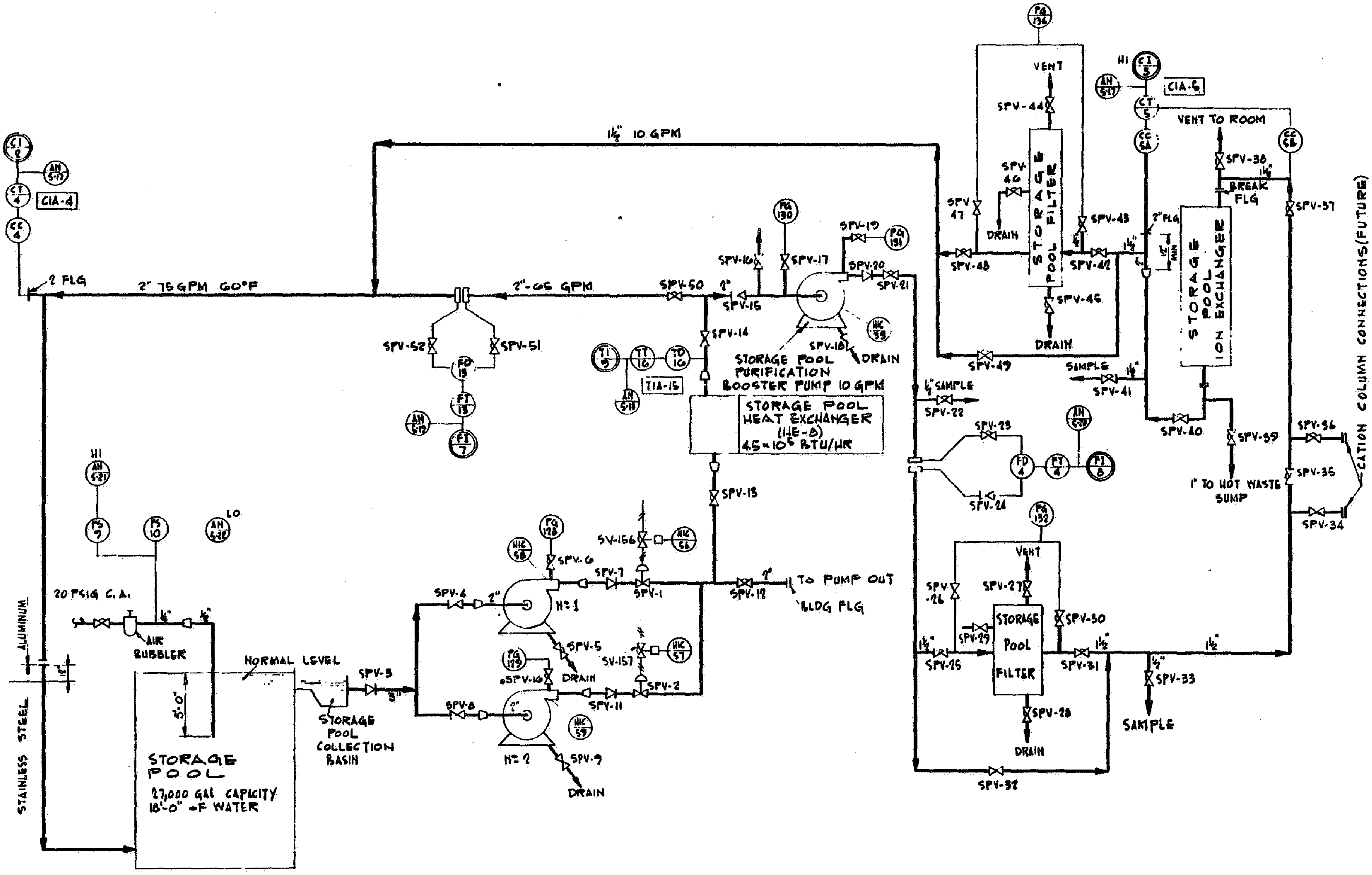

Figure 7.10 Storage pool cooling system.

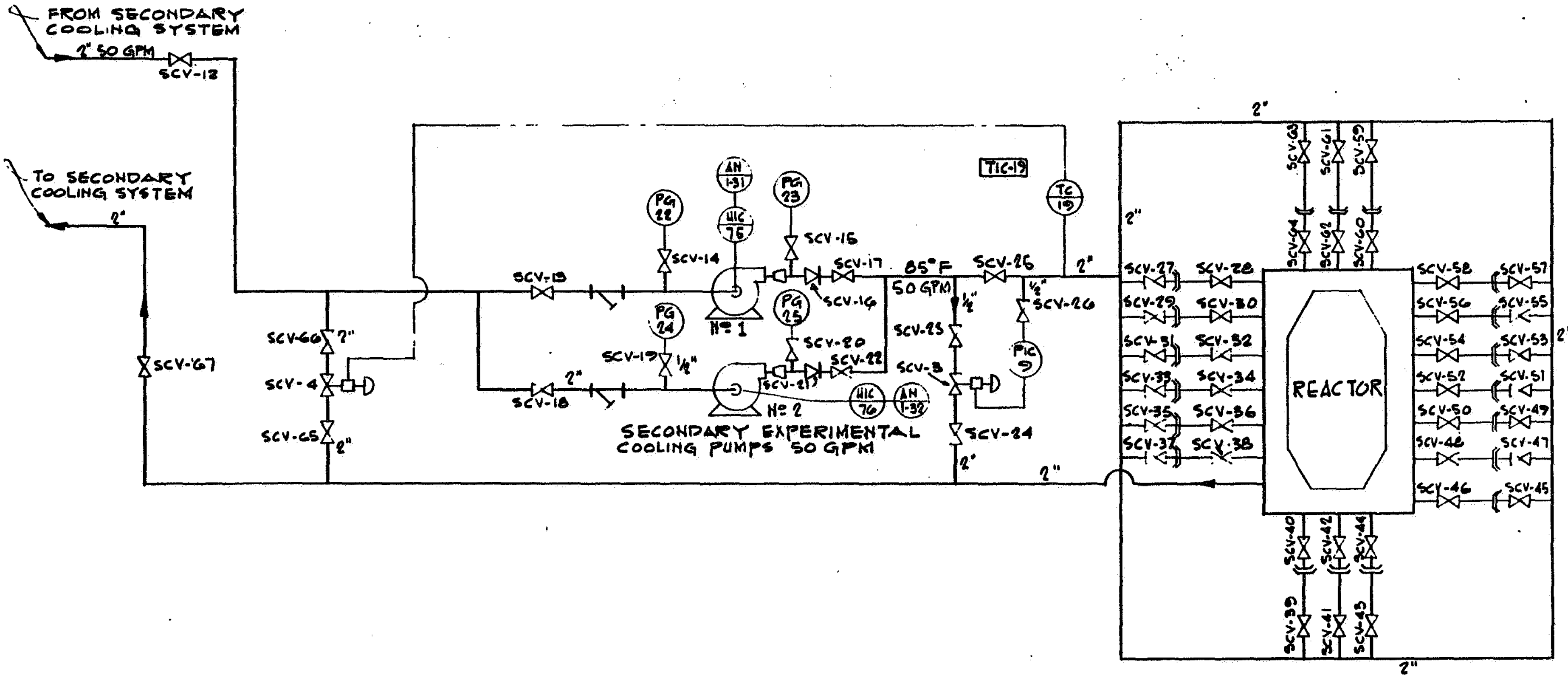

Figure 7.11 Secondary water experimental cooling system.

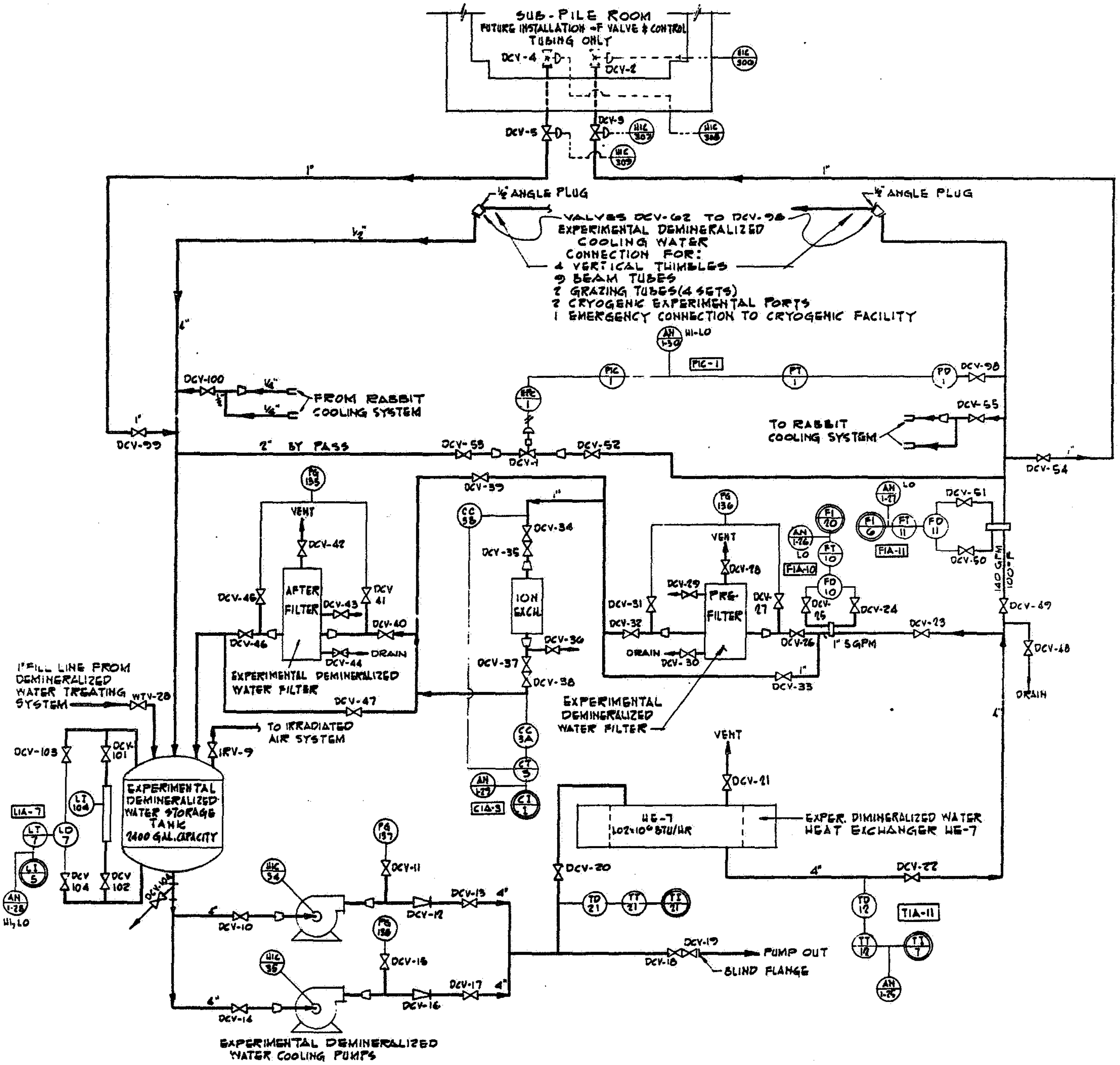

Figure 7.12 Demineralized water experimental cooling system.

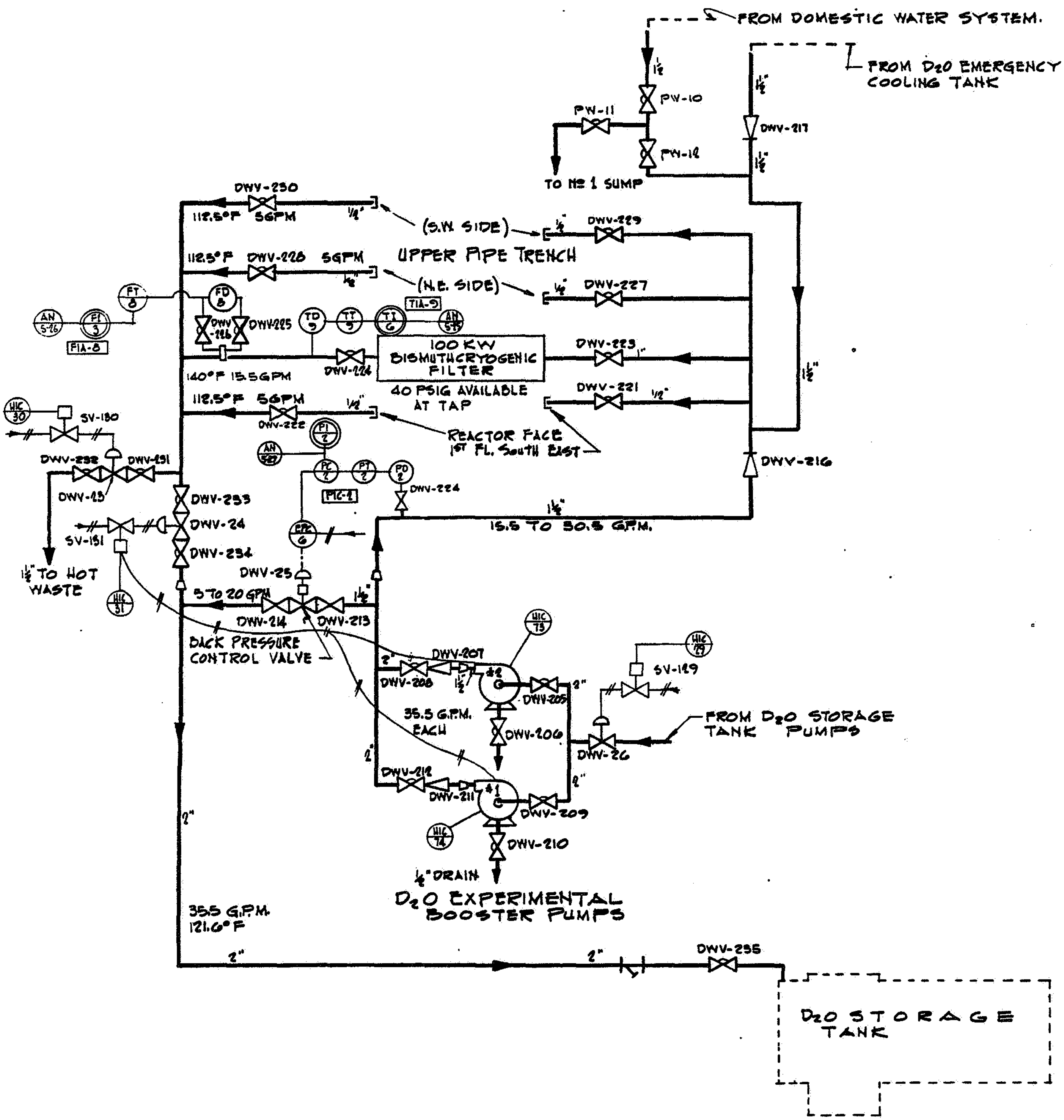

Figure 7.13 D_2O experimental cooling system.

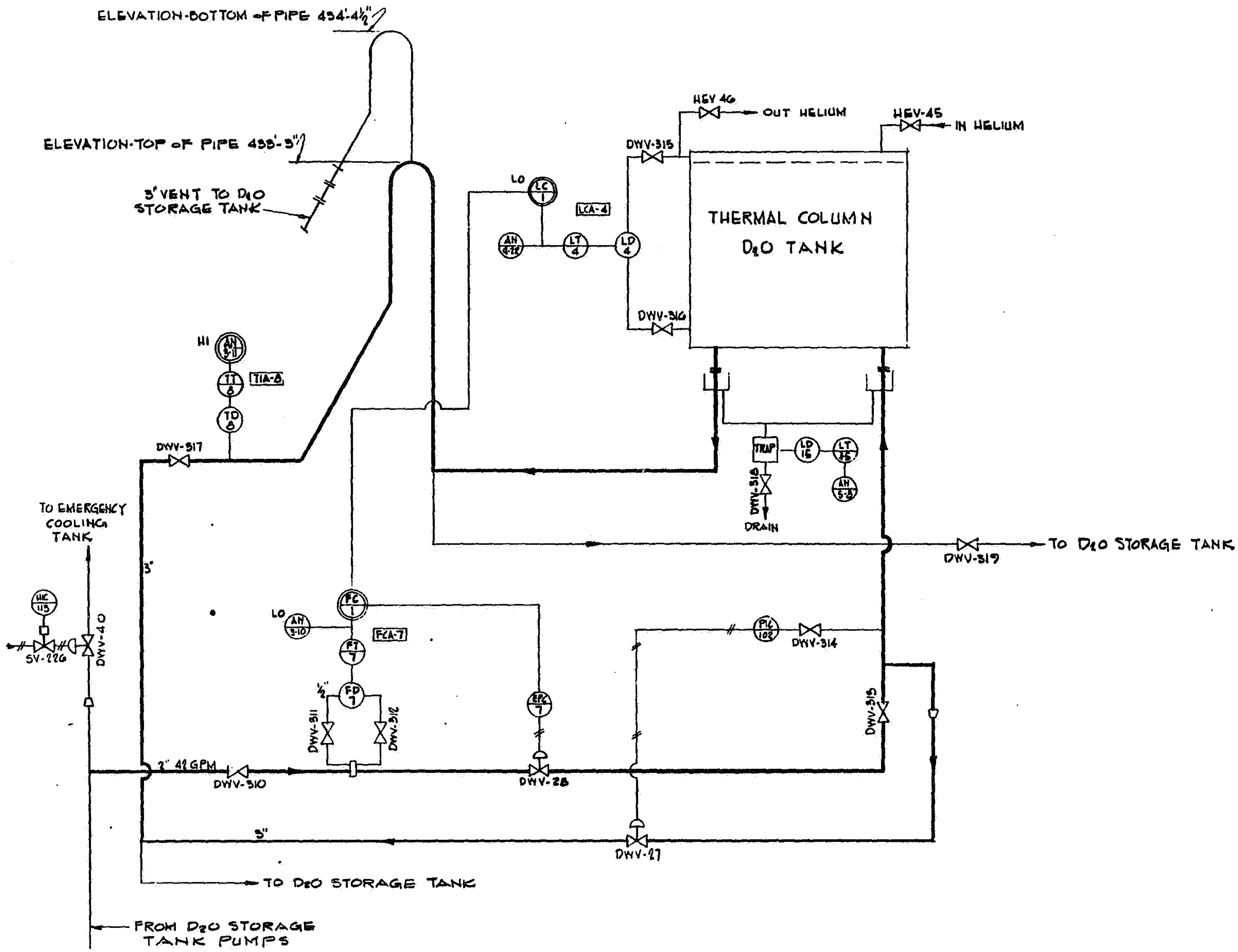

Figure 7.14 Thermal column tank cooling system.

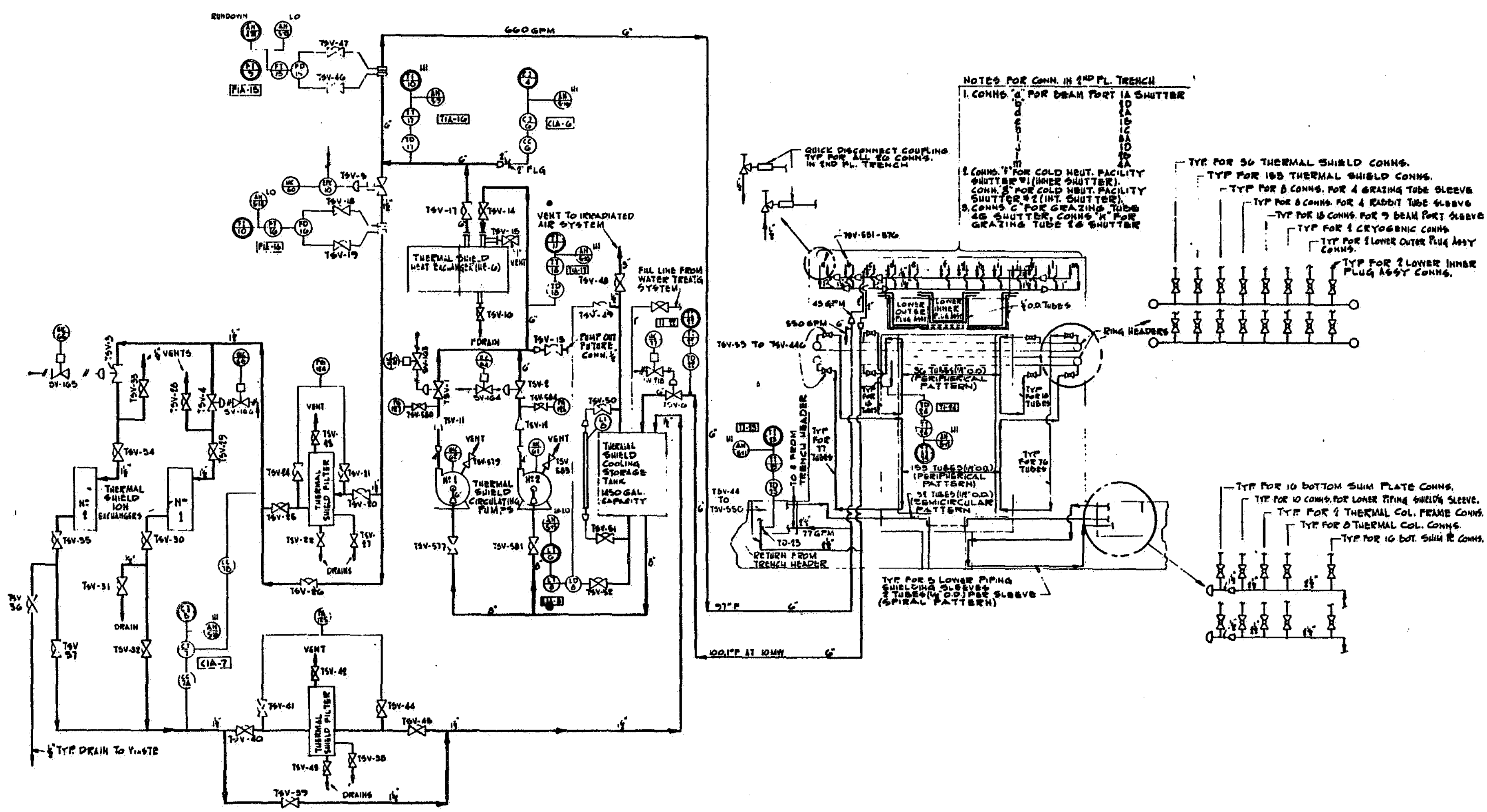

Figure 7.15 Thermal shield cooling system.

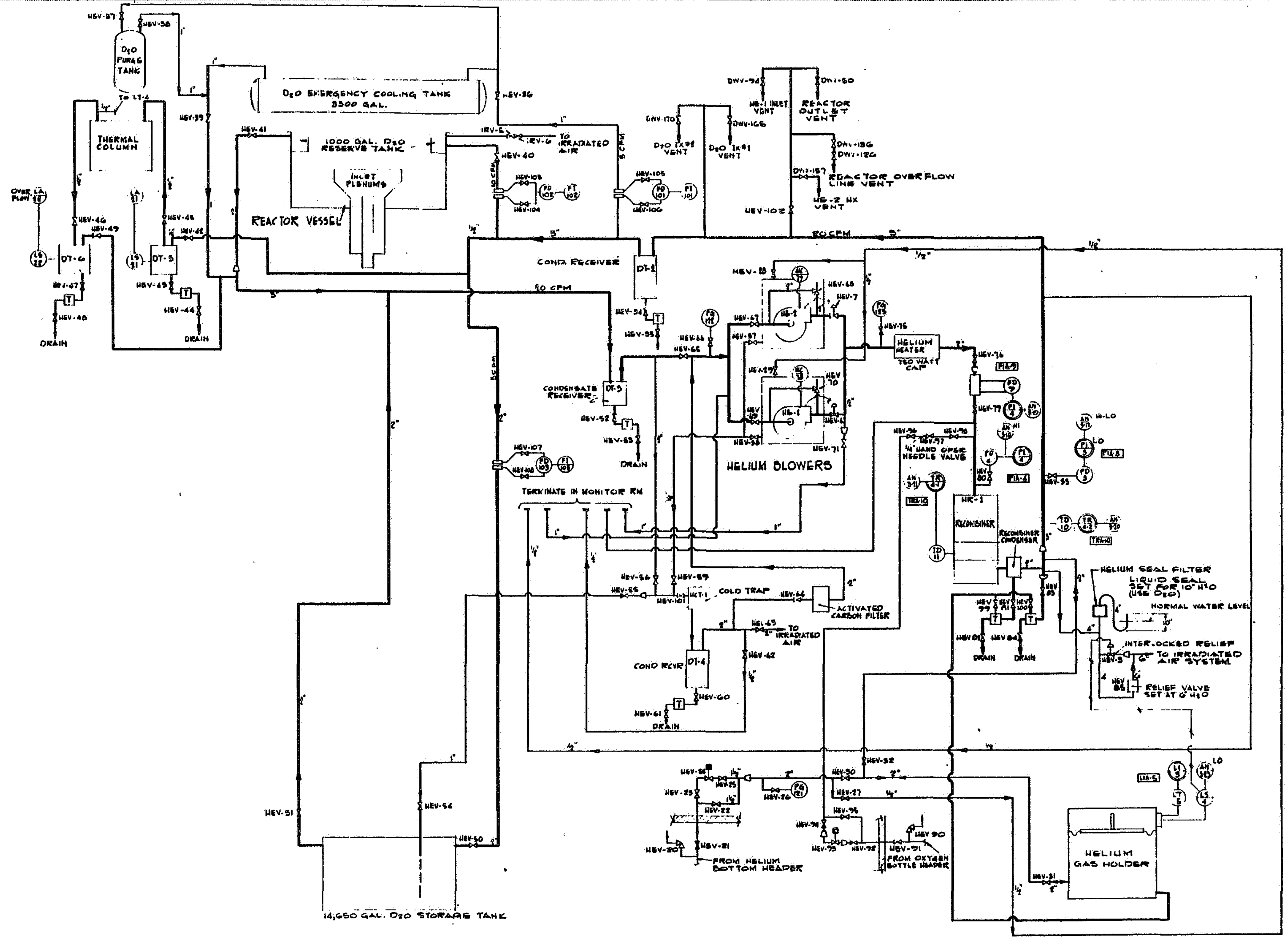

Figure 7.16 Helium sweep gas system.

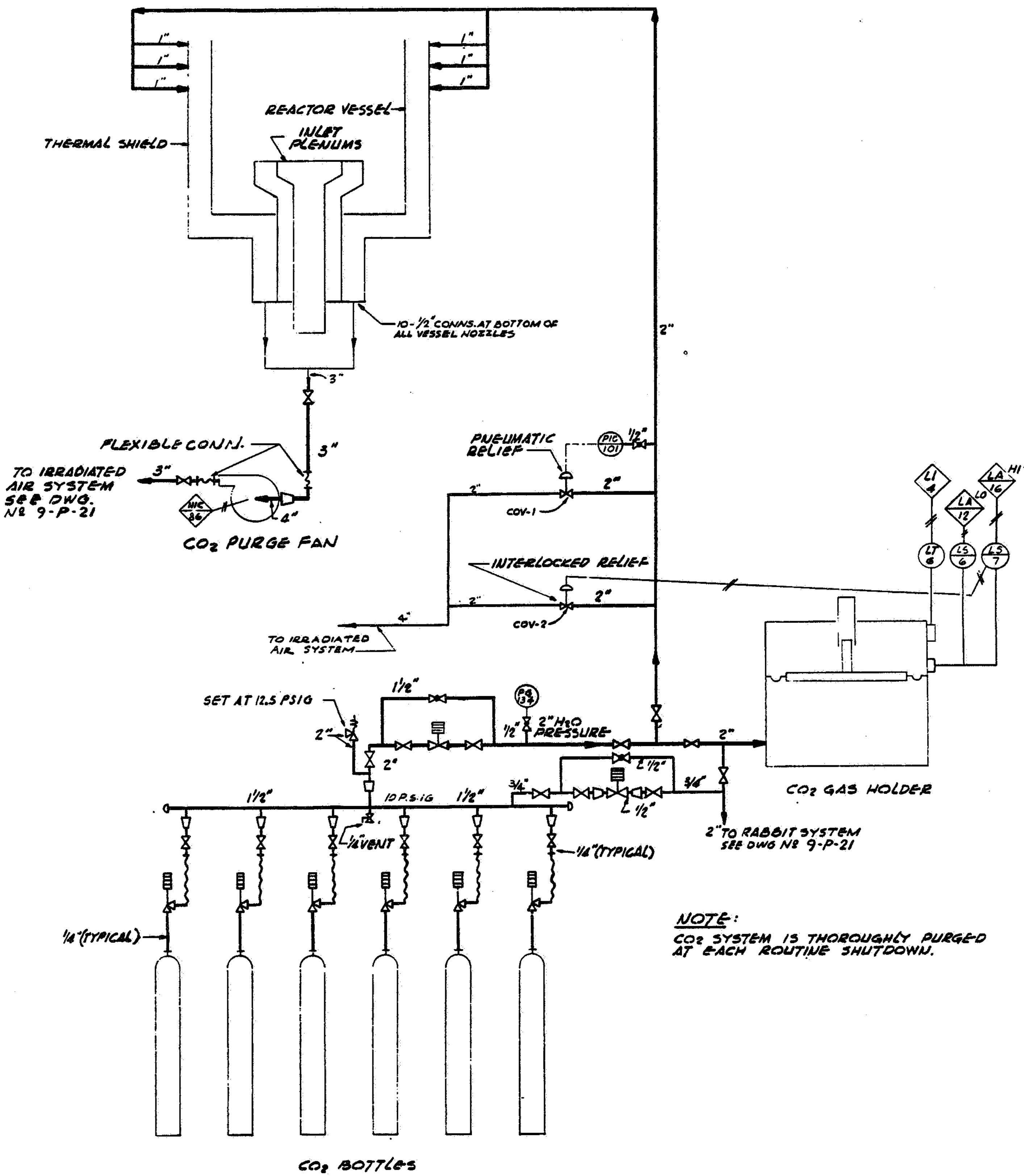

Figure 7.17 CO_2 purge system.

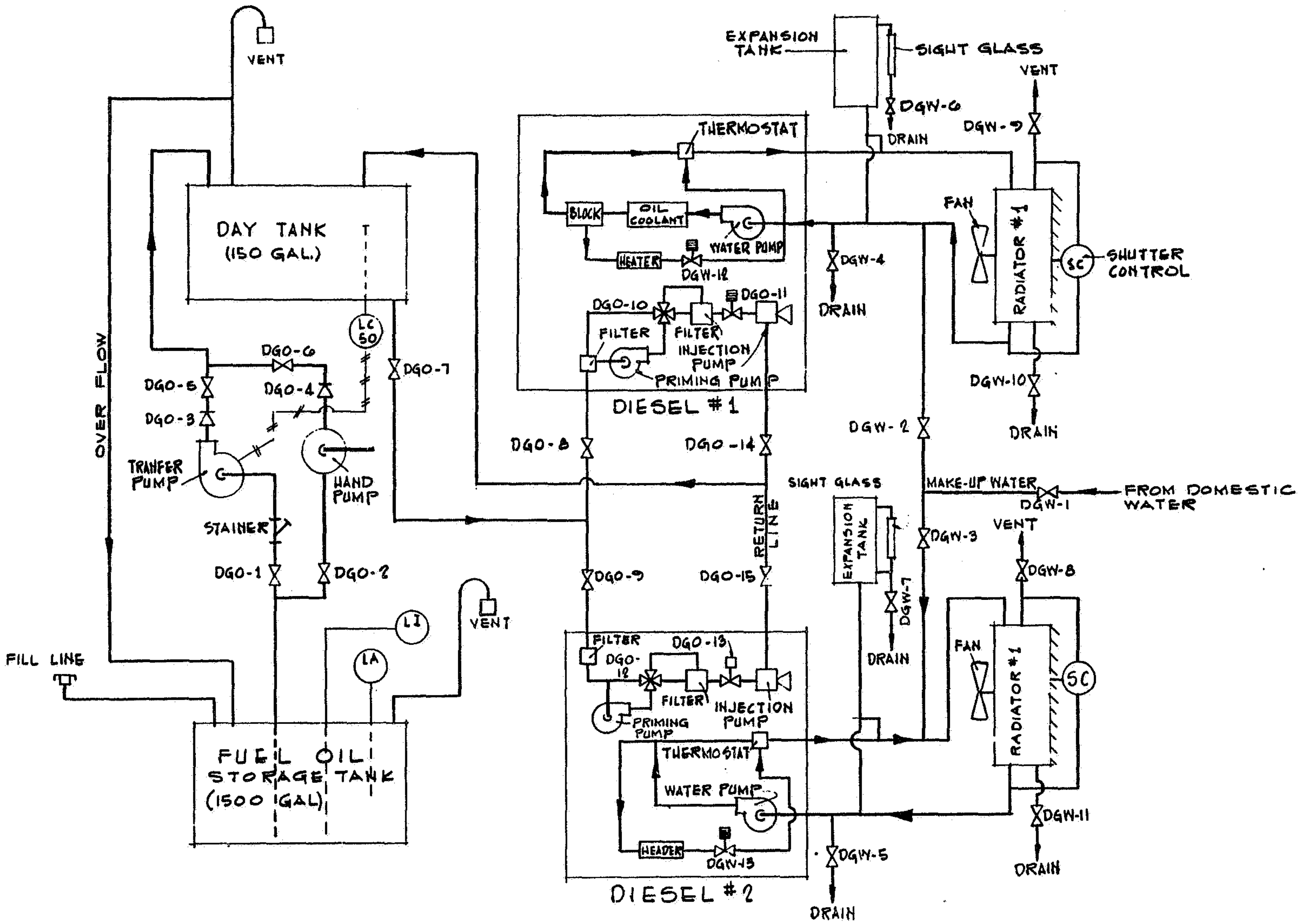

Figure 7.18 Diesèl fuel oil supply and cooling system.

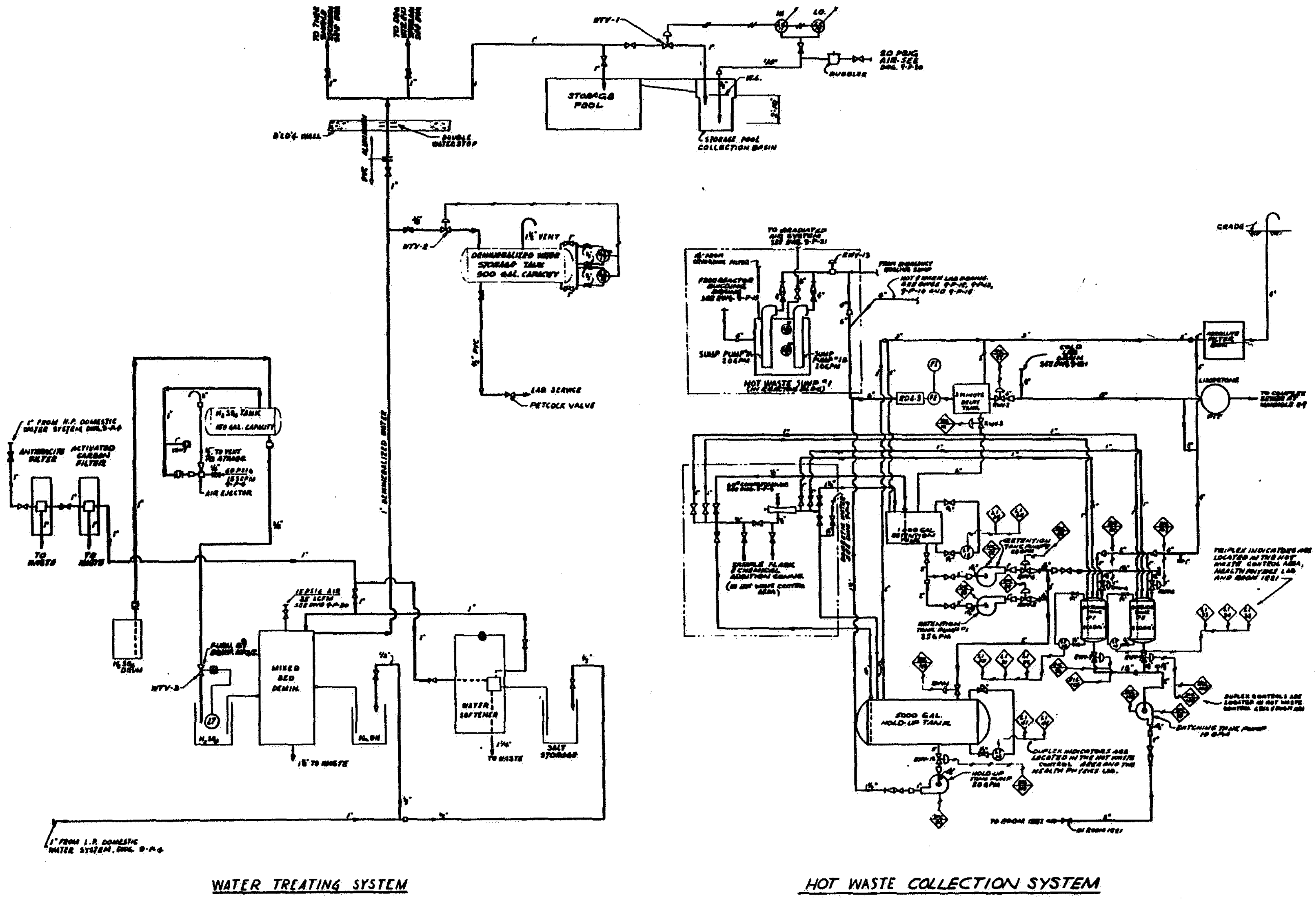

Figure 7.19 Demineralized water treatment system and hot waste system.

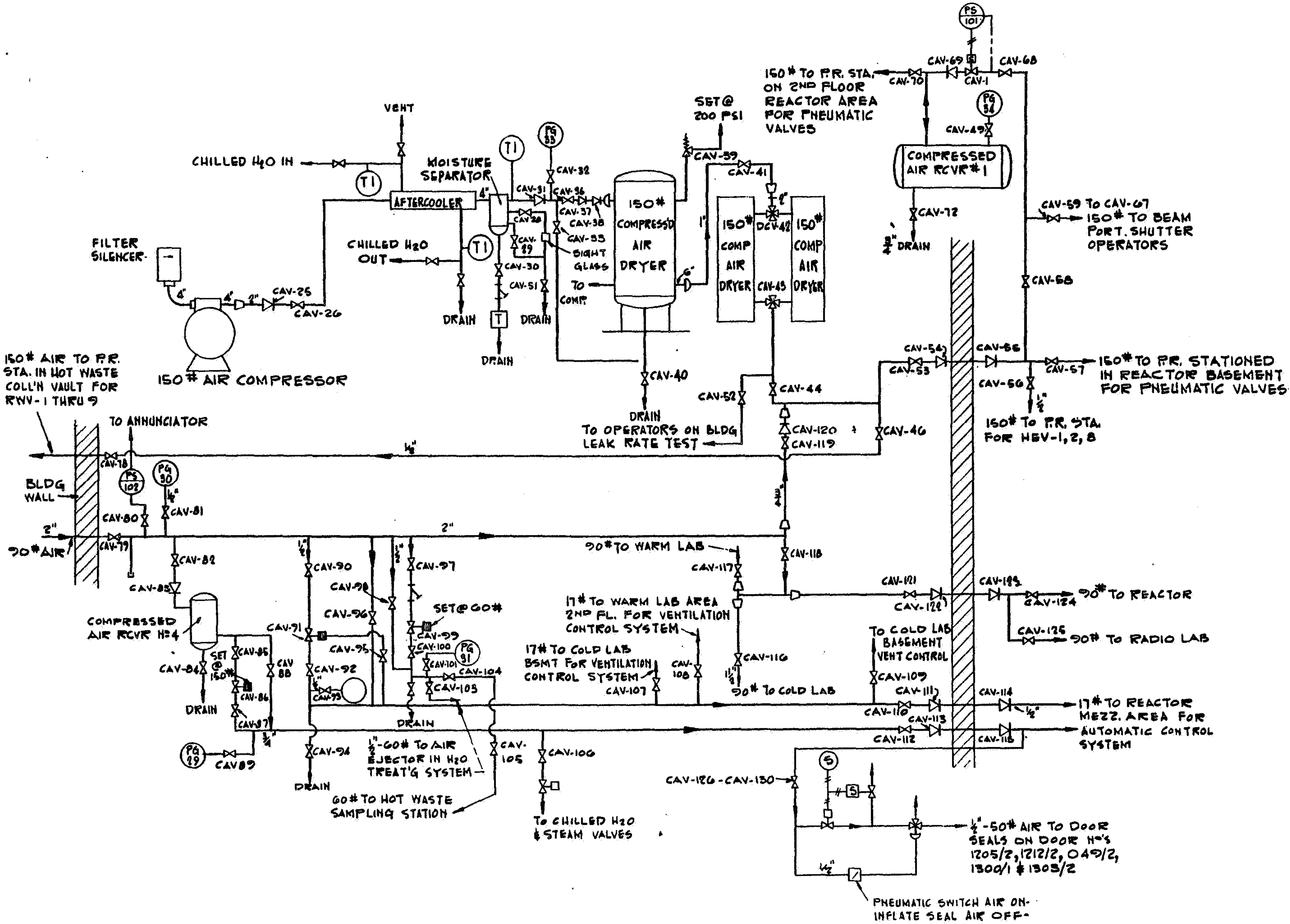

Figure 7.20 Compressed air system.

SECTION 8. EXPERIMENTAL FACILITIES AND SERVICES

8.1 EXPERIMENTAL FACILITIES

8.1.1 INTRODUCTION. The experimental facilities built into the NBSR can be seen in Figures 4.1 and 4.2. Ten positions not including the regulating rod thimble are available within the core structure itself for the insertion of experiments and seven positions are available in the reflector. Nine beam tubes are arranged radially in the central plane and look into the unfueled gap region of the core. Two beam tubes run completely through the reactor on either side of the core just below the radial tubes. The reactor includes a large re-entrant hole which is designed to permit the installation of a low temperature moderator close to the core to increase the intensity of long wavelength neutrons available to the two beam ports which look at it. Four pneumatic tubes which operate on CO_2 allow the rapid insertion and removal of small samples into various parts of the core and reflector. A large volume of well-thermalized neutrons will be available in the graphite thermal column.

Table 8.1-1 summarizes the experimental facilities which have been incorporated into the NBSR design.

Table 8.1-1 Experimental Facilities

Description	Number of Units
Beam Tubes	
6" I.D. radial tubes	4
5" I.D. radial tubes	3
5" I.D. truncated radial tubes	2
4" I.D. through beam tubes	2
Cryogenic Facility	
Large rolling plug for insertion and support of low temperature moderator	1
6" beam tubes viewing cold moderator	2
Thermal Column, 54" x 52" x 37" deep graphite	1
Vertical Thimbles	
3-1/2" I.D. in core	6
2-1/2" I.D. in core	4
4" in reflector	7
Pneumatic Tubes - All 1" I.D.	
Penetrates 6" into core below fuel gap	1
Central plane at core surface	2
15" below center plane and 34" from core center	1

8.1.2 RADIAL BEAM TUBES. There are a total of nine radial beam tubes. Four have an inside diameter of 6" from the shutter through the thermal shield with a 6-1/2" I.D. re-entrant tube in the reactor vessel proper. Similarly there are five tubes with corresponding dimensions of 5" and 5-1/2". With the exception of one beam tube whose shutter had to be different to avoid interferences, all 9 radial beam tubes are identical from the shutter out to the face of the biological shield. The thermal neutron flux at the core end of the beam holes is expected to be slightly less than under 10^{14} n/cm^2-sec for 10 Mw operating power.

A typical beam tube is shown in Figure 8.1. The region which is filled with CO_2, between the reactor vessel and the thermal shield is sealed by an aluminum window bolted to the outer face of the thermal shield plug. The thermal shield plug consists of an aluminum liner surrounded by water-cooled lead set in a steel plug which penetrates the thermal shield. All radial beam tubes have vertical shutters located as close to the thermal shield as possible. With the exception mentioned above, all radial beam tube shutters are identical and consist of 13" of lead, 3" of hydrogenous material (masonite), and 2" of steel for a total thickness of 18". The shutters and shutter cavities are lined with boral and the shutters are water-cooled. The shutters are designed to provide enough shielding when the reactor is shutdown to allow the easy removal and insertion of plugs and collimators in the region outside of the shutters. Since the shutters absorb most of the core gamma ray energy, no cooling is required in the outer shield plugs. Beyond the shutter is an 8" I.D. hole followed by a 14" I.D. hole and the beam tube terminates in a 2' x 2' x 6" deep recess in the reactor face. This recess has many conduits leading into it which allow access to all the services provided for the beam tube.

Since, when the reactor is operating, the shutter will not be adequate to completely suppress many of the experimental beams that will be used, provisions have been made for placing a water tank shield between the core and the shutter. A shallow trench along the bottom of the beam tube and shutter cavity allows as many as four 1/2" tubes to be led to the region beyond the shutter. So it is possible to replace the aluminum window with an aluminum tank that can be filled and drained to act as an auxiliary shutter.

In order to not preclude the possibility of neutron time-of-flight studies requiring a long flight path, a port has been built into the confinement building wall opposite one of the beam tubes. A thin neutron window may be used to replace the normal closure if a long neutron flight path is to be used.

8.1.3 THROUGH TUBES. Two 4" tubes pass completely through the reactor close to the bottom half of the core. The arrangement is shown in Figure 8.2. The arrangement is very similar to that for the radial beams except there is only one shutter for each tube which had to be located near the outer end of the beam. The opening through the thermal shield is 4" and the tube through the reactor vessel is 4-1/2" I.D.

8.1.4 CRYOGENIC FACILITY. The cryogenic facility provides a large re-entrant port that can be used to place a large volume of low temperature moderating material close to the core. Its purpose is to increase the intensity of long wavelength neutrons ($\lambda > 4A$) available for use in certain neutron scattering experiments. This will be accomplished by refrigerating a large block of ice to about 25°K by means of cold helium gas. The neutrons will then attempt to come into thermal equilibrium with the cold ice and, consequently, the low energy neutron intensity will be increased. The volume of ice will be about one cubic foot. If this volume were not shielded from core gamma radiation, the cryogenic cooling requirements would be too large to manage. Therefore, it is planned to place a bismuth gamma ray shield around the cold moderator. The bismuth shield will be attached to a large rolling shielding plug which will make it possible to insert and remove the bismuth. Within the large plug will be another shielding plug on the end of which will be mounted the moderator cryostat. The detailed design of the bismuth shield and moderator cryostat have not been completed; consequently, they are not being presented for review at this time.

During the early stages of reactor operation a simple D_2O filled tank will be mounted on the end of the large rolling plug. The arrangement is shown in Figure 8.3. The water tank is penetrated by two 6" I.D. tubes which mate with the two cryogenic beam ports in the biological shield. In addition there are two smaller 2" tubes which mate with small beam holes in the inner plug. The flange of the water tank carries a gasket which forms the CO_2 seal by pressure from the outer plug.

The shutter arrangement shown is to aid in the withdrawal of the plugs during reactor shutdown, if necessary. The small inner shutter can be lowered after the inner plug is partially removed to allow its complete withdrawal. To withdraw the large rolling plug, it is rolled part way out and the narrow shutter on the core side is lowered. Then the plug is withdrawn further and the remaining combination of inner and outer shutter is lowered together allowing the whole plug assembly to be removed into a shielded cask. These shutters are not available for use when the facility is in operation.

Heavy water is supplied to the service box at the outside of the large plug. Lines through the plug make it available for circulation through the tip. Flow and temperature transducers transmit signals to the reactor panel so the D_2O flow and outlet temperature may be read directly in the control room.

8.1.5 THERMAL COLUMN. The thermal column can be seen in Figure 4.2. The initial section of the thermal column is a D_2O filled water tank. This tank is part of the main D_2O system and so is completely closed. It is used because the high heat generation rate and radiation damage to graphite make the use of graphite in this region, close to the core, very difficult. The D_2O is, of course, easily cooled and is an even better thermal column material than graphite.

To reduce the gamma ray contamination of the thermal flux in the graphite section, a 7-1/2" thick bismuth shield follows the D_2O. It is actually located in the D_2O. The bismuth itself, does not come directly into contact with the D_2O, but is cast into an aluminum container which is located in the D_2O tank. The bismuth reduces the core and capture gamma rays to the point where most of the gamma radiation in the graphite region comes from neutron capture in the graphite itself.

The graphite region is 54" x 52" in cross section and 37" deep. It is made up of 4" x 4" stringers. Certain groups of stringers are made to be readily removable through plugs in the thermal column shield door. In addition, the access is provided to the graphite from the top through a vertical hole about one foot on a side. The shielding door can be rolled back to provide access to the whole graphite region if desired. The graphite is surrounded by boral which is backed up by water cooled lead.

In order to facilitate limited access to the thermal column during reactor operation, there is a boral curtain the size of the thermal column which can be lowered between the bismuth shield and the graphite.

8.1.6 PNEUMATIC TUBE SYSTEM. Four pneumatic tubes penetrate the reactor vessel. They are designed for the rapid insertion and removal of samples in the high flux regions of the reactor. The rabbits which carry the samples are about 1" in diameter and travel at speeds from 30 to 45 ft/sec. Automatic timing and control devices can be set for exposures from a few seconds to 20 minutes.

Each tube has two sending and receiving stations; one in each of two radiological laboratories located in the reactor basement. All the receiving stations are located in radiological hoods.

