LICENSE APPLICATION

FOR THE

GENERAL ELECTRIC TEST REACTOR

License No. TR-1
Docket No. 50-70 ORDD

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GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT

LICENSE APPLICATION
for
GENERAL ELECTRIC TEST REACTOR

I. INTRODUCTION

General Electric hereby applies for a Facility License Applicable to the General Electric Test Reactor. This application combines an application pursuant to 10CFR Part 50 for an Operating License, an application pursuant to 10CFR Part 70 for License to receive, possess, and use the special nuclear material required in connection with and which results from operation of the reactor, and an application for a by-product material license pursuant to 10 CFR Part 30 to receive possession of and title to, and to transfer to those authorized to receive the by-product material which results from the operation of the reactor.

The General Electric Test Reactor was constructed and has operated as part of the experimental facilities at the Vallecitos Atomic Laboratory in Alameda County, California. It has successfully performed as a high neutron flux facility to further the research, development, and commercial programs of General Electric and its customers. The reactor design, experimental facilities, site, and operating methods are described and evaluated in APED-5000, General Electric Test Reactor Final Hazards Summary Report which is made a part hereof.

II. APPLICATION FOR OPERATING LICENSE PURSUANT TO TITLE 10, CODE OF FEDERAL REGULATIONS, PART 50

A. Information Required by Section 50.33

1. Corporate and financial information regarding the General Electric Company is contained in Section I-A-1 of Amendment No. 41 to License Application for Vallecitos Boiling Water Reactor (Docket 50-18) which by reference is made a part hereof. Copies of General Electric's latest Annual Report were submitted to the Commission by letter dated April 3, 1962.

2. This application is for an operating license for the General Electric Test Reactor licensed under Section 104(b) of the Atomic Energy Act of 1954.

3. General Electric requests the license be issued for a period of ten (10) years from date of issue.

B. Information Required by Section 50.34

Information required by Section 50.34 is contained in APED-5000.

C. Technical Specification as Provided in Section 50.36

A list of technical specifications proposed for adoption as Appendix A to the license are submitted with this application.

D. Agreement Limiting Access to Restricted Data in Accordance with Section 50.37

General Electric will not permit any individual to have access to Restricted Data until the Civil Service Commission shall have made an investigation and report to the Atomic Energy Commission on the character associations and loyalty of such individual, and until the Atomic Energy Commission shall have determined that permitting such person to have access to restricted data will not endanger the common defense and security.

E. Schedule of Receipts and Transfers of Special Nuclear Material in Accordance with Section 50.60

The special nuclear material required as fuel for operation of the GETR is U-235 contained in highly enriched uranium. A schedule which estimates receipts and transfers of special nuclear material, annual average Plutonium production and U-235 consumption and operating losses is submitted with this application and proposed for adoption as Appendix B to the license.

III. APPLICATION FOR LICENSE TO RECEIVE, POSSESS, AND USE SPECIAL NUCLEAR MATERIAL PURSUANT TO TITLE 10, CODE OF FEDERAL REGULATIONS, PART 70

A. Information Required by Section 70.22

1. Corporate and financial information regarding the General Electric Company is referenced in Section II, A, 1 of this application.

2. The general plan for use of special nuclear material required in connection with operation of the GETR is described in APED-5000.

3. General Electric requests the license be issued for a period of ten (10) years from date of issue.

4. The reactor fuel is described in Section 2.3.1 of APED-5000, and use of special nuclear material in experimental and test programs is described in Section 5 of APED-5000.

5. Information required by 10CFR70.22 (a) (5) is included in Section II, E of this application.

6. General Electric Company has more than 20 years of experience in the field of atomic energy. Uranium-235 was isolated in a General Electric laboratory in 1940. General Electric was active in the work of the Manhattan District Project during World War II. Since 1946, General Electric has operated the Hanford Plant for the AEC. General Electric also operates the AEC's Knolls Atomic Power Laboratory and the Nuclear Materials and Propulsion Operation.

   The department has designed and constructed several power, test, and research reactors, including a 200 electrical megawatt nuclear power plant for the Commonwealth Edison Company in Illinois, a 62 thermal megawatt reactor for the Allgemeine Electricitats Gesellschaft in Germany, a 50 electrical megawatt reactor for the Consumers Power Company, a 50 electrical megawatt reactor for the Pacific Gas and Electric Company, and the VBWR, EVESR, NTR and CA and MSCA critical assemblies in addition to the GETR at the Vallecitos Atomic Laboratory.

   The qualifications of Laboratory Management are illustrated by the following descriptions of personnel with responsibilities regarding the GETR:

   Mr. S. W. Akin: Mr. Akin, Manager of Reactor Irradiation, has been responsible for the technical work supporting the completion of irradiation programs at the General Electric Test Reactor, and with the operation of the reactor plant. From 1954 to 1959, he was responsible for the design and procurement of mechanical equipment for the Submarine Advance Reactor project which provided the atomic power plant for the nation's largest nuclear submarine, the Triton. He received his BSME from Oregon State College in 1942. From 1947 to 1954, he was responsible for conceptual design of the sodium cooling system for the Seawolf power plant and development and testing of special heat exchange equipment for this system.

Mr. J. O. Arterburn: Mr. Arterburn, Manager of Reactor Operational Physics, is responsible for the physics evaluations related to the operation, experiments and irradiations of the GETR, VBWR, and shortly, the EVESR. Mr. Arterburn has been with the Department since September, 1955. His work for the Department has included design analysis and theoretical and experimental work with the Laboratory Critical Experimental Facility. From 1952 until 1955, Mr. Arterburn was employed by the Aircraft Nuclear Propulsion Department of General Electric in reactor physics work at Idaho, Oak Ridge, and Cincinnati regarding such facilities as the ETR and RER.

Mr. L. Kornblith: Mr. Kornblith, Manager of the Reactor Technical Operations, joined the Vallecitos Atomic Laboratory in 1956 with responsibility for the design, construction, and operation of the nuclear portions of the Vallecitos Boiling Water Reactor. Subsequently, he was responsible for operation and maintenance of the entire facility.

His present responsibilities include safeguards evaluation and licensing of the Vallecitos

Boiling Water Reactor and the General Electric Test Reactor and the experiments to be performed in them. He also is responsible for auditing the operation of nuclear facilities for compliance with licenses and safety standards and for providing a technical consulting service to the operating components. He is a member of the Laboratory Safeguards Group.

Mr. Kornblith joined the Enrico Fermi Institute for Nuclear Studies at the University of Chicago in 1947. Here he was first employed as Chief Electrical Engineer responsible for electrical and electronic aspects of the design, manufacture, construction and operation of the 170 inch Synchro-Cyclotron. This included power, control, and instrumentation systems. In 1952, he was appointed Chief Engineer with responsibility for all operation and maintenance of the machine, as well as for the design of auxiliary and experimental apparatus. His duties also included installation and operation of the 100 MEV Betatron and consulting on other engineering projects.

Mr. J. H. M. Miller: Mr. Miller, Manager, General Electric Test Reactor Operation, was associated with the design, construction, startup, operation, modification, and maintenance of Hanford production and test reactors from 1945 until 1961.

Startup experience includes supervision of a shift startup crew on two of the Hanford production reactors as well as direct responsibility for the reactor operating personnel associated with the design, critical and power tests performed during the startup of Hanford's Plutonium Recycle Test Reactor.

Since November 20, 1961, he has been employed at the Vallecitos Atomic Laboratory in his present capacity. He is a graduate of Albright College and has received a Master of Science degree from Lehigh University in Chemistry.

Mr. E. W. O'Rorke: Mr. O'Rorke, Manager of the Vallecitos Irradiation Services Operation, joined the Department in 1958. From 1944 to 1958, he was employed at the Hanford Atomic Products Operation where he received extensive training in reactor operations and health physics. The positions he held included reactor shift supervisor, Assistant Chief Supervisor of Reactor Operations, Manager of Fuel Process Development and Section Manager of the dual reactor plutonium production area. Mr. O'Rorke was responsible for Fuel and Materials Development Engineering prior to his current assignment which includes responsibility for Reactor Operations, hot laboratory work, nuclear safety, and site services.

Dr. T. M. Snyder: Dr. Snyder, Manager of Physics at the Laboratory and Chairman of the Laboratory Safeguards Group, began his career in nuclear research at Princeton University, where he participated in developing the concepts of the first chain-reacting pile. At Los Alamos, he assisted in developing the physics of nuclear weapons. He joined General Electric in 1946 in the development of the preliminary pile assembly, becoming Manager of Research in 1956. He came to the Vallecitos Laboratory in 1957.

Dr. Snyder has been a member of the Advisory Committee on Reactor Physics of the Atomic Energy Commission since 1950, and participated in the first United States technical mission to the United Kingdom, in the United States mission to review Calder Hall, and the United States AEC mission to Belgium under the United States - Belgium bilateral agreement. He is a charter member of the American Nuclear Society and was on the Editorial Advisory Committee of Nuclear Science and Engineering 1958 - 1960. He received a citation from Secretary of War Henry L. Stinson and a Navy "E" for work on the atomic bomb in 1945.

Mr. R. C. Thorburn: Mr. Thorburn, Manager of Nuclear Safety, is responsible for establishing Department health and safety standards; providing consultant and audit services with respect to reactor technical operation, reactor operational physics, health physics, and the Department's AEC license program. From 1946 to 1952, he was employed by the Radiological Sciences Department at the Hanford Atomic Products Operation in research on health physics problems, operational monitoring, and development work for reactor and separations areas. From 1952 to 1954, he was Health Physics Supervisor for the California Research and Development Company with complete charge of the Health Physics and Reactor Safeguard Program. In 1956, he returned to General Electric as an Engineer in Reactor Safeguards. From 1956 to 1960 he served as Department Consulting Health Physicist. Mr. Thorburn is a certified Health Physicist, a member of the Board of Directors of the Health Physics Society, a member of the ASA Sectional Committee N-7 on Radiation Protection, and a member of the Laboratory Safeguards Group.

General Electric may replace the above named individuals with others of similar experience and competence, change position titles, and reassign responsibilities without amendment of this application.

7. A description of the equipment and facilities used to protect health and minimize danger to life or property is included in APED-5000.

8. A description of the procedures to protect health and minimize danger to life and property is included in APED-5000.

IV. APPLICATION AMENDMENTS

Application Amendments will be required to describe future designs and operation as necessary to conform to 10CFR50.59. This application, including the Final Hazards Summary Report, APED-5000, has been prepared in loose leaf form to allow continuous consolidation of the application and future amendment in a single document for greater clarity and efficient use.

Subsequent application amendments will consist of a cover section, signed under oath, which will describe the application amendment, its purpose, and identify the new or revised pages therewith included. The cover sections will thereby provide a method for determining that the application is complete at any time. A decimal system of page numbering will be used to provide for expansion. The upper corner of each page will be labeled with the number of the amendment wherewith that page was included. Asterisks will mark the beginning and end of a new or revised description on pages submitted in future amendments.

PROPOSED APPENDIX "B" TO
GENERAL ELECTRIC COMPANY FACILITY LICENSE NO. TR-1

Estimated Schedule of Transfers of Special Nuclear Material from the Commission to General Electric Company and to the Commission from General Electric Company:

Date of Transfer (Calendar Year)	Total Receipts Kg U-235	Pu Production Kg	U-235 Consumed Reactor Kg	Fuel Fab Losses Kg U-235	Scrap Returned For Recovery U-235 Kg	Spent Fuel Returned For Recovery Kg. U-235	Total Inventory Including Reactor Load Kg U-235
1963	13.19	.014	11.17	.07	.46	-0-	59.00
1964	38.92	.014	11.17	.19	1.31	32.8	52.40
1965	38.92	.014	11.17	.19	1.31	32.8	45.85
1966	38.92	.014	11.17	.19	1.31	-0-	72.10
1967	38.92	.014	11.17	.19	1.31	32.8	65.55
1968	38.92	.014	11.17	.19	1.31	32.8	59.00
1969	38.92	.014	11.17	.19	1.31	-0-	85.25
1970	38.92	.014	11.17	.19	1.31	32.8	68.70
1971	38.92	.014	11.17	.19	1.31	32.8	62.15
1972	38.92	.014	11.17	.19	1.31	-0-	88.40

The above estimates are based on a manufacturing yield of 96% an average burn-up of 35% per fuel element, expending 60-460 gram U-235 fuel assemblies and 20-215 gram control assemblies per year.

APED - 5000

Classification I

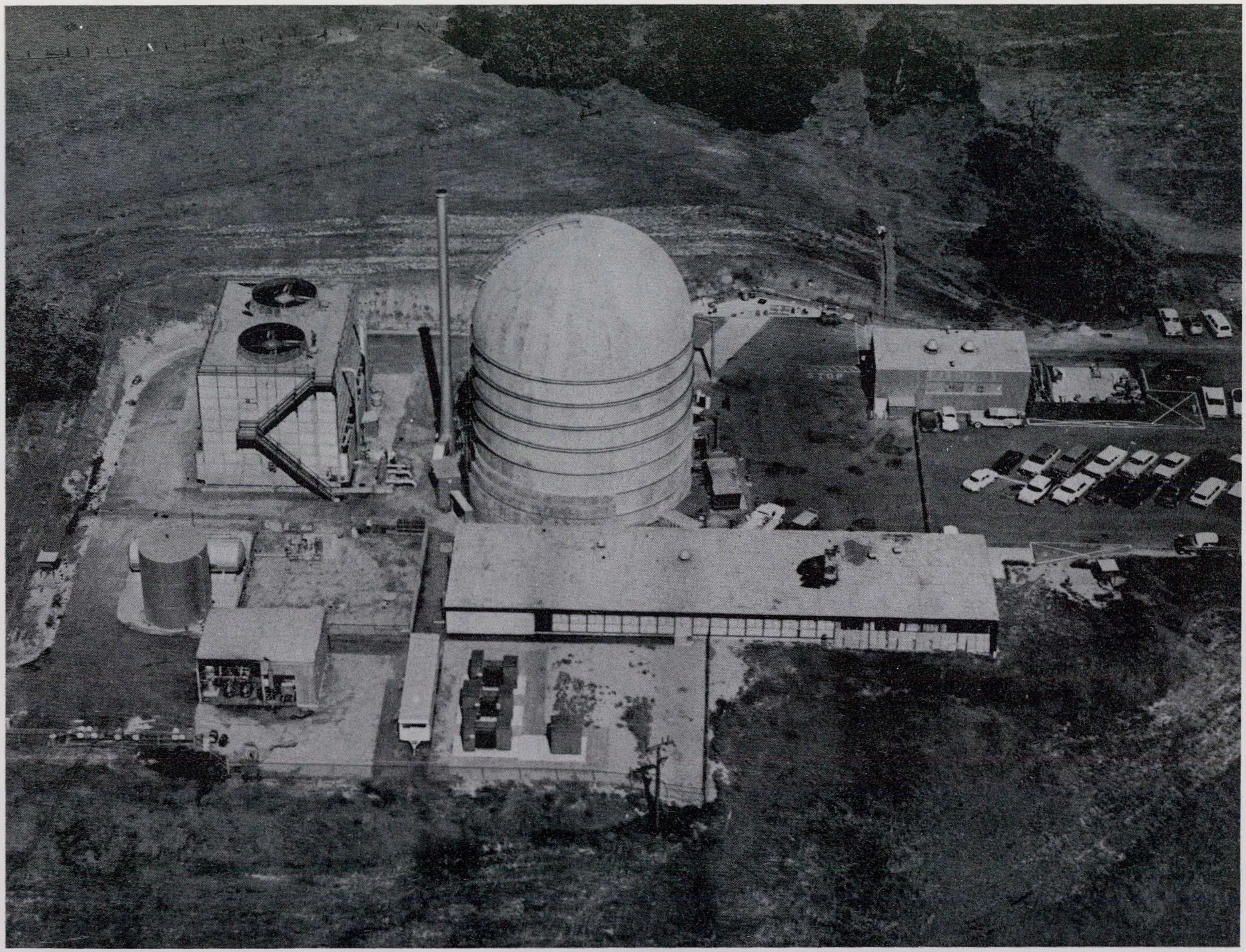
FINAL HAZARDS SUMMARY REPORT

FOR THE

GENERAL ELECTRIC TEST REACTOR

GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT



GENERAL ELECTRIC TEST REACTOR

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SECTION 1

INTRODUCTION

1.1 Introduction

The General Electric Company constructed and is operating the General Electric Test Reactor as a part of the experimental facilities at its Vallecitos Atomic Laboratory in Alameda County, California. The reactor was designed to provide high neutron flux irradiation capabilities suitable to further the research, development, and commercial programs of General Electric and its customers. Over four years of operating experience with the GETR has demonstrated the safety and effectiveness inherent in the reactor's design and operating methods.

1.2 Purpose

The General Electric Test Reactor and experimental facility designs, operations, and evaluations are described within this Final Hazards Summary Report which has been prepared as part of the Reactor's License Application in accordance with Title 10 of the Code of Federal Regulations, Part 50. This report supercedes GEAP-2064, Final Summary Safeguards Report for the General Electric Test Reactor, which previously served the same purpose.

1.3 General Description of the Facility

The General Electric Test Reactor facility, shown in Figure 1.1, is capable of a wide variety of irradiations under varying environmental conditions. The facility consists of a pressurized light-water cooled and moderated reactor and supporting auxiliaries. The reactor and experimental facilities are housed in a containment vessel designed to withstand all credible forces of nature and accident conditions.

The reactor core is contained within the pressure vessel which is submerged in a light water pool. Experimental facilities are located both inside the pressure vessel and in the pool. The reactor is designed to include three through loops in the reactor core, hairpin loops external to the core, a beam port, hydraulic shuttle, both in-core and pool capsule facilities,

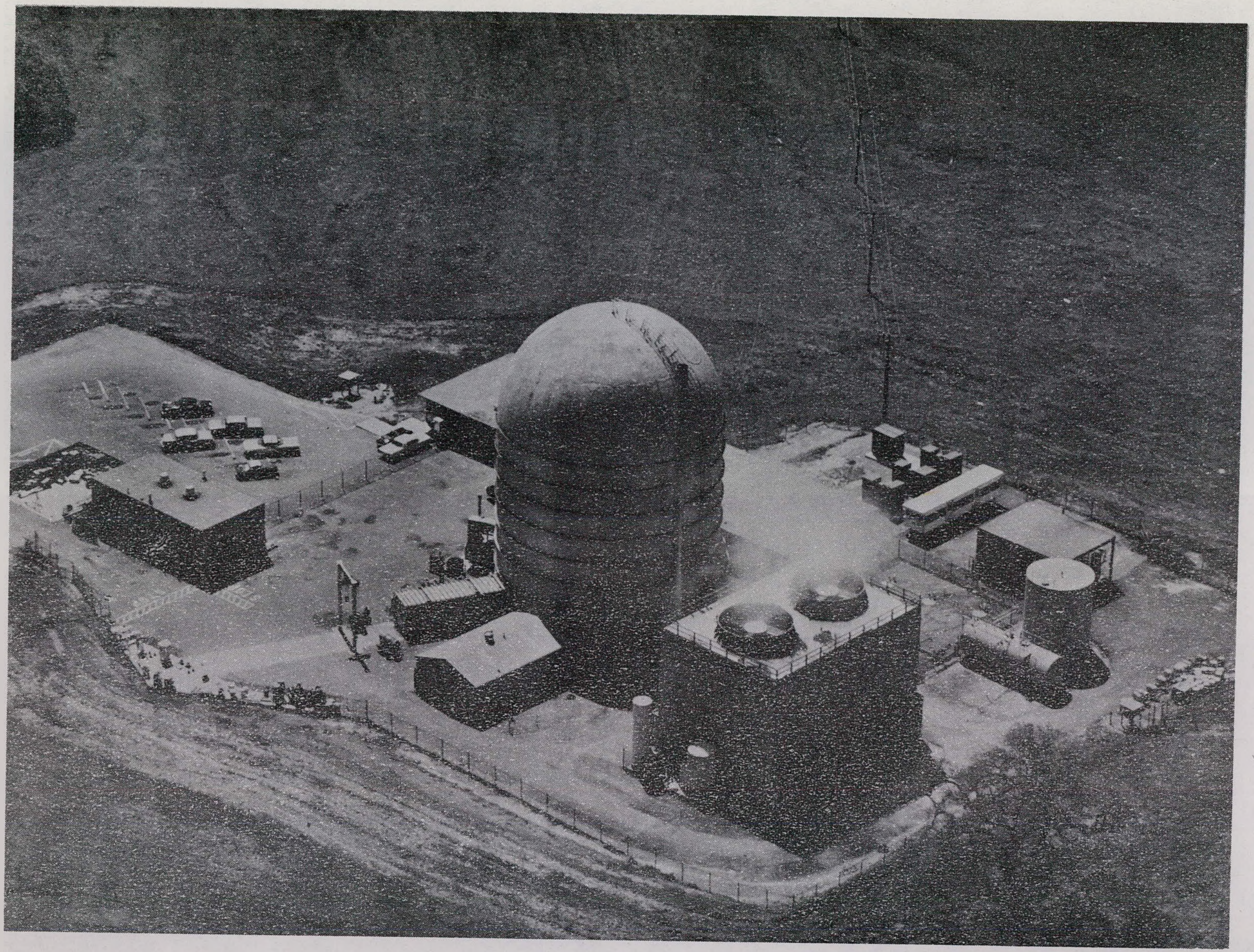

Figure 1.1

and is adaptable to in-core hairpin loops. Bulk irradiations can be accomplished in the pool.

The core for the reactor contains plate type fuel assemblies utilizing fully-enriched uranium alloyed and clad with aluminum. The core loading is limited to 11.5 percent excess reactivity.

The reactor is controlled by six bottom mounted, top entry control rods penetrating the core with a total design control worth of approximately 17 percent. The rods consist of both a poison section and a fuel section.

The reactor is located in an aluminum pressure vessel designed for 150 psig at 200°F. Surrounding the reactor core and within the pressure vessel is a beryllium-aluminum reflector cooled by circulating primary water. Reactor shielding is provided by both the pool water and approximately eight feet of concrete.

The reactor core is forced convection cooled by demineralized water circulated in the primary system. Heat is transferred from the primary to the secondary cooling system through a heat exchanger and dissipated to the atmosphere from the cooling tower. Emergency cooling of the reactor is accomplished by opening the primary system to the reactor pool which serves as a heat sink and allows additional circulation by thermal convection.

The reactor and experiments are instrumented to indicate, record, and control important variables, and automatically shut down the reactor and experiments if assigned operating limits are exceeded.

Department personnel and facilities engaged in research, development, analytical, manufacturing, and safeguards activities provide extensive support for the operation of the reactor.

1.4 Operation of the Facility

The reactor was constructed under AEC Construction Permit No. CPTR-2 as requested by General Electric's application and Preliminary Safeguards and

Hazards Report, GEAP-0984. Initial loading began on December 24, 1958 and Criticality was achieved on December 26, 1958 as authorized by License No. R-48 dated December 22, 1958. Full power operation was authorized by License No. TR-1 initially issued January 7, 1959. The start up program for thorough testing of the reactor was successfully completed on March 14, 1959 with full power operation of 30 megawatts first achieved on February 28, 1959. Operation since that time has been in accordance with License No. TR-1 which, together with the application and Final Summary Safeguards Report, GEAP-2064, has been amended as necessary to authorize the wide range of experimental programs conducted and the improvements which have resulted from operating experience and development programs.

The reactor has operated more than 22,500 hours and produced approximately 26,500 MWD of power.

The safe and efficient operation of the GETR, now in its fifth year of operation, is evident in the four Reports on Operation Safety submitted to the Commission to describe the initial start-up phases and the four Annual Reports on Operating Experience Pertinent to Safety submitted to the Commission since that time. The organization, designs, controls, and evaluations responsible for this performance are described in the remaining sections of this report.

SECTION 2

FACILITY DESCRIPTION

2.1 Introduction

This section describes the GETR facility and the components of the plant which are essential to safe operation.

The design philosophy of the test reactor is based on providing maximum flexibility for experimental irradiation. Proven technology is used to the maximum degree possible in designs of the reactor and its experiments. All designs and materials used in the reactor conform to standards essential for a safe working facility and comply with the intent of the General Industry Safety Orders of the State of California, Division of Industrial Safety.

2.2 General Description of the Test Reactor

The test reactor is a light-water-cooled and moderated reactor using highly enriched uranium fuel. It has operated to date with a normal power of 30MW.

The core is a two-foot diameter matrix with an active length of three feet. The normal core loading contains twenty flat-plate type fuel elements utilizing aluminum clad, fully-enriched uranium-aluminum alloy fuel. Appropriately shaped beryllium and aluminum reflector pieces round the core out to a cylinder. There are six bottom-mounted, top-entry control rods which use separate fuel and poison sections. Provision is made in the core for three experimental through tubes of approximately three inch diameter and sixteen experimental capsule spaces.

The reactor core is housed in a twenty-four inch diameter aluminum vessel which is positioned on the bottom of a nine foot diameter pool. Considerable external experimental space is available in the pool. Typical facilities in the pool include an eight-inch beam port, a two-inch hydraulic shuttle, and thirty-one capsule holders. Provision is made for installation of various sizes and types of irradiation loops in the pool. The pool design allows refueling through the top of the reactor vessel after head removal. Eleven feet of water between the top of the vessel and the surface of the pool provides an effective shield for this purpose. A storage and service canal used for working on irradiated experiments and temporary storage of depleted fuel elements is separated from the pool by a water-tight gate. Figures 2.1 and 2.2 show longitudinal and horizontal cross-sections of the reactor core and typical experimental facilities.

Coolant for the reactor is high-purity demineralized water. The water enters near the top of the pressure vessel, flows downward through the core, and is discharged near the bottom of the vessel. This water then flows through the primary heat exchanger where it is cooled by the secondary water system which dumps its heat to the cooling tower. The system also incorporates adequate emergency cooling capacity and bypass demineralization equipment as described in Sections 2.10.5 and 2.10.1.

Instrumentation for proper startup, operation, and safety of the reactor is provided. Radiation detection and alarm systems are installed for personnel protection.

The reactor, complete primary coolant system, and experimental facilities are housed in a vessel designed to contain radioactive contamination which could result from an accident as severe as the maximum credible accident.

Shielding is provided to limit radiation levels in all areas of continuous occupancy to less than 1 mr/hr during operation at reactor power up to 60 MW.

Normal utilities for the plant site are augmented by a 500,000 gallon water storage tank and a heavy duty diesel-driven emergency power supply.

The reactor is operated from a control room located in the office building adjacent to the containment vessel.

Design parameters for the reactor are listed in Table 2.1.

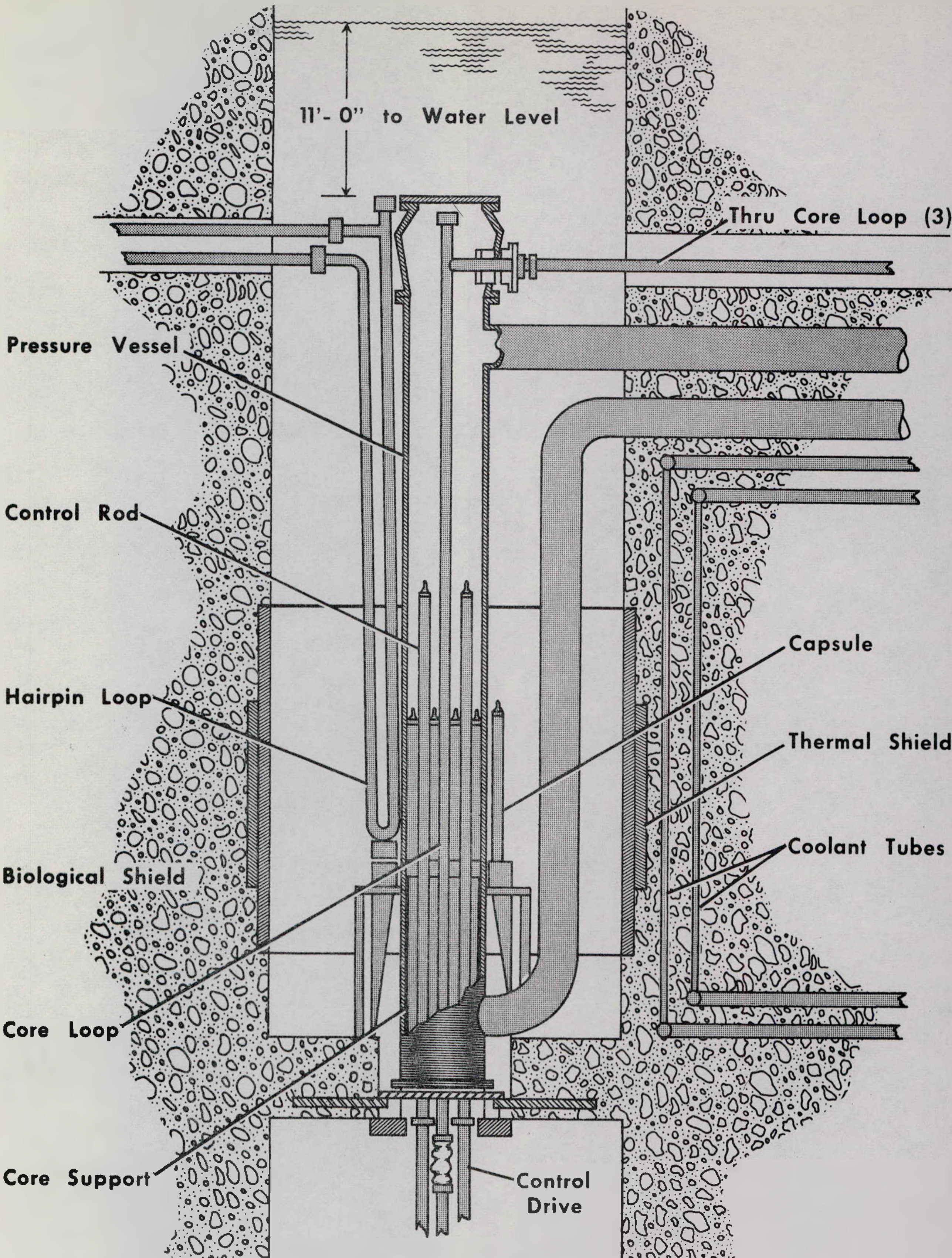

Figure 2.1 REACTOR LONGITUDINAL VIEW

VAL 3338

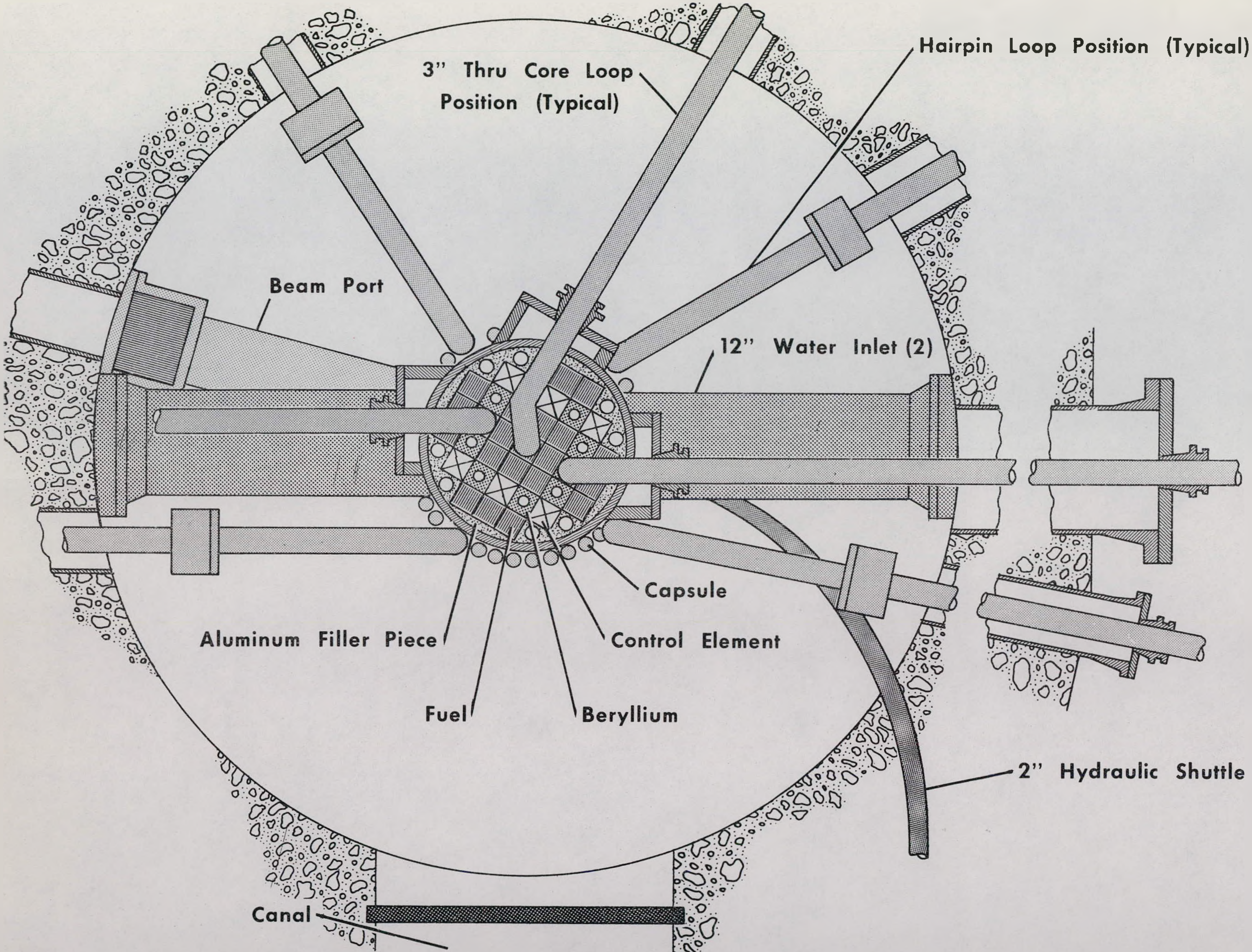

Figure 2.2 REACTOR PLAN VIEW

TABLE 2.1

GETR DESIGN PARAMETERS

1.	Power generation in core	60 MW (overpower trip)
2.	Midplane thermal flux in experiments	Max. 4×10^{14} nv
3.	Burnout safety margin	1.5
4.	Fuel elements MTR - ETR type	3″ x 3″ x 36″ 19 plates Al clad Al U-235 meat
5.	Control	6 fuel-poison rods
6.	Fuel enrichment	93%
7.	Moderator	Light water
8.	Coolant	Light water
9.	Coolant flow	10,000 gpm
10.	Reflector	Beryllium, water
11.	Core size	24″ diameter
12.	System pressure	135 psig
13.	Temperature (reactor outlet)	180°F
14.	Flow through the fuel (20 element core)	7,700 gpm
15.	Flow through components of core (20 element core)	2,300 gpm
16.	Flow velocity in fuel (20 element core)	20 ft/sec
17.	Pressure drop through the reactor	20 psi

2.3 Reactor Core Assembly

The reactor core consists of a matrix of thirty-seven core positions each three inches square. Fuel normally occupies approximately twenty of these positions, as shown in Figure 2.3, but may occupy up to twenty-eight positions. Six positions are used for control rods. Beryllium, aluminum or experimental pieces occupy the matrix positions not used by the control rods or the fuel assemblies and surround the core to round out the cylindrical shape and provide neutron reflection.

2.3.1 Reactor Fuel

The fuel assemblies used in the core are ETR-type, flat-plate, uranium-aluminum assemblies arranged in a core pattern as shown in Figure 2.3. As designed, each such fuel assembly contains nineteen 0.05 inch thick, 2.57 inch wide, and 37 inch long fuel plates. The number of fuel plates may be altered as required by the experimental programs. Each plate consists of a 0.02 inch thick layer of uranium-aluminum alloy with 0.015 inch thick aluminum cladding. The fuel plates are roll swaged onto aluminum side pieces which hold the plates at a spacing that allows 0.110±0.010 inch wide water passages between fuel plates and a metal to water ratio in the fuel assemblies of 0.65 to 5. An aluminum "comb" plate is inserted to maintain fuel plate spacing. A nose piece on the lower end of the fuel assembly seats and aligns the fuel element in the grid plate. The top of the fuel assembly is a square end box equipped with a cross bar for fuel handling. These fuel assemblies are fabricated to written specifications which assure the integrity of the assembly during operations in the reactor. In the future, burnable poison may be used in the fuel assemblies.

A typical fuel assembly contains between 460 and 510 grams of U-235 contained in uranium at a minimum enrichment of 93%, although other fuel loads may be used as the situation dictates. A typical control rod fuel assembly contains 215 grams of U-235 contained in uranium at a minimum enrichment of 93%, although other fuel loads may be used in the future.

Fuel elements are replaced or changed primarily on the basis of reactivity worth. A burnup level of approximately 50% is not exceeded for this type of fuel assembly. New fuel is normally inserted around the periphery and, after partial burnup, is moved to the central region of the core. Each fuel element is usually in the core for three operating cycles.