Of the four tubes, two penetrate the reactor vessel up to the surface of the core in the central plane. The expected thermal flux for these tubes is about 9×10^{13} n/cm^2-sec and the fast (epi-cadmium) flux will be about 3×10^{13}. The third pneumatic tube actually passes between two fuel elements into the core itself below the unfueled gap. Its thermal flux is about 9×10^{13} n/cm^2-sec and its fast flux about 10^{14}. The remaining tube is located near the outer edge of the reflector and is in a flux of about 3×10^{13} thermal and 2×10^{12} fast.

8.1.7 VERTICAL THIMBLES. The position of the in-core and reflector thimbles can be seen in Figure 4.2. In the regular hexagonal arrangement of the core, there are seven positions which will normally be used for 3-1/2" diameter thimbles. In addition, within the core there are four smaller, 2-1/2" thimbles located between adjacent fuel elements. Seven more positions up to 4" in diameter are available in the reflector.

8.1.7.1 Three and One-Half Inch Experimental Thimbles. Each of the seven 3-1/2" positions in the core is occupied by a 4" O.D. x 3-1/2" I.D. cylinder which has a bottom end fitting designed to fit the standard grid plate opening. The end fitting largely blocks the normal flow, but contains a small opening which allows about an 8 gpm flow up through the tube to cool it and any experiment that may be in it. This flow should suffice to cool most samples which are inserted, but they can be cooled by auxiliary lines from above if necessary. The cylinders are held down by the posioned tubes shown in Figure 4.1. To insert an experiment, the shielding plug is removed from inside of the top of the hold down tube and an experiment inserted down into the core. Only six of the seven positions are available for experiments, because one position is used for the regulating rod.

These thimbles are actually small islands of D_2O surrounded by fuel elements with a radius of 7" to the center of the adjacent elements. Consequently, the thermal flux is somewhat enhanced and the fast flux depressed. The ratio of thermal to fast flux is estimated to be about two.

8.1.7.2 Two and One-Half Inch Experimental Thimbles. The 2-1/2" I.D. experimental thimbles are defined by semi-permanent cylinders held in place by the top grid plate. They can only be removed by removing the top grid plate first. They do not fit into a standard fuel element hole in the bottom grid plate, but special, smaller receivers have been built into the bottom grid plate. These smaller sockets have a small hole at the bottom which allows about 10 gpm of plenum cooling water to flow up through the experimental thimble.

The cylinders are located between adjacent elements so they are as close as possible to fuel. This should enhance their fast flux spectrum and provide a better flux for radiation damage than is available in the 3-1/2" thimbles.

Although samples and supporting equipment may have a diameter up to 2-1/2" in the core, any supporting tubes, leads, etc., coming from the top plug must be confined within a 1-1/2" diameter in the region between the bottom of the top plug and the top grid plate. Although there are normally no interferences in this region, this restriction is necessary so that fuel elements being transferred by the transfer arms do not touch the supports of the experiments in the 2-1/2" thimbles.

8.1.7.3 Reflector Thimbles. The top shielding plug has seven ports in it designed to allow up to 3-1/2" diameter experiments to be inserted into the reflector. They are located as shown in Figure 4.2, about half way between the core surface and the vessel wall. There are no thimbles constructed in the vessel as part of the reflector facilities. Instead, any sample or experimental device is supported from the top by a tube inserted through the plug. Each experiment will require its own design and construction--only the openings in the top plug are provided initially. No cooling other than free convection is provided in the reflector, but cooling can be provided from outside the plug.

Not all seven positions will always be available for experiments since two of them will probably be used during shutdown for a periscope.

8.2 EXPERIMENTAL FACILITY SERVICES

8.2.1 FIRST FLOOR SERVICES. The utilities and services available at beam tubes are shown in Figure 8.4. In this figure, one-half of the reactor face has been unfolded to display the beam tubes, utility supply boxes, and interconnecting conduit. In a typical arrangement, utilities and electric outlets are located in a recess in the reactor face. The bottom of the recess is eight feet above floor level. One side of the recess contains two double duplex outlets. One is 120 volt instrument power and the other is 120 volt regular power. In addition a duplex outlet of each type is mounted flush on the face of the biological shield near each recess. Also at this level, one three-phase 208 volt outlet is mounted flush with the shielding face for approximately every two beam holes. All supplies have 20 amp circuit breakers.

The other side of the recess contains valves for demineralized water, secondary cooling water, and compressed air. The demineralized water is supplied by a separate system provided for experimental use. The secondary cooling water is supplied by a take off from the reactor secondary cooling system incorporating the cooling towers and the main reactor heat exchanger. D_2O supply and return is provided on opposite sides of the reactor. One of these is incorporated directly in the cryogenic facility which will require D_2O cooling for its bismuth shield. These water supplies can be used only in aluminum or stainless steel systems so, if they are needed for other type systems, intermediate heat exchangers must be used.

Other recesses, most of which are 2' x 2' x 6" deep, are at the ends of each beam hole. These are connected to the utility recesses by curved conduit embedded in the concrete. Thus, the utilities can be supplied to the interior of the shielding system around the beam hole without interfering with the shielding itself.

A third recess is located directly below each hole and is connected to it by more embedded conduit. One side contains the valved water returns plus a valved drain which leads into the reactor hot waste system. The other side is essentially a junction or pull box for any signal leads, tubes, wires, etc., which it may be desirable to run from the vicinity of the beam hole to a control station. Conduit runs through the floor from the building wall to these recesses under the beam holes. An intermediate access box is supplied in the reactor floor for each set of conduit about 10' from the reactor face.

Slightly above each beam hole, a small recess designated by G in Figure 8.4, provides access to a reactor scram circuit. Thus, if necessary, a component of an experimental system can be connected directly to the reactor scram system.

In addition to the utilities in the reactor face, electric power and domestic water are available along the building wall.

8.2.2 SECOND FLOOR SERVICES. Figure 8.5 is a plan view of the utility trench and the top outer plug or doughnut. The center plug containing the transfer mechanisms and access ports to the core is not shown but it fits into the center hole and is recessed about 2' below the second floor level. A typical cross section of this region is contained in Figure 4.1. In addition to the utility trench, there is a raceway about 12" deep by 12" wide formed by the plug liner and the top outer doughnut. There are holes in the raceway walls providing direct access to the utility trench. Conduit from the trench to the experimental hole recesses in the upper doughnut as well as conduit directly from the trench to the center plug region are shown in the dotted lines in Figure 8.5. The experimental hole recesses are connected to the center plug region by short 6" diameter openings. Conduit from the trench to pull boxes in the floor and on out to the wall allow signal cables, etc., to be run between instrumentation at the wall and the center plug region. This system of conduit, trenches and openings allows a great deal of freedom in the arrangements necessary to service the in-core experiments.

The location of utilities available in the trench are shown schematically in Figure 8.5. Demineralized water and secondary cooling water supply and returns are available at four locations in the trench, and D_2O supply and return at two. Each line is valved and also equipped with a quick disconnect fitting. Four compressed air outlets are provided. Flexible tubes can be run through the passages and conduit provided to connect these utilities with the experiments inserted through the top plugs of the reactor. Instrument and regular 120 volt electric power is available both on the outer trench walls and in the raceway. More instrument and regular outlets are mounted on the room walls along with some 208 volt three-phase outlets.

In addition to the utilities, the trench contains the pneumatic drive system for the shutters. A pneumatic cylinder is mounted above each shutter and the air to drive it is supplied from the trench. The water cooling for the shutters is also connected and valved individually in the trench.

All trenches, raceways and other recesses are fitted with cover plates which fit flush with the floor. They are sufficiently strong to match the load bearing capacity of the floor. The floor plate over the center plug consists of six inches of steel and is designed to provide shielding in conjunction with a vertical transfer cask for the removal of irradiated materials up through the open space above the center plug and into the transfer cask.

8.3 OPERATING CONSIDERATIONS

8.3.1 EXPERIMENTAL COOLING. Most experiments using the beam holes will have very little direct interaction with the operation of the reactor. A few beam hole experiments and those inserted into the core from the top will need cooling for components near the core. It is anticipated that most of the in-core irradiations will be cooled by the plenum water flowing up through the experimental thimbles in the core. Certain experimental components, however, heated by the core, will need additional cooling. The D_2O services mentioned earlier are for this purpose. Each cooling system for reactor heated components will be monitored for temperature or flow or both. Low flow or high temperature will alarm at an annunciator panel in the control room whenever such condition presents serious hazard to experimental equipment or any hazard whatsoever to the reactor. Whenever necessary, indication of faulty operation of these cooling systems can be hooked directly into the reactor scram system.

8.3.2 <u>EXPERIMENTAL INTERACTION WITH CORE</u>. The NBSR is a research reactor rather than a testing reactor. Thus, the emphasis is on neutron beam programs and small sample irradiations. It is never anticipated to undertake fuel element development studies or the irradiation of other bulk type components which might introduce large amounts of fissionable materials into the core.

The beam tubes will be used almost exclusively to extract neutron beams from the reactor which will have no significant interaction with the core. In some cases water tanks may be placed behind the shutters to serve as additional neutron shutters. Filling and draining these tanks may alter the reflector efficiency slightly, but will have negligible effect on the core reactivity. In some cases beam holes may be filled or partially filled with moderator. These would have small effects on reactivity as discussed in Section 4.6.7, but would be semi-permanent installations.

The pneumatic tube system does have a limited interaction with the core. Since the rabbits are inserted and removed rapidly, they can introduce sudden, but small changes in reactivity. These changes will be small (see Section 4.6.5) compared to the reactivity controlled by the regulation rod and so will introduce no difficulties into reactor control.

The vertical thimbles will be used primarily for small sample irradiation. Some of these samples may be highly instrumented, including cryogenic cooling, but they will have relatively low reactivity worth. Most of the in-core facilities will be of such a nature that they can be inserted or removed only during shutdown. It may be possible to insert and remove a few low reactivity experiments during operation, but the reactivity of such experiments would be limited to values less than the worth of the regulation rod.

In every case in which the failure of any experimental component or physical phenomena would affect the operation of the reactor, the effects will be monitored, displayed in the control room where desirable, and connected directly into the scram system if necessary.

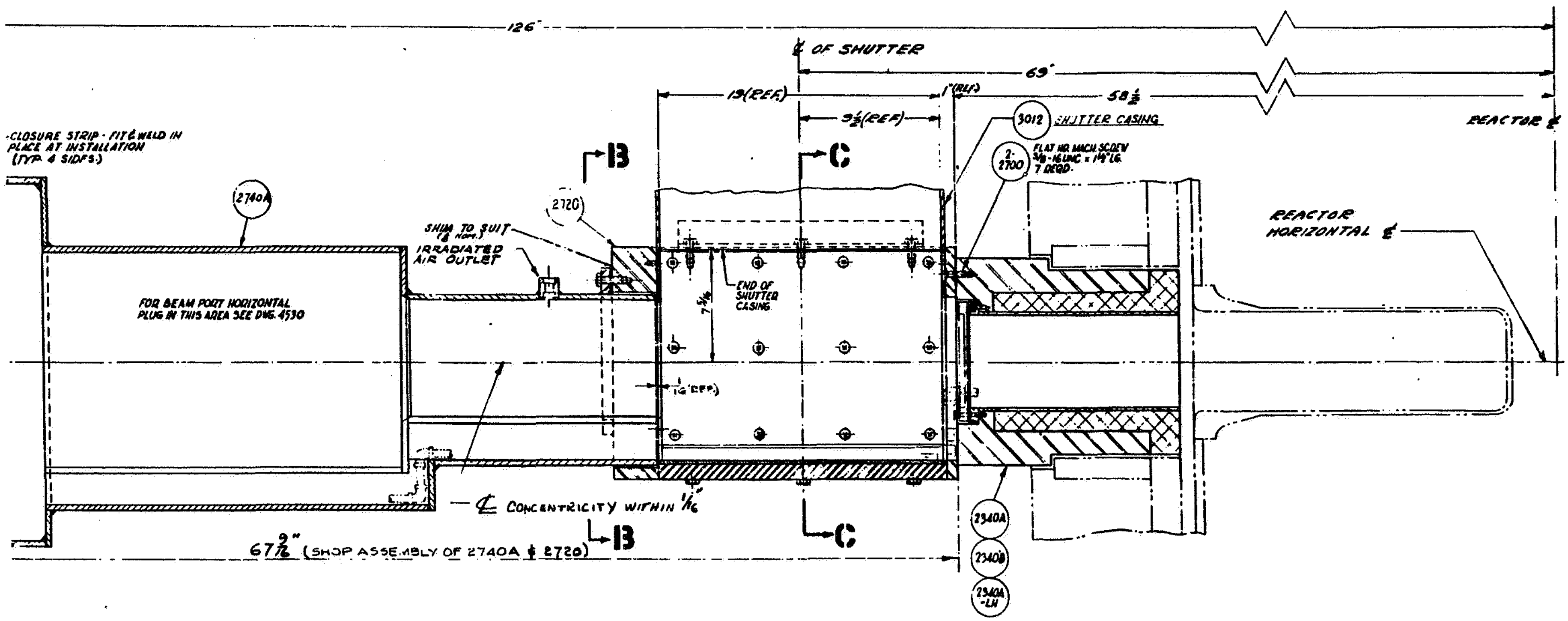

Figure 8.1 Typical radial beam tube sectional elevation.

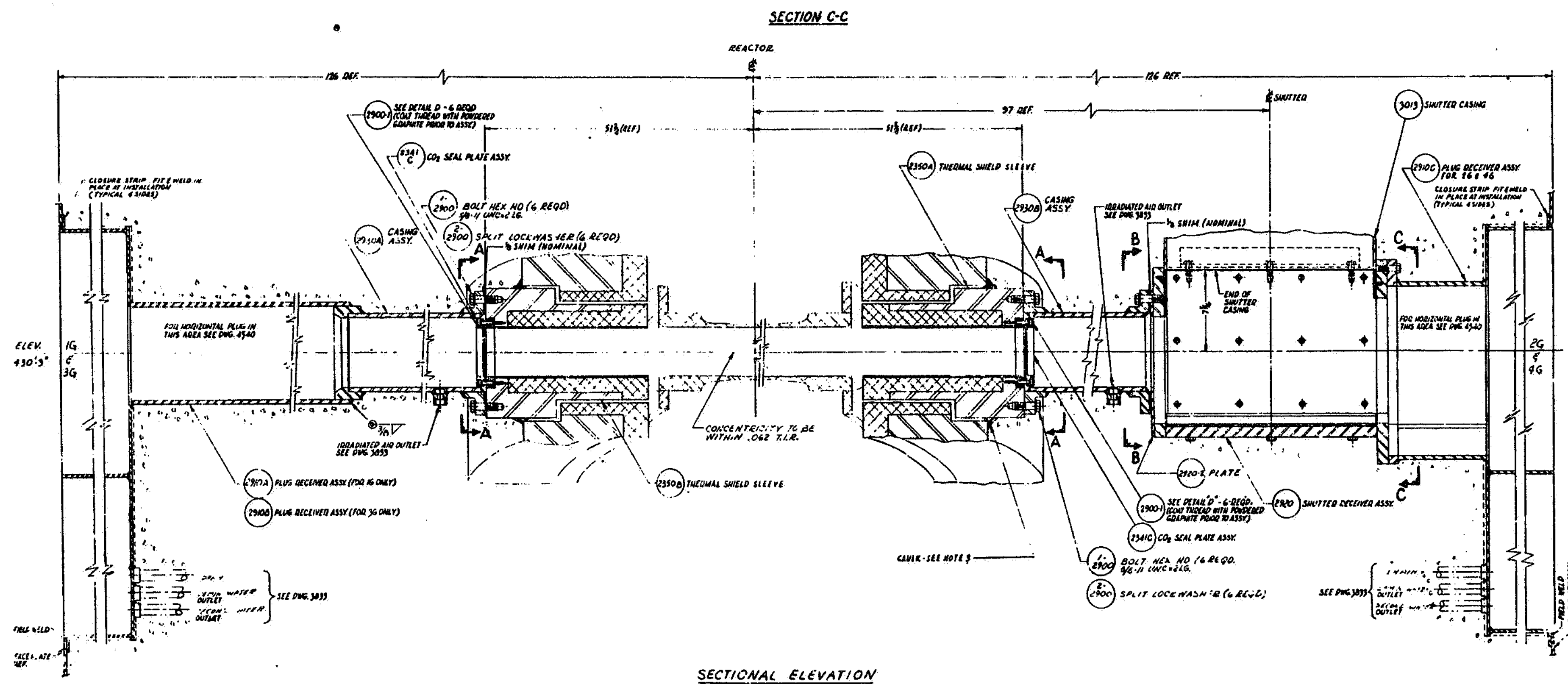

Figure 8.2 Typical through tube sectional elevation.

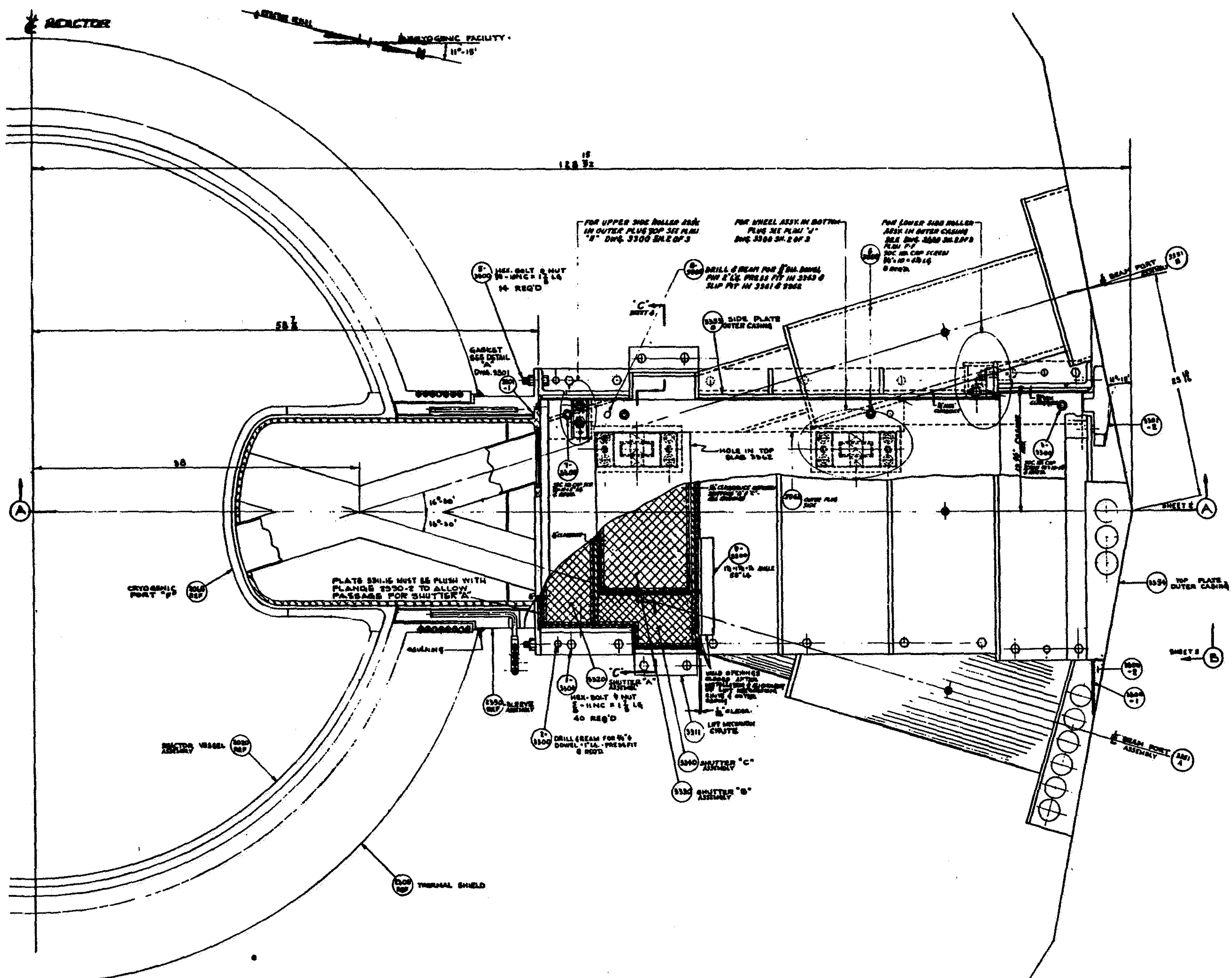

Figure 8.3 Cryogenic facility details.

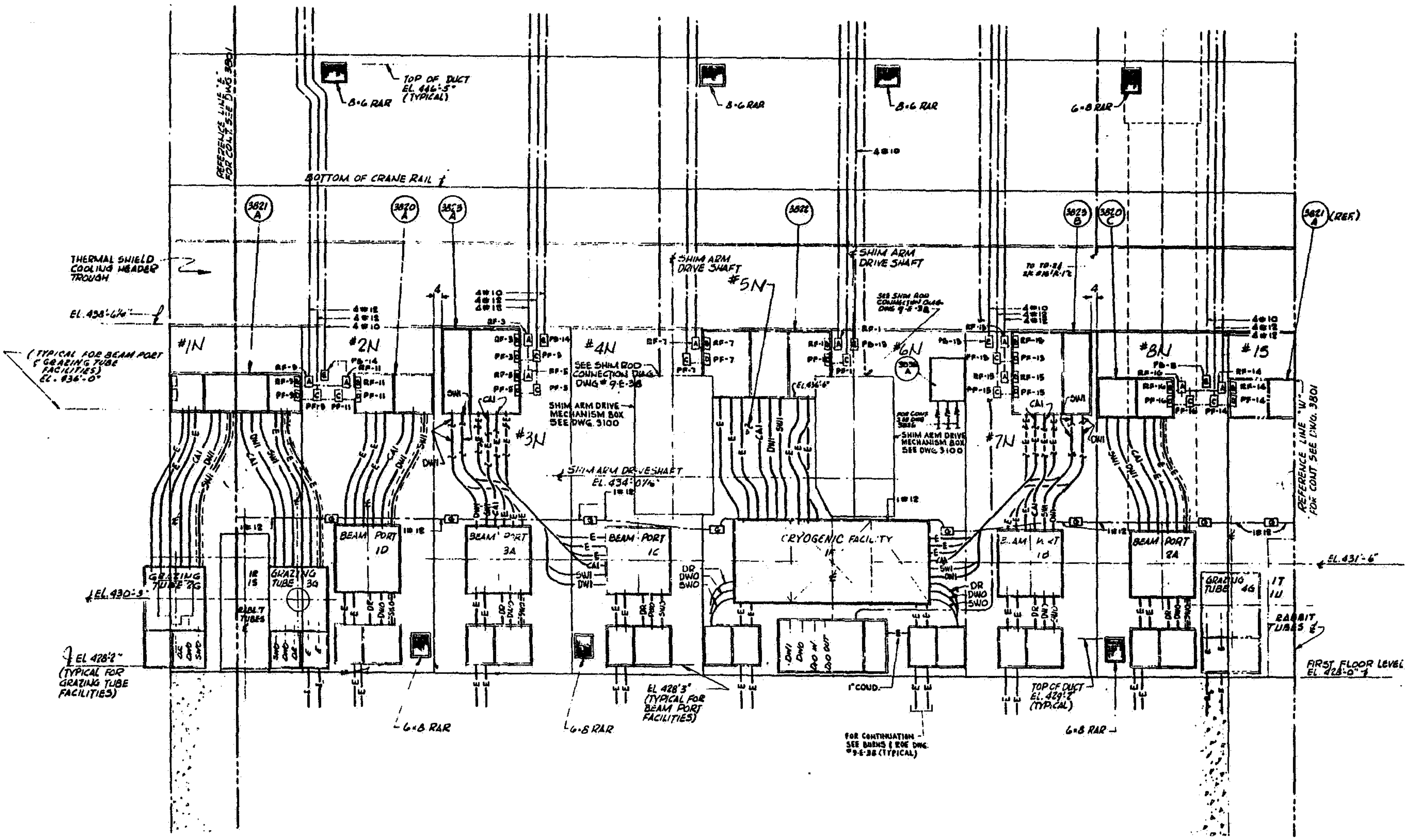

Figure 8.4 Beam tube utilities.

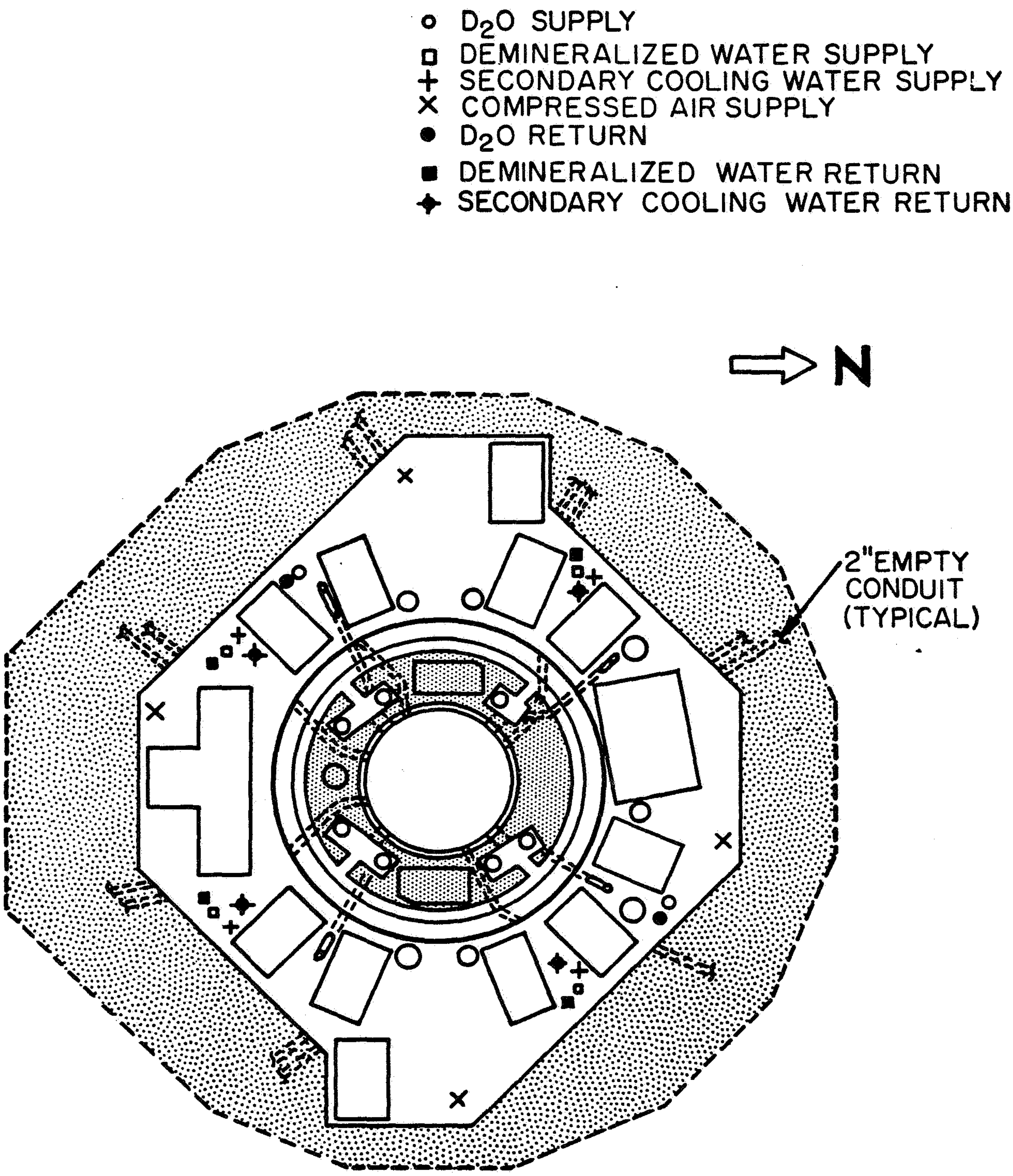

Figure 8.5 Plan view of top utility trenches and services.

SECTION 9. INSTRUMENTATION AND CONTROL

9.1 REACTIVITY CONTROL

The basic point of view taken in the control philosophy of the NBSR stems from the nature of the xenon poison problem characteristic of high flux thermal reactors. As discussed in Section 4.6.11 xenon poison builds up after shutdown to levels which are exceedingly high. In order to be able to restart the reactor after shutdown without major core reloading, large excess reactivity and rapid rod withdrawal would be necessary, both of which intensify the hazards problem. It has, therefore, been decided to attempt to design a control system which stresses reliable operation, and which minimizes inadvertent reactor shutdown. Little, if any, reactivity will be carried for xenon override, and rod withdrawal rates will be conservatively slow. This approach suggests the use of redundant circuitry, the outputs of which are placed in coincidence to reduce false scrams from individual circuit failures. The slight decrease in safety inherent in this approach is more than compensated for by the prior decision to limit reactivity and reactivity change rate.

The thermal neutron flux level is the primary quantity which is measured for reactor operation and safety. The flux measuring instruments indicate the level and the rate of change of flux from source level to full power and signal the safety system if the reactor power level changes too rapidly or if it exceeds a pre-set maximum power level.

Reactivity can be controlled in the NBSR by means of four separate systems; the regulating rod, the shim safety arms, the top reflector dump, and the void shutdown or bubbler system. The latter three can be individually or collectively brought into action for reactor scram purposes.

9.1.1 REGULATING ROD. Normally the reactor at power will be controlled by the regulating rod which has been described from the mechanical point of view in Section 4.3.3. The control circuitry for manual, automatic and emergency operation is described in the succeeding Section 9.5.3.7. The rod is driven by a reversible 115 volt, 60 cycle AC servomotor in a drive assembly which positions a lead screw. Simultaneously, selsyn transmitters on the top of the motor transmit electrical signals to a position indicator assembly located on the control panel. Both coarse and fine control indication is transmitted by separately geared torque transmitters. Gear reducers, giving speed reductions of 10:1 and 30:1 are coupled to the torque transmitters so that the "fine" dial makes one revolution per inch of travel and the coarse dial makes one revolution per thirty inches of travel. Position indication to better than 0.01" has been achieved.

The regulating rod drive has a travel of approximately 29" and the servomotor drives the rod at a speed of approximately 30" per minute. Since the rod worth is approximately 0.5% reactivity, the maximum rate of reactivity control with the regulating rod is approximately 0.009% per second.

As previously mentioned the regulating rod does not release and drop during scram action. However, since it does insert during rundown, it also inserts at normal speed during a scram condition.

Annunciator indication is included for both near up and near down position in addition to full up and full down position. This feature will permit operators to take shim action before achieving full up or down conditions thereby permitting more nearly linear reactivity control with rod position. These near up and near down limit switch positions are adjustable. In the event the regulating rod should bind, a jam switch is included in the drive mechanism to provide control panel annunciation.

9.1.2 SHIM SAFETY ARMS. The shim safety arms are of the "semaphore" type such as used at the CP-5. The mechanical description of the drive system and the reactor physics reactivity calculations are discussed in Section 4.3 The purpose of the shim safety arms are twofold: (1) to adjust for gross changes in reactivity, (2) to shut the reactor down quickly. These arms will have a control worth greater than the maximum excess reactivity of the

reactor and will, therefore, shut the reactor down and hold it sub-critical even after the water is cooled, the fission products have decayed, and new fuel has been loaded. The design of the drive includes a spring which accelerates the arms into the core upon release of an electromagnetic clutch. The release time is expected to be less than 0.050 seconds as measured from the time of scram action to release of the clutch plate. Preliminary tests upon the clutch and power source have confirmed this release time.

The withdrawal rate of the shim safety arms will be 0.04 deg/sec, which corresponds to a maximum reactivity change rate of 6×10^{-4} per second. The arms can be inserted or withdrawn individually or as a group. Position indication is transmitted by torque transmitters to the control panel for each arm individually. The control circuitry for electromagnetic clutch activation is discussed in Section 9.5.1.

9.1.3 TOP REFLECTOR DUMP. In the remote event of difficulty of shim safety arm insertion of all four safety arms, provision is made for lowering of the top reflector to a level just above the top of the core. Reactivity calculations show the top reflector worth under normal operating conditions to be approximately 6% reactivity. This so called "moderator dump" can be initiated manually by the reactor operator at the console by switch S-6 of Figure 9.12. The relay control for the valve which operates the dump pipe line from the reactor tank is interlocked with the rundown when S-6 is in the automatic position. If the top reflector level falls below the alarm set of LA-3, a startup prohibit is reinstituted. The relay control is also interlocked with the control switch for the void shutdown control which is described in the next section.

9.1.4 BUBBLER OR VOID SHUTDOWN. An additional mechanism for providing large amounts of negative reactivity is provided in the form of an in-core helium gas bubbler system. On the appropriate signal this system will blow helium gas into the core moderator at a rate of approximately 180 cfm. Approximately 4 ft^3 of void space can be accumulated in a mean storage time of about 1.3 seconds. This amount of void is worth approximately the reactivity of the top reflector. Its overall time response is similar to, and only slightly slower than the shim safety arm system. Though this system could be used as a fast emergency back-up for the mechanical shim safety arms, it is planned to use it only to augment the moderator dump. This is so since it takes approximately 40 seconds for the top reflector level to reach its lowest position above the core after the dump valve has opened.

A schematic representation of bubbler system flow diagram is shown in Figure 7.2. The bubbler system requires a deliberate operator action to initiate through switch HIC-67. A moderator dump signal follows automatically. The indicated flow rate previals at the pressure of 5 psig, which pressure is controlled by the controller PC-3 of channel PIC-5. The flow rate is sufficient to exhaust the contents of one helium bottle in about one minute, a time longer than the drop time of the top reflector level. Several bottles will be installed in addition to the fact that a back-up instrument air system can be initiated should helium flow fall below the prescribed rate of 180 cfm.

9.2 OPERATION CONTROL STATIONS

9.2.1 CONTROL ROOM. The control room is located on the second floor of the reactor building at the operations level. The arrangement and location of the control room are shown on building drawings, Figures 3.1 and 3.5. The shift supervisor's office is adjacent to the control room. A window in the front wall of the control room permits observation of the operations floor and the top of the reactor shield. Access to the control room will be limited to authorized personnel.

Remote instrumentation and systems controls are utilized throughout the plant to permit control of the reactor and the essential associated systems from the control room. Instrument panels display the process and nuclear variables required for proper plant operation. When seated at the control desk the operator has an unobstructed view of all important insturment panels, and by means of the controls can fully operate the reactor and associated systems during all phases of reactor operation. These controls permit the operator to regulate primary and secondary system flows, operate all primary and support system pumps, operate the control rods and cause immediate shutdown of the reactor in the event such action is required.

9.2.1.1 Control Panel. There are 11 instrument panels which make up the main instrument display in front of the console. These panels are designated as shown in Figure 9.1. Panels "K" and "L" contain controls and instruments for the thermal shield, storage pool, and experimental cooling systems. Panels "H" and "J" house helium and D_2O purification controls and instruments. Panels "F" and "G" display radiation monitoring, emergency ventilation and AC and DC power controls. Primary coolant instrumentation is located on panel "E". The main control station is panel "D" where reactor shim safety arm and regulating rod controls are located. Panel "C" contains secondary cooling system equipment controls and instruments. Additional radiation monitors are located on panel "B". Panel "A" houses nuclear instrumentation equipment. A list of individual instruments on each panel is given below.

9.2.1.1.1 Panel "L" - Vertical Sections

1)	AN-1	Annunciator panel #1
2)	TI-21	Experimental demineralized water heat exchanger, inlet temperature indicator
3)	TI-7	Experimental demineralized water heat exchanger, outlet temperature indicator
4)	PIC-1	Experimental demineralized water pressure indicator-controller
5)	FI-20	Experimental demineralized water ion exchanger, inlet flow indicator
6)	FI-6	Experimental demineralized cooling water flow indicator
7)	FI-7	Storage pool, cooling flow indicator
8)	LI-5	Experimental demineralized water tank level indicator
9)	TI-9	Storage pool, water temperature indicator
10)	CI-3	Storage pool ion exchanger inlet-outlet conductivity indicator
11)	CI-1	Experimental demineralizer, ion exchanger inlet-outlet conductivity indicator
12)	FI-8	Storage pool ion exchanger flow indicator
13)	CI-2	Storage pool water conductivity indicator

9.2.1.1.2 Panel "L" - Sloping Section

1)	CS-1	Thermal shield ion exchanger conductivity selector switch
2)		Power key switch, rabbit tube system blower
3)	HIC-76	Secondary experimental cooling pump #1, switch and lights
4)	HIC-75	Secondary experimental cooling pump #2, switch and lights
5)	HIC-57	Storage pool pump #1 isolation valve, SPV-2, switch and lights
6)	HIC-56	Storage pool pump #2 isolation valve, SPV-1, switch and lights
7)	HIC-35	Experimental demineralized water pump #1, switch and lights
8)	HIC-34	Experimental demineralized water pump #2, switch and lights
9)	HIC-59	Storage pool pump #1, switch and lights
10)	HIC-58	Storage pool pump #2, switch and lights

9.2.1.1.3 Panel "K" Vertical Section

1)	FI-10	Thermal shield, ion exchanger flow indicator
2)	CI-4	Thermal shield, water conductivity indicator
3)	CI-5	Thermal shield, ion exchanger inlet-outlet conductivity indicator
4)	FI-9	Thermal shield, cooling water flow indicator
5)	TI-11	Thermal shield, heat exchanger inlet temperature indicator
6)	TI-10	Thermal shield, heat exchanger outlet temperature indicator
7)	TI-23	Thermal shield, floor header outlet temperature indicator
8)	TI-24	Thermal shield, ring header outlet temperature indicator
9)	TI-22	Thermal shield, storage tank inlet temperature indicator
10)	HIC-60	Thermal shield, ion exchanger throttle valve TSV-5, controller
11)	LI-6	Thermal shield, storage tank level indicator

9.2.1.1.4 Panel "K" Sloping Section

1)	HIC-33	Storage pool booster pump, switch and lights
2)	HIC-77	Thermal shield return isolation valve, TSV-6, switch and lights
3)	HIC-64	Thermal shield pump #1 isolation valve, TSV-2, switch and lights
4)	HIC-63	Thermal shield pump #2 isolation valve, TSV-1, switch and lights

5) HIC-66 Thermal shield ion exchanger #1, isolation valve, TSV-4, switch and lights
6) HIC-65 Thermal shield ion exchanger #2, isolation valve, TSV-3, switch and lights
7) HIC-61 Thermal shield pump #1, switch and lights
8) HIC-62 Thermal shield pump #2, switch and lights

9.2.1.1.5 Panel "J" Vertical Section

1) SPKR-1 Process room noise level, speaker
2) TI-5 Thermal column, tank outlet temperature indicator
3) TI-6 Cryogenic, experimental outlet temperature indicator
4) LI-7 Thermal column, tank level, indicator
5) FI-3 Cryogenic, cooling flow, indicator
6) PIC-2 D_2O experimental cooling pressure, indicator-controller
7) FIC-1 Thermal column, flow, indicator-controller
8) R-1 Process room noise level speaker, switch
9) VU-1 Process room noise level, indicator
10) LR5/6 Helium and CO_2 gas holders level, recorder

9.2.1.1.6 Panel "J" Sloping Section

1) HIC-29 D_2O experimental cooling isolation valve, DWV-26, switch and lights
2) HIC-31 D_2O experimental return, isolation valve, DWV-24, switch and lights
3) HIC-79 Helium blower, #1, discharge valve, HEV-7, switch and lights
4) HIC-78 Helium blower, #2, discharge valve, HEV-6, switch and lights
5) HIC-74 D_2O experimental cooling, #1 booster pump, switch and lights
6) HIC-73 D_2O experimental cooling #2 booster pump, switch and lights
7) HIC-80 D_2O experimental cooling, to sump #1, Valve DWV-23, switch and lights
8) HIC-86 CO_2 purge fan, switch and lights

9.2.1.1.7 Panel "H" Vertical Section

1) CR-1 D_2O ion exchanger, inlet-outlet conductivity, recorder
2) TI-4 D_2O heat exchanger, HE-2, inlet temperature indicator
3) TI-3 D_2O heat exchanger, HE-2, outlet temperature indicator
4) PI-4 Recombiner inlet pressure indicator
5) PI-3 Recombiner outlet pressure indicator
6) FI-1 D_2O to purification heat exchanger, HE-2, flow indicator
7) FI-4 Helium flow indicator
8) FI-18 D_2O purification to primary system, flow indicator
9) FI-2 D_2O ion exchanger flow
10) TR-4 Recombiner temperature recorder
11) HIC-112 D_2O purification to primary system valve, DWV-39, controller
12) HIC-26 D_2O ion exchanger bypass valve, DWV-22, controller

9.2.1.1.8 Panel "H" Sloping Section

1) HIC-113 Emergency tank, make-up valve, DWV-40, switch and lights
2) HIC-23 D_2O ion exchanger #2, isolation valve, DWV-17, switch and lights
3) HIC-22 D_2O ion exchanger #1, isolation valve, DWV-19 switch and lights
4) HIC-25 D_2O purification after-filter bypass valve, DWV-19, switch and lights
5) HIC-24 D_2O purification after-filter isolation valve, DWV-18, switch and lights
6) HIC-21 D_2O purification prefilter bypass valve, DWV-15, switch and lights
7) HIC-20 D_2O purification prefilter isolation valve, DWV-14, switch and lights
8) HIC-19 D_2O ion exchanger #1 and 2, bypass valve, DWV-38, switch and lights
9) HIC-71 D_2O storage tank pump #1, switch and lights
10) HIC-72 D_2O storage tank pump #2, switch and lights

9.2.1.1.9 Panel "G" Vertical Section

1) Door 049/2 Elevator door position, lights
2) Door 1212/2 South door position, lights
3) Door 1205/5 North door position, lights
4) Door 1300/1 Truck door position, lights

5) Door 1303/2 Back door position, lights
6) Door 062/1 Sub-pile room door position, lights
7) Door 069/1 Process room door position, lights
8) SF-27 Basement exhaust fan EF-27, switch and lights
9) SF-11 Basement supply fan SE-11, switch and lights
10) SF-1 Second floor booster fan, SF-1, switch and lights
11) SF-3 First floor booster fan, SF-3, switch and lights
12) SF-2 First and second floor supply fan, SF-2, switch and lights
13) SF-12 Radiological laboratory and storage pool booster fan, SF-12, switch and lights
14) EF-2 Dilution fan EF-2, switch and lights
15) EF-3 First and second floor exhaust fan, EF-3, switch and lights
16) EF-4 Irradiated air exhaust fan, EF-4, switch and lights
17) SF-19 Decontamination recirculation fan, SF-19, switch and lights
18) EF-5AC AC power to emergency exhaust fan, EF-5, switch and lights
19) EF-6AC AC power to emergency exhaust fan, EF-6, switch and lights
20) EF-5DC DC power to emergency exhaust fan, EF-5, switch and lights
21 EF-6DC DC power to emergency exhaust fan, EF-6, switch and lights

9.2.1.1.10 Panel "G" Sloping Section

1) HIC-18 Reactor overflow valve, DWV-37, switch and lights
2) HIC-13 Experimental D_2O emergency cooling valves, DWV-29 and 30, switch and lights
3) HIC-15 D_2O emergency cooling to plenum valves, DWV-32 and 35, switch and lights
4) HIC-14 D_2O emergency cooling to reserve tank valves, DWV-32 and 33, switch and lights
5) HIC-11 Reactor fill valve, DWV-11, switch and lights
6) HIC-10 Reactor fill valve, DWV-12, switch and lights
7) HIC-27 Sump to emergency tank valve, DWV-20, switch and lights
8) HIC-28 Sump to storage tank valve, DWV-21, switch and lights
9) HIC-9 Emergency tank drain valve, DWV-13, switch and lights
10) HIC-12 Reactor fill valve, DWV-31, switch and lights
11) HIC-69 Sump pump to hot waste, switch and lights
12) HIC-70 Emergency sump pump, switch and lights

9.2.1.1.11 Panel "F" Vertical Section

1) AN-2 Annunciator panel #2
2) CRM-1-11 Log rate-meter, radiation detector
3) CRM-1-12 Log rate-meter, radiation detector
4) CRM-1-13 Log rate-meter, radiation detector
5) RD-1-1 First floor area monitor #1, indicator
6) RD-1-2 First floor area monitor #2, indicator
7) RD-1-3 First floor area monitor #3, indicator
8) RD-1-4 First floor area monitor #4, indicator
9) RD-1-5 Second floor area monitor #1, indicator
10) RD-1-6 Second floor area monitor #2, indicator
11) RD-1-7 Storage pool area monitor #1, indicator
12) RD-1-8 D_2O pump room area monitor #1, indicator
13) RD-1-9 D_2O pump room area monitor #2, indicator
14) RD-1-10 Control room area monitor #1, indicator
15) RI-1 to RI-10 Area monitor, indicator and selector switch
16) LI-1 Emergency cooling tank level, indicator
17) FI-11 Void shutdown flow, indicator
18) PIC-5 Helium void pressure regulator valve HEV-2, controller
19) S-75 Void and dump reset button

9.2.1.1.12 Panel "F" Sloping Section

1) HIC-68 Air void valve, HEV-8, switch and lights
2) HIC-67 Helium void valve, HEV-1, switch and lights
3) HIC-8 Shutdown cooling pump #1 isolation valve DWV-8, switch and lights

4) HIC-7 Shutdown cooling pump #2 isolation valve DWV-7, switch and lights
5) HIC-17 Moderator dump valve DWV-9, switch and lights
6) S-6 Evacuation alarm, button
7) HIC-54 AC power to shutdown cooling pump #1, switch and lights
8) HIC-52 AC power to shutdown cooling pump #2, switch and lights
9) HIC-55 DC power to shutdown cooling pump #1, switch and lights
10) HIC-53 DC power to shutdown cooling pump #2, switch and lights

9.2.1.1.13 <u>Panel "E" Vertical Section</u>

1) AN-3 Annunciator panel #3
2) Reactor "ON" light
3) TI-2 Reactor outlet, inlet temperature differential
4) TI-1 D_2O heat exchanger HE-1, outlet temperature indicator
5) LI-2 D_2O storage tank level, indicator
6) FI-5 Reactor overflow, indicator
7) TR-2 Reactor outlet temperature recorder
8) LR-1 Reactor level, recorder
9) FR-2 Outer plenum flow, recorder
10) FR-3 Inner plenum flow, recorder
11) TR-3 Reactor inlet temperature, recorder
12) TC-1 Reactor inlet temperature, controller
13) HIC-1 Outer plenum valve, DWV-1, switch and lights
14) HIC-2 Inner plenum valve, DWV-2, switch and lights

9.2.1.1.14 <u>Panel "E" Sloping Section</u>

1) S-80 Annunciator test, button
2) HIC-6 (future)
3) HIC-5 Primary pump #3 isolation valve, DWV-5, switch and lights
4) HIC-4 Primary pump #2 isolation valve, DWV-4, switch and lights
5) HIC-3 Primary pump #1 isolation valve, DWV-3, switch and lights
6) HIC-51 (future)
7) HIC-50 Primary pump #3, switch and lights
8) HIC-49 Primary pump #2, switch and lights
9) HIC-48 Primary pump #1, switch and lights
10) HIC-16 Reactor overflow valve, DWV-10, switch and lights

9.2.1.1.15 <u>Panel "D" Vertical Section</u>

1) AN-4 Annunciator panel #4
2) NI-1 Log count rate, indicator
3) AN-6 Annunciator panel #6
4) NI-3 Log flux level, indicator
5) NI-2 Period, indicator source range
6) S-71 NC-1, NC-2, selector switch
7) NR-1-4 Flux level, recorder
8) NI-4 Period indicator
9) S-72 NC-3, NC-4, selector switch
10) XI-1 Shim rod #1 position, indicator
11) XI-2 Shim rod #2 position, indicator
12) XI-5 Regulating rod position, indicator
13) XI-3 Shim rod #3 position, indicator
14) SI-4 Shim rod #4 position, indicator

9.2.1.1.16 <u>Panel "D" Sloping Section</u>

1) S-3 Major scram, button
2) NI-6 Flux level, indicator
3) NI-5 Servo deviation, indicator
4) S-77 Power demand setting, dial
5) S-2 Scram button
6) S-78 Annunciator reset, button

7) S-79 Annunciator acknowledge, button
8) S-9 Shim rod ganged withdrawal, switch
9) S-8 Man-auto control transfer, switch and lights
10) S-70 Range selector, switch
11) S-73 NC-6, 7, and 8, selector switch
12) S-10 Shim rod #1 control, switch
13) S-20 Shim rod #2 control, switch
14) S-50 Regulating rod control, switch
15) S-30 Shim rod #3 control, switch
16) S-40 Shim rod #4 control, switch

9.2.1.1.17 Panel "C" Vertical Section

1) AN-5 Annunciator panel #5
2) FR-1 Reactor outlet flow, recorder
3) TR-1 Reactor ΔT recorder
4) KWH-1 Reactor thermal power, recorder
5) BTUR-1 SW Reactor thermal power calibration switch
6) BTUR-1 Reactor thermal power, integrator
7) FI-12 Secondary cooling water flow, indicator
8) TI-8 Cooling tower temperature, indicator
9) HIC-44 Cooling tower fan #1, switch and lights
10) HIC-45 Cooling tower fan #2, switch and lights
11) TR-14 Secondary inlet to heat exchanger, HE-1, temperature recorder
12) TI-13 Secondary outlet from heat exchanger, HE-1, temperature recorder
13) HIC-46 (future)
14) HIC-47 (future)

9.2.1.1.18 Panel "C" Sloping Section

1) S-4 Control power, switch and lights
2) S-1 Rod drive power, switch and lights
3) S-76 Scram reset, button
4) HIC-83 Secondary shutdown cooling pump, switch and lights
5) HIC-40 Secondary booster pump #1, switch and lights
6) HIC-41 Secondary booster pump #2, switch and lights
7) HIC-42 Secondary booster pump #3, switch and lights
8) HIC-31 (future)
9) HIC-36 Secondary main pump #1, switch and lights
10) HIC-37 Secondary main pump #2, switch and lights
11) HIC-38 Secondary main pump #3, switch and lights
12) HIC-39 (future)

9.2.1.1.19 Panel "B" Vertical Section

1) Clock
2) Scram logic selector switch position, lights
3) Experimental ports by-pass position, lights
4) Rod drop test by-pass position, lights
5) Power range set point selector position, lights
6) Future position lights
7) Future position lights
8) Future position lights
9) Future position lights
10) RR-1 Radiation level, recorder
11) RR-3 Radiation level, recorder

9.2.1.2 Nuclear Instrumentation Cabinet - Panel "A". Figure 9.1 shows the location of this cabinet with respect to the main control board. This cabinet contains all the amplifiers, calibration units and power supplies for the reactor. Refer to Section 9.3 for a detailed description of this equipment.

9.2.1.3 Leak Detector Panel. This control room mounted equipment provides a means of locating heavy water leaks by means of 15 selector switches, each with 5 positions, and indicator lights. Refer to Section 5.3.3 for a description of the leak detectors.

9.2.1.4 Annunciator Panels. Visual and audible alarms are provided in the control room for all variables which cause control rod action, and for variables which require the prompt attention of the operator. Visual alarms are given by lighted windows which are arranged in 6 panels mounted on the control board. Audible alarms are given by a horn which sounds when the annunciator is activated. Figures 9.2 through 9.5 show the face display for each panel.

9.2.2 MANUAL SCRAM STATIONS. Five manual scram stations are located throughout the reactor building. They consist of momentary push-buttons and are located on each wall of the first floor or experimental level, and on the north exterior wall of the sub-pile room.

9.2.3 EMERGENCY VENTILATION CONTROL STATION. This control panel is located in the waste storage vault, which is located underground, approximately 200 feet east of the main entrance of the building. A list of displayed instruments and switches is given below:

1) ACV-11 Basement recirculation isolation valve, position lights
2) ACV-1 First and second floor intake valve, position lights
3) ACV-2 Basement intake valve, position lights
4) ACV-3 Basement exhaust valve, position lights
5) ACV-6 Irradiated air exhaust valve, position lights
6) ACV-7 First and second floor exhaust valve; position lights
7) ACV-10 EF-5 and 6 basement isolation valves, position lights
8) ACV-12 Building ventilation valve, switch and position lights
9) SF-19 Recirculation fan, switch and position lights
10) EF-5AC AC power, building exhaust fan, switch and position lights
11) EF-5DC DC power, building exhaust fan, switch and position lights
12) ACV-4 Inlet to EF-5, position lights
13) ACV-8 Outlet of EF-5, position lights
14) EF-2 Dilution exhaust fan, switch and position lights
15) EF-6AC AC power to EF-6, switch and position lights
16) EF-6DC DC power to EF-6, switch and position lights
17) ACV-5 Inlet to EF-6, position lights
18) ACV-9 Outlet of EF-6, position lights
19) PI-10 Reactor building pressure indicator
20) FI-19 Emergency exhaust indicator
21) RI-1-8 First floor area monitor indicator

9.2.4 MECHANICAL AND ELECTRICAL SUPERVISORY PANELS

9.2.4.1 First Floor Corridor Annunciator Panel. This annunciator panel is located in the front corridor of the "cold" office wing. This panel contains 30 annunciator plates, with annunciator test and reset buttons. Figure 9.6 illustrates the layout of the panel. Actuation of any annunciator in this panel also sounds a common control room alarm.

9.2.4.2 Basement Annunciator Panel. This panel is located in the basement of the "cold" wing, in the vicinity of the emergency diesel generators. It is similar to the first floor corridor panel. Figure 9.6 shows the panel. Any alarms received also sounds a common control room annunciator.

9.2.5 WASTE SYSTEM CONTROL PANELS

9.2.5.1 Basement Control Panel. This control board serves the hot waste system. It is located in the basement of the "cold" wing. Instruments and switches on this panel are as follows:

1) FI-30 Batching tank pump discharge flow, light
2) LI-32 Batching tank #1 level, indicator
3) LI-35 Batching tank #2 level, indicator
4) LI-40 Retention tank level indicator
5) LI-42 Holdup tank level indicator

6) SV-217 RWV-13, valve, indicating lights
7) HIC-90 (future)
8) HIC-96 RWV-7 valve, switch and lights
9) HIC-106 RWV-6 valve, switch and lights
10) HIC-91 (future)
11) HIC-97 RWV-2 valve, switch and lights
12) HIC-107 RWV-8 valve, switch and lights
13) HIC-92 Retention tank pump #2, switch and lights
14) HIC-98 RWV-4 valve, switch and lights
15) HIC-93 Retention tank pump #1, switch and lights
16) HIC-99 RWV-5 valve, switch and lights
17) HIC-109 RWV-12 valve, switch and lights
18) HIC-94 Hold-up tank pump, switch and lights
19) HIC-100 RWV-9 valve, switch and lights
20) HIC-110 RWV-1 valve, switch and lights
21) HIC-95 Batching tank pump, switch and lights
22) HIC-111 RWV-3 valve, switch and lights

9.2.5.2 Hot Lab Control Panel. This panel serves the hot waste system also. It is located in the hot lab area. Below is a listing of displayed instrumentation and available switches.

1) LI-31 Batching tank #1 level, indicator
2) LI-34 Batching tank #2 level, indicator
3) FI-31 Batching tank pump discharge flow, indicator
4) HIC-102 Batching tank pump, switch and lights
5) HIC-103 RWV-7 valve, switch and lights
6) HIC-104 RWV-9 valve, switch and lights

9.2.5.3 Health Physics Lab Panel. Four level indicators for the hot waste system are located in the health physics lab; they are as follows:

1) LI-30 Batching tank #1 level, indicator
2) LI-33 Batching tank #2 level, indicator
3) LI-39 Retention tank level, indicator
4) LI-41 Hold-up tank level, indicator

An additional area monitor panel with 21 indicators, which display duct filter activity, is provided in the Health Physics Office.

9.3 NUCLEAR INSTRUMENTATION

9.3.1 NUCLEAR FLUX MEASUREMENT

9.3.1.1 General. The nuclear instrumentation utilized for the NBSR is solid state and plug-in modular construction. It provides surveillance of reactor power from source level through intermediate, to full power by monitoring thermal neutron flux. It provides signals to the safety system and for reactor operation. The instrumentation consists of 8 measuring channels contained in one cabinet.

With the exception of the detector the nuclear instrumentation may be tested, and calibrations and trip points verified (without scramming the reactor) by built in test circuits. Testing of each channel may be performed by operation of the controls located on the front panels. Calibration of each channel requires removal of the drawers or modules to gain access to the calibration controls.

With few exceptions most modules may be removed during operation without causing or preventing a reactor scram. In all cases removing a module will not prevent a bonafide scram from occurring (provided safety system is in 2 of 3 coincidence) for maintenance or testing. Most maintenance may be accomplished by module replacement by spares and the defective module may be repaired on the bench. Over 95% of the servicing can be handled with 15 different plug-in modules.

The design is a hybrid transistor and magnetic amplifier design. In most cases the magnetic amplifier is used to convert the analog signal (when it is above 1 microampere) to bistate output. The transistor amplifiers are used only to amplify the currents below 1 microampere and in the power range channel bistable trips (input units) accept the signal directly from the ion chamber without the use of transistor amplifiers. The transistor is used in the bistable trip here as an inverter (change DC to square output) to supply the magnetic amplifier with 100 kc.

The NBSR nuclear safety system is designed to be as failsafe as possible (a design objective that is difficult to be achieved) and to tolerate single failures without a significant reduction in safety and monitor for failures which do not fail in the safe direction. Testing features are incorporated to locate the failed component as quickly as possible and modular construction permits replacement of the unit in a minimum of time. In the event of card failure, it should be possible to locate, replace and verify the operation of the safety systems in a short interval of time.

9.3.1.2 Log N Amplifiers - 07-15 and 07-16. The Leeds and Northrup (L&N) Log N Amplifier is a solid-state, modulator stabilized, DC current amplifier employing a logarithmic DC feedback. The difference between the input signal and the feedback current is modulated, amplified, demodulated, and fed back to the input through a logarithmic element. The carrier frequency of the amplifier is 455 kc. Power for the modulator and synchronous demodulator is derived from a crystal controlled transistor oscillator.

The modulator is a capacitance bridge suppressed carrier type employing voltage sensitive silicon diode capacitors. The output of the modulator is amplified by a transistorized AC amplifier, synchronously demodulated by a diode switch and filtered. The DC from the demodulator is fed back through a temperature compensated logarithmic element. In addition, this DC signal is amplified by a magnetic amplifier to produce a 0-4 ma DC signal proportional to the logarithm of the full input range. The magnetic amplifier has a current balanced output circuit isolated from ground.

One of the outstanding characteristics of the L&N Log N amplifier is the unique self-biasing feature of the modulator which makes the zero stability virtually independent of the capacitance diode saturation current. This feature greatly improves the long-term zero stability of the unit and eliminates the periodic zero-balance adjustments required on many other units. Another important feature of the modulator is that it is operated at 455 kc rather than at some lower frequency to reduce the output impedance of the modulator so that conventional components and techniques can be used for coupling the modulator to the transistor amplifier. The higher operating speed also improves the speed of response of the unit.

The modulator, in addition to its modulation function, provides power gain with a very low noise level. The modulator is capable of modulating currents of less than 10^{-12} amp with a signal to noise ratio of better than 10:1.

The transistorized AC amplifier is an untuned bandpass amplifier, the gain of which is stabilized by local AC feedback so that standard non-selected components can be used and no adjustments are required in production, operation, or normal maintenance. The synchronous demodulator is a transformer coupled diode switch. Again no selected components are required and no adjustments are necessary.

The logarithmic feedback is a solid state silicon diode network which is completely temperature compensated for both zero and range stability over the operating temperature range of 10°C to 40°C. This is important when considering the accuracy and stability of the Log N amplifier since all silicon diodes are susceptible to zero and range shifts with temperature variations. The seven decade operating range of 10^{-10} to 10^{-3} amp is good over the temperature range stated above.

To produce the 0-4 ma, DC, proportional to the logarithm of the current input, a linear, fixed current gain magnetic amplifier measures the output of the demodulator. Magnetic feedback is used to fix the current gain of the device over the load range of 500 to 2500 ohms.

The front panel meter has a logarithmic scale of 10^{-10} to 10^{-3} amp. A front panel switch switches the meter to read the trip unit set point currents so that the set point adjustments

are made on the same scale on which the variable is read. Another front panel switch inserts either of the two calibration currents to check the operation of the amplifier. The calibration points are at 10^{-9} amp and at 10^{-4} amp. In addition, test jacks and a "Power On" light are also included on the front panel. The calibrate switches are connected to the startup permit circuit to prevent reactor startup unless all the switches are in operate position.

The specifications for the Intermediate Range or Log N Amplifier are shown in Table 9.3-1.

Table 9.3-1

Manufacturer:	Leeds & Northrup
Model No.:	6932
Range:	10^{-10} to 10^{-3} amp
Accuracy:	1/5 decade, 10^{-10} to 2.5×10^{-4} amp 1/3 decade, 2.5×10^{-4} to 10^{-3} amp
Response Time:	Varies with signal level: less than 20 milli-seconds at 10^{-3} amp; adjustable from 1.0 to 10.0 seconds by component selection at 10^{-10} amp.
Output:	0-4 ma. DC, into 500 to 2500 ohms, proportional to the logarithm of the input signal. Output ungrounded.
Calibration:	Two built-in calibration points at 10^{-9} and 10^{-4}amp
Ambient Temperature:	10°C to 40°C
Power Supply:	± 10 volts, ± 1.0%, DC

9.3.1.3 Period Amplifier - 07-11, 07-12, 07-13, and 07-14. The period amplifier is a solid-state rate amplifier designed to operate from any 0-4 ma DC signal such as the output of the Log N amplifier and to develop an output proportional to the time rate of change of that input.

The rate circuit is an R-C differentiating circuit with a variable time constant adjustable from 0.2 to 20 sec. The input is completely isolated from ground. The output of the rate circuit is fed into a linear magnetic current amplifier with an adjustable gain which permits changing range from 2.5 to 25 decades per minute (10% per sec to 160% per sec). The final stage is a fixed current gain magnetic amplifier producing a 4.0 ma. full range signal. Magnetic feedback is used to fix the current gain over the load range of 500 to 2500 ohms.

The front panel meter calibrated in seconds gives a reading of the rate variable. A front panel switch allows selection of the remote trip unit set point current for indication on the above meter so that the set point adjustment is made on the same scale on which the variable is read. Another front panel switch selects the three calibration points and inserts built-in ramp functions into the circuit. These calibration points check the calibration of the amplifier and also permit checking of a period trip unit. In addition, test jacks and a "Power On" light are also included on the front panel. The calibration switch is connected to the startup permit circuit to prevent startup if calibration switch is not in the operate position. The specifications for the period amplifier are shown in Table 9.3-2.

9.3.1.4 Compensated Ionization Chambers. The compensated ionization chambers ND-3, ND-4, and ND-5 are Westinghouse type 8074, which are boron coated, electrically compensated ionization chambers. The chamber is cylinderical, 3-3/16" in diameter and 23-13/16" in length. The case and electrodes are constructed of an alloy of 97% magnesium and 3% aluminum. The chamber filling gas is 1 atmosphere of nitrogen. The signal electrode is coated with 96% B-10 (enriched boron). The chamber insulation is stabilized polystryene with HN type coaxial cable connectors.

The neutron sensitivity is 4×10^{-14} amp/nv. The gamma sensitivity is 3×10^{-11} amp/R/hr uncompensated and 3×10^{-13} amp/R/hr compensated. The compensating voltage required is from 0 to 100 volts negative. The minimum insulation leakage resistance from the signal electrode to ground is 10^{13} ohms. The chambers have a plateau length of at least 300 volts at 2×10^{10} nv and the slope does not exceed 5%.

Table 9.3-2. Specifications for Startup Channels NC-1, NC-2, NC-3, and NC-4

Manufacturer:	Leeds & Northrup
Model:	6933
Input Requirements:	
Input Signal:	0-4 ma, DC
Input Power	+10 v, DC (+0.1%): 10 w
Output Characteristics:	
Range:	-30 to ∞ to +3 sec
Output Signal:	0-4 ma into 500 to 2500 ohms
Drift:	0.1% including warmup, less than 0.05%/24 hr after 1 hr (at infinite period)
Accuracy:	±5% full scale
Indications:	Period front panel meter -30 to ∞ to +3
Auxiliary Outputs:	
Remote Meter:	50 mv (ungrounded)
Remote Recorder	10 mv
Ambient Temperature:	10°C to 40°C
Response Time:	Adjustable from 0.2 to 20.0 sec time constant
Calibration:	Three built-in ramp signals

9.3.1.5 Uncompensated Ionization Chambers. The uncompensated ionization chambers ND-6, ND-7, and ND-8 are Westinghouse, type 8075, which are boron coated chambers. The chambers are cylindrical 3-1/6" in diameter and 13-7/8" in length. The case and electrodes are constructed of aluminum. The chamber filling is an argon-nitrogen mixture at 1 atmosphere pressure. The signal electrode is coated with 96% B-10 (enriched boron). The chamber insulation is high purity alumina with HN type of coaxial cable connectors.

The neutron sensitivity is 4.4×10^{-14} amp/nv and the gamma sensitivity does not exceed 5×10^{-11} amp/R/hr. The minimum insulation leakage resistance from signal electrode to ground is 10^{11} ohms. The chambers have a plateau length of 750 volts at 7.9×10^{9} nv with less than a 5% slope.

9.3.1.6 Compensated Ion Chamber Power Supply - 07-3, 07-4, 07-5, 07-6, 07-7, 07-8, and 07-9. The solid-state compensated ion chamber (C.I.C.) power supply produces both the high level positive voltage and the medium level compensating negative voltage for the operation of a compensated ion chamber. The positive voltage is 100 to 1000 volts DC adjustable in 100 volt steps and the negative voltage is continuously adjustable from 0 to 100 volts DC.

The input power from the channel 10 volt DC power supply is regulated to better than 0.1% for line and load variations. The C.I.C. power supply further regulates this input power to compensate for load variations.

The output voltages are developed by a multivibrator circuit consisting of 2 transistors and a special transformer which produces a square wave voltage from the DC source. This voltage is stepped up by conventional transformer action to the high voltages desired. The transformer is wound with two secondary coils each having ten taps. One coil produces the high level positive voltage selectable from 100 to 1000 volts in 100 volt steps. The second coil produces the medium level negative voltage variable from 0 to 100 volts. A ten-turn potentiometer adjustment on the second coil permits continuous adjustment from 0 to -100 volts. The outputs of both the secondary coils are rectified and filtered to produce the low ripple, regulated DC voltages at the outputs. A front panel meter permits reading and adjustment of both the voltage levels. A front panel two-position switch permits selection of either voltage for indication. There are 2 front panel output voltage adjustments:

1) A ten-position switch to select high voltage in 100 v steps.

2) A ten-turn potentiometer to continuously adjust the negative voltage between 0 to 100 v. A red indicating lamp indicates that the 10 v DC input is connected.

The specifications for the power supplies are shown in Table 9.3-3.

Table 9.3-3 C.I.C., U.I.C. and Fission Chamber Power Supply

Manufacturer:	Leeds & Northrup
Model No.:	6934
Type:	Dual, High Voltage and Compensating Voltage
Output Voltage Range and Polarity:	
Positive:	100 to 1000 in ten steps
Negative	0 to 100 continuously adjustable
Output Voltage Regulation:	
Positive:	600 to 1000 v; 0-2 ma better than 3%
Negative:	30 to 100 v; 0-2 ma better than 0.5%
Ripple:	2 mv peak to peak
Controls:	
Positive:	10 position switch
Negative:	10 turn potentiometer
Meter Range:	switch
Power Input:	10 v (± 0.5%); DC
Power Consumption:	6 watts

9.3.1.7 Instrument Power Distribution. The instrument and control power distribution for the NBSR is illustrated by the single line diagram Figure 9.7. The power to supply vital equipment is obtained from the critical power panel mounted on the control room wall. The power for this panel may be obtained from either motor control center MCC-A5 or MCC-B6 by the manual transfer switch which is mounted on the control room wall.

The normal operation is to obtain power from motor control center MCC-A5 through transformer T-10. In the event that power fails to this motor control center the inverter-diverter will start and run as a DC motor driving an AC generator which will supply power until either of the diesel generators start.

The safety system power supplies, nuclear instrumentation, process instrument power supplies, area monitors, radiation monitors, public address system, and rod position indicators obtain power from this panel. Thus upon loss of power from the motor control centers the vital monitoring systems and emergency systems will be powered from the inverter-diverter or the diesel generators. The power for the critical power panel is 3 phase 208/120 v, AC. Each of the process instrument power supplies and safety system power supplies are connected so that the loss of a single phase will not shutdown the plant. This is discussed under the respective sections for these devices.

9.3.1.8 Nuclear Instrumentation Power Monitor. The nuclear instrumentation is powered from ± 10 volt supplies. The output of the +10 volt supply is monitored by relay K07-19 and the output of the -10 volt supply is monitored by K07-20. If either of these supplies fail a contact of these two relays will de-energize relay K103 in the safety system which in turn will shutoff the shim rod clutch current. A set of contacts of these relays will produce an alarm, if either relay is de-energized, via Annunciator 4-5.

9.3.1.9 Source, Intermediate, and Power Range Indicators and Recorders. The operator may monitor the reactor nuclear parameters by indicators and recorders mounted in the reactor console. The operator may select the channel he choses to monitor via switches mounted on the console.

The source ranges are monitored by the meters NI-1 and NI-2 and recorder NR-1. The meters and recorder may be switched from channel NC-1 to NC-2 by a selector switch S-71. The meters are General Electric (G.E.) type D.B.40 taut band indicators of ± 1% accuracy and a 320° scale. The log count rate indicator NI-1 has a logarithmic scale calibrated from 1 c.p.s. to 10^5 c.p.s. The period indicator NI-2 has a scale calibrated from -30 to ∞ to +3 sec.

The recorder NR-1 is a Honeywell class 17 strip-chart recorder which records the log count rate by indicating the 0-10 mv drop across a resistor in series with the log count rate meter output. The recorder has an accuracy of 0.25% of span and the scale is calibrated from 1 to 10^5 c.p.s. The response time for full scale indication is 1 sec.

The intermediate ranges are monitored by the meters NI-3, NI-4, and recorder NR-2. The meters and recorder may be switched from channel NC-3 to NC-4 by a selector switch S-72. The meters are the G.E., type D.B. 40. The indicator NI-3 has a logarithmic scale calibrated from 10^{-10} amp to 10^{-3} amp and accepts the 0 to 4 ma output of Log N amplifiers 07-15 or 07-16. The period indicator has a scale calibrated from -30 to ∞ to +30 sec and accepts the 0-4 ma output of period amplifiers 07-13 or 07-14.

The recorder NR-2 is a Honeywell class 17 strip-chart recorder which records the 0-10 mv output of the Log N amplifiers 07-15 or 07-16. The recorder scale is calibrated 10^{-10} amp to 10^{-3} amp.

The output of the G.E. picoammeter is recorded and indicated by the recorder NR-3, which is a Honeywell class 17 strip-chart recorder. The recorder has a scale calibrated 0 to 100% of scale.

The power level of the NBSR from 1% to 150% of full power may be monitored by the indicator NI-6 which may be switched from NC-6, NC-7, and NC-8 by the selector switch S-73. Since the power range channels do not have an amplifier the output of the ion chambers are displayed directly on this indicator.

The recorder NR-4 is a Honeywell class 17 strip-chart recorder and is also switched between channels NC-6, NC-7, and NC-8 by switch S-73. The recorder measures a 10 mv drop across a resistor which is in series with the chamber output.

The servo deviation meter NI-5 is a G.E., type D.B. 40 meter which is calibrated -15% to 0 to +15% with zero center scale. The meter indicates the difference between setpoint of the power demand potentiometer and the reactor power level. The meter is shorted by relay contacts KD7-13 to prevent damage to the meter if the deviation is excessive.

9.3.2 POWER RANGE INSTRUMENTATION. The power range channels NC-6, NC-7, and NC-8 provide surveillance of reactor power over the upper 2 decades of power level, i.e., 1% to 100%. A selection of 2 power level scram points (13% and 130%) is possible by a switch located on the set point panel. Amplifiers are not necessary in these channels and the signal from the uncompensated ion chambers are fed directly to the bistable trip circuits. Each power range channel consists of a chamber, bistable trips and high voltage supplies as illustrated in Figure 9.8.

The high voltage supply is the same supply as utilized for the intermediate range channels. (See Section 9.3.1.6) The compensating voltage function portion of this supply is not utilized in this application.

The current from the uncompensated chambers (see Section 9.3.1.5) flows through a calibrate-operate switch through the bistable trips through resistors to ground. The resistors provide a 0-50 mv drop and a 0-10 mv drop to operate the console mounted indicator and recorder respectively. An indicator mounted in the nuclear rack provides level indication in the individual channels. A switch mounted under each of these indicators provides indication of the trip set point in one position and power level indication in the other position.

The calibrate-operate switch in the calibrate position permits the introduction of a test current into the channel via a 10-turn calibration potentiometer. The test current may be monitored by a test meter which measures the current into each channel. The calibration current may be used to check the channel for calibration and verify the trip settings by raising the input current above the set point without providing a reactor shutdown (if the safety system is in 2 of 3 coincidence). The calibrate-operate switches have contacts connected into the startup permit circuit which prevents reactor startup, if one of the switches is inadvertently left in calibrate, and sounds an alarm, Annunciator 4-27.

A power range deviation comparator compares the output of NC-6, NC-7 or NC-8. If a deviation in output current greater than ± 10% is detected an alarm will operate Annunciator 4-35. The purpose of this comparator is to warn the operator that an excessive deviation in current has occurred which may be due to one of the following causes:

a) Excessive flux shift
b) Chamber does not give required output current due to faulty chamber
c) Voltage on a chamber has fallen below the amount necessary to keep the chamber saturated
d) Excessive chamber signal cable leakage
e) A cable open circuit due to defective connectors, etc.

The comparator deviation alarm function is accomplished by the bistable trip circuits 07-31, 07-32, 07-33, 07-34, 07-35, and 07-36. The bistable trips are identical to those used elsewhere in the system. Each bistable compares the current flowing in each of 2 channels and will trip when the current difference exceeds the set point. A trip of any one of these bistables, i.e., output goes to "1" state will produce a "0" state of NOD-7, de-energize relay K07-12 and produce an alarm, Annunciator 4-43.

The comparator trips may be tested by means of the trip test selector switch and test push button #3. The test trips each bistable trip thus verifying its operation and operation of the NOD logic circuit, relay K07-12 and Annunciator 4-43.

9.3.3 POWER LEVEL CALIBRATION. The BTUR system will be used to give an accurate power level indication from 100 Kw to 10 Mw. This indication will be used to provide calorimetric data which will be used to position the neutron detectors and achieve power level calibration. These data will mostly be used to position the ion chambers for the power level channels. The intermediate range chambers may then be calibrated at the high level end. (See Section 9.4.2.5 for details of this system.)

9.3.4 LINEAR POWER LEVEL CHANNEL AND AUTOMATIC CONTROL CHANNEL. The linear power level channel permits measurement of reactor power level from 60 x 10^{-8} to 150 percent in 18 steps. This channel also provides automatic power level control to ±0.5% of level from 0.1% to 100% power. A remote range change switch mounted on the console permits the operator to select the proper range. The power level is displayed to the operator via recorder NR-3. Also, signals from this instrument are used as the error signal for the automatic control channel.

The pico-ammeter is a G.E. model NC01 which is a vibrating reed stabilizing pico-ammeter with the following specifications:

Range	10^{-12}amp to 10^{-3}amp full scale	
Accuracy	10^{-12}amp to 10^{-11}amp	4%
	2.5 x 10^{-11} to 2.5 x 10^{-7}amp	3%
	10^{-6} to 10^{-3}amp	3%
Stability	better than ±0.25% long term	
Response Time	10^{-9}amp to 10^{-3}amp	2 milliseconds
	10^{-10}amp to 10^{-3}amp	20 "
	10^{-11}amp to 10^{-3}amp	200 "
	10^{-12}amp to 10^{-3}amp	2000 "
Recorder Output	0 to 10 mv and 0 to 1 ma	

The 0 to 10 volt pico-ammeter output is compared to a signal from the power demand potentiometer and the difference is indicated by servo deviation indicator NI-5 which is mounted on the console. The difference is also monitored by the bistable trips 07-29 and 07-30. If an excessive deviation trips one of the bistable trip circuits an output will be sent to NOD-8 on card 07-65. If either of the bistables trips, relay K07-13 will be de-energized and the reactor will go off automatic control, and Annunciator 4-44 will alarm.

The power demand potentiometer is adjustable from 0 to 100% of range. The difference signal is amplified by two operational magnetic amplifiers with an overall amplification factor of 60. This amplified error signal is then fed to the DIAT (Directional Impulse Adjusting Type) controller which drives the regulating rod to raise or lower reactor power to match demand set point.

The proportional bank, reset, and rate settings will be determined by experiment with an analog model of the reactor and should not require any further adjustment.

Proportional band action is that in which the regulating rod is moved by an amount proportional to the deviation in reactor flux from set point. The proportional band dial determines the ratio of deviation of reactor flux to regulating rod position change. The proportional band dial is calibrated from 0 to 500% of recorder scale. Reset action corrects rod position at a rate proportional to the deviation of the reactor flux from the control point setting. It is adjusted by the reset dial which is calibrated in repeats per minute. Reset action, therefore, repeats the proportional action until the reactor flux returns to the control point setting. The dial is calibrated from .01 to 100 repeats per minute. Rate action changes the regulating rod position by an amount proportional to the rate of change of reactor flux. Generally, very little rate control can be used with a reactor.

A gain dial adjusts the dead band of the controller. To reduce wear on the controller and drive mechanism this control will be adjusted for a 1/2% dead band. The dead band is a band in which no control action will occur. The gain dial is calibrated from 0 to 10.

The DIAT controller simulates the action of the control rod so that a tachometer feedback or a position transmitting slidewire on the regulating rod is not necessary. The relay that operates the control rod drive motor relays K-56 and K-57 starts an integrating action in simulating motion of the rod drive.

The proportional position control is defined as:

$$Y = -100\ \theta/A$$

Y = change in rod position (% of full travel)
θ = change in flux (% of controller scale)
A = proportional band (% of controller scale)

For a 100% proportional band the rod is fully stroked when the pico-ammeter changes full scale. A servo deviation ctonrol limits this to 10% of span.

The change in flux θ will be ±0.5% of power level.

The regulating rod is limited to approximately 0.5% reactivity to prevent the reactor from going prompt critical in the event of a controller malfunction. The regulating rod position is displayed on the console by a fine and course position indicator.

The regulating rod may be operated manually by means of S-50 mounted on the console. The reactor is placed on automatic control by the manual-auto transfer switch, S-8. The reactor may be returned to manual control at any time by operation of the S-50 switch.

9.3.5 SOURCE RANGE OF STARTUP CHANNELS NC-1 AND NC-2

9.3.5.1 Source Range Detectors. The source range or startup channel detectors consist of a fission counter ND-1, and BF_3 counters ND-2. The fission counter is a Westinghouse type 8073 which is coated with enriched U-235. The counter is cylindrical 2-1/16" in diameter and 9-3/4" long. The case and electrodes are constructed of aluminum. The counter is filled with argon-nitrogen mixture at 1 atmosphere pressure. The counter is coated with U-235 (90% enrichment) to a thickness of 2 mg/cm^2 with a total quantity of 1.72 grams. The counter insulation is alumina with a HN coaxial cable connector.

The minimum sensitivity is 0.7 cps/nv with a natural background of 5 cps. The detector may be operated in gamma fields of at least 3.3×10^4 r/hr before gamma pile-up becomes a problem. The output pulse amplitude is 200 microvolts with 0.2 microsecond rise time. The neutron flux range is to 10^5 nv as a counter but may be operated in fluxes up to 10^{10} nv without permanent damage.

The BF_3 proportional counter is comprised of 4 Westinghouse type 6307 counters operated in parallel. The parallel operation is necessary to increase the sensitivity to approximately 16 cps/nv to "see" the source during cold clean core conditions. The individual counters are 1-3/32" in diameter and 12-1/8" long, and constructed of aluminum. The counters are filled with boron trifloride enriched to 96% B-10 to a pressure of 55 cm Hg. The counter insulation is alumina with HN type of coaxial cable connectors.

The minimum sensitivity is 16 cps/nv. The output pulse is 1 to 3 mv (loaded with a 0.1 microsecond rise time. The neutron flux range is up to 10^5 cps. The plateau is at least 200 volts at 2000 volts with less than a 4% slope.

The high voltage is removed from these detectors when both the intermediate range channels indicate greater than 10^{-9} amp. This action is accomplished by means of the bistable trips 07-46 and 07-47 when reactor power exceeds approximately 0.0003% of full power. When both bistables 07-46 and 07-47 trip, a "0" state exists at the NOD-4 logic input which de-energizes the K07-03 relay and removes the ND-1 and ND-2 high voltage.

9.3.5.2 Log Count Rate Amplifier 07-1 and 07-2. The Log Count Rate Amplifier is a solid state device with an output signal proportional to the logarithm of the pulse counting rate. The normally separate functions of linear amplifier and log count rate amplifier have been combined on a single chassis. The output signal is 0 to 4 ma DC into 500 to 2500 ohms impedance for a 1 to 10^5 cps range. Power to operate the chassis is obtained from a separate -10 volt DC power supply.

The first stage of the linear amplifier module, 101145, incorporates a low noise transistor as a low input impedance current amplifier. Current amplification techniques improve the signal-to-noise characteristics since the low impedance termination reduces significantly the amplitude of the noise induced along the line or appearing at the input terminals. The voltage amplifiers following the input circuit are doublets. Each doublet has its own stabilization by feedback from the output emitter follower. Nominal gain for the doublet for the large pulse rise times involved will be approximately five. A gain control is provided to prevent overloading the amplifier since operation from both fission and BF_3 chambers will be necessary. The discriminator section module, 444119, consists of a differentiator containing a variable capacitance diode. A bias voltage is adjustable to give a desired level. A derivitive amplifier triggers a scale-of-two circuit.

The rectangular shaped pulse output of this circuit feeds the log synthesizer module, 444120. The log synthesizer circuit consists of 6 solid state diode pumps (Cooke-Yarborough) with the output being a direct current summation of the outputs of the individual pumps. The resultant current is then proportional to the logarithim of the count rate over the 5 decade range. A 5 Kc operational magnetic amplifier module, 444067, accepts this output directly and produces a 0-4 ma output current. Zero offset is balanced out by the adjustment on the magnetic amplifiers.

The front panel meter will have a logarithmic scale of 1 to 10^5 cps. A front panel switch changes the meter to read the trip unit set point currents so that the set point adjustments are made on the same scale on which the variable is read. In addition, test jacks and a "Power On" light are included on the front panel. A test pulse generator is incorporated which supplies a 10 cps and 50 Kc signal. A switch selects the level of the signal supplied to the BF_3 or fission counter. The detailed specifications of the log count rate amplifiers are given in Table 9.3-4.

Table 9.3-4 Specifications of Log Count Rate Amplifiers 07-1 and 07-2

Manufacturer:	Leeds & Northrup
Model:	6931
Range:	1 to 10^5 cps
Input Impedance:	75 ohms
Rise Time:	0.1×10^{-6} sec
Pair Resolution:	0.6×10^{-6} sec
Pulse Height Discrimination Range:	10 to 100%
Accurace:	± 0.2 decades
Response Time:	Varies with signal level; 1st decade 6 sec, last decade 70 milliseconds
Output:	0-4 ma, DC into 500 to 2500 ohms; proportional to logarithm of counting rate
Ambient Temperature:	10°C to 40°C
Power Supply:	± 10 v, ± 1.0%, DC

9.3.6 DETECTOR LOCATIONS. The NBSR detectors are arranged to provide monitoring of neutron flux of the reactor from source level to full power as illustrated by Figure 9.9. After the reactor has been operated for some time the photo neutron production is expected to produce approximately 10^5 nv at the detector locations so that the source range channels will be off scale. The intermediate range channels will then provide the required reactor protection to 10% of full power after which the power range channels will provide protection.

The detectors (excepting ND-1) are located in vertical thimbles in the biological shield just beyond the thermal shield. Lead windows are provided in the thermal shield to reduce the gamma flux at the detector. The gamma flux is calculated to be 40 r/hr at full power and 0.2 r/hr at shutdown at the detector locations.

The calculated temperature at full power is not expected to exceed 160°F under worst case conditions. The normal operating ambient temperature is approximately 115°F. The detectors are designed to operate at 175°F.

The arrangement of the detectors in respect to the source and shim arms is illustrated in Figure 4.2.

The source range channel detector ND-1 is inserted in one of the 4" experimental thimbles to provide a more sensitive channel to fission neutrons when a cold clean core is loaded. After the core has built up sufficient fission products to produce a high enough shutdown photo-neutron flux, this detector will be withdrawn to a low flux region. This detector is housed in a watertight housing which will permit submersion in water up to 30' deep. The outer case is aluminum with stainless steel fittings.

The location of the ionization chambers is such that the flux at the chamber will be relatively independent of movement of the shim rods. The source range channels NC-1 and NC-2 "look" through the core at the startup source and both are required to produce 10 cps before a withdraw permit is obtained.

9.3.7 DETECTOR POSITIONING. The detectors are positioned manually in their thimbles by raising and lowering the detector housing by a flexible steel cable.

The upper end of each instrument tube provides a stepped section for a shield plug. The chamber positioning cable and coaxial cables are routed through a recess in the side of the instrument wells. The connections for the coaxial cables are made in the boxes provided on top of the biological shield. The cables are routed to the control room through individual conduit through the biological shield. The length of the coaxial cables are approximately 100 ft.

The coaxial cable employed for the signal leads on the ion chambers is a special low noise cable, RG71. The RG71 coaxial cable has a capacitance of approximately 13.5 pico farads per foot. The coaxial cable has a layer of graphite between the shield and the insulation to reduce the introduction of electrostatic charges in the cable due to vibration. The source range channels employ RG149 for NC-2 and RG150 for NC-1. The polarizing voltage for the chambers is supplied by RG59 cable which is rated at 2300 volts. The connectors employed for signal leads on ND-3, ND-4 and ND-5 are ceramic type HN connectors. The minimum signal leakage resistance to ground on these detectors is greater than 10^{12} ohms.

9.3.8 NUCLEAR INSTRUMENTATION FOR INITIAL LOADING AND STARTUP. During the initial approach to criticality and subsequent approaches to criticality after a complete refueling, an auxiliary neutron source will be utilized. The source will be of adequate strength to obtain the required count rate to clear the startup permit (10 cps). The source will be placed in one of the vertical experimental thimbles.

For the initial approach to criticality additional instruments will be installed to provide the required experimental data. The normal reactor instrumentation will be used to provide reactor safety by moving some of the chambers into the through tubes. This change will increase the neutron flux at the chambers by a factor of approximately 10^4 and will allow the reactor scram levels to be set at very lower power levels, i.e. 100 w. The period scrams will be active from approximately 0.1 w to 1 Mw. The chambers will be moved in the through tubes to establish the required reactor protection levels as the low power experiments require increased power levels.

Two of the power range channel detectors will be moved to the through tubes and the other detector will remain in its permanent position. This will be done so that a correlation between the temporary detector positions and the permanent positions will be obtained. The power level channels NC-6, NC-7, or NC-8 in their temporary positions will provide trip levels down to approximately 100 w. The remaining power range channel trip level may be set down to approximately 100 Kw. The safety system logic will be operated in the 1 of 3 mode during critical and low power testing so that a scram level in any one channel will provide reactor shutdown.

The linear power level channel NC-5 will provide surveillance from source level to approximately 25 Kw when the ion chamber is placed in a through tube. This channel will be used to establish initial power level calibrations at startup and to provide reference data each time the other chambers are moved. The chamber ND-5 will remain in one position up to approximately 100 Kw before being moved to its permanent position.

9.4 NON-NUCLEAR INSTRUMENTATION

The process instrumentation utilized for the NBSR reactor is of solid state design utilizing electrical transmission of signals. The transistor amplifiers (oscillator position detectors), have been removed from the transmitter bodies in high radiation areas and are mounted in a cabinet outside the process equipment room. The transmission signal is a 4 to 20 ma direct current signal which does not require any elaborate shielding of cables. The receivers, recorders, controllers, and bistable trips (monitor switches) are connected in series forming a series circuit. The series loops are powered by a ± 42 volt ungrounded power supply.

The advantages of this system are that only two wires are needed for signal transmission and power to the field mounted instruments. The instrument calibrations are unaffected by variations in the load resistance or line resistance. All the power for the instrument loops connected to the reactor safety system is obtained from a central source, ± 42 v, thus power to the field is not required. The supply voltage is monitored by voltage monitoring relays K111 and K111a, which produces a reactor scram via the K103 relay upon loss of both supplies. The power supplies are connected in parallel through two isolating diodes so that the loss of one supply will not shutdown the reactor but produce an alarm signal. Each supply is capable of supply power to all the instrument loops.

The process instruments connected to the reactor safety system for scram and rod interlock functions are shown in Figure 9.10. The channels which do not produce safety functions but relay alarm or indication are shown on the individual flow sheets (refer to Section 7). The transmitters, receivers, recorders, indicators, and monitor switch specifications which are most common in the system are indicated in the following paragraphs.

In the NBSR most of the process variables, flow, and level are measured by Honeywell Model 29211 differential pressure cell, Δ P/I. The specifications for this unit are as follows:

Range:	Adjustable from 0 to 20" to 0 to 250" of H_2O
Displacement:	Less than 0.1 cu. in. for 250" differential
Temperature:	-40 to +250°F
Materials:	316 stainless steel
Maximum working pressure:	1500 psi
Frequency response:	1 cps
Accuracy:	± 1/2%

The Honeywell Model 30000 millivolt to current converter is used on the NBSR for most temperature measurements. The specifications for this unit are as follows:

Range:	0 to 2.5 mv to 0 to 100 mv
Output:	4 to 20 ma into 1200 ohms maximum
Temperature:	+ 40°F to 120°F
Burnout:	Upscale Burnout (if input opens instrument fails up scale)
Frequency Response:	9 cps with 1.2 to 410 ohms source impedance
Accurace:	± 1/4%

Most of the process pressures are measured in NBSR by the Honeywell Model 30200 process pressure to current transmitter ($PP/_I$). The specifications for this unit are as follows:

Input:	10 to 10,000 psi
Output:	4 to 20 ma into 800 ohms maximum load
Temperature:	-40 to +150°F
Materials:	Stainless steel
Frequency Response:	5.0 cps
Accuracy:	3/4%

The recorder receivers most utilized on the NBSR are for recording functions only, or for recording and control functions. Some of the recorders are single pen, Model 32301 and come are dual pen, Model 32351. The specifications for the recorders are as follows:

Input Signal:	4 to 20 ma DC
Chart Speeds:	3/4 in/hr to 12 in/hr
Response:	3 sec for 98% deflection
Temperature Range:	40° to 120°
Chart Size:	4"
Accuracy:	± 0.5% of span

The indicators used on the NBSR are 5-1/2" rectangular 4 to 20 millampere meters. The devices are used in every loop except where recorders or Vertical Scale Indicators (VSI) are employed.

Most of the controllers employed on the NBSR to hold process variables constant are the Honeywell 33311 series. The control mode, i.e., one mode, two mode, or three mode, depends upon the requirements of the process variable. The specifications are as follows:

Proportional Band Adjust:	one mode 5% to 100% or 50% to 1000%; two and three mode 1% to 100% or 10% to 1000%
Reset Adjust:	manual reset ± 50% of span; auto reset 0.01 to 10 repeats/minute for proportional band of 1 to 100%; 0.1 to 100 repeats/minute for proportional bank 10% to 1000%
Rate:	0 to 20 minutes
Control Action:	direct or reverse

There are two types of control elements used for the NBSR; a current to pressure (I/P) Honeywell, Model 31200, valve operator for diaphragm control valves and the electro-pneumatic valve positioner. The I/P transducer converts a 4 to 20 ma DC signal into a 3 to 15 psi air pressure signal to operate the NBSR valves. The specifications are as follows:

Input:	4 to 20 ma
Output:	3 to 15 psi
Air Consumption:	less than 0.35 scfm
Temperature Range:	-40°F to + 150°F
Accuracy:	± 1% of span

The electro-pneumatic positioners mounted on the NBSR control valves use the force-balance principle to feed or bleed air to the valve diaphragm until the force from the feed-back spring balances the 4 to 20 ma signal flowing through the force coil. The specifications are as follows:

Air Consumption:	0.3 scfm (maximum)
Linearity:	1% of span
Maximum Stroke Speed:	1 in per sec

The Honeywell monitor switch, 33400, single or dual point is used to convert an analog current signal to a relay contact output to operate alarms or relays in the NBSR safety system. The monitor switch operates from 117 volt AC from the critical power panel.

The monitor switches are connected in series with the process instrument loops. The loop 4 to 20 ma current signal is converted to a voltage by an input resistor. This voltage is compared to an adjustable reference voltage, i.e., set point and fed to a two-stage transistor amplifier which operates a single pole double throw relay. The unit may be connected for "high" trip or "low" trip operation. The monitor switch is operated in a "fail safe" fashion so that this relay is de-energized upon loss of power.

The dual set point unit is two single units mounted in one case. The specifications for the monitor switch are as follows:

Signal input:	4 to 20 ma
Set point:	full range 0 to 100%
Output:	SPDT relay
Temperature:	40° to 130°F
Drift:	± 1%
Operating Differential	1%
Response Time:	0.050 to 0.150 sec for step changes*

The process instrumentation connected to the safety system is powered by 2 separate DC power supplies rated at 600 ma at 42 volts DC. The power supplies are Honeywell 33901 which are transistor regulated power supplies of ± 1% regulation for a 100% load change. Each supply is connected to a 1 Kva sola transformer to a different phase of the critical power supply. The output of each supply is connected in parallel through an isolating diode so that the loss of one supply or phase will not shutdown the reactor. The output of each power supply is monitored by a relay K111 and K111a which have contacts connected in parallel in the scram circuit, K103, so that the loss of both supplies will produce a reactor scram. The voltage monitoring relays also have contacts connected in series to an alarm, Annunciator 4-5, so that the loss of either supply will produce an alarm signal.

Each of the process instrument loops is fused so that a single loop fault will not disable the system; i.e., drag down the output of the loop power supplies. (See Figure 9.10)

9.4.1 PROCESS SCRAM RELAY LOGIC CIRCUITRY. Each channel of the process instrumentation connected to the scram system controls the coil of a single relay mounted in a safety system relay drawer #2. Each of these single relays have duplicate sets of isolated contacts connected in series with the two final scram relays. (See Figure 9.10) When any of the channels reach the pre-set scram limits, the scram relay coil is de-energized and the corresponding contacts open. The relays remain de-energized as long as the abnormal condition exists.

The final scram relays K2000 and K2001 have 4 sets of contacts each connected in series with the shim rod clutch power supplies. When these relays are de-energized, the contacts open and de-energize the shim rod clutches. Thus no single failure, i.e., a relay contact welding closed, can prevent a bonafide scram from occurring. These relays cannot be energized again until all scram conditions are cleared and the safety system is manually reset by S76. All relays in the system are operated in a "fail safe" fashion, i.e., loss of power or opening of a relay coil will de-energize the relays.

The scram relay coils, K2000 and K2001, as do all relays in the rod control system, receive 48 volt DC power which is obtained from two supplies connected in parallel via isolation diodes. Each of the two supplies is powered from a different phase of the critical power. The failure of one of the DC supplies will not de-energize the scram relay or rod control relays but will produce an alarm.

The process instrument and rod control safety system are powered from two separate DC power supplies rated at 10 amp at 48 volt DC. These power supplies are Sola 281561 power supplies which consist of a Sola stepdown transformer and a full wave bridge rectifier with appropriate filtering. Each power supply is connected to a different phase of the critical power supply. Since each power supply is able to supply the total load the supply will continue to operate even if one of the phases is de-energized or one supply fails. Two relays monitor the output of the power supplies and actuate an alarm, Annunciator 5-5, if power from either supply should fail.

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* Depends upon magnitude of step input.

9.4.1.1 Reactor Outlet Flow Rate, FRC-1. The reactor outlet flow is monitored by flow element FE-1, a venturi tube calibrated to 1/2%. The differential pressure across FE-1 operates a Honeywell 29211 differential pressure transmitter. The 4 to 20 ma signal output, a square root function, is transmitted to the control console. This signal operates a Honeywell Model 33400 monitor switch FA-1 which scrams the reactor if the flow rate drops below the set point. The flow transmitter also provides a signal to a Honeywell Model 31101 square root extractor, FX-1. The functional block diagram of the system is shown in Figure 9.10.

The monitor switch de-energizes the auxiliary relay Ka-01-283 when the signal exceeds set point. The auxiliary relay de-energizes the scram relay K2000 and K2001 and operates Annunciator 4-9.

The square root extractor accepts the input signal from the flow transmitter, a square root function, and converts it to a linear function. The square root extractor consists of an input amplifier which changes its gain as a function of the input amplitude by varying gain through a switching circuit composed of eight parallel diode switches. The accuracy from 15% to 100% of full scale is within ±1/2%. The frequency response is one cycle per second and operates over a 40°F range. The output of the square root extractor feeds the flow recorder, FR-1, which is calibrated from 0 to 6000 gpm, a linear scale. The 4 to 20 ma signal, proportional to flow rate, flows through a slide wire on the differential temperature recorder TR-1, and is transmitted to the BTUR recorder.

9.4.1.2 Primary Coolant Liquid Level, LRC-1. The reactor coolant liquid level is monitored by a liquid level transmitter, LD-1. The transmitter sends a signal to the control room which operates a recorder, LR-1, and 4 level switches, LA-1A, LA-1B, LA-2A, and LA-2B. The level switches provide annunciation if level is too high or low, provide shim rod rundown if level reaches a low condition, and scram the reactor if level reaches low low level. In addition, if the moderator level is very low, the primary circulating pumps will stop and shutdown cooling pumps will start. The shutdown cooling may be stopped manually by the operator.

The instrumentation associated with this system consists of a Honeywell 30300 differential pressure transmitter, LD-1 (similar in principle to Honeywell 29211 except for range) transmitting a 4 to 20 ma signal to a console mounted recorder, LR-1 and two dual monitor switches LA-1A, LA-1B, LA-2B, and LA-2A. The differential pressure cell has a range of 400" H_2O of which approximately 165" H_2O of the range will be suppressed.

The recorder is a Honeywell 32301 strip chart recorder and is calibrated from 0 to 200" D_2O.

The Honeywell 33400 dual monitor switch provides signals to operate Annunciators 3-1 and 3-2 for high and low level alarm. Another dual monitor switch LA-2A and LA-2B operates auxiliary relays Ka-01-251A and Ka-01-251 which are connected in the safety system to provide scram and shim rod inserts safety functions. (See Figure 9.10).

The normal operating level for the primary coolant is 180.25" D_2O. The high level alarm LA-1A will operate at 2" D_2O above normal and the low level alarm LA-1B will operate at 3" D_2O below normal. A shim rod rundown will occur if level drops to 6" D_2O below normal and a scram will occur at 12" D_2O below normal level. The response time of the instrumentation is approximately 1 cycle per second and is short compared to the time to produce the above referenced changes in liquid level.