2.3.2 Beryllium and Aluminum Reflector Pieces

As shown in the core loading plan, Figure 2.3, eight beryllium and four aluminum reflector pieces fill the spaces between the fuel elements and the inside wall of the pressure vessel. They are shaped to conform to the faces of the fuel elements and the wall of the pressure vessel. In addition, beryllium filler pieces, 3" x 3" x 36" long, may be inserted in any of the fuel assembly

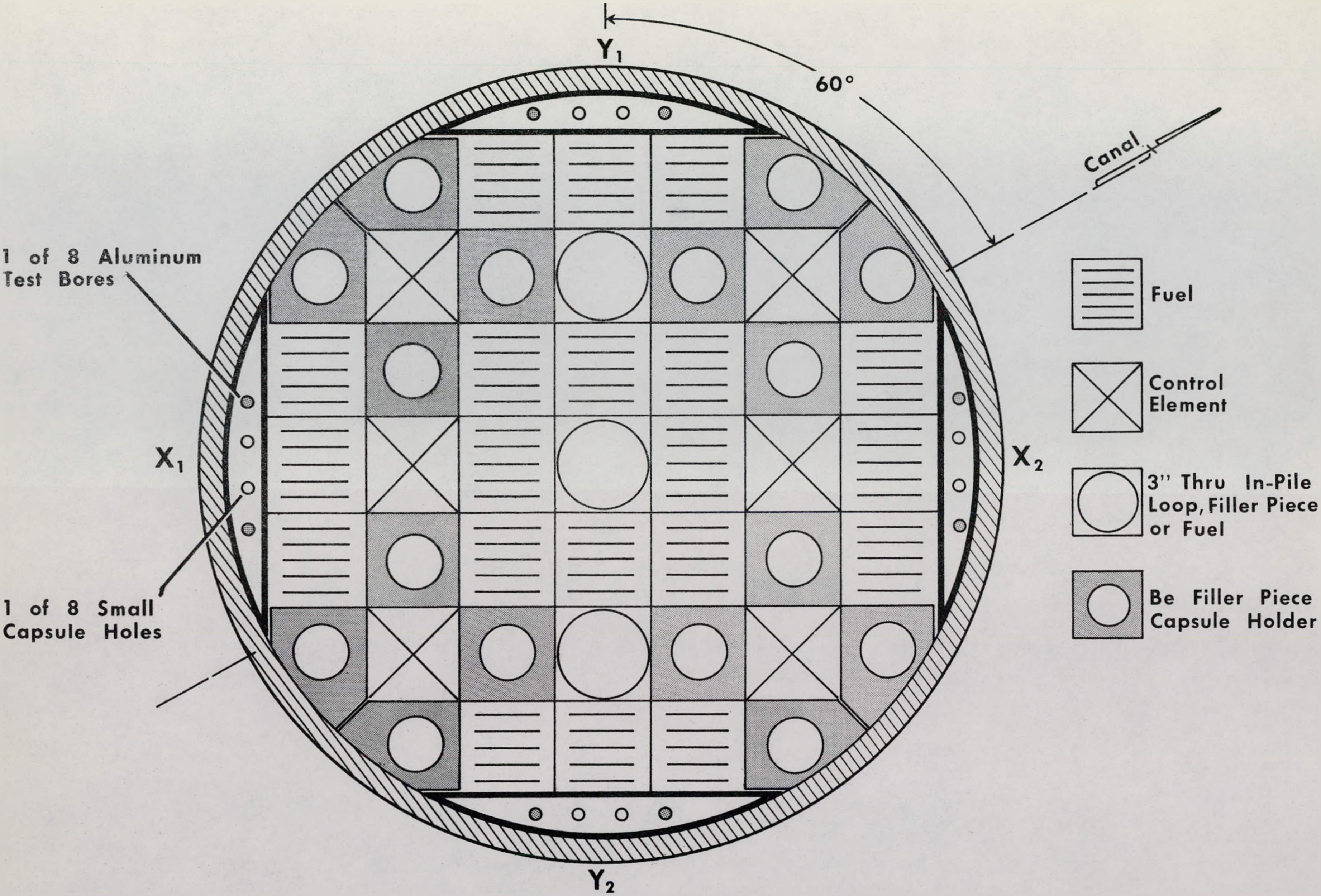

Figure 2.3 TYPICAL REACTOR CORE PLAN

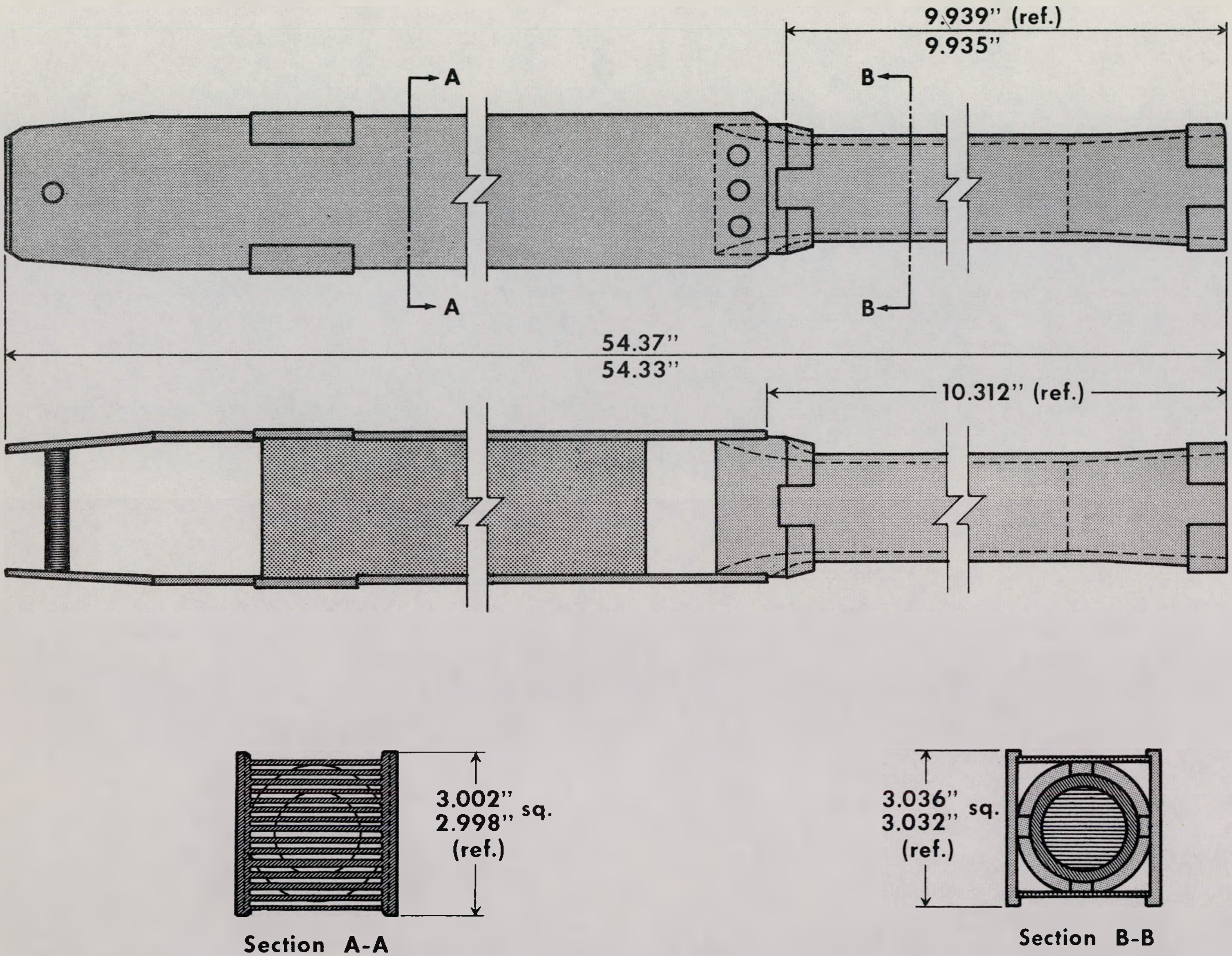

Figure 2.4 FUEL ELEMENT

positions in the core. The aluminum and beryllium reflector and filler pieces are supported on the grid plate. These pieces are cooled by the downward flow of core coolant water. The beryllium filler pieces have openings one and one-half inch in diameter to accommodate capsule experiments although larger or smaller holes may be used in special experiments.

There are sixteen smaller holes located in the aluminum peripheral filler pieces. Eight holes are approximately 0.560 inches in diameter and extend the length of the core. Eight holes are approximately 0.250 inches in diameter and extend eighteen inches down from the top of the core.

2.3.3 Support Structure

A stainless steel cylinder approximately 80 inches high is welded to the pressure vessel bottom head. This core support structure extends to the grid plate. Holes in the sides of the cylinder allow passage of coolant water. This support structure provides both vertical and lateral stability for the core. The control rod guide tubes extend from the core to the bottom head of the reactor inside the support cylinder. The support structure is securely welded to the bottom head of the pressure vessel. The grid plate rests directly on top of the support structure.

2.3.4 Grid Plate

The grid plate is fabricated from 304 stainless steel. The assembly is made up of two parallel flat plates held approximately nine inches apart by means of welded spacer bars. Matched circular holes in the 2.25 inch thick upper plate and the 0.5 inch thick lower plate provide support and alignment for the fuel elements and the core filler pieces. The upper and lower plates contain openings for the penetration and lateral alignment of the control rods and the experimental through loops.

2.4 Reactor Vessel

The reactor pressure vessel is a 24-inch diameter by approximately 20-foot long aluminum cylinder. It is equipped with a 2-foot long stainless steel top spool extension, and stainless steel top and bottom flat heads. The vessel houses the core and incore experimental facilities.

2.4.1 Design

The reactor's aluminum pressure vessel design is based on a service limit of 150 psig at 200°F. The maximum coolant operating condition is 135 psig at 160°F. The design and construction requirements are in conformity with the ASME Boiler and Pressure Vessel Code and the vessel bears the official code stamp. In areas where the code does not provide complete guidance in design, a thorough stress analysis has been made to assure that stresses are within those allowed by the Code.

The temperature gradient through the pressure vessel wall results in thermal stresses. At the original design power level of 30 MW, the temperature induced stress is 3300 psi and the pressure stress is 2400 psi, for a combined maximum stress of 5700 psi. The allowable ASME Code stress at 200°F is 6200 psi.

This vessel was constructed under Nuclear Code case 1234 which permits stresses up to $1\frac{1}{2}$ times the allowable code values when both thermal and pressure stresses are calculated. Under Nuclear Code case 1270N the allowable combined stress for 5052 aluminum at 200°F is 9300 psi. At 60 MW the thermal stress will be 6400 psi. With a pressure stress of 2400 psi this makes a total stress of 8800 psi. Therefore, the pressure vessel is adequate for 60 MW operation.

The effects of circumferential and axial variation of heating in the pressure vessel wall have been investigated. Because of the symmetrical core, the pressure vessel wall heating is symmetrical about the axis and stresses due to circumferential variation of heating in the cylinder are negligible. The axial variation of the pressure vessel wall heating does not increase the maximum thermal stresses generated by the radial temperature gradient. The axial variation of heat generation has its greatest effect in the regions at the top and bottom of the reactor core. However, in these regions the heat generation at the wall has decreased to less than 20% of the maximum value at the core centerline. The combined total of the thermal stresses produced by the axial and radial temperature distribution at the core inlet and outlet regions is considerably less than the stresses generated by the radial temperature distribution at the core centerline.

2.4.2 Vessel Components

The reactor vessel shell is a straight-walled cylinder 24-inches in diameter by approximately 20 feet long. The wall of the vessel is 0.75 inches thick. Flanges are welded to the ends of the cylinder for mounting of the spool extension and the bottom head. Two 12-inch diameter coolant inlet nozzles are located approximately 1 foot below the top flange and 180° apart. Two 12-inch diameter outlet nozzles are located approximately 2 feet above the bottom flange and 180° apart. A siphon-breaking device approximately 6 feet above the core prevents accidental draining of the vessel in the event of a coolant-pipe failure.

Seven test specimens of 52-S aluminum are being irradiated in the high flux regions near the core. These will be tested for ductility and tensile strength at different integrated radiation exposures to predict the effect of neutrons and gamma flux on the strength of the pressure vessel. One unirradiated specimen has been retained to provide reference data on mechanical properties.

A flanged stainless steel extension spool, two feet in diameter, two feet long, and 0.50 inches in wall thickness, is bolted to the top flange of the aluminum vessel shell. This spool contains three 9-inch diameter flanged nozzles for penetration of in-core experimental loops and eight 3-1/8-inch flanged nozzles for in-core capsule lead penetrations. The spool piece vent line permits venting of the pressure vessel.

A flow deflector has been installed in the pressure vessel below the spool piece to reduce the water turbulence in the pressure vessel. The deflector is a 20-inch stainless steel cylinder with a 3/8-inch wall. The inlet coolant flow to the reactor impinges on the outside of the deflector cylinder and is directed through the annular space in the direction of the core before it is released to the main vessel cavity. In-core lead tubes may be attached to the deflector for support.

The top head is a flat, circular, stainless steel plate. Captured bolts are used to fasten the top head to the upper flange of the spool extension. The top head is removable for refueling of the reactor and for servicing of experiments. During shutdown it can be replaced with a viewing head for observing hydraulic phenomenon inside the vessel. The presently used viewing head has 4 window-like openings approximately 4 inches in diameter. The transparent material used is an acrylic plastic.

The bottom head serves as the lower closure and supports the reactor and its internals. The outer edge of the plate is fastened to a ring set in the foundation of the pool. The core support cylinder is welded to, and supported on, the inner face of the bottom head. Penetration nozzles are provided through the bottom head for three experimental facilities and six control rods. One additional penetration nozzle, approximately 1-inch in diameter, is available for lowering water level in the pressure vessel, when required.

2.5 Control Rods and Drives

The GETR is controlled by six control rods located in the core as shown in Figure 2.3. The control rods are actuated individually by six control rod drive mechanisms mounted on the bottom of the reactor pressure vessel. The control rods move in an upward direction to increase core reactivity and are scrammed by releasing each rod from its mechanism thereby allowing it to fall into the core. Gravity and the force of the primary coolant flow are the scram forces on the rods. Each control rod consists of a poison section, a fuel follower section, and a shock absorber section. The over-all length of the entire rod (composed of these three sections) is about 11 ft. 3 inches.

There are two basic types of control rods. Mark I and Mark II. The Mark I and Mark II are quite similar and either type can be used in the core at one time. The rods and drive mechanisms are described in Section 2.5.1 and 2.5.2 and the performance characteristics are given in Section 2.6.

2.5.1 Control Rods

The Mark I control rod was used in the GETR for the first four and one-half years. This rod (as well as the Mark II design) consists of three sections, i.e. poison, fuel and shock absorber, which latch into a single integral rod approximately 11 ft. 3 inches in length.

The upper most part of the rod is the poison section. The poison sections are box shaped, about 2.5 inches square and 38 inches in length, with 0.22 inch thick walls. The walls of the box are fabricated of type 304 stainless steel containing 1% boron enriched to 92% in boron-10. The plates making up the poison section are welded together. Rollers located on all four sides both top and bottom center the poison piece in the guide tube. These rollers also reduce sliding friction. The rollers on adjacent sides (half of the rollers) are spring mounted and the opposite rollers are fixed rollers. The spring mounted rollers are slightly depressed when the rod is in the guide tube, thereby making the rod fit snugly in the guide tube. The poison section has a latch which couples with the fuel follower section.

The middle portion of the control rod is the fuel follower section. This section is a 2.5 inch square assembly of 14 fuel plates loaded to about 46% of the regular fuel element loading. The fuel follower sections are fabricated to the same specifications as the core fuel elements. The fuel follower section has latches which couple to the poison section above it and to the shock absorber section below it. There are no rollers on the fuel follower sections.

The shock absorber section is the lower most part of the control rod. This part provides the mechanical linkage between the rod drive mechanism and the control rod. The shock absorber section also is designed to decelerate the rod (scrammed rod) as it enters the receiver section of the drive mechanism. The shock absorber section is an assembly approximately 56-$\frac{1}{2}$ inches long consisting of a 2.5 inch square aluminum box and a cylindrical stainless steel shock tube. The general configuration of this assembly can be seen in Figure 2.5.1 (Mark II Control Rod). Slots in the shock section provide a means for coolant to leave the control rod. The shock absorber section is equipped with rollers (both fixed and spring mounted) similar to those on the poison piece. There is a latch device at the upper end of the shock absorber section which couples this piece to the fuel follower section. The control rod drive mechanism engages the lower end of the shock section (see Figures 2.5.1 and 2.5).

The control rod guide tubes are not really a part of the control rods but are described here to maintain continuity. The guide tube is a square tube extending from the reactor vessel bottom head, through the core region, and terminating about 40 inches above the core. The guide tubes provide alignment and support for the control rods along their entire length. The rollers on both the poison and shock absorber sections bear against the guide tubes. Coolant flow enters the top end of the guide tube, flows down through the control rod and exits through slots in the lower part of the guide tubes below the reactor core. Support brackets and hold down rods have been used to provide support to the top of the six rod guide tubes although use of such devices is optional. The guide tubes are removable and either Mark I or Mark II rods can be used in them.

The Mark II control rod is a redesigned version of the Mark I control rod. The two types are quite similar, employing the same design principles, and the two types have been used concurrently. The differences involve the construction of the poison piece, the type of latch used on all sections, and minor changes in the rollers and other small parts. The attachment to the drive mechanism is identical for both rod types.

The Mark II poison section, like the Mark I is a square box assembly 2.5 inches square and about 38 inches long. A view of the entire Mark II control rod is shown in Figure 2.5.1. The poison piece is composed of boron stainless steel (1% enriched boron) pieces attached to the outside of a stainless steel box. This box is a rigid all welded assembly designed to provide structural support for the poison piece. The boron stainless steel is not a load bearing member of the rod. The boron stainless pieces are in turn covered with a thin (0.015 inch) stainless steel "skin". The control material is therefore fully contained with a sandwich type construction. The Mark II poison pieces have been used successfully in the reactor since May, 1963.

The Mark II design uses a latch similar to those used in the ETR control rods. (The two reactors have the same type of control rods). The latch is a spring finger type which latches into square holes in the mating piece. Note in Figure 2.5.1 that latch fingers are on the poison and shock sections and the mating holes are on the fuel section. The rod sections are latched by butting one piece against the other in the guide tube and unlatched by raising the rod partially out of the guide tube and rotating it 90° to disengage the latch finger. The Mark II latch has been very successful.

2.5.2 <u>Control Rod Drive Mechanism</u>

The six control rods are actuated from below by individual control rod drives. These devices are identical in principle, and nearly identical to the ETR drive mechanisms which, in turn, is a modification of the ORNL Reasearch Reactor drive mechanism. Experience gained in the testing and operation of this design has resulted in improvements that have increased its reliability. The drive mechanism consists of a drive motor, position indicator, shock absorber, bottom position indicator, ball coupling, lead screw assembly, scram magnet assembly, limit switches, and mechanical stops. These components are located in the sub-pile room and the control rod drive access room. The control rod drive access room may be entered for a limited time during reactor operation for inspection and minor adjustments to control rod drives. The drive mechanisms are attached to the reactor bottom head at the shock absorber section.

The control rod is positioned by a non-rotating lead screw and a motor-driven nut. A ball coupling at the end of the lead screw assembly connects the rod to the drive as shown in Figure 2.5. The position of the drive is indicated by a selsyn on the motor shaft which transmits a signal to an indicator in the reactor control room. The drive motion is limited by use of limit switches and mechanical stops.

Scram operation is accomplished by de-energizing the release magnet. The resultant axial motion of a cam permits the coupling balls to retract and thereby free the control rod from the drive.

To return the drive to normal operation, the ball coupling is reset by driving the mechanism to the lower limit of its travel. This action causes the reset pin to strike the spring retainer, causing the armature to engage the release magnet as the scram spring is compressed (see Figure 2.6). Since the control rod is also at its lowest position, the ball's outward movement engages the control rod. In case of malfunction when the control rod is not in its lowest position, the outward movement of the balls would not engage the control rod. This situation is indicated by the "rod seated" lights which indicate when a control rod is fully inserted and the poison section thus positioned entirely in the core. Referring back to the scram operation when the coupling balls retract and allow the control rod to drop, the drive tube is left extended up inside the control rod at a distance corresponding to the amount of rod

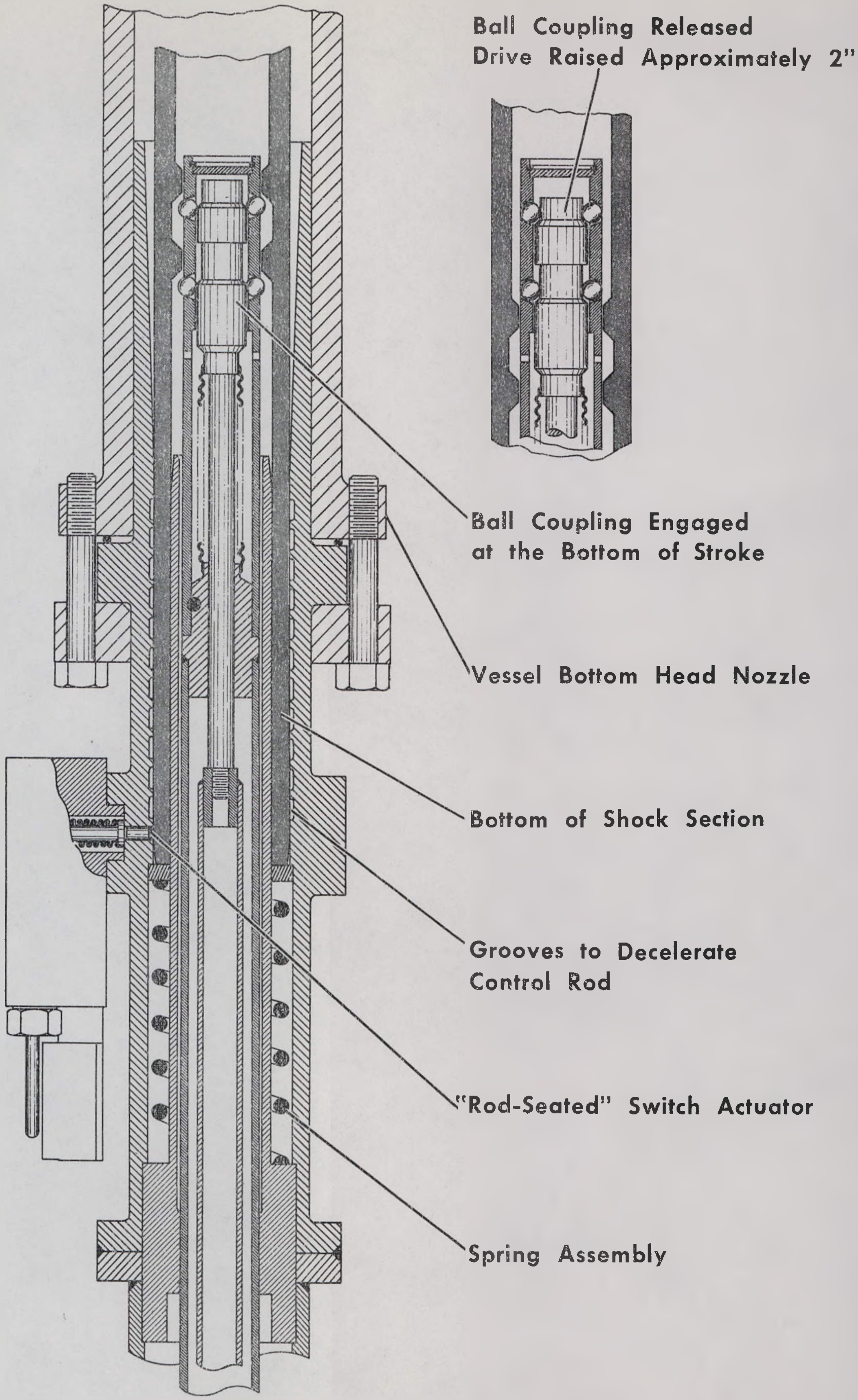

Figure 2.5 CONTROL ROD DRIVE BALL COUPLING

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Figure 2.5.1. GENERAL ELECTRIC TEST REACTOR CONTROL ROD ASSEMBLY MARK II

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withdrawal at the time of the scram. Since the coupling ball will remain retracted until the drive reaches the lower limit of its travel, force will not be transmitted between the drive and the control rod until both are at the lower limit of travel.

The upper part of the drive mechanism provides the deceleration for the control rod after it has been released during scram. The control rod shock section passes through the orifice section before coming to rest on the spring assembly. This spring assembly supports the control rod when it is not attached to the drive mechanism.

In the lead-screw assembly, the motor-driven worm gear causes the worm wheel to rotate. This worm wheel is threaded to receive the drive screw. The drive screw is caused to move axially by a key which rides in a slot of the screw. Action of the worm wheel is one to produce a non-rotating axial displacement of the lead screw to which is attached the plunger guide assembly. The ball coupling is at the upper end of the plunger guide assembly.

Operation of the scram magnet assembly causes the cam surface of the ball coupling assembly to be properly positioned for either latching or releasing of the control rod. Energizing the magnet coil holds the scram spring compressed; opening the circuit for de-energizing the magnet causes the scram spring to be released. This release will cause the coupling balls to retract. A switch indicates if the magnet has relatched after a scram. Switches are used to limit the upper and lower travel of the drive. If a limit switch fails, mechanical stops will limit control rod travel.

2.6 Reactor Performance

Nuclear physics, heat transfer, and fluid flow are discussed in this section. Table 2.2 summarizes the important core parameters and compares original design values with those measured during operation.

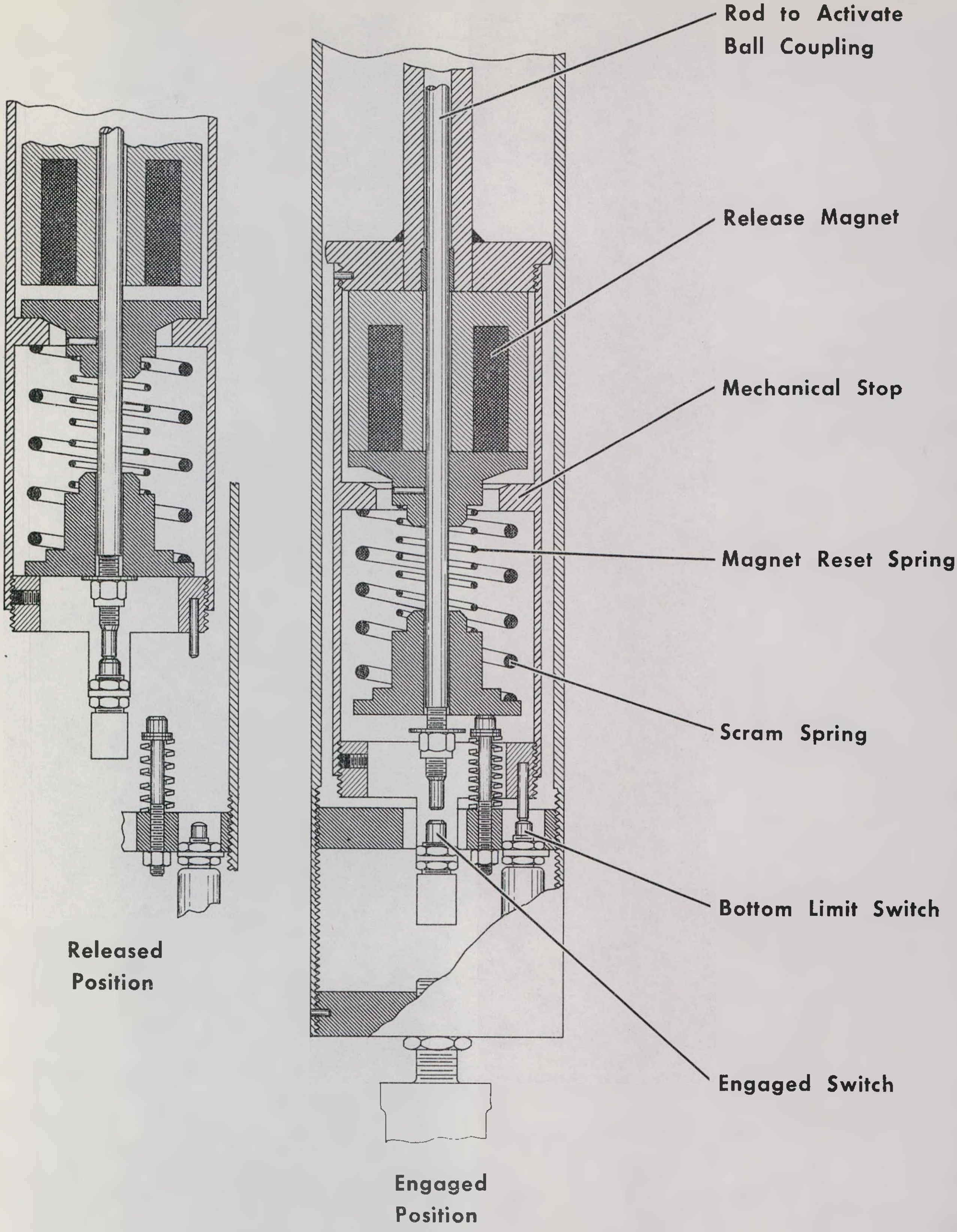


Figure 2.6 SCRAM SPRING AND RELEASE MAGNET

TABLE 2.2

GETR Physics Parameters at 30 MW Power

1.	Vertical peak-to-average power in regular fuel elements (integrated over a cycle)	1.60
2.	Average thermal flux in fuel over 36 inches	
	a. Beginning of cycle	1.1×10^{14} nv
	b. Average over cycle	1.15×10^{14} nv
3.	Average void coefficient in fuel sections	
	a. Measured during initial critical	-3×10^{-3} $\frac{\Delta k}{k}$/% void in water
	b. Design value (start of cycle, 25°C)	-1.3×10^{-3} $\frac{\Delta k}{k}$/% void in water
4.	Temperature coefficient in fuel section	
	a. Measured during reactor operation	-1.8×10^{-4} $\frac{\Delta k}{k}$/°C
	b. Design value (start of cycle)	-2.1×10^{-4} $\frac{\Delta k}{k}$/oC
5.	Void coefficient at core hot spot	
	a. Calculated from measurements	-3.9×10^{-7} $\frac{\Delta k}{k}/cm^3$/% void in water
	b. Design value	-1.3×10^{-7} $\frac{\Delta k}{k}/cm^3$/% void in water
6.	Neutron lifetime	5.5×10^{-5} seconds
7.	Maximum available excess Δk effective at reactor startup	11.5% Δk/k
8.	Cycle time based on 65% operating effiency for 35 day cycle	22.7 days on line
9.	Time available for restart after shutdown with 8.7% Δk/k available excess at startup	
	Start of cycle (Xe equilibrium)	1.2 hours
	10 days	0.7 hours
	15 days	0.3 hours
	20 days	0.0 hours
10.	Effect of flooding beam port	
	a. Measured during initial critical	$+4 \times 10^{-4}$ $\frac{\Delta k}{k}$
	b. Design value	$+18 \times 10^{-4}$ $\frac{\Delta k}{k}$
11.	Control system strength	
	a. Mark I Design value	17% $\frac{\Delta k}{k}$
	b. Mark II Design value	16.6% $\frac{\Delta k}{k}$

2.6.1 Nuclear Physics and Gamma Heat

The reactor has the unique feature that part of the reflector is outside of the pressure vessel. The complete assembly of the reactor core is shown in Figure 2.3 where it can be seen that the pressure vessel is close to the core. Considerable neutron leakage from the core occurs and thermal flux peaking takes place in the water reflector outside the pressure vessel. This reflector region outside the pressure vessel can be used for irradiations to make advantage of the unperturbed average thermal flux of approximately 7×10^{13} nv at 30 MW power level. In addition to this experimental region in the pool, positions are available for experiments within the reactor core where the maximum mid-plane unperturbed thermal flux is approximately 3×10^{14} nv at 30 MW power level.

Since the maximum available excess Δk effective at startup for the fuel is 11.5% $\Delta k/k$ and the Mark I control rod strength is 17.2% $\Delta k/k$ and 16.6 for Mark II, the shutdown margin is 5.7% and 5.1% $\Delta k/k$ respectively. When the control rods are withdrawn 12.5 inches and the reactor is just critical, the highest value control rod is worth 4.3% $\Delta k/k$. This leaves a worth between 12.9% and 12.3% $\Delta k/k$ in the remaining five rods, which is sufficient to shutdown the reactor.

The physics group performs two-dimensional, three-group, neutron diffusion theory calculations to provide flux level values at each experiment position for each reactor cycle. The calculated flux level values are confirmed by measurement at frequent intervals. The perturbed neutron flux in an experiment is calculated by two-dimensional calculations with the experiment in its location relative to the reactor core. Typical detectors used for low power flux measurements are cobalt wires, 0.03 inch in diameter. The detectors used for integrated flux measurements are 0.01% cobalt in aluminum wire, 0.03 inch in diameter.

Neutron spectrum measurements have been made in core locations with sulphur, magnesium, and aluminum threshold detectors.

For physics calculations of the reactor and experiments, basic neutron cross sections are obtained from existing data. These microscopic cross sections are converted to three group macroscopic values, after hardening for temperature and absorption, and weighting the flux. Perturbation factors, as presented by the experimenter's design calculations, are checked by use of a P-3 approximation to transport theory. Interaction effects of experiments and the reactor are determined by two-dimensional calculations. With the exception of loop facility tubes within the core, the effect of experiments on the reactor is generally small. Even the higher absorption materials have little reactivity effect when positioned in reflector pool facilities. Isolation of experiments with sufficient thickness of water greatly reduces the interaction flux perturbation between experiments.

The reactor was operated without experiments during initial startup phases. The core was composed of 20 fresh fuel elements with a total fuel loading of 8.5 kg of uranium-235. This fuel loading would have resulted in an initial excess reactivity greater than 11.5%. Therefore, special high cross section filler plugs were placed in the through holes and capsule locations in order to reduce the excess reactivity. This eliminated the need of procuring special fuel elements with uranium content sufficiently reduced to compensate for the missing poisoning effect of the experiments. These special plugs contained substantially more stainless steel than normal plugs and served to poison the core to an initial excess reactivity of 11.5%.

During the critical testing phase of initial startup operations, a series of flux measurements were made for a variety of core conditions by irradiation of Cu-Mn flux wires. These conditions corresponded roughly to the conditions expected to prevail at the beginning and end of a core cycle as well as intermediate conditions. The detectors used for the flux measurements were 0.030" diameter, 80% Mn - 20% Cu wires. These wires were strung vertically in several locations through the core extending over the 36-inch active length. After irradiation, the wires were removed from the core and counted on gamma-sensitive scintillation channels. Each of the flux tranverses was integrated and the average saturated activity along the length of the wire was determined. The saturated activity of the detector was related to the incident thermal flux by activating bare and cadmium-covered gold foils at points where the Mn-Cu activities were known and then counting the gold foils in calibrated proportional counters. Using the relative flux distributions obtained from the wires and applying the activity to flux relationship, the thermal power for each of the flux runs was determined. All of the measurements were than normalized as fluxes at 30 MW.

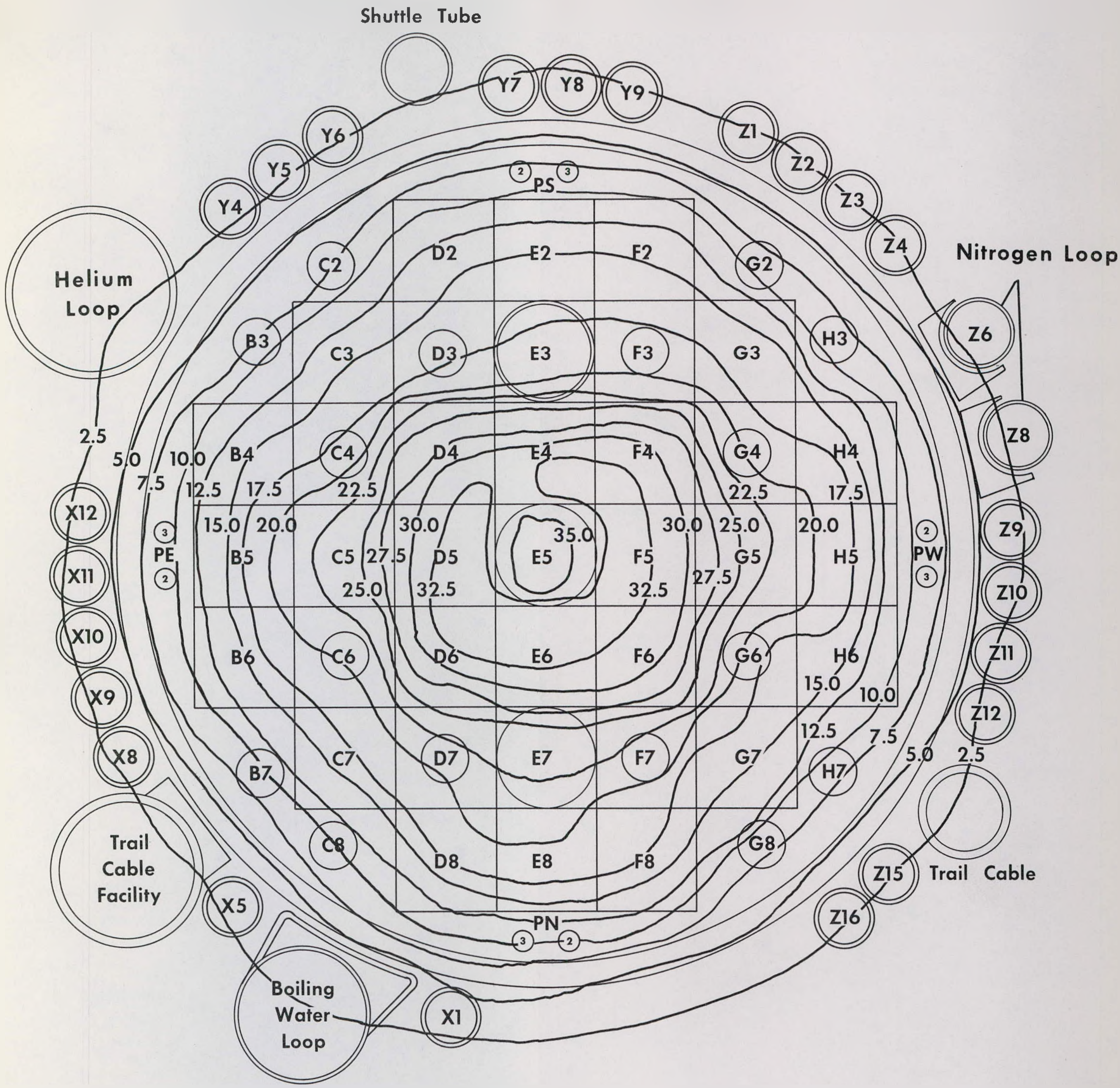
Thermal power was also determined by a series of heat balance measurements in order to verify the accuracy of the flux measurement values. Thermal power, as calculated from the flux distributions, was within 7% of the value determined by the heat balance. The estimated uncertainty prescribed by all the fluxes reported herein is ~15%.

The reactor temperature coefficient was estimated from the control rod movements required to hold the power level constant while the primary coolant temperature was varied by changing the secondary cooling water flow rate. The control rod movement was evaluated using control rod calibration curves to determine the reactivity change. A value of -1.8×10^{-4} $\Delta k/k$ per °C was measured by this method. This compares with a value of -2.1×10^{-4} $\Delta k/k$ per °C which was calculated during reactor design.

The reactor void coefficient was determined by comparing the difference in control rod position required to bring the reactor critical with known voids in the core and to bring the reactor critical without voids in the core. Criticality tests were run in three ways for this evaluation: normal operating condition, solid plastic strips in the flow channels of fuel assemblies, and known voids in the plastic strips in the flow channels of the fuel assemblies. There was no measurable difference in the control rod positions at critical for the case of water in the channels and the case of the solid plastic strips in the channels. There was a change in the critical control rod position when the plastic strips with known voids were in the fuel assemblies. The change observed in the control rod position was evaluated using the control rod calibration curves to determine the reactivity change. A value of -3×10^{-3} $\Delta k/k$ per % void in water was measured by this method. This average void coefficient in the fuel sections compares with -1.3×10^{-3} $\Delta k/k$ per % void in water from design calculations.

Typical measured flux distributions are shown in Figures 2.7, 2.8, and 2.9. The maximum thermal flux (0 to 0.17 ev) of 2×10^{14} nv shown in Figure 2.9 applies to the load condition experienced for reactor cycle 32 averaged over the cycle and averaged over the core height. The reactor contained a typical experimental loading during this cycle. Considerable flux variations are experienced as experimental loadings change. Figure 2.8 shows the intermediate flux (0.17 ev - 0.18 Mev) and Figure 2.7 shows the fast flux (0.18 to 10 Mev). Figures 2.10 and 2.11 show the results of one-dimensional studies of the flux in the reactor made for a typical reactor cycle, No. 32, at 30 MW power. The Y axis is through the experimental loop (PWL) and the X axis is at 90° to the Y axis. At a 50 MW power level the flux would be 5/3 that shown in these figures. At the trip setting of 60 MW the flux would be double that shown in these figures. It is planned to use a burnable poison in the core at a later date. The objective is to provide a near constant flux for experiments by preventing the normal shift in flux over the reactor cycle otherwise caused by the control rod movement. It is presently planned to alloy the burnable poison with the fuel, although it may be placed in the fuel element side plates.