9.4.1.3 Void Shutdown Flow, FCA-17. The function of this channel is to measure and monitor the flow at the inlet to the bubbler. The flow rate is measured by a differnetial pressure transmitter, FE-17, an indicator FI-11 mounted on the console and monitored by flow switches FA-17A and FA-17B. The flow switches operate auxiliary relays Ka-01-299 and Ka-01-299A which provide the necessary scram and alarm functions (See Figure 7.2).

The flow rate to the inlet of the bubbler 0-200 cfm is measured by the differential pressure drop across an orifice plate FE-17. The pressure drop is measured by a Honeywell 29211 differential pressure transmitter FD-17 which has a span of 0 to 100" H_2O. The flow rate is indicated on the control board by a Honeywell 4 to 20 ma meter.

The flow alarms and safety actions are provided by a Honeywell 33400 dual monitor switch. The monitor switch relays are connected to auxiliary relays Ka-01-299 and Ka-01-299A in the reactor safety system. The relay Ka-01-299 will scram the reactor if the flow rate exceeds set point 20 cfm and produce an alarm via Annunciator 4-8. The other relay Ka-01-299A will produce an alarm via Annunciator 2-6, excepting when the air void switch or helium void switches HIC-68 or HIC-67 are intentionally operated.

9.4.1.4 Reactor Inner Plenum Flow, FRC-3. The reactor inner plenum flow is measured by a 12" venturi tube FE-3 which develops a 33" differential at 3500 gpm. This differential is measured by a differential pressure transmitter FD-3 which transmits a signal to a console mounted recorder FR-2, and monitored by a flow switch FA-3. The flow switch operates an auxiliary relay Ka-01-285 in the safety system which produces the scram function and alarm, Annunciator 4-10.

The differential pressure drop across FE-3 is measured by a Honeywell 29211 differential pressure transmitter FD-3 which transmits a 4 to 20 ma signal to the console. The differential pressure transmitter has a span of 0 to 33" H_2O.

The recorder FR-2 is a Honeywell 32301 strip chart recorder calibrated 0 to 3500 gpm, a square root function.

The flow switch FA-3 is a Honeywell 33400 monitor switch. The monitor switch relay de-energizes an auxiliary relay Ka-01-285 which is connected to the reactor safety system. This relay will scram the reactor if the flow is less than 2000 gpm and produce an alarm via Annunciator 4-10.

9.4.1.5 Reactor Outer Plenum Flow, FRC-4. The reactor outer plenum flow is measured by a 10" venturi tube FE-4, which develops a 27.3" H_2O differential at 1600 gpm flow. This differential is measured by a differential pressure transmitter FD-4 which transmits a signal to a console mounted recorder FR-3, and a flow switch FA-4. The flow switch FA-4 is connected to the process instrument safety system and provides an alarm, Annunciator 4-11.

The differential pressure drop across FE-4 is measured by a Honeywell 29211 differential pressure transmitter FD-4 which transmits a 4 to 20 ma signal to the console. The differential pressure transmitter has a span of 0 to 27.3" H_2O. The recorder FR-3 is a Honeywell 32301 strip chart recorder with a scale calibrated 0 to 1600 gpm, a square root function.

The flow switch FA-4 is a Honeywell 33400 monitor switch. The monitor switch de-energizes an auxiliary relay Ka-01-286 in the reactor safety system. This relay will scram the reactor at the pre-set scram point by de-energizing K103, K2000, and K2001 relays and produce an alarm via Annunciator 4-11.

9.4.1.6 Reactor Differential Temperature, TCA-100. The differential temperature channel measures the differential temperature across the core and scrams the reactor if the differential temperature exceeds the set point. Thermocouples mounted in the inlet line of the reactor are connected so that the differential between the two (100°F to 130°F) is transmitted to the control room to provide indication and signal to the safety system to scram the reactor and sound an alarm.

The thermocouples TD-101 and TD-102 are type J (Iron-Constantan) which are mounted in the same wells as the resistance bulbs for channel TR-1.

A 0 to 1 mv to current converter (TT-100) accepts the differential signal from the two thermocouples and transmits a 4 to 20 ma signal to the control room. A 4 to 20 ma indicator (TI-100) mounted on the console is calibrated from 0 to 35°F full scale. Connected in series with the loop is a monitor switch (TA-100) Honeywell 33400 which opens a set of contacts in series with relay Ka-01-58 to de-energize the scram relays K103, K2000, and K2001 if the differential temperature exceeds set point. Contacts of this relay also operate Annunciator 4-13.

9.4.2 RUNDOWN SYSTEM. The process instrumentation connected to the rundown system causes all rods to be inserted in the core at normal speed until the signal causing the condition is corrected. Most of the systems connected to this system monitor variables that have slow response or variables that preceed a scram condition, i.e., reactor power level greater than 115% or reactor power greater than 11.5 Mw.

9.4.2.1 Primary Coolant Temperature, TRC-2. This channel monitors the primary coolant outlet temperature by a resistance bulb element TD-2, which is mounted in the 18" outlet line from the reactor. The signal from this detector is fed to a temperature transmitter TT-2 which supplies a signal to the control room to operate a temperature recorder TR-2 and a temperature alarm TA-2. The temperature alarm operates an auxiliary relay Ka-01-267 which has a set of contacts in the rundown circuit K-107 and one set of contacts to operate the Annunciator 4-19.

The temperature detector is a Honeywell type A resistance bulb the same as that employed in the channel TRC-1. The change in resistance of this bulb for temperature change is converted to a 4 to 20 ma signal by the resistance to current converter, Honeywell 30020. The 4 to 20 ma signal is fed to the control room where it operates a temperature recorder, Honeywell 32301 strip chart recorder calibrated from 50 to 150°F. The 4 to 20 ma signal also operates a temperature alarm switch, Honeywell 33400. The relay in the monitor switch operates the auxiliary relay Ka-01-267 in the reactor safety system to provide an all rod rundown if the temperature exceeds 118°F.

9.4.2.2 Thermal Column D_2O Level, LCA-4. This channel monitors the level in the thermal column tank and provides a signal to the reactor safety system to produce an all rod rundown in the event the level drops below the set point. The level is measured by a differential pressure cell LD-4, which transmits a signal to a console mounted indicator LI-7, and to a level alarm switch LA-6. The flow switch operates an auxiliary relay Ka-01-262 which has contacts connected into the safety system and to Annunciator 4-22.

The hydrostatic head of the D_2O in the thermal column tank is measured by a Honeywell 29211 differential pressure transmitter with a span of 25 to 55" D_2O. The transmitter transmits a 4 to 20 ma signal to the control console to operate a Honeywell 4 to 20 ma meter which indicates the tank level and is calibrated from 25" to 55".

The current then flows through a level switch, Honeywell 33400 monitor switch. The monitor switch relay de-energizes the relay Ka-01-262 in the reactor safety system, which will rundown all shim rods until the level condition is corrected. The contacts of the relay Ka-01-262 also operate Annunciator 4-22. (See Figure 9.10)

9.4.2.3 Primary Coolant Level, LRC-1. This channel is the same channel that provides the reactor scram function discussed in Section 9.4.1.2. The 4 to 20 ma signal from this channel operates the level switch LA-2A which provides an all rod rundown signal if the level falls below 6" of normal and sounds an alarm Annunciator 4-21.

The level alarm Honeywell 33400 monitor switch de-energizes an auxiliary relay Ka-01-251 in the reactor safety system. This relay is connected in the rod rundown system as illustrated in Figure 9.10 and also operates the Annunciator 4-21.

9.4.2.4 Thermal Shield Coolant Flow, FIA-15. This instrument channel accepts a differential pressure signal from a venturi tube FE-15, and operates a flow indicator on the console FI-9, and a double flow switch FA-15A and FA-15B to produce a pre-warning alarm Annunciator 5-13, and finally a rod rundown and Annunciator 4-24 in the event the flow falls below the set point. (See Figure 7.15)

The flow tube FE-15 produces a 250" differential pressure at 750 gpm. This differential pressure operates a Honeywell 29211 differential pressure transmitter. The transmitter supplies a signal to operate a Honeywell 4 to 20 ma indicator with a square root function scale calibrated 0 to 750 gpm. The Honeywell 33400 dual monitor switch relays de-energize the relays Ka-01-298 and Ka-01-298A in the safety system. The relay Ka-01-298A when de-energized produces an all rod rundown until the low flow condition is corrected.

9.4.2.5 Reactor Thermal Power, BTUR-1. The BTUR-1 system computes the reactor thermal power, records and integrates the power level. The system also provides a signal to the reactor safety system to produce an all rod rundown in case the power level exceeds 11.5 Mw.

A signal is taken from the reactor outlet flow loop square root extractor FX-1, 0 to 16 ma, and is fed to a retransmitting slide wire on the differential temperature recorder TR-1.

The output signal from this retransmitting slide wire is then a product of differential temperature and flow rate and feeds the BTUR-1 recorder.

The differential temperature recorder TR-1 measures the change in resistance of two resistance bulb temperature detectors. One of the temperature bulbs is mounted in the inlet line to the reactor TD-1-1 and one is mounted in the outlet line TD-1-2.

The resistance bulbs are Honeywell type A and have been calibrated at the National Bureau of Standards. A recording Wheatstone bridge, differential temperature recorder TR-1 indicates and records the difference in resistance between the two bulbs. The recorder is a Honeywell Class 18 strip chart recorder with full scale calibration of 20°F.

The BTUR recorder accepts a millivolt signal from the differential temperature recorder. The BTUR recorder is a Honeywell Class 17 strip chart recorder with the following specifications:

Accuracy:	± 0.25% of span
Linearity:	± 0.1% of span
Dead Band:	± 0.15% of span
Response Time:	approximately 5cps for 10% change

A ball and disc integrator mounted in this recorder integrates the pen position in respect to time and displays the results on a "Veeder Root" type of register on the console. A transistorized trip circuit mounted in the recorder de-energizes a relay Ka-01-60 in the safety system when the input signal exceeds the set point. This relay has one set of contacts connected in the rod rundown circuit and another set operates Annunciator 4-23.

A calibrate switch mounted on the console permits the operator to check the calibration of the system at any time. This switch, when placed in the calibrate position, places a current into the system simulating a given flow rate and fixed resistors into the differential temperature recorder to simulate a known ΔT. Thus a known product may be introduced into the input of the BTUR recorder which will check the operation and calibration of the BTUR system.

9.4.2.6 Process Instrumentation Testing. The electric transmission of the process instrumentation signals makes it possible to introduce, test, and calibrate signals during operation and prior to routine reactor startups. The system is designed to permit introduction of test currents into the loop to verify calibration of the entire loop (excepting transmitter) and operation of the bistable trip circuits, i.e., monitor switchs.

The system has not been designed utilizing coincidence circuits as is the nuclear instrumentation to permit operational testing without bypassing since redundancy is present in the process itself; i.e., low flow scram is backed up by differential temperature scram, etc. The philosophy is: 1) to permit bypassing a channel for testing as long as an equivalent channel is monitoring the process; 2) require acknowledgement by the operator that a bypass is present; and 3) prevent reactor startup if a bypass condition exists.

An 11 position selector switch mounted in the nuclear instrumentation rack permits the operator to select a channel for testing and calibration. The selection of a channel will sound an alarm and require the operator to acknowledge the annunciator. A test current 4 to 20 ma from a test box may then be introduced in the channel to be tested by a test jack which opens the loop to the transmitter and introduces the test current in series with the receiver and bistable trip (monitor switch). The test current will then permit checking the calibration of the receivers and verify the operability of the monitor switch and the safety set point. The rotary selector switch will inhibit the final safety action by supplying 48 volts DC directly to the safety system relay coil. The rotary switch has additional contacts which are connected into the startup permit circuit to prevent reactor startup if the switch is inadvertently left in a bypass position. (See Figure 9.10)

During shutdown when the radiation levels in the process pump room are low enough, pneumatic test signals may be put into the transmitter which will enable checking of the entire loop. The transmitters have been installed with isolation valves and a quick disconnect fitting in the transmitter body. A pneumatic test set then may be plugged into the quick disconnect fitting and calibration of the entire channel may be checked.

9.4.3 PROCESS INSTRUMENTATION IN THE ALARM SYSTEM. The process instrumentation in the alarm system includes all instrumentation which appear on the annunciator alarm panel. A number of these alarm points have already been discussed in Section 9.4.1 which are the pre-alarms and alarms for the scram and rundown safety functions. The remaining instruments are discussed below. Tables 9.4-1 thru 9.4-5 provide lists of those instruments which alarm only.

9.4.3.1 Temperature Alarms. There are 14 alarm only temperature points in the NBSR system. All these temperature points are monitored by iron-constantan thermocouples mounted in stainless steel wells. The thermocouples are connected to millivolt to current converters, mv/I, Honeywell 30000, which are mounted in an instrumentation rack in the basement of the reactor building outside the process equipment room. The 4 to 20 ma output signal is transmitted to the control room to operate indicators and monitor switches, Honeywell 33400. It should be noted that this equipment is the same as employed in the safety system.

9.4.3.2 Flow Rate Instruments. Flow rate instruments in the alarm system include all instruments which appear on the control room annunciator system shown in Figures 9.2 thru 9.5. The alarm points for scram and safety functions have already been discussed in Sections 9.4.1 and 9.4.2. All flow elements are orifice plates excepting FD-5, FD-9, and FD-15. The flow elements FD-5 and FD-15 are flow tubes and the element FD-9 is a Brooks roto-meter. The differential pressure across the flow elements is measured by differential pressure transducers, Honeywell 29211 Δ P/I. The indicator and alarm switch are the same as employed in the safety system. The Brooks roto-meter, FV-1110-5522 employed for FD-9 has a detector for remote indication of the float position. An alarm unit is mounted in the remote electronics panel and provides a set of N.O. contacts to the annunciator system.

9.4.3.3 Level Instrumentation. Level instruments in the alarm system include all instruments which appear on the control room annunciator system shown in Figures 9.2 thru 9.5. The alarm points for scram and other safety functions have already been discussed in Sections 9.4.1 and 9.4.2. All level instruments are differential pressure transmitters measuring hydrostatic head, excepting LT-5 and LT-6 which are the helium and carbon dioxide gas holder level transmitters and the sump pit alarms. The sump pit alarms are GEM adjustable float operated switches.

9.4.3.4 Pressure Instrumentation. Pressure instruments in the alarm system include all instruments which appear on the control room annunciator system shown in Figures 9.2 thru 9.5. All pressure transmitters are Honeywell 30200 process pressure to current transmitters, PP/I. The pressure alarm switches are the same monitor switch as used elsewhere.

9.4.3.5 Conductivity. The electrical conductivity of the primary coolant D_2O storage pool, thermal shield, demineralized water is measured by a stainless steel conductivity cell which is temperature compensated over the operating range of 0 to 10 micromhos/cm. A stabilized AC voltage is applied to the cell and the resultant current flow is directly proportional to the conductivity of the water. The current flow is amplified and then is utilized to operate external meters, recorders, and alarms. The conductivity cells are Industrial Instrument cells with the following specifications:

Material:	Stainless steel
Range:	0 to 10 micromhos/cm
Fluid Temperature:	80° to 120°F with integral temp. compensation
Fluid Pressure:	200 psi
Cell Constant:	.01
Electrodes:	Platinized gold plate nickle disc shaped to prevent fouling by resins. Ceramic insulation.
Mechanical:	Flange mounted

The conductivity monitor is a Leeds & Northrup Model 4957-2-1 with the following specifications:

Range:	0 to 10 micromhos/cm (non-linear)
Ambient Temperature:	40° to 120°F
Temperature Compensation:	80° to 120°F fluid temperature
Accuracy:	± 3%
Output:	0 to 10 mv, remote meter
Alarm:	Adjustable over full range
Alarm Contacts:	S.P.D.T. 5 amp 120 volts

Table 9.4-1 Temperature Instruments in Annunciator System (Alarm Only)

Instrument No.	Service	Annunciator No.
TRC-3 (TA-3) Temperature	Reactor Inlet Low	3-4
TIA-4 (TA-4) Temperature	Reactor Inlet High	3-3
TIA-5	Future HE-3 Outlet	
TIA-6 (TA-6) Temperature	HE-2 Outlet	3-5
TIA-8 (TA-7) Temperature	Thermal Column Tank Outlet	3-11
TIA-9 (TA-8) Temperature	Cryogenic Experimental Outlet	5-25
TRA-10 (TA-9) Temperature	Recombiner Outlet	3-20
(TA-10) Temperature	Recombiner Internal	3-21
TIA-11 (TA-11) Temperature	Experimental Demineralized Water Heat Exchanger	1-25
TIA-12 (TA-12) Temperature	Cooling Tower	5-1
TIA-15 (TA-13) Temperature	Storage Pool Water	5-18
TIA-16 (TA-14) Temperature	Thermal Shield Heat Exchanger Outlet	5-9
TIA-17 (TA-15) Temperature	Thermal Shield Heat Exchanger Inlet	5-10
TIA-23 (TA-23) Temperature	Thermal Shield Floor Header Outlet	5-11
TIA-24 (TA-24) Temperature	Thermal Shield Ring Header	5-12

Table 9.4-2 Flow Instruments in Annunciator System (Alarm Only)

Instrument No.	Service	Annunciator No.
FIA-5 (FA-5) Flow	Heat Exchanger -2 Inlet	3-9
FIA-9 (FA-6) Flow	Reactor Ion Exchanger	3-6
FCA-7 (FA-7) Flow	Thermal Column	3-10
FIA-8 (FA-8) Flow	Cryogenic Inlet	5-26
FIA-9 (FA-9) Flow	Helium System	3-19
FIA-10 (FA-10) Flow	Demineralized Experimental Cooling Ion Exchanger	1-26
FIA-11 (FA-11) Flow	Demineralized Experimental Cooling	1-27
FIA-12 (FA-12) Flow	Secondary Cooling	5-4
FIA-13 (FA-13) Flow	Storage Pool	5-19
FIA-14 (FA-14) Flow	Storage Pool Ion Exchanger	5-20
FIA-15 (FA-15 A & B) Flow	Thermal Shield	5-13
FIA-16 (FA-16) Flow	Thermal Shield Ion Exchanger	5-14
FCA-17 (FA-17B) Flow	Void Shutdown Flow	2-6
FS-1 (FA-18) Flow	Plenum	2-2
FS-2 (FA-19) Flow	Reserve Tank	2-1

Table 9.4-3 Level Instruments in Annunciator System (Alarm Only)

Instrument No.	Service	Annunciator No.
LRC-1 (LA-1A) Level	Reactor D_2O	3-1
(LA-1B) Level	Reactor D_2O	3-2
LIA-3 (LA-4) Level	D_2O Storage Tank	3-12
(LA-5) Level	D_2O Storage Tank	3-13
LCA-4 (LA-21/22) Level		
LIA-7 (LA-13A) Level	Experimental Demineralized Water Tank	1-28
(LA-13B) Level	Experimental Demineralized Water Tank	
LIA-8 (LA-14A) Level	Thermal Shield Storage Tank	3-15
(LA-14B) Level	Thermal Shield Storage Tank	
LIC-9 (LA-15A) Level	Cooling Tower	5-2
(LA-15B) Level		5-3
LRA-5 (LS-5) Level	Helium Gas Holder	3-23
LRA-6 (LS-6) Level	CO_2 Gas Holder	3-25
LS-1 (LA-7) Level	Sump Pit High	2-3
LS-2 (LA-7) Level	Sump Pit Med	2-4
LS-3 (LA-7) Level	Sump Pit Low	2-5
LA-18 (PS-9) Level	Storage Pool High	5-21
(PS-10) Level	Storage Pool Low	5-22
LA-19 (LS-11) Level	Pump Pit Low	5-23
(LS-12) Level	Pump Pit High	5-24

Table 9.4-4 Pressure Instruments in Annunciator System (Alarm Only)

Instrument No.	Service	Annunciator No.
PIC-1 (PA-1A) Pressure	Experimental Demineralized Water	1-30
(PA-1B) Pressure	Experimental Demineralized Water	
PIC-2 (PA-2) Pressure	D_2O Experimental Cooling	5-27
PIC-3 (PA-3A) Pressure (PA-3B)	Recombiner Outlet	3-17
PIA-4 (PA-4) Pressure	Recombiner Inlet	3-18
PIC-5 (PA-5) Pressure	Helium Void	2-8

Table 9.4-5 Conductivity Instruments in Annunciator System (Alarm Only)

Instrument No.	Service	Annunciator No.
CRA-1-1 (CA-1)	Inlet Primary Ion Exchanger	3-7
CRA-1-2B (CA-2)	Outlet Primary Ion Exchanger	3-8
CIA-3 (CA-3)	Demineralized H_2O Experimental Cooling	1-29
CIA-4 (CA-4)	Storage Pool Recirculation System	5-17
CIA-5 (CA-5)	Storage Pool Demineralizer Inlet and Outlet	5-17
CIA-6 (CA-6)	Thermal Shield Inlet	5-16
CIA-7 (CA-7)	Thermal Shield Outlet	5-16

The primary coolant conductivity of the reactor outlet, CRA-1-2A, and the primary coolant ion exchanger outlet, CRA-1-2B, are continuously recorded in the control room by a 2 pen, Honeywell 32351 strip chart recorder. The 0 to 10 mv output of the conductivity monitor is converted to a 4 to 20 ma signal necessary to operate the recorder by a Honeywell 3000 mv to current converter.

The other channels CIA-3, CIA-4, CIA-5, CIA-6, and CIA-7 operate console mounted indicators and the annunciator system.

9.4.3.6 Beam Ports. When a shutter to a beam port is opened the limit, switch LS-4 contacts close and energize a relay K4. A set of N.C. contacts operate Annunciators 1-1 thru 1-9.

9.4.3.7 Grazing Tubes. The circuitry to operate these alarms is the same as for the shutter control, Section 9.4.3.6. The alarms are given by Annunciators 1-10 and 1-11.

9.4.3.8 Cryogenic Shutters. The cryogenic shutter when opened operates a relay to produce control room annunciation via the same circuitry as the shutters discussed in Section 9.4.3.6. Annunciators 1-13 and 1-14 are utilized for this application.

9.4.3.9 Process Room Door. The latch on this door operates a limit switch which is connected to the control room annunciator system. If this door is not latched Annunciator 1-16 will operate.

9.4.3.10 Experimental Alarms. A block of annunciator points have been set aside to provide alarms as required by the various experiments which do not affect reactor safety but are required to protect the experimental equipment.

9.4.3.11 Secondary Experimental Cooling Pumps. Auxiliary contacts of the motor starters for the secondary experimental cooling pumps #1 and #2 are connected to Annunciator 1-32. A set of contacts on the pump control switches "bypass" these contacts when the switch is in the "off" position. When either of the switches are in the "on" position the de-energizing of a motor starter by overload or etc., will operate the appropriate annunciator.

9.4.3.12 Sump Pit Levels. The sump pit levels are monitored by float operated switches at the low, intermediate, and high level positions. The float switches are magnetic float operated switches. A float with an Alnico magnet attached operates magnetic reed switches as the float moves by the switch. The N.O. contacts of these switches are connected to Annunciator 2-3, 2-4, and 2-5.

9.4.3.13 D_2O Leak Detection System. The NBSR leak detection system monitors flanges, joints, seals, and other places of likely leakage in the D_2O systems. If a leak is detected a control room annunciation is obtained.

When a leak occurs the water completes a circuit causing a change in the output of an AC bridge circuit which is amplified by a transistorized AC amplifier and de-energizes a relay. The relay contacts actuate points on the control room annunciator.

The leak detector sensing unit for pump seals, flanges, and other joints is a printed circuit grid on a fiberglass board. The board has a 1" rim to contain the water leakage. The board is approximately 3-1/2" by 2-1/2" with 1/16" width metallic strips which are spaced 1/16" apart so that a drop of water completes the circuit.

The valve bonnet leak detectors are automotive type spark plugs. If a leak occurs in the valve diaphragm the spark plug electrodes will be wetted and complete the circuit to the amplifier.

The bridge circuit and amplifier is a Honeywell Versa-Tran R7088C relative humidity controller. This controller has an AC bridge input circuit which utilizes the resistance of the sensor as one leg. A drop of water on the sensor changes the resistance and causes the bridge to unbalance. At the pre-set alarm point a S.P.D.T. relay is de-energized. The unit is "fail safe" to loss of power, control signal, or relay coil opening. The controller has the following specifications:

Range:	3 KΩ to 3 MΩ adjustable set point
Ambient Temperature:	20 to 125°F
Output:	S.P.D.T. relay, 3 amp, 120 volt, AC resistive

The controllers have been mounted in a panel together with manually operated switches adjacent to the control panel. Normally each controller will monitor up to five points. If any one of the 5 sensors detects a drop of water the controller relay is de-energized causing a control room alarm and lights a red lamp adjacent to the switch. The operator then rotates the switch until the light is extinguished indicating the trouble point. A test position on this switch substitutes a known value of resistance which permits the operator to verify the set point, calibration, and operation of the annunciator at any time. The system design will permit the monitoring of up to 80 leak detector points.

9.4.3.14 Pressure Relief Valve. The pressure relief valves for the helium system HEV-3, and the CO_2 relief valve COV-2, operate limit switches when the valves leave the seated position and operate the appropriate alarm Annunciator 3-22 or 3-26. (See Figure 7.16 and 7.17)

9.4.3.15 Gas Holder Levels. The He and CO_2 gas holder positions are transmitted to the control room where they are recorded on a 2 pen recorder LR-5 and LR-6. The gas holders actuate limit switches which operate the He and CO_2 relief valves HEV-3 and COV-2. A low limit switch and a high limit switch actuate Annunciators 3-23 and 3-25.

A chain attached to the gas holders operates a sprocket which is attached to the shaft of a 1000 ohm precision potentiometer. A Honeywell 30020 resistance to current converter transforms the resistance of the potentiometer, which is proportional to level, to a 4 to 20 ma signal to operate a Honeywell 32351 two pen strip chart recorder in the control room. (See Figure 7.16)

9.4.4 "NON-ALARM" PROCESS INSTRUMENTATION

9.4.4.1 Pressure Gages. Pressure instruments which do not produce a safety action or an alarm are listed below. Unless otherwise indicated by notes on the tables the instruments are locally mounted. The symbols used on the flow diagrams indicate whether the instrument is locally mounted or board mounted. The locally mounted pressure gages are 6" gages with 316 S.S. Bourdon tubes or Monel diaphragms. All pressure instruments mounted on the console produce alarms.

Table 9.4-6 Pressure Instruments

Instrument no.	Service	Range	Figure no.
101	D_2O Pump DP-1 Discharge Pressure	100 psi	5.2
102	D_2O Pump DP-2 Discharge Pressure	100 psi	5.2
103	D_2O Pump DP-3 Discharge Pressure	100 psi	5.2
105	D_2O Pump DP-5 Discharge Pressure	30 psi	5.2
106	D_2O Pump DP-6 Discharge Pressure	30 psi	5.2
107	D_2O Pump DP-1 Inlet Pressure	30 psi	5.2
108	D_2O Pump DP-2 Inlet Pressure	30 psi	5.2
109	D_2O Pump DP-3 Inlet Pressure	30 psi	5.2
111	D_2O Pump DP-5 Inlet Pressure	30 psi	5.2
112	D_2O Pump DP-6 Inlet Pressure	30 psi	5.2

Table 9.4-6 (cont'd)

Instrument no.	Service	Range	Figure no.
113	D_2O Storage Tank Pump DP-8 Discharge Pressure	160 psi	5.2
114	D_2O Storage Tank Pump DP-7 Discharge Pressure	160 psi	5.2
115	D_2O Filter DF-2 Differential Pressure	60 psi differential	7.9
116	D_2O Filter DF-1 Differential Pressure	60 psi differential	7.9
117	Emergency Sump Pump Discharge Pressure	100 psi	7.1
118	Sump Pump #4 Discharge Pressure	30 psi	7.1
119	Heat Exchanger #1 Differential Pressure	60 psi differential	5.1
121	Helium Manifold Pressure	10" H_2O	7.16
122	Helium Pump Inlet Pressure	10" H_2O	7.16
123	Helium Pump Discharge Pressure	30" H_2O	7.16
124	Thermal Shield Cooling System Filter Differential Pressure	60 psi differential	7.15
125	Thermal Shield Cooling System Filter Differential Pressure	60 psi differential	7.15
126	Thermal Shield Pump #2 Discharge Pressure	100 psi	7.15
127	Thermal Shield Pump #1 Discharge Pressure	100 psi	7.15
128	Storage Pool Pump #1 Discharge Pressure	100 psi	7.10
129	Storage Pool Pump #2 Discharge Pressure	100 psi	7.10
130	Storage Pool Purification Boost Pump Inlet Pressure	30 psi	7.10
131	Storage Pool Purification Boost Pump Discharge Pressure	60 psi	7.10
132	Storage Pool Filter Differential Pressure	60 psi differential	7.10
133	Storage Pool Filter Differential Pressure	60 psi differential	7.10
134	CO_2 System Manifold Pressure	10" H_2O	7.17
135	Experimental Demineralized H_2O Filter Differential Pressure	60 psi differential	7.12
136	Experimental Demineralized H_2O Filter Differential Pressure	60 psi differential	7.12
137	Experimental Demineralized H_2O Pump #2 Discharge Pressure	160 psi	7.12

Table 9.4-6 (cont'd)

Instrument no.	Service	Range	Figure
138	Experimental Demineralized H_2O Pump #1 Discharge Pressure	160 psi	7.12
139	He Refueling Cooling System Manifold Pressure	10" H_2O	7.16
PIC-9	Secondary Experimental Cooling Pressure Control		7.11

9.4.4.2 Flow. Flow instruments which do not produce a safety action or an alarm are listed below. The symbols used on the flow diagrams indicate whether the instrument is locally mounted or board mounted.

The FI-18 channel is the only channel that has an indicator which is mounted on the console. This channel consists of a Honeywell 29211 differential pressure transmitter which monitors a 0 to 150" H_2O pressure drop across an orfice plate, FE-18. The flow is indicated on the console by a 4 to 20 ma Honeywell meter which is calibrated from 0 to 60 gpm.

The other flow indicators FI-101, FI-102 and F-103 are mounted on a panel near the storage pool. These units are Republic Type VDP differential pressure transmitters which measure the 0 to 1/2 in. H_2O pressure drop across an orfice plate. The differential pressure cell transmits a 3 - 12 psi signal to vertical scale indicators which are mounted on the control board.

Table 9.4-7 Flow Instruments

Instrument no.	Service	Range	Figure
F1-18	D_2O Ion Exchange Injection Flow	0-60 gpm	7.9
F1-101	Emergency Cooling Tank Helium Flow	0-6 SCFM	7.16
F-102	Reactor Vessel Helium Flow	0-15 SCFM	7.16
F-103	D_2O Storage Tank Helium Flow	0-6 SCFM	7.16

9.4.4.3 Temperature. Temperature instruments which do not produce a safety action or an alarm are listed below.

All the channels with the exception of T1-50, T1-51 and T1-52 consist of iron-constantan type J thermocouples which supply a signal to Honeywell 30000 millivolt to current converters, MV/I. The 4 to 20 ma output signal from the MV/I is indicated on the console by Honeywell 4 to 20 ma indicators.

The channels T1-50, T1-51, and T1-52 are locally mounted thermocouples manufactured by Weiss.

Table 9.4-8 Temperature Instruments

Instrument No.	Service	Range	Figure No.
T1-4	Temperature, HE2 D_2O Inlet	50-125°F	7.9
T1-13	Temperature, HE1 Secondary Outlet	25-125°F	6.1
T1-21	Temperature, Experimental Demineralized Water Inlet	50-125°F	7.12
T1-22	Temperature, Thermal Shield Storage Tank Inlet	50-125°F	7.15
T1-14	Temperature, HE-1 Secondary Inlet	0-100°F	6.1
T1-50	Temperature, Thermal Shield Heat Exchanger Outlet	0-120°F	6.1
T1-51	Temperature, Experimental Demineralized Water Heat Exchanger Outlet	0-120°F	6.1
T1-52	Temperature, D_2O Purification Heat Exchanger Outlet	0-120°F	6.1

9.4.4.4 Level. Level instruments which do not produce a safety action or an alarm are listed below.

All the instruments are locally mounted Jerguson magnetic liquid level gages. This device consists of a float in an external guide tube. The float contains a magnet which actuates a small shutter as the float passes by.

Table 9.4-9 Level Instruments

Instrument No.	Service	Range	Figure No.
L1-101	Level, D_2O Emergency Cooling Tank	0 to 5'	7.1
L1-102	Level, D_2O Storage Tank	0 to 10'6"	7.9
L1-103	Level, Thermal Shield Storage Tank	0 to 11'	7.15
L1-105	Level, D_2O Purge Tank	0 to 5'6"	7.1

9.4.5 PROCESS INSTRUMENTATION FOR INITIAL LOADING AND STARTUP. The initial loading of fuel into the reactor and the "zero power" physics measurements during startup will be carried out with the reactor D_2O level up but no flow in the system. Because the process instrumentation is a "live zero" system; i.e., 4 to 20 ma, it will not be necessary to bypass any of the scrams or other safety functions. As long as the instrumentation loop is complete and functioning a minimum of 4 ma will flow and it will only be necessary to change the set point to 4 ma. Therefore, all process instrument indicating functions will be operative.

As the power level is raised for various tests and the proper set point will be adjusted on each svstem to meet the safety requirements.

9.5 NUCLEAR SAFETY SYSTEM

9.5.1 GENERAL. The conversion of the analog signals from the nuclear instrumentation to a bi-state signal ; i.e., two stable states, to turn the shim rod clutch current on or off is accomplished by the bistable trip or "input unit". The bistable trip outputs are fed to logic circuits which perform the preprogrammed logic operation to satisfy reactor safety requirements.

The "logical operations" are performed by the logic unit for the operation of the output unit or the NOD logic unit for operation of relays. The relays perform rod interlock functions in the relay safety system described in Section 9.5.3. The logic required to satisfy NBSR safety is a selectable 1 of 3 or 2 of 3 logic for scram and 2 of 3 for shim rod rundown of power range signals. Either of the period trips will scram the reactor below 10% of full power.

The final output device that de-energizes the clutch upon receipt of a scram signal is the output unit. The output units require a safe signal from both logic units to energize the clutches. The loss of either signal will shut off the clutch current. In addition, each output unit may only supply two clutches through a diode isolation circuit so that the failure of one output unit to turn off would not prevent the release of at least two rods which would safely shut down the reactor. To further guard against a failure of the output units to de-energize the clutches a NOD logic circuit is fed from the two scram buses so that a scram signal (0 volts) on either bus "A" or "B" (see Figure 9.11) will de-energize a relay when contacts are connected to the relay safety system and will de-energize the safety system relays which in turn interrupt the power to the output units.

The nuclear scram system is continuously monitored by an automatic testing system. This is accomplished by inserting pulse signals into the bistable trips in all possible combinations, and checking the trips through the logic units wiring and through the output unit. The testing is a continuous action and when the schedule is completed the monitor automatically starts the cycle again. If a component failure is detected the tester stops, sounds an alarm and keeps repeating the sequence. Thus with the help of the pulse sequence indicators the failed component may be found and replaced by a spare in a short time.

In addition a complete manual testing scheme is incorporated to test the scram logic and NOD logic circuits; i.e., the interlock functions, during operation. Thus it is possible to locate a failed module in the NBSR safety system during operation, replace the failed module and verify the operability of the replacement by internal testing means.

The NBSR nuclear safety system and nuclear instrument has undergone a testing program of approximately 4 months at the vendors plant under observation of Honeywell and NBSR inspectors. The tests included performance, response time, drift, component failure, elevated temperature, voltage and frequency.

The equipment after being installed was once again performance tested as an integrated system and has been in almost continuous operation since that time with no unsafe failures.

The terminology tabulated below will be used throughout this section of the report.

"1" State	- The "1" State is defined as a nominal ground or positive potential.
"0" State	- The "0" State is defined as a nominal -10 volts or negative potential.
Inputs and Outputs	- The inputs and outputs of the logic unit are restricted to either the 1 or 0 states.
OR Gate	- An OR Gate is a resistor-diode structure arranged so that when one or more inputs change from 0 to 1 the output changes from 0 to 1.

AND Gate	- An AND Gate is a resistor-diode structure arranged so that when all inputs are in the 1 state simultaneously, the output will be in the 1 state.
Switch	- A Switch is a transistorized circuit, the output of which changes from 0 to 1 as its input changes from 0 to 1 and vice-versa.
Inverter	- An Inverter is a transistorized circuit, the output of which changes from 1 to 0 when its input changes from 0 to 1 and vice-versa.
Tripped Logic Unit	- A tripped state of a logic unit is defined as switch output (T OUT) being in the 1 state, and the inverter output ($\bar{T}$ OUT) is in the 0 state.
Tripped Gate, Switch or Inverter	- A tripped gate, switch or inverter in the logic unit is defined by its output being in the 1 state.

9.5.1.1 Bistable Trip or Input Unit Module 444028. The 444028 bistable trip or input is a printed-circuit module used in nuclear reactor instrumentation. It compares one input representing a variable against another input representing a set point. Whenever the variable input exceeds the set point input, the unit is "tripped" and provides a change in its output voltage level that is applied to other units. A blue signal lamp on the input unit indicates the tripped condition.

The bistable trip is used in the NBSR safety system to convert analog current signals to bistate, 0 or 1. In channels NC-6, 7 & 8 the signal inputs to three bistable trips are accepted directly from ion chambers in the reactor, while the set point inputs are obtained from current sources that are adjusted to reference levels determined by the requirements of the system.

By means of appropriate external connections, the bistable trips can be made to memorize (i.e., remain in) its tripped condition until an external resetting signal is applied, or it can be prevented from operating while an external inhibiting signal is applied.

The bistable trip consists of a magnetic amplifier and a switch unit that are provided on two printed-circuit assemblies. The magnetic amplifier is provided on one printed-circuit sub-assembly card and mounted on a card containing the switch unit. A metal cover and bottom plate shield the magnetic amplifier from other units. Interconnections between the two printed-circuit cards are made by means of eight mating connectors on each card.

Two keyed slots in the connector end of the printed-circuit card for the switch unit prevent it from being plugged into the wrong connector. A metal handle on one end of the card facilitates insertion and removal of the input unit. The handle is blue so that the input unit can be readily distinguished from other units whose handles are of different colors. A signal lamp mounted on the handle is also blue. The lamp provides a visual indication of three conditions: when it is on continuously, the input unit is in its tripped condition; and when it is flashing, a component of the input unit has failed. It should be noted that the signal lamp will flash only if testing and resetting pulses are applied to the input unit in the scram system. The specifications for the input units are given in Table 9.5-1.

Table 9.5-1 Bistable Trip Specifications

Manufacturer:	Leeds & Northrup
Model No.:	444028
Description:	Composite high gain magnetic differential operational amplifier and transistor switching unit.
Input Signal:	0 to 5 ma
Reference Signal (Trip Point)	0 to 5 ma
Input Impedence:	200 ohm, both input and reference circuits

Table 9.5-1 (cont'd)

Trip Sensitivity:	±1 microampere
Output Signal:	"1" state, -0.5 to 0 volts, 100 ma max. "0" state, -8 to -10 volts
Ambient Temperature:	-18°C to + 55°C
Response Time:	1 millisecond max.

9.5.1.2 Logic Unit. The logic unit is a bistable (0 or 1 state) decision-making element consisting of a diode gating structure coupled with transistor switching. The unit has 8 input gates with 4 inputs per gate. The input-output logic relations of these logic units as wired in this system are tabulated below as follows:

Input			Output 2 of 3	Output 1 of 3
0	0	0	0	0
1	0	0	0	1
0	1	0	0	1
0	0	1	0	1
1	1	0	1	1
1	0	1	1	1
0	1	1	1	1
1	1	1	1	1

With the scram-logic selector switch, located on the linear-level calibration panel, the operator can select either "1 of 3" or "2 of 3" logic. With "1 of 3" logic, a "1" from any of the 07-39, 07-42, or 07-45 input units on the power level channels the logic-unit output will be a "1". With "2 of 3" logic for "1's" from any two of the above input units, the logic-unit output will be a "1." The direction manual for the logic unit is Leeds and Northrup No. DB 2711.

The logic circuit consists of eight OR gates each having four inputs operating on a single AND gate. The switch senses the state of the AND gate and provides the same state at the T output at a power level sufficient to drive other devices in a system. Certain applications of the logic unit will require the state opposite to that of the AND gate. An inverter operates from the switch to provide a phase inverted level at $\bar{T}$ output at a power level equivalent to that of the switch. Both the switch and the inverter therefore are power amplifiers. When one or more input signals to each OR gate are in the "1" state simultaneously, all eight input signals to the AND gate are in the 1 state, thus providing the 1 state at the T output and the 0 state at the $\bar{T}$ output. These conditions must be fulfilled whenever the unit is to trip.

The specifications for the logic unit are given in Table 9.5-2.

Table 9.5-2 Logic Unit Specifications

Power Requirements	- -10 v DC ±5%, 60 ma (untripped), 130 ma (tripped); +10 v DC +5%, 10 ma.
Gate Input Load	- 2.2 ma max. (each gate).

Table 9.5-2 (cont'd)

Gate Input Signal	- -0.5 to 0.0 v (short circuit); -8.0 to -10.0 v (open circuit).
Memory, Reset, Inhibit Signals	- -0.5 to 0.0 v (short circuit); -8.0 to -10.0 v (open circuit).
Memory, Reset, Inhibit Load	- Ind. Reset, T Reset, $\bar{T}$ Inhibit, $\bar{T}$ Memory 12 ma; T Memory 18 ma; $\bar{T}$ Reset, $\bar{T}$ Inhibit 10 ma; Ind. Memory 7 ma.
Logic Unit Identification	- Yellow handle and light on front edge of printed circuit card.
Ambient Temperature Limits	- -18°to 55°C (0 to -131°F).
Logic Unit Connector	- 25 dual-contact (50 contacts total) LN part no. 040282, Precision connector No. 093-25.

9.5.1.3 NOD Logic. To perform the logical functions necessary for operation of relays connected to the relay safety circuit the NOD (Negative OR Diode) logic is used. Each NOD circuit consists of two input diode gates which feed a transistor switch circuit. With either or both inputs in the "1" state (0 volts) both diodes conduct which turns off the transistor switch and the output current becomes 0 thus no current flows through the relay and it is de-energized.

The characteristic equation of this logic circuit is:

$$T = \bar{A} \times \bar{B}$$

where T (output) = logical "1" state and A, B (inputs) = logical "0" state. The specifications for this unit are given in Table 9.5-3.

Table 9.5-3 NOD Circuit Specifications

Manufacturer:	Leeds & Northrup
Model No.:	444067
Unit No.:	07-48, 07-49, 07-50, 07-51, 07-52
Service:	Coincidence Logic, Safety Monitor Logic
Ambient temperature:	-18°C to +55°C
Input Signal:	"1" state, -0.5 to 0 volts
	"0" state, -8 to -10 volts
Output Signal:	"1" state, -0.5 to 0 volts, 80 ma max. (Each NOR)
	"0" state, -8 to -10 volts
Power:	±10 volt DC (±5%)

9.5.1.4 Output Unit. The output unit provides the clutch power for the NBSR shim safety rods. Upon receipt of a scram signal from the logic units the clutch power is cutoff and the shim rods are rapidly inserted into the core.

The output unit consists of two independent transistor switches feeding an OR circuit which controls the current input to an inverter. The inverter supplies a 2 Kc square wave to a transformer which steps up the voltage to that required by the shim rod clutches. The 2 Kc square wave voltage is rectified by a full wave rectifier and is supplied to the clutches.

A separate winding on the inverter transformer supplies an indicator light which lights when the unit is tripped and also supplies a signal to the safety monitor. The indicator is reset by the safety monitor which grounds the indicator reset circuit.

The specifications for the output unit are given in Table 9.5-4.

Table 9.5-4 Output Unit Specifications

Manufacturer:	Leeds & Northrup
Model No.:	444030
Unit No.:	07-53, 07-54, 07-55, 07-56
Service:	Clutch Power Supply
Ambient temperature	-18°C to +55°C
Input Signal:	"1" state -0.5 to 0 volts "0" state -8 to -10 volts
Output Signal:	28 volts DC at 700 ma
Power:	+ 10 volts DC (± 5%)

9.5.2 RELAY SAFETY SYSTEM. The process instruments that produce reactor scram and the nuclear backup scrams are connected in series in two redundant scram circuits so that a single contact failing to open will not prevent a scram from occurring in the other redundant string. The relays are used in the "fail safe" mode; i.e., the relay is de-energized upon receipt of a scram signal. Thus the relay contacts will open on loss of power or the opening of the relay coil. The relay contacts are employed at approximately 10% of their rated capacity and all the relay coils utilize voltage suppressors (Varistors) connected across their coils to limit the voltage peak when coil circuit is opened. See Figure 9.11 for the scram string arrangement and Section 9.4.1

The relays employed for the NBSR safety system are mercury wetted contact relays which are high speed sealed relays with the following specifications:

Manufacturer:	C. P. Clare Co.
Model No.:	HGX-1022
Normal Operating Voltage:	48 volts DC
Normal Operating Power:	3.5w
Release Time:	5.5 milliseconds
Contacts:	4 Form C
Contact Rating	5 amp @ 500 volts

The scram relay coils K-2001 receive power through the K-103 contacts and the redundant process instrument scram relays as depicted in Figure 9.11.

The relay backup nuclear instrument scrams are connected in series with the scram relay K-103. The relay K07-14 contacts open on receipt of a scram signal in the "A" or "B" scram bus, (see Section 9.5.2.1 for system operation).

The period relay backup scram is also connected in series with the K-103 scram relay, (see Section 9.5.2.2 for system operation).

The nuclear instrument power monitor consists of relays K07-19 and K07-20 which will open upon loss of nuclear instrument power as discussed in Section 9.3.1.8.

The relay that interrupts the clutch power are the relays K-2000 and K-2001. These are heavy duty power relays manufactured by Struthers Dunn Type 8DXXH3. The contacts of these relays are connected in series as shown on Figure 9.11 to interrupt directly the 10 volt power to the inverter. Thus the failure of one relay to open or the welding of one contact will not prevent the de-energizing of the clutches. The specifications for these relays are as follows:

Manufacturer:	Struthers Dunn
Model No.:	8 DXXH3
Normal Operating Voltage:	48 volts DC
Normal Operating Power	0.5 amp
Release Time	Less than 30 milliseconds
Contacts	4P. ST
Contact Rating:	30 amp @ 24 volts DC max. voltage 600 volts

9.5.2.1 Power Range Safety System. The power range scram function is provided directly by bistable trip circuits with no intermediate amplifiers. The measuring function of this channel is discussed in Section 9.3.2.

The power range scram function is performed by the bistable trips 07-39, 07-42, and 07-45. These bistables have two set points 13% and 130% scram. The set points are selected by the switch S74, mounted on the set point panel which selects the 13% set point potentiometer or the 130% set point potentiometer. A contact of this switch permits the selection of Annunciators, 4-2, or 4-3. Lights on the console indicate to the operator the position of this switch.

The output of the bistable trips 07-39, 07-42, and 07-45 feed the logic units 07-62 and 07-63. The logic function, 1 of 3 or 2 of 3, may be selected by a key operated switch mounted in the nuclear instrumentation panel. With "1 of 3" logic, a "1" or 0 volts from any one of the bistables 07-39, 07-42,or 07-45 will cause the output of the logic units to be "1" or 0 volts. With the 2 of 3 logic any two of the bistable trips with "1" outputs or 0 volts will produce a "1" output or 0 volts out of the logic unit.

The logic unit outputs are connected to two separate busses "A" and "B". A "1" or 0 volts output of either logic unit will cutoff current to the shim rod clutches since the output units require both inputs to be "0" or -10 volts to produce an output to the clutches. If either of the inputs goes to "1" or 0 volts the output is turned off. The operation of the output unit is discussed in Section 9.5.1.4.

The outputs of the output units are connected to the shim clutches through current monitoring relays and a rheostat which permits adjustment of the current through the clutches to the minimum value to drive the shim rods. The output of the output units is connected through isolation diodes which will permit one output unit to operate two clutches if one output unit fails, (see Figure 9.11). Thus it may be seen that the failure of any one output unit to shutoff current to the clutches will not prevent the de-energization of the other two clutches.

The current monitoring meter relays have a high and low contact which connect to the K07-100 relay. Contacts of this relay are connected in the startup permit circuit and to Annunciator 4-35. Thus if an output fails to supply the proper current a startup prohibit will occur and an alarm.

An additional scram path is provided via NOD logic curcuits 10 and 11 monitoring the "A" and "B" scram bus. This is a 1 of 2 logic so that if either bus goes to "L" or 0 volts the relay K07-14 will be de-energized. A set of normally open contacts of this relay de-energize the relay K103 in the non-nuclear safety system which de-energizes the K2000 and K2001 relays whose contacts open the -10 volt supply to the output units. Thus a backup scram is provided for an unsafe failure of the output units.

The overall response time of the power range scram system for a one decade step change input does not exceed 50 milliseconds, including the shim rod clutch release time. The overall response time of the input unit and output unit is better than 200 microseconds.

9.5.2.2 Period Scram. The intermediate range channels furnish a signal to bistable trip units 07-27 and 07-28 which provide period scram safety. The bistable trip outputs are connected directly into the "A" and "B" scram bases. A separate output is connected to the NOD 8 and 9 circuits on module card 07-64 to provide relay scram backup similar to that employed in the power range channels, (see Figure 9.11).

The outputs of bistable trip 07-27 feeds "B" bus and the output of bistable trio 07-28 feeds "A" bus. A scram output "1" from either of these units will produce a "1" output of the output units to de-energize the shim rod clutches.