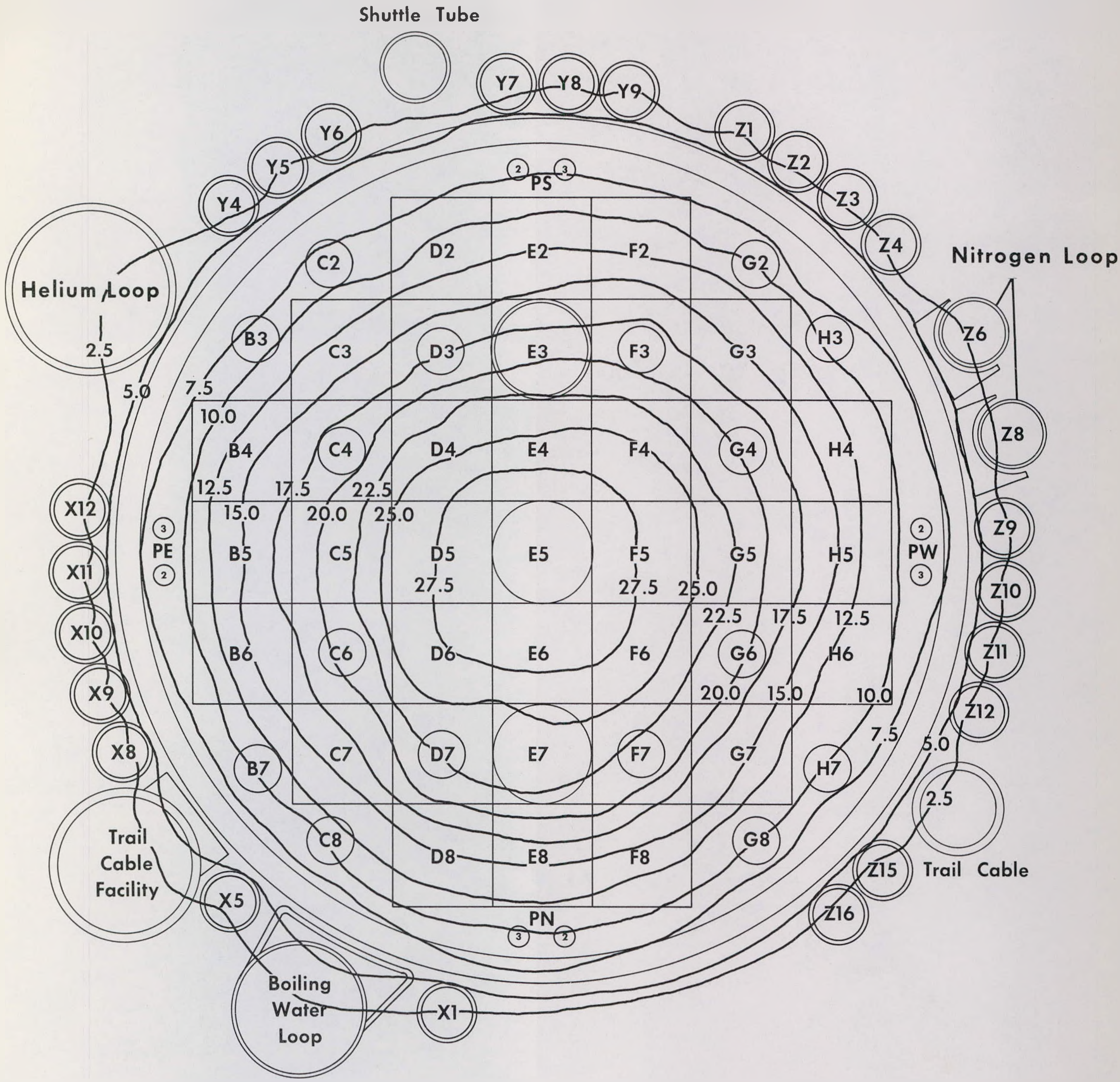
The temperature coefficient, void coefficient, and neutron lifetime, which affect safety of the reactor, are not significantly altered by the experiments as more fully described in Section 5.



.18 MEV to - ∞

Fluxes × 10^{13}

Figure 2.7. FAST FLUX CONTOUR



.17 EV to .18 MEV

Fluxes × 10^{13}

Figure 2.8. INTERMEDIATE FLUX CONTOURS

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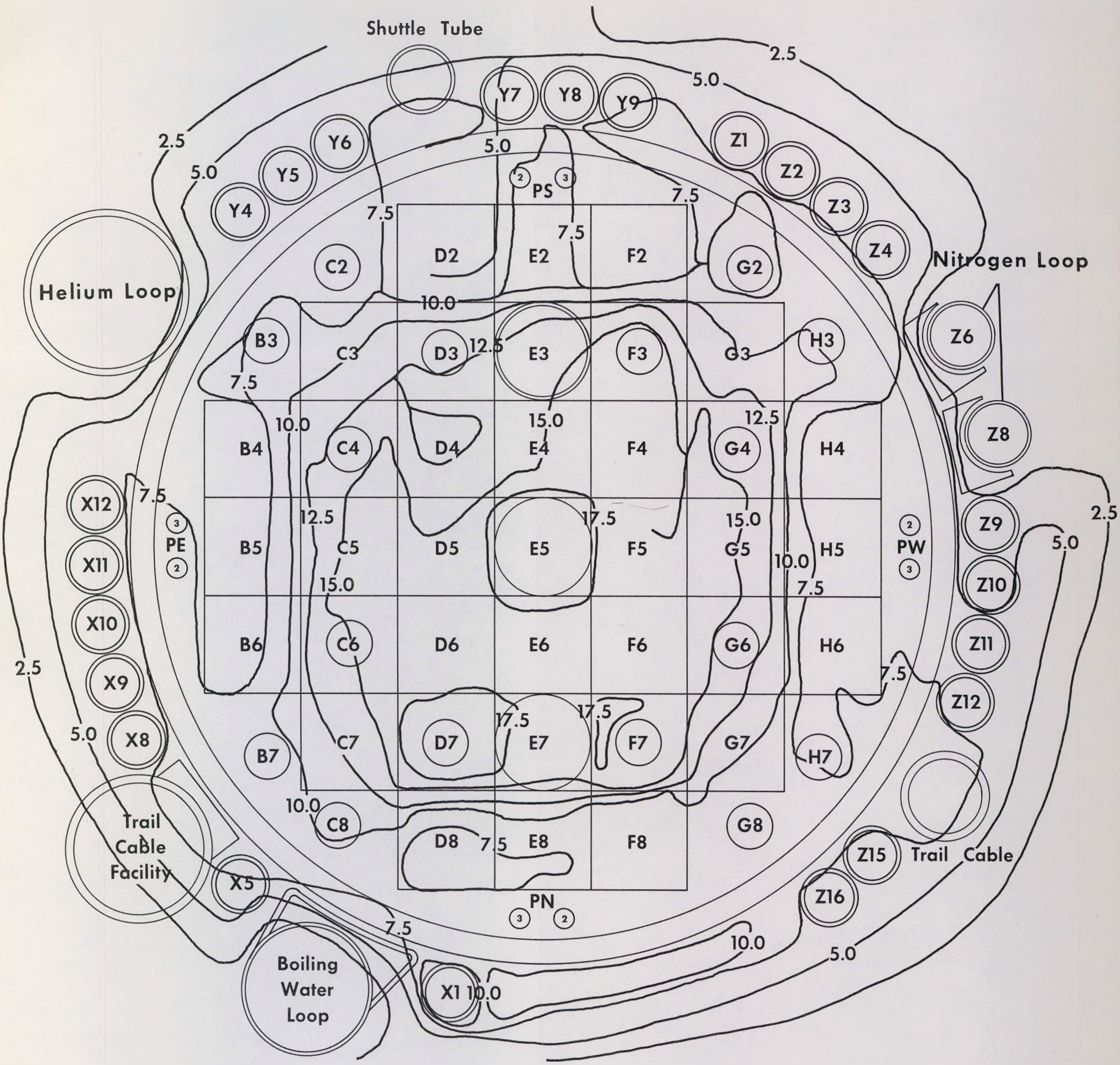

0 to .17 EV

Fluxes × 10^{13}

Figure 2.9. THERMAL FLUX CONTOURS

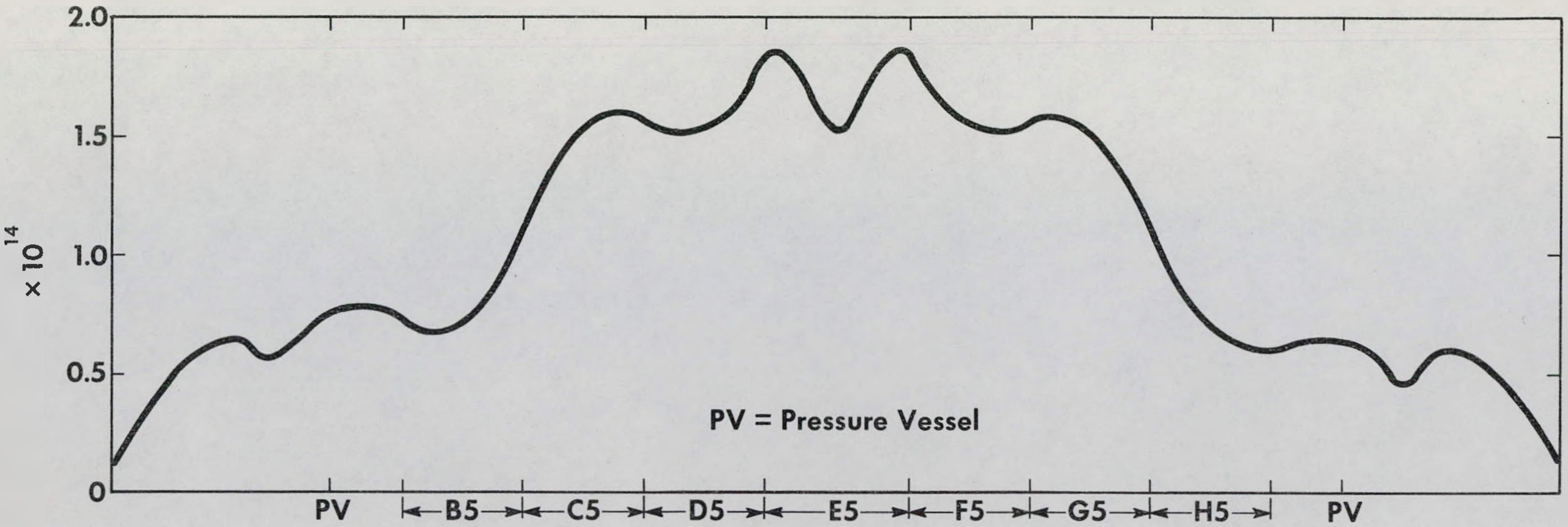

Figure 2.10 GETR FLUX PROFILE - CYCLE 32 (X - AXIS, THERMAL FLUX)

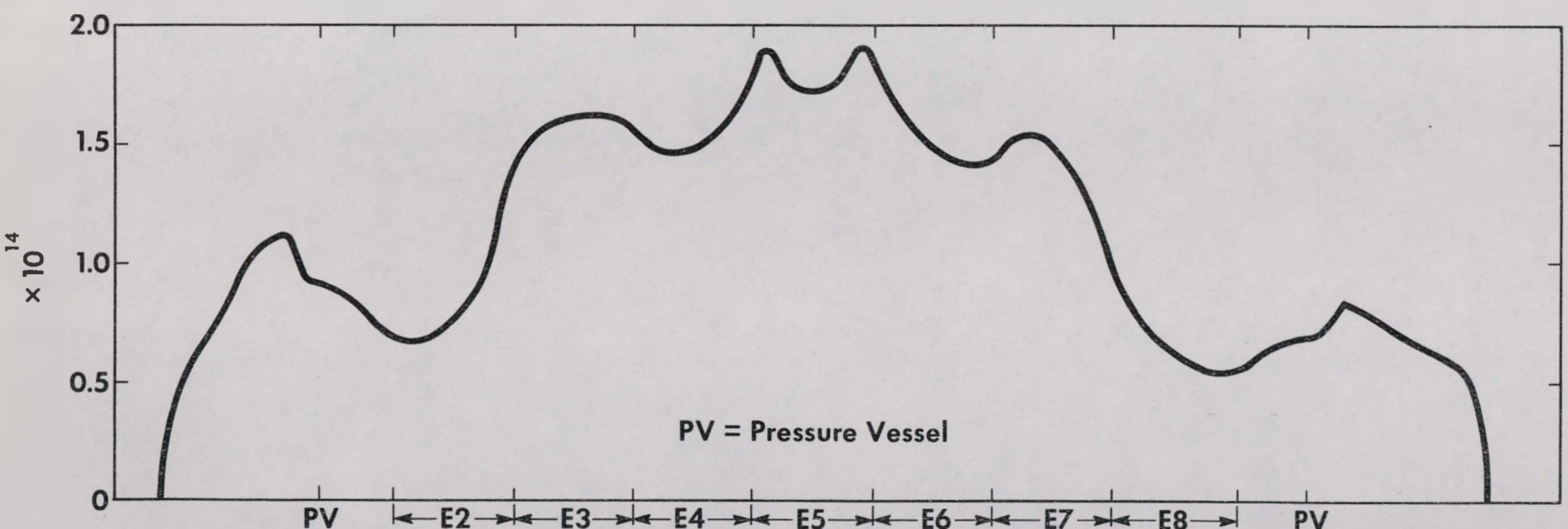

Figure 2.11 GETR FLUX PROFILE - CYCLE 32 (Y - AXIS, THERMAL FLUX)

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In the characteristic manner, the equilibrium xenon concentration increases at a rapid rate after a scram unless the reactor is restarted in a short interval of time. Otherwise, startup must be delayed until the poisoning effect is reduced by xenon decay or the reactor is refueled. Figure 2.12 shows the time available for the restart after a scram as a function of operating time before the scram. No xenon instabilities have been noted and because of the small core size none are expected.

The reactor and the experimental facilities designs take into account the effect of gamma ray heating. In particular, the reactor pressure vessel and the in-core loops are designed for operating pressures plus temperature gradients produced by gamma ray heating. The results of the stress analysis on the pressure vessel presented in Section 2.4 indicate the vessel is adequately designed for all conditions of operation.

Gamma ray heating in the core depends upon the power distribution within the core as shown in Figure 2.13. A number of gamma heating measurements have been made in the reactor during operation. During 30 MW operation gamma heating in reactor position E-5 was measured by a calorimeter to be 5.0 watts per gram.

A heat transfer correlation, made for the same core position, gave a 5.3 watts per gram value. A heat balance of the loop experiment in position E-3 has yielded a value of 4.5 watts per gram due to gamma heat. A calorimeter in the Trail Cable, which is located in positions Z-13 and Z-14, has measured a gamma heat value of 0.9 watts per gram. The above measured values are in close agreement with calculated gamma heat flux values.

2.6.2 Heat Transfer Fluid Flow

The reactor core is cooled by the downward flow of demineralized water. The reactor is designed to remove up to 60 megawatts of thermal energy, (overpower trip) with a coolant flow of 10,000 gpm, at an inlet temperature of approximately 140°F, and with an average rise of 40°F in the core.

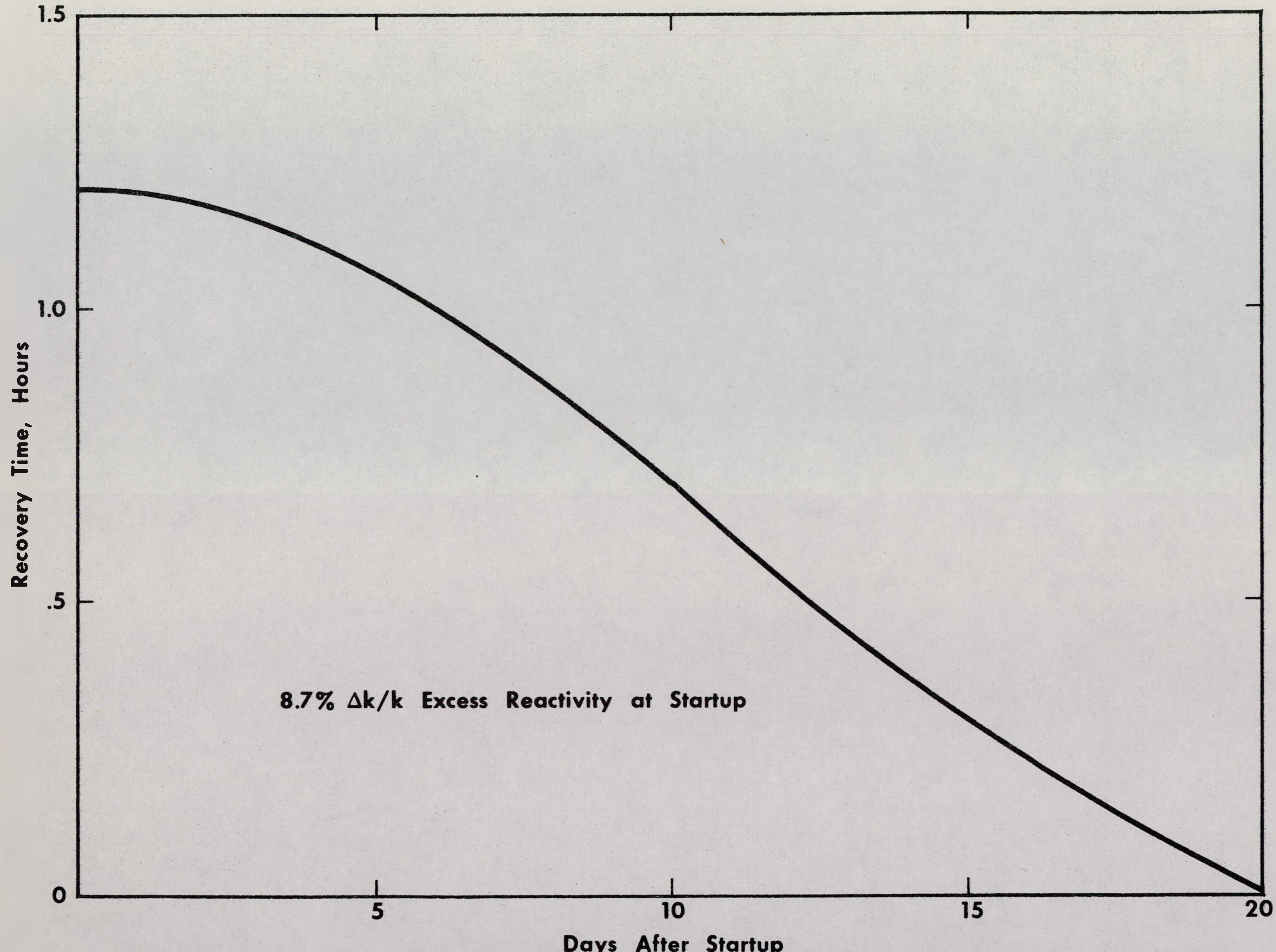
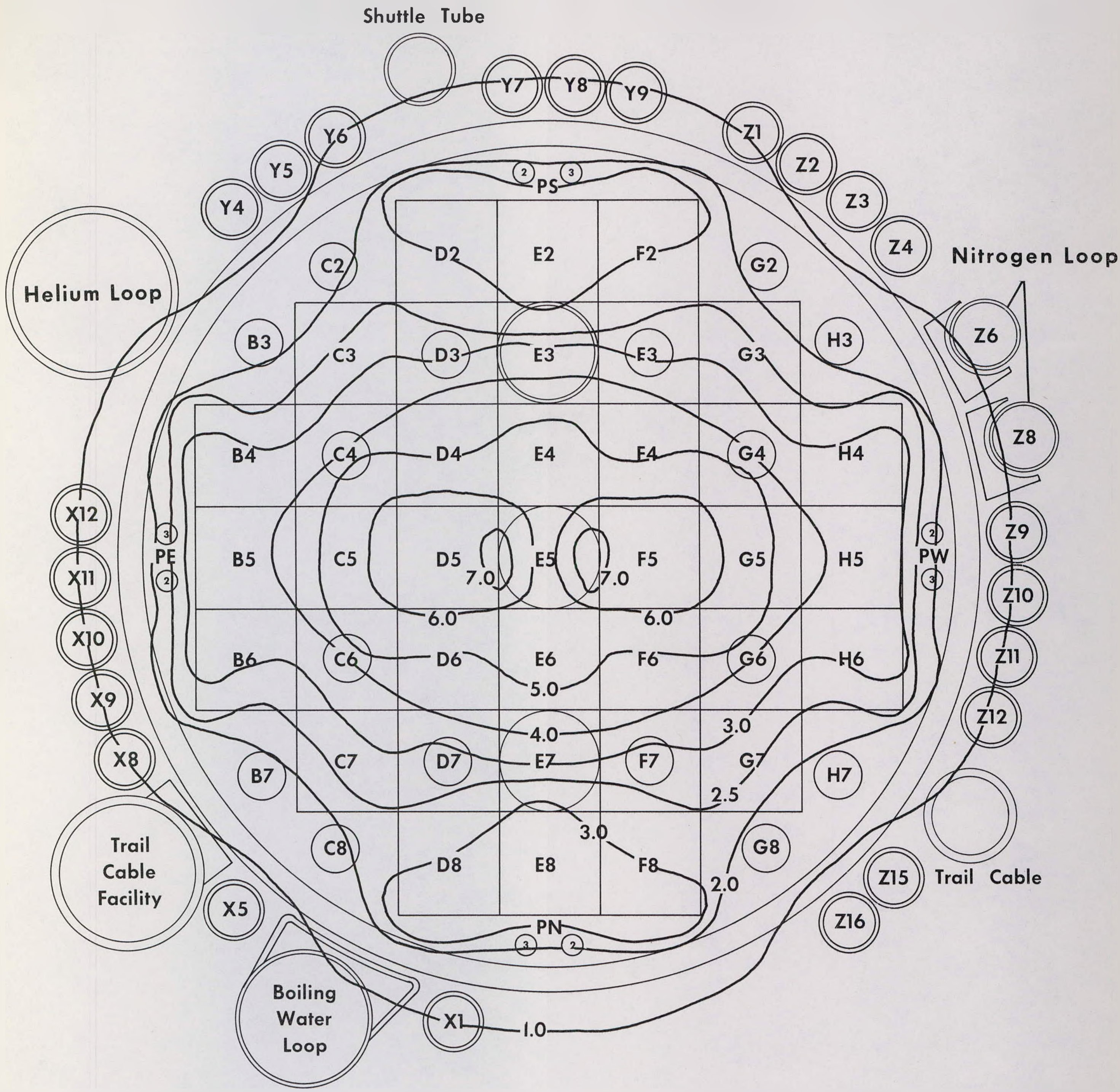


Figure 2.12 TIME AVAILABLE FOR RESTART AFTER SHUTDOWN



Watts per Gram

Figure 2.13. AVERAGE GAMMA HEATING 30MW

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The average velocity of coolant flow through the fuel channel is 21 feet per second with an over-all pressure drop through the reactor of 30 psi. Hydraulic test data for these fuel elements are available as a result of tests performed for the Engineering Test Reactor fuel. The fuel elements successfully withstood flow velocities under sustained operation of 47 feet per second.

Fuel element surface temperatures and burnout heat fluxes have been calculated for the reactor at 60 megawatts. Results indicate that there will be no nucleate boiling at 60 megawatts when the inlet water temperature to the core is 140°F and pump suction pressure is 100 psig. The burnout safety factor for these conditions is approximately 2.4 based on the Savannah River Laboratory correlation.[1]

Surface temperatures were calculated for three cases at the overpower trip level of 60 MW.

Case I This is the most critical case. The highest power generation is assumed to be associated with the flow channel between fuel elements and flow is assumed to be 85% of design flow. The width of the flow channel is 0.098 inches.

Case II Power generation in the flow channel is the same as in Case I, but flow is the design value and channel width is 0.110 inches.

Case III Power generation is 82% of the channel power for Cases I and II. This is the average power within the hottest fuel element. Channel width is 0.110 inches and flow is the design.

For Case I, fuel surface temperature exceeds saturation temperature by about 20°F. Case I could apply to only four channels in the core, but only one of these four channels would be at Case I heat fluxes.

(1) Heat Flux at Burnout, DP-355, Feb. 1959
S. Marshals, W. S. Durant, R. H. Towell

Generally, the highest heat fluxes have been in fuel plates adjacent to other fuel elements. This heat transfer condition is much less severe than Case I because channel water temperatures are lower since only one fuel plate heats the water.

At the reactor scram trip power level of 60 MW, the peak heat flux on the fuel plate is 1.21×10^6 BTU/ft^2-hr assuming 87% of the heating load is transferred through the fuel bearing surface. The heat transfer surface of the active portion of a 20 element core is 470 square feet. The calculated peak surface temperature is 370°F for the channel with maximum heat generated with the reactor operating at 60 MW. The saturation temperature at the maximum heat flux position is approximately 350°F. Boiling is not likely to occur under these conditions.

For operation of the reactor as a critical assembly at power levels up to 50 kilowatts, the reactor may be run unpressurized, with the pressure head removed, and without forced primary circulation. Natural convection circulation in the emergency cooling system is adequate to remove all the heat generated at this power although, in performing critical experiments, the primary system can be pressurized with full primary coolant flow.

The pool temperature will normally be held at approximately 100°F with a maximum allowable temperature of 135°F. This latter temperature limitation has been raised from 120°F because test data from the National Reactor Test Station demonstrates that fuel element failure will not result due to boiling in the fuel element coolant channels as might occur during emergency cooling conditions at the pool temperature of 135°F. The pool cooling pumps are operated in series providing 2000 gpm flow. Tests have shown that a flow of 1300 gpm can be maintained when one pump is operating and the other pump rotor is locked.

The reactor uses the pool as a heat sink for emergency and normal shut-down cooling purposes. When the normal reactor heat generation rate is low or circumstances require shutdown of the reactor, a natural circulation scheme is used to transfer the reactor core energy to the reactor pool. Heat addition in the core applies a driving force so that normal downward flow is reversed by means of natural internal circulation and flow is established in an upward direction. The reactor may be brought from any phase of normal operation to equilibrium shutdown temperatures with this cooling scheme without damage.

The thermal stress load on the pressure vessel is primarily from gamma heating. The heating from experiments adjacent to the pressure vessel is of a minor nature. The pressure vessel wall is water-cooled by forced convection on the inside surface and natural convection on the outside surface. The internal cooling passages are maintained by a spacer constructed integrally with the internal core components. These heat fluxes lead to a maximum internal wall temperature of 188°F which is a safe operating temperature for the pressure vessel.

In the event that coolant flow is lost to the fuel elements, and consequently, to the inner surface of the pressure vessel, the reactor is automatically scrammed, and the system pressure is released. The temperature rise during this transient is not sufficient to cause excessive temperatures or thermal stresses in the pressure vessel and adequate cooling is provided during all portions of the power decay.

2.7 Instrumentation

Instrumentation essential to the control and safe operation of the facility is described in this section. Instrumentation will warn of potentially unsafe trends and scram the reactor before a dangerous condition occurs. Instrumentation also initiates other automatic actions such as isolating the reactor containment vessel or preventing control rod withdrawal. Measuring operating parameters and control of the process is also accomplished by the instrumentation.

A gamma radiation monitoring system with sensors located at selected locations gives warning of high activity.

The instrumentation is centrally located in the control room to permit the operator to control the reactor, observe trends and check the performance of equipment.

The intercommunication system consists of a master station in the control room at the nuclear console and remote stations located at all important experimental and reactor operation points in the containment building.

2.7.1 Nuclear Instrumentation

The nuclear instrumentation provides continuous flux level monitoring and protects the reactor against an excessive power level or a rapid rate of power rise. The nuclear instrumentation operation is diagrammed in Figure 2.14. It includes:

Two wide range gamma compensated ion chamber channels.
Two uncompensated ion chamber channels.
Two fission counter channels.
A Log N and period channel.

The four safety channels (i.e., two compensated and two uncompensated ion chambers) are at locations near the mid-plane of the core in the pool. The chambers are contained in 4-inch aluminum tubes sealed at the bottom and extending up above the surface of the pool water. The

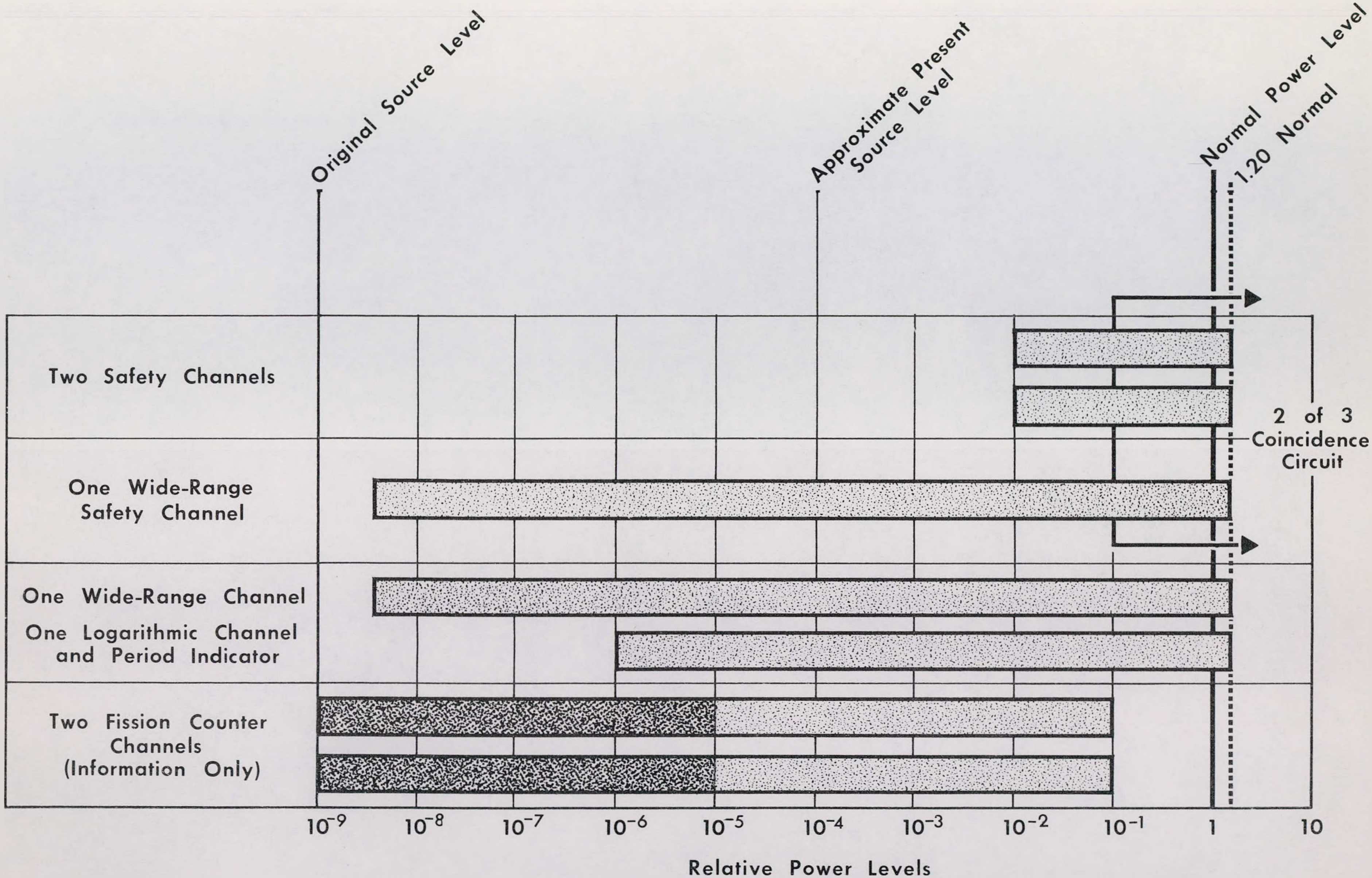

Figure 2.14 INSTRUMENTATION & CONTROL CHANNELS

tubes have a jog bend to prevent escape of high radiation to the pool surface. The chambers may be lowered or raised in the tubes to obtain the optimum neutron intensity for proper response of the chamber. During power operation only one of the chambers is moved at a time.

The two wide range gamma compensated ion chamber channels have six decades of response. Each wide range channel micromicroammeter has an adjustable trip setting which can be used during startup to scram the reactor at any level within the six decade range. A linear power recorder working from either micromicroammeter is used to record reactor power.

The coupled safety amplifier monitors two uncompensated ion chambers and controls the magnet current to the six control rods. The safety amplifier has several features which contribute to its reliability. The signal from each ion chamber is monitored by two vacuum tube sections and two plate relays. Each plate relay has a slave relay which interrupts power to the magnet power supply,giving a second mode of scram. The relay output from the micromicroammeter couples into the ion chamber input of the safety amplifier in order to operate the trip. To effect the trip, the relay contacts close and short circuit the input to the safety amplifier, causing a scram. If the trip point on either the two uncompensated ion chambers or either one of the two wide range channels is exceeded before the reactor operating level exceeds 10% of normal power, as indicated by the Log N and period recorder, a scram will occur. Above 10% of full power, three of the channels operate a coincidence circuit so that two of the three must trip in order to initiate a scram. The coincidence feature requiring two out of three trips to initiate a scram is built into the safety amplifier by special arrangement of relay contact.

The Log N and period channel consists of a gamma compensated ion chamber, a Log N and period amplifier and a Log N recorder. The channel monitors flux level from about 10^{-6} to 300% of full reactor power. Relative power level is indicated and recorded on a logarithmic scale. When the reactor period is too fast, the relay circuit in the period amplifier trips resulting in a reactor scram. Above 10% reactor power, the period signal does not result in a scram.

The startup channel consists of a sensor, a preamplifier, a linear amplifier, log count rate meter, a log count rate meter recorder, and a scaler. During startup, the fission counter is positioned in a lead shield near the mid-plane of the core in the reactor pool. After startup the fission counter is withdrawn to a position of low flux to prevent damage to the counter. The fission counter is raised and lowered from the third floor adjacent to the pool. The log count rate meter monitors level for four decades and drives the log count rate meter recorder. The startup channel is not connected to the scram circuit. The function of the startup channel is to give the operator information on the behavior of the reactor at low levels. Gamma effects on the channel are minimuzed by pulse height discrimination.

The channel used during refueling or other low level experimental work consists of a portable fission counter, a preamplifier, an amplifier, and a scaler. The counter may be inserted in the lead shield previously described, but is withdrawn prior to high power operation.

The fission counter channels are used to monitor neutron flux from source level to approximately 10^{-5} times full power. Although these channels do not actuate the safety circuit, they do contribute to the safe operation of the reactor by furnishing low level indication of power to the operator. Source level within the reactor is relatively high because of the gamma-neutron reaction in the beryllium.

When the reactor is operated as a critical assembly at power levels up to 50 kilowatts, it will be protected by low level trip settings on two wide range safety channels and by the Log N period channel.

2.7.2 Process Instrumentation

The process instrumentation protects against flow instability, sustained flow oscillations, provides backup for the nuclear instrumentation, and generally indicates any system malfunction of consequence. The system responds under fast transient conditions as well as more stable conditions. Additionally, the instrumentation will record, energize alarm relays, and actuate valves as required to satisfy the requirements for safe plant operation. The quantities measured include flow, temperature, pressure, pH, conductivity, fission product activity and radiation.

Special fast response instrumentation is provided to measure primary coolant flow, differential pressure across the reactor, and pressure at the primary pump inlet. A typical response time for the instrumentation is less than 0.15 seconds. This response time is defined as the time interval between the occurrence of the condition which calls for a scram and the initiation of the scram rod acceleration. The temperature instrumentation had a response time of less than 2.0 seconds.

When operating above 50 KW power, the following conditions will scram the reactor: low primary coolant flow; high reactor coolant outlet temperature; high reactor coolant inlet temperature; high and low reactor differential pressure; low pressure at the primary pump suction; low pool capsule header pressure; high seismic activity; experimental loop scrams; and incoming power loss. Undervoltage or frequency fluctuation will scram the reactor if they continue for more than 30 seconds. In most cases alarm settings precede the scram levels. Alarms indicate such conditions as low primary pressurizer pressure, high-low primary pressurizer water level, low nitrogen manifold supply pressure, high conductivity in the primary system, high activity in the primary system, and low demineralizer water tank level. There are more process alarms not listed here. A listing of scram and alarm conditions and settings is provided in Table 2.3.

The accuracies and performance of the instruments used in the reactor systems are of a high quality reactor grade instrument. Resistance thermometers are used at the reactor inlet and outlet for their inherent accuracy. Temperatures throughout the primary cooling system, the pool cooling system, and the secondary cooling water system are measured in places of interest. Mercury is not used in any reactor system transmitter. Duplication of instrumentation is used when a high degree of system reliability is necessary.

Conductivity cells are installed in demineralized and raw water systems to monitor the fluid for ionic purity. The cooling tower basin water is monitored for pH.

The quality of water in the various cooling systems is determined by periodic sampling and analysis.

2.7.3 Alarm Signals

The purpose of the alarm system is to warn the operators of approaching abnormal or potentially unsafe operating conditions. The system, generally, consists of a sensing element, an adjustable set point relay, and a horn and push button set for the purpose of acknowledgement and test. The annunciators are located on the control console. When an alarm signal is given, backlighted nameplates are illuminated and a horn is sounded.

The signals which initiate alarms, trips, scrams, or rod run-in are listed in Table 2.3, together with the action level and accuracy measurement. The values specified are those for operation of the

TABLE 2.3

ALARM, TRIP, SCRAM, AND ROD RUN-IN LEVELS

Item	Alarm Level	Trip Level	Scram Level	Rod Run-In
1. High Flux	107.5 + 2.5% normal		120 + 2.5% normal	
2. Fast Reactor Period	30 sec.		8 sec.	
3. Low Primary Coolant Flow (Emergency Cooling Trip)	9500 ±600 gpm	3000 ±600 gpm	8500 ±600 gpm	
4. High Reactor Coolant Outlet Temperature	174° ±2°F		180° ±2°F	
5. High Reactor Coolant Inlet Temperature	135° ±2°F		140° ±2°F	
6. Low Reactor Differential Pressure (Emergency Cooling Trip)	24 ± 1 psi	7 ± 0.5 psi	20 ± 1 psi	
7. Low Primary Pump Suction Pressure	97 ± 6 psig		90 ±6 psig	
8. High Seismographic Intensity			Modified Mercalli IV	
9. Low Primary Pressurizer Pressure	115 ± 3 psig			
10. High-Low Primary Pressurizer Level (From Tank Centerline)	+13 and -13 inches			
11. Low Instrument Air Pressure (Below Compressor Cut-In)	5 psi			
12. High-Low Demineralized Water Supply Level (From Tank Base)	+22 and +1 feet			
13. Low Pool Water Level (From Overflow)	-6 inches			
14. Low Canal Water Level (From Overflow)	-6 inches			
15. Low Discharge from Cooling Tower Pump (Shutdown Pump Trip)	7200 ±500 gpm	Loss of 1 or both cooling tower pumps		
16. High Retention Tank Level (From Bottom of Tank)	9.5 feet			
17. High Conductivity in Primary Loop	1.5 ± 0.2 micro-mhos			
18. High Stack Particulate Activity	40 μc/sec			
19. High Stack Gas Activity (Isolation Valve Trip)	6000 μc/sec	60,000 μc/sec		
20. High Containment Vessel Pressure (Isolation Valve Trip)		2 psig		
21. Low Capsule Header Pressure	6 psi		4 psi	
22. Loss of Emergency Power			Power Breaker Drop-Out	
23. Loss of Incoming Line Power			Power Breaker Drop-Out	

reactor at 30 megawatts. These values will change for 50 megawatt operation as experience dictates and within the limits specified in the technical specifications. The various set points have been determined to maintain safe operation and to protect the reactor in case of malfunction.