In addition a backup is provided by supplying an inverted output "0" on a trip signal to the NOD circuits 8 or 9. The output of these NOD circuits will be an inverted "0" or "1" which will provide a "0" output to NOD circuit 7 and de-energize the relay K07-06. A normally open set of contacts of this relay de-energize the K103 relay which interrupts the power to the output units.

The period bistable trips are inhibited by a signal from the three power range channels when the reactor power is above 10% of full power. All 3 channels are required to be above 10% to inhibit the period safety functions. When the bistable trips 07-37, 07-40, and 07-43 are in a tripped condition or "0" output, the "0" inputs into NOD circuit 1 will produce a "1" output which in turn will inhibit the bistables 07-21, 07-22, 07-23, 07-24, 07-25, 07-26 from tripping. To prevent reactor startup with a period inhibit condition a contact of the relay K07-10 is incorporated in the startup permit circuit. The relay coil is supplied from NOD circuit 2 on unit 07-65 which has a "0" output for the "1" input from NOD 1. A contact of relay K07-10 also supplies an alarm that the period trips have been inhibited, Annunciator 4-37.

The response time of the intermediate range channel for one decade step change input is as follows:

	Time Constant		Period Trip
	Min*	Max*	
10^{-10} to 10^{-9} amps	40	200	6 sec
10^{-7} to 10^{-6}	16	80	6 sec
10^{-4} to 10^{-3}	12	60	6 sec

* Time in milliseconds with 195 pfd input capacity.

9.5.2.3 Testing

9.5.2.3.1 Automatic Testing of Scram System. The system to be tested consists of three power level scram bistable trips 07-39, 07-42, and 07-45 whose outputs are combined into a selected 1 of 2 or 2 of 3 logic in doubly-redundant logic units 07-62 and 07-63. The output of each bistable trip connects to both logic units. Two period-scram bistable trips, 07-27 and 07-28, each connect to one of the reactor safety buses. Each logic unit also connects to one of the above two buses. Each output unit 07-53, 07-54, 07-55, 07-56 receives an input from each bus but in a 1 of 2 logic and can be turned off by a trip signal on either or both safety buses.

Test procedure: for the following sections, each component in the scram safety system will be assigned a letter as follows:

Component	Letter Destination
Level Bistable Trip 07-39	A
Level Bistable Trip 07-42	B
Level Bistable Trip 07-45	C
Period Bistable Trip 07-27	D
Period Bistable Trip 07-28	E
Logic Unit 07-62	F
Logic Unit 07-63	G
Output Unit 07-53	H
Output Unit 07-54	J
Output Unit 07-55	K
Output Unit 07-56	L

Pulse Program I: The automatic testing routine will insert all combinations of three pulses into power range trip units A, B, C and the system response will be read at the outputs at the logic units F, G.

Pulse Program II: The automatic testing routine will insert all combinations of two pulses (third pulse not used) into the period bistable trips D, E and the system response will be read at the outputs of the output units H, J, K, L.

Circuit Operation: The binary counter 07-60 sends signals to the pulser pacer 07-59 which inhibit the generation of pulses in all combinations, (see Figure 9.11) A "1" from the binary counter inhibits pulse generation. Conversely, a "0" from the binary counter permits pulse generation. The binary counter may be advanced automatically or manually. The automatic advance is accomplished by connecting the gate #2 output to pulse generator #5. The binary-advance pulse output from pulse generator #5 connects to the binary counter through the closed contact of "trouble" relay K07-16 (closed when no trouble) and the closed contact of the auto-manual selector switch (closed when switch is in "auto" position), allowing the binary counter to advance automatically. Manual advance is accomplished by switching the auto-manual selector switch to "man" which opens the path to the manual advance pushbutton, which then can advance the binary counter.

The pulser pacer acts as the automatic testing system clock. The output from gate #1 initiates pulses from the pulse generators which are permitted or inhibited by the binary counter. The output from gate #2 through pulse generator #5 resets the indicating lights on the tested components. The operation of gates G_1 and G_2 steps the system through the automatic testing routine.

The pulses are routed to the level scram and period scram bistable trips by NOD circuit card 07-48, switchgate. The state of the fourth digit of the binary counter determines the selection of the pulse program I or II and whether the pulses go to the level scram or the period scram bistable trip. NOD circuits 11, 12, 13 and 14 will bypass the period scram bistable trip in the event of a period-trip block signal (which inhibits bistable trip) such that the output units are tested directly.

The first three digits of the binary counter both permit and inhibit pulse operation as described above and indicate to the monitor readout NOD circuits 07-49, 07-51, and 07-52 the particular combination of test pulses which have been inserted for a given test. The logic configuration on the NOD circuits on the above cards is determined by a "trouble" equation which defines all the possible configurations of component malfunctions that can occur in the system.

When there is no trouble in the system (all components functioning correctly for a given test) the state of NOD circuit 11 output on 07-50 is a "1" (ground) energizing relay K07-15. When trouble occurs, the state of NOD circuit 11 output is a "0" (-10 volts DC), de-energizing relay K07-15, closing a contact in series with the coil of relay K07-16, and opening a contact to Annunciator 4-27. The relay K07-16 is energized, opening one contact in the automatic advance circuit from the pulser pacer to the binary counter, which prohibits any further advance. A second K07-16 contact closes in the "trouble" light circuit which lights the trouble light. When a component malfunction is detected the same combination of alarms will continue to flash until the defective unit has been replaced. A trouble alarm will prohibit reactor startup by de-energizing K500 which in turn de-energizes K104 relay. (See Section 9.5.3.3).

The scram logic selector switch determines whether 1 of 3 or 2 of 3 logic is being tested automatically, dependent on the operator's logic selection.

Monitor Readout, Program I: The correct response to program I is for both F and G to trip whenever two or more pulses are inserted into A, B and C if 2 of 3 logic is selected. When 1 of 3 logic is selected, both F and G should trip when one or more pulses are inserted and neither F or G should trip when no pulses are inserted.

Program II: The correct response to program II is for H and J, and K and L to trip when one or more pulses are inserted into D and E and neither H nor J, nor K nor L to trip when no pulses are inserted into D and E.

Monitor Panel Lights: Lights P1, P2, and P3 each light when a pulse is inserted. When light I is on, program I is being tested. When light II is on program II is being tested.

Special Components: (Pacer Pulser). The pacer pulser is essentially a timing signal source with controllable multi-channel output. Its output is 200 microsecond pulses which can be either "1" or "0" depending upon connection. These pulses are sufficient to momentarily trip a given component being tested without hindering its normal operation.

Binary Counter. The binary counter generates binary numbers up to and including six digits. Either phase of a binary digit ("0" or "1") is available to use as inputs to the automatic testing routine. The binary counter thus generates all combinations of "0's" and "1's" up to and including six digits. These outputs then can be used to test for all configurations of faults. The binary is advanced by grounding pin 13.

9.5.2.3.2 Manual Testing of Safety System. In addition to the ability to introduce test signals into the inputs of the nuclear instruments during operation, provision has been made to manually test the bistable trips, logic circuits, relays, annunciators, etc. which are not a part of the scram system. The philosophy of manual testing is the same as that of the automatic testing excepting the set point is reduced to signal level by a test switch instead of a pulse generator. The testing is divided into two portions, that of testing (1) startup prohibit and withdraw prohibit, and (2) rundown and comparator.

The input units 07-30, 07-19, 07-21, 07-23, 07-25, 07-37, and 07-46 are connected to a test bus. When the push button #1 is pressed a +10 volts is applied to this bus which trips these bistables and operates the Annunciators 4-17, 4-25, 4-26, and 4-36.

The pressing of push button #2 will trip bistables 07-20, 07-22, 07-24, 07-26, 07-29, 07-40, 07-47 and the same annunciators as above will operate.

A selector switch and push button #3 enable the testing of the rundown trips and the comparators which may be tested in the different combinations as noted in the Table on Figure 9.11.

9.5.3 SHIM SAFETY ARM CONTROL RELAY LOGIC. The philosophy for the control of the NBSR is shown in Figure 9.12. The diagram shows the various conditions that have to be satisfied to obtain rod withdrawal. The logic used is a negative logic since the relays are connected in a fail safe manner. The end operation; i.e., energizing a relay is termed an inhibit or a permit. The term inhibit is utilized with the opposite logical condition being the protective action; i.e., scram, rundown, etc. The power for the relays is obtained from the same power supplies as the process scram safety system. The term permit is used with prohibit being the opposite relay state (de-energized).

The design parameters for shim safety arm control are as follows:

(a) The shim safety arms may be individually operated or as a group of four by use of G.E. SBM type selector switches, S10, S20, S30, and S40.

(b) It is possible to position a shim safety arm while the regulating rod is on automatic. The automatic rod must be overridden by a safety action.

(c) The regulating rod may be operated manually by a SBM type of swtich.

(d) The scram interlocks have been made redundant to provide a scram path even if one string should fail. The scram conditions will directly actuate the K103 circuit and at the same time actuate redundant contacts in the K2000 relay coil circuit.

(e) The rundown of all shim safety arms occurs upon a scram condition which will positively drive the shim safety arm in. Thus this condition serves as a scram backup in case a shim safety arm clutch fails to disengage.

(f) The withdraw permit condition prevents shim safety arm withdrawal unless safe operating conditions exist.

(g) The startup permit requires that certain conditions must be fulfilled. After a

startup permit is obtained only an alarm will be provided if the prescribed conditions are not fulfilled.

9.5.3.1 Experiment Interlocks. The experiment interlock string is provided to provide reactor scram from safety circuits incorporated into individual experiments if required by the NBSR safety committee. The relays K201 thru K213 have coils which may be de-energized by protective instrumentation in the experiments. These relays have contacts connected to the annunciator system, Annunciator 1-10 thru 1-24, which will provide the operator information to which experiment scrammed the reactor. The experiment interlock relay has contacts which are wired into the scram circuit.

9.5.3.2 Manual Scram Circuit. The manual scram circuit is connected to normally closed contacts of push buttons mounted around the walls on the operating floor and one in the D_2O pump room. The "Major Scram" and "Reactor Scram" switches are located on the console. The manual scram relays K101 will be energized through a set of N.O. contacts. A momentary depression of the scram reset push button will energize the relay and it will hold in through N.O. contacts. Thus a momentary depression of any one of the scram buttons will de-energize this relay and require manual reset by the reactor operator.

This relay, K101, has two sets of redundant N.O. contacts connected in the K103 and K2000 circuits which energize the shim safety arm clutches. (See Section 9.4.1 for discussion of these relay functions.) Another set of N.O. contacts of the relay K101 operate Annunciator 4-1 which indicates a manual scram has occurred.

9.5.3.3 Startup Permit. A startup permit will be obtained if the following conditions have been satisfied:

(a) Control power on - The control power on switch is a key operated switch mounted on the control console.

(b) All shim safety arms seated - All shim safety arms and the regulating rod must be seated. This is a contact of relay K105 whose coil is energized by the shim safety arm switch (through auxiliary relays). A "seal in" circuit energizes the relay K105.

(c) Emergency cooling tank level - The level in the emergency cooling tank must be normal to achieve a startup permit.

(d) Primary coolant overflow FIA-2 - The primary coolant flow in the reactor overflow pipe must be normal.

(e) NC-1 and NC-2 Count rate > 10 cps - This interlock, K07-01, prevents reactor startup unless both count rate channels NC-1 and NC-2 are counting at least 10 cps which indicates that high voltage is present and those channels are counting fission neutrons.

(f) Period not bypassed - This interlock prohibits reactor startup if a period inhibit signal is present from the nuclear safety system.

(g) No automatic and calibration monitoring trouble - If any of the calibrate-operate switches is not in the operate position relay K500 will be de-energized and prohibit reactor startup and produce an alarm, Annunciator 4-27.

Also connected in series with the same relay coil is a contact of the automatic monitoring relay K07-15. If trouble is detected this contact opens and prohibits reactor startup. See Section 9.5.2.3.1 on automatic testing.

(h) Clutch current normal - The shim safety arm clutches are monitored for a high or low current condition by indicating meter relays. If the clutch current is too high or too low the relay K07-100 will be de-energized and open contacts which will prohibit reactor startup and produce an alarm, Annunciator 4-27.

(i) Experimental interlock - This is a contact of the relay logic discussed in Section 9.5.3.1.

(j) Process and nuclear backup scram - All scram conditions must be satisfied, i.e., inhibited.

If all the above conditions are satisfied, relay K104 may be energizer by momentarily pressing the "Scram Reset" push button. When relay K104 is energized a set of normally open contacts close and "seal in" the relay. Another set of contacts of this relay bypass the conditions (a) to (i) after a startup permit is obtained. However, if a scram occurs the conditions above must again be satisfied to obtain a reactor startup permit.

9.5.3.4 Rundown Inhibit. The next set of conditions that must be satisfied after the scrams are cleared and a startup permit has been obtained is a rundown inhibit. If at any time anyone of the conditions is not satisfied the shim safety arms will rundown until the condition is cleared.

The following conditions have to be satisfied to inhibit shim safety arm rundown:

(a) NC-3 and NC-4 period > 10 seconds - The reactor period as measured by the intermediate range channels NC-3 and NC-4 must be greater than 10 sec if this interlock is to be satisfied or the reactor power level must be greater than 10% of reactor full power.

The interlock relay K07-05 from the nuclear safety system opens when the reactor period is less than 10 sec. This relay is energized from NOD-6 (see Figure 9.11). The bistable trips 07-25 and 07-26 provide a logical "0" to the NOD circuit if a safe condition is indicated. When the reactor power level exceeds 10% of full power as indicated by channels NC-6, NC-7, and NC-8 the bistable trips 07-37, 07-40, and 07-43 will provide a logical "0" safe signal to the NOD circuit 1 on module 07-65 and a logical "1" will inhibit the tripping of the period safety system bistables 07-25 and 07-26.

(b) Reactor power < 115% - The reactor power level as measured by the power range channels must be less than 115% of full power in 2 of 3 channels to satisfy this interlock. This interlock function is supplied by relay K07-11 in the nuclear safety system. The power to energize this relay is obtained from the NOD circuit 6, unit 07-65, (see Figure 9.11). A logical "0" signal from 2 of 3 of the bistables 07-38, 07-41, and 07-44 must be present at the input of the NOD circuit to energize the K07-11 relay.

(c) Thermal column D_2O level normal - This channel function is described in Section 9.4.2.2. If the conditions are normal this condition will be satisfied and the contacts will be closed.

(d) Primary coolant level normal - This channel function is described in Section 9.4.2.3. If the level signal is above set point this contact will be closed.

(e) Thermal shield flow normal - This channel function is described in Section 9.4.2.4. If the flow signal is above the set point this contact will be closed.

(f) Moderator cump inhibit - If the conditions causing a moderator dump are not satisfied then the dump relay K109 is energized and this contact will be closed.

(g) All control rods not seated - All shim safety arms and the regulating rod must be seated. This is another contact of the relay K105 which has a contact in the startup permit circuit.

(h) Withdraw permit - If all the conditions for a withdraw permit have been satisfied this contact of K104 closes and satisfies the interlock.

(i) Shim safety arms gang switch "off" or "withdraw" - This condition is satisfied as long as the manual shim safety arm gang withdraw and rundown switch is in a condition other than rundown. Normally closed contacts in the "off" and "withdraw" position of this switch S9 provide this function.

(j) Reactor thermal power - This channel function BTUR-1 is described in Section 9.4.2.5. If the thermal power recorder indicates less than 11.5 Mw then this contact will be closed.

If all the aforementioned conditions are satisfied then the relay K107 will be energized and a rundown of all rods will be inhibited. Whenever one of these conditions are not satisfied the shim safety arms will be inserted until the condition is rectified.

9.5.3.5 Withdraw Permit. To achieve a withdraw permit it is necessary to satisfy the following conditions:

(a) NC-3 and NC-4 period > 15 sec - The reactor period as indicated by the intermediate range channels NC-3 and NC-4 must be greater than 15 sec or the reactor power level must be greater than 15% of full reactor power.

The relay K07-04 operated by the NOD circuit 3 on card 07-64 in the nuclear safety system is energized when the intermediate range period bistable trips 07-23 and 07-24 provide a logical "0" to the NOD circuit if a safe condition is indicated. When the reactor power level exceeds 10% of full power as indicated by the power range channels NC-6, NC-7, and NC-8 the bistable trips 07-37, 07-40, and 07-43 provide a logical "0" to the NOD circuit 1 on card 07-65 which provides a logical "1" to inhibit the trip of the bistable trips 07-23 and 07-24. The relay contacts also operate Annunciator 4-26 when the period signal is less than set point.

(b) NC-1 and NC-2 period > 20 sec - The reactor period as measured by the source range channels NC-1 and NC-2 must be greater than 20 sec or the reactor power level must be greater than 10% of full power.

The relay K07-02 operated by the NOD circuit 2 on unit 07-64 in the nuclear safety system is energized when the source range period bistable trips 07-21 and 07-22 provide a logical "0" to the NOD circuit if a safe condition is indicated. When the reactor power level exceeds 10% of full power, as indicated by power range channels NC-6, NC-7, and NC-8, the bistable trip units 07-37, 07-40, and 07-43 will provide a logical "0" to the NOD 1 on unit 07-65 which provides a logical "1" to inhibit the trip of the bistable trip units 07-21 and 07-22.

(c) Shim safety arms seated - All shim safety arms must initially be seated to satisfy this condition closing this contact of K105.

(d) Shim safety arm rundown - This condition has to be satisfied, i.e., K107 energized, thus no rundown condition is present.

If the withdraw permit condition is satisfied the shim safety arms or the regulating rod may be withdrawn. If one of the required conditions is not satisfied rod withdrawal will be stopped until the condition clears.

9.5.3.6 Automatic Control. To operate the reactor on automatic control it is necessary to satisfy the automatic control permit conditions which are as follows:

(a) The manual-auto transfer switch is a spring loaded to center (G.E. SBM type of switch). If all of the following conditions are satisfied the switch, when momentarily thrown to the auto position, will energize relay K50 which will close a set of back contacts and will remain energized through a set of normally closed contacts on the manual-auto transfer switch.

Reactor control will revert to manual whenever one of the following conditions occur:

(a) Manual operation of regulating rod - this function is provided by a set of normally closed contacts of S50, the regulating rod manual control switch. The operation of this switch opens the N.C. contacts and de-energizes relay K50.

(b) Regulating rod within operating limits - If the regulating rod exceeds the travel limits, the limit switches will de-energize relays K53 and K54. These relays have normally closed contacts which open if the limit switches operate, de-energize the relay K50, the reactor reverts to manual control and provides an alarm, Annunciator 4-45.

(c) Servo deviation exceeds 10% - This function is provided by contacts of K07-13 relay in the nuclear instrumentation rack. This relay coil is energized by NOD circuit 8 on unit 07-65 which receives inputs from the bistable trip units 07-29 and 07-30. If the deviation exceeds 10% of the set point, which indicates the controller is unable to control the reactor, the NOD circuit receives a logical "1" input which de-energizes this relay and opens the contacts which de-energizes the automatic permit relay K50 and returns the reactor to manual control with an alarm. (See Figure 9.8)

(d) Rod rundown inhibit - If a rod rundown condition occurs through the de-energizing of K107 then a contact of K107 opens and de-energizes the automatic permit relay K50.

(e) Withdraw permit - If a rod withdraw prohibit condition occurs, contacts of K108 will open and de-energize relay K50.

9.3.5.7 Regulating Rod. The regulating rod may be operated manually or automatically. The manual operation is by a G. E. SBM type of switch (S50). The switch is spring return to the "off" position and the operation of this rod on manual is the same as the shim safety arms.

9.5.3.8 Rods Seated. The shim safety arms and the regulating rod, when seated, i.e., the insert limit switches are actuated, will de-energize a relay K-14, K-24, K-34, K-44 or K-54. A set or normally open contacts of each of these relays is wired in series with the relay K-105 as illustrated in Figure 9.12. This circuit prevents reactor startup unless all rods are initially seated by contacts connected in the rundown inhibit circuit K-107, withdraw permit K-108, and the startup permit circuit K-105. Another set of contacts of the rod seated relay K-105 operate the Annunciator 4-34.

9.5.3.9 Manual Control of the Shim Safety Arms. The shim safety arms may be manually operated by individual switches S-10, S-20, S-30, or S-40 or the gang switch S-9. The switches are G. E. SBM type of switches with spring return to the "off" position. The switches operate auxiliary relays to energize the appropriate winding of the shim safety arm drive motor.

The shim safety arm drive motors are two phase motors as described in Section 4.3. A change in direction of rotation of the drive motors is obtained by reversing the phases applied to the motor windings. The two phase power for the drive motors is obtained from a Scott T Transformer which converts the incoming three phase voltage to required two phase voltage. The use of the two phase voltage provides a positive instantaneous phase reversal of the drive motors.

The shim safety arms are protected from exceeding their mechanical travel limits by two redundant limit switches. These switches are wired in series with a control relay. The actuation of either switch will de-energize this relay. The use of relays here reduces the arcing of the limit switch contacts.

A shim safety arm may be withdrawn when the following conditions are satisfied. At any time these conditions are not satisfied the shim safety arm motors will be de-energized. At all times a rundown signal will override the withdraw condition and insert all the shim safety arms.

(a) Individual shim safety arm switch in the withdraw position or the gang switch in the withdraw position.

(b) The withdraw permit relay, K-108, energized. The energizing of this relay indicates that all the conditions of Section 9.5.3.5 have been satisfied.

(c) Shim safety arms may be withdrawn until the withdraw limit switches on the respective arm is actuated. The limit switch will interrupt the current to the appropriate motor phase and stop the motor.

The shim safety arms may be inserted without satisfying any conditions until the insert limit switches are actuated. Thus the manual insert of the shim rods may be accomplished by actuation of the switches S-10, S-20, S-30, or S-50 or by the gang switch, S-9. Whenever a rundown condition occurs the contacts of K-107, the rundown relay, will close and insert the rods until the condition is cleared.

9.5.4 MAJOR SCRAM, MODERATOR DUMP AND VOID SCRAM BUS. This bus is powered from the 125v DC batteries described in Section 7.3.1. The 125 volt DC power is supplied to this bus through the DC power panel mounted on the control room wall. (See Figure 9.7).

9.5.4.1 Moderator Dump. The operator may choose to dump the moderator manually or have the moderator dumped automatically when a reactor scram occurs. The moderator dump will continue until the void and dump reset push button, S75, is actuated.

The conditions necessary to inhibit the dump are as follows:

(a) If the moderator dump switch is in automatic then the startup permit must be satisfied (a contact of relay K104). If in the moderator dump switch is in the manual position the automatic dump feature on scram is by-passed.

(b) The helium void switch HIC-67 must be off.

(c) The air void switch HIC-68 must be off.

If the above conditions are satisfied then pressing the "void and dump" reset switch S75 will momentarily energize the relay K109 which will close a set of N.O. contacts and seal in. Thus an inhibit of the dump will be obtained by the closure of a set of contacts which will energize the solenoid pilot valve, SV117, on dump valve DWV-9. Other contacts are connected to Annunciator 4-20, which will produce an alarm if K109 is de-energized. A set of contacts also connect to the rundown inhibit circuit K107 (see Section 9.5.3.4).

9.5.4.2 Helium Void Shutdown. The helium void shutdown system (see Section 7.1.2) is manually actuated with no automatic features. This switch, when actuated, de-energizes a solenoid pilot valve which opens the valve HEV-1 and admits a flow of helium from helium storage flasks to the bubbler pipes. When this switch is actuated the reactor is scrammed and a D_2O moderator dump is initiated.

The helium void shutdown inhibit will be satisfied if the following conditions are satisfied:

(a) Helium void switch "off." The helium void switch HIC-67 is a G.E. SB-1 type of switch with two position and spring return to the off position. Normally closed contacts this switch are connected in the scram relay circuit K2000, the moderator dump circuit K109, the helium void relay circuit K112, and the helium void valve solenoid circuit SV-167.

(b) When the "void and dump" reset switch is depressed a set of N.O. contacts close and energize the K112 relay which closes a set of N.O. contacts of this relay to maintain the relay. Contacts of this relay are connected, N.O., in series with N.C. contacts of the switch HIC-67 which energize the solenoid pilot valve SV-167 for the helium void valve HEV-1.

9.5.4.3 Air Void Shutdown. The philosophy of operation of this system is the same as the operation of the helium void system excepting that air is admitted to the bubbler pipes via valve HEV-8 from the compressed air system (see Section 7.1.2).

The air void inhibit will be satisfied if the following conditions are satisfied:

(a) The "air void" switch is a SB-1 switch with the same development as the helium void switch HIC-67. This switch has normally closed contacts in the scram relay circuit K2000, the moderator dump circuit K109, the air void relay circuit K113, and the air void solenoid valve circuit SV-168.

(b) When the "void and dump" reset switch is depressed a set of N.O. contacts close and energize the K113 relay which is maintained through a set of N.O. contacts. A set of N.O. contacts are connected in series with a set of N.C. contscts of the switch HIC-68 which energizes the solenoid pilot valve SV-168 for the air void valve HEV-8.

9.5.4.4 Major Scram. A major scram seals the reactor confinement building by closing all personnel doors and drains, shutting off domestic services, and stopping ventilation fans. A major scram will occur whenever a high radiation level exists in the ventilation system or may be manually initiated by the operator at the console.

The main scram bus is powered from the 125 volt DC station batteries. The major scram relay, K114, is the same type of relay that is used elsewhere in the reactor safety system excepting with a 125 volt DC coil. This relay is de-energized by the operation of the "major scram" switch, S3, a spring loaded N.C. switch, or opening of contacts of the irradiated air monitor RRC-3-4 or normal air monitor RRC-3-5. This relay when de-energized requires resetting by the scram reset push button S76. Contacts of the K114 rela , N.O., de-energize the DSR and FSR relays located in motor control center MCCA-3 upon receipt of a scram signal.

The DSR relays have contacts, N.O., that de-energize the door operator circuits and isolation valves. The following doors and isolation valves are operated by the DSR relay:

a. Elevator doors
b. South door
c. North door
d. Sump #1 isolation valve RWV-13
e. Hot waste isolation valve RWV-14
f. Stack drain valve RWV-15
g. Hot waste sump pump #4 isolation valve RWV-16
h. Domestic cold water valve RUV-1
i. Low pressure natural gas valve RUV-2
j. Domestic hot water valve RUV-3

The FSR relays have N.O. contacts that de-energize the motor starters for the following fans (see Section 3.6).

a. Second floor supply fan SF-1
b. Supply fan SF-2
c. First floor supply fan SF-3
d. Basement supply fan SF-11
e. Lab and pool supply fan SF-12
f. Normal exhaust fan EF-3
g. Irradiated air exhaust fan EF-4
h. Radiological lab #1 exhaust fan EF-23
i. Radiological lab #2 exhaust fan EF-24
j. Reactor basement exhaust fan EF-27

The FSR relay has N.C. contacts which operate the following emergency fans when the major scram relay is de-energized to exhaust the reactor building (see Section 3.6).

a. Emergency exhaust fan EF-5
b. Emergency exhaust fan EF-6
c. Decontamination recirculation fan SF-19

9.5.4.5 Evacuation alarm. The evacuation alarm is manually operated by a push button mounted on the console. This switch de-energizes a relay K115 which has contacts connected to an auxiliary relay which operates the evacuation alarms. The evacuation alarms are placed in strategic locations throughout the building. The alarm will continue until manually reset by the operator via the void and dump reset switch, S75.

The evacuation alarm switch, HIC-84, is a mushroom head push button switch with momentary contacts. The relay K115 is the same type of relay as employed elsewhere in the safety system. A set of normally open contacts of K115 operates an auxiliary 2P 20 ampere relay which actuates the evacuation alarm.

The evacuation alarms are Federal Sign and Signal Corp. 120 volt DC sirens rated at 1/2 HP. The evacuation alarms operate directly from the 125 volt DC station battery.

9.6 GENERAL CONSIDERATIONS

9.6.1 EMERGENCY POWER SOURCES. As stated in Section 7.3.1 and shown schematically in Figure 3.12 there are two emergency power motor control centers A5 and B6 which normally supply power to critical or backup systems for the reactor. The two centers are normally tied together through a normally closed manually operated bus tie circuit breaker, while

power is fed to the bus through either of two breakers #1 or #4. Breaker #1 controls the power fed to the emergency power bus from motor control center A3 while breaker #4 can supply power to the bus from motor control center B4. Under normal operating conditions only one of the breakers #1 and #4 is closed. The voltage of the emergency power bus is monitored by an under voltage relay which controls the breakers #1 and #4 automatically. If the bus voltage should drop, a number of actions are initiated. First, breaker #1 opens and breaker #4 closes. If the voltage to the bus is not restored after an adjustable time delay by the closing of breaker #4 a signal is transmitted to start one of two standby diesel generators, either of which can feed the emergency power bus through breakers #2 and #3. In the meantime a separate "inverter-diverter" battery powered motor generator set can feed the Critical Power and Console Power Panel. The diesel generators and inverter-diverter systems are described in the following subsections.

9.6.1.1 Inverter-Diverter Motor Generator Set. Upon failure of the normal source of power to the Emergency Power Motor Control Centers (MCC A5 and B6) an undervoltage relay operates to open the line contactor in the MCC A5 to disconnect and isolate the Critical Power Panel from its normal AC power system, and to immediately feed power without interruption of voltage or frequency to the panels from the Station Battery, the equipment running as a DC to AC inverter. This undervoltage control is independent of the undervoltage control for breakers #1 and #4 and operates without delay.

The motor generator set consists of a 18.75 Kw, AC synchronous generator, 3 phase, 60 cycles, 480 volts with a direct connected exciter, flexibly coupled to and mounted on a common bedplate with a 25 housepower DC motor 125/105 volt shunt wound, constant speed, diverter pole type. The set normally operates as an AC synchronous motor driving a 25 Kw DC generator at 1200 rpm to develop 125 volt power to feed the DC distribution board.

The Station Battery is of the lead-acid type rated for 125 volts DC, with an output of at least 830 ampere-hours at an 8 hour discharge rate.

The phase of the output voltage of the motor generator set is automatically synchronized with power on the emergency power bus when the line contactor is operated. When voltage is restored to the emergency bus from either the diesel generators or from normal breaker closure the voltage phase is again automatically synchronized before the inverter diverts back to an AC motor DC generator condition.

9.6.1.2 Diesel Generators. The two diesel generators A and B each consist of a six cylinder engine with direct starting. The engines are four cycle valve-in-head design, with replaceable sleeves, heavy duty crankshaft, twin intake valves and twin exhaust valves. Conventional starting equipment is supplied consisting of a switch operated starting motor with drive geared to the flywheel.

The rotating member of the generator is directly connected to the engine flywheel by means of a steel disc. The generator is a single bearing brushless synchronous generator. The generator armature windings supplying AC power are on the stator of the main machine, and the DC field windings on the rotor. The DC field power is supplied by rotating rectifiers mounted on one of the fans. The rectifiers are supplied from the armature windings of the AC exciter which are on the rotor. The exciter field windings are stationary. Exciter field power is obtained from the diesel generator output which is rectified and controlled by an automatic regulator.

The previous design details contribute to the overall reliability of the emergency power system while allowing a minimum of moving parts and routine maintenance.

One of the diesels is selected by a control switch to automatically start cranking if voltage is not restored to the emergency power bus by the closure of breaker #4. At the same time the second diesel is put into standby condition to automatically crank if the first diesel fails to start after four attempts or if the first diesel should fail after coming up to speed for any reason. The closure of breakers #2 and #3 is automatically controlled by the respective voltage output of each diesel A and B to supply power to the emergency power bus.

9.6.2 AUTOMATIC CONTROL SIMULATION. Prior to reactor startup a series of studies of the dynamic characteristics of the automatic control channel is planned. A reactor simulator has been constructed using DC operational amplifiers in an analog computer program to permit closed loop operation of all components of the reactor control system without including the reactor itself. It is expected from these studies that the range of operational parameters for the three mode automatic controller can be set before an attempt is made to bring a critical reactor configuration under control.

The servo permit features of the control circuitry can also be tested prior to reactor startup.

The system will use the electrical output signal of the simulator as an input to the micro-microammeter of the automatic control channel. The automatic controller will compare this signal to a set point voltage and drive the servo motor of the regulating rod to minimize an error signal. The regulating rod servo motor will drive a potentiometer which will be chosen to represent the reactivity change introduced by the regulating rod under the equivalent servo motor rotation. The voltage from the potentiometer simulating reactivity will be programmed in the analog computer to induce the corresponding reactor flux change in the real reactor configuration, thus completing the control loop.

9.7 RADIATION MONITORING SYSTEMS

9.7.1 REACTOR BUILDING MONITORS. Figure 9.13 shows the reactor building monitoring system which consists of ten γ sensitive area monitors and three β-γ detectors adjacent to high efficiency filters in air exhaust ducts.

9.7.1.1 Area Monitors. The area monitors are in the following locations:

RD-1-1	First floor, north wall
RD-1-2	First floor, east wall
RD-1-3	First floor, south wall
RD-1-4	First floor, west wall
RD-1-5	Second floor, ceiling above reactor
RD-1-6	Second floor, west wall
RD-1-7	Fuel storage pool area, south wall
RD-1-8	Process room, east wall
RD-1-9	Process room, west wall
RD-1-10	Reactor control room

The detectors are Tracerlab type TA-6A, using halogen-quenched G-M tubes, and each detector assembly is equipped with an alarm light, an alarm buzzer, and a solenoid operated check source. Detectors 1 through 4, 6, 7, and 10 have local dose rate indicators with ranges of 0.01 mr/hr to 10 R/hr. Detector 5 has the same range, but no local dose rate indicator. Detectors 8 and 9 have ranges of 0.1 mr/hr to 100 R/hr and have remote lights, buzzers and dose rate indicators located in the storage pool area. Dose rates at detectors 1 and 8 are also indicated on follow meters located on the emergency ventilation control panel.

Individual Tracerlab type TA-3 meter relay stations in the control room indicate the dose rates at the detectors. When the measured dose rates exceed levels preset on each indicator, visual and audible alarms at the detector, control room, and health physics office are energized. The check sources, actuated individually from the control room, provide a means for checking the operation of each area monitor.

RD-1-11 Irradiated air exhaust from the reactor.

RD-1-12 Normal air exhaust (unirradiated room air taken from the face of the reactor).

RD-1-13 Basement air exhaust.

The activity in these three ducts is indicated on Tracerlab type MM-6B log ratemeters with independently set alarm points. The MM-6B ratemeter has a range of 20 cpm to 200,000 cpm and time constants that vary from 60 sec at 20 cpm to 0.05 sec at 200,000 cpm. Visible

and audible alarms in the control room and health physic office are energized by the activity exceeding the levels preset on the MM-6B meter relays.

9.7.2 FILTER RADIATION MONITORS. (Figure 9.14). Tracerlab type MD-12B thin wall β-γ G.M. detectors are adjacent to the high effeciency filters in ductwork serving the following areas:

RD-2-1	Hot lab #1	Room B-154
RD-2-2	Hot lab #2	Room B-153
RD-2-3	Hot lab #3	Room B-152
RD-2-4	Warm lab #1	Room B-147
RD-2-5	Warm lab #2	Room B-145
RD-2-6	Warm lab #3	Room B-143
RD-2-7	Warm lab #4	Room B-144
RD-2-8	Warm lab #5	Room B-142
RD-2-9	Warm lab #6	Room B-140
RD-2-10	Radiological lab #1	Room C-001
RD-2-11	Radiological lab #2	Room C-002
RD-2-16	Semi-warm lab #1	Room B-119
RD-2-17	Semi-warm lab #2	Room B-120
RD-2-18	Semi-warm lab #3	Room B-121
RD-2-19	Semi-warm lab #4	Room B-122
RD-2-20	Semi-warm lab #5	Room B-123
RD-2-21	Semi-warm lab #6	Room B-124
RD-2-22	Warm shop	Room B-141
RD-2-23	Service area, filter F52	Room B-142
RD-2-24	Service area, filter F50	Room B-148
RD-2-26	Hot cell assembly shop	Room B-146

Radiation levels at the filters are indicated on individual Tracerlab type MM-6B log ratemeters located in the Health Physics lab. These levels are also recorded in the same lab. Count rates exceeding levels preset on the MM-6B meter relays actuate visible and audible alarms in the Health Physics lab and also in the lab that the detector is serving.

9.7.3 REACTOR PLANT RADIATION MONITORS. (Figure 9.15). The reactor plant radiation monitoring system consists of shielded detectors monitoring the secondary coolant, helium sweep gas, normal air exhaust and irradiated air exhaust.

9.7.3.1 Secondary Coolant Monitor, RD-3-1. The secondary coolant downstream from the main D_2O heat exchanger HE-1 is monitored for radioactivity to indicate a heat exchanger tube failure. The monitor consists of a Tracerlab type MD-12B detector inside a Tracerlab MW-4 shielded liquid sampler located in the secondary coolant pump room. The output of this detector is indicated on a Tracerlab RM-20B log ratemeter in the monitor room and is recorded in the control room. The RM-20B ratemeter has a range of 10 cpm to 10^6 cpm and time constants that vary from 72 sec at 10 cpm to 0.012 sec at 10^6 cpm. Activity exceeding a level preset on the RM-20B meter relay actuates an annunciator in the control room. A concentration of 8×10^{-6} µCi/ml of Sr^{90} - Y^{90} activity will produce a net signal of 40 cpm above a background of 40 cpm in this monitor.

9.7.3.2 Gaseous Fission Product Monitor, RD-3-2. The helium sweep gas is continuously monitored for radioactivity to provide an indication of a fuel element cladding failure. A sample of the helium is pumped through a Tracerlab type MG-2 shielded gas sampler with a Tracerlab MD-12B detector inside. A concentration of 7.6×10^{-7} µCi/Cm^3 of Kr^{85} will produce a net signal of 40 cpm above a background signal of 40 cpm in this monitor. A Tracerlab type RM-20B log ratemeter, located with the detector in the monitor room indicates the radioactivity of the gas. This level is then recorded in the control room. Activity exceeding a level preset on the ratemeter activates a control room annunciator.

9.7.3.3 Gas Monitors, RD-3-4 and RD-3-5. This system is comprised of a gas pump; two Tracerlab type MG-2 shielded gas samplers, containing Tracerlab type MD-12B detectors; and two Tracerlab type RM-20B log ratemeters in a single housing on the reactor basement mezzanine. A concentration of 5.5×10^{-7} µCi/CC of A^{41} activity will produce a net signal of 40 cpm above a background signal of 40 cpm in these monitors. Gas samples are continuously pumped

from the irradiated air and normal air exhaust ducts. When the activity in either duct exceeds a preset value, a major scram is initiated and audible as well as visual alarms in the control room are energized.

9.7.4 PLANT EFFLUENT MONITORS. (Figure 9.16).

9.7.4.1 Stack Gas Monitor, RD-4-1. The stack gas monitor consists of four paralleled Tracerlab MD-12B thin wall β sensitive G.M. detectors located inside the stack about two-thirds of the way up it. The radiation level is indicated on a Tracerlab RM-20B log ratemeter in the monitor room and is recorded in the control room. Activity exceeding a preset level will actuate the control room and Health Physics annunciators.

9.7.4.2 Liquid Effluent Monitor, RD-4-3. The liquid effluent monitor RD-4-3 is a thin walled β sensitive Tracerlab MD-12B G.M. detector located in the hot waste vault as described in Section 7.4. The radiation level is indicated on a Tracerlab RM-20B log ratemeter in the hot waste control area of the cold laboratory basement. Activity exceeding a preset level will actuate the Health Physics annunciator and operate the control valves of the hot waste system shown in Figure 7.19.

9.7.5 HEALTH PHYSICS INSTRUMENTATION

9.7.5.1 Portable Insturments. Portable instrumentation will include six G.M. type β-γ sensitive survey meters with ranges of 0-0.5-5-50 mR/hr, four ionization type β-γ survey meters with ranges of 0-50-500-5000 mR/hr and 0-250-2500-25,000 mR/hr, three ionization type survey instruments with ranges of 0-0.5-5-50-500 R/hr, two thermal and fast neutron survey meters with ranges of $0\text{-}10\text{-}10^2\text{-}10^3\text{-}10^4$ n/cm^{2}sec, two air-proportional type γ survey meters with ranges of $0\text{-}10^3\text{-}10^4\text{-}10^5\text{-}10^6$ cpm, and a dose-rate sensitive neutron survey meter with ranges of 0-1-10-100-1000 mrem/hr. A semiportable, γ-compensated, ionization chamber type tritium monitor and semi-portable high volume air samplers are also included. The tritium monitor has ranges of $0\text{-}10^2\text{-}10^3\text{-}10^4\text{-}10^5$ μCi/m^3 of tritium concentration.

9.7.5.2 Fixed Instruments. The fixed instrumentation include two γ-β sensitive gas-flow proportional counters for counting smears and filters from the air samplers, a β-γ sensitive hand and foot monitor with a frisking probe for monitoring clothing, and four β-γ sensitive foot monitors with frisking probes. These five instruments are used to monitor personnel leaving potentially contaminated areas, such as the confinement structure; hot cell area; and hot and warm labs.

Ten Savannah River Plant type dosimeters are located throughout the containment structure and are to be evaluated in the event of a criticality accident. The dosimeters have bare In, and Cd covered In and Cu foils; sulfur powder; and a γ sensitive chemical dosimeter.

A 400 channel analyzer with a 3 x 3 NaI crystal is available for health physics use.

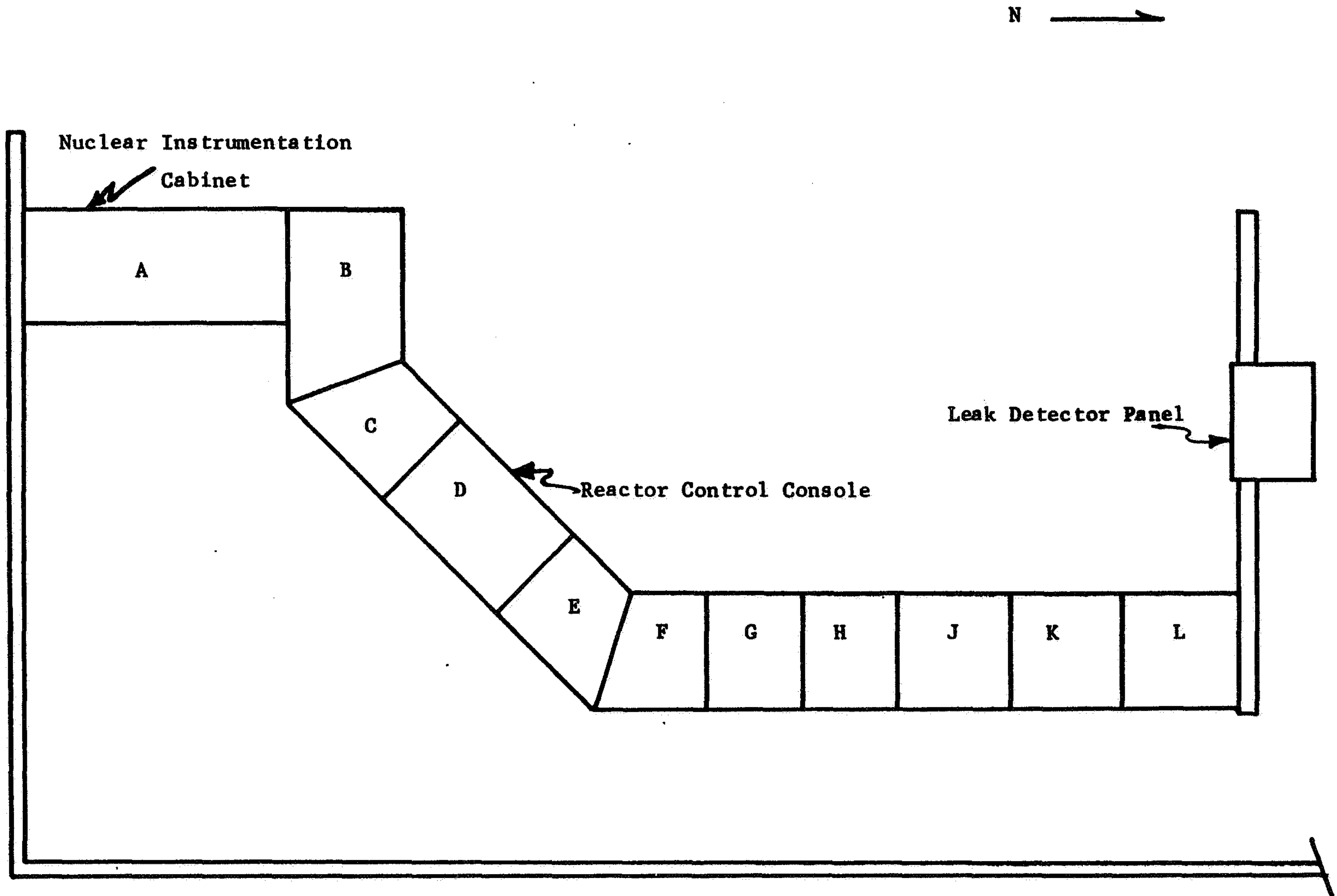


Figure 9.1 Control room panel location.

Beam Port 1A Open	Beam Port 1B Open	Beam Port 1C Open	Beam Port 1D Open	Beam Port 2A Open	Beam Port 2B Open
Beam Port 2C Open	Beam Port 3A Open	Beam Port 4A Open	Grazing Tube 2G Open	Grazing Tube 4G Open	
Cryogenic A Shutter Open	Cryogenic B/C Shutter Open	Rabbit Blower On	Process Room Door Open	Sub-Pile Room Door Open	
Experiment #1	Experiment #2	Experiment #3	Experiment #4	Experiment #5	Experiment #6
Demin.Exp. Cooling Hi Outlet Temp.	Demin.Exp. Cooling Lox IX Flow	Demin. Exp. Cooling Low Flow	Demin. Storage Tnk Abnormal Level	Demin. IX Outlet Hi Cond.	Demin. Exp. Cooling Abnormal Press.
Sec. Exp. Cooling Pump #1 Off	Sec. Exp. Cooling Pump #2 Off	Rabbit In Lower West 14" R	Rabbit In Upper West 21" S	Rabbit In Lower East 23" T	Rabbit In Upper East 34" U

Figure 9.2 Annunciator panel number 1.

Reserve Tank Low Flow	Plenum Low Flow	Sump Pit #4 Hi Level	Sump Pit #4 Med. Level	Sump Pit #4 Lo Level	Void Shutdown No Flow	Void He Bottles Low Pressure	Bubbler Inlet Low Pressure
D_2O Leak 1-5	D_2O Leak 6-10	D_2O Leak 11-15	D_2O Leak 16-20	D_2O Leak 21-25	D_2O Leak 26-30	D_2O Leak 31-35	D_2O Leak 36-40
D_2O Leak 41-45	D_2O Leak 46-50	D_2O Leak 51-55	D_2O Leak 56-60	D_2O Leak 61-65	D_2O Leak 66-70	D_2O Leak 71-75	Thermal Column Tank Leak
Liquid Waste High Rad.	Stack Monitor Hi Rad.	Area Monitor Hi Rad.	He Sweep Gas Hi Rad.	Sec. Cool. Hi Rad.	Hi Tritium Activity	H. P. Office Ann.	Plant Ann.

Hi Reactor D_2O Level	Lo Reactor D_2O Level	Reactor Inlet Hi Temp.	Reactor Inlet Lo Temp.	IX Inlet Hi Temp.	Reactor IX Low Flow	IX Inlet Hi Cond.	IX Outlet Hi Cond.
HX. #2 Inlet Low Flow	Thermal Column Lo Flow	Thermal Column Hi Temp.	D_2O Storage Tank Low Level	D_2O Storage Tank Hi Level	O_2 Makeup Low Press.		
Recombiner Outlet Abnormal Press.	Recombiner Inlet Hi Press.	He System Low Flow	He Outlet Hi Temp.	Recombiner Internal Lo Temp.	He Blower Off Valve Open	He Gas Holder Low Level	He Sweep Makeup Low Press.
CO_2 Gas Holder Lo Level	CO_2 Press. Relief Valve Open	CO_2 Blower Off Valve Open	CO_2 Makeup Low Pressure	Cont. Board Normal Pwr. Failure	Mg Set Trouble	DC Ground	

Figure 9.3 Annunciator panel number 2 and 3.

Manual Scram	Hi Flux Low Set Point	Hi Flux NC 6,7OR8	Period Scram	Nuclear Inst. Power Off	Bldg. Exh. Hi Activity	Irradiated Air Hi Activity	Bubbler Inlet Gas Leak
Low Flow Outlet	Low Flow 14" Inlet	Low Flow 10" Inlet	Exp. Interlock	Hi Diff. Temp.	Low Reactor D_2O Level		
Period Rundown	Hi Flux Rundown	Hi Reactor Outlet Temp.	Moderator Dump	Low Reactor D_2O Level	Thermal Column Low D_2O Level	Hi Thermal Power	Thermal Shield Low Flow
Source Period Rod Stop	Period Rod Stop		Reactor On Manual	Servo > 10%	Reg. Rod Control Limit	Reg. Rod Jammed	
Start-Up Chamber Not In	Rods Not Seated	Clutch Power	Low Count Rate	Period Bypassed	Exp. Interlocks	Rod Test On	Low Level Emergency Tank
Low D_2O Overflow		NC 6, 7, 8 Pwr. Dev. $> \pm 10\%$					

Figure 9.4 Annunciator panel number 4.