2.7.4 Communication System

The intercommunication system has a master station in the control room at the nuclear console and remote stations at each experimental area and in other strategic places. The system operates on the normal electrical power system. If this system fails, the unit automatically switches to battery power which can supply continuous power to the system for over an hour. The battery is trickle charged and ready for service at all times. The control room, office building, and strategic locations in the containment building are also equipped with telephones.

2.7.5 Radiation Monitoring System

Radiation level is monitored at selected locations and gives warning of high activity in accordance with 10CFR20. Some of the monitors are located at work stations to indicate existing dose rates; others are used to monitor the process and detect a release. The radiation levels at these stations are recorded on a multipoint recorder in the control room. If at any time the radiation level of a station exceeds a pre-set level, a warning light and, usually, an audible alarm at the chamber location, are activated.

A scintillation counter is used to monitor reactor coolant radioactivity. The levels are continuously recorded and alarms are actuated by high activity.

A constant air monitor records the activity level of gases and particulates being discharged from the stack. The particulates are monitored after collection on filter paper and the filtered gas is monitored separately in a shielded collection chamber.

2.8 Shielding

Biological shielding was desired to limit radiation levels to less than one mrem per hour in all areas of continuous occupancy. Missile shielding was provided to assure the integrity of the containment vessel from high energy missiles in the remote event of a severe incident.

2.8.1 Missile Shield

Missile shielding is provided over the top of the reactor pool and between the pool and the canal. In addition, blast mats consisting of alternate layers of steel and redwood are placed in the biological shield to minimize damage to the concrete. Both of the shields over the reactor and between the reactor and the canal can be removed during refueling operations.

The shield over the pool consists of an octagonal-shaped laminated steel slab. It is 11 feet across flats and 13 inches thick, mounted on wheels to permit removal and restrained by four bolts, 2-7/8" diameter by 23" long.

To guard against the possibility that test capsules, fragments, or other high-energy missiles may be projected through the opening between the pool and the canal, a steel shield is mounted in that opening. The shield is used as a gate leading to the canal. It is approximately three feet wide with three sections which form a gate 20 feet high. Its thickness of six inches is adequate to prevent penetration by missiles.

A blast mat consisting of three layers of redwood placed behind steel plates was built into the inside surface of the biological shield and encircles the reactor. It extends five feet above and below the core centerline. The inside surface of the first steel layer coincides with the inside surface of the biological shield and was made 2.75 inches thick to be sufficient as a thermal shield to protect the wood and concrete from excessive radiation heating. A two-inch layer of redwood was placed adjacent to this layer of steel. The second and third stages of the blast mat were placed in the concrete about 14 inches from the first stage. Each of the second and third stages consists of a 0.5 inch thick steel plate followed by a two-inch thickness of redwood. Two rows of cooling tubes were cast into the concrete between the first and second stages.

2.8.2 Biological Shield

The reactor pool wall is composed of normal and heavy aggregate reinforced concrete with a blast mat as described in the preceeding section. This biological shield has a thickness of eight feet at a point directly opposite the core. At other points, a thinner section of concrete was adequate.

The highest gamma level detected on the outer face of the biological shield at 30 MW power has been 0.5 mr/hr. At the increased power of 50 MW the gamma level should remain less than 1 mr/hr. Neutron activity has not been detected at the outside face of the biological shield using an instrument capable of detecting 0.01 mrem/hr of thermal or intermediate flux or 2 mrem/hr of fast flux.

In limited access areas, such as the equipment space and the control rod access room, the radiation is not limited to 1 mr/hr. Personnel access is limited within these areas in order to control radiation exposure in accordance with permissible standards. These spaces were provided with shielding to limit the outside dose rate to a level which will allow continuous exposure.

2.9 Containment Building and Arrangement

The containment building is a steel tank 66 feet in diameter, approximately 105 feet high, within which is constructed a reinforced concrete structure. This enclosure has a hemispherical head and flat bottom extending approximately 15 feet below grade. The bottom head of the containment vessel is a reinforced concrete mat. Access is obtained into the containment building through two airlock openings. Within the structure, a steel stairway, an elevator, and a hatchway are provided between the basement and the third floor. In addition, emergency hatches and ladders are provided to prevent personnel from being stranded in the event a fire or other hazard has blocked the elevator and the stairway.

2.9.1 Containment Vessel

The containment vessel was designed to contain radioactive contamination resulting from an incident as severe as a maximum credible accident. Permissible design stresses would be achieved by an external pressure of 0.2 psig and a wind of 75 miles per hour. Periodic testing assures the vessel's leak tightness is maintained. The design pressure of the vessel is 5 psig with a coincident temperature of 250°F. It has a free volume of 230,000 cubic feet. In addition to the reactor, the containment vessel houses the experimental equipment and the primary coolant loop equipment for the reactor.

The basic code used for the design, construction, and testing of the containment vessel was a modification of the American Petroleum Institute tentative standard 620, first edition, dated February, 1956, for the "Design and construction of large, low pressure storage tanks". The modifications of this code developed for this particular application impose additional requirements. This instruction is in accordance with the California Boiler, Safety Orders and the vessel has been designated as California Special No. 704.

The vessel rests on a reinforced concrete base pad. This pad, which is in effect the bottom head of the containment vessel, has been designed to act as a pressure vessel head independent of any other loading. A continuous plate is anchored into the concrete base pad and the cylinder of the containment vessel rests on and is welded to this plate. This steel plate forms the vessel bottom. Dowels necessary to anchor internal concrete construction to the base pad extend from the base pad through the vessel bottom. The steel vessel bottom is protected from mechanical damage by means of an internal concrete liner.

Rigorous stress analyses have been made of the containment vessel superstructure design to show that stresses at points where stress concentration must be anticipated do not exceed the allowable stress for the metal. The base ring was satisfactorily tested at an impact force of 15 foot lbs. at 30°F. This force was selected since the base ring is located 15 feet below grade. The steel vessel bottom is not used to restrain internal pressure and may be considered only as a membrane. All the material used in the vessel other than the bottom plate has been tested to assure that it will withstand an impact force of at least 15 foot lbs. at a temperature of -11°F, which is 30 degrees below the lowest ambient ever recorded at this locality.

All wall and roof joints of the vessel are double butt-welded and the vessel has single welded butt joints in the steel tank bottom. All vessel penetrations are reinforced and installed in accordance with the requirements of the ASME Code for unfired pressure vessels. Stringent welding requirements were specified to assure the quality of welds and minimize notches which may be considered as possible causes of brittle fracture failures. The vessel is insulated externally with a 0.75-inch thickness of mastic for purposes of minimizing stress imposed by uneven heating of the shell as a result of a solar load. By the use of external insulation and internal heating of the structure, the steel shell has a minimum temperature during winter operation of approximately 40°F. Ultrasonic techniques were used to detect flaws which were confirmed by further radiographic examination and repaired as required.

Upon completion of the containment vessel, it was successfully leak tested at 1.25 times design pressure, or 6.25 psig. Additional tests are performed periodically to assure that the maximum allowable leak rate of 2% per 24 hours at 5 psig pressure is not exceeded.

The integrity of the containment vessel is maintained by the use of airlocks for personnel and equipment access and by the use of isolation valves at all points where piping penetrates the containment vessel wall. The personnel and equipment access airlocks are each provided with mechanical interlocked pairs of doors so arranged that only one door in each airlock can be opened at one time. The personnel air lock has an opening 3 feet by 6.5 feet, and the equipment airlock has an opening 8 feet by 10 feet.

Containment vessel construction details are shown on Figure 2.15.

2.9.2 Building Arrangement

The reactor rests in a pool of water formed by the biological shield and located at approximately the center of the containment building. The canal extends radially from the pool to the periphery of the containment building as shown in Figure 2.16. The reactor coolant equipment occupies a 30° segment of the structure about the canal center line which extends from the underside of the third floor to the basement. All components of the primary coolant system are contained in a shielded space within that area.

The four floor structure within the containment vessel is constructed of reinforced concrete. The second and third floor design was dictated to a greater extent by shielding requirements than by structural requirements. These floors are three feet thick which has provided adequate shielding for all experimental uses to date.

Space on all floors other than that occupied by reactor cooling equipment, the pool, and the biological shield is available for experimental equipment. The basement, first and second floors each provide two experimental areas of approximately 800 square feet each. The second floor also provides space for use of the beam port facility. The third floor provides experimental space in addition to the space taken up by the canal, the pool and its missile shield, and building service equipment. The third floor is served by a crane of 15 ton capacity. It provides service to the other floors through the equipment hatch which is expandable to an 8 foot by 14 foot opening if the demountable stairs are

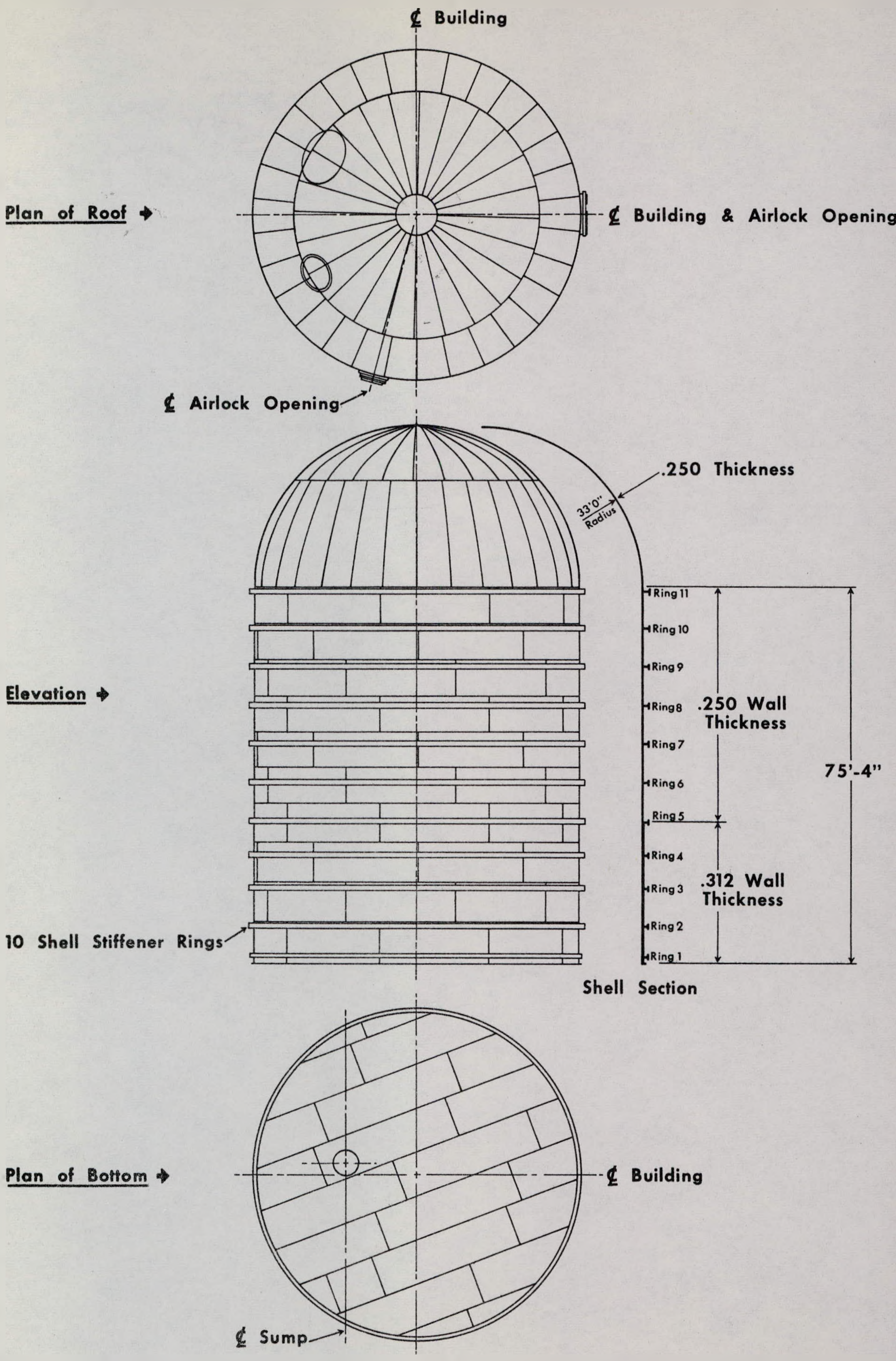

Figure 2.15 CONTAINMENT VESSEL

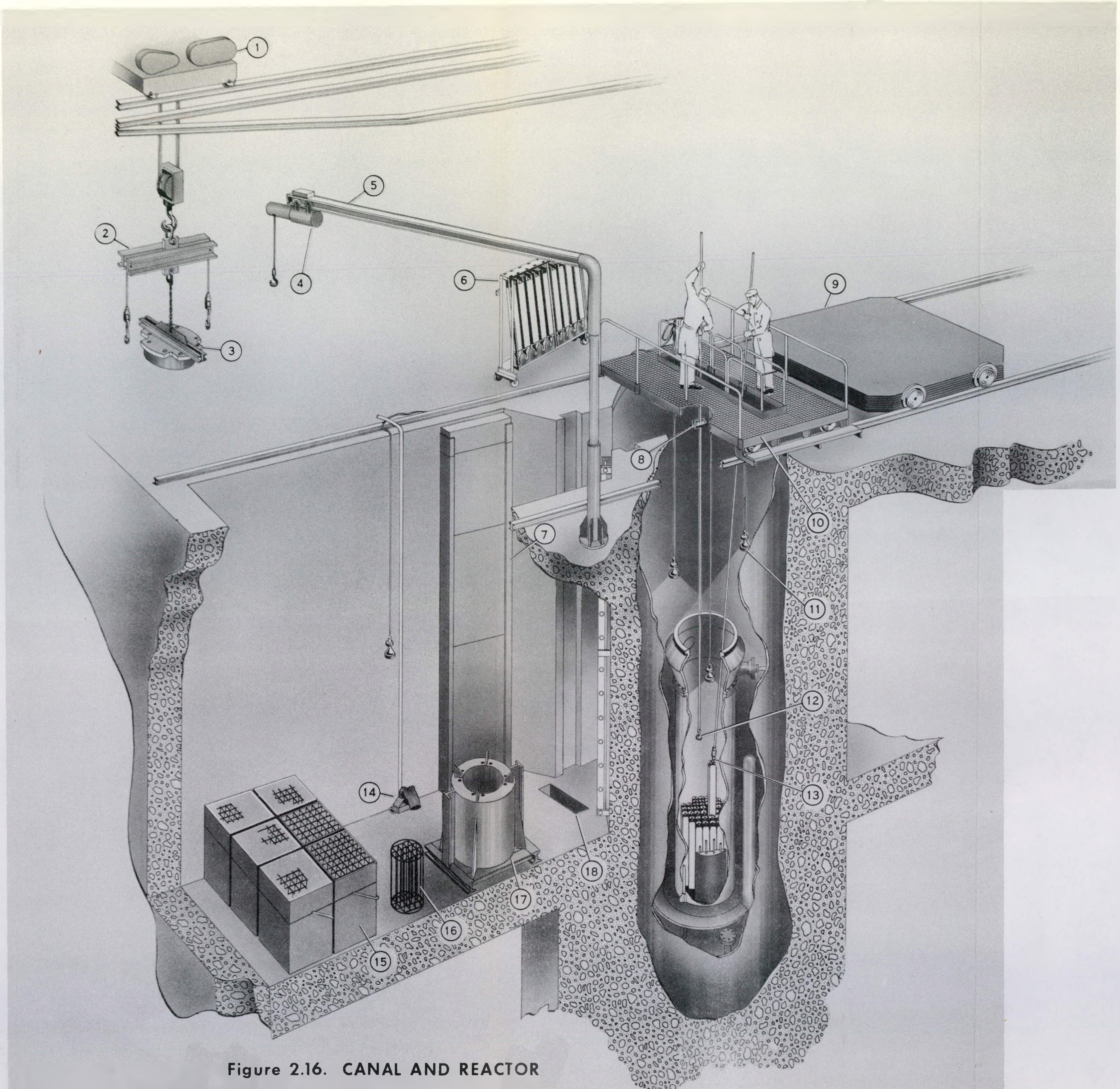

Figure 2.16. CANAL AND REACTOR

KEY

(1) Polar Crane
(2) Cask Lifting Bail
(3) Shipping Cask Cover
(4) 1000 Lbs. Hoist
(5) Jib Crane
(6) Fuel Storage Rack
(7) Canal Gate Storage
(8) Viewing Aid
(9) Missile Shield
(10) Refueling Platform
(11) Underwater Lights
(12) Hook Grapple
(13) Element Grapple
(14) Underwater Vacuum Cleaner
(15) Fuel Storage Racks
(16) Shipping Cask Basket
(17) Shipping Cask
(18) Canal Storage Pit

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Figure 2.16

removed. Manholes are provided on each floor and located to permit extension of loop piping from any of the experimental facilities to any one of the seven experimental areas throughout the containment building.

Figures 2.17, 2.18 and 2.19 show the containment building arrangement.

2.10 Reactor Coolant System

The cooling system consists of a primary system which absorbs the reactor heat and a secondary system which rejects the reactor heat to the atmosphere through a cooling tower. A shell and tube heat exchanger is used to transfer the heat from the primary loop to the secondary loop. In addition to the reactor cooling system, there is a separate pool cooling system and a separate canal cooling system. The pool and canal cooling systems reject the heat through a heat exchanger into the secondary system. Water is the coolant used in all systems.

These cooling systems make up the major portion of mechanical and electrical equipment which serves the facility.

Current water storage is provided by three retention tanks and a hold tank. Two "fill and flush" pumps are used for flushing water from the primary loop and the reactor pool into underground storage tanks. The water is held in these tanks, if necessary, prior to being returned to the hold tank for reuse. Demineralizers are used to maintain coolant purity in the primary and pool water systems. A deaerator system has been designed for use in degassing the system, and will be installed at a later date.

All piping, heat exchangers and vessels in the primary loop are of aluminum construction. Pumps and valves are generally of cast-iron construction with stainless steel trim. In circulating systems where coolant purity is not a problem, conventional iron piping and valves are used. Figure 2.20 shows a simplified schematic of the primary system.

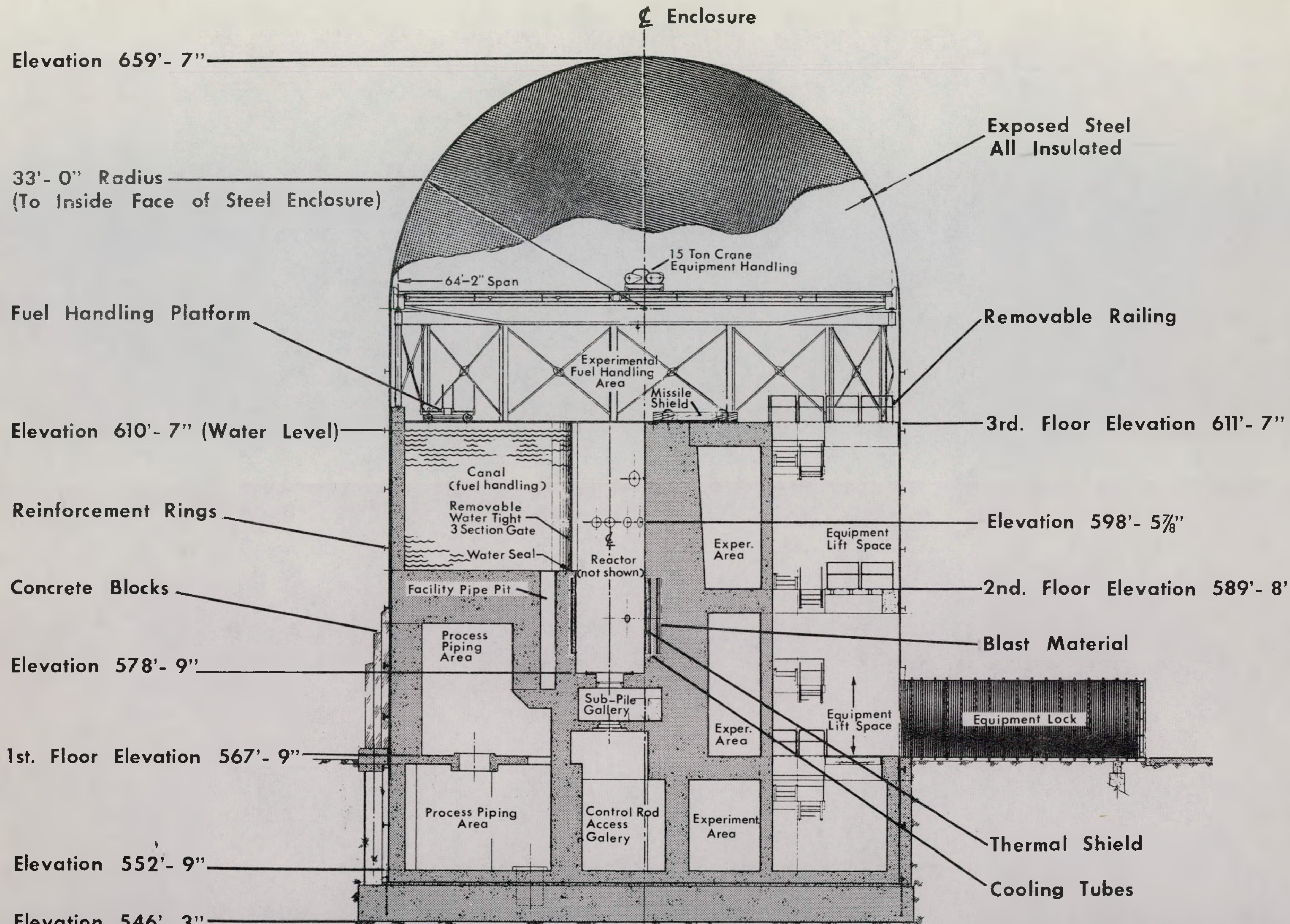

Figure 2.17 REACTOR BUILDING CENTERLINE SECTION

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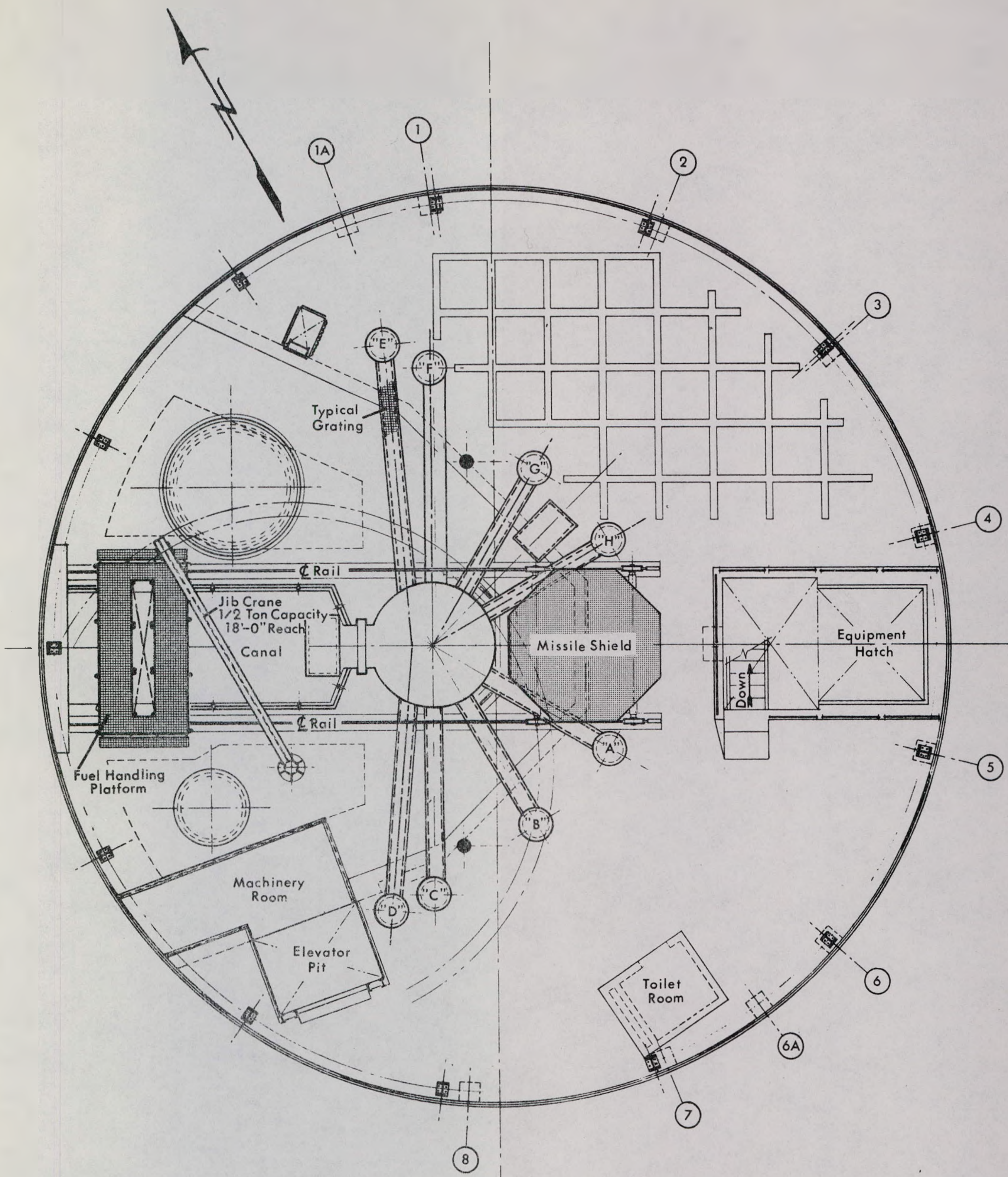

Figure 2.18. REACTOR BUILDING THIRD FLOOR PLAN

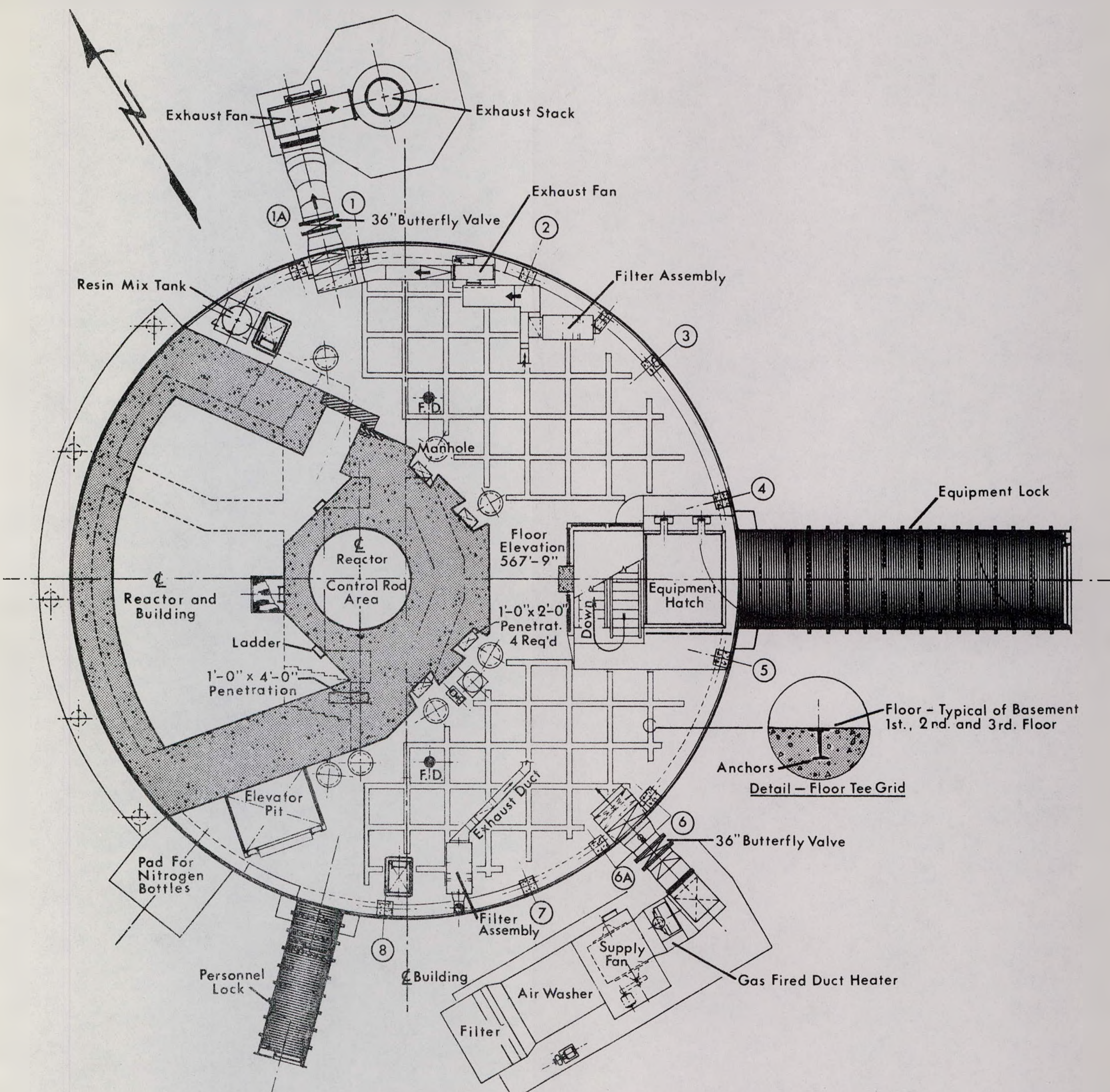

Figure 2.19 REACTOR BUILDING FIRST FLOOR PLAN

2.10.1 Primary System

The reactor is cooled by a pressurized closed loop of 5,000 gallon capacity housed within the containment building. One vertical centrifugal pump circulates approximately 10,000 gpm of demineralized water through the reactor core. Water leaving the reactor passes through the tubes of the primary heat exchanger and is returned to the reactor by the primary coolant pump. The system is pressurized at approximately 130 psia by means of a nitrogen pressurizer. The primary system piping is seamless aluminum. The primary heat exchanger is an aluminum U-tube type.

The primary heat exchanger for the reactor was replaced in 1961. The replacement unit has 40% more heat transfer area than the original unit. The new exchanger has a 200 psig primary side pressure rating at 250°F (ASME Unfired Pressure Vessel Code rating). The original unit was rated at 150 psig at 140°F. The rating of the secondary side is 80 psig at 175°F for the replacement heat exchanger, compared to 80 psig at 115°F for the original exchanger. The exchanger was designed to meet the ASME Code. The replacement heat exchanger is designed to operate at reactor power levels in excess of 60 MW.

A full flow stainless steel strainer has been installed in the reactor primary system at the inlet line to the primary heat exchanger. The strainer is a truncated cone sized to fit the 20-inch primary system flange. The strainer screen is 1/16 inch mesh. A 3-inch inspection flange has been installed in the piping to permit inspection of the strainer. Orifice flanges have been installed in both reactor vessel inlet lines to check flow to the reactor core within specified limits.

Coolant conditions which result in automatic alarms or reactor scrams are listed in Table 2.3. Instrumentation is provided to initiate a reactor scram on low primary coolant flow, high reactor coolant outlet temperature, high reactor coolant inlet temperature, low reactor differential pressure, and low primary pump suction pressure. All critical operating parameters are indicated on the process console in the control room.

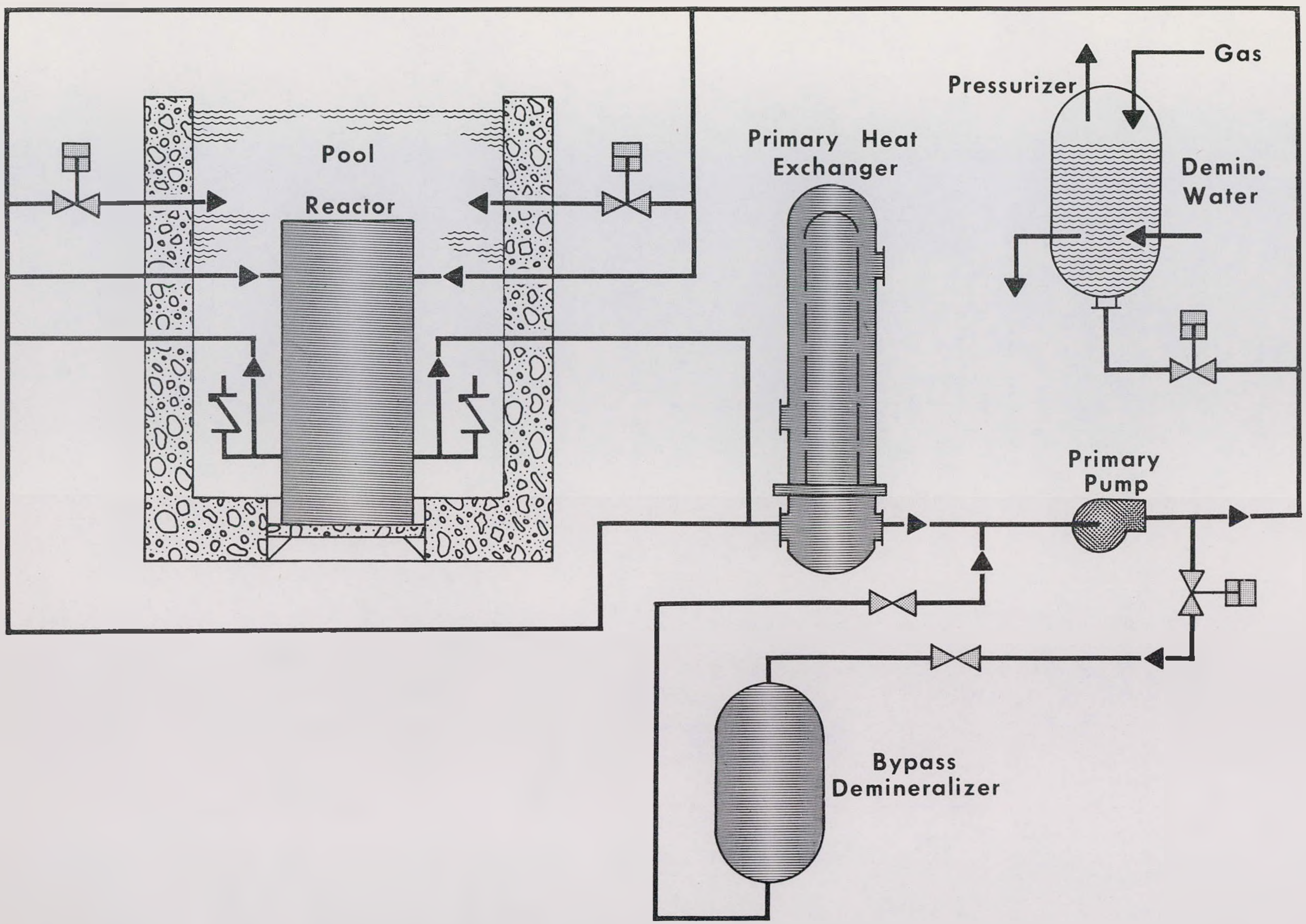

Figure 2.20 PRIMARY COOLING SYSTEM CLOSED LOOP

Approximately 50 gpm of the primary cooling water flow bypasses the reactor cooling pump and flows through a demineralizer and filter which maintain the water purity at a high level. The water conductivity is normally held to below one micromho per centimeter but may increase as high as two micromhos per centimeter during operation of the reactor. Experience has indicated that radioactivity in the primary system may be allowed to increase to 5 times background before reactor shutdown is advisable. This increase in primary activity does not result in dose rates which exceed the maximum permissible for continuous exposure. Normal background for operation at 30 MW is equivalent to 2×10^3 dpm/ml of I^{131} which is used to gage activity level. This value is equivalent to a total activity of about 10^6 dpm/ml.

A system is provided for remotely flushing expended resin from the demineralizer into an underground storage tank, and a resin mixing tank is provided to enable flushing of the slurry of new resin and water into this device. Filling of the primary loop or makeup of demineralized water into the primary loop is accomplished by discharging water from the fill and flush pumps into the loop. The fill and flush pumps take suction from the above-ground demineralized water hold tank.

A hairpin loop is provided in each of the reactor outlets to prevent uncovering the reactor core in the event of pipe rupture in the primary loop. Antisiphon devices are provided in each hairpin loop.

2.10.2 Secondary System

The heat from the reactor is transferred from the primary system to the secondary system through the primary heat exchanger. The heat is then dissipated to the atmosphere through a cooling tower. Two pumps in the secondary system, each having a designed capacity of 4,800 gpm, take suction from the cooling tower basin and discharge through the shell side of the primary heat exchanger for the return flow to the cooling tower. In addition to providing a heat sink for the primary cooling system, the secondary system also provides a heat sink for the pool cooling system, the experimental loops, the canal cooling system, the biological shield cooling system, and service equipment such as air compressors. The cooling tower, associated pumping equipment and piping are sized for heat loads in excess of 180 x 10^6 Btu per hour at 9,000 gpm. This rating is greater than that listed in the previous GETR Hazards Report, GEAP-2064, due to an increase in the water inlet temperature to the cooling tower. The cooling tower has a water outlet temperature of 87°F and a water inlet temperature of 127°F at 50 MW operation. This permits a 15° approach to a 70°F wet bulb temperature which has been measured during operation of the reactor. The cooling tower capacity, which is in excess of that required for dissipating reactor heat, allows reserve capacity for other requirements. The cooling pump normally operates at 9,000 gpm, although the discharge may range from 6,000 gpm to 11,000 gpm depending upon the operating level and atmospheric conditions. If both cooling tower pumps fail, a shutdown pump operable by emergency power will supply coolant to the experimental facilities as necessary. Temperature actuated flow control is provided in the secondary loop at the outlets of the primary heat exchanger and the pool cooling heat exchanger to maintain constant temperature of outlet water from these heat exchangers. The flow control valves will open in the event of a malfunction of the control system.

2.10.3 Pool System

Pool water is circulated and cooled at a flow rate of approximately 2,000 gpm. A portion of the circulated water is utilized to cool the external experimental capsule tubes which surround the reactor pressure vessel in the pool. The two centrifugal pool circulating pumps have stainless steel impeller shafts and trim. They operate in series as described in Section 2.6.2. A demineralizer is used to maintain the purity of the pool water. The pool cooling pumps take suction from the reactor pool near the bottom and discharge through the tube side of the pool heat exchanger and thence into the reactor pool at an elevation below the pool water level. At the point of discharge into the pool, the flow is diverted for partial flow through the capsule tube header and partial flow directly into the pool. Heat transferred into the pool heat exchanger is dissipated into the secondary system by circulating the secondary water through the shell side of the heat exchanger. At 30 MW power the pool temperature has been held to approximately 100°F. At 50 MW operation the pool temperature will be held to a maximum of 135°F.

The pool water provides shielding, reflection, and a heat sink for emergency cooling. For critical assembly experiments under 100 watts of power, the pool water level is held to a minimum of 6 feet above the reactor core. At higher power levels, the gamma radiation from the reactor core requires that the pool water level be within 2 feet of the overflow which is 20.5 feet above the core. At this level the pool contains approximately 16,000 gallons of water.

In order to maintain adequate water for emergency cooling, the pool level must be within 2 feet of the overflow line at all power levels above 100 watts. Normally, the required level will be maintained between the overflow line and -0.5 feet. The piping is arranged to prevent draining of the pool below a minimum level in the event of a pipe rupture. An aluminum liner is provided to prevent leakage around the penetrations or through the biological shield.

2.10.4 Canal System

Subsequent to the initial reactor startup, it was found desirable to install a heat exchanger and demineralizer system for the canal water. The canal water is circulated and cooled at a maximum flow rate of approximately 50 gpm. A heat exchanger is used in the system. A 50 gpm pump circulates water through 2" aluminum piping. With the exchanger in operation, the canal water is maintained at approximately 85°F. When filled, the canal contains approximately 20,000 gallons of water.

2.10.5 Emergency Cooling System

A natural convection cooling system as shown in Figure 2.21 provides emergency cooling for the reactor in the event of a coolant system failure and also provides cooling during normal shutdown periods. An emergency cooling valve is provided adjacent to the outside surface of the biological shield in each leg of the two reactor inlet coolant lines. A line is extended from each of these valves through the biological shield and into the pool. These valves are electro-pneumatically operated. Check valves are installed vertically and adjacent to each of the vessel outlet nozzles in the reactor pool in such a manner that with no pressure in the primary loop the check valves fall open due to gravity (see Figure 2.21).

During forced cooling operation, the pneumatic valves on each reactor inlet are closed and full flow passes downward through the reactor. The pressurization of the primary loop closes the check valves, which forces full reactor flow to pass through the heat exchanger. The emergency cooling system will be actuated when the primary flow drops to 3,000 gpm, 30% of normal flow, or when the differential pressure across the reactor drops to 3.5 psi, or 15% of the normal differential reactor pressure. In the event of decreasing primary flow, as an example, the reactor scrams at 8,500 gpm. At 3,000 gpm, the two pneumatically operated valves referred to above open the inlet primary system to the reactor pool. Simultaneously, the nitrogen pressurizer supply valve controlling the flow of nitrogen is closed, and the pressurizer is isolated from the primary system. As these functions take place, the system pressure drops off and the check valves referred to above on each of the reactor outlet lines drop open by gravity. A convection cooling loop is established as soon as the kinetic energy of the primary loop reaches zero. Cool water in the bottom of the reactor pool enters the two check valve openings at the bottom of the reactor, flows up through the reactor into the reactor inlet pipe, and in turn flows back through the pneumatically operated valve into the upper region of the reactor pool. The volume of the reactor pool is sufficient to absorb the decay heat of the reactor following a scram from the trip point setting of 60 MW. The temperature of the water and the concrete increases until the loss of heat by evaporation from the pool surface balances the decay heat rate.

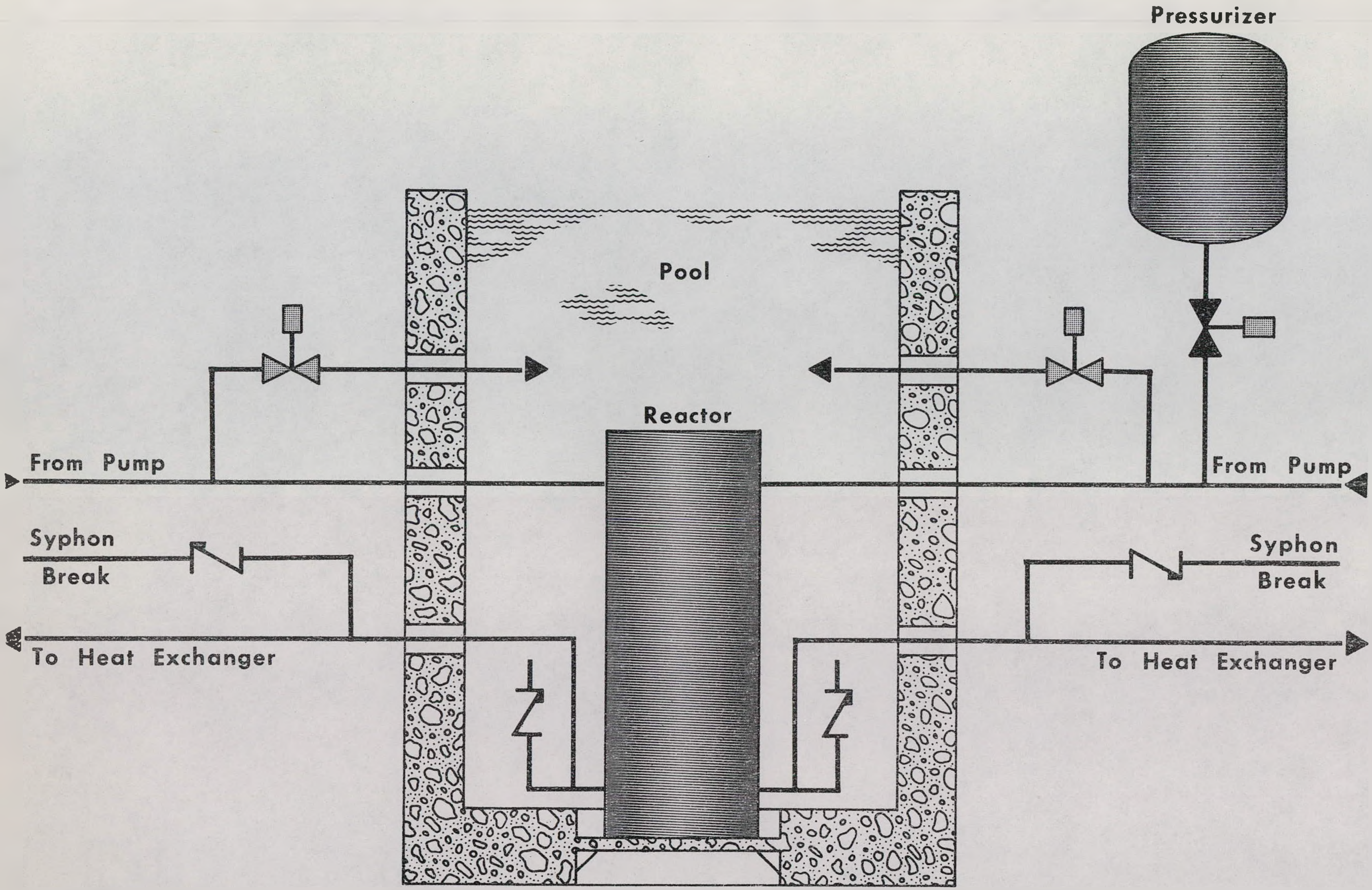

Figure 2.21 EMERGENCY COOLING SYSTEM

If, through some set of unusual circumstances, the shutdown cooling does not occur in the normal manner, the operator has additional cooling capacity available. If either emergency power or normal power is available, the operator may run a fill and flush pump and transfer demineralized water from the hold tank into the pool and drain pool water back to the hold tank. In this manner, the storage capacity of the hold tank is added to the heat sink capacity of the pool. The hold tank has a normal capacity of 50,000 gallons and must contain at least 25,000 gallons in reserve before the reactor can be operated.

If no source of power is available, raw water from the fire protection system may be flushed into the pool. A manually operated valve in the control room controls the flow. In such an event water would be drained from the pool to the retention tanks to maintain pool water at a satisfactory level.

To provide adequate drainage capacity, the reactor is not operated unless there is at least 25,000 gallons available storage capacity between the three 25,000 gallon retention tanks.

During normal shutdown, the primary cooling system pump is operated after the rods have been driven into the reactor and until decay heat has significantly decreased. At this time, the operator may undertake the flushing operation which is described in Section 2.10.6.

The remainder of the decay heat is then accommodated in the reactor pool. As flow decays in the primary loop, the emergency cooling valves will operate and a convection cooling system will provide sufficient shutdown cooling. Pneumatically operated valves open to allow emergency cooling in the event of an electrical power failure. These valves close by pneumatic power and are sized so that any one pneumatically operated valve and any one of the check valves at the reactor outlet will allow sufficient emergency cooling. In the event both pneumatically operated valves were to open inadvertently during full power operation, 75% of the flow would continue through the reactor core and 25% of the flow would bypass into the reactor pool. The reactor is instrumented to scram under these conditions due to loss of pressure even though the reactor would be adequately cooled by the 75% flow.

2.10.6 Fill and Flush System

After an extended period of operation, activity from sodium-24 and other corrosion products may be higher than desired for immediate working access to the reactor. The fill and flush system provides a means for the operators to flush the primary system and pool cooling system with demineralized water to minimize this activity. This operation utilizes up to three 25,000 gallon underground retention tanks, the hold tank make-up pump, the make-up demineralizer, the 50,000 gallon above-ground demineralized water hold tank, and the two fill and flush pumps.

The operator may use one or both fill and flush pumps to transfer water from the demineralized water hold tank in order to flush the pool, the pool cooling loop, and the primary cooling loop. As water is added from the demineralized water hold tank to the pool or to one of the loops, an equivalent amount of water is drained into the underground retention tanks. When the coolant activity has been decreased in this fashion to an acceptable working level, the operation is stopped and the missile shield removed from above the reactor pool. Refueling and other operations can then start immediately after the primary system is shut down and depressurized.

Following the fill and flush operation, radioactivity in the flushed water is allowed to decay. The water is then pumped through the makeup demineralizer and into the hold tank or clean retention tanks for storage until reused. All raw water added to the plant system is first processed in this manner. The clean retention tanks and the hold tank are conventional steel tanks with a plastic lining to prevent contamination of the water by the tank shell.

2.10.7 Experiment System

Cooling water for experimental facilities is provided by the main reactor systems. Cooling tower water, demineralized water, and raw water are made available for experimental facilities. Similar high standards of design, operation and maintenance are established for the experimental cooling systems as for the reactor systems as more fully described in Section 5.

2.11 Ventilation

The containment vessel ventilation system, illustrated in Figure 2.22, was designed to provide approximately five changes of air per hour within the structure. Approximately 18,000 cubic feet per minute of air flows through the system. Supply air filters, an air washer, a gas-fired duct heater, and the supply air fan with associated dampers are installed outside the containment vessel. All exhaust filters are located inside the containment vessel. A small exhaust system, including an exhaust fan for the experimental cubicles, is located inside the containment vessel. The exhaust isolation valve and the exhaust fan are located between the containment vessel and the stack.

The inlet air is charged by the supply fan, directed through an isolation valve, and then through a 36" penetration in the containment vessel. Within the containment vessel ducts direct the air to appropriate areas of the building. A system of return ducts with outlets strategically located is used to collect the air from the experimental and operating areas as well as the mechanical equipment space. At each duct intake point in this return air system, roughing filters, high efficiency filters, and dampers are provided. A separate return system is provided to collect air exhausted over the reactor pool and air exhausted from the experimental areas throughout the building. This latter system also uses high efficiency filters which are rated to be 99.95% efficient in removing 0.3 micron particles. It discharges into an exhaust air trunk duct where the two exhaust air systems merge into one system. The air leaves the containment building through a 36" diameter penetration, flows through an isolation valve, and into an exhaust air fan external to the containment building. The discharge of the fan is directed into a 95 foot tall,

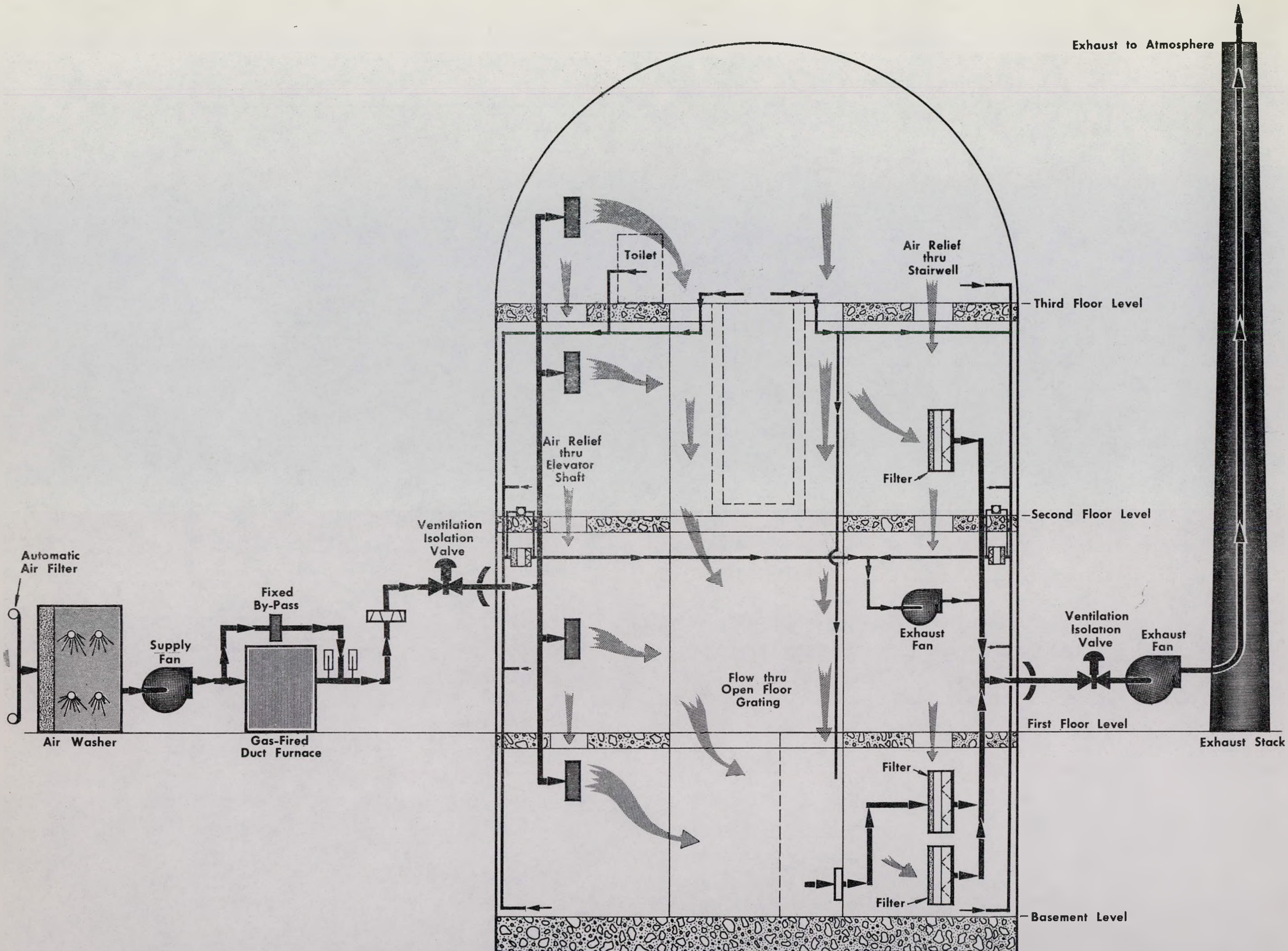

Figure 2.22 VENTILATION DIAGRAM

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stack. Radiation monitoring of and release limitations on the exhaust air from the stack are described in Section 6.3.

2.12 Utilities

Official inspections of utilities have indicated that both the GETR and the site comply with local and state standards for constructing the systems described in this section.

2.12.1 Water Supply

A 6" raw water line from the 500,000 gallon tank described in Section 7.4.3 feeds the reactor systems. During normal operation the primary system will contain approximately 6,000 gallons of water. The canal and pool can hold approximately 20,000 and 16,000 gallons of water respectively. These systems are serviced directly from the demineralized water hold tank which contains a minimum of 25,000 gallons of water for a normal startup. The underground retention tanks form a receiving system to hold waters from these systems as required.

2.12.2 Electrical System

The site electrical power supply to the GETR sub-station is described in Section 7.4.5. Load center #1 out of the GETR sub-station feeds the generator control panel, mock-up shop, the cooling tower fans, and load centers #4 and #5. Load center #4 furnishes power to a number of small pumps in the containment building, heating and ventilating equipment for the containment building and office building, and certain miscellaneous equipment such as the containment building crane. Load center #5 is designed to distribute emergency power. Upon loss of power in load center #1, the diesel generator will automatically pick up the power distribution to load center #5 as described in Section 2.12.3. Load center #3 feeds power to the experimental bus on each floor of the containment building.

The reactor is scrammed if the incoming electrical power frequency changes by more than 0.5 of a cycle. A voltage drop to the equivalent of 92 out of a 120 volt circuit will also scram the reactor.

2.12.3 Emergency Power

If the normal power supply is lost, an emergency power supply prevents interruption in power to such items as the elevator, secondary shutdown pump, fill and flush pumps, air compressors, instrumentation system, vital equipment in experimenters' systems, and building ventilation system. The system is wired as necessary to assure that continuous power is supplied to all devices which must function in the event of a power outage for personnel or plant safety.

The emergency power is provided by a heavy duty, 150 kw diesel generator set which is suitable for continuous operation . It is rated at 480 volts with an 0.8 power factor. During reactor operation above 50 kw, the diesel-generator is operated at approximately 30% of capacity. In the event normal power is lost, the diesel generator will automatically pick up the emergency load through a reversed power relay which clears the generator of all non-essential loads for shutdown conditions. Controls for the emergency generator are located in the control room. The diesel generator set installation is complete with the necessary underground diesel fuel storage tanks, fuel transfer pumps, and filters. The fuel oil storage capacity will permit 24 hours of full-power operation when the fuel storage tank reaches its lowest allowed level. Failure of the emergency power source, even though normal power is available, will scram the reactor.

2.12.4 Lighting System

The interior lighting system consists of fluorescent lighting for general illumination and incandescent portable lighting for supplemental use in the canal and pool areas. Exterior lighting is provided for security and personnel safety.

The emergency lighting for the plant consists of semi-portable units with a self-contained battery system which maintains its charge from the normal power system. On loss of normal power, the units are energized by an automatic relay. The units are located such that the lighting is adequate for personnel exit, instrument reading, and control room operation. The emergency lighting is designed to operate for 8 hours after the loss of normal power.

2.12.5 Instrument Air

Instrument air is supplied from two horizontal, single stage, double acting compressors capable of a discharge pressure of 100 psig. Each compressor will deliver 50 SCFM of free air. The air receiver tank is 36" in diameter and 9 feet high. One of the compressors operates between 70 and 90 psi and the other operates between 80 and 100 psi. This system provides one air compressor as a reserve in case of an excessive load. The relief valve on the air receiver is set at 120 psi.

2.12.6 Nitrogen System

A nitrogen distribution system has been added to the plant with outlets on each floor in the containment vessel. The system is protected by a safety relief valve. Traps with blowdown valves are located at the lowest points in the system.

This nitrogen system supplies the primary system pressurizer and experimental equipment. The liquid nitrogen receiver is located on a pad adjacent to the cooling tower. The storage capacity of this receiver, considering the nitrogen as gas under standard conditions, is 125,000 cubic feet. The system is capable of delivering 2,000 cubic feet per hour of gas at standard conditions at a regulated pressure of 175 psig.

2.13 Control and Office Building

The reactor control and office building is located adjacent to the containment building as shown in Figure 1.1. The layout details are shown in Figure 2.23.

The major portion of the structure is of conventional frame construction and accommodates office space for operating personnel and experimenters' personnel in addition to decontamination and health physics facilities. The vault is constructed of concrete and used to store special nuclear materials. The control room provides space for the nuclear and process control room walls and floors are of concrete construction and concrete blocks on the side of the containment building adjacent to the control room, as shown in Figure 2.17 and the frontispiece photograph, provide supplementary shielding between the containment vessel and the control room for reactor operators in the event of an incident.

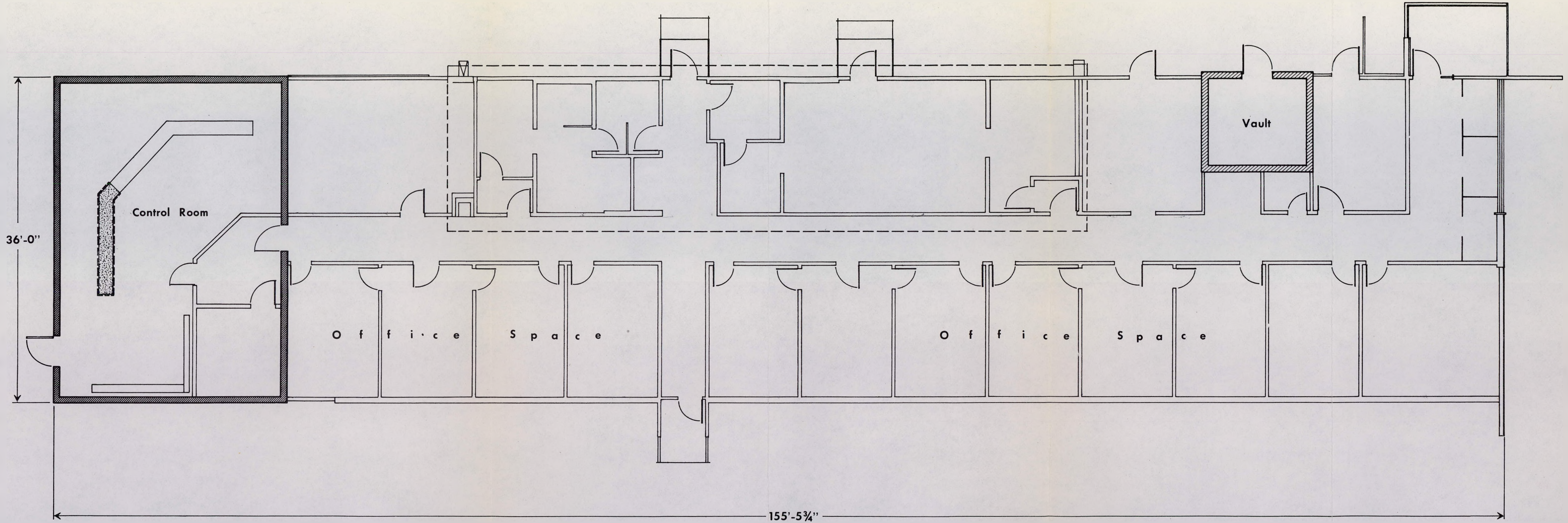

Figure 2.23 OFFICE BUILDING

Figure 2.23

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SECTION 3

REACTOR SAFEGUARDS EVALUATION

3.1 Introduction

Significant accidents which conceivably could occur during operation of the reactor are discussed in this section. An evaluation of the sequence of events which would bring the reactor under control demonstrates that the reactor can be safely scrammed and cooled following any single or double combination of mechanical failures or operational errors. The maximum credible accident is derived by assuming that three equipment components fail resulting in the simultaneous failure of two major and critical reactor systems.

The several years of GETR operations have demonstrated the adequacy of established controls to protect against the described accidents. These controls are continuously improved as experience is gained on the reactor and experimental facilities which results in further reductions in the probability of such accidents.

3.2 Mechanical Accidents in Reactor System

Significant mechanical accidents involving the reactor system and which have a reasonable probability of occurrence are described in the paragraphs below. The severity of these accidents vary in terms of maximum temperatures attained, time or cost to restore reactor to normal operation, or potential danger to personnel; but none of the accidents result in reactor core damage severe enough to release fission products to the primary coolant. The margin of safety is sufficiently great in every case that even though initial power, flow, pressure, and temperature are at the most unfavorable limits of their respective operating ranges, as determined by control tolerances and operating procedures, the reactor core is not damaged during the course of the accident. In every accident analyzed, the only effect due to initial power, flow, and other variables being at their most unfavorable limits is a slightly higher temperature, or slightly more extensive nucleate boiling during the course of the accident.

3.2.1 Loss of Normal Electrical Power

A loss of the normal electrical power source results in a reactor scram when the primary collant flow decays to 85% of normal. The burnout safety factor for the hot spot on reactor fuel is approximately two. The upper emergency cooling valves open upon loss of electrical power. This permits the reactor to be subcritical by the time the pressure drops as a result of opening the emergency cooling valves. Downward flow through the reactor vessel gradually decreases and finally reverses. Bulk boiling may occur in the hottest fuel channels, but the energy release by them would be mostly decay heat and natural convection cooling is adequate to protect the core.

3.2.2 Loss of Secondary Flow

Loss of secondary flow could result from failure of secondary coolant pumps, a secondary pipe rupture, or accidental closure of secondary valves. The accident results in a slow increase in temperature of the primary coolant leaving the heat exchanger. The reactor is protected against high temperature by an automatic scram on high reactor inlet temperature.

3.2.3 Loss of Pressure

The loss of pressure accident results from either a pressurizer rupture or a pressurizer valve failure whereby pressurizer pressure is reduced to atmospheric. A scram is initiated upon loss of pressure, but reactor coolant flow is unchanged. The scram is sufficiently rapid so that there is no chance for damage to fuel elements from thermal effects.

3.2.4 Loss of Instrument Air

Normal instrument air pressure is maintained at approximately 100 psig and the reactor is manually shut down when this pressure becomes less than 50 psig. There is no hazard from rapid loss of instrument air and thus no need for automatic reactor scram.

3.2.5 Rupture of Primary Coolant System

The loss of primary coolant pressure due to a moderate break in the primary coolant system would not interrupt or divert a significant portion of the normal coolant flow. Coolant loss would be made up by pool water flowing in through the reactor emergency cooling check valves, and positive flow through the reactor continues at a normal or almost normal rate. Such an accident would proceed in the same manner as the accident described in Section 3.2.3. In the event of a large break in the primary coolant system, such as one inlet line completely severed at a point outside the biological shield, there would be a loss of pressure and a decrease in flow. A reactor scram would be initiated by the pressure loss, the emergency cooling valves would open when the pressure differential across the reactor vessel dropped to 10 percent of normal, and the siphon break valves would open when the pressure at the highest elevation in the primary system inlet line fell below atmospheric. If the pool was drained through the break so that air was admitted to the system, a free surface would form inside the reactor vessel which could not recede below approximately five feet above the core. Net flow through the reactor vessel either by forced or natural circulation would stop and there would be bulk boiling in the core until decay heat dropped to a sufficiently low level. Pool water would flow in through the emergency cooling check valves to replace that which boiled away.

3.2.6 Loss of Pool Water

An audible alarm gives warning if the pool water recedes, and the operator manually scrams the reactor and opens valves to admit demineralized water (and raw water if necessary) to the pool. Only an event such as a violent earthquake could damage the structure in the vicinity of the reactor severely enough to result in an outward flow of pool water at a rate in excess of that which could be made up by the fill and flush pumps (400 gpm) and the raw water gravity supply (initially over 700 gpm depending on the water level in the tank). The primary cooling system, if still intact, would continue to cool the reactor effectively. If the primary system and pool were severely ruptured beyond makeup capacity, the accident would proceed similarly to the maximum credible accident described in Section 3.4, except that the energy release would be limited to less than 75 percent of that considered in the maximum credible accident.

3.2.7 Failure to Scram

A scram may be initiated manually by the operator or automatically as indicated in Table 2.4. Transducers are selected and placed so that more than one instrument will scram the reactor if an important malfunction in the nuclear or cooling system occurs. In the case of mechanical accidents, for example, loss of flow is sensed by one differential pressure pickup across an orifice and by another differential pressure pickup across the reactor vessel. The circuitry between the transducers and the control rod magnets is fail-safe in that failure of the circuit activates a reactor scram. The six control rods have independent release mechanisms to prevent any influence on other control rods if one failed.

A scram failure is considered highly improbable since it requires a coincident failure of at least two relay armatures, or the accumulation of at least four, burned-closed contacts during the period of operation following the last preventive maintenance check. Failure of the reactor to scram due to operating conditions slightly in excess of the established limits does not place the reactor in serious danger since these limits are established with a margin of safety. The reactor could be scrammed manually under such conditions with adequate safety. Operational monitoring of critical parameters is required at short intervals during operation.

If failure to scram were associated with a malfunction of a serious nature, such as loss of primary coolant flow from pump motor failure, the situation would lead to the maximum credible accident as described in Section 3.4. The probability of several such coincident failures is almost negligible.

3.3 Operational Accidents

Three types of operational accidents could conceivably occur: (1) dropping a fuel element on a fully-loaded core during critical testing; (2) reactor startup with improper core loading; and (3) a startup accident resulting from the maximum rate of control rod withdrawal combined with simultaneous failure of the period scram circuit.

3.3.1 Dropping a Fuel Element on a Fully Loaded Core

Operating procedures specify that fuel elements will not be allowed in the vicinity of the reactor pool immediately prior to nor during any operation of the reactor as a critical assembly with the pressure vessel head removed. However, this accident postulates that this procedure has been violated and a fuel element is accidently dropped on a fully-loaded core during critical assembly testing. Depending upon the resting position of the fuel element, a maximum reactivity insertion of 0.5 percent can be introduced. The period trip and flux trip would scram the cocked rods and limit the excursion. The Borax tests demonstrated that reactors of the type that include the GETR can withstand a reactivity increase of 1.7 percent without damage to the fuel. Since the 0.5 percent insertion from this accident is well below the demonstrated 1.7 percent, no physical damage would be expected.

3.3.2 Reactor Startup with Improper Core Loading

Precautions against core arrangements and loadings which could add more reactivity than planned are described in Section 4.4. In the improbable event that these controls fail and a large reactivity addition is thereby allowed, the period and flux instrumentation, provided during refueling operations, would scram the cocked rods. A smaller erroneous reactivity addition would be detected during startup when criticality occurred sooner than predicted. The cautious approach to criticality and startup instrumentation described in Section 4.4 protect against any damaging results. Should such a situation occur, the reactor will be shut down and the cause investigated.

3.3.3 Reactor Startup Accident

A startup accident can be one of the most severe operating accidents involving an increase in reactivity. In formulating such an accident, it is assumed that the reactor is started with normal coolant flow from source level by withdrawing the control rods at the maximum rate of 7.5×10^{-4} Δk/sec, the period scram circuit and the low power trips fail to operate, and the resulting excursion is limited by a high neutron flux level scan at 120% of reactor power. The effect of fuel and moderator temperature coefficients of reactivity is assumed to be negligible. This accident was used as the basis for the specified maximum control rod withdrawal rate and the 60 millisecond scram delay time.

Other parameters used in the evaluation include:

Ratio of source power to rated design power	1×10^{-7}
Overflux trip level	60 MW
Initial scram acceleration	3.1 $\Delta k/sec^2$
Thermal neutron lifetime	5.5×10^{-5} sec
Reactor period	0.12 sec

The accident evaluation assumed no contribution of negative reactivity from energy absorption within the core. This led to an excursion with about 25 MW - sec and a peak power of 110 MW. The burnout margin for the hot spot of the fuel elements would be about 1.1 at the peak of the transient. Thus, burnout of the fuel is not possible although there will be transient nucleate boiling.

3.4 Maximum Credible Accident

The probability is extremely small that an accident could reach the magnitude of the maximum credible accident as described in this section. Such an accident would require at least concurrent failures of the cooling system and scram system.

3.4.1 Summary

The minimum number of coincident component failures which could lead to a maximum credible accident is three. Based on design and operating performance, these three components are among the most reliable in the system. Two of these failures concern relays which must coincidentally fail with the armature stuck in the energized position. Any lesser failure or malfunction of either relay will not prevent a scram. The third failure must be a pump failure, pump motor failure, or pipe break. Loss of the main power source does not qualify as a failure because this would scram the reactor by de-energizing the rod magnets.

With the reactor operating at full power, the loss of coolant would result in a rapid rise in fuel element temperature. The fuel first becomes steam blanketed in the central regions of the core which causes the fuel temperature to rise rapidly above the melting point. A chemical reaction is assumed to occur between aluminum core material and the water when the fuel reaches a critical temperature somewhat above its melting point. It is postulated that sufficient energy is generated to melt the entire reactor core before the reactor becomes subcritical. It is further postulated that a 25 percent chemical reaction occurs between the molten aluminum and the water. A high temperature gas bubble containing the reaction products expands forming steam, ruptures the pressure vessel, expels water from the pressure vessel and pool, and does work on the missile shield. Some of the gas may go into the equipment space and some may rise up through the pool. In either event, the hydrogen evolved in the reaction is assumed to ignite and burn when it mixes with the oxygen of the containment vessel. Building air is heated by the gas bubble and hydrogen combustion. The total energy released amounts to 730,000 Btu which causes a containment vessel pressure of 4.3 psig and temperature of 250°F which is less than the containment vessel design values. The nuclear excursion contributes 200,000 Btu, the metal-water reaction contributes 280,000 Btu and the hydrogen combustion contributes 250,000 Btu. Chemical Reactions involving the experiments are relatively insignificant. Use of missile shielding prevents any high energy particles from penetrating the containment vessel wall. The maximum radiation dose rate at the nearest site boundary to the reactor would be 325 millirem per hour ten minutes after the accident.

3.4.2 Pressure from the Maximum Credible Accident

The pressure which results during the maximum credible accident, as defined in Section 3.4, has been analytically determined. The analysis is conservative since it considers that all events occur which could contribute although the probability of this is extremely small, and assumptions regarding these events are indicated.

3.4.2.1 Nuclear Incident

The magnitude of the nuclear incident is derived from the energy required to melt the complete reactor core. At this point, the core is considered to be sufficiently deformed to cause it to become subcritical. In an actual incident, it would be expected that as the central portions of the core melted, the resulting steam explosion, which has been shown to occur in experiments by Long[1] would disperse the core components and stop the nuclear reaction. The 200,000 Btu figure for complete core melting is considered an approximate equivalent to the actual case. The destructive experiment with Borax as reported in AECD-3668 released approximately 135,000 Btu.

3.4.2.2 Chemical Reaction

A chemical reaction between the water in the core and 25 percent of the fuel and its associated cladding has been considered in this accident analysis. While there are still a number of important questions on the nature of the metal-water reaction that are unanswered, the assumption of a 25% reaction between aluminum and water seems to be overly pessimistic and thus to leave a wide margin of safety. Considerable information has been accumulated on the metal-water reaction (2) over the last few years.

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(1) Long, George, "Explosions of Molten Aluminum in Water - Cause and Prevention", Metal Progress, Vol. 71, No. 5, May, 1957.

(2) Epstein, Leo F., "Recent Developments in the Study of Metal-Water Reactions". Chapter 7-7, p. 461 in Progress in Nuclear Energy, Series IV, Vol. 4, Technology Engineering and Safety. Pergaman Press (Oxford 1961). See also Reactor Technology 3 273 (Feb. 1962).

For example, from Figure 3 of the reference cited, it would require molten aluminum, with a 500 to 600 micron mean droplet diameter, at 1400°C to produce a reaction of this extent. In order to obtain a violent aluminum-water reaction, it is necessary that the metal be not only molten, but, according to some studies, actually above the melting point (932°K) by several hundred degrees - up to 1440°K perhaps.[3] In addition, a fine state of subdivision in the micron range for droplet sizes is required. How this exact combination of conditions can be achieved in a reactor incident is rather difficult to imagine and there is some question as to whether a chemical reaction of this kind has ever, in reality, been observed in a reactor accident. If the reaction

$$2Al + H_2O = Al_2O_3 + 3H_2$$

did occur, about 4200 calories of heat would be released for each gram of metal consumed.

3.4.2.3 Gas Bubble Formation

The gas bubble formed from the chemical reaction consists of 81.5 pounds of aluminum oxide and 4.8 pounds of hydrogen. It is assumed that there is no steam formation nor mixing with water due to the short time intervals involved in the process. Consideration of water and steam formation results in reduced amounts of heat getting to the air in the containment building and a resultant lower building pressure.

3.4.2.4 Expansion of the Gas Bubble

The expansion of the gas bubble which initially contains water vapor, aluminum oxide, and hydrogen is largely dependent upon the thermodynamic state of the aluminum oxide. The bubble expands as it travels upward and the increase in volume forces water ahead of it. Water vapor formation, discussed above, has been considered insignificant in the bubble thermodynamic calculations. It will be accounted for, however, in obtaining the final building pressure.

(3) Epstein, Leo F., Nuclear Science and Engineering 10 247 (July, 1961)

The path of the bubble expansion in the actual case is in two directions; downward and upward. It may be shown that the bottom of the reactor pool fails before the anchors of the upper missile shield are moved. Thus, in an actual incident, large amounts of gas are moved into the sub pile rooms and the machinery access rooms rather than into the containment space above the pool. This is not considered in the calculations since conceivably all the heat of gases going into the lower portions of the building could become communicative with the space above the third floor level. The resulting expansion considered in the calculations gives a pessimistic result since more heat is added to the air in the space above the third floor than would be expected.

The high temperature thermodynamic properties of aluminum oxides are well known[4], both below and above the melting point (2288° - 2318°K). From these data, and the corresponding characteristics of the hydrogen gas which is the other product of the chemical reaction, it is possible to compute the total energy release and the amount delivered to the air.

The vaporization behavior of Al_2O_3 is also known[5]. It vaporizes by disassociating into a variety of gaseous species: Al(g), Al_2O(g), AlO(g), Al_2O_2(g), O(g) and O_2(g). The principal reaction in the range of 2300° to 2600°K is

$$Al_2O_3(e) \;=\; 2AlO(g) \;+\; O(g)$$

The total pressure above Al_2O_3(e) is given by the equation[6]

(4) See for example JANAF Interim Thermochemical Tublos, Volume 1, prepared by the Thermal Laboratory, Dow Chemical Co. (Midland, Mich.) where data for Al_2O_3, crystal and liquid are given up to 6000°K, based on the best information available at the date of issue, September 30, 1961.

(5) Elliott, J. F. and Gleisov, Molly, Thermochemistry for Steelmaking, Vol. 1. Addison-Wesley (Reading, Mass., 1960), p. 277. See also (1) Brewer, Land Searey, A. W., J. Am. Chem. Soc. 73 5308 (1951); and (2) DeMaria, G., Drowart, J. and Ingraham, M. J., J. Chem. Phys. 39 318 (1959).

(6) Kelley, K. K., Contributions to the Data on Theoretical Metallurgy. III The Free Energies of Vaporization and Vapor Pressures of Inorganic Substances. U. S. Bureau of Mines Bulletin 383 (Washington, 1935).

$$\log_{10} P \text{ (atm.)} = +8.415 - 27320 \text{ (1/T)}$$

where T is in °K. From this, the following values are computed:

Pressure (atm.)	Temperature (°K)
0.001	2395
0.01	2624
0.1	2901
0.25	3030
0.5	3137
1.0	3250 (Boiling Point)

Also from the equation given, the heat associated with the vaporization process is 125,000 calories/mole or 1226 calories/g (2207 BTU/lb). Note that the "entropy of vaporization" at the normal boiling point is 38.5 cal./mole - °K instead of the Trautau's Rule value of about 21 entropy units; this arises because of the complex value of the vaporization process in this case - disassociation, etc.

By using these parameters for Al_2O_3 and the available information on hydrogen, a temperature of 4000°F was calculated using the conservative conditions outlined above and considering that the gas bubble expansion has filled the tank from the core centerline to the missile shield.

3.4.2.5 Emission of the Hydrogen Gas Bubble

The hydrogen gas bubble is emitted from the pool space at a temperature of 4000°F and may be expected to ignite on passage into the containment vessel air.

3.4.2.6 Heating of the Building Air

The 230,000 ft^3 of containment vessel air is heated by the hydrogen gas and aluminum oxide entering from the reactor pool and by the combustion of the hydrogen gas. Pressure in the building is determined from the equation of state once the air temperature is calculated. Partial pressures of water vapor are added to the pressures calculated.

3.4.3 Missile Shielding

This section presents a discussion of the consequences of an explosive energy release from the maximum credible accident and a description of the shielding designed to protect the containment building from damage by the variety of high-energy missiles and shock waves which might be initiated. Missile shielding is provided over the top of the reactor and between the pool and canal. In addition, blast mats were placed in the biological shield to minimize damage to the concrete. Both the shield over the reactor and the shield in the canal can be removed during refueling operations.