Cooling Tower Low Temp.	Cooling Tower Low Level	Cooling Tower Hi Level	Sec. Cooling Low Flow	48 Volt Relay Power Failure	Start-Up Chamber Not Withdrawn	Press.Diff. Sec. D_2O	
Thermal Shield Hi Inlet Temp.	HX-6 Hi Inlet Temp.	Floor Outlet Header Hi Temp.	Ring Outlet Header Hi Temp.	Thermal Shield Low Flow	Thermal Shield IX Low Flow	Thermal Shield Storage Tk.Ab.Level	Thermal Shield Hi Cond.
Storage Pool Hi Cond.	Storage Pool HX 8 Hi Temp.	Storage Pool Low Flow	Storage Pool IX Low Flow	Storage Pool Hi Level	Storage Pool Lo Level	Pump Pit Hi Level	Pump Pit Lo Level
Cryogenic Shield Cool. Hi Temp.	Cryogenic Shield Cool. Low Flow	D_2O Exp. Cool. Low	Diesel A Failure To Start	Diesel B Failure To Start	150# Air Comp. Trouble	10# Air Comp. Trouble	Refueling Helium Low Pressure

Scram	Run Down All Shims	Withdraw Prohibit

Figure 9.5 Annunciator panel number 5 and 6.

System Wiring Trouble				Transformer TA Hi Temp.
Transformer TB Hi Temp.	Transformer TC Hi Temp.	Bus "A" Network Prot. Open	Bus "B" Network Prot. Open	Bus "C" Network Prot. Open
Bus "A" Tie Bkr. Open	Bus "B" Tie Bkr. Open	Bus "C" Tie Bkr. Open	Emer. Des. Gen. "A" Fail To St.	Emer. Des. Gen. "B" Fail To St.
				Steam Low Pressure
Chilled Wtr. Low Pressure	Comp. Air Low Pressure	Gas Low Pressure	Dom. Wtr. Low Pressure	
				DC Failure

Power

Super. System | On Light | Super. System

Current | Ann. | Ann. | Potential

Test | Power Switch | Ack.

System Wiring Trouble	DC Failure	Bus "A" Network Prot. Open	Bus "B" Network Prot. Open	Bus "C" Network Prot. Open
Transformer TA Hi Temp.	Transformer TB Hi Temp.	Transformer TC Hi Temp.		
Bus "A" Tie Bkr. Open	Bus "B" Tie Bkr. Open	Bus "C" Tie Bkr. Open		
Chilled Wtr Low Pressure	Comp. Air Low Pressure	Incom. Gas Low Pressure	Dom. Wtr. Low Pressure	Steam Low Pressure

Power

Super. System | On Light | Super System

Current | Ann. | Ann. | Potential

Test | Power Switch | Ack.

Figure 9.6 Annunciator panel at Diesel generators and building services annunciator panel.

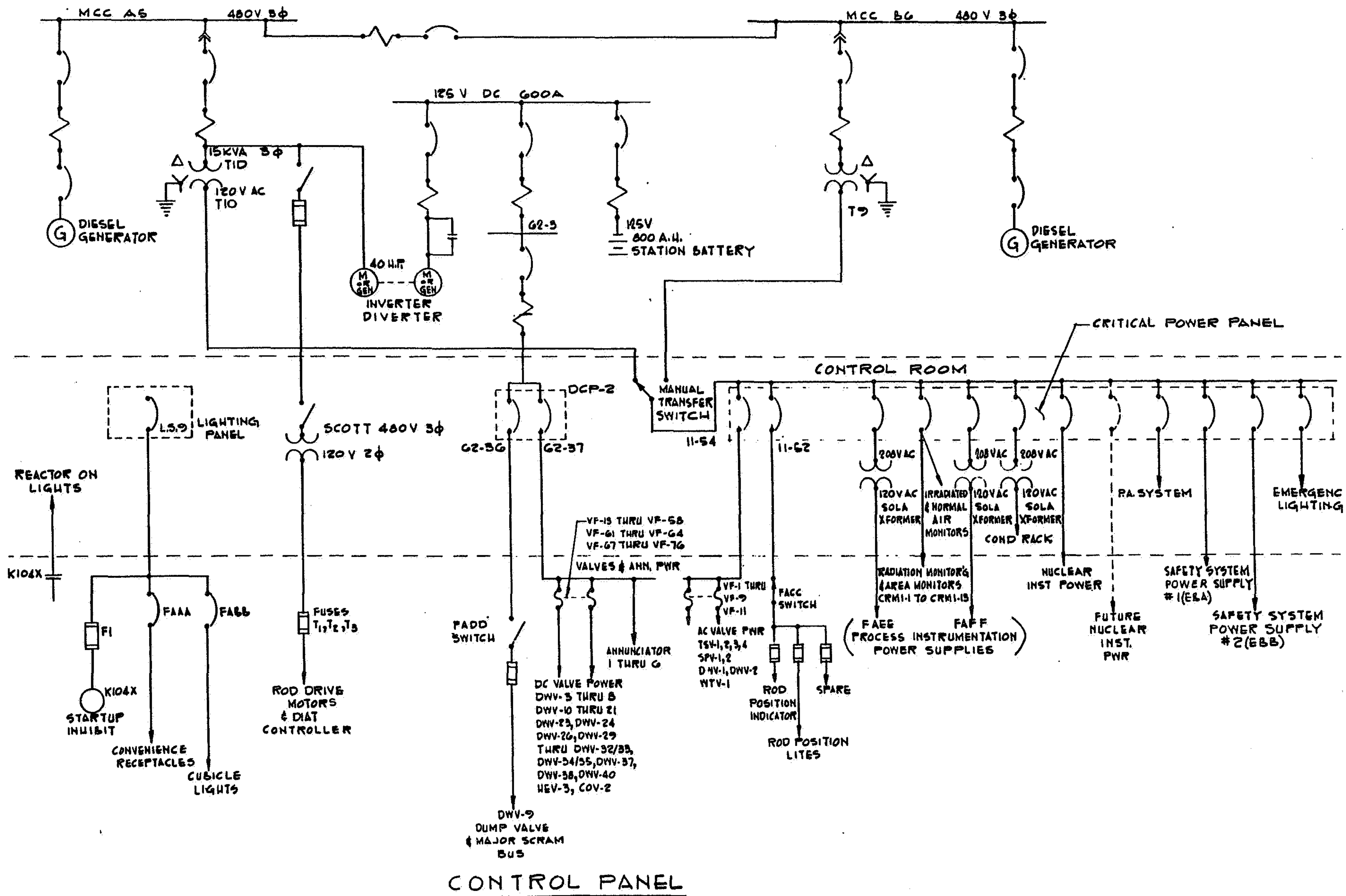


Figure 9.7 Single line instrument power.

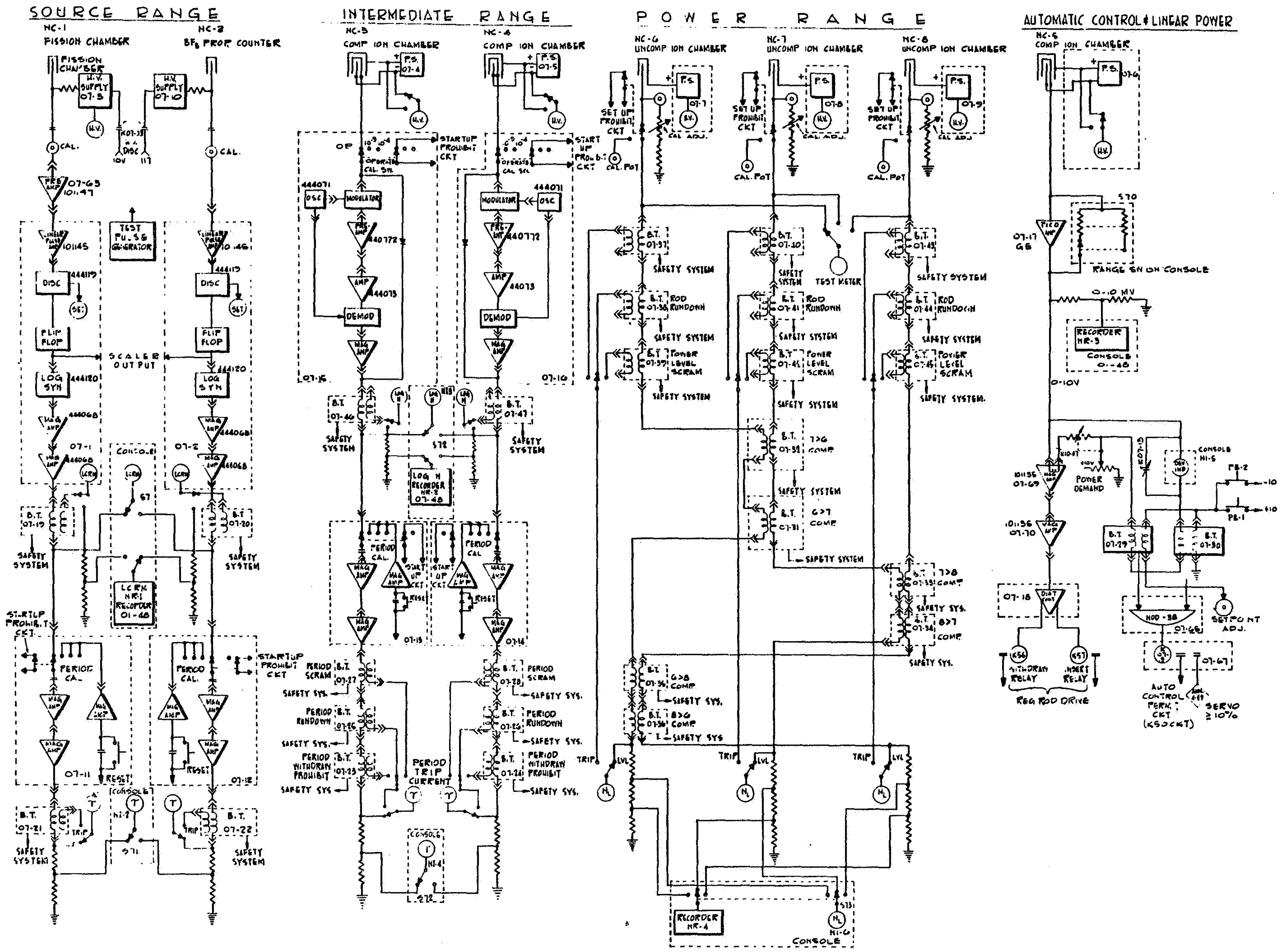


Figure 9.8 Functional block diagram of the nuclear instrumentation.

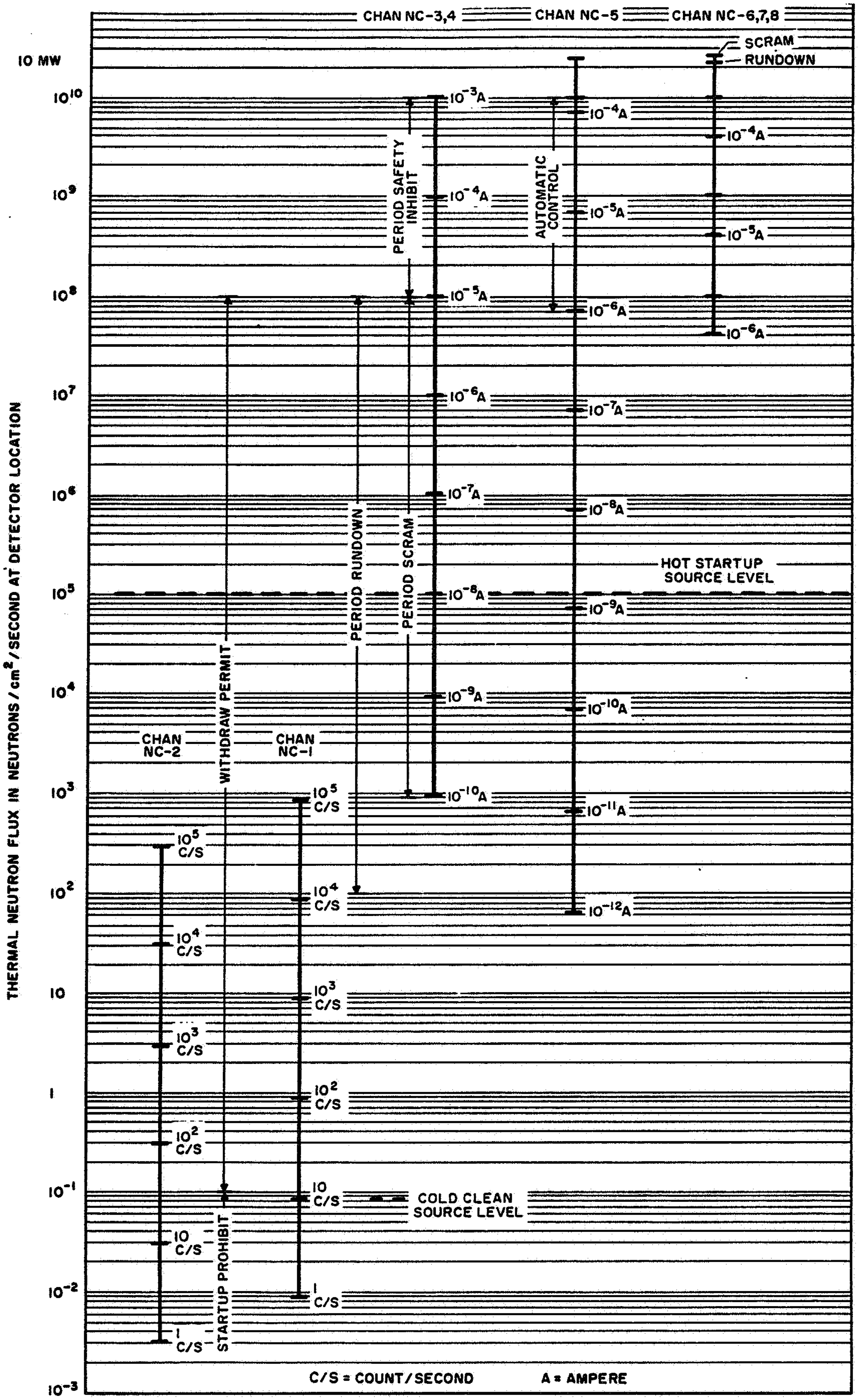


Figure 9.9 Flux coverage of NBSR.

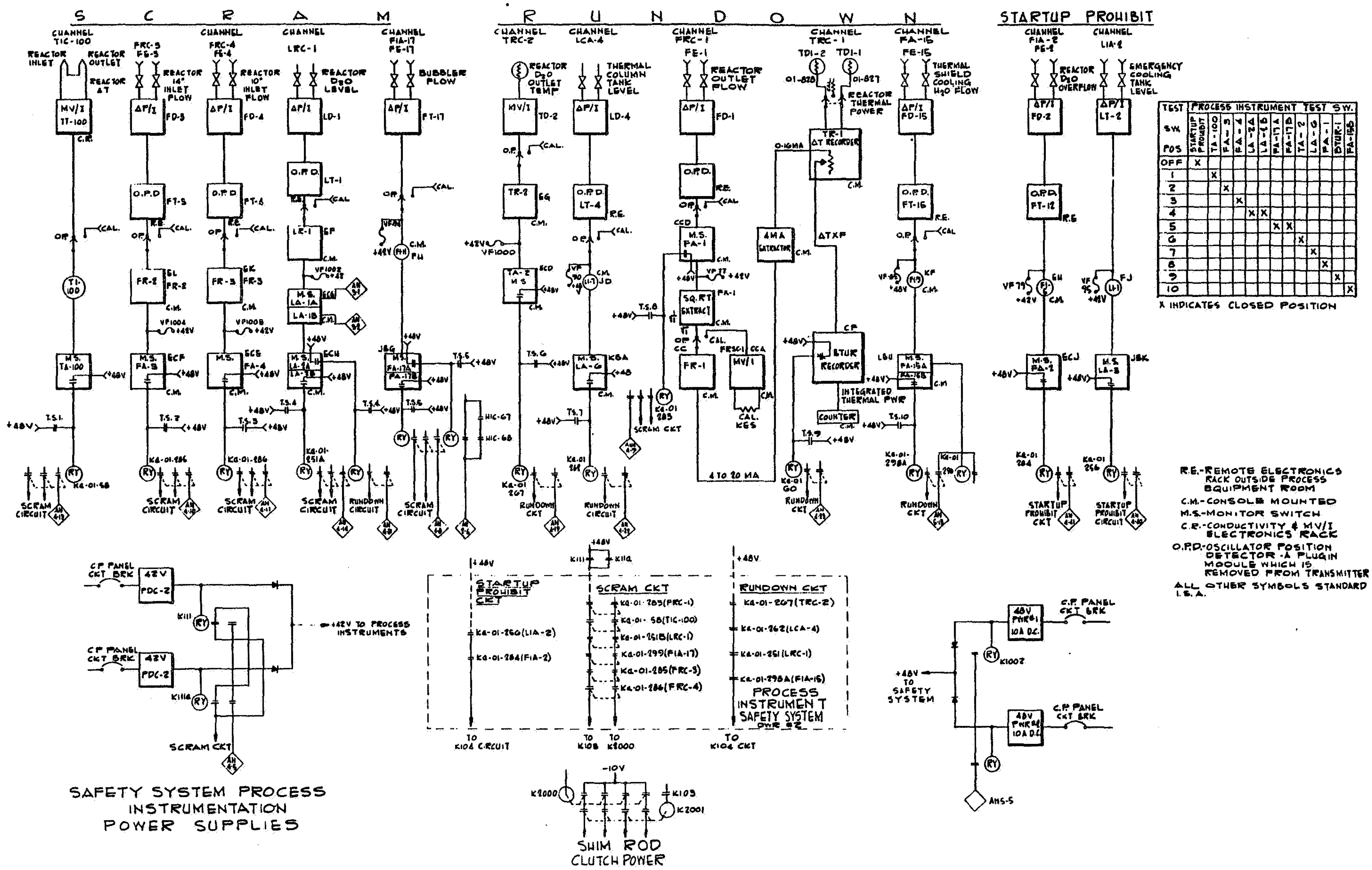
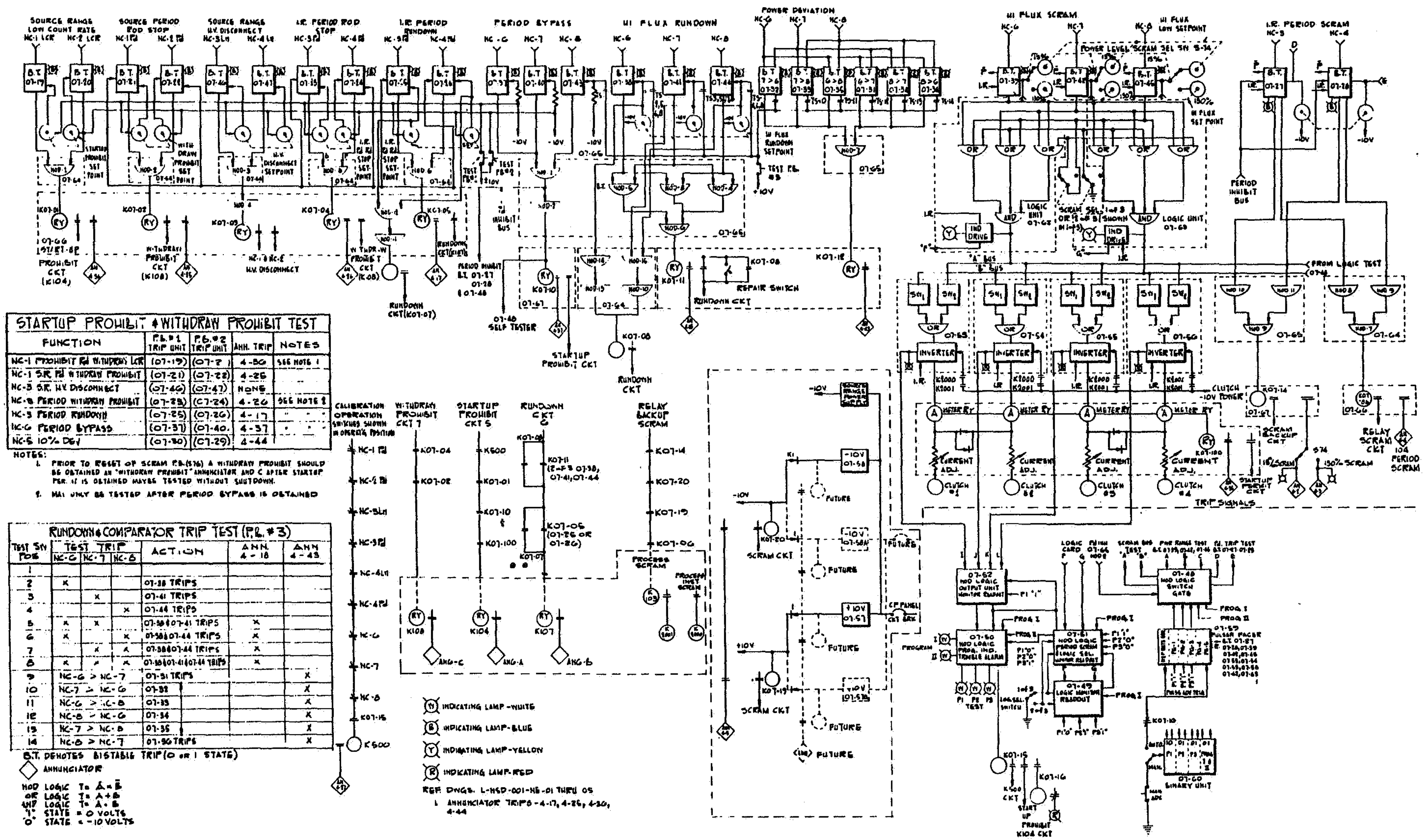


Figure 9.10 Functional block diagram of the process safety system.



STARTUP PROHIBIT & WITHDRAW PROHIBIT TEST				
FUNCTION	P.B. #1 TRIP UNIT	P.B. #2 TRIP UNIT	ANN. TRIP	NOTES
NC-1 PROHIBIT Rd WITHDRAW LCR	(07-19)	(07-2)	4-36	SEE NOTE 1
NC-1 S.R. Rd WITHDRAW PROHIBIT	(07-21)	(07-22)	4-25	"
NC-3 S.R. H.V. DISCONNECT	(07-46)	(07-47)	NONE	
NC-3 PERIOD WITHDRAW PROHIBIT	(07-23)	(07-24)	4-26	SEE NOTE 2
NC-3 PERIOD RUNDOWN	(07-25)	(07-26)	4-17	" "
NC-6 PERIOD BYPASS	(07-37)	(07-40)	4-37	" "
NC-5 10% DEV	(07-30)	(07-29)	4-44	

NOTES:

1. PRIOR TO RESET OF SCRAM P.B.(S76) A WITHDRAW PROHIBIT SHOULD BE OBTAINED AN "WITHDRAW PROHIBIT" ANNUNCIATOR AND C AFTER STARTUP PER. IF IS OBTAINED MAYBE TESTED WITHOUT SHUTDOWN.
2. MAY ONLY BE TESTED AFTER PERIOD BYPASS IS OBTAINED

RUNDOWN & COMPARATOR TRIP TEST (P.B. #3)						
TEST SW POS	TEST TRIP NC-6	NC-7	NC-8	ACTION	ANN. 4-18	ANN 4-43
1						
2	X			07-38 TRIPS		
3		X		07-41 TRIPS		
4			X	07-44 TRIPS		
5	X	X		07-38 & 07-41 TRIPS	X	
6	X		X	07-38 & 07-44 TRIPS	X	
7		X	X	07-38 & 07-44 TRIPS	X	
8	X	X	X	07-38 & 07-41 & 07-44 TRIPS	X	
9	NC-6 > NC-7			07-31 TRIPS		X
10	NC-7 > NC-6			07-32		X
11	NC-6 > NC-8			07-33		X
12	NC-8 > NC-6			07-34		X
13	NC-7 > NC-8			07-35		X
14	NC-8 > NC-7			07-36 TRIPS		X

B.T. DENOTES BISTABLE TRIP (0 or 1 STATE)

ANNUNCIATOR

NOR LOGIC T = $\bar{A} \cdot \bar{B}$
OR LOGIC T = A + B
AND LOGIC T = A · B
"1" STATE = 0 VOLTS
"0" STATE = -10 VOLTS

REF. DWGS. L-HSD-001-HS-01 THRU 05
1. ANNUNCIATOR TRIPS - 4-17, 4-26, 4-36, 4-44

Figure 9.11 Safety system logic block diagram.

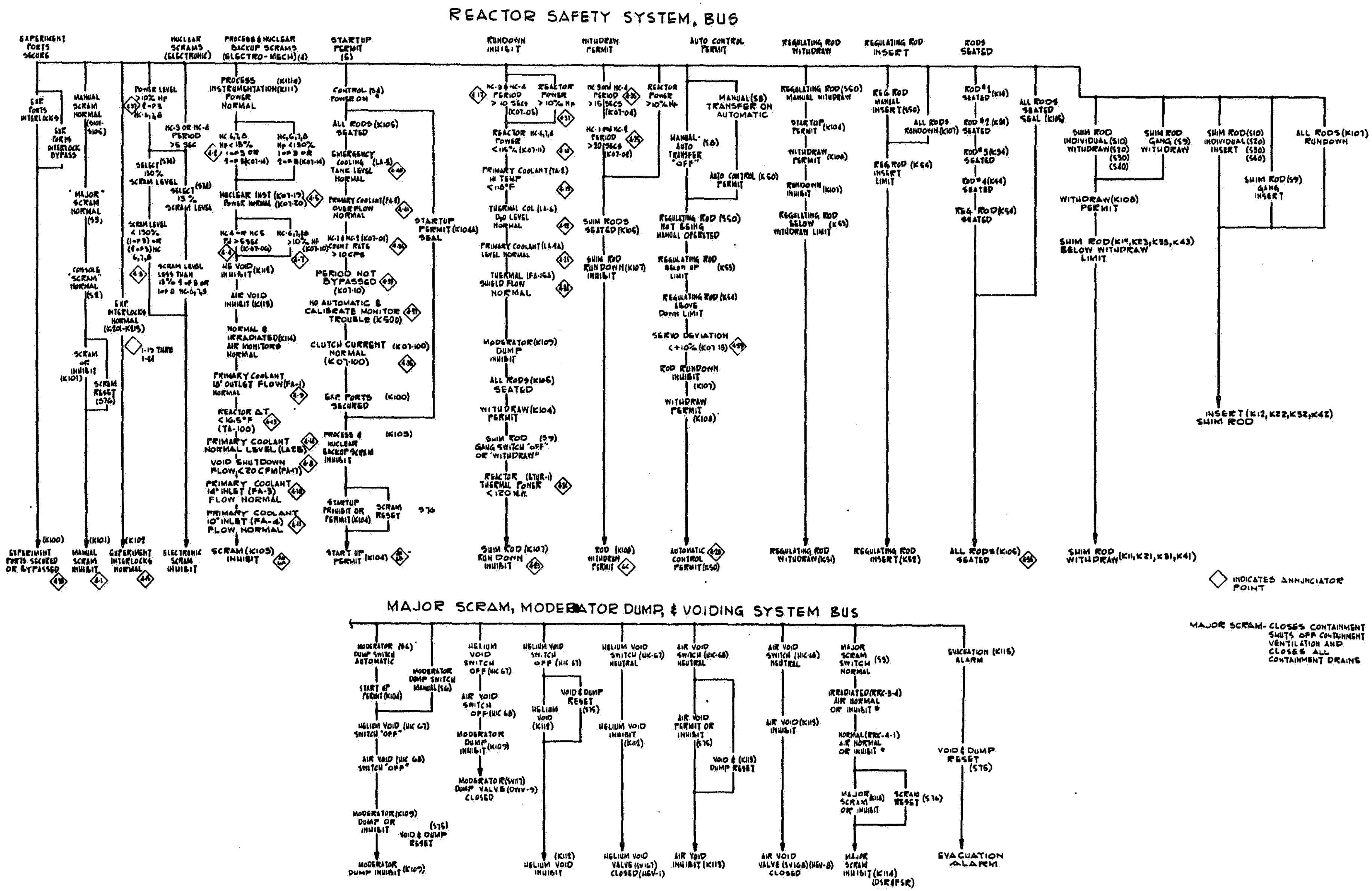


Figure 9.12 Control logic diagram.

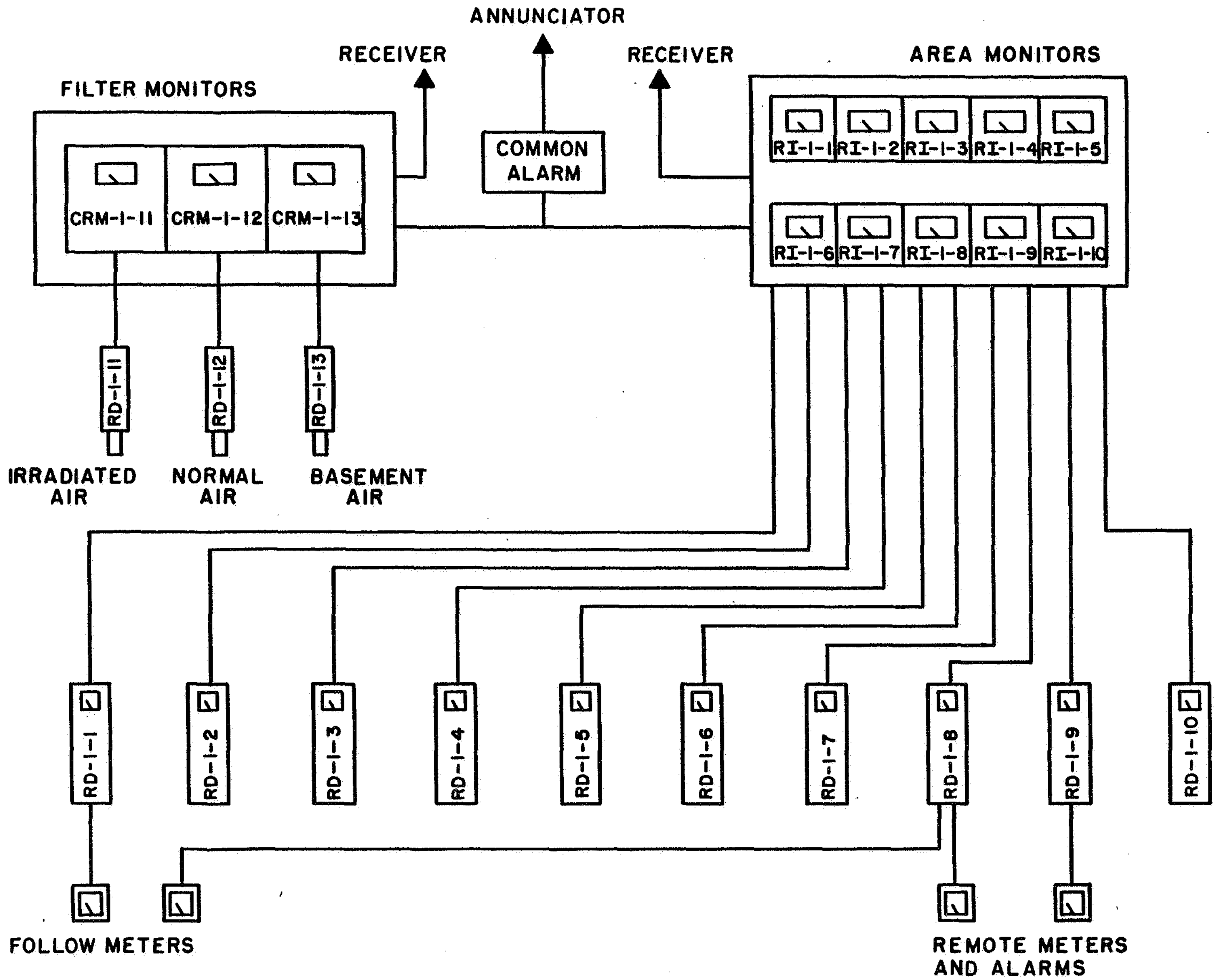


Figure 9.13 Reactor building radiation monitoring.

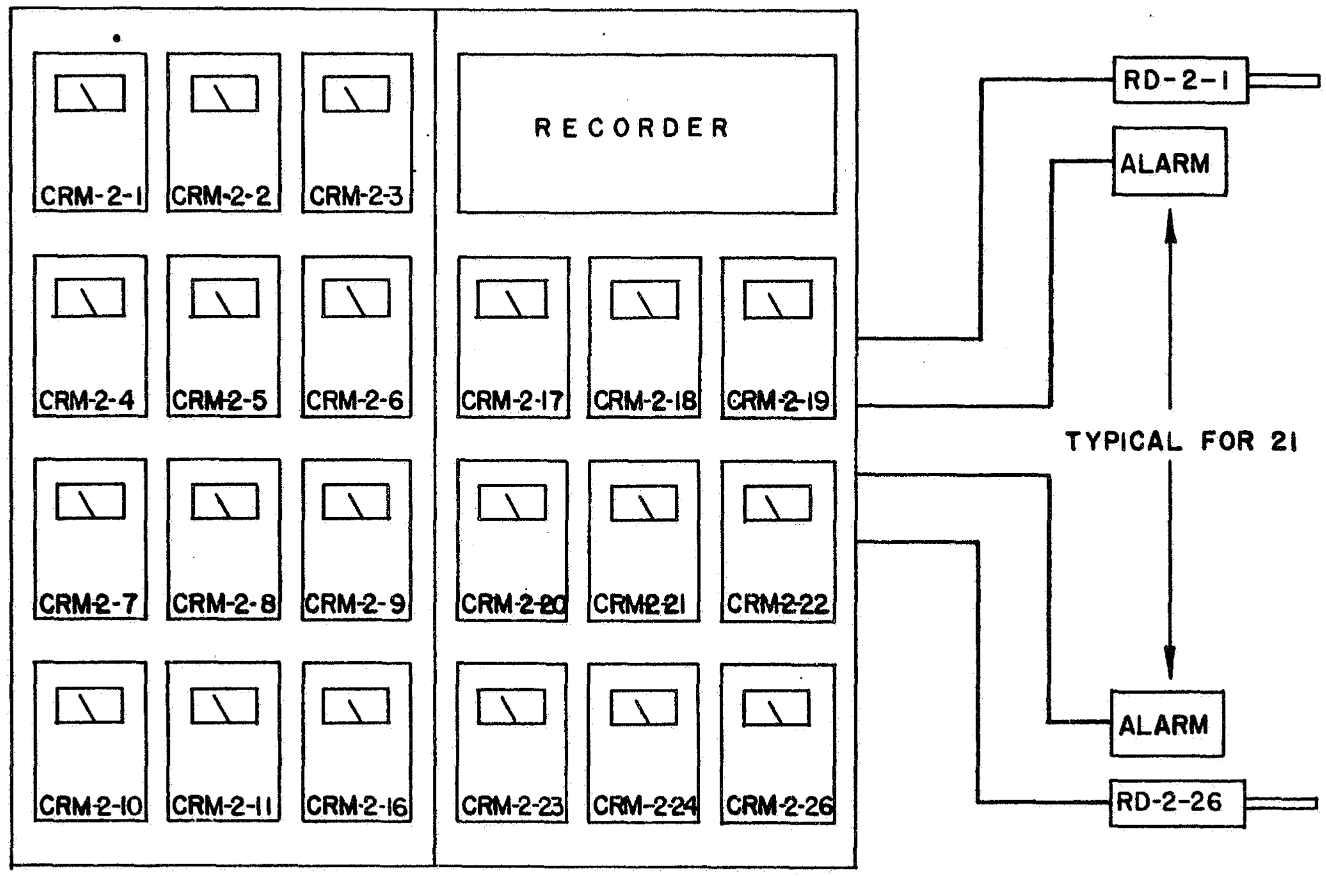
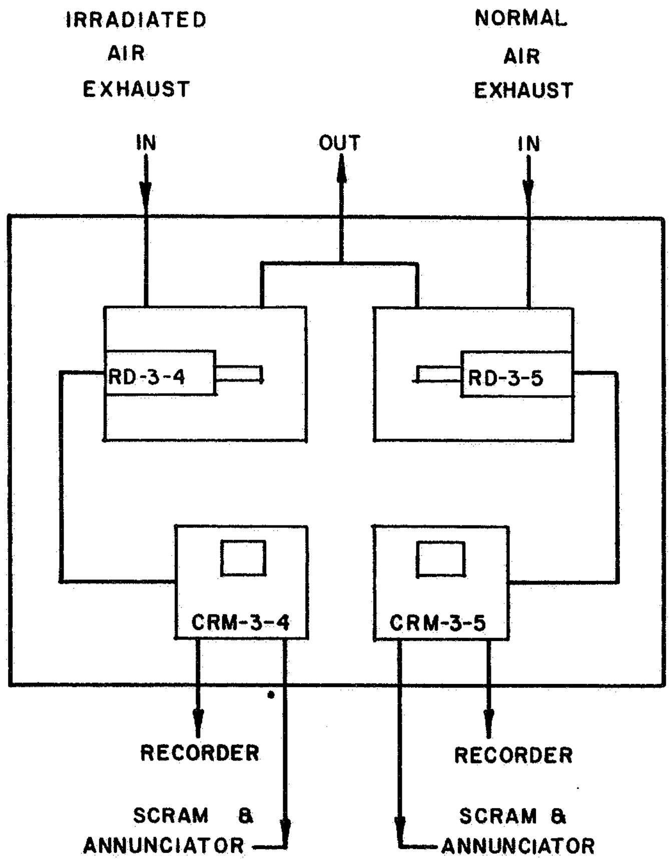
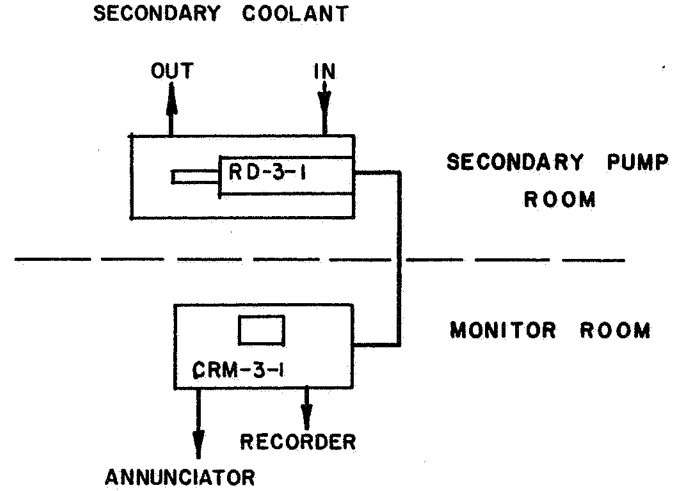


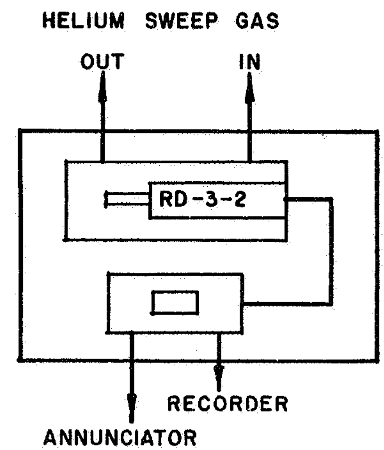
Figure 9.14 Fume hood monitoring.



IRRADIATED AIR EXHAUST AND NORMAL AIR EXHAUST MONITOR



SECONDARY COOLING MONITOR



GASEOUS FISSION PRODUCT MONITOR

Figure 9.15 Reactor plant radiation monitoring.

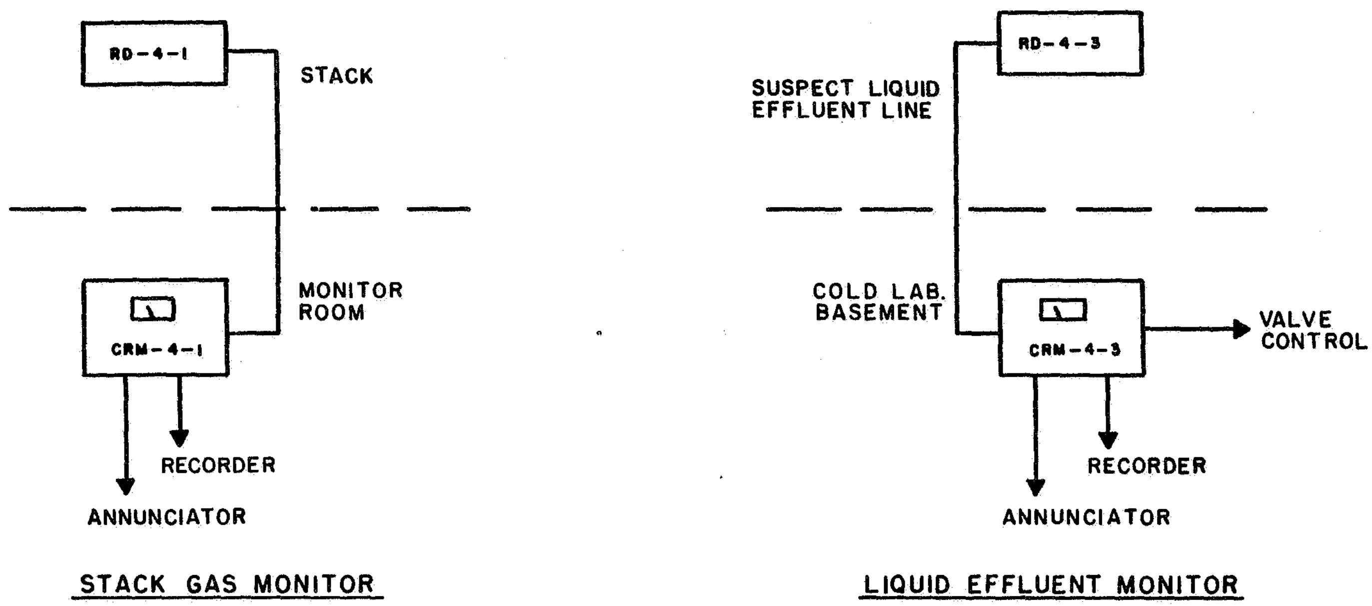


Figure 9.16 Plant effluent radiation monitoring.

SECTION 10. SHIELDING

10.1 REACTOR SHIELDING

10.1.1 INTRODUCTION. The reactor core is surrounded by 20" of heavy water reflector. In addition to acting as a reflector, this provides the first stage of shielding. Next, there is a steel and lead thermal shield surrounding the reactor vessel. The thermal shield removes most of the gamma ray energy and protects the concrete shield from excessive heating. The concrete shield then reduces the remaining radiation to biologically and technically acceptable levels at the outside edge of the shield.

10.1.2 THERMAL SHIELD

10.1.2.1 Design Criteria. The primary function of the thermal shield is to reduce the radiant energy into the concrete to a point where the heating of the concrete will not cause temperature differentials within it which will cause cracking. A gamma energy flux of less than 15 mw/cm^2 on the inner surface of the NBSR biological shield will produce a maximum temperature differential of less than 50°F within the concrete. This value of the temperature differential is a conservative upper limit which assures that the concrete will not crack from thermal stresses. Thus, the thermal shield must reduce the gamma energy flux to less than 15 mw/cm^2 and at the same time reduce the neutron flux to a level such that neutron capture gamma rays contribute only a small fraction of this energy.

Another function of the thermal shield is to support the reactor vessel. The thermal shield rests on a concrete foundation. The reactor vessel hangs from its top flange which rests on the top of the thermal shield. The top shielding plugs rest on the vessel flange and so are also supported by the thermal shield.

One additional function of the thermal shield design was to minimize the neutron capture gamma ray flux around the reactor vessel walls and particularly in the vicinity of the beam holes, and at the same time to reduce the activation gamma rays reaching the interior of the reactor vessel from the thermal shield after shutdown. These criteria are the reasons that the interior of the thermal shield and beam port liners are lined with lead.

10.1.2.2 Shielding Design. The thermal shield consists of 2" of lead followed by 8" of steel for a height of 8'4" starting 4'6" below the central plane of the core. The upper section of the shield has 2" of lead and 6" of steel. The lead thickness was chosen to minimize the gamma ray flux at the vessel wall. Since the thermal neutron current entering the thermal shield is quite high (1.5×10^{12} n/cm^2-sec), a large fraction of the gamma ray flux both in the thermal shield and in the vessel wall come from capture neutrons. With no lead between the steel and the tank wall, the capture gamma ray flux from the steel contributed over half the gamma ray flux at the tank wall. The use of lead on the inside of the steel greatly reduced the steel capture gamma ray contribution. The use of too much lead, however, increased the thermal flux in the aluminum and caused the gamma flux at the vessel wall to start increasing again from capture in the lead and aluminum. It was found that 2" of lead gave the minimum capture gamma ray contribution at the vessel wall, being a factor of four below the no-lead case.

The total energy absorbed by the lead and the steel is summarized in Table 10.1-1. The summary shows clearly that the soft core gamma rays are almost entirely absorbed by the lead and that the bulk of the steel is required primarily to shield its own capture gamma rays.

The heating rate in the lead is approximately uniform at about 0.3 $watts/cm^3$. The heating rate in the steel is more complicated since the exponential absorption is modified by the neutron capture distribution. The heating rate in the steel is shown in Figure 10.1. Near the inside edge, the neutron flux is still important which accounts for the bending over of the curve. After the first few centimeters, the curve follows an exponential quite closely.

The total gamma ray flux including core and capture gamma rays penetrating the thermal shield is only 5.6 mw/cm^2 which is well below the value of 15 set as a conservative upper limit.

Table 10.1-1 Summary of Energy Deposition in Lead and Steel of Thermal Shield

Source of gamma rays		Energy deposited/cm^2 of surface	
		Lead	Steel
Core		.63 w/cm^2	.035 w/cm^2
Aluminum Capture		.14	.035
Lead Capture		.18	.040
Steel Capture		.57	.90
	Total	1.52	1.01

The energy carried by thermal neutrons penetrating the thermal shield is negligible. The thermal flux is reduced to 1.5 x 10^5 n/cm^2-sec which converts to no more than 2 x 10^{-4} mw/cm^2.

The fast neutron flux distribution at the reactor vessel wall is shown in Figure 4.18. From these data it is possible to show that the total fast neutron flux (E > 0.23 ev) is 2.8 x 10^9 n/cm^2-sec with an average energy of 100 Kev. The flux of neutrons with energy greater than 1 Mev is about 5 x 10^7 n/cm^2-sec. In this energy range, the production of the 850 Kev gamma ray by inelastic neutron scattering in iron is important. As a consequence, the neutron spectrum is significantly moderated. The number of neutrons penetrating the thermal shield above 1 Mev is reduced to 2 x 10^7 and the number above 5 Mev is reduced from 1.6 x 10^7 to 5 x 10^6 n/cm^2-sec.

10.1.2.3 Temperature Distribution in Thermal Shield. The thermal shield is cooled at the interface between the lead and the steel by copper tubes. The tubes are 1/2" O.D. by .049" wall spaced on 1.6" centers in the central part of the shield. They are soldered to the steel and the lead is bonded both to the steel and the tubes.

The heating rates of the previous section were used in conjunction with the heat transfer properties of the materials to determine the temperature distributions in the lead and steel. Since the rate of energy deposition in the lead is approximately uniform, the temperature distribution in the lead is parabolic with the maximum temperature at the inside face. The maximum temperature differential across the lead is calculated to be 20°F.

The rate of energy deposition in the steel is close to an exponential with the maximum rate near the cooled face. The temperature distribution, therefore, is well approximated by an expression of the form $A(1-e^{-ax})$ for the temperature difference between the cooled face and some point x distance from the cooled face. The coefficients A and a have the values 14.5 and 0.28 respectively for ΔT in degrees F and x in cm. Thus, the maximum temperature differential across the steel is 14.5°F.

When the effects of local heat transfer in the vicinity of the cooling tubes and from the tube walls into the water is allowed for, the maximum temperatures in the thermal shield are about 150°F in the lead and 145°F in the steel.

Thermocouples have been imbedded in the lead and attached to the steel of the thermal shield to monitor the shield temperatures as the reactor is brought to power.

10.1.2.4 Mechanical Design and Fabrication. The mechanical loadings on the NBSR thermal shield result from the weights of the reactor vessel, the core components, the vessel D_2O content and the weight of the top shielding plugs. Additional stresses result from axial and radial thermal gradients and the attendant thermal expansions. In all instances, however, evaluation of the order of magnitude of these combined stresses has shown that they are much too low to warrant further calculation. Thermally induced movements are of the same magnitude as the local concrete shrinkage which will occur at the thermal shield--biological shield interface.

A general view of the NBSR thermal shield is shown in Figure 10.2. This shield was fabricated to the appropriate Burns and Roe, Incorporated drawings by O. G. Kelley and Company of Boston, Massachusetts. In-process inspection and final acceptance testing was carried out under the supervision of members of the NBSR staff.

The steel portions of the shield were formed from ASTM-A-7 carbon steel. These steel sections were then welded together to form a one piece steel shell. Cooling coils were then attached to the inside surface of the steel by clips which were tack welded to the steel. These cooling coils were formed of 1/2" diameter by .049" wall copper tube which conformed to ASTM specifification B-75, soft annealed, type DHP. The inner surface of the steel was then prepared and a lining of St. Joe Doe Run mined lead was metallurgically bonded to the prepared surface and to the cooling tubes.

The maximum allowable unbonded area in any square inch of surface was specified not to exceed 20 percent of that area. Test samples demonstrated that this specification was met or exceeded in all areas of the shield.

10.1.3 BIOLOGICAL SHIELD. The biological or bulk shield of heavy concrete surrounds the thermal shield and reduces the radiation which still remains at the outside of the thermal shield to acceptable levels at the shield face in the accessible areas. The bulk shield is designed to reduce the radiation to levels which will yield normally insignificant amounts of instrument background. This requirement is more stringent than that set by personnel exposure limitations.

The bulk shielding completely surrounds the reactor becoming an integral part of the first and second floors. Access to the reactor is made possible from the top floor. These plugs consist of two doughnut shaped plugs, one above the other, and a stepped cylindrical plug which fits into the doughnut. The center plug is 5' thick which is thinner than the doughnut combination leaving a 2' deep well in the center of the floor over the reactor. This well is covered with a removable steel floor plate 6" thick.