The thickness of the missile shields and blast mats were based on results of the analysis described in this section. The energy released by the nuclear incident is converted to an "equivalent" amount of TNT, and the pressures and impulses from the shock wave are estimated from data given by R. D. Cole[1]. It is known that the results obtained in this manner provide greater shield thicknesses than would be required by an actual accident involving an energy release of 480,000 Btu due to the core meltdown and chemical reaction as postulated for this maximum credible accident.

3.4.3.1 Upper Shield

The shield over the reactor pool consists of an octagonal-shaped steel slab eleven feet wide and approximately thirteen inches thick. It is mounted on wheels to permit removal during refueling and restrained when in place to withstand a total upward force of about 800,000 lbs. This is believed to be adequate, since the bottom of the pool would collapse and relieve the pressure before a force of this magnitude could be attained.

3.4.3.2 Canal Door

To guard against the possibility that test capsules, vessel fragments, or other high-energy missiles may be projected through the opening between the pool and canal

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(1) Cole, R. D., "Underwater Explosions", Princeton Univ. Press, 1948

a steel shield is mounted in the opening and used as a gate leading to the canal. The shield is approximately three feet wide and twenty feet high. Its thickness of six inches was determined to be adequate as protection against penetration by missiles based on the anticipated energy spectrum of missiles and shock wave pressures.

3.4.3.3 Blast Mats in Biological Shield

It is estimated from data given by R. D. Cole that the peak pressures in the shock wave impinging upon the inner surface of the biological shield is in the neighborhood of 23,000 psi. In order to minimize damage to the concrete, the blast mat described in Section 2.8.2 was incorporated in the biological shield. The blast mat configuration is based on information presented in ANL-5651[2], which describes the work done by the Armour Research Foundation on the blast effects of internal explosions.

3.4.3.4 Pool Bottom

The bottom of the reactor pool is a reinforced concrete slab two feet thick and designed to carry a total weight of approximately 150,000 lbs. It is estimated that a total force of about 450,000 lbs. will cause failure. This force is substantially less than required for missile shield failure.

3.4.3.5 Beam Port

If an explosive energy release should occur, part of the shock wave energy would be channeled into the beam port and absorbed by the installed experimental equipment. If experimental equipment is not installed, a plug consisting of alternate layers of steel and redwood is placed in the outer end of the port and backed up by a thick steel plate anchored into the biological shield.

3.4.4 Radiological Effects of the Maximum Credible Accident

The radiological effects of the maximum credible accident due to

(2) Porzel, F. B., "Design Evaluation of BER (Boiling Experimental Reactor) in regard to Internal Explosions" January, 1957

the release of fission products are described in this Section.

3.4.4.1 Radioactive Materials Available for Environs Effects

The maximum credible accident would cause fission products to be dispersed by the energy of the assumed nuclear excursion and chemical reaction. Some of the soluble fission products would then be mixed with the water surrounding the reactor; others would be carried upward with the bubble of hydrogen and water. The bubble would be deflected by the missile shield and disperse into the enclosure volume. Additional dispersion would also result from the hydrogen combustion. As the bubble expanded it would mix with the air in the enclosure. Then, as it made contact with the enclosure shell, the bubble would be cooled; the pressure within the enclosure would be reduced and the water which had been vaporized by the occurrence would be condensed. Some of the soluble radiogases and particulates would be collected by the condensate as it dropped back to the lower regions of the enclosure. Radioactive halogens and particulate material would also be removed from the gaseous phase through plating or settling. The quantity of radioactive material in the vapor space would also be reduced through natural radioactive decay and leakage of the gas would be reduced with reduction of the enclosure pressure.

A small fraction of the gaseous fission products would leak out of the enclosure and diffuse downwind from the enclosures. This leakage would not exceed 2% per day at rated enclosure pressure and would be substantially less at the reduced pressure existing in the enclosure after the gas came to thermal equilibrium with the metal of the enclosure shell. The evaluation of off-site exposures resulting from the release of radioactive gases from the enclosure was made on the basis of transport of fission products from the fuel to the enclosure free volume. As discussed in connection with Table 5.6.6.1 and Table 5.6.6.2, the release of fission products is not expected to exceed the following percentages:

100% of the noble gases (Kr and Xe)

25% of the halogen (I and Br)

15% of the volatile solids (Te, Se, Ru, Cs)

0.3% of all other solid fission products

The inventory of fission products in the enclosure vapor space at various times after the accident is shown in Table 3.1. For this evaluation, no credit was taken for activity washdown or plate-out during the period.

Table 3.1

Inventory of Fission Products in the Enclosure Vapor Space Following Maximum Credible Accident - 50 MW Power

	Inventory - Megacuries						
Time After Accident	10 min.	1 hr.	3 hrs.	10 hrs.	1 day	10 days	30 days
Noble gas	11.0	8.1	7.2	6.0	4.0	1.0	.07
Iodines	2.8	2.5	2.2	1.5	1.1	.12	.01
Solids	1.9	1.1	.8	.6	.5	.2	.1

3.4.4.2 Direct External Gamma Radiation from Enclosure

The quantities of fission products described in Section 3.4.4.1 were also assumed as the contributors to the direct external gamma radiation from the enclosure. This direct radiation is a sensitive function of the gamma energy levels of the radioisotopes present due to the variable shielding effect provided for the different gamma energies by the large thickness of air available between the enclosure and the site boundary. Therefore, this evaluation was made by calculating the exposure contribution from each gamma radiation level from each isotope in the noble gas, halogen, and volatile solid fission product categories, together with their appropriate daughter fission products, and by considering shielding and build-up factors for both air and the steel enclosure wall. Because of the non-symmetrical shielding provided at the reactor, approximately 50% of the enclosure free volume would be visible from the west and northwest of the site and up to 85% of the free volume could be visible from other directions.

The results of these evaluations at distances ranging from 2700 feet which is the distance to the nearest site perimeter to two miles is shown in Table 3.2. It may be noted that the maximum dose rate at the site perimeter is 325 millirem per hour and at distances greater than 1 mile the maximum dose rate is only 15 millirem per hour. The table also shows how the dose rates are reduced as a function of time. It may be seen that the total direct gamma radiation from the enclosure would contribute no more than 625 millirem to any off-site individual.

Table 3.2

Direct External Gamma Radiation From Enclosure
Dose Rates After Accident-Milliroentgen Per Hour

Distance	10 min.	1 hr.	10 hr.	10 day
2700 feet	325	280	170	30
1 mile	15	11	8	2
2 miles	.1	-	-	-

3.4.4.3 Leakage from Enclosure

The leakage rate of various fission product groups was determined based upon the enclosure free volume fission product inventory, as outlined above, and upon the instantaneous enclosure pressure. The radiological effects of the leakage were evaluated for four atmospheric conditions: strong inversion, neutral and unstable conditions, with a wind speed of 1 meter/sec and, for neutral conditions, with a wind speed of 5 meters/sec.

3.4.4.4 Meteorological Diffusion Evaluation Methods

The atmospheric diffusion methods of Sutton were used for the neutral and unstable cases. Due to the empirically indicated inadequacies of the Sutton method for inversion conditions, calculation methods based on Hanford diffusion results, as outlined in Report HW-54128[7] were used for the inversion cases.

Weather Conditions: This evaluation assumed that the weather conditions involved no precipitation and that the incident occurred during hot summer weather. Precipitation would deposit more contamination close to the plant than this evaluation indicates, thus reducing contamination levels further away.

Elevation of Release: Leakage from the enclosure is considered to occur near the ground level. This appears reasonable as most enclosure penetrations are near grade. If the postulated leakage occurred at some significantly different height, such as by emission from the stack, the off-plant consequences of passing cloud dose, ground deposition, and possible inhalation would be vastly reduced because much of the radioactivity would remain aloft until gaseous diffusion resulted in substantial dilution of the material.

Initial Dilution by Building Wake: This evaluation recognizes that initial immediate dilution of the leakage will occur due to the turbulent wake of the enclosure structure produced by the passing wind. It is estimated that the effective wake cross-section is of the order of one-half of the vertical cross-section of the enclosure structure. No additional immediate dilution by other nearby structures is considered.

This effective wake has been equated to a semi-circle of equivalent area centered at ground level. Centering the initially diluted leakage at some greater height would reduce the off-plant effects of leakage from those evaluated. It is noted that the radius of the equivalent

(7) HW-54128, "Calculations on Environmental Consequences of Reactor Accidents", Interim Report by J. W. Healy, December 11, 1957.

semi-circle is about the same as the enclosure radius. To obtain an estimate of this initial dilution of the leakage, the radius of the equivalent semi-circle was estimated to represent about 1-1/2 standard deviations of the cloud width. From these considerations, virtual source points were calculated at various upward distances dependent upon the diffusion conditions, and are:

200 meters for strong inversion - 1 m/s wind speed
110 meters for neutral conditions- 1 m/s wind speed
50 meters for unstable conditions - 1 m/s wind speed

These estimates of the virtual source distances agree generally with the methods of Holland for the neutral and unstable cases and are more conservative for the strong inversion case.

3.4.4.5 External Radiation Dose from Passing Cloud

Evaluation of effects of passing cloud air concentrations downwind were estimated using the Sutton and Hanford methods as outlined in Section 3.4.4.5. Particular emphasis was taken in this evaluation of the conversion from air concentration to integrated dose for the passing cloud effect. Due to the radioactive decay of the equilibrium fission product mixture which occurs during the post-accident period, the conversion from concentration to dose becomes more favorable in reducing dose as the decay period available increases. For the noble gas, halogen, and solid fission product groups, the concentration required in an infinite cloud to produce a certain dose was evaluated for the radioactive decay periods of interest in the post-accident period. Selected values of the air concentration in an infinite cloud, in units of microcuries per cubic centimeter, which will produce a dose rate of one mrad per hour with hemispherical geometry are given in Table 3.3.

Radiation from ground deposition for the duration of the accident is shown in Table 3.4.

Table 3.3

AIR CONCENTRATIONS ($\mu c/cc$) GIVING ONE MRAD PER HOUR DOSE RATE

Decay Time	Noble Gases	Halogens	Solids
1 Hour	1.6×10^{-6}	0.78×10^{-6}	1.4×10^{-6}
4 Hours	2.3×10^{-6}	0.79×10^{-6}	2.5×10^{-6}
8 Hours	3.1×10^{-6}	0.88×10^{-6}	2.5×10^{-6}
16 Hours	4.2×10^{-6}	1.0×10^{-6}	2.6×10^{-6}

Table 3.4

INTEGRATED DIRECT RADIATION DOSE FROM GROUND DEPOSITON (R)

Diffusion Condition	Distance from Reactor 1000	2000	2700	5000
Inversion 1 m/s windspeed	1	0.6	0.5	0.3

The dose from the passing cloud based on uniform concentration and infinite cloud considerations was then corrected for the finite cloud size and Gaussian distribution of cloud concentration. For the various diffusions evaluated, and for cloud sizes calculated at the 2700 feet to two miles distance, the ratio of finite cloud dose to infinite cloud dose was found to range from 0.07 to 0.7. The reduction of cloud concentration at the distance evaluated, because of prior deposition on the ground of halogens and solids, was factored into the dose from the passing cloud. This correction was of small magnitude since most of the passing cloud dose was due to noble gases.

The results of these evaluations are shown on Table 3.5.

Table 3.5

AVERAGE DOSE RATE RECEIVED FROM PASSING CLOUD OF RADIOGASES (mrads/hr)

Atmospheric Condition	Wind Speed	Distance From Reactor 2700 ft.	1 mile	2 miles
Strong Inversion	1 meter/sec	75 mr/hr	60 mr/hr	40 mr/hr
Unstable	1 meter/sec	6	3	1
Neutral	1 meter/sec	16	12	8
Neutral	5 meter/sec	8	2	1

At the nearest site boundary the passing cloud dose, for inversion conditions with low wind speed, is approximately 150 mrads in the two hours during evacuation. For other conditions of higher wind speed, the dose rate is much lower depending upon the diffusion condition existing. At the one-mile distance, a two-hour dose of about 120 mrads is indicated with the least favorable wind speed and diffusion conditions with similar reductions for the more probable higher wind speeds. These dose calculations assume that the receptor is on the center of the cloud path continuously for the period evaluated and that no incidental shielding, such as that provided by housing, is available.

3.4.4.6 Internal Dose to Thyroid

Internal exposure to the thyroid gland from inhalation of the fission product mixture in the passing cloud is primarily due to iodine radioisotopes. This exposure was evaluated considering the dose from thyroid deposition of iodine-131, 133 and 135. Other iodine radioisotopes with half lives of 2.3 hours or less were not included, considering their low rem-per-microcuries ratio for lifetime dosage considerations and because of the estimated 3 to 6 hour thyroid uptake time after the material is inhaled. The lifetime thyroid dose was evaluated for the three iodine isotopes considering a breathing rate of 20 liters per minute and a thyroid deposition of 23% of that which was inhaled as recommended by the International Commission on Radiological Protection.

The infinite thyroid dose for inhalation during the two hour evacuation period, at various distances and atmospheric diffusion conditions, is shown on Table 3.6. At the site perimeter directly downwind of the reactor, the infinite thyroid dose from inhalation is less than 104 rads for exposure during the first two hours under the least favorable diffusion conditions. For the other more probable diffusion conditions, the dose is much smaller. The similar infinite thyroid doses for inhalation at a distance of one mile during unstable atmospheric conditions is less than 1 rad even without considering any change of wind direction during the period.

Table 3.6

INFINITE THYROID DOSE RECEIVED FROM BREATHING RADIOIODIDES CONTINUOUSLY FOLLOWING MCA (RADS)

		Distance from Reactor		
Atmospheric Condition	Wind Speed	2700	1 mile	2 miles
Strong inversion	1 meter/sec	104	6.7	18
Unstable	1 meter/sec	1.8	1.0	0.2
Neutral	1 meter/sec	7.3	4.5	0.7
Neutral	5 meter/sec	3.7	0.7	0.1

The analysis considers reduction of the airborne halogen fission product inventory by deposition at a removal rate of 1 x 10^{-3} fraction per second. This rate is established on the basis of an estimated deposition velocity of 1 cm/sec and an average 10 meter distance to a deposition surface, but does not consider washout by natural condensation.

3.5 Radiation Hazards Evaluation

Credible accidents that may release radioisotopes to the containment vessel and to the atmosphere, thus posing a potential radiation exposure, are described below. The discussion includes an appraisal of the likelihood of a given release, an evaluation of the consequences, an a discussion of ways and means for bringing the release under control. The hazards involved range from a very low level release of small quantities of short half-lived nitrogen-16 to the maximum credible accident involving a large portion of the fission products in the core as described in Section 3.4.4. The amount and rate of release for most accidents is limited by (1) automatic reactor shutdown to prevent further production of fission products, and (2) automatic controls to contain the fission products already released from the fuel.

Several years of GETR operating experience have established the character of radiation hazards in operation and maintenance of the GETR. The plant has proved to be extremely amenable to control of radiation problems including the case of operating with defective fuel in the reactor core. Background of I-131 in the primary water has been about 10^3 d/m/ml, and this level of fission product radioactivity has not interfered with plant operation. Release of radioactivity from the reactor water to the enclosure as a result of such things as spills, leaks, emergency cooling trips, has also not presented unmanageable hazards.

3.5.1 Leak in the Primary Coolant System

The primary piping system including the heat exchanger, pumps, and valves is completely contained within the shielded equipment space inside the containment vessel. This space is not accessible during operation, due to the radiation from nitrogen-16 in the primary water. The ventilation system for the building is so designed that the air sweeps through this equipment space directly into an exhaust duct to the stack.

Since the primary system coolant water is at a pressure of 125 psig, there may be some leakage from mechanical pump seals, valve stems, and flange gaskets. This leakage carries with it nitrogen-16, sodium-24, and perhaps traces of other radioactive corrosion products from the primary water. Gases released from this leakage pass from the shielded cell into the ventilation exhaust for release from the stack. A radiation monitor system warns operators of increasing levels of radioactive gases within the stack and closes the containment vessel if a preset radiation level is exceeded. Water leakage containing non-volatile fission products drains to a sump in the basement; from there it is pumped to the contaminated waste retention tank. Experience has been that these small leaks present no significant hazard. Nitrogen-16 has a 7.4 second half-life so that travel time from the source of leak to the stack allows decay by 1 to 10 half-lives which further minimizes the hazard. Residue from the water consists primarily of sodium-24 with a 15-hour half-life which will be eliminated from the floor by standard cleanup procedures that avoid spreading the radioisotopes.

If a major leak occurs and, for example, spills 60 gpm of primary water on the floor, the gas activity in the stack will rise to approximately 0.2 μc/cc of air. This assumes that the nitrogen-16 flashes immediately just below a ventilation opening and is pulled into the stack as a small finite cloud. Radioactive gas of this concentration will trip the monitor in the stack, sound an alarm, and close all containment vessel isolation valves. Radioactive gas which escapes from the stack during the time required to close the ventilation isolation valves will expose personnel standing 10 feet from the stack to a radiation dose rate of 35 mr/hr. If the gas diffuses into the atmosphere within two minutes, the total integrated dose will amount to an almost negligible 1 mr. If the reactor is shut down at the time of the radiation alarm, the integrated dose within the containment vessel will be less than 1 mr.

If the leak is of the order of 200 gpm, the primary cooling system depressurizes and automatically scrams the reactor. In this case the exposures external to the containment vessel and within the containment vessel are proportionately higher but are limited to a few seconds duration at this level. The radiation exposure levels are slightly higher than values quoted above for the smaller leak, and remain almost negligible.

3.5.2 Emergency Cooling Condition

Emergency cooling of the GETR is by natural convection cooling of the core. Primary water mixes with pool water immediately following reactor scram and originally it was anticipated that this could lead to substantial release of nitrogen-16 into the containment building. Experience has proved this is not the case. This probably is because the pool flow continues during this time so that water at the surface of the pool has practically no nitrogen-16 even immediately following emergency cooling trips.

3.5.3 Radiation Hazards During Reactor Shutdown Operation

The reactor coolant loop is normally purged and the water held for decay in retention tanks before the system is opened up for shutdown operations such as refueling. This procedure has been effective in virtually eliminating undue hazards during the initial phases of shutdown operations.

SECTION 4

ADMINISTRATIVE AND PROCEDURAL CONTROLS

4.1 Introduction

Administrative and procedural controls have been established to assure safety at the GETR. The organization, policies, procedural controls, and review and approval requirements which govern operation of the GETR are described in this section.

4.2 Organization

The general organization of the Atomic Power Equipment Department is illustrated in Figure 4.1 which shows the components which operate or directly support operation of the reactor. To assure safe and efficient operation, the organization is modified as necessary to reflect changes in laboratory programs and objectives.

The Manager-Reactor Irradiations has overall responsibility for the safe, efficient operation of the GETR within the operating standards and procedures. His organization is assisted by the service components at the Laboratory who provide technical, analytical, maintenance, and administrative supporting functions. The Manager-GETR Operations is responsible for operation of the Plant and directing the activities of the shift supervisors who directly supervise the operation and maintenance activities, train and supervise their crews, and control access to the enclosure during their assigned shifts. The Test Engineers provide technical support related to the experimental programs of the GETR.

The Manager-Nuclear Safety provides counsel and service to the reactor staff in the fields of reactor engineering and analysis, operational physics, nuclear safety, and licensing. The Manager-Reactor Technical Operations provides technical guidance through continuous evaluation of operations and determination of safeguard criteria. His organization develops technical standards for the operation, provides engineering services, and audits. The Manager-Reactor Operational Physics is responsible for determining the physics aspects of the operation which are necessary for the safe and efficient control of the reactor process.

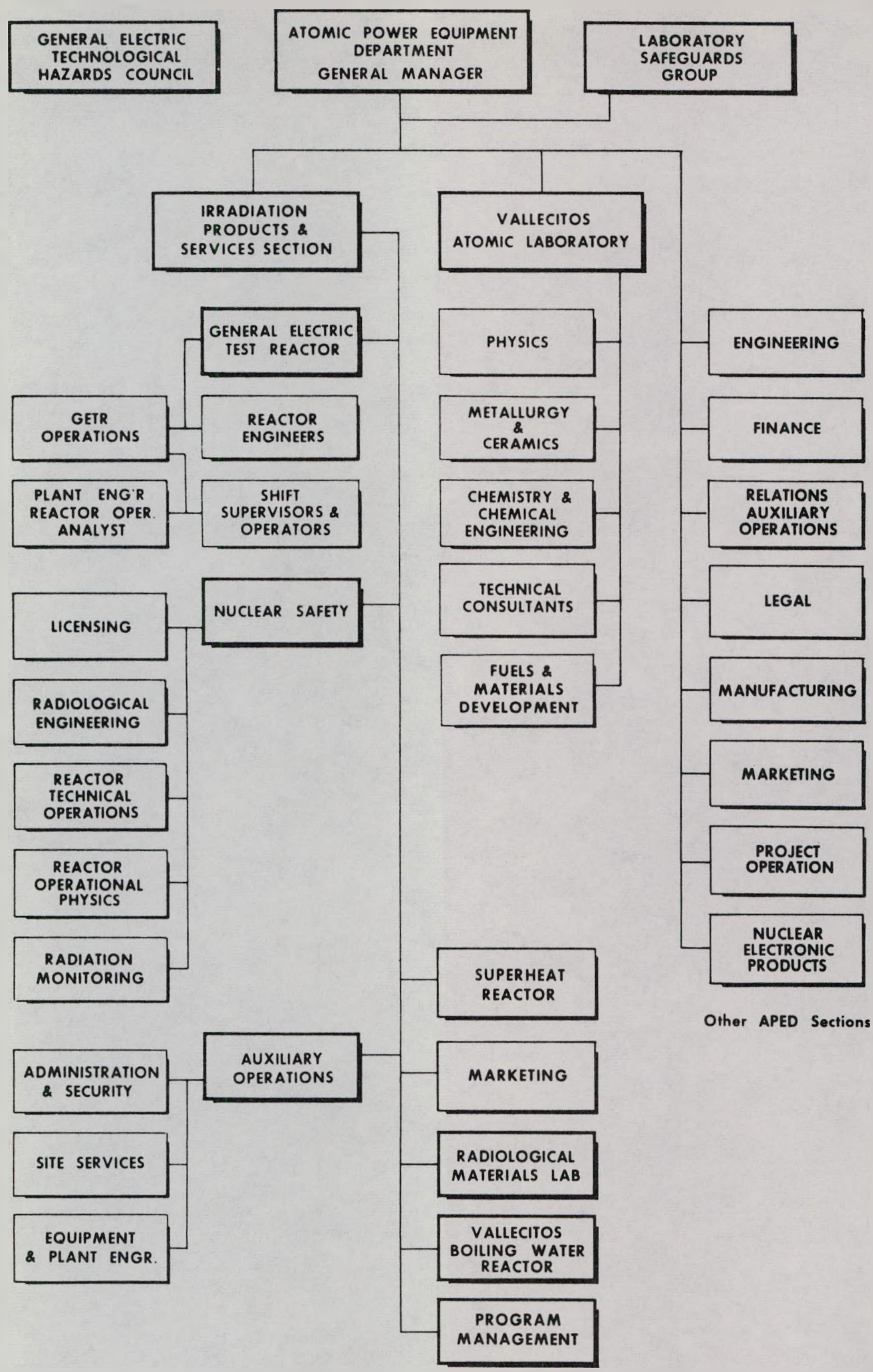

Figure 4.1 ORGANIZATION CHART

The General Electric Technological Hazards Council consists of representatives from each department in the Company's Atomic Products Division and several other Company components particularly suited for this assignment by reason of their experience and knowledge in the field. The functions of the Council include furnishing advice to Company managers on all matters relating to reactor safeguards, reviewing and recommending reactor safeguard design criteria and operational limits, and participating in safeguard studies conducted or sponsored by government agencies or industry-wide activities.

The Vallecitos Laboratory Safeguards Group consists of senior personnel selected by management on the basis of their experience, judgement, and knowledge. The group counsels operating organizations of the Laboratory in regard to safety aspects of proposed reactor operation and experiments. This includes review functions as described in Section 4.4.

4.3 Policies

Established administrative policies related to reactor safety include the following:

1. Responsibility for the safe operation of the reactor within policy limits is assigned to the Manager of Reactor Irradiations.

2. The reactor shall be operated only in accordance with the License Technical Specifications. No change shall be made in the Technical Specifications unless authorized by the Commission.

3. Reactor conditions and variables shall be monitored by reactor and experiment operator observation and by automatic alarm and recording systems.

4. Critical variables shall be indicated in the control room.

5. Automatic, irreversible reactor shutdowns are required when operating variables exceed the limits established to assure the safety of personnel, the reactor, or experiments.

6. Written directions shall be issued for all operations which may affect nuclear safety and for emergencies.

7. Procedures and designs which may affect nuclear safety shall be reviewed by management independent ofand in addition to the management of the facility.

8. Personnel shall be trained in the details of and necessity for strict observance of the written procedures.

9. Management shall regularly audit performance. Occurrences which cause or threaten to cause a disabling injury, overexposure to radiation, or significant property damage shall be investigated. The objectives of these investigations shall be to determine the cause and to recommend action to be taken to eliminate the cause or reduce the effect should it recur.

4.4 Procedural Controls

Procedural controls have been developed to assure that all operations that may affect nuclear safety are conducted in a safe manner. Procedural controls are specified in the Operating Standards, Operating Procedures, Test Procedures, Operational Request Forms, and Laboratory Instructions. They reflect four years of experience with the GETR and are modified as operating conditions dictate. All procedures are subject to Laboratory Safeguards Group review.

The Operating Standards are established to specify operating limits or procedures for safe control of nuclear, experimental, or process variables within the Technical Specification limits. They are prepared by the Reactor Technical Operation, approved by the Manager- Irradiation Services Product Section, and subject to Laboratory Safeguards Group review.

The Operating Standards cover the following general areas:

a. Specific limits for operating variables within the Technical Specification limitations.

b. Specification of and schedules for testing safety devices and instrumentation.

c. Scram by-pass limitations for situations where by-passing is necessary as specified in this report.

d. Any limitation which the Manager-ISPS wishes to place on otherwise authorized operations.

The Operating Procedures are established in conformance with Operating Standards to give detailed steps for operating the plant and for emergencies. They are prepared and issued by the reactor operating organization, and reviewed by the Reactor Technical Operation.

The Test Procedures are established to give the detailed steps for establishing experimental conditions and performing experimental work in a safe manner.

Operation Request Forms are used to request and specify detailed instructions for tasks related to operation of the GETR and its experiments. These requests are approved by operations and subject to review by the Reactor Technical Operations.

The Laboratory Instructions include policies, instruction, and administrative procedures which are applicable to more than one group at the Laboratory. Basic nuclear safety criteria, such as radiation protection requirements, are issued in this manner.

Certain procedures of prime interest are described in the remainder of this section. Procedures which govern experiments are described in Section 5.

4.4.1 Cold Startup

The procedure governing routine startup of the reactor and experiments includes the following general steps:

1. Fuel and experimental loadings in the reactor and auxiliaries are evaluated for conformance to the excess reactivity and shutdown control limitations.
2. The pressure vessel is completely reassembled, the pool level is adjusted to the operating level, and the missile shield is locked into position over the pool if the power level is to exceed 50 kilowatts.
3. The reactor and its auxiliaries, including instrumentation, controls, and shielding are inspected to assure complete reassembly and readiness for operation.
4. The height of the reactor pool and storage canal are visually checked.
5. The operation of control rods is observed as each one is individually tested by manually tripping safety circuits to insert the rods.

6. Reactor plant auxiliaries and emergency backup equipment necessary for safe startup and operation of the reactor are placed into operation. An adequate emergency supply of demineralized water must be available for the reactor and experiments and the reserve supply of water in the raw water tank must be above the minimum emergency level.

7. All experiments are loaded according to approved schedules, and operated satisfactorily under cold conditions. Trip settings are set in accordance with approved standards.

8. The performance of the ventilation, air monitoring, and isolation systems are checked.

9. Instrumentation is carefully inspected and calibrated with the trip settings. The proper setting and response of critical alarms, emergency equipment trips, and scram trips throughout the process are checked. The flux level scram trip settings are set not in excess of 125 percent of full scale on the decade selected for operation and never in excess of 60 megawatts. At least two flux-monitoring safety channels, one fission counter, and one period meter are tested for satisfactory working condition.

10. All personnel not required to facilitate the startup are cleared from the containment building.

11. The indicators and recorders are checked to assure conditions are proper for startup and that data are recorded.

12. The startup check sheets, indicating the status of the above items, are reviewed by the Shift Supervisor prior to startup.

13. The critical position is predicted by Operational Physics.

Normal startup proceeds by withdrawing control rods in a specified sequence with waiting periods after each withdrawal step to assure instrument readings have stabilized. Finer rod control is used to approach criticality as predetermined from calculations by Operational Physics and as indicated on the period and neutron level meters. The rod withdrawal rate is such that the apparent period is normally greater than 30 seconds. The reactor will automatically scram at a period of 8 seconds.

After the reactor becomes critical, the power is increased to 50 kilowatts and then leveled off for at least 5 minutes in order to establish that all systems are operating normally. The reactor power is then increased to 1 megawatt and again the power is leveled off and systems checked. Subsequent power increases are made at a maximum rate of 5 megawatts per minute until the desired operating power level is reached. Rod adjustments are made to compensate for the negative temperature coefficient and xenon effects. When the power increases to about 10 percent of rated, the period trip becomes inactive. Protection from reactivity surges is afforded by the regular flux channels.

The operating conditions of the reactor and experimental facilities are constantly checked as the power level is gradually increased. Should control of the reactor or any experiment become uncertain, immediate steps will be taken to correct the condition or shut down the reactor.

4.4.2 Hot Startup

Hot startups are made, normally, following reactor scrams and when there is no doubt as to the safety of the personnel or the facility. These startups must be made within the scram recovery time which is fixed primarily by the rate of xenon production and available excess reactivity as described in Section 2.6.1. The cause of any unplanned scram is investigated and conditions analyzed before a restart is attempted.

Hot startups will consist principally of resetting the trips and ranges of primary flux monitors and proceeding with rod withdrawal until the rod positions are the same as before the scram. Additional rod withdrawals will be made in increments with short waiting period until criticality is achieved and a positive rising period of approximately 30 seconds is obtained. Safety circuits described in Section 4.4.1 and used for cold startups are also operative during these recovery operations.

Further rod adjustments will be made to maintain this rising period and increasing power level until a power level of approximately 50 percent of the previous normal level is obtained. At this point, the power rise will be leveled off and all critical instruments again checked for proper indication and control. Following this, the power will be increased gradually to the previous normal level.

In the event of a reactor scram, all experiments will remain on the line for reactor recovery. If recovery is not possible, the experiments may be taken off the line in accordance with the shutdown procedures.

4.4.3 Shutdown

The shutdown procedure prescribes the essential steps in preparation for fuel changes, servicing of experiments, or maintenance of equipment. These steps will vary according to the shutdown activity planned. Common to all, however, are the precautionary measures taken to assure that criticality is controlled, that exposure of personnel to radioactivity is minimized, and that equipment is placed in such condition that personnel may work on it safely.

The poison sections of all control rods are fully inserted in the reactor core. The power supply is disconnected and locked out during long periods of shutdown; otherwise, normal operating requirements apply. Should operation or removal of a control rod for maintenance or testing become necessary while the core contains fuel, only one rod at a time will be moved. Such work will not be done concurrent with fuel adjustments. The rods may be cocked during refueling operations as described in Section 4.4.5.

If refueling or maintenance work is required, the following activities are conducted during reactor shutdown:

1. All accumulated gases in the pressure vessel are vented to the stack and the vessel is depressurized.

2. The circulating pump on the primary cooling loop is stopped.

3. The primary and pool loop water may be flushed to the retention tanks and replaced with demineralized water if deemed necessary.

4. The missile shield may be removed from over the pool and the top head may be removed from the pressure vessel.

Step 3 will normally be used where fast cooling of the reactor and pool is important and where rapid reduction of pool and primary loop radioactivity levels will result in lower personnel exposures or reduce the probability of contamination during shutdown work.

4.4.4 Routine Operation

The reactor will normally be operated at power levels of 30 to 50 megawatts, although the power may be varied up to a steady-state power of 55 megawatts. Automatic shutdown of the reactor is required in the event of transient power operation in excess of 60 megawatts. This power level is not measured directly but will be controlled by proper calibration of the flux safety channels.

Close surveillance of nuclear controls, critical process variables and plant equipment are made during all reactor operating periods. Such surveillance includes:

1. constant attendance and observation by a licensed operator of nuclear controls and critical process variables that are indicated,

recorded, and controlled from the reactor control room,

2. periodic recording of significant process data by the operators,

3. periodic checks of equipment operation to optimize performance, and

4. periodic review of accumulated process operating data by management to assist in obtaining optimum performance.

Upon any indication of an unsafe condition the situation shall be analyzed and corrected, if necessary, or the reactor shall be shut down. The reactor shall not be restarted until the conditions are analyzed.

Emergency power and water supplies are available to the reactor at all times; loss of either during operation above 50 kilowatts is cause for immediate reactor shutdown.

Procedures for operation at normal levels include constant surveillance and control of experiments by reactor operating personnel. Any changes in experimental conditions are accomplished in such a manner as to minimize possible effects on the reactor operation and assure compliance with approved standards.

4.4.5 Refueling

The core is refueled by manual loading through the top of the reactor. The fuel loading for each operating cycle is calculated using perturbation theory to evaluate the reactivity effects of experiments and the change in fuel from the end of the previous cycle. The total U^{235} content in the core is used as a gross check on the predicted loading. Criticality checks are made to confirm the loading and are compared with the predicted critical control rod positions.

Prior to refueling operations, the Operational Physics group prepares a list of fuel movements which identifies each fuel element to be moved by number and specifies the sequence of fuel movements. These refueling lists are approved by the Manager, GETR Operation, and copies are provided to the fuel loading crew, and the shift supervisor. The loading crew identifies each fuel element by number and notifies the control room of the transfer to be made. Transfers are then made in accordance with instructions on the refueling list. The control room copy is the master control list; and, therefore, all movements are double checked. The Operating Standards and Procedures require that:

a. loading of the reactor is performed in accordance with a written procedure,

b. prior to refueling, control rods be withdrawn to provide 3% reserve shutdown margin,

c. at least one neutron sensing channel be operated in such a way that the control rod scram circuit will be activated by an increase in neutron flux at the equivalent of 100 Kw,

d. a fission chamber is used to provide a visual indication in the control room of the neutron count rate,

e. licensed operators be on duty at all times in the control room during refueling operations, and in the containment building to supervise activities which involve the movement of fuel or control rods in the pressure vessel or in the pool if the pressure vessel head is removed,

f. only one fuel element be moved at a time, and

g. the reactor pool be maintained within 24 inches of the overflow line to afford maximum radiation protection. (When the fuel will not be transferred out of the vessel, the pool level may be lowered to the top of the vessel if the vessel remains filled with water.)

Grapples and lights are inserted into the reactor vessel and fuel elements are removed with a grapple. Each element is lifted over the side of the pressure vessel while maintaining the maximum practical water coverage. The elements are then transported under water into the storage canal where they are placed into numbered slots in the fuel storage racks as previously specified in writing. Complete records are maintained of the fuel element inventory in the storage racks. This record includes calculated percentage depletion.

New fuel may be handled and loaded into the reactor without shielding, but when reloading a previously irradiated element, the element is transported under water from the storage rack to the reactor vessel with the same precautions as used for removal.

All fuel removed from the reactor is stored in the canal racks designed for safe storage of fuel. The elements remain in the canal for radioactive decay and possible re-use in the reactor. When the usefulness of the fuel is complete and radioactivity has decayed to specified levels, the element end adapters are removed and the elements loaded into a shielded cask in an underwater loading operation on the floor of the canal. When the cask is loaded, the lid is installed and the cask removed from the canal to a transfer dolly by use of a crane. The transfer dolly is removed from the containment vessel through the equipment air lock. Elements may also be transferred to other site facilities for analysis, examination, or preparation for transfer.

4.4.6 Maintenance

Maintenance or repair of internal reactor components or control rod drives shall be performed by or under the guidance of GETR supervisory personnel in accordance with detailed instructions specified on Operations Request Forms.

When maintenance, inspection, or replacement operations are to be performed on the control rods:

a. The reactor must be subcritical by at least twice the expected reactivity change resulting from the alteration,

b. neutron sensing instrumentation similar to that described for refueling operations must be provided, and,

c. fuel elements specified by operational physics must be removed from the core to maintain a minimum subcritical margin of 3% $\Delta k/k$ if control rod movement is required. During shutdown operations which do not involve control rod movement, the minimum subcritical margin of the core is 3% $\Delta k/k$ maintained by insertion of control rods.

Functional tests will be performed following maintenance or replacement of critical reactor components, controls, and instrumentation to assure proper and reliable operation prior to restarting the reactor.

Normally, the equipment will be transferred through the containment vessel equipment air locks. If necessary, the bolted patches will be removed for access. The integrity of the patches will be tested after reinstallation and before startup.

A continuous preventive maintenance program is conducted and includes the periodic testing, replacement, and servicing of systems and equipment important to the safe and efficient operation of the reactor and experiments.

4.4.7 Health and Safety Procedures and Equipment

Procedures at the Laboratory for protection of the employees from radiation have been established to conform with the radiation exposure and control limits required by 10 CFR 20, the recommendations of the National Committee on Radiation Protection and Measurements, the International Commission on Radiological Protection, and those which the State of California presently require, except in cases where more stringent limits have been voluntarily imposed. Changes are made in the Laboratory procedures as required to conform to changes in any of the above.

The Laboratory's radiation protection policies, procedures, equipment, and emergency plans were most recently described in the Approved License Application for Chemistry, Metallurgy and Ceramics Laboratory, (SNM-420, Docket 70-445), Sections 4.2.1 through 4.2.7 and Section 5.1. These sections, which are applicable to operation of the GETR, describe exposure limits; audits; personnel monitoring and dosimetry; internal deposition control; contamination control; training; type, use and calibration of instruments; air sampling; protective clothing; bioassay;

control of entry to and removal of material from radiation areas; the health physics and fire trucks; industrial safety; and emergency and disaster plans for fire, explosions, mechanical or operational failure, air attack, and earthquakes.

The remote area monitoring stations at the GETR, as described in Section 2.7.5, are set between 1 and 100 mr/hr to give audible and visible alarm in the event normal radiation levels are exceeded. The radiation levels at these stations are recorded on a multipoint recorder. At any time a station exceeds the pre-set level, an alarm is sounded in the control room and a warning light and audible alarm are activated at the station location.

Each entrance to an area where personnel might receive a radiation dose of 100 mrem in one hour is posted and equipped to sound alarms, when an entrance is made, at that entrance and in the control room, and actuate a light in the control room which identifies that entrance. Currently, these areas are the Boiling Water Loop cubicle, nitrogen loop cubicle, pressurized water loop cubicle, first floor equipment space, and the rod access gallery.

Mobile continuous air monitors are used and relocated as necessary to serve the needs of the work in progress. They continuously sample and record airborne particulate activity, iodine and noble gases, and activate alarms when high activity is detected. Other portable air samplers continuously collect air samples for periodic analysis of particulate activity and iodine.

4.4.8 Material Handling

Radioactive material at the facility is received, used, stored, and transferred in accordance with written procedures. The identity and location of all radioactive material except activated structural materials are recorded. Storage and handling procedures, methods, equipment and locations are approved by GETR supervisors and nuclear safety personnel. Each movement of radioactive material is approved by GETR supervision.