Below the reactor, the shield is penetrated by the large primary cooling pipes. The area of these penetrations below the reactor is enclosed in the subpile room which has walls with a minimum thickness of 3' of regular concrete. The subpile room is located in the much larger process room which is also heavily shielded by 4' and 5' thick walls.

The bulk reactor shield is made of magnitite concrete with a minimum dry density of 240 lbs/ft^3. Its minimum thickness in the high flux central plane region of the reactor is 74". The concrete was formed directly against the thermal shield on the inside and the 1/2" thick steel face plates on the outside. The top plugs are made of stainless steel and filled with 3 inches of lead on the bottom followed by magnitite concrete.

10.1.3.1 Radial Shielding Calculations. The radiation fluxes entering the biological shield in the central plane are given in Table 10.1-2. In addition to these incident fluxes, neutron capture in the concrete produces additional gamma rays that must be shielded.

Table 10.1-2 Radiation Entering Biological Shield

Neutron Flux		Gamma Flux
Energy Interval	Flux	$E \sim 6$ Mev
Thermal	1.5×10^5 n/cm^2 - sec	
Thermal - 1 Mev	2.8×10^9	5.6 mw/cm^2
1 - 5 Mev	1.5×10^7	
5 - 10	4×10^6	
10 - 15	1.3×10^6	

The shielding properties of the magnitite concrete are given in Table 10.1-3. Since most of the gamma flux is generated by neutron capture in either the steel of the thermal shield or the iron in the magnitite concrete, an average energy of 6 Mev has been used to determine the proper gamma ray attenuation coefficient and buildup factor. The results of the calculation yeild a fast neutron flux 1.4×10^{-3} n/cm^2-sec and a gamma flux of 10^{-7} mw/cm^2 at the face of the biological shield. The contribution to the gamma flux from neutron capture in the concrete is only about 25%; the rest is from the neutron capture in the thermal shield. The gamma ray flux corresponds to a dose rate of about 1 mr/hr.

Table 10.1-3 Shielding Coefficients of NBSR Magnitite Concrete

Neutron Σ_r		Gamma Attenuation Coefficients
Energy Interval	Σ_r	E ~ 6 Mev
Thermal - 1 Mev	.194 cm^{-1}	
1 - 5 Mev	.132	0.107 cm^{-1}
5 - 10	.126	
10 - 15	.112	

These calculations are based on a simple, unperturbed concrete shield. They take no credit for structural steel in the actual shield and make no allowance for voids and streaming through cracks around beam plugs. In the design of the shield, care was taken to minimize the effects of voids. Wherever a void was necessary due to some structural feature such as a pipe or shutter well, enough lead was added to the inside of the void to compensate for the gamma stopping power of the concrete that was removed. The beam holes are, of course, designed to extract intense radiation beams from the reactor. These will require extensive individual shielding to meet individual requirements. Each separate case will be reviewed at the design stage and checked upon installation.

10.1.3.2 Top Plug Shielding Calculations. The top plug shielding presents no problem during reactor operation since the reactor core is covered by 10-1/2' of heavy water which effectively reduces the fast neutron flux to negligible proportions. The poison shrouds in the top reflector effectively reduce the thermal flux, and the few neutrons that do get through the water are captured in the boral on the bottom of the top plugs producing negligible gamma radiation. The major contribution to the gamma ray flux during operation comes from neutron capture in the poison sleeves in the top reflector. This contribution, however, is less than 1/2 mw/cm^2.

The most severe condition for the top plug shielding comes eight hours after shutdown when the water level is lowered for fuel element removal. The gamma flux from the core is then of the order of 9 mw/cm^2 and the contribution from the suspended fuel element is 15 mw/cm^2 giving a total of 24 mw/cm^2.

The thinnest part of the top plug system is the center plug. It is 5' thick and filled with heavy concrete except for the bottom 3" which are lead filled. The plug is penetrated by many pick-up and transfer tools. The bulk of the center plug shield, including the head, is more than adequate, reducing the gamma dose rate at the top of the center plug to .01 mr/hr. A greater source of radiation is that coming through the pick-up tool penetrations. The center of each pick-up tool penetration consists largely of aluminum over a 2" diameter region. Also, the requirements that the pick-up tools slide vertically introduces a small crack which is only partially stopped by tight fitting bushings and a step in the aluminum pick-up tool. Estimates have been made allowing for the aluminum and limited streaming which indicate that the field near the top of the center plug in the immediate vicinity of the pick-up tool holding the suspended element will be about 4 mr/hr.

This field constitutes no health hazard since it is in the well in the top floor which is covered with a 6" steel plate. This plate is an integral part of the transfer system

and is always in place when fuel elements are being transferred. Over each pick-up tool the plate is penetrated by openings up to 6" diameter which are normally plugged. The hole for the tool in use is, of course, open and an estimate made for the field 1' above the floor plate over the open hole indicates a radiation field of less than 1 mr/hr.

It is difficult to calculate accurately the effects of such streaming through cracks and less dense materials, so the numbers given above must be accepted only as order of magnitude estimates which indicate that a serious problem does not exist. It should be emphasized, that the conditions discussed here are only temporary, existing only when the water is lowered for fuel element removal and an element is withdrawn above the core.

10.2 PROCESS ROOM SHIELDING

The process room contains all the reactor process equipment and auxiliary systems. It occupies about two-thirds of the available volume in the confinement building basement and is separated from the rest of the basement by concrete shielding walls. It is entered through a single steel shielding door. The region within the process room directly under the reactor is further shielded by the 3' thick concrete walls of the subpile room. The primary cooling loop, including pumps and heat exchanger, runs north from the subpile room and is shielded from the radiological laboratories by a 5' thick concrete wall forming the east side of the process room. The south wall of the process room is 4' thick shielding the pool area. This wall is thinner because the subpile room walls provide additional shielding between it and the bulk of the primary loop. The ceiling is 5' of concrete.

The primary radiation source in the process room is the N^{16} formed in the primary loop water. Its disintegration rate is about 1.5×10^5 disintegrations/cm^3-sec in the first half of the loop. When this is multiplied by the appropriate area of the pipes involved it may be approximated by an infinite cylindrical source having a strength of 1.2×10^9 Mev/sec per centimeter of length of pipe. The geometry of the system combined with the shielding of the portland concrete walls and ceiling reduce the maximum dose rate outside the process room to about .3 mr/hr.

When the reactor is not operating, the N^{16} activity is no longer present and the primary source of radiation is the core decay gamma rays streaming through the piping penetrations under the reactor. This radiation is conservatively estimated to be of the order of 100 r/hr within about 4' of the pipe shortly after shutdown and about 20 to 50 r/hr in the lower regions of the room. It is for this reason that the subpile room shielding was built in order to allow entrance into the rest of the process room.

Residual activities will cause certain areas in the process room outside the subpile room to require individual shielding. This will be particularly true of the main ion exchangers. These areas will be shielded with lead or heavy concrete bricks as required to allow necessary access to and maintenance of process room equipment.

Only one area of the process system which might be a source of radiation during operation is located outside the process room. This is the D_2O emergency cooling tank. This tank is mounted on a concrete slab just under the containment building roof in the south-east corner of the top floor. It is enclosed by a concrete wall and the piping and associated valves are either shielded by concrete bricks or lead sleeves. Very little activity is anticipated from this system, but the shielding has been provided as a precautionary measure.

A special condition exists during fuel element transfer. The transfer chute passes through the subpile room and the 3' of normal concrete making up the room walls does not provide adequate shielding for the process room. Even the additional 4' of concrete separating the process room from the transfer control panel located in the pool area is not quite adequate to reduce the radiation level at the control panel to an acceptable level. An additional shield of 4-1/2" of lead is required around the transfer chute to reduce the field at the control panel to about 1 mr/hr. Even with the lead shield, the radiation level in the process room is expected to be too high to allow occupancy during fuel element transfer.

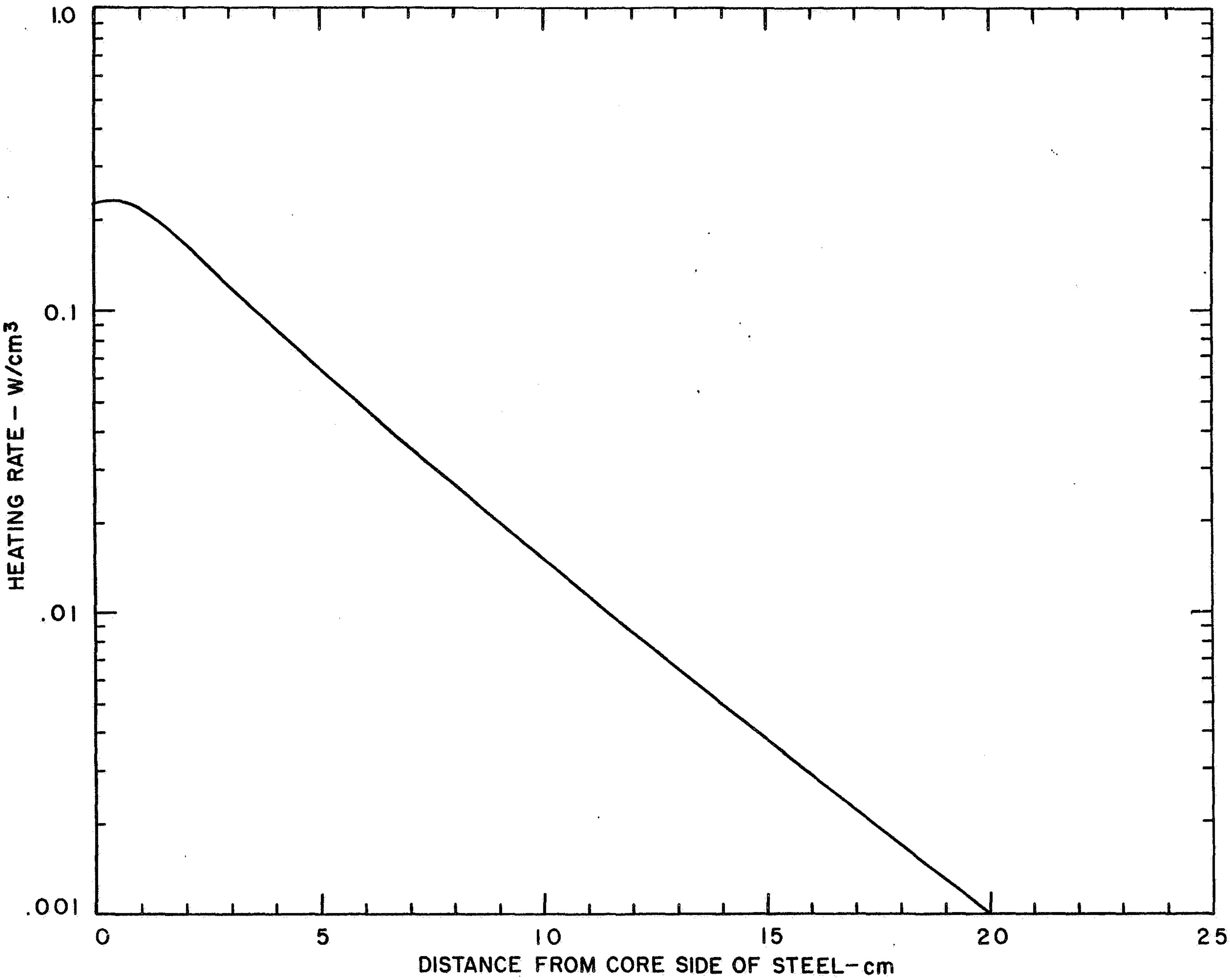


Figure 10.1 Heating rate in the center plane of the steel of the thermal shield as a function of distance from the core side of the steel.

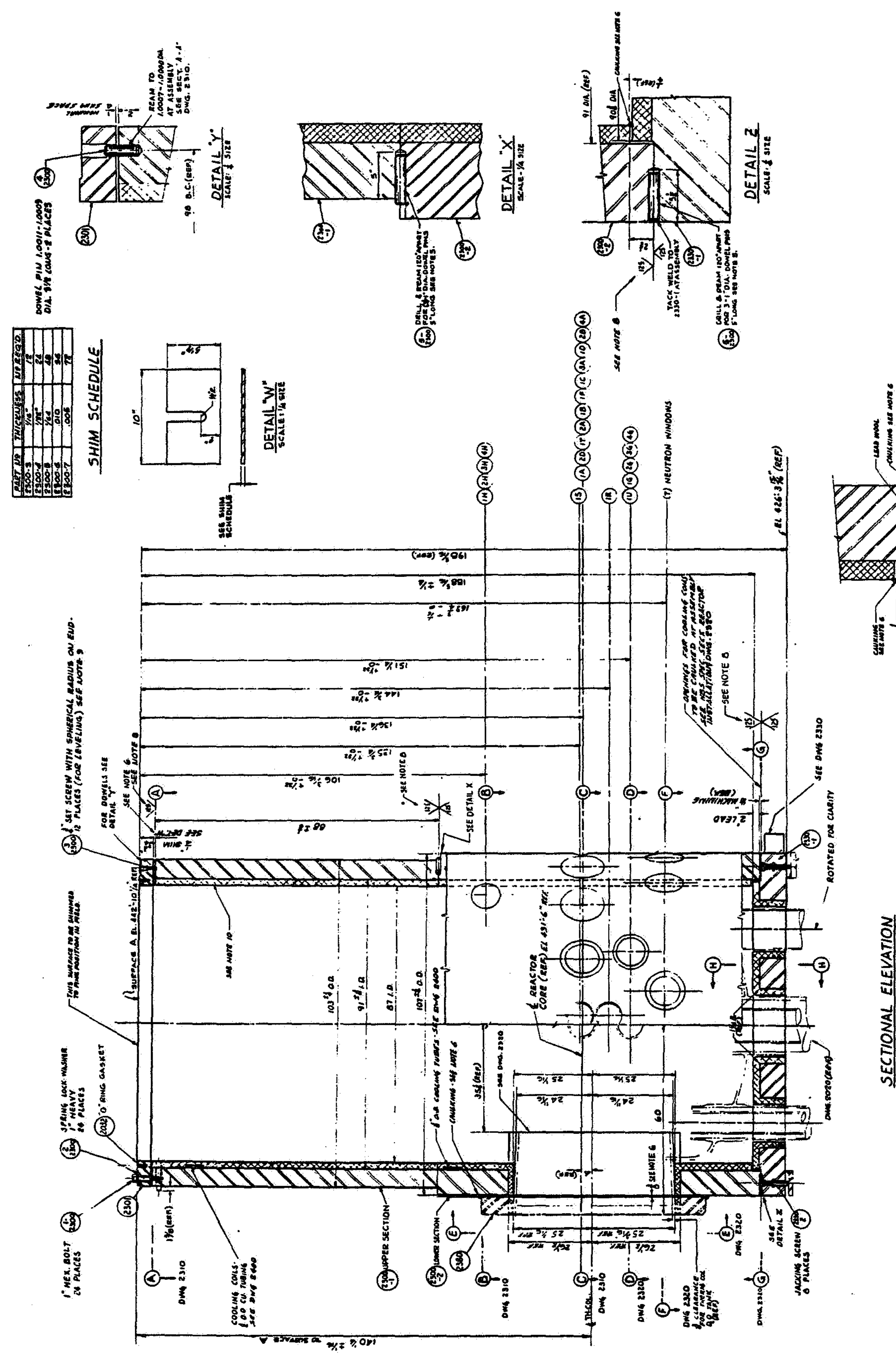


Figure 10.2 NBSR thermal shield.

SECTION 11. INITIAL TESTS AND OPERATING PROCEDURES

11.1 PREOPERATIONAL TESTS

11.1.1 GENERAL. The preoperational tests began during the acceptance period with the general contractor and have continued since. They will end at such time as all systems are proven and the reactor is filled with heavy water. To accomplish this, all water systems are cleaned, filled with demineralized light water, and operated under a variety of appropriate conditions. In addition, the instrumentation and building service systems are also tested and proven. Following this, the entire plant is operationally tested for an extended period of time. Finally, the light water is drained from the primary system, dried, purged with helium and filled with heavy water. All tests will be performed and documented using detailed written procedures.

11.1.2 INSTRUMENTATION. The nuclear and process instrumentation is tested, calibrated, and adjusted after complete installation and hookup to the various sensing elements. Prior to this, the temperature, pressure, and flow elements had been individually calibrated and shown to generate appropriate signals. Proper trip points are set for the nuclear instrumentation by exposing the neutron detectors to portable neutron sources. In addition, all of the self-testing features of the instrumentation are proven. Finally, the nuclear control system itself is tested and proven by closing the loop with a reactor simulator.

11.1.3 PROCESS SYSTEM CLEANING AND FILLING. Prior to cleaning, the process systems are helium leak tested and hydrostatically pressure tested for integrity.

Certain components of the process system, such as the 10 Mw aluminum heat exchanger, were supplied in a clean condition. In the initial flush with potable water these components are blanked off and not connected to the system until the various debris and undesirable disolved material have been flushed or otherwise removed from the system. Final flush is accomplished with demineralized water. It is required that this water maintain an adequately low conductivity and not otherwise disclose the presence of disolved material via chemical analysis.

11.1.4 PRIMARY SYSTEM. Following the achievement of satisfactory cleanliness, the primary system is filled with demineralized water, orifice restrictions to simulate fuel elements are placed in the lower grid plate, and the water circulated at operational flows by the main pumps.

At this juncture the control blades and rod mechanisms are installed, and the system is hydraulically tested; that is, pressure drops for various flow rates are observed as well as the performance of the lower plenum control valves. In addition, coast down characteristics, power delivered to the water by the pumps, and other such features are observed.

The shim blade mechanisms will be tested for insertion and drop times, lateral stability, etc., and the regulating rod mechanism will be tested at full flow conditions. Similarly, the dump and void shutdown systems will be tested for flow, etc. In the latter case the void fraction and its effectiveness cannot be measured until after criticality has been achieved.

11.1.5 LIGHT WATER OPERATIONAL RUN. Having achieved a clean tight system with all process instrument components functioning properly, the plant will be run for an extended period of time. During this time, rechecks will be made of the capacities of all components, instruments, etc. Also, at the start of this run the resins will be placed in the ion exchange columns, and the chemistry of the primary and secondary systems maintained throughout the test.

This period will also serve to train operational personnel on the working of the plant and to effect further detail in maintenance and operation.

11.1.6 HEAVY WATER FILL. After successful preoperational running with light water, the primary system will be drained, dried, purged with helium and filled with heavy water. The resins will be deuterized, and the system filled and retested for a limited period of time. Following such tests, the plant is ready for fuel loading.

11.2 INITIAL CRITICALITY TESTS

11.2.1 PRELOADING REQUIREMENTS. The basic preloading requirements are the same as for a regular loading in that all systems have been checked out according to written procedure. Certain initial startup conditions are different, however.

11.2.2 STARTUP CONDITIONS. The startup conditions are the same as for regular startup except that: (a) no low-flow is required, (b) a neutron source of approximately 10^7 n/sec is required, and (c) the extra startup channels are operative.

The system is closed in the normal manner, except that air is used in place of the helium blanket since the fuel access port is open.

11.2.3 FUEL ADDITION. With at least one shim arm in a cocked position, the first fuel element is inserted in the top access port and transferred to the core where its position is logged. The neutron multiplication is then observed for at least three positions of the shim arms; (a) all in, (b) approximately half out, and (c) all out. This procedure is repeated for each fuel element addition. In these observations the shim arms are moved at regular startup rates. The usual inverse multiplication plots are made for each of the shim positions cited so that three independent approaches to criticality may bo observed. Of course, criticality will only be realized for the all-out shim position.

11.2.4 PROCEDURES AND PRECAUTIONS. As has already been indicated, detailed procedures for fuel handling and startup exist in the Operations Manual. These are followed in the initial startup. Care is exercised to; (a) ascertain that the fuel element has been transferred from the pick-up tool to the grid plates, (b) that the element is indeed locked between the plates, and (c) careful records are kept which identifies each element and its position in the core.

11.3 ZERO AND LOW POWER TESTS

11.3.1 CORE CONFIGURATIONS. At "zero" power it is expected that the first critical core loadings will be realized at $\sim$17 elements. At this point the reactivity worths cited in 11.3.2 will be measured. Thereafter it is anticipated that two elements will have to be added to achieve a core capable of operating at full power. At this point the control rod worths will be remeasured.

In these configurations an aluminum thimble will occupy the axial core position. It is planned to make observations with and without this thimble flooded with heavy water, and also to make flux measurements in the thimble. Since this constitutes the subject matter of the following two paragraphs, it will be treated in greater detail there. In this paragraph it is noted that such an arrangement does not constitute a separate core configuration. This follows from the fact that the reactivity worth of the void contemplated is estimated to be only 0.3% $\Delta k/k$.

11.3.2 REACTIVITY WORTHS. Several different types of reactivity worths will be measured at "zero" power. These include: poison, void, upper heavy water reflector, helium void, shim blade and regulating rod worths. Measurement of the temperature coefficient of reactivity will not be undertaken until some sensible low power is achieved; i.e. 1 Kw or higher.

Access to the thimble tubes and the special axial tube may be gained through the top plug. It is planned only to void the central or axial tube for the purpose of making a void coefficient measurement. Poisons, however, may be placed in any of these tubes. Small amounts will be introduced for the purpose of determining the absorption coefficient

The axial thimble is a special initial device and will not constitute a permanent reactor feature. It is shown conceptually in Figure 11.1. The bottom end seats in the axial

grid position, but is closed; i.e., not accessible to primary water flow. It is filled or voided of heavy water from the top of the reactor and through the central plug. The device will be removed at the conclusion of the zero power tests.

The principal reactivity worths to be measured are those of the control rods, all others being determined from the rod calibrations. Though one or two reactivity measurements may be made by direct observation of the stable reactor period, most measurements will be obtained with the use of an analogue reactivity computer. In this method the rod is scramed from a critical setting while the computer reads out the reactivity worth.

11.3.3 FLUX MEASUREMENTS. It is planned to make flux and neutron density measurements in a variety of thimble positions, for both core configurations and by perhaps as many as three methods. The methods involve the activation of small non-perturbing materials such as gold foils, cobalt wires, and glass beads.

11.4 TESTS AT INCREASING POWER

11.4.1 GENERAL. At each power level the reactor will be operated for several days to several weeks during which time the normal function of the system will be observed; i.e., according to written procedure. In addition, other tests appropriate to that power level will be made during or following the regular run.

11.4.2 TESTS AT VARIOUS POWER LEVELS. A minimal list of the tests to be performed follows. Also shown below, in Table 11.4-1, are the power levels at which the tests will be made.

TESTS

1. Operate for normal function.

2. Shutdown, remove all elements, scan elements for power distribution, and reinsert by reversing dropout procedure.

3. Search process system for active contamination.

4. Measure temperature coefficient.

5. Measure temperatures in the thermal and biological shields with the imbedded thermocouples and predict temperatures for higher powers.

6. Measure and evaluate process shielding for higher power.

7. Calibrate neutron chambers against power.

8. Survey radiation fields through the biological shields and evaluate for higher power.

9. Vary control valves in the lower plena and obtain performance curves relating flow and temperature.

10. Observe coastdown and evaluate proximity to boiling for higher power.

11. Shutdown for various periods of time, reestablish cirticality, and measure reactivity worth of xenon.

12. Shutdown, cool system for an appropriate period of time, withdraw instrumented fuel element into helium blanket, and observe equilibrium temperature rise in plates. Predict temperature for higher powers.

13. Shutdown, remove all elements, drain the heavy water, measure the subpile room field, scan elements for power distribution, and reinsert by reversing the dropout procedure. When the partially spent elements are in the dropout tube measure the γ-field in the pool storage room.

14. Measure N^{16} activity in the primary heat exchanger and estimate sensitivity of a down line detector to a primary-to-secondary leak.

15. Observe noise level in the nuclear instrumentation and compare with throttled helium gas flow through the bubbler system.

16. Throttle pumps via the flow valves to minimum permissible flow to demonstrate absence of nucleate boiling based on 15.

Table 11.4-1 Tests at Various Powers

Test Level	1 Kw	10 Kw	100 Kw	1 Mw	5 Mw	10 Mw
1	X	X	X	X	X	X
2	X					
3	X	X	X	X	X	X
4		X	X	X		
5			X	X	X	X
6			X	X	X	X
7				X	X	X
8				X	X	X
9				X	X	X
10				X	X	X
11				X	X	X
12				X	X	X
13				X	X	X
14						X
15						X
16						X

11.4.3 TRANSITION TO CONTINUOUS OPERATION. Following the successful completion of the above cited tests which includes replenishment of three fuel elements, the core has approached but not achieved its final burned-in fuel distribution. It is planned to add one or two 170 gm elements at a time to the original core of 19 elements as required by fuel burnup until a core of 24 elements is achieved. Thereafter three 205 gm elements will be added every three weeks as described in Section 4.6.4. Each element will ultimately spend 24 weeks in the core at 10 Mw. The exact transition from about 19 elements to the anticipated 24 element core cannot be precisely stated at this time; however, the above cited tests and especially those measuring the power distribution, should clarify this procedure.

11.5 INITIAL EXPERIMENTAL FACILITY TESTS

11.5.1 GENERAL. Concurrent with the plant tests certain general and specific experimental facility tests will be performed. A number of these have no effect on reactivity and otherwise do not constitute a significant hazard. Such will be conducted by scientists setting up equipment at the reactor and include such items as the alinement of neutron diffraction equipment and the measurement of beam flux. Other experimental facility tests are more significant to reactor operations and will accordingly be undertaken by the operating staff. A brief description of these follow.

11.5.2 BEAM TUBES. In at least one beam tube not occupied by experimental equipment an aluminum tank will be inserted which may be voided or flooded with heavy water for the purpose of determining its reactivity worth. Flux and neutron current measurements will also be made in this beam tube.

11.5.3 THIMBLE TUBES. In addition to flux measurements a γ-ray calorimeter will be used in one or more thimble tubes to measure the γ-ray heating rates which prevail.

11.5.4 PNEUMATIC TUBES. Flux measurements will be made in the various pneumatic irradiation tubes. Additionally it is planned to make a study of the cooling capacity of the rabbit system including the method of mounting samples inside capsules by observing the temperature rise in select samples at increasing power.

11.5.5 THERMAL COLUMN. Flux traverses and other plots will be made to map the thermal column flux distribution for several interesting graphite configurations. These include the fully stacked condition and an internal cavity condition.

11.5.6 COLD NEUTRON FACILTIY. Initially this facility will include neither the neutron moderating cryostat nor the bismuth shield. Instead a simple heavy water moderated and cooled tank will be present to provide pile temperature neutrons in its two beam ports. The adequacy of the shutter system and shielding will be tested and flux measurements will be performed.

11.6 OPERATING PROCEDURES

The NBSR will operate in accordance with detailed written procedures. Refer to Section 12.3 for the method of review and approval of these procedures.

Since certain systems must be in operation before nuclear startup, step-by-step procedures will be followed to assure that each system is in proper operation before startup proceeds.

11.6.1 NORMAL OPERATING CYCLE. During normal operation of the reactor, a 21 day fuel cycle is used. After a 2 day shutdown to discharge 3 fuel elements and load experiments, power operation is resumed. In the following section, this operating sequence is described.

11.6.2 REACTOR STARTUP. The reactor contains 24 fuel assemblies, of which the inner 21 have been partially burned in previous cycles. Three of these partially burned elements are repositioned toward the center of the core, being replaced by 3 new elements in the periphery of the core. All control rods are fully inserted and cooling flow is at 800 gpm as maintained by a shutdown coolant pump. The reactor D_2O level is at the new fuel loading position, which is at the bottom of the lower top plug.

In preparation for startup the vessel is carefully inspected for the proper position of all fuel elements and control rods. The D_2O level in the core tank is lowered to the normal level by opening DWV-10. Helium gas is admitted to this top void and the sweep gas system placed in operation. A 25 gpm overflow is then established from the emergency cooling tank to the reactor vessel, using the D_2O purification pumps. The primary and secondary coolant pumps are placed in operation having secured the shutdown coolant pumps. Next the experimental facility cooling, thermal shield cooling, and other necessary auxiliary systems are placed in operation. In addition to the steps outlined above, Operations has checked to see that all shield plugs and experiment shields around beam experiments are in place. The building ventilation system has been checked and is operating normally. All instruments show normal temperatures, flows, levels, and pressures. After a final inspection of equipment, the process room will be locked.

The nuclear safety system and other instrumentation, which has been tested and calibrated during the shutdown, is given a final check to provide assurance that all trip levels are set and functioning. The nuclear instrument channels indicate "on scale" readings due to photo-neutron production in the D_2O or by an installed neutron source. Control rod drive power is then turned "on" and each shim rod checked for proper drive and scram by raising one rod at a time, approximately 5 degrees and manually scramming. All alarm and scram system panels will then be checked for proper operation and the indicated conditions verified. Each of the reactor start permits will be realized sequentially.

Before rod withdrawal is permitted as estimate of the critical rod position is made, based on the estimated available reactivity, experimental loading, xenon concentration, water temperature, and previous critical rod positions.

The reactor is now ready for startup. The regulating rod is withdrawn to its "1/2 out" position. The reactor is then brought critical by withdrawing the four shim blades. The reactor power is allowed to increase on a period greater than 15 seconds, with the regulating rod used for control. Finally after the power level reaches approximately 0.1% of full power, automatic control of the regulating rod can be utilized to bring the reactor to full power.

11.6.3 OPERATION AT NORMAL POWER LEVEL. As the fuel burns up and the xenon concentration increases, the regulating rod gradually withdraws to maintain a constant flux level at the control channel ion chamber. When the regulating rod has withdrawn to approximately its "2/3 out" position, the four shim blades are withdrawn slightly until the regulating rod reaches its "1/3 out" position. This position is repeated until all rods are near the full out position at the end of the fuel cycle.

11.6.4 SHUTDOWN. A normal reactor shutdown will be accomplished either by inserting all control rods or by deliberately feeding test signals into the scram circuitry to simulate abnormal conditions. After all rods are inserted, the drive power is turned off.

The primary pumps maintain full flow until system temperature has reached equilibrium, at which time they are secured and the shutdown cooling pumps placed in operation. After approximately 12 hours has elapsed to allow for fission product decay heat to decrease and other short lived contaminents to decay, operations personnel will have access to the process room so maintenance and refueling operations can begin.

11.7 PERIODIC TESTING

11.7.1 CONFINEMENT AND VENTILATION TESTS. The Reactor Building will be tested as described in Section 3.7.2 upon completion of construction. It will be tested again approximately six months later, then at least annually thereafter. Post-critical tests will be conducted during extended shutdown periods required for maintenance, using the same methods employed in the initial tests, incorporating any refinements which may prove to be necessary.

11.7.2 EMERGENCY POWER. The diesel generators will be tested for proper performance on at least a monthly basis, at which time they will be operated for a minimum of 4 hours to check running temperature conditions. The automatic start feature is also to be tested. Batteries will be tested monthly to establish their proper charge. Specific gravity measurements of each cell will be made periodically.

11.7.3 SAFETY SYSTEMS. Plant nuclear instrumentation will be checked for proper calibration prior to each startup of the reactor. Various inputs to reactor safety instrumentation, which initiate scram, will be simulated to cause safety system action when the reactor is being shutdown. Each input will be tested by this method at least annually. Calibration of linear power channels by heat balance will be performed at least quarterly.

A test of the control rod drive system will be made at approximately six month intervals during which each control rod will be driven through its full travel with the maximum rod speeds recorded and any significant deviation from the pre-set values analyzed and eliminated. Scram times of each control rod will be measured at each scheduled shutdown.

The helium void shutdown will be initiated and checked during each scheduled shutdown to insure proper operation.

11.7.4 RADIATION MONITORS. Area monitors will be calibrated at least semi-annually using a standard source.

Effluent release monitors will be checked once a month by means of an internal source or by background counting where no internal source is installed. An external source check will be performed at least every three (3) months where no internal source is supplied.

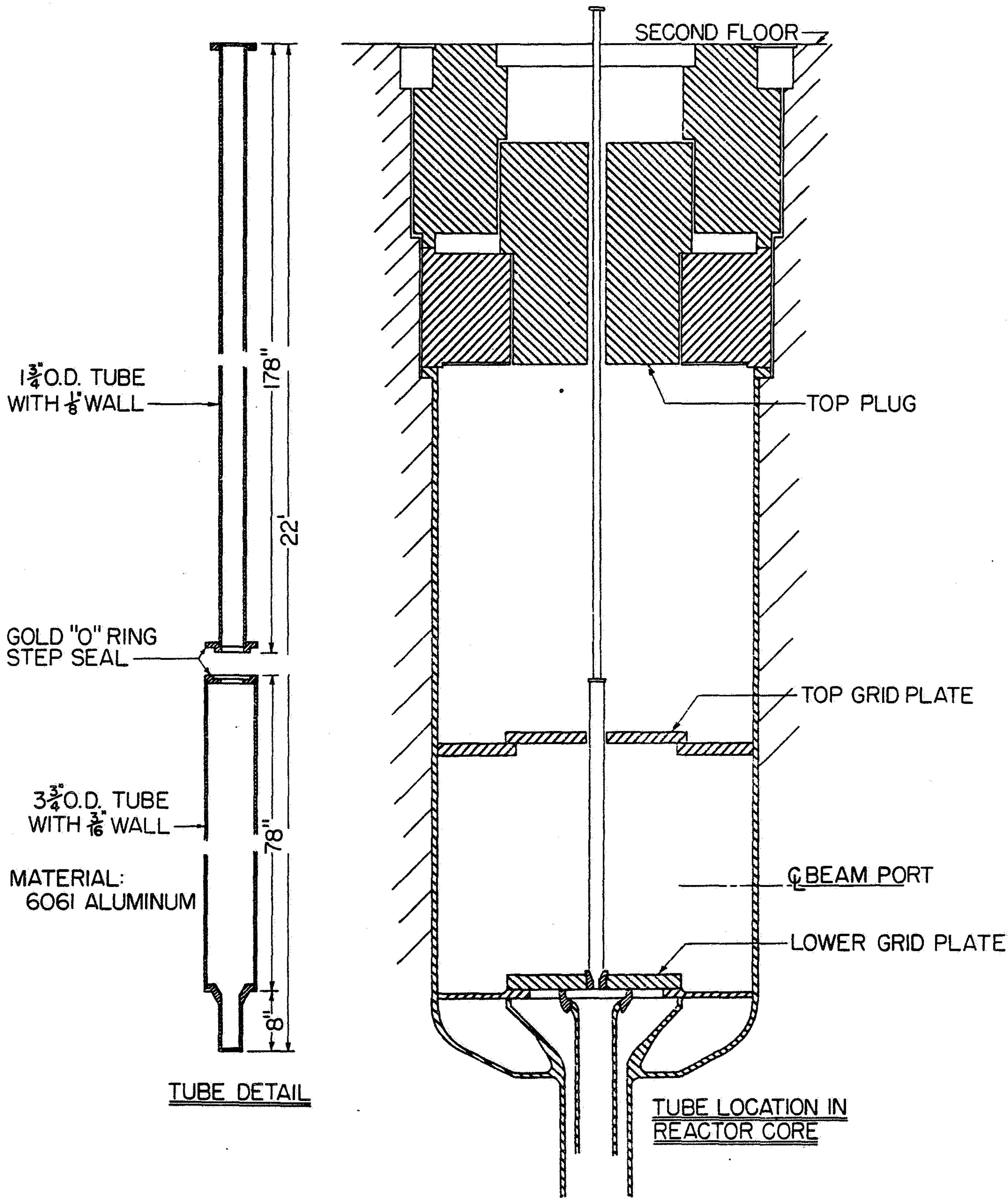


Figure 11.1 Axial test thimble.

SECTION 12. ADMINISTRATIVE AND PROCEDURAL SAFEGUARDS

12.1 OPERATIONS SECTION ORGANIZATION AND RESPONSIBILITIES

The operation and safety of the National Bureau of Standards Reactor are line responsibilities of the Chief of the Reactor Radiations Division and the Chief Nuclear Engineer of the Reactor Operations Section. The organization of the Operations Section for startup and initial power operation is shown in Figure 12.1.

1) Manipulating controls of the facility.
2) Maintaining and calibrating plant instrumentation.
3) Maintaining all facility electrical and mechanical systems and components.
4) Keeping plant records and logs.
5) Controlling plant chemistry.
6) Refueling and fuel handling operations.
7) Operating and controlling hot cell activity.
8) Maintaining fuel and D_2O inventory.
9) Scheduling of plant testing and experimental programs.
10) Procurement of materials and parts as necessary for the facility experimental and operating program.
11) Handling, storage, and disposal of radioactive materials associated with the reactor.
12) Contamination control of high bay, reactor and hot cell areas.
13) Personnel training and licensing.
14) Exercising control over the installation and operation of experiments placed in the irradiation facilities.
15) Compiling sufficient records to establish the excess and shutdown reactivity of the reactor under all conditions of plant operation.
16) Compiling data on the reactivity balance on the reactor, including the effective core loading and experimental facilities.
17) Keeping sufficient records of changes in reactivity coefficients.
18) Keeping records on the power distribution within the reactor core.
19) Review and formulate operating, maintenance, test, and emergency procedures.

12.1.1 ADMINISTRATION

12.1.1.1 Chief Nuclear Engineer. The Chief Nuclear Engineer reports directly to the NBSR Division Chief. He has the responsibility for:

1) Directing the safe and efficient operation and maintenance of the NBSR facility.
2) Maintaining adequate records of plant operations for inspection by AEC and other personnel.
3) Membership on the Reactor Safety Committee.
4) Collaborating with NBSR Engineering Staff, representatives of cooperating area institutions and others in the formulation and execution of the experimental program to be carried out at the NBSR.
5) Directing the training of operating personnel.
6) Recruiting of new operating personnel.
7) Maintaining budget and operating expense controls for the Operations Section.
8) Preparation of monthly and annual operations reports.
9) Directing the efforts of facility operating personnel in the operation and maintenance of the NBSR.

12.1.1.2 Deputy Chief Nuclear Engineer. The Deputy Chief Nuclear Engineer reports directly to the Chief Nuclear Engineer. He has the responsibility for:

1) Reviewing operating and abnormal procedures, final hazards summary report, technical specifications and other documents.
2) Keeping sufficient records of plant thermal, hydraulic and nuclear operating parameters.
3) Scheduling of plant testing and experimental programs.

4) Assuming responsibilities of the Chief Nuclear Engineer during his absence.
5) Assisting in preparation of periodic operating reports.
6) Additional duties as deemed necessary by the Chief Nuclear Engineer or the Division Chief.

12.1.1.3 Administrative and Technical Assistant. The Administrative and Technical Assistant reports directly to the Chief Nuclear Engineer. He has the responsibility for:

1) Determining possible sources of supply and fabrication of highly specialized reactor and experiment components and materials.
2) Generating correspondence and requests for quotations and specifications and making recommendations on selections.
3) Evaluation of needs for procurement of spare parts and equipment.
4) Maintaining records of D_2O and reactor fuel for accountability purposes.
5) Scheduling, expediting, and monitoring various internal technical services such as shop fabrication and plant repair.
6) Maintaining current drawings and information on facility equipment and systems.
7) Maintaining records of Section expenditures for budget purposes.

12.1.2 REACTOR MAINTENANCE, CHEMISTRY, INSTRUMENT AND OPERATIONS

12.1.2.1 Reactor Supervisor-Mechanical. The Reactor Supervisor-Mechanical reports directly to the Chief Nuclear Engineer. He has the responsibility for:

1) Reactor shift operation.
   a) Directing the efforts of shift personnel in the operation of NBSR equipment.
   b) Protecting the reactor and the safety of personnel during his respective shift.
   c) Supervision of all changes in the reactivity of the reactor, whether in the form of experiments or fuel addition.
   d) Exercising control over personnel access to the reactor area of the facility.
   e) Refueling the reactor.
   f) Exercising control over access to and activities in the process room.
   g) Contamination control of all three levels of the reactor area.
   h) Compliance with all facility procedures, administrative rules and regulations.
   i) Recommending changes in procedures.
   j) Assist in training of operating personnel.
   k) Maintaining a current senior "Operator's License."
2) Facility Maintenance.
   a) Establishing a facility maintenance program.
   b) Routine maintenance of all plant process equipment.
   c) Preparation of periodic reports to the Chief Nuclear Engineer concerning maintenance.
   d) Removal and replacement of all absolute filters in plant radiological ventilation equipment.
   e) Securing necessary services from plant division to perform maintenance on facility service systems.

12.1.2.2 Reactor Supervisor-Chemical. The Reactor Supervisor-Chemical reports directly to the Chief Nuclear Engineer. He has the responsibility for:

1) Reactor shift operation.
   a) Directing the efforts of shift personnel in the operation of NBSR equipment.
   b) Protecting the reactor and the safety of personnel during his respective shift.
   c) Supervision of all changes in the reactivity of the reactor whether in the form of experiments or fuel additions.
   d) Exercising control over personnel access to the reactor area of the facility.
   e) Refueling the reactor.
   f) Exercising control over access to and activities in the process room.
   g) Contamination control of all three levels of the reactor area.
   h) Compliance with all facility procedures, administrative rules and procedures.
   i) Recommending changes in procedures.

j) Assist in training of operating personnel.
k) Maintaining a current senior "Operator's License."

2) Reactor chemistry.
   a) Establishing a facility chemistry program.
   b) Regeneration of the demineralized water supply equipment.
   c) Controlling process systems chemistry.
   d) Supervision and assisting in the removal and replacement of all ion exchangers.
   e) Preparation of periodic reports to the Chief Nuclear Engineer concerning plant chemistry.

12.1.2.3 Reactor Supervisor-Instrument and Electrical. The Reactor Supervisor-Instrument and Electrical reports directly to the Chief Nuclear Engineer. He has the responsibility for:

1) Reactor shift operation.
   a) Directing the efforts of shift personnel in the operation of NBSR equipment.
   b) Protecting the reactor and the safety of personnel during his respective shift.
   c) Supervision of all changes in the reactivity of the reactor, whether in the form of experiments or fuel additions.
   d) Exercising control over personnel access to the reactor area of the facility.
   e) Refueling the reactor.
   f) Exercising control over access to and activities in the process room.
   g) Contamination control of all three levels of the reactor area.
   h) Compliance with all facility procedures, administrative rules and regulations.
   i) Recommending changes in procedures.
   j) Assist in training of operating personnel.
   k) Maintaining a current senior "Operator's License."
2) Instrument and Electrical Program.
   a) Assisting the electronic design group of the Engineering Services Section in the establishment of a facility electrical and instrument maintenance program.
   b) Coordinating activities for the program with members of the Engineering Services Section.
   c) Keeping records of all plant electrical equipment and instruments.
   d) Preparation of periodic reports to the Chief Nuclear Engineer on instrument and electrical maintenance.

12.1.2.4 Reactor Supervisor-Nuclear. The Reactor Supervisor-Nuclear reports directly to the Chief Nuclear Engineer. He has the responsibility for:

1) Reactor shift operation.
   a) Directing the efforts of shift personnel in the operation of NBSR equipment.
   b) Protecting the reactor and the safety of personnel during his respective shift.
   c) Supervision of all changes in the reactivity of the reactor, whether in the form of experiments or fuel additions.
   d) Exercising control over personnel access to the reactor area of the facility.
   e) Refueling the reactor.
   f) Exercising control over access to and activities in the process room.
   g) Contamination control of all three levels of the reactor area.
   h) Compliance with all facility procedures, administrative rules and regulations.
   i) Recommending changes in procedures.
   j) Assist in training of operating personnel.
   k) Maintaining a current senior "Operator's License."
2) Reactor facility program.
   a) Preparation of all necessary procedures for the operation of facility systems.
   b) Operating and controlling radioactive liquid and gaseous waste disposal systems.
   c) Maintaining records of the amounts and levels of solid, liquid and gaseous wastes disposal.
   d) Receiving and storage of all D_2O and fuel.
   e) Preparing period reports to the Chief Nuclear Engineer on operational performance of the reactor.
   f) Supervising canal fuel handling and refueling operations.

12.2 MAINTENANCE OF EQUIPMENT

12.2.1 GENERAL. Major maintenance procedures on all portions of the reactor plant will be detailed. Most of the equipment is of a conventional nature which requires no special tools. The components within the reactor vessel do, however, require special handling tools. These have been provided as part of the reactor equipment. These special tools include long-handled wrenches and clamping devices.

Maintenance work which involves opening D_2O systems will be carried out using temporary hoods and exhaust ducts connected to the building exhaust ventilation system to prevent tritium vapor from diffusing into the building atmosphere. The surveying of contaminated equipment and of radiation exposure during maintenance work will be done by the Health Physics Section. Maintenance of contaminated equipment must be performed under a "Special Work Permit," (SWP), which requires Shift Supervisor approval and Health Physics recommendations.

Additional maintenance personnel will be available from NBS Plant Services Division should the need be required.

12.2.2 PUMPS AND MOTORS. Pump motors should require very little maintenance beyond periodic oil changes in bearings. Initial resistance readings were taken of motor windings, which will be checked in the future and compared with installed values.

Pumps will require periodic maintenance particularly on the shaft seals. Replacement of wearing rings, pump shafts, or impellers will require dismounting the motors and draining of associated D_2O lines.

12.2.3 PRIMARY HEAT EXCHANGER. Should it become necessary to open the primary heat exchanger to plug a leaking tube, it will be isolated, drained, and the head removed, presenting no serious problems. The tube is then plugged, tested and the head reinstalled.

12.2.4 VALVES. In the event a valve diaphragm ruptures, that portion of the system will be drained, the valve bonnet removed, and the diaphragm replaced.

12.3 PROCEDURES AND CONTROLS

12.3.1 OPERATING AND TEST PROCEDURES. It is the responsibility of the Operations Section to generate all operating and test procedures; however, various conditions and tests may warrant cooperation among various facility personnel.

Copies of proposed procedures are to be distributed to members of the Reactor Safety Committee for their review. Procedures which rasie objections are to be reviewed and revised by the Safety Committee, meeting as a group, all others proceed with formal approval.

The Chief Nuclear Engineer approves all operating and test procedures to be used at the NBSR. This depends upon prior review by the Reactor Safety Committee members and Operations personnel. Approval is to be based on agreement with the Technical Specifications, the plant operating limits and set points, system design factors, compliance with good engineering and operating practices, and nuclear considerations.

Major revisions to procedures which effect the philosophy of operation and all changes effecting nuclear safety must be approved as described above.

Only the minor details of an approved operating procedure may be revised by the Shift Supervisor or the Deputy Chief Nuclear Engineer. The person making the change, notes and initials the change on the master copy of the operating manual, completes an "Operating Procedure Change Notice," and transmits the original to the Chief Nuclear Engineer, with carbon copies sent to other operations supervisors.

Persons receiving this change notice are to immediately review the change made. Objections to the change are to be written in the place so designated on the change form and returned to the Chief Nuclear Engineer for appropriate action. When the original is returned with no objections, the Chief Nuclear Engineer makes the change permanent and signed copies of the change form are distributed to all persons concerned.

Only minor details, which do not affect the method of performance of a procedure may be changed during the performance of a test by the Test Engineer or the Shift Supervisor. All such changes must be noted on the Control Room copy of the test procedure and operating procedures.

12.3.2 ADMINISTRATIVE RULES. In addition to controls by operating procedures, certain specific methods or philosophies are governed by administrative rules. These rules are issued by the Chief Nuclear Engineer as controls over various areas of NBSR operations. They describe and define operating limitations. A list of such rules, but not limited to, are as follows:

1) Responsibilities of operating personnel.
2) Personnel requirements for reactor operation.
3) Nuclear instrumentation requirements for reactor operation.
4) NBSR logs, records and instruction books.
5) Operating and test procedures review and approval.
6) Refueling operation personnel and instrument requirements.
7) Process room entry.
8) Receipt, storage and handling of radioactive materials.
9) NBSR operating manual--scope, approval and revision.
10) Fuel transfer and canal operations.
11) Tagging and clearance procedure of equipment for maintenance.
12) Special work permit for maintenance of radioactive equipment.
13) Reactor startup and operation requirements.

12.3.3 OPERATING RECORDS

12.3.3.1 Shift Supervisor's Instruction Book. This book is to be used primarily by the Chief Nuclear Engineer. It contains specific instructions for duties to be carried out during a given shift. These matters may apply to preventative maintenance work, experimental sample handling, and similar matters which are not connected with reactor operating limits or nuclear safety. Other personnel authorized to place instructions in this book include the Deputy Chief Nuclear Engineer, the Reactor Supervisor-Mechanical, the Reactor Supervisor-Chemical, the Reactor Supervisor-Instrument and Electrical, and the Reactor Supervisor-Nuclear.

All entries written in this instruction book are dated and signed by the person making the entries.

12.3.3.2 Shift Supervisor's Logbook. A shift logbook is kept by the Shift Supervisor. Its purpose is to summarize all significant plant operations. Entries are to be in sufficient detail for oncoming shifts to understand the status of all plant operations. For example, all significant events and all abnormal operations are to be entered, including those involving tests. Whenever a test is being performed, references to the test number and progress during the shift are to be logged. All scrams are to be logged with time, cause, corrective action taken, and other important aspects. Also included is the date, shift worked, and operating personnel present. All entries must be in ink and signed by the person in charge of the shift. A complete record of the amounts and levels of all gaseous effluent is also included in the logbook.

12.3.3.3 Reactor Console Log. The log includes a chronological sequence of all operations of the reactor. All entries are the responsibility of the Console Room Operator. For example, this log includes the time and date of any significant changes in power level, reason therefore, and control rod positions. All scrams are entered with time, cause, corrective action taken, and other important aspects. All scrams and criticality times are underlined in red. All annunciators which indicate abnormal conditions in the plant are logged with time, cause and corrective action taken.