Radioactive materials other than waste, which is described in Chapter 6, are transferred to other on-site licensed facilities or off-site to persons authorized to receive as necessary to support operation of the facility or complete irradiation programs. Shipments are labeled, packaged, and handled in accordance with federal, state, and local regulations and carrier tariffs. All shipments are surveyed prior to transfer. Requests for authorization to transfer special nuclear material as required by 10 CFR 71 or, for significantly irradiated special nuclear material, the proposed 10 CFR 72, are submitted as amendments to special nuclear material licenses. The first authorization to transfer spent fuel elements from GETR was granted by Amendment to SNM-130, Docket 70-154.

Preparation of the material for shipment and packaging may be accomplished at the GETR or other adequate site areas.

4.4.9 Administrative Procedure

Access to GETR areas other than the office area is controlled by the shift supervisor. Access is restricted to those persons with a valid interest in the operation and service of the reactor and experiments.

All significant information related to the operation of the reactor and experiments is recorded.

Personnel assigned to the GETR are trained in the procedures and requirements which assure the safe performance of their duties. Periodic safety meetings are held to discuss general industrial and nuclear safety as well as the specific safety requirements of the facility.

4.4.10 Initial Increase of Reactor Power from 33 to 55 Megawatts

The program for initial increase of reactor power from the previous steady-state maximum value of 33 MW to the current value of 55 MW will be accomplished in power increments and at a rate that will demonstrate the accuracy of the predicted reactor and experiment performance at each power step. The procedure which will govern the initial power increase shall provide that:

1. prior to the initial power increase, all experiments will be designed for maximum reactor power with respect to their location,

2. scram trips will be set at no greater than 60 MW,

3. power will be increased in increments not to exceed 5 MW,

4. the rate of increase will not exceed one increment per day,

5. each increment will be approved by the Manager of GETR operations and the Manager of Operational Physics, who will also observe the activities necessary to obtain the additional increment, and

6. immediate shutdown will be required by abnormal and unpredicted changes in reactivity or an abnormal increase in radioactivity in the primary coolant system.

Normal operating requirements will also apply during the initial increase in reactor power from 33 to 55 megawatts.

SECTION 5

DESCRIPTION AND SAFETY ANALYSIS OF THE EXPERIMENTAL FACILITIES

The GETR was designed and is operated for the purpose of conducting experiments. These experiments for the most part involve materials irradiations rather than reactor experiments. The reactor serves only as a source of neutrons for the installed experiments. As pointed out in Section 2, the GETR has facilities to accommodate a wide variety of experiments. These facilities as shown in Figure 5.1.1 are:

1) In-core capsules
2) Pool capsules
3) Miscellaneous capsules (hydraulic shuttles, trail cable, bulk, gamma)
4) Beam port
5) Loops

These facilities are described below in detail along with other information pertaining to procedures, operating experience, limits and set points, and safety analysis.

5.1 In-Core Capsules

5.1.1 General

The reactor core has an assortment of special filler pieces designed to accept both capsule and loop experiments, as shown in Figure 5.1.2. A typical experiment loading of the core includes: (a) sixteen large capsule locations, (b) sixteen small capsule locations (isotope holes in the peripheral filler pieces), and (c) three in-core loop locations. At times this loading may be changed by removing experimental filler pieces and adding additional fuel elements. Also the in-core loop locations may be used for capsule experiments. Core capsules are cooled by primary reactor water. The maximum perturbed thermal neutron flux available in these locations is about 5 x 10^{14}nv at 50 MW power. Both instrumented and non-instrumented capsules have been successfully irradiated in the reactor core.

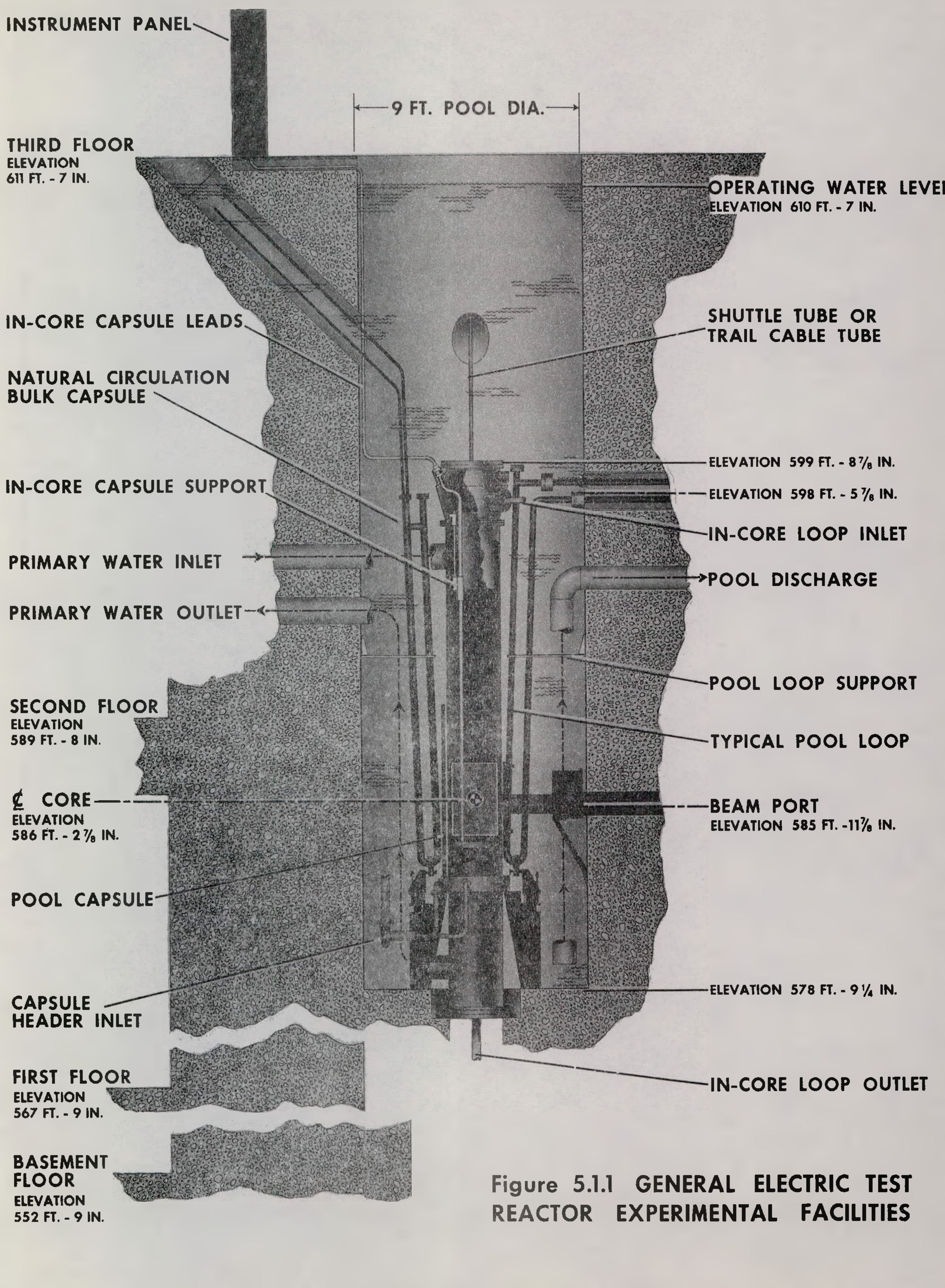

Figure 5.1.1 GENERAL ELECTRIC TEST REACTOR EXPERIMENTAL FACILITIES

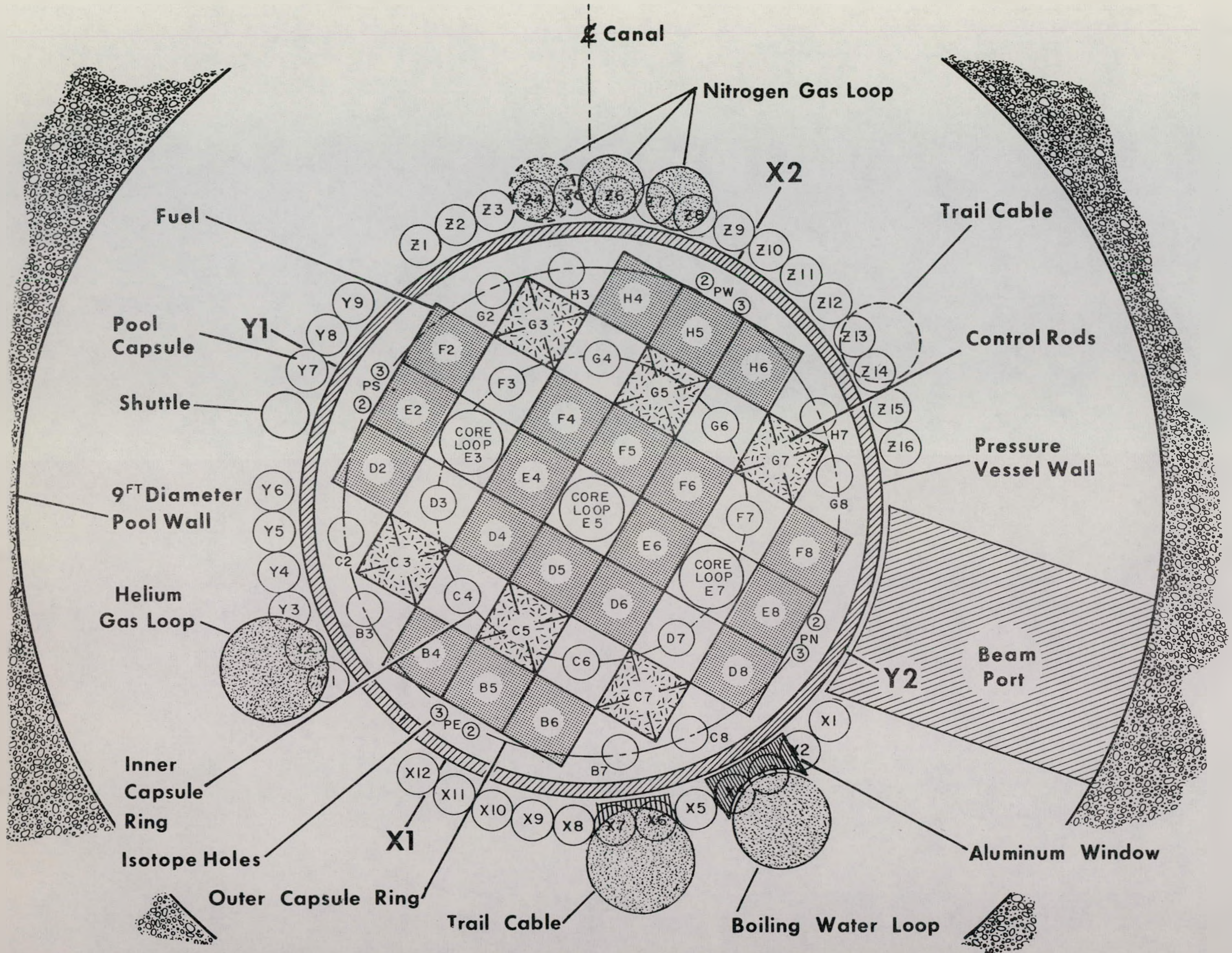

Figure 5.1.2 RADIAL LOCATIONS OF GETR EXPERIMENTAL FACILITIES

5.1.2 In-Core Capsule Facility Description

All in-core capsules are irradiated in core filler pieces. These filler pieces, as described in Section 2, may be either aluminum or beryllium, and are interchangeable with the reactor fuel elements (with the exception of the special shaped peripheral pieces). The typical filler piece has a 1-1/2 inch diameter hole bored longitudinally through the piece to accept the capsules or capsule baskets, although filler pieces with larger holes can be used. The over-all length of a filler piece is about 54 inches with the main body being approximately 39 inches in length. A retaining plug is located at the bottom of the capsule hole to prevent installed capsules from dropping out the bottom of the filler piece.

In-core capsules can be installed in the filler pieces directly or in a capsule basket which, in turn, is installed in the filler piece. The capsule basket is a tube about 45 inches long which locks into the filler piece (usually one with a 1-1/2 inch diameter capsule hole). The capsule basket has holes in the side wall near the top to receive the capsule hold down piece. The capsule basket has a fluted bottom to permit water flow through the basket and prevent capsules from dropping out. Capsule baskets are particularly useful if several capsules are to be loaded in the same core filler piece. Capsules and capsule baskets are equipped with metal weld lugs (about 3/32 inch high) spaced along their length to maintain the proper annular spaces for cooling water flow. Figure 5.1.3 shows a typical in-core non-instrumented capsule assembly.

In-core capsules can be either instrumented or non-instrumented. Non-instrumented capsules vary in length from a few inches up to about 40 inches. Their diameter is usually 1-1/4 inches although for special designs larger diameters may be used. Both types of capsules have weld metal lugs, as mentioned above, to maintain proper annular spacing for cooling water. If non-instrumented capsules are not loaded in capsule baskets, they often have special lifting knobs to facilitate handling with grappling tools. All capsules loaded into the core are numbered for proper identification and inventory purposes. Numbers are usually etched or "vibra tooled" on the side of the capsule and are large enough to be read

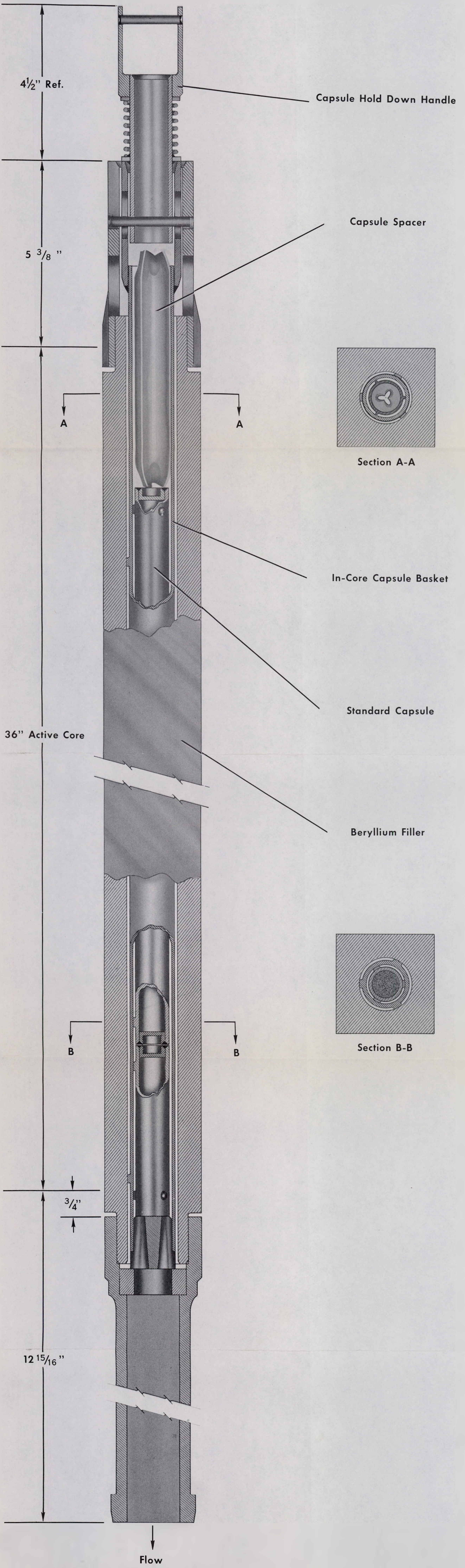

Figure 5.1.3. TYPICAL IN-CORE NON-INSTRUMENTED CAPSULE ASSEMBLY (Inner Ring)

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through ten feet of water. Instrumented capsules usually occupy one entire filler piece, although for special designs only a portion of the available space in a filler piece may be used. In-core capsule leads may be either thermocouples, electrical heater gas sampling lines, etc. Leads are enclosed in a lead tube for protection and ease of handling. Additional support tubes may be used for the lead tube, depending on the actual lead tube design. The lead tube is attached to the primary flow diffuser baffle located in the upper portion of the reactor pressure vessel (see Figure 5.1.1). From this region the leads penetrate the spool piece through special flanges and then traverse to the top of the reactor pool and on to their respective instrument consoles via the floor trenches on the reactor building third floor.

All capsules and capsule apparatus which are exposed to reactor primary water are made of inert materials particularly selected for their low corrosion properties. Aluminum, stainless steel, zirconium, and nickel are examples of capsule materials.

In-core capsules are cooled by primary reactor coolant flowing downward past the capsule in the annular spaces provided by the weld metal lugs. The reactor is automatically scrammed if primary coolant flow is lost. The inlet water temperature is about 130°F.

5.1.3 Procedures and Operation

Loading: Non-instrumented in-core capsules are loaded into their respective core location by operators using grappling tools. If a capsule is to be loaded into a capsule basket and then into a filler piece, the loading into the capsule basket is usually performed in the service canal and the loaded basket is placed in its final core location. Loading sheets are provided stating the loading location and sequence for all core capsules (as well as other core components). Check and sign-off sheets are used to assure that each capsule is located in the core properly. A final core inventory is made after all core components have been loaded and before the head is installed on the reactor pressure vessel. Instrumented capsules are loaded into the core in the same manner although the lead tube may be used to maneuver the capsule into its proper core position. The lead is fitted into its penetration in the reactor vessel spool piece and the necessary supports are attached to the lead tube in the pool. The respective connections to the control consoles are made and instrument checks completed prior to reactor startup. If all in-core experimental locations are not filled with experiments, it may be necessary to install dummy plugs to produce the required hydraulic effects in the core. Capsule experiments may also be positioned vertically in a filler piece by the use of spacers (or dummy plugs) to make optimum use of the peak neutron fluxes.

Operation: During reactor operation all in-core instrumented capsules are monitored by the capsule operator and required data are recorded. Alarms are provided on many indicators and for critical parameters automatic reactor scram circuits may be installed. In all cases, limits and set points which have a bearing on safety are determined before the capsule is approved for insertion in the reactor. In the event that alarm or abnormal conditions are reached, the operator can attempt to re-establish normal conditions by using procedures described in the capsule operating instructions. If all such efforts do not affect the abnormal conditions and if the maximum safe condition previously specified is being approached, a reactor shutdown may be required. In any event, critical parameters are monitored, safe limits are specified, and explicit instructions are at hand to be followed for all contemplated abnormal conditions.

Unloading: In-core capsules are unloaded in much the same manner as they are loaded. Instructions defining what capsules are to be unloaded, their core location, and the unloading sequence are issued prior to shutdown of the reactor. Cross-checks and sign-off sheets are used to prevent removal of the wrong capsule or capsule baskets. Capsules and capsule baskets are moved from the core to the reactor service canal for storage and subsequent shipment or disposal. While transferring a capsule to the canal, the operator is shielded by at least 10 feet of water. The leads to instrumented in-core capsules are usually cut (and crimped if required) prior to transferring the experiment to the canal. Dummy plugs, spacers, and capsule baskets are stored in the canal for future use.

5.1.4 <u>In-Core Capsule Safety Analysis</u>

<u>Review</u>: Each in-core capsule is reviewed and approved for operation prior to loading it into the reactor. This review involves the following items:

a) Power produced
b) Heat transfer
c) Hydraulics and cooling
d) Materials and type of construction
e) Pressure and stresses
f) Core reactivity effects
g) Instrumentation
h) Operating instructions

The above items are typical for all capsules. Certain specialized capsules may have other critical areas which enter into the review. This review also establishes any operational limits and instrument set points which are required to assure the safety of the experiment and the reactor. The required limits for all in-core capsules are listed in this section.

<u>Reactivity</u>: In-core capsules can have either a positive or negative effect on the reactor core reactivity although in-core capsules are usually neutron absorbers and the reactivity effect is negative. For any one experimental location (including the center core position) the maximum reactivity effect as a result of the removal of an absorbing experiment from a filler piece is limited to 0.6% or less. To date the effect of any in-core capsules has been less than this value.

<u>Construction</u>: As pointed out above, each in-core capsule is reviewed to determine the materials involved, the internal arrangement, the type of construction, the stresses involved, etc. Basic rules which are followed are:

a) All exterior materials in contact with the reactor primary coolant shall be corrosion resistant and compatible with water,

b) All internal materials shall be compatible with their expected usage and with adjacent materials,

c) Capsules shall be constructed to prevent rupture of or leakage from the capsule in the event of internal capsule member failure.

In-core capsules are normally of an all welded type construction, although on some inert isotope capsules screw tops and gaskets may be used to seal the capsule.

Cooling: In-core capsules are cooled by the downward flow of reactor primary coolant. The flow rate through the core is 10,000 gpm (8800 gpm scram point) and of this about 1600 gpm flows through the experimental in-core capsule positions. The inlet temperature is 130°F and the bulk reactor outlet temperature is 170°F. The pressure drop across the core is about 20 psi. In the event of loss of primary flow, the reactor will scram. The above values are for 50 MW reactor power.

The most important items associated with the cooling of in-core capsules are to prevent internal capsule temperatures from exceeding safe limits (and perhaps melting or rupturing), and to prevent film boiling in the core. The latter consideration is important since the core has a negative void coefficient and void formation by film boiling could cause reactor power oscillations. Internal capsule temperatures are calculated prior to loading the capsule and reviewed as part of the approval procedure mentioned above. These temperatures can be regulated by proper selection of fuel enrichments, for example, or by locating the capsule in a certain flux zone. Instrumented capsule temperatures are monitored during operation and compared with predicted values. If the capsule burnout ratio is maintained sufficiently high (greater than 1.0), film boiling and void formation cannot take place. The burnout ratio for all in-core capsules is maintained at 1.5 (or greater) at the following conditions:

a) 125% normal capsule power (or the reactor overflux scram point.)

b) 88% normal flow (reactor scram point)

c) 130°F bulk inlet water temperature

The probability of these conditions occurring simultaneously is extremely small. However, if all these conditions should occur, the burnout ratio would still be 1.5 (or greater) providing an additional margin of safety to prevent film boiling.

5.1.5 In-core Capsule Limits

The limiting conditions for all in-core capsules are:

1) All in-core capsules shall be reviewed and approved for operation by the Manager - Reactor Technical Operation or his designated alternate prior to inserting the capsules in the reactor. Special limits or restructions which may be stipulated as part of this review shall be observed.

2) The calculated burnout ratio for in-core capsules shall be at least 1.5 at the following conditions: 125% power (or the reactor over-flux scram point), low reactor flow scram, and high reactor inlet temperature rundown.

3) The maximum reactivity effect of any fully loaded in-core capsule position shall not exceed 0.6% $\Delta K/K$.

4) All materials used in capsule construction shall be compatible with the intended usage.

5.2 Pool Capsules

5.2.1 General

The GETR pool is equipped to irradiate capsule experiments. The pool has more free space than the core and accordingly a larger variety of capsule sizes and shapes can be installed in the pool. The reactor cross sectional view and the reactor elevation, Figures 5.1.1 and 5.1.2, show the pool irradiation space. There are spaces for thirty-seven capsule tubes outside the reactor pressure vessel although these tubes are removable and may be replaced by pool loop facility tubes or large diameter capsule tubes. The capsule tubes are cooled with water from a header supplied by the pool coolant pumps. The average unperturbed thermal neutron flux in the pool capsule facility is about 1 x 10^{14} nv. Both instrumented and non-instrumented capsules can be irradiated in the reactor pool.

5.2.2 Pool Capsule Facility Description

Pool capsules are normally irradiated in the capsule tubes positioned adjacent to the reactor pressure vessel. Those pool capsules not irradiated in these tubes are described in Section 5.3. The basic pool capsule tube is 1-1/2 inches in diameter and about forty-five inches long. Larger tubes (2-1/4 and 2-7/8 inches in diameter) may also be used. The larger tubes occupy two small capsule tube spaces. All capsule tubes have a special end piece which fits into the capsule tube coolant header. This header has three sections and the coolant is supplied by the pool cooling pumps. The inlet temperature to the pool capsules is about 100°F and the flow through a 1-1/2 inch capsule tube is about 13 gpm. The larger tubes have correspondingly more flow. Water flow through the capsule tubes is from bottom to top, requiring that each capsule be weighted or otherwise held down to prevent the water flow from lifting the capsule.

As in the case of in-core capsules, pool capsules can be loaded directly into a capsule tube or loaded into a standard capsule basket then loaded into the capsule tube. Capsules and capsule baskets are equipped with metal weld lugs (about 3/32 inch high) spaced along their length to maintain the proper annular spaces for cooling water flow. Pool capsules can be either instrumented or non-instrumented. The loading and handling of pool capsules is practically identical to in-core capsules as described in Section 5.1.2. The process of bringing leads from pool capsules is much simpler since no penetrations through the reactor pressure vessel are required. Pool capsule lead out tubes pass directly from the capsule, up the side of the pool to the floor trenches on the third floor of the reactor building, and then to their respective instrument consoles. Pool capsules may vary in length from a few inches up to over thirty-six inches (the length of the fuel in the core). Spacers can be used to position capsules vertically in their capsule tube or basket. If a capsule tube is to be vacant during operation or if it is to be replaced by a loop facility tube, its orifice hole to the capsule header is plugged to stop capsule header flow to that particular tube. Figure 5.2.1 shows a typical non-instrumented pool capsule assembly.

5.2.3 <u>Procedures and Operation</u>

<u>Loading</u>: Non-instrumented pool capsules are loaded in about the same manner, using similar handling tools, procedures, and check lists as for in-core capsules (see Section 5.1.3). Pool capsules are weighted to hold them in their respective capsule tubes. Instrument lead tubes are attached to the reactor pool liner after the capsule has been loaded into the proper location. Instrument checks are made after the capsule hook-up and prior to reactor startup. A pool capsule inventory is made prior to reactor startup.

<u>Operation</u>: The operation of pool capsules is identical to in-core capsules during reactor operation. Please refer to Section 5.1.3.

<u>Unloading</u>: Unloading of pool and in-core capsules is essentially the same. (See Section 5.1.3.)

5.2.4 <u>Pool Capsule Safety Analysis</u>

<u>Review</u>: The review and approval procedure for pool capsules is identical to in-core capsules as discussed in Section 5.1.4.

<u>Reactivity</u>: Experience has shown that pool capsules have a very small effect on core reactivity. The maximum reactivity for any fully loaded pool capsule tube is limited to 0.6% $\Delta K/K$. If film boiling should occur simultaneously in all the pool capsule tubes, the net reactivity effect would be about 0.04% $\Delta K/K$.

<u>Construction</u>: The construction requirements and considerations for pool capsules are identical to those for in-core capsules (refer to Section 5.1.4).

<u>Cooling</u>: Pool capsules are forced convection cooled by water from the capsule tube header. The flow rate through a typical pool capsule is about 13 gpm with a 10 psi pressure drop. The bulk inlet water temperature is about 100°F.

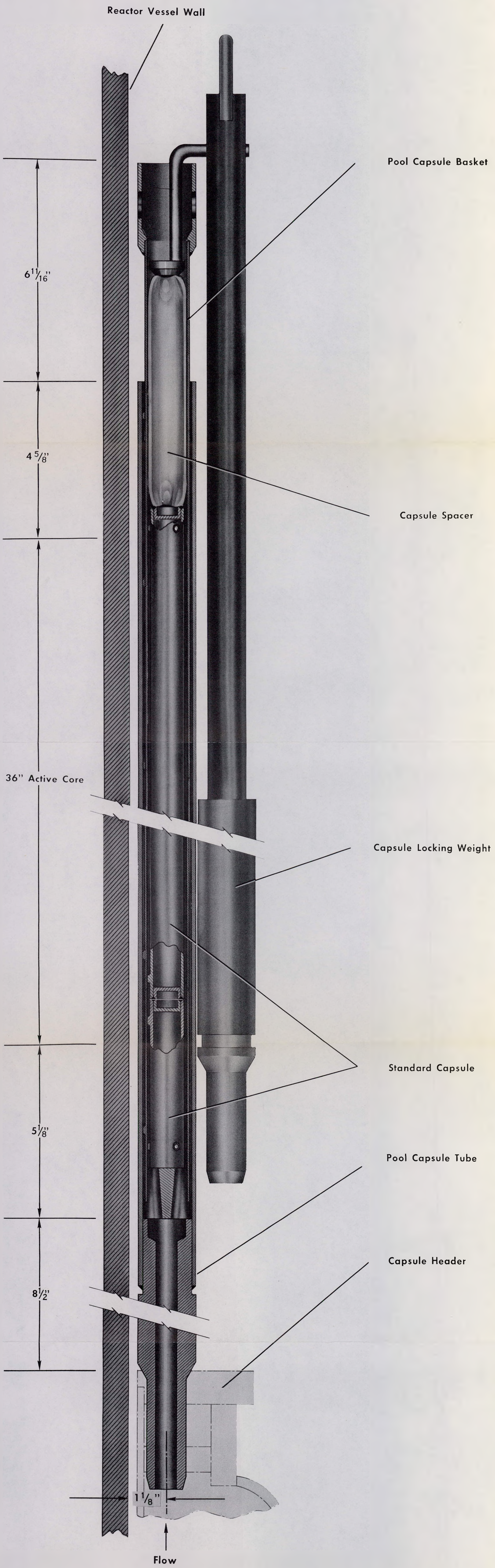

Figure 5.2.1. TYPICAL POOL CAPSULE ASSEMBLY

The most important item associated with the cooling of pool capsules is to prevent the capsule from overheating to the extent that burnout might occur or internal melting would take place. The reactivity effects of film boiling and void formation in the pool are essentially inconsequential as pointed out above. To safeguard against overheating pool capsules, the in-core capsule limit on burnout is imposed on pool capsules also. If the capsule burnout ratio is maintained sufficiently high (greater than 1.0) burnout and subsequent capsule damage cannot occur. The burnout ratio limit for all pool capsules is 1.5 (or greater) at the following conditions:

a) 125% capsule power (or the reactor over-flux scram point)
b) 50% normal capsule header flow (scram value)
c) 110°F bulk inlet water temperature

As in the case of in-core capsules, the probability of these scram conditions occurring simultaneously is extremely small. However, if they do occur, the burnout ratio would still be 1.5 (or greater) providing an additional margin of safety to prevent capsule damage.

5.2.5 Pool Capsule Limits

The limiting conditions for all pool capsules are:

1. All pool capsules shall be reviewed and approved for operation by the Manager-Reactor Technical Operation or his designated alternate prior to inserting the capsules in the reactor. Special limits or restrictions which may be stipulated as part of this review shall be observed.

2. The calculated burnout ratio for pool capsules shall be at least 1.5 at the following conditions: 125% power (or the reactor over-flux scram point), low capsule header flow (50% normal), and high pool coolant temperature (110°F).

3. The maximum reactivity effect of any fully loaded pool capsule position shall not exceed 0.6% $\Delta K/K$.

4. All materials used in capsule construction shall be compatible with the intended usage.

5.3 Miscellaneous Capsule Experiments

5.3.1 General

In addition to in-core and pool capsules, as described in Section

5.1 and 5.2, several other classes of capsule experiments can be performed in the GETR. These experiments are:

a) Hydraulic shuttle capsules
b) Trail cable capsules
c) Radial adjustable facility tubes (RAFT) in the capsule header
d) Bulk pool experiments
e) Gamma experiments in the service canal.

These experiments are normally irradiated in the pool (or canal) but under certain circumstances they could be adapted to core locations.

The Hydraulic Shuttle Facility provides a means to insert and withdraw small capsule experiments during power operation without affecting the operation of the reactor. These irradiations range from several minutes to several days in duration. Hydraulic shuttle capsules are non-instrumented experiments. The Trail Cable Facilities also provides a means to insert and remove capsules during operation. Trail cable capsules can be instrumented. Special auxiliary cooling systems may be used for these experiments. Trail Cable irradiations from minutes to days can be performed. The Radial Adjustable Facility Tube (RAFT) is similar to a standard pool capsule tube (see Section 5.2) except that its radial position (with respect to the core) can be adjusted remotely during reactor operation. Coolant for RAFT is received from the normal capsule header. This facility is useful in adjusting capsule power levels during reactor operation. RAFT capsules are usually instrumented. Similar facilities for verticle movement (RAFT) or a combination of verticle and radial movements are also available. Bulk Pool Experiments include all pool experiments which are not in the capsule tubes, the Hydraulic Shuttle Facility, the Trail Cable Facility, RAFT, or pool loop facilities. Bulk pool experiments take advantage of the pool flexibility since they often involve large bulky equipment. Examples of bulk pool experiments are a "thermal harp" experiment which has its own coolant system circulating by thermal convection, or a small gas-cooled capsule with a blower and integral gas cooling system built into the capsule. Bulk pool experiments can be either instrumented or non-instrumented.

Figure 5.1.1 shows typical miscellaneous pool capsule experiments.

5.3.2 Description

5.3.2.1 Hydraulic Shuttle Facility

The Hydraulic Shuttle Facility provides a means of rapidly transmitting small capsules to and from the reactor. The capsules to be irradiated are placed in shuttles or the capsule itself can be designed to be its own shuttle. The shuttle tube assembly is a tube extending from the service canal to the capsule tube area in the pool (see Figure 5.3.1). The tube extends upward from the capsule tube support system to the pool wall at a point about 10 feet below the surface of the pool water. The tube then passes through the concrete wall and extends along the canal wall (in a trench set in the wall) to the canal control station. Large radius bends in the tube permits passage of capsules approximately four inches long. There is presently one tube although additional tubes may be installed.

The control station includes a ball valve on the end of the shuttle tube, a basket to catch shuttles as they are ejected from the tube, a four-way valve for directing the water flow through the shuttle tube, and a water flow monitor. The ball valve and basket are located on a shelf about ten feet below the water level.

5.3.2.2 Trail Cable Facility

The Trail Cable Facility as shown in Figure 5.1.1 is a tube extending from the reactor building third floor area through a 45-degree penetration in the biological shield to the pool capsule tube area. The tube is supported by the capsule support system and is cooled by water from the capsule header. Different sized trail cable tubes have been used successfully during the past four years. The tube has several holes below the pool water level to permit free exchange of water with the pool. A lifting cable is attached to trail cable experiments and is used to lower and withdraw the experiment. Instrument leads can also be attached to these capsules. Cooling is

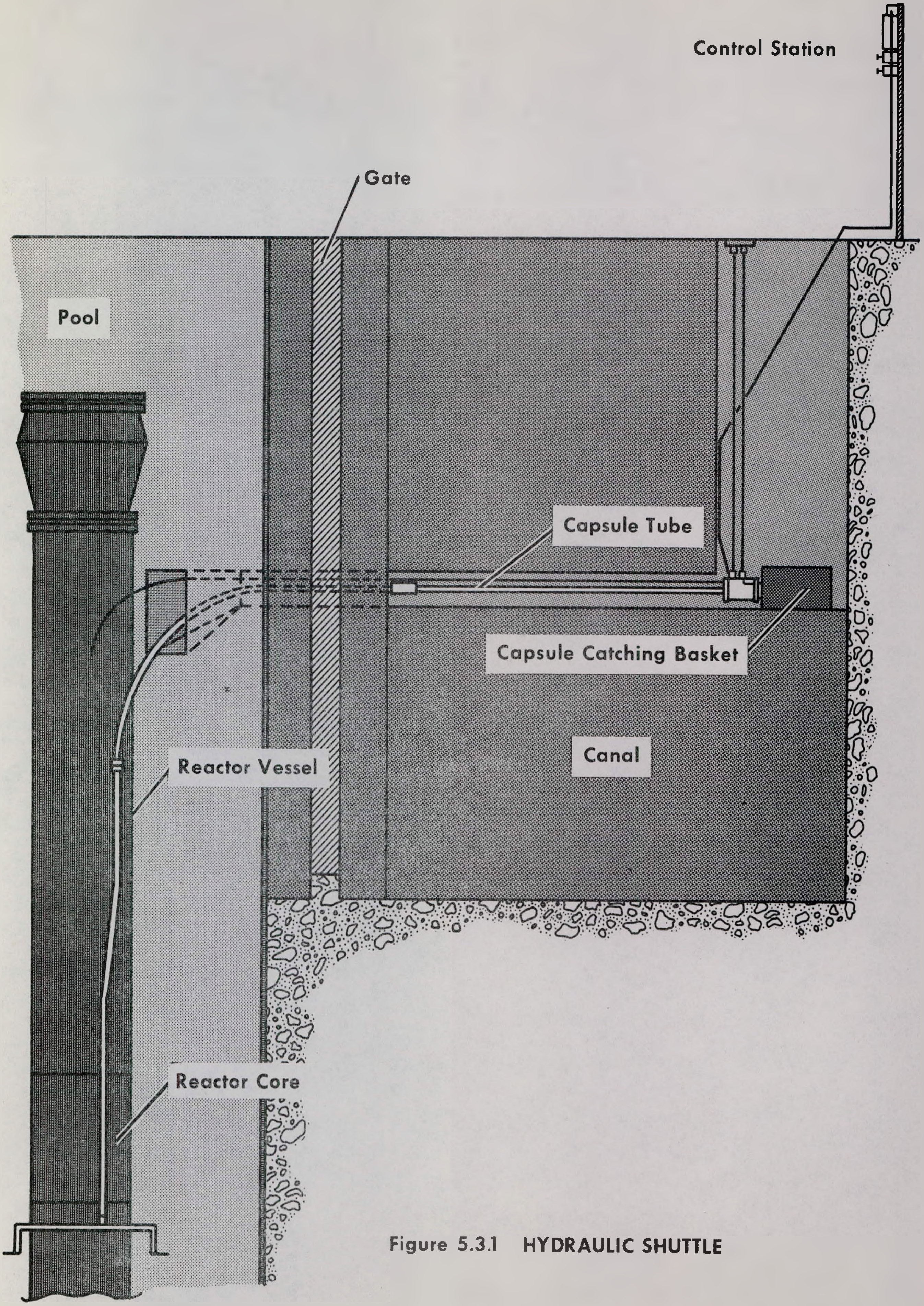

Figure 5.3.1 HYDRAULIC SHUTTLE

provided by either capsule header water flow or from a separate supply such as a flexible water hose extending down the tube from the third floor area. Radioactive capsules can be pulled from the tube into special shielded casks for subsequent shipment or removed from the tube underwater during a reactor shutdown period. More than one such facility can be used in the reactor pool at the same time.

5.3.2.3 Radial Adjustable Facility Tube (RAFT)

The RAFT is identical to the standard pool capsule tube as described previously in Section 5.2.2 except that it may be repositioned remotely during reactor operation. The direction of movement is radially outward from the reactor pressure vessel although verticle movement as well is scanned. The total travel of the capsule tube is about six inches. The actuating mechanism may be either mechanical (flexible shaft and screw linkage) or hydraulic. RAFT capsules are usually instrumented and radial adjustments are made to alter the power of the capsules. Several RAFT capsules have been operated successfully.

5.3.2.4 Bulk Irradiation Capsules

Bulk irradiation capsules include large bulky ezperiments which, due to their physical size or configuration, cannot fit into any of the standard facilities. The free space in the reactor pool is ideally suited for bulk irradiations. Capsules consisting of a fuel specimen cooled by an integral forced gas circulating system are an example of bulk experiments which have been successfully operated in the pool. At the completion of the irradiation, the entire capsule is unloaded and shipped for subsequent evaluation of the fuel specimen. A thermal harp consisting basically of a hot leg, a cold leg, a fuel specimen and associated piping is another example of a bulk irradiation capsule.

5.3.2.5 Gamma Irradiation Facility

The spent reactor fuel elements stored in the service canal are used as a source of gamma irradiation. Capsules are positioned near these fuel elements or certain arrangements of elements may be made to provide the required conditions for irradiation. Special fixtures for positioning gamma capsules are used as needed.

5.3.3 <u>Procedures and Operation</u>

<u>Hydraulic Shuttle Facility</u>. This facility is operated by first inserting a shuttle capsule through the ball valve located near the control station (see Figure 5.3.1). The ball valve is closed and the coolant flow is turned on forcing the capsule through the shuttle tube to the terminal adjacent to the reactor pressure vessel. Several shuttle capsules may be loaded and inserted in this manner at the same time. Coolant flow through the tube is maintained during the irradiation. To remove the shuttles, the flow in the tube is reversed and the capsules are returned to the basket at the loading station. This facility has performed very successfully since start-up of the reactor in 1959.

<u>Trail Cable Facility</u>. Trail cable capsules (all of which have a lifting cable attached) are inserted into the flux zone by first lowering the capsule to within a few feet of the core area. With the capsule at this location, final instrument and cooling checks are made (if these are integral parts of the experiment). The capsule is then lowered into the flux zone where it may remain or be cycled for the duration of the test after which it is pulled up to an area above the core to cool prior to removal from the facility tube. Special casks, adapted to receive a trail cable capsule, are often used to remove capsules from this facility.

<u>RAFT</u>. Capsules are loaded and unloaded from RAFT in the same manner as a standard pool capsule (see Section 5.2.3). During power operation the position of the capsule tube may be adjusted by actuating the RAFT mechanism.

<u>Bulk Irradiations</u>. The operation of bulk irradiation capsules is similar to any pool capsule. Loading is performed in the standard

manner - special apparatus may be required to locate and fix the bulk capsule in its required position. Instrumentation, if present, is attached in the same manner as instrumented pool capsules. Checks of such instrumentation are made prior to startup. During operation readings are taken from the instrumented capsules and performance is monitored. Bulk capsules are unloaded and transported to the canal for subsequent disassembly and shipment upon completion of the irradiation.

Gamma Irradiations. These capsules are normally attached to a cable or lifting wire which is used to insert and remove the experiment from the gamma facility. Capsule irradiations vary from relatively short exposure times to days.

5.3.4 Safety Analysis

5.3.4.1 Review

Capsule experiments discussed in this section are reviewed prior to their acceptance for insertion in the reactor. This review is similar to that mentioned previously for in-core and pool capsules. All capsules which are loaded into these facilities must be approved for operation. Depending on the specific design and hazards of the capsule, limits on operations such as maximum recorded temperatures or required reactor locations may be specified and, as such, are a condition of the approval. Specific instructions on the operation, handling, or unloading may also be specified as a condition of the approval. Records of these reviews and the conditions of approval are kept for the duration of the specific experimental program.

5.3.4.2 Reactivity

The reactivity effect for any capsule of the type described in this section is no greater than that for a typical pool capsule (see Section 5.2.4). Shuttle capsules, for example, have a maximum reactivity value of about 0.6% $\Delta K/K$.

5.3.4.3 Construction

The requirements for capsule construction, materials, fabrication, strength, etc., depend to a large extent on the specific purposes of the experiment. For example, if a small piece of aluminum-cobalt wire is to be irradiated in the hydraulic shuttle facility for ten minutes, the capsule construction would be much different than a fueled capsule in the RAFT facility. The aluminum-cobalt wire could safely be capsulated in quartz and in an aluminum outer container whereas the fueled experiment might require double containment in stainless steel. During the pre-irradiation review of each capsule the purpose, operating conditions, material requirements, lifetime, etc., are all examined to determine the hazards of each capsule program. The safety of each capsule and its construction, use of materials, etc., is determined based on the purpose and experiment requirements. Limits such as maximum operational temperatures, required cooling flow, and maximum pressures are stipulated if needed to assure safety of the program.

5.3.4.4 Capsule Cooling

Hydraulic Shuttle Facility. Shuttle capsules are cooled by water flow through the shuttle tube. Normally the flow is 10 gpm during steady state operation, however, when the flow is reversed to drive the shuttles out of the facility the flow drops to zero. The time that the water in the shuttle tube is stagnant is quite short. The temperature of the cooling water is 100°F or less. The requirement used to assure that shuttle capsules will not be overheated and damaged is a limit on the burnout ratio. A burnout ratio of 1.5 (or greater) with no flow and 100°F coolant temperature is required for all hydraulic shuttle capsules.

Trail Cable Facility. Trail cable capsules can be cooled by flow from the capsule header, by special cooling systems built into the capsule apparatus, or by natural convection of pool water in the trail cable tube. The basic requirement for these capsules is to maintain a burnout ratio of 1.5 (or greater) at 125% capsule power (or the reactor high power scram trip) and at the highest normal coolant temperature expected (usually 100°F).

RAFT. These capsules are cooled by flow from the capsule header - identical to pool capsules (see Section 5.2.4). The cooling requirements specified for pool capsules in Section 5.2.4 is also used for RAFT capsules.

Bulk Capsules. The surfaces of bulk capsules are cooled by pool coolant, although for some bulk capsules special coolant systems have been built into the apparatus. The basic requirement of a burnout ratio of 1.5 (or greater) at 125% capsule power (or the reactor high power scram value) and at the highest normal coolant temperature expected is used for all bulk capsules.

Gamma Capsules. Gamma capsules do not for all practical purposes generate heat. Such capsules are submerged in canal water while being irradiated and no further requirements are specified.

5.3.5 Miscellaneous Capsule Limits

The limiting conditions for all pool capsules are:

1) All miscellaneous pool capsules shall be reviewed and approved for operation by the Manager-Reactor Technical Operation or his designated alternate prior to inserting the capsules in the reactor. Special limits or restrictions which may be stipulated as part of this review shall be observed.

2) The calculated burnout ratio for miscellaneous capsules shall be at least 1.5 at the following conditions: 125% normal power (or the reactor over-flux trip point), low capsule flow (no flow for shuttles, low capsule header flow for trail cable and RAFT) and high coolant temperature (110°F).

3) The maximum reactivity effect of any miscellaneous capsule shall not exceed 0.6% $\Delta K/K$.

4) All materials used in capsule construction shall be compatible with the intended usage.

5.4 Beam Port

5.4.1 General

The beam port is a facility for special purpose testing, usually requiring an appreciable fast neutron flux and a fairly large irradiation space. The beam port is located in the reactor pool in connection with a penetration in the biological shield. The beam port, when in operation, provides a relatively unperturbed beam of neutrons into the beam port experimental space. The beam port is shown in Fig.5.1.1.

5.4.2 Beam Port Description

The beam port consists of three main components: the forward compartment, the shutter, and the biological shield penetration. The forward compartment is an eight inch diameter aluminum tube closed at both ends. One face of the forward compartment fits close to the reactor pressure vessel, the other face is attached to the shutter. The forward compartment has vent, fill, and drain lines attached to it. The shutter is a thick lead door which can be raised and lowered in an enclosed aluminum case. The shutter and case are located between the forward compartment and the pool wall. The shutter is raised and lowered by a hydraulic cylinder attached to the pool wall. The penetration in the biological shield contains several stepped plugs ranging in size from 12 inches to 14 inches in diameter. An experimental area is located adjacent to the beam port penetration outside the biological shield. A two foot diameter nozzle with a blind flange is located in the containment vessel for possible future extensions and experimental use. Figure 5.4.1 shows a cross-sectional view of the beam port.

Use of the beam port facility requires special equipment and shielding. The receiver part of the experiment can be located either in the biological shield penetration or in the experimental

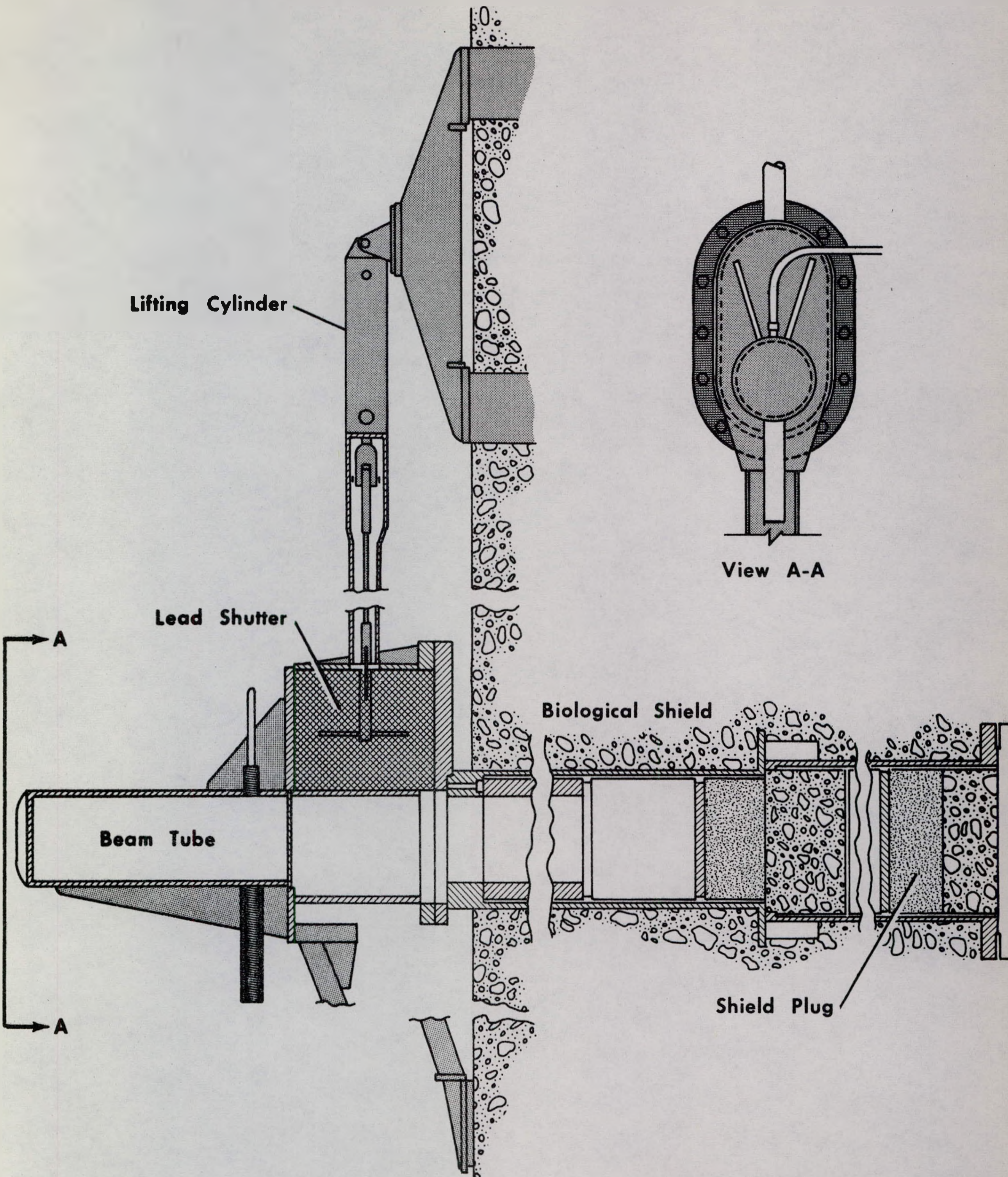

Figure 5.4.1 BEAM PORT

area outside the shield. Filters or special absorbers can be positioned ahead of the experiment receiver to obtain the desired neutron flux. Special shielding, usually lead and borated polyethylene, is used to reduce the external dose to acceptable levels in the working areas around the beam port apparatus.

5.4.3 Procedures and Operation

The beam port is operated by first raising the lead shutter and then draining the water from the forward compartment. This effectively removes all the absorber material between the reactor pressure vessel and the stepped penetration in the biological shield. To turn off the beam, the forward compartment is flooded with water and the shutter is lowered. This sequence is followed to assure that water is in the forward compartment when the shutter is down to prevent excess gamma heating in the shutter. The beam port is operated only during steady state operation of the reactor. The beam may be "on" for periods of time ranging from a few minutes to several days depending on the experimental requirements.

Prior to operating the beam port, the experiment often requires certain preparations. These preparations are, of course, entirely dependent on the specific experiment and equipment being used. For example, instrumentation should be on and checked out, cooling equipment, heaters, and power supplies (if present) are normally all functioning prior to turning on the beam. The dose to operators is periodically checked, as it is for all experimental work locations throughout the reactor building. The beam port experimental area can be made a personnel exclusion area during beam port operation, if required.

5.4.4 Safety Analysis

Review. Beam port experiments, like all reactor experiments, are reviewed and must be approved for operation prior to accepting the experiment. Typical items checked during this review are:

The purpose of the test and full program.
The shielding required and expected dose rates.
The operation of the experiment and frequency.

Sampling requirements.
Utilities required.
Experiment power and cooling requirements.
Materials, intended uses, and compatability.
Reactivity or other reactor effects.
Program or equipment changes and alterations.

Reactivity: Flooding the forward compartment has a positive reactivity effect of about 0.04% $\Delta K/K$. There is no reactivity effect involved in raising or lowering the shutter.

Cooling: Typical beam port experiments do not generate power to the extent that cooling becomes a matter of safety. If fissionable target materials are used the power is only a few watts and does not require a special cooling system. At times, low temperature environments are a part of the test program, necessitating special cooling apparatus. However, these environmental conditions are usually not a safety requirement. The beam port apparatus located in the pool area is cooled by natural circulation of pool coolant.

5.5 Loop Facilities

The GETR, being expressly designed and operated as a testing reactor, has the potential to operate several loop experiments concurrently with the other facilities mentioned above. Loop experiments are divided into two general classes; in-core and pool loops. Both types have been successfully operated in the reactor. Basically, a loop consists of a facility tube (in or adjacent to the reactor core) and the out-of-pile supporting equipment. The supporting equipment for GETR loops may be located on any of the reactor building floors. A variety of facility tube types may be installed in the GETR, such as through-tubes (usually in-core) hairpin tubes, and re-entrant tubes. The supporting equipment is in a centralized area and is contained in a cubicle or shielded room. Loops are designed for fairly specific purposes or test programs although, with modifications, a loop can easily be adapted to produce different test conditions for new experimental programs. To date, loops using gas (helium, nitrogen, and air) and boiling and non-boiling pressurized water as the primary coolant have been operated in the GETR. Other loop coolants such as fused salt or liquid metals may also be used. The following sections provide a description and safeguards analysis for each of the loop experiments in the GETR.

5.6 Pressurized Water Loop

5.6.1 Introduction

The GETR Pressurized Water Loop (PWL) is a general purpose in-core loop using water as the primary coolant. The facility tube is the through-tube type in core position E-3 (see Figure 5.1.2). The equipment cubicle and the loop control console are located on the reactor building's second floor. Figure 5.1.1 is a schematic elevation of the reactor showing a typical in-core loop such as PWL. The average thermal neutron flux available in the PWL is about 1.2×10^{14} nv which, with the proper selection of fuel element enrichment and design, can produce surface heat fluxes in the range of 1×10^{6} Btu/hr-ft^{2}.

5.6.2 PWL Systems

The PWL is composed of several individual systems - integrated to form the entire facility. Each system is used for some purpose in the experimental program although all systems may not be operated concurrently. Presented here is a listing of the PWL systems, their purpose and function, instrumentation, location, and relationship to the loop as a complete facility.

Main Loop System. The purpose of the main loop system is to circulate the primary coolant at the required conditions past or through the fuel test element. All components associated with the full main loop flow are considered part of this system. These components are: the facility tube, the particle trap, the main heat exchanger, the main pumps, the heater, the coupon station and the associated valves, piping, and instrumentation. The main loop system and its components are shown schematically in Figure 5.6.1. A description of these components is given in Section 5.6.4. The main loop components are located in several places in the GETR reactor building. The facility tube and fuel test piece are in the reactor pressure vessel, the particle trap and some piping is in the sub-pile room (below the reactor), all other main loop systems (with the exception of the instrument console) are in the PWL cubicle on the reactor building's second floor. Associated with the main loop system is most of the PWL instrumentation. The basic safety circuits to scram the reactor or run down the control rods are a part of this instrumentation. Main loop conditions of flow, pressure and temperature are monitored at important locations in the system. The radioactivity of the primary coolant is also monitored. Read-out for this instrumentation is at the main loop operating console adjacent to the shielded cubicle. A more complete description of the instrumentation is given in Section 5.6.4. Necessary cooling of the test fuel element is dependent on proper operation of the main loop. All other loop systems can be considered as supporting systems to the main loop system.

Pressurizer System. The purpose of the pressurizer system is to maintain the main loop at the desired pressure. This system consists of a pressurizer with built-in electrical heaters and water spray nozzles, a liquid level control, a vent condenser, and associated valves and piping. This system is shown schematically in Figure 5.6.1. The pressurizer system operates by boiling water in the pressurizer and condensing the steam at a controlled rate in the steam dome of the pressurizer by the spray nozzles. The spray nozzles for steam condensation receive flow from the main loop pumps. Flow to the spray nozzles is regulated to maintain loop pressure automatically by a pressure control valve which receives a signal from the main loop pressure recorder controller. Non-condensable gases can be vented from the pressurizer by venting gas to the vent condenser for subsequent release to the reactor stack exhaust system. The pressurizer system is located in the PWL cubicle.

The instrumentation associated with this system includes a pressurizer liquid level indicator, pressure indicators, radiation indicator and an automatic pressure control valve in the spray supply line. Pressure relief valves and a rupture disc are also a part of the pressurizer system.

Clean-up System. The purpose of the clean-up system is to provide a means for primary water chemistry control. About one gpm flows

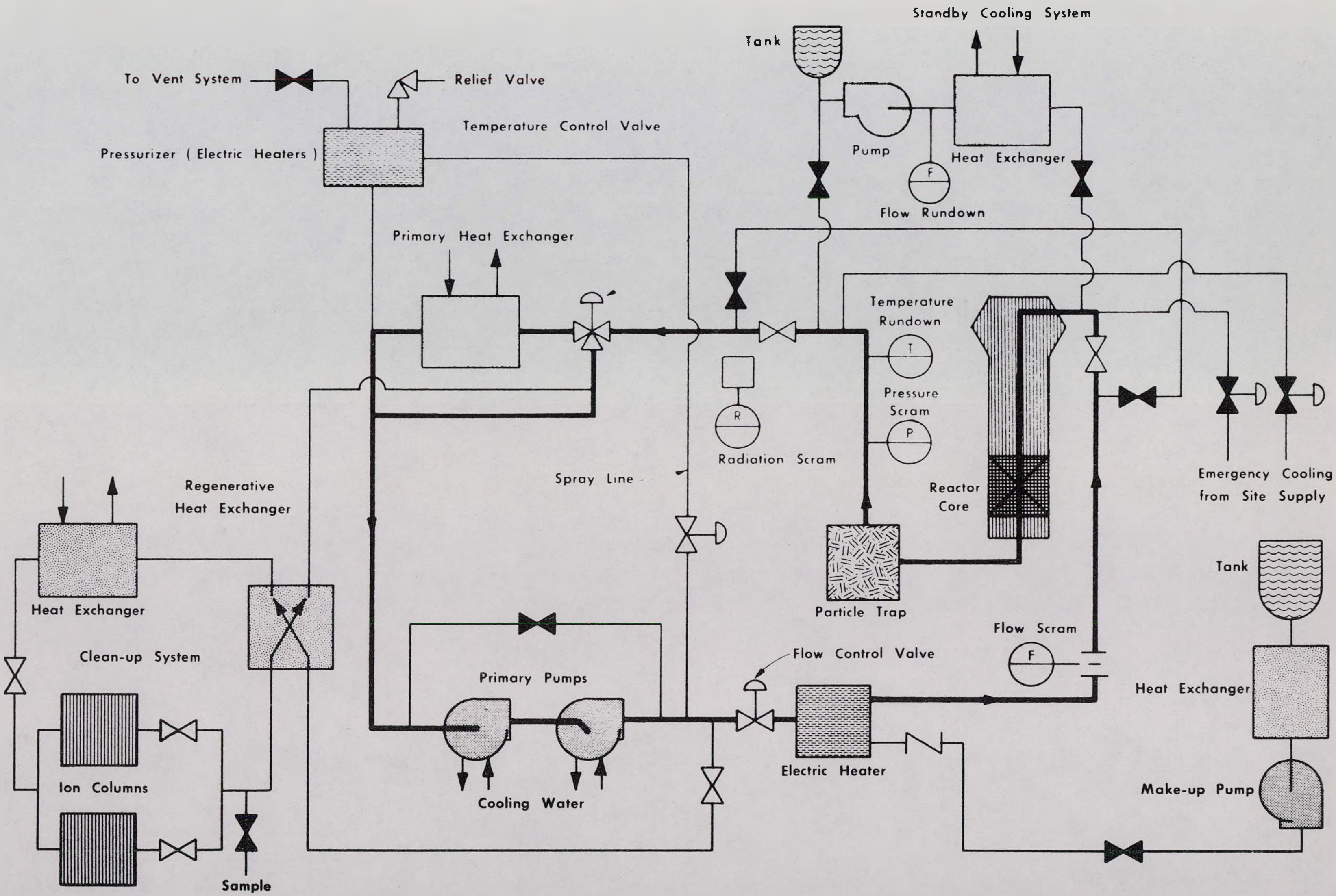

Figure 5.6.1 GETR - PRESSURIZED WATER LOOP FLOW DIAGRAM

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through the clean-up system, as shown in Figure 5.6.1. The system consists of a regenerative heat exchanger, a heat exchanger, two ion exchange columns, a sample station and associated valves and piping. The clean-up system instrumentation includes temperature measuring system, a radioactivity monitor, and a flow meter. The clean-up system is located in the PWL cubicle.

Make-up System. The purpose of the make-up system is to add primary coolant to the loop as needed during operation. The system consists of a make-up pump, a heat exchanger, a deaerator, and associated valves and piping.

Demineralized water from the GETR is supplied to the deaerator tank. Make-up coolant from the deaerator passes through the heat exchanger and is pumped into the loop by the high pressure make-up pump. Make-up coolant enters the loop via the main loop heater. The make-up system is operated manually. The instrumentation for this system consists of a level indicator and thermocouples in the deaerator tank. The PWL make-up system is located on the third floor of the reactor building and is connected to the Boiling Water Loop make-up system. Either make-up system can be used for either loop.

Standby Cooling System. The standby cooling system is used to remove gamma heat from the test section of the facility tube during operation of the reactor without fuel in the loop. This system permits the main loop to be shut down (for maintenance or cleanup as required) while the reactor is operated at power. The system consists of a heat exchanger,a pump, a surge tank and associated valves and piping as shown in Figure 5.6.1. During operation of the standby cooling system, the main loop stop valves are closed, isolating the main loop from the facility tube, and gamma heat produced in the test section is removed by operation of the standby cooling system. With the exception of the facility tube section, the standby cooling system is located in the PWL cubicle. The instrumentation included is temperature and flow measuring devices.

Emergency Cooling System. The PWL emergency cooling system consists of two lines to the site water storage tank. These lines, as shown in Figure 5.6.1, lead to both the upstream and downstream side of the test section. In the event of low loop pressure (causing a reactor scram and indicating a possible loop line rupture) automatic valves in these lines open. If the loop pressure drops to about 60 psi, the check valves in the lines open, permitting water to enter the test section and cool the fuel element. The basic components of the system are valves (automatic and check valves) and piping. No instrumentation, other than that provided for the automatic valves, is included. The components for this system are located in the PWL cubicle.

Decontamination System. The PWL has a system for decontaminating loop piping and components, should this be necessary following a loop fuel element failure. The system consists of a chemical mix tank, a pump, and associated valves and piping. If it should be desirable to decontaminate the loop, chemicals can be added via this system. The basic components are located on the third floor of the reactor building with lines leading to the main loop.

5.6.3 Typical PWL Test Conditions

The PWL is a general purpose irradiation facility capable of producing a wide range of test conditions. The loop has been used as a proof test facility for various experimental programs involving a variety of test conditions. Presented in Table 5.6.1 is a set of operating conditions for a typical PWL test program. Section 5.6.7 lists the operational limits for PWL.

TABLE 5.6.1

Typical PWL Operating Conditions

Power	208 KW
Peak heat flux	6×10^5 Btu/hr-ft^2
Main loop flow	90 gpm
Main loop pressure	1000 psi
Inlet temp (to fuel)	345°F
Outlet temp (from fuel)	372°F
Burnout ratio of fuel	3.0*

*This is the burnout ratio at the conditions of 125% loop power, low loop flow scram, and low loop pressure scram values.

For the test conditions given in Table 5.6.1, the primary coolant enters the test section at 345°F or 200°F subcooled. The temperature rise across the fuel is about 27°F (170°F subcooled at this point). The primary coolant then passes on to the main heat exchanger where a portion of the flow can be bypassed around the heat exchanger as needed to maintain the bulk water temperature of the coolant entering the main pumps at about 340°F. On the discharge side of the pumps the main heater raises the bulk coolant temperature to 345°F providing an inlet temperature of 345°F to the fuel element. The differential pressure across the pumps is about 170 psi and 30 psi across the fuel element. The primary loop radioactivity is about 1 R/hr at contact on the surface of the loop piping. As pointed out above, these conditions are typical and may change for other test programs.

5.6.4 Loop Component Description

Fuel Element. The loop is specifically operated to proof-test elements, concepts, fabrication techniques, new materials, etc.- all of which require new and different element designs. The typical fuel test element contains UO_2 as the fuel material fabricated in rod form, clad with either Zircaloy or stainless steel. Fuel rods are nominally one-half inch in diameter and usually 36 inches long. The fuel rods are fixed to the element structure at one end and loosely coupled at the other end to permit axial expansion due to temperature changes. Flow shroud tubes have been used to produce desired mass coolant velocities past the fuel rods. Elements rest on a special seat in the facility tube test section. Instrumentation leads (if present) traverse up the facility tube to the loading head and leave the loop through special seals in the loading head (see Figure 5.1.1). Fuel elements may be irradiated in the PWL for a period of a few days to a year depending on the program requirements. Both defective (or leaker) and non-defective fuel tests may be performed in this facility.

Facility Tube: The PWL facility tube is that part of the main loop system inside the reactor pressure vessel. The test section is the part of the facility tube which is in the reactor core (about 36" long). The facility tube is basically two concentric tubes - the inner tube contains the loop coolant at operating conditions and the outer tube prevents reactor coolant from contacting the inner tube. The annulus formed between these two tubes is filled with nitrogen gas (150 psi) which provides the required insulation to minimize the loss of heat from the loop coolant and also to prevent boiling of the reactor primary water. The inner tube is 316 type stainless steel; the outer tube is 304 type stainless steel. The inner tube is 2.625 inch O.D. The test section is machined to provide a wall thickness of 0.110 inches. The inside diameter of the test section is 2.375 inches. The outer tube is a 3-inch O.D. tube with a wall thickness of 0.083 inches. The outer tube has filler pieces in the core section to make a square assembly about three inches square. This facility tube occupies core position E-3. The inner tube is designed to contain the entire loop system pressure without considering any benefit from the annulus pressure. The facility tube is a pressure vessel, fully coded for operation at 1500 psig with a wall temperature of up to 600°F (Section VIII of ASME Code). The outer tube meets the Code requirements for 680 psig with a rupture strength of over 3000 psig. The facility tube annulus is equipped with an overpressure rupture disc rated at 250 psi rupture pressure. At the lower end of the facility tube, just below the reactor pressure vessel, lower head steel bellows seal the gas annulus between the two tubes and permit differential expansion between the two tubes. In the design and construction of the facility tube, every precaution was taken to produce a tube of the highest quality. For example, in addition to ASME Code requirements, the entire tube was ultrasonically inspected. The inner tube material meets ASME Code Specification SA-312.

<u>Particle Trap</u>. The particle trap is a small in-line full flow cyclone type unit located in the shielded area in the reactor subpile room. The unit has a six-inch schedule-120 type 304 body about two feet long and will collect particles as small as a few microns in diameter. The particle trap is removable and may be taken out of service during some tests.

<u>Main Heat Exchanger</u>. The main heat exchanger, located in the PWL cubicle, is of the shell and tube type. It is constructed of

carbon steel. The design pressure (tube side) is 1800 psig, and the pressure drop is about 30 psi for the loop coolant. Duty is about 2,000,000 Btu/hr.

Pressurizer: The loop pressurizer is a carbon steel vessel about 24 inches O.D. and six feet in length. The design pressure is 1800 psig at 650°F. The vessel has six 10 KW heaters. The spray line is a 3/4-inch schedule 160 pipe with ten spray nozzles. The pressurizer is located in the cubicle.

Main Circulating Pumps: There are two pumps in series in the main loop. These pumps are the 304 stainless steel centrifugal, canned rotor type designed to deliver 150 gpm at 207 feet head each. Their design pressure is 1800 psig at 600°F. These pumps are located in the cubicle.

Main Loop Heater: The heater is a carbon steel eight inch schedule 120 pipe with flange end fittings. There are five 10 kilowatt heaters. The design conditions are 1800 psig at 650°F. The heater is in the cubicle and is manually controlled.

Coupon Station: The coupon station is a 2-1/2 inch schedule 80 type 304 stainless steel pipe about forty-six inches long. Coupons, if used in the experiment, are held in place in the primary loop coolant by a stainless steel coupon holder.

Clean-up System: The clean-up loop contains two heat exchangers and two ion exchange columns. The regenerative heat exchanger has a carbon steel shell and is of coiled tube type. The shell and tube conditions are 190,000 Btu/hr duty and 1800 psig at 650°F design conditions. The second heat exchanger has a carbon steel shell and is of tube and shell type with a duty of 20,000 Btu/hr and design conditions of 1800 psig at 650°F. The ion exchange columns (two) are lead shielded and are made from four inch schedule 80 type 304 stainless steel pipes. The resin volume is about 0.25 cubic feet each. The design conditions are 1650 psig at 200°F.

Standby Cooling System: Basically, this system contains a pump, a heat exchanger and a surge tank. The pump is a stainless steel centrifugal type with a capacity of 20 gpm at 100 feet head. The design conditions are 150 psig at 215°F. The heat exchanger is a Heliflow type made of 304 stainless steel rated at 180,000 Btu/hr. The surge tank is made from a section of a ten inch, schedule ten, stainless steel pipe about two feet long.

Makeup System: The makeup system contains a pump, a heat exchanger, and a deaerator tank. The pump is a positive displacement type with a discharge pressure of 1400 psig and a design capacity of 515 gph. It is made of stainless steel with design conditions of 2500 psig at 250°F. The heat exchanger is a small Heliflow unit with stainless steel tubes. The deaerator tank is stainless steel and about 30 inches in diameter and six feet long. There are four electrical heaters in the tank. This portion of the system is not operated at pressure.

Instrumentation and Control: The PWL, being a general purpose irradiation facility, is equipped with instrumentation to assure safe operation and to provide operational data for the experimenter. The type of instrumentation and data required may change depending on the requirements of the test program. Listed below are the basic safety circuit (reactor scram and rundown instruments), general or experimental data instrumentation, and the control systems for the loop.

PWL Safety Circuit: The safety circuit contains all the loop instrumentation which can cause an automatic reactor scram or rundown in the event of loop operating problems. In all cases scram and rundown signals are preceded by an alarm signal which, in many occasions, permits the operator to take corrective action before any danger exists or the reactor is scrammed. The PWL safety circuit consists of the following items:

	Alarm	Rundown	Scram
1. Main Loop Flow	Yes	Yes	Yes
2. Main Loop Pressure (Low)	Yes	Yes	Yes
3. Main Loop Pressure (High)	Yes	Yes	No
4. Coolant Radiation	Yes	No	Yes
5. Coolant Outlet Temp.	Yes	Yes	No
6. Standby Cooling Flow	Yes	Yes	No

The scram circuits are a "2 out of 3" coincident circuit which requires three transmitters for each parameter. For example, on low pressure there are three pressure transmitters and two of the three must trip to cause a reactor scram. This type of circuit has been used successfully at GETR for over three years. The rundown instrumentation does not have the "2 out of 3" circuit. The relative location of the safety circuit sensing elements is shown on Figure 5.6.1 - the loop piping schematic drawing. The set points and safety criteria are presented in Sections 5.6.6 and 5.6.7.

General Experimental Instrumentation: The PWL is a well-instrumented facility for obtaining performance data for post-irradiation evaluation. The general classes of data produced are: flow, temperature, pressure, radioactivity, and level (tank). Some of these instruments are a part of the safety circuit mentioned above. Flow is measured in the main loop (safety circuit), the clean-up system, the standby cooling system, and in the secondary cooling water system to the various heat exchangers. In addition, differential pressure, which can be related to fluid flow, is measured across the facility tube and across the main pumps. Temperatures are measured in the following locations: facility tube outlet (safety circuit), main heat exchanger outlet, main pump discharge, facility tube inlet, pressurizer, standby cooling system, clean-up system, make-up system, and the secondary cooling water inlet to the heat exchangers. For specific test programs the fuel element may be instrumented to obtain additional data. Pressures are measured at the facility tube inlet and outlet (safety circuit), in the pressurizer and in the make-up system. In addition, differential pressures across the facility tube and across the main pumps are measured. Radioactivity of the loop coolant is measured at the facility tube outlet (safety circuit), the pressurizer, and the clean-up system. (Additional chambers are in the loop cubicle and at the loop control console for operator protection.) Liquid level indicators are on the pressurizer and deaerator tanks. These are the basic experiment instruments; additions or deletions may be made to fit the needs of the test program.

Loop Controls: In addition to the automatic actions associated with the safety circuit, the PWL has automatic controls for main heat exchanger outlet temperature (pump suction temperature) main loop pressure, and initiation of emergency cooling. Manual control of the pressurizer heaters, main loop heater, main loop flow, and make-up flow is provided. All automatic controls may be operated on manual if desired.

5.6.5 PWL Operating Procedures

The PWL operating procedures are contained in a book titled "PWL Operations and Instruction Manual". This manual is prepared by the reactor organization with assistance from the design group and the safeguards personnel as required. Upon completion of the manual, it is reviewed by the operations group, safeguards personnel, and it is subject to review by the Vallecitos Laboratory Safeguards Group. Operating procedures are to some extent dependent on the test program requirements. Prior to the initiation of a new test program, the "O and I Manual" is reviewed and changes made as needed. Listed below is the outline for the PWL "O and I Manual".

1.0 PWL Description
- 1.1 General Description and Figures
- 1.2 Component and System Description (Approximately 20 items)
- 1.3 Valve List
- 1.4 Instrumentation Summary
- Figures and Tables

2.0 Operating Procedures
- 2.1 Start-up (check lists and tables)
- 2.2 Normal Operation (levels, data)
- 2.3 Shutdown

3.0 Emergency Procedures
- 3.1 General Standards
- 3.2 Alarm Causes and Responses
- 3.3 Emergency Shutdown Procedures

4.0 Safety and Radiation Precautions

APPENDIX
- A. Drawings and Diagrams
- B. Tables
- C. Instrument Correction and Calibration Data

Procedures of this general type have been used for the PWL over the past three years. Changes and additions have been made as needed based on experience and use of the Manual. Special tests, such as changes in loop water chemistry or sampling frequency are always reviewed and approved prior to their enactment.

5.6.6 PWL Safety Analysis

5.6.6.1 Review

All test programs to be performed in the PWL irradiation facility are reviewed and approved for operation prior to their initiation. Typical items incorporated in these reviews are: (a) Performance - the element power, heat flux, desired operating conditions, etc.; (b) Limits - establishment of safety circuit set points for the program; (c) Instrumentation - review the loop instrumentation, make changes when required; (d) Fabrication - review the construction details to assure high quality equipment; (e) Inspection - physically inspect loop equipment, instrumentation, and the fuel test pieces; (f) Equipment Changes - determine all equipment changes which may be necessary to perform the test program; (g) establish a PWL Operating Standard (see Section 4); (h) review the PWL Operating Instruction Manual; (i) audit performance of the test program periodically.

5.6.6.2 Reactivity

The presence of the PWL facility tube in core location E-3 has a negative reactivity effect on the reactor core. The empty tube has a negative reactivity worth of about -1.3% $\Delta K/K$, and when loaded with a heavily loaded fuel element, the value is -0.1% $\Delta K/K$.

5.6.6.3 Cooling Experiments

The pressurized water loop is cooled by the downward flow of water through the facility tube from the main loop system as described in Section 5.3. Test fuel elements generating up to 500 KW of power may be operated in the facility. The loop cooling system is designed to remove up to 500 KW of thermal power. Fuel elements are usually designed to operate with a burnout ratio of 1.5 or greater under steady state conditions although for specific tests, lower burnout ratios may be permitted. Limits on flow, pressure, and power are set accordingly to assure that the desired burnout ratio will be met. For example, a typical set of test conditions is:

Pressure	1000 psia
Element Power	288 KW
Mass Velocity	5.0×10^6 lbs/hr-ft^2
Weight Flow	38,500 lbs/hr
Inlet subcooling	200°F
Test section temperature rise	24°F
Peak heat flux	1.0×10^6 Btu/hr-ft^2

and the corresponding loop scram values will be 90% of this flow and pressure.

The above values provide a burnout ratio greater than 1.5 at the scram point including the effect of the reactor overpower scram (a maximum of 125% normal full power).

A second method of cooling the fuel element and facility tube is provided by the standby cooling system (see Section 5.6.2). This system may be used to remove decay heat from a test element after shutdown of the reactor or the system may be used to remove gamma heat from an empty facility tube during operation of the reactor. Either mode of operation frees the main loop from operation and permits maintenance or decontamination of main loop components, if necessary. The standby cooling system is instrumented to determine the facility tube outlet water temperature and flow. Low flow in this system will cause an automatic reactor rundown. This safety circuit is provided to stop reactor operation before boiling takes place in the facility tube test section. The bulk outlet temperature from the facility tube during PWL standby cooling at 50 MW reactor power is well below the saturation temperature.

The Pressurized Water Loop also has an emergency cooling system to provide a long term supply of water in the event of a loop rupture and subsequent loss of loop coolant. Two supply lines, connected respectively to the main loop inlet and outlet from the fuel element, admit water to the loop. These lines are equipped with automatic shut-off valves and check valves. The automatic valves open on a loop low pressure signal and the check valves open when the loop pressure drops to about 60 psig. This system provides a supply of site water to assure that the fuel will always be covered with water should the loop piping rupture cause loss of loop coolant.

5.6.6.4 Reactor Startup Accident Effects on PWL

The effect of the reactor startup accident (see Section 3) on typical PWL test fuel elements has been investigated to determine the peak transient temperatures. Starting with an initial cladding temperature of 550°F, at the time of reactor startup and assuming no heat transfer from the fuel, the maximum cladding temperature was calculated to be approximately 1050°F. This temperature is well below the melting point of any cladding materials which will be used. All assumptions used in this calculation were conservative. Since the fuel material within the clad would be about the same initial temperature at startup, there is no concern of internal fuel melting as a result of this accident.

5.6.6.5 Mechanical Accidents

In the design and operation of the PWL every precaution has been taken to prevent unplanned or accidental occurrences of any type. Each loop experimental test program is evaluated prior to initiation to determine the type, magnitude, and consequences of credible accidents. Of particular significance is the evaluation of the steady state thermal burnout ratio and the instrument set points for alarm and automatic reactor scram. In the evaluation of mechanical accidents in the PWL, it may be possible that the fuel element integrity will not be maintained for certain test programs. For example, a test program involving fuel elements with a large thermal time constant and a high peak heat flux may experience thermal burnout