This log gives the date, shift worked, and operating personnel present. All entries are in ink and are signed upon completion by the person assigned to the console.

12.3.3.4 Process Systems Log Sheets. These log sheets contain a record of liquid levels, flow rates, pressures, ΔP's, temperatures, conductivities, radiation levels, and other conditions in all process and utility systems throughout the plant. They are the responsibility of both the Reactor Technician and the Auxiliary Operator.

These log sheets give the date and the shift, and are signed by the person taking the readings. The log sheets for one day's operation are collected in the Shift Supervisor's office, placed in the file folder with the console log sheets and returned to the Deputy Chief Nuclear Engineer for filing.

12.3.3.5 Canal Logbook. Initially, this logbook will be maintained by a Shift Supervisor. A summary of canal operations, including the shifting of spent fuel from storage racks to the cutoff saw station and subsequent operations are given in detail. Also included is a complete record of the location of all fuel elements stored in the vault as well as the canal. At some future date when the hot cells are added, this logbook will be the responsibility of the Canal and Hot cell Supervisor.

All entries are in ink, dated, and signed by the Shift Supervisor. A summary of hot cell activities is also kept in the logbook, as well as a complete record of all amounts of solid and liquid wastes effluent from the facility.

12.3.3.6 Fuel Charging and Discharging Record Sheets. These sheets, issued by the Chief Nuclear Engineer prior to each shutdown, contain explicit instructions for the movement of spent fuel out of the reactor and new fuel into the reactor. They are the responsibility of the Shift Supervisor.

Specific fuel movements are signed as they are completed. Sheets are returned to the Deputy Chief Nuclear Engineer upon completion.

12.3.3.7 Chemistry Logbook. This logbook is the responsibility of the Reactor Supervisor-Chemical. It reflects routine and special chemistry analysis with the date, time, and origin of each sample.

Reports of water treatment system operation are included as well as reports of all "rabbit" tube irradiations.

12.3.3.8 Maintenance Log. The mechanical maintenance log is the responsibility of the Reactor Supervisor-Mechanical. This log is kept in sufficiently detailed form to provide a complete history of maintenance of all NBSR equipment. It includes such items as the date of preventive or "breakdown" maintenance, parts replaced, and other details of work performed. Information is transferred from this log to individual cards kept on each piece of equipment.

12.3.3.9 Instrument and Electrical Logbook. This record is the responsibility of the Reactor Supervisor-Instrument and Electrical. It is kept in sufficient detail to provide a complete history of maintenance of all NBSR instrument and electrical components. It includes such information as the date of preventative maintenance or repairs, parts replaced and other work performed.

12.3.4 EMERGENCY PROCEDURES. A local NBSR emergency plan has been prepared and integrated with the overall NBS site emergency plan. The procedures of the NBSR emergency plan will be tested by periodic drills.

In the event of a major reactor accident involving radiation or the release of radioactivity, a prepared emergency plan will be put into effect to protect the reactor, NBS personnel and the general public.

The following is included in the plan of action:

1) An evaluation of the emergency situation will be made, remedial action will be taken and necessary aid summoned.
2) Radiation survey coverage and emergency equipment will be used for rescue operations.
3) Personnel will be accounted for.

4) Any injured personnel will receive medical aid.
5) Personnel will be monitored for contamination and decontamination measures will be carried out.
6) An evaluation of dose will be made.

In accordance with the emergency plan, various Bureau and public agencies (Guard Services, Montgomery County Fire Board, Ambulance and First Aid Services, etc.) will be immediately advised of the emergency situation.

If an evacuation is ordered as a result of this emergency, all reactor and NBS personnel will assemble at previously designated locations.

The Atomic Energy Commission will be notified, as required, of all reactor plant accidents, over-exposures, etc., and necessary reports will be submitted.

Procedures will be written, prior to plant startup, that give the proper actions to be taken in response to annunciator signals and alarms which indicate abnormal or emergency conditions of the plant.

12.4 REPORTS

12.4.1 REACTOR OPERATIONS REPORTS. The Reactor Group will prepare a reactor operations monthly report. It is prepared the month following the report period. This report will cover, for example, the following topics:

1) Plant operating performance.
   a) Summary of various operations.
   b) Tabulation of operating statistics
   c) Unusual occurrances
2) Plant chemistry.
3) Plant maintenance.
4) Health and safety.
5) Experimental program carried out.
6) Technical reports issued.

An additional annual report will be prepared serving as a composite of the previous twelve (12) months' operating report.

12.4.2 SPECIAL REPORTS. Occasionally, reports will be prepared of the information of NBS personnel concerning current reactor operating characteristics. This type of report would include, for example:

1) Current control rod worth measurements.
2) Coefficient measurements.
3) Power calibration.
4) Systems performance.
5) Containment performance.

These reports are the results of tests conducted primarily during initial startup of the reactor.

12.5 SERVICE ORGANIZATION

12.5.1 REACTOR SAFEGUARDS COMMITTEE. The Reactor Safeguards Committee will review original operating and test procedures, proposed experiments, and design modifications which relate to the operational safety of the plant. In addition, the committee will investigate the causes of any accidents or any over-exposure of persons to radiation.

The Reactor Safeguards Committee shall be appointed by and report directly to the Division Chief.

The entire operation of the plant, such as administrative procedures, the procedures for operation, plant tests, modifications to the plant, any incidents, accidents and any

over-exposure of persons to radiation shall be subject to review by the Reactor Safeguards Committee.

The committee shall give special attention to all problems concerned with reactor equipment and system malfunctions, operations which would tend to approach safety limits, unusual maintenance problems, or any case of abnormal release of radioactivity to the environment.

The Reactor Safeguards Committee shall keep all records necessary for proper performance of their tasks. A secretary to the committee shall be designated by the Division Chief.

All recommendations of the committee must be approved by the Division Chief before they are placed in effect.

12.5.2 HEALTH PHYSICS. The Health Physics Section is concerned with all actions involving radiation and general safety. Where appropriate, in each area of responsibility, recommendations are made to those who may be subject to any exposure to radiation. All exposure experience is promptly reported to the proper supervisors.

Additional duties include:

1) Radiation control: monitor entries into high radiation areas, supply and evaluate film badges and dosimeters for all personnel.
2) Contamination control: monitor the building for possible contamination. Internal exposure for tritium will be monitored by urinalysis techniques.
3) Airborne and liquid effluent contamination control: monitor the reactor building, the stack discharge, and the liquid effluent for amount and levels of radioactivity released.
4) Environmental monitoring: maintain the program of on-and-off site area surveys.
5) Advise the reactor operations section on proper methods and exposures involved in performing various operations and maintenance.

12.5.3 ENGINEERING SERVICES SECTION. Following the startup and initial operation, as the reactor achieves full power routine operation, this section within the division will provide technical assistance to the operations section on a continuing basis as required. The extensive participation of this section in the NBSR design and startup ideally qualify it for this service. Should major problems involving the mechanical or electronic redesign or repair arise, this section will provide the necessary support and assistance required.

12.6 OPERATIONS PERSONNEL

12.6.1 QUALIFICATIONS. The operating personnel of the NBSR have been selected on the basis of their varied and extensive experience to provide a well balanced organization.

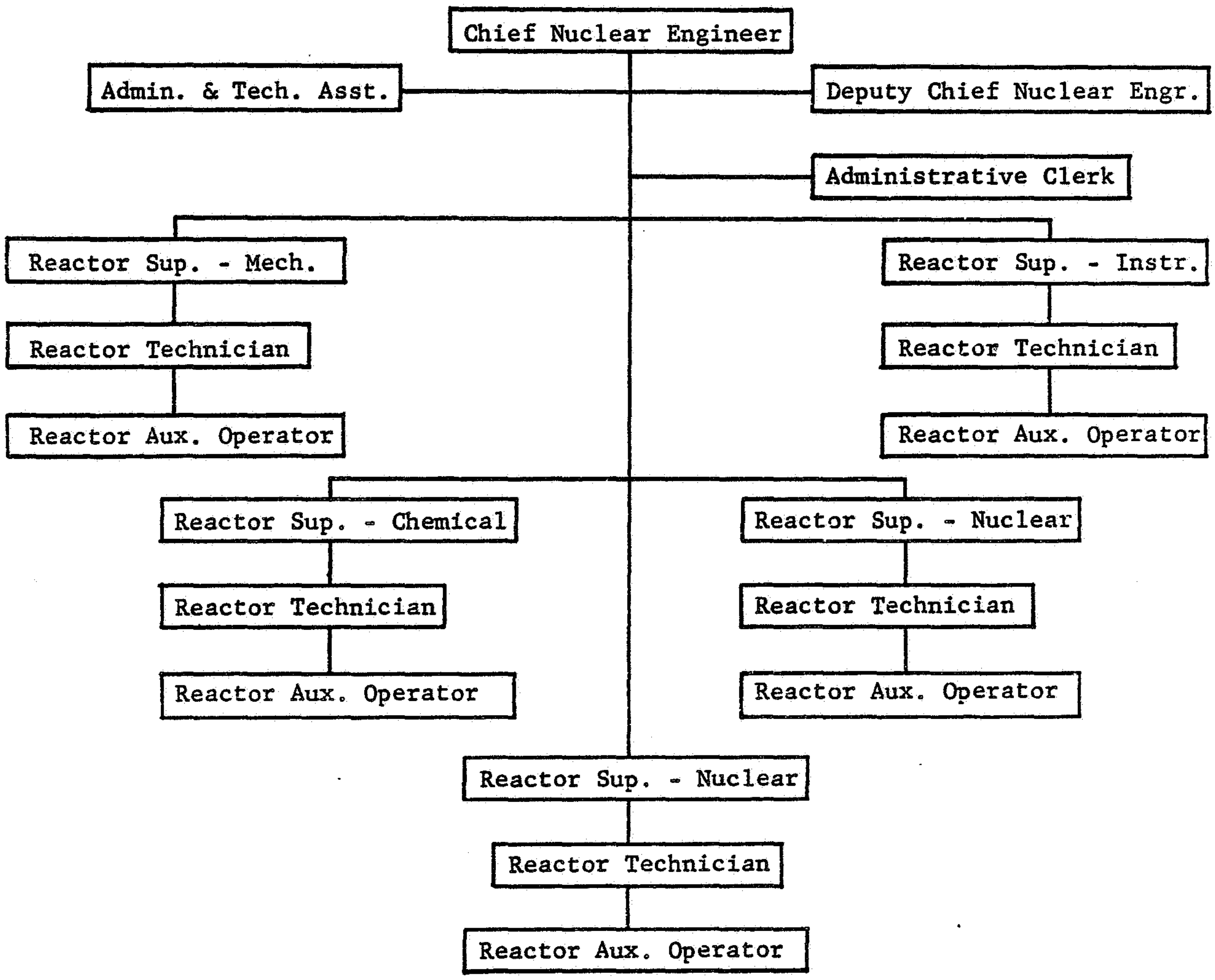
The Chief Nuclear Engineer, the Deputy Chief Nuclear Engineer and the Reactor Supervisors were selected on the basis of their past experience at facilities such as: MTR, CVTR, ERR, EBWR, JANUS and CP-5; giving an excellent cross section of power, research, test and comparable reactors. In addition, some Operations personnel have gained experience in the U. S. Navy Nuclear Submarine program. For supervisors and above, years of nuclear experience range from five to ten with reactor technicians having more than two years of prior nuclear experience.

12.6.2 TRAINING. Training and experience has been provided, as necessary, for personnel presently on the operating staff to assure that adequate experience exists for safe operation of the NBSR.

All site personnel have received training in radiation safety in accordance with the operational needs of the plant and the requirements of Article 20.206 of 10 CFR Part 20, Standards for Protection Against Radiation.

A familiarization course on NBSR design and operation has been presented to operations personnel. Additional experience with NBSR systems has been gained during the pre-operational check-out and testing of equipment.

During the course of reactor operation, a vigorous program of continuous training will be conducted. This program will be designed to fit the needs of each member of the staff. Material coverage will include, for example, radiation safety; procedures review and drills; emergency procedures; basic and intermediate electrical, mechanical, and reactor theory; and practical demonstrations of equipment operation and care. This continuing training program will be directed by the Chief Nuclear Engineer and will utilize academic courses provided by the National Bureau of Standards Technical Career Program for Employee Development.



Date: December 29, 1965

Figure 12.1 NBSR Operations Section organization for startup and initial power operation.

SECTION 13. HAZARDS ANALYSIS

13.1 INTRODUCTION

Following a long running-period at a steady power of 10 megawatts the NBSR will have a fission product inventory in the megacurie range. It is, consequently, absolutely essential that there be adequate assurance that nothing other than a very small fraction ever be allowed to leak from the facility under the worst conceivable circumstances. The design of the facility, therefore, has included a number of barriers which must be traversed before any release is possible. These barriers include at least fuel element cladding, main reactor tank, thermal and/or biological shield, reactor confinement building and/or filters and absorbers. Every effort has been made in the detailed design to make the failure of a barrier incredible. The facility description shows the number of measures taken to accomplish this. The following sections of this report contain a discussion of the likelihood of a release of fission products through failure of the fuel element cladding in the event of a whole series of improbable operational errors, accidents, and malfunctions. Included, in addition, is a discussion of the consequences of the "maximum credible accident" which, though highly unlikely, conceivably may lead to a release of fission products to the interior of the reactor building. The accidents discussed include the startup accident, the maximum reactivity insertion accident, and the loss of coolant accident. The loss of coolant accident in its worst conceivable form appears to lead to the "maximum credible accident," a slow core meltdown with fission product release.

13.2 STARTUP ACCIDENT

It has been well recognized that the control of a reactor during startup is difficult and that, as a consequence of this difficulty, potentially hazardous. It is during this time of startup that large reactivity changes are being deliberately made. It is the time in which nuclear measuring instruments are required to operate under the least favorable conditions. Since the startup level can be ten or more orders of magnitude below the operating or scram levels, the basic startup problem is one of raising the initial neutron level sufficiently so that the detecting instruments can measure the level with good statistics. Control rod withdrawal rates must be set such that there is no possibility that dangerously short reactor periods can be achieved before instrument levels are adequately high. A prohibitively slow rate is undesirable, however, because of the large reactivity change which must be effected during the startup. This problem has been studied by numerous people and discussed fully by M. A. Schultz (13.1).

It should be noted that a D_2O moderated reactor is uniquely characterized in this matter. The fact of photoneutron production in D_2O by fission product gamma emitters makes the startup problem simpler in a very important way. The neutron level of the NBSR reactor after eight hours of shutdown from full power will be in the range corresponding to a 10-100 watt power level. This means that the dynamic range of the startup instrumentation need cover only approximately 5 decades under the most serious conditions, that of startup with large fission product inventory. Initial startup remains more difficult but it need not be made under the most hazardous fully poisoned condition.

As stated in Section 9.1, Reactivity Control, the philosophy of operation places emphasis on the achievement of instrument reliability. With the achievement of reliability a reduction of inadvertent shutdown and consequent re-start can be expected. The minimization of the number of undesirable reactor re-starts permits the relaxation of the compromise between safe but slow startup because of slow rod withdrawal and fast but less safe startup because of rapid rod withdrawal. The choice of conservatively slow rod withdrawal, when combined with the fact of a reduced dynamic power range implicit with D_2O reactors, means the so-called startup accident is much more tractable than would otherwise have been the case. Increased instrument reliability also permits the relaxation of the urgency for fast rod* withdrawal to overcome the rapid xenon poison build-up characteristic of this reactor.

* See the discussion in Section 4.6.11.

An analysis of the so-called "startup accident" has been made for this reactor.* The conventional, simple, and conservative estimate of minimum reactor period that is given by Newson's (13.2) equation has been made for the cases in which the reactor starts either from a power level of 100 watts or from a low level of 10^{-4} watts. This estimate was made assuming a neutron lifetime of 500μsec, a value which was chosen as a probable lower limit to a somewhat variable parameter.** The reactivity withdrawal rate chosen was 7×10^{-4} $\delta k/k$ per sec. The minimum reactor period resulting from steady rod withdrawal interrupted only by power level trip at 150%*** full power (15 Mw) was 0.140 sec for startup from 100 watts and 0.110 sec for startup from 10^{-4} watts.

In order to more fully examine the kinetic behavior of the reactor under conditions of reactivity insertion an analog computer study was undertaken. In particular, step and linear ramp reactivity insertion programs were applied to the analog computer. The kinetic equations for the reactor were written with multigroup delayed neutron terms for the step insertion studies. An appropriately averaged single group approximation was used for the linear ramp insertion studies. The results were initially tested by comparison to an exact analytic integration (13.3) of the kinetic equations. The integration by analog technique was considered preferable because of the relative simplicity for the degree of accuracy required.

The kinetic behavior of the reactor during the startup accident was simulated by the analog program. With this program it was possible to extend the problem to include the simulation of the shim safety arm insertion following maximum power level trip at 15 Mw. For these studies the minimum period as approximated by Newson's equation was established in the program by step insertion of the appropriate reactivity given by the "in-hour" equation. A series of runs were performed with various delay times inserted before shim safety arm insertion was started to simulate the actual delay expected to be encountered in arm drop. The actual shim safety arm insertion of reactivity was approximated by two linear ramp functions, each started after appropriate delay. In this manner a considerably better approximation to the actual reactivity insertion of a freely falling arm**** of variable worth was achieved as compared to a simple linear insertion without delay. A comparison of the reactivity insertion function as approximated in this manner was made to a curve of rod worth published for the CP-5. The results for a reactor period of 0.100 sec are shown in Figure 13.1

The integrated energy released in these reactor excursions is shown on the figure for two separate rod drop delays. The degree of danger to the reactor from the worst of these cases can be judged by comparing the total energy released to that required to melt the fuel elements in the core. A calculation of the energy required to adiabatically bring only the metal of the element structure within the core to the melting point yields 34 Mw-sec, a factor of three larger than the most severe excursion of Figure 13.1. This factor of three is very conservatively estimated when it is considered that the reactor coolant is capable under normal flow and heat transfer of removing 30% of the energy from the maximum excursion. The hot spot of the reactor would probably reach burnout momentarily at the peak of the maximum excursion. This case is a pessimistic description of the startup accident for a least

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* The startup accident is defined as that accident caused by failure of all short period trip instrumentation to arrest an excursion caused by continuous rod withdrawal at a maximum rate.

** See Section 4.5.12.

*** 150% was chosen to allow conservatively an error in the actual set point which will be at 130%.

**** The shim safety arms will not be freely falling since they will be decending through D_2O. However, they are spring loaded in order to increase the initial acceleration. Only approximately 1.2% reactivity is required to return the reactor to just critical and only 0.4% to return to prompt critical. Since the total shim safety arm worth is greater than 20% it is the initial insertion rate which is important in achieving shutdown. The calculations have been conservative in assuming that the worth is initially zero at the instant of rod drop and increases as the square of the sine $\frac{\Pi g t^2}{4L}$ where t is time, L is total arm drop and g is acceleration due to gravity.

two further reasons, however. As indicated, this case results if a 0.050 sec drop delay is assumed. An arm drop delay of less than 0.050 sec is most certainly achievable. Moreover, in these calculations no credit has been taken for negative reactivity insertion which might result from coolant boiling, radiolytic gas formation, or temperature coefficient. These factors, which will be more fully discussed in a following section, would tend to reduce the magnitude of the excursion.

13.3 MAXIMUM REACTIVITY INSERTION

The analog computer studies were extended beyond the "startup accident" study to include reactivity insertions by linear ramp starting from a just critical reactor. In analyzing what are the ways and the maximum rate by which reactivity can be inserted into the reactor a number of possible causes were considered. Step insertion of reactivity was rejected "a priori." Shim safety arm motion was the first source of reactivity insertion considered. The startup accident already studied, however, represents the worst possible consequence of inadvertent shim rod withdrawal. The withdrawal rate of the regulating rod is permitted to be greater than that of the shim safety arms but its total reactivity worth is fixed at a magnitude less than the delayed neutron fraction, β. The ultimate period that can be achieved by removal of the regulating rod is greater than the ultimate period from the startup accident. The consequences of this type reactivity insertion are fully covered by the previous calculations.

Of the other possible means of reactivity insertion, such as beam tube collapse and flooding,* posion removal, cold coolant addition, ** fuel insertion, etc., the maximum rate of insertion, and maximum total insertion of reactivity, appears to come from inadvertent rapid insertion of a fuel element. Whether such an accident as might be caused by an element freely falling into a just critical reactor is credible or not is difficult to prove. In any case, it can be said that it would violate the principle of operation of the fuel element changer and, in addition, require the violation of normal operating procedures and the ignoring of non-normal instrumentation indications. The consequences of this reactivity insertion have been analyzed, however, as the apparent upper limit of a reasonable reactiviy insertion accident. It should, perhaps, be noted that this type of accident could not reasonable happen except following a shutdown period of approximately two days during which the xenon poison, which prohibits startup, is decaying. The decay of the short period activities of the fission product inventory would be significant during this interval and consequently mitigates partially the hazard.

The reactivity increase caused by a falling fuel element was approximated as a linear ramp insertion of 2% $\delta k/k$ total reactivity in a time of approximately 0.4 sec. The time of 0.4 sec is the total time for an element to fall freely through only the height of the core. A constant insertion rate, which is the linear ramp approximation, is a conservative assumption for these calculations. Shim safety arm drop was simulated in the same manner as for the startup accident calculations. Rod drop was initiated by power level trips at 150% of full power and calculations for two separate delay times were undertaken. The results are shown in Figure 13.2. The integrated energy released in the case for which a 0.050 sec arm drop delay was assumed was 9.2 Mw-sec. This is a factor of 4 less than the energy required to heat the whole core to melting adiabatically. Since nucleate boiling would increase the heat transfer coefficient as the power rises, there would be very little fuel element temperature increase until burnout heat flux is reached. Burnout heat flux occurs at the core hot spot at approximately 40 Mw. The maximum power level attained during the excursion was 50 Mw. The time interval for which burnout would be exceeded at the hot spot was 0.7 sec and the integrated energy released to the whole core during this time is 3.5 Mw-sec. The temperature rise in the fuel element at the hot spot would be approximately 130°C if all this energy goes

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* An estimate of beam tube flooding was made allowing flooding to take place at a rate corresponding to water freely falling to fill the void of the beam tube. The rate of reactivity insertion was greatest for the flooding of the cryogenic hole, a hole which will be worth approximately 1.3% reactivity. This is a hole which will not normally be voided. In any case, however, the rate of reactivity insertion and the total reactivity insertible by the flooding of this hole is less than that calculated for the freely falling fuel element.

** Cold coolant addition cannot add more than approximately 1.1% unless chilled water were somehow added. Moreover, it could not be accomplished in a system such as the NBSR without complete change of moderator, which requires a minimum of approximately 45 sec.

to heating the metal of the element. This is adequately confirmed by the "Borax" experiments (13.4). The maximum fuel plate temperature rise observed in a core constructed of nearly the same number of very similar 18 plate elements for an excursion which released approximately the same total energy of 9 Mw-sec was only 110°C. The maximum power in this "Borax" excursion was more than a factor of 2 higher than for the case studied for the NBSR, the period was correspondingly shorter. The film boiling regime was, therefore, more important and a larger fraction of the total energy should have remained in the element.

Results are shown plotted for a case in which a delay of 0.100 sec was allowed before shim safety arm release. It can be shown that even with such a conservative assumption the total energy release of 16.4 Mw-sec is not sufficient to cause element melting. These calculations are furthermore conservative if it is recognized that the presence of boiling heat transfer should add negative reactivity to the system even before the excursions have reached maximum power.

13.4 UNRESTRICTED EXCURSION

The calculation of the magnitude of a reactor excursion which results from failure of all control devices simultaneously with the worst possible reactivity insertion is presently one which has not been performed definitely for this reactor. A considerable amount of data is available from the "Borax" and "Spert" experiments, however, which can be used in a semi-quantitative manner to set limits to the magnitude under certain conditions. The basic problem is the understanding of the transient heat transfer. Heat is generated by fission in the fuel element "meat" and is transferred under non-steady state conditions to the coolant where shutdown steam void is generated. Difficulty arises in a calculation because of the lack of sufficient data to permit description of the heat transfer during the transient. This heat transfer involves successively conduction, nucleate boiling and film boiling regimes. An analytic solution would involve, in addition to a heat transfer model, a coupling of the void forming mechanisms as a negative feedback term to the reactivity of the reactor kinetic equations. The primary objective of the calculations is a conservative, if not exact, estimate of the total energy release during a transient or excursion. A more difficult objective, which is thereby usually secondary, is a description of the fuel element temperature variation. A solution in closed analytic form has not been attempted except for the simplified case involving exponential time variation of reactor power (13.5). The more complete solutions utilizing reactivity feedback involve the use of analog computing machines. A rather extensive calculation has been performed (13.6) using a major analog computing facility. The complete time dependent partial differential heat transfer equation was integrated. The spatial part of the differential equation was approximated by differential-difference equations applied to a finite difference grid. The released energy was included through a void coefficient as negative feedback to the reactor kinetic equations. The results rely, however, even in this extensive calculation on a fitting to the "Borax" and "Spert" experiments.

A calculation of this type using a simplified model in a less extensive analog computer has been performed for the NBSR using core parameters which obtained in the preliminary design stage. These calculations have not yet been repeated for the final core parameters. The original results were fitted to Spert I data to establish the transient heat transfer as all such calculations must do at this stage of our knowledge of the problem. The results show, as have all such calculations, the great capacity of the core to be self-limiting in power excursions because of the negative void coefficients.

A number of arguments based on Spert type data can be used in predicting results for the NBSR reactor.* The ultimate period that results from the uninterrupted ramp insertion of 2 percent reactivity to a critical NBSR core configuration is approximately 0.060 sec. assuming a lifetime ℓ = 700 μsec. Neglecting all shutdown mechanisms other than steam void formation, 30 liters of void uniformly distributed throughout the reactor are required to return the reactor to prompt critical. The energy required to make this steam void at atmospheric pressure is a small fraction of the total energy release in corresponding period transients in Spert I and II. The question of the fraction of the energy release which remains within the element to raise its temperature is the object of a proper calculation. As stated in NBSR 7 calculations indicate that the energy necessary to adiabatically heat the elements to the melting point, namely 34 Mw-sec, will not be reached in transients having a period approximately 0.060 sec.

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* See NBSR 7, Chapter VI.

Experiments on a D_2O moderated core of similar characteristics to the NBSR have been reported (13.7). In particular, transient No. 00582 for an unpressurized expanded core of Spert II gives results for a 0.062 sec period. The total energy release in this excursion was 26 Mw-sec. The total reactivity compensated in this excursion was greater than 6 per cent reactivity. The maximum measured fuel plate temperature was less than 250°C. No melting or damage to the core was observed. It is difficult to be certain that the NBSR would behave in the exact manner but clearly even shorter period transients in Spert II core did not lead to fuel plate melting.

A correlation study of maximum reactor fuel plate temperature versus a parameter which is inversely proportional to reactor period has been made for the GTRR (13.8) using Spert II data. These results show that reactivity insertions of greater than 2.7 percent reactivity are required in the GTRR for fuel plate melting temperatures to be reached. In view of the fact that an unrestricted excursion is so highly unlikely and with the indication that no fuel element melting will take place in any event, it seems adequately safe to assume that no major damage to the core is possible with this kind of accident.

13.5 LOSS OF COOLANT

In the analysis of reactor accidents consideration should be given to the possibility of loss of coolant from the reactor vessel. For the NBSR the loss of coolant, itself, will terminate any power generation due to neutron fission but the residual decay heat from previous fission is a source of heat to the core structure with which it is necessary to contend. Emergency cooling provisions are included in the facility design but the consequences of the failure of all such provision has been studied. The actual problem of time-temperature history of the fuel elements following a sudden loss of all coolant is complex and an analysis would depend on the exact nature of the conditions which obtain following the accident. A number of calculations and the results from recent experiments give an approximate picture of the accident, however.

As has been previously stated, the energy necessary to bring the metal of the core immediately contiguous with the fuel adiabatically to the melting temperature is 34 Mw-sec. The time required to do this following a long operating period at 10 Mw is approximately two minutes if estimated from an integration of the Way-Wigner decay formula. This is, of course, a very conservative estimate of the time. First, it is difficult to imagine a loss of coolant in a time less than 45 seconds, the time for pumping the coolant from the reactor tank, without a major disruption of the tank. This alone would extend the melting time to approximately four minutes. The inner emergency cooling tank is designed to supply cooling automatically without necessity of valve operation. In addition, the core is surrounded by a retaining tank which will keep the core partially immersed unless a major disruption has taken place. Finally, even if all coolant cover to the core is lost, the element can transfer energy and, thus, be cooled by two mechanisms. It can conduct heat to the helium and/or air which would fill the tank following loss of coolant and which, in turn, would transfer heat by convection. It can conduct heat by metallic conduction of its structural supports. Finally, it can radiate energy both as gamma radiation and infrared radiation. The relative importance of these factors can be estimated.

A conservative calculation which allows only heat conduction to the lower plenum plate structure shows that each element is capable of transferring heat at a rate of approximately 300 watts before any melting occurs. An estimate of the rate of radiant energy loss of an element at the melting temperature depends directly on the emissivity but a conservative estimate of 0.1 emissivity gives approximately 600 watts as the rate at which a single element could radiate to its surroundings. Experiments indicate that a single element loses approximately 30-50% of its decay heat by direct radiation of its gamma rays. Finally, the element is capable of losing a large fraction of its decay heat by the convective cooling of air. This fact has been proven by experiments carried out at both Harwell and Oak Ridge. In order to investigate these effects a detailed analysis has been made of the experiments (13.9) performed at Oak Ridge.

A complete discussion of these calculations was presented in NBSR-7, Chapter VI. Further discussion of the question of fuel element heating has been presented in Section 7.2.1.6 of this report.

The results of these calculations do not show that a fuel element will not melt when held in the core structure following a sudden loss of coolant. The results depend on how much initial heat can be expected to be dissipated before the elements are uncovered with coolant and to what degree convective cooling will be effective with the element held in its grid structure.

It is clear that several minutes would be available for emergency procedures, including flooding with H_2O coolant, before melting would be expected to begin. It has been assumed, however, that a slow meltdown of the NBSR core following loss of coolant is the "maximum credible accident." An analysis of the consequences of this meltdown is given in the following section of this report.

13.6 FUEL ELEMENT MELT

If it is assumed that immediately after a long irradiation at 10 Mw the entire reactor core is allowed to reach and maintain for a considerable time a temperature of 1290°F, a fraction of the fission product inventory may be released from the melted elements. A number of calculations have been made in the past and presented in the NBSR 7 and in the record of public hearings during the construction permit application stage of the licensing procedures. The calculations have pertained to both short and long term irradiation of persons located at the reactor site boundary and beyond and have dealt with iodine inhalation and subsequent thyroid gland dose and whole body irradiation. The following paragraphs will summarize the calculations and the assumptions made in performing them.

The question of the expected fractional release of fission produce inventory from fuel elements at or near the melting temperature has been the subject of considerable research. For this summary it will be assumed that the release fractions as stated in TID 14844 (13.1) and cited in Title 10, Code of Federal Regulations Part 100 (10 CFR 100) February 11, 1961, are applicable to the NBSR. It is assumed that the reactor core will release into the reactor building 100 percent of the noble gases and 50 percent of the halogens. It is further assumed that 50 percent of the iodine released to the building remains available for release to the atmosphere.

The source of data on fission inventory has been taken from ANL-WHZ-299, (13.11) TID-14844, and ORNL-2127 (13.12). The original calculations presented in NBSR 7 did not use a format in which the inventory in curies was explicitly stated, as the number of curies of activity in and of itself does not represent a number from which dose can be calculated, nor is the number meaningful unless the exact time after shutdown is stated. For the NBSR after a long irradiation at 10 Mw the total inventory of xenon and krypton in the core is approximately 2.9×10^6 curies based on a one-hour post release average. The corresponding number of curies of iodine is approximately 2.2×10^6 of which only a fraction is iodine 131. Iodine 131 becomes the main contributor to thyroid dose calculations because of its longer half life however.

Several dosages have been calculated pertaining to different conditions of expsoure for varying times. The following calculations have been called hypothetical since several factors which would tend to mitigate the consequences of the above stated release to the reactor building have not been taken into account for the purpose of establishing upper limits in the analyses. Some of these factors are:

(a) The internal recirculation filter system has been assumed to be inoperative, which system would reduce the iodine available to be released.

(b) The high efficiency "absolute" and charcoal filters which are included in the building exhaust system are assumed to be non-effective in reducing fission produce release to the building stack.

(c) The building stack itself is treated as equivalent in height to its actual physical height although the high air velocity in the stack should result in a significant increase in the effective stack height.

(d) The confinement building is assumed to be sealed in conjunction with a falling barometer. The rate of decrease in the barometric pressure is taken to be 5 percent in twenty-four hours, which rate is the greatest observed rate in recorded history for the weather station at the Washington National Airport.

Initially a thyroid dose was calculated for exposure of a person standing at the nearest site boundary a distance of approximately 400 meters. The breathing rate of such a person was taken as the same as in TID-14844. The thyroid uptake factor for iodine was taken as 0.23. The diffusion coefficients used have been discussed in Section2.3.7. Neutral coefficients were chosen since the barometer was assumed to be rapidly falling. The actual stack exhaust rate was taken to be 6.25 percent of the total building contents per day which rate is greater than that caused by the 5 percent barometer change to allow for continuous building in-leakage under the negative pressure condition for the building which would obtain.

The calculated dose is approximately 10 rad to the thyroid. Slightly larger doses would be calculated for the strong inversion condition but this dose would not pertain to the 400 meter distance to the site boundary. For the latter calculation a reduced barometric pressure change would be expected to obtain.

Calculations have been made of the expected whole body dose under the above conditions. Two aspects of the whole body exposure were considered. The exposure was calculated for gamma radiation emanating from the sealed building by penetration through the concrete walls. A calculation of the gamma ray exposure of persons at the site boundary due to immersion in a cloud of Xe, Kr, and I as would be expected to be present as a diffusing stack plume were made. The latter dose was shown to be small compared to the direct radiation through the reactor building. The direct radiation was calculated in two ways. In the NBSR 7 the building was treated as a concrete spherical shell one foot thick. A later calculation takes into account the fact that the reactor building roof is less than one foot thick and that "sky-shine" of gamma rays penetrating the roof must be considered.

The dose for the above calculation assumes the previous fission product release, although since the building is constructed of concrete whose walls and floors are up to five feet thick in places, the entire source would not be effective if it is uniformly distributed throughout the building. The source was considered to be located only in the upper half of the reactor building where the walls are one foot thick and was reduced by the fraction of the total volume that this space occupies. The ceiling itself is surrounded by a parapet and is therefore not visible as a source from the site boundary. A mean gamma ray energy of 1 Mev was assumed in calculating the concrete penetration and build-up factors. No gamma ray energy spectrum degradation was assumed in the concrete, so that the air penetration and air build-up factors are conservatively estimated.

The total dose to a person on the boundary continuously for 30 days was calculated as the extreme. The whole body dose is approximately 1.5 rad of which approximately one-half is direct radiation. The remainder is a conservative estimate of "sky-shine" radiation which penetrates the building ceiling and is scattered by air in reaching the person at the boundary.

The calculations of whole body dose due to gamma radiation from the diffusing cloud are less than one percent of the above dose and can be neglected.

Finally a calculation was performed to define the maximum thyroid dose to a person exposed for an extended time in the stack plume. In order to perform this calculation a choice of assumption must be made as to the nature of the long term or persistent wind conditions one might conservatively expect. In order to define a wind direction frequency distribution which is indicative of a period of high wind persistence records* covering a twenty-seven month period beginning with January 1961 were examined. The month of January 1963 was observed to exhibit a direction frequency distribution with a south wind direction more than twice as frequent as was observed over a ten-year average from 1951-1960. The average wind speed for the south winds observed during this month was 6.3 mph, approximately two thirds as fast as normally observed from this direction. The combination of wind frequency and wind speed makes the choice of weather data for this month more pessimistic from the point of view of wind dispersion characteristics than for any other month of the examined period.

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* The weather data was that recorded at the Washington National Airport. The appropriatness of this data to the NBSR site has been previously discussed in Section 2.3.

It was therefore assumed that following an accident the wind would blow into a 22.5° sector 25 percent of the time with an average wind speed of 6.3 mph. Since this speed is typical of neutral diffusion coefficients, parameters for Sutton's equation were chosen for this regime as defined in Section 2.3.7.

The internal clean-up system continued to be denied as well as all stack effluent filter operation. The calculation was performed as though the nearest site boundary at 400 meters was in the direction of the persistent wind movement, a fact which is actually not true. The building leakage factor was taken to be the expected average over a 30 day period, namely 1.25%. The total dose to a thyroid gland for a person remaining in this cloud for the entire 30 day period would be approximately 30 rad.

If this dose is combined with the short period dose calculated under the rapidly falling barometer condition, a net dose of approximately 40 rad to the thyroid is estimated. This is expected to be higher than the combination of a slightly larger strong inversion regime dose at large distance with the correspondingly lower dose under the persistent neutral diffusion condition.

Several other calculations have been presented in the record of the proceedings of the NBSR public hearings during the construction permit stage, but the above discussed cases serve to set upper limits to possible exposures to persons located at or near the site boundary. It should be noted that the results remain well within the guides of 10 CFR 100 even for the hypothetical assumptions examined.

13.7 SUMMARY

In conclusion it can be stated that a D_2O moderated and cooled reactor of the NBSR type is basically a relatively safe reactor. In addition to possessing a negative temperature coefficient it is protected by an inherently negative void coefficient. The intrinsic safety of such systems for properly limited reactivity insertions has been thoroughly demonstrated by the Borax I and Spert I experiments for H_2O systems and confirmed more recently by the Spert II experiments for D_2O systems. In addition, however, the D_2O reactor is relatively more safe for at least three important reasons. First, the prompt neutron lifetime of the NBSR will be nearly an order of magnitude longer than for H_2O systems. This means that nearly an order of magnitude greater reactivity insertion is required to achieve a given reactor period in the range of periods above prompt critical. Second, the delayed neutron fraction is augmented by photoneutron production in the D_2O moderator. This gives the reactor an increased range between delayed critical and prompt critical. Finally, the fact of photoneutron production in the moderator by fission product gamma emitters makes the startup problem simpler in a very important way. As previously stated the dynamic range of the startup instrumentation need cover only approximately five decades under the most serious conditions, that of startup with large fission product inventory. Initial startup remains difficult but it need not be made under the most hazardous fully poisoned condition.

The previous discussion of this section has shown the nature of the "maximum credible accident" to the NBSR to be a slow meltdown of the fuel elements of the core. The basic point of view taken in the design of the facility has been that in the event of failure of the most important barrier, the fuel element cladding, the final barrier, the reactor building, shall be capable of mitigating the one approach to achieving an adequate final barrier. Any structure that is designed to be sealed must ultimately contend with the possibility of positive internal pressure relative to the external atmosphere because of barometric pressure changes. These pressure changes can amount to one or more pounds per square inch. Containment buildings to withstand these pressures with leakage rates of the order of one percent or less of the building volume per day have been built. Such small leakage from a building with the number and variety of penetrations characteristic of a research reactor building is difficult to achieve in design and especially difficult to maintain. It was decided that it was better not to attempt to limit the leakage under the stringent condition of positive pressure differential, but rather to control the pressure differential. Since the barometric pressure changes are of the order of a few percent per day, the volume of exhaust needed to maintain negligible or negative pressure differential across the building barrier are of the same order. Such an approach is possible when it is not necessary to contend with large positive pressure buildup in the containment building

following an accident. The exhaust volume need not, therefore, greatly exceed that which could be expected from a sealed building under the same condition of falling barometer. A distinct advantage can accrue from such an approach. Since the exhaust effluent is controlled (in contrast to the uncontrolled leak), it can be filtered to remove much of the most serious fission products, in particular iodine. As indicated, filter transmissions of the order of one part per thousand or better can be achieved. Secondly, the filtered effluent can be released high in the atmosphere where turbulent diffusion is more likely to be achieved as compared to a ground level release. The combination of these two factors should result in a considerable improvement over a sealed building unless the sealed building is designed for much less than 1% leakage per day.

In summary, therefore, it would appear that the NBSR is adequately designed and constructed to mitigate the results of any credible accident to such a degree that the operation of such a facility should impose no undue danger to the health and safety of the most seriously affected public.

References

(13.1) M A. Schultz, Control of Nuclear Reactors and Power Plants, McGraw-Hill (1955).

(13.2) H. Newson, Mon P-271

(13.3) H. B. Smets, Bulletin de la Classe des Sciences, 5[e] serie, Tome XLV (1959).

(13.4) J. R. Dietrich, "Experimental Investigation of the Self Limitation of Power During Reactivity Transients in a Sub-cooled, Water Moderated Reactor, BORAX-1 Experiment," AECD 3668 (1954).

(13.5) S. E. Golian, et. al., "Transient Response of Plane Parallel Fuel Assemblies to Exponential Power Excursions," NRL 4495.

(13.6) W. A. Vogele and S. O. Johnson, "Simulation of Prompt-Critical Excursions in the Water Moderated Reactors," WAPD-T-1059 (1959).

(13.7) V. W. Goldsberry, "Self-Limiting Excursion Tests of the Spert II Highly Enriched Plate-type D_2O Moderated Reactor," Part IV IDO-16990 (1964) p. 42.

(13.8) Supplement No. 2, "Final Safeguards Report for the Georgia Tech Research Reactor," Appendix A, December 1, 1964.

(13.9) J. F. Wett, Jr., "Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air," ORNL 2892 (1960).

(13.10) J. J. DiNunno, et. al. "Calculations of Distance Factors for Power and Test Reactor Sites," TID-14844, U.S.A.E.C. (1962).

(13.11) J. M. West and J. T. Weills, "Supplement to a Report to the Atomic Energy Commission on the Proposed Argonne Research Reactor, (CP-5)," ANL-WHZ-299, (1951).

(13.12) J. O. Blomeke and M. F. Todd, "Uranium-235 Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time," ORNL-2127 (1957).

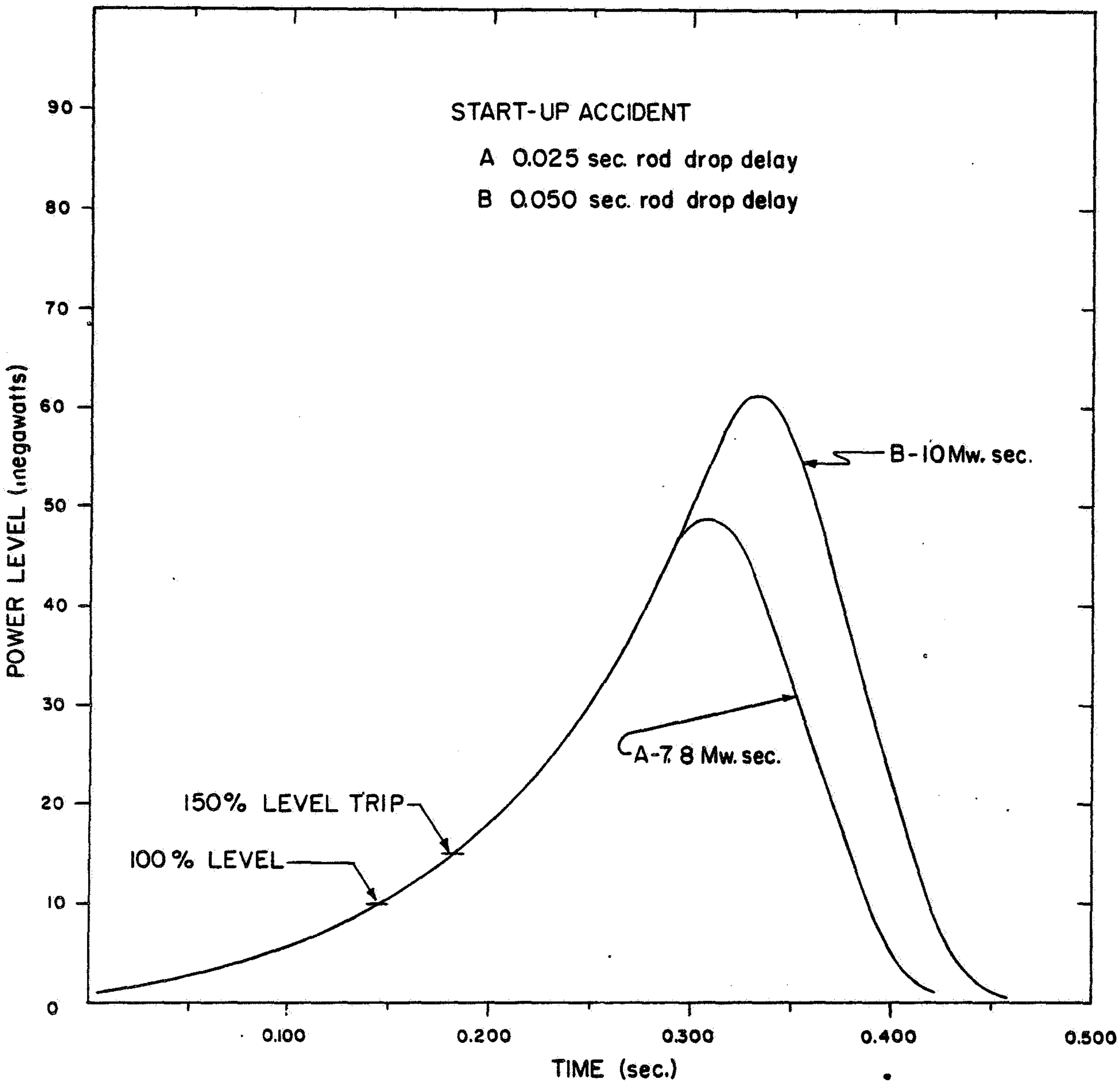

Figure 13.1 Startup accident.

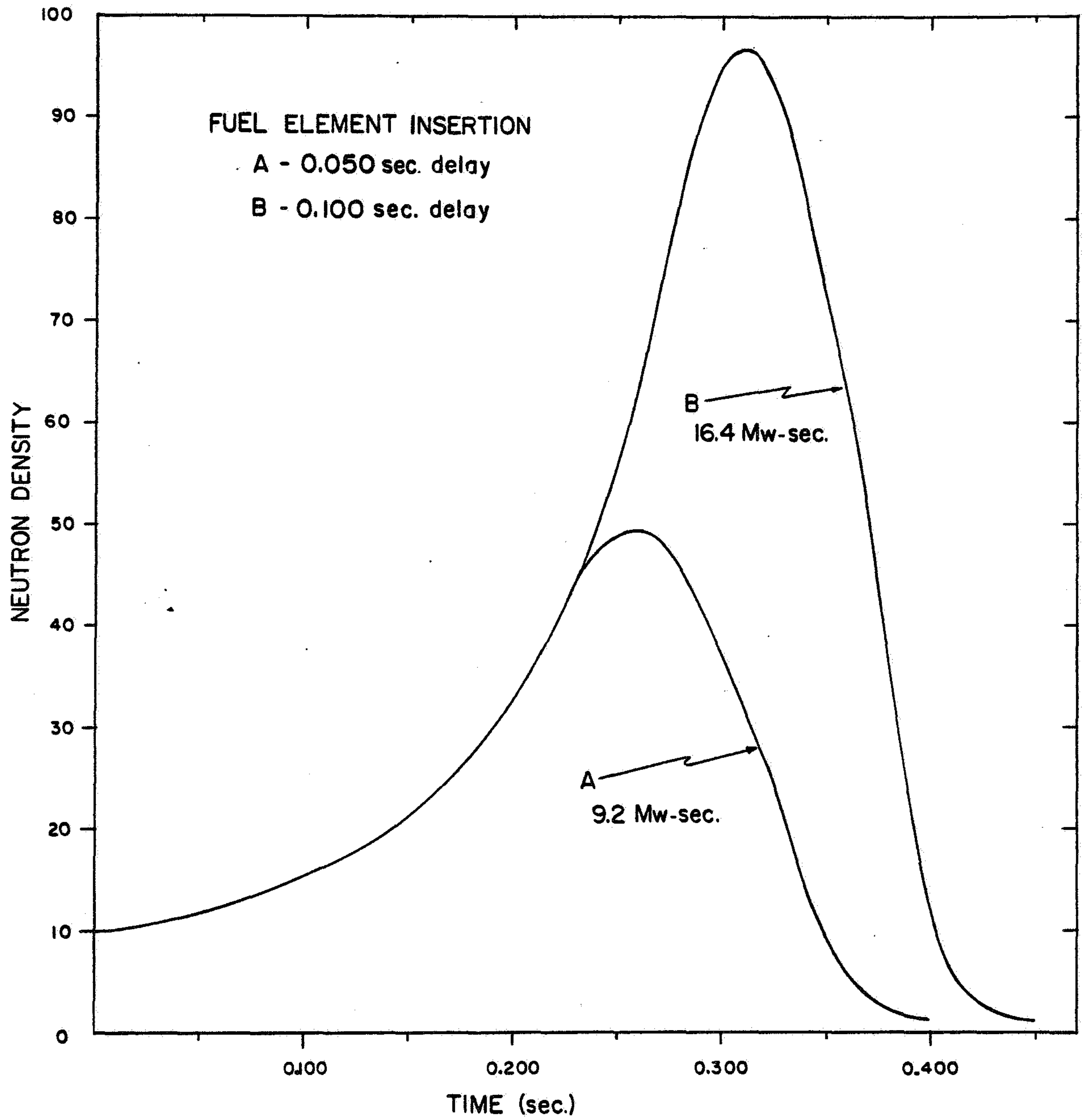

Figure 13.2 Fuel loading accident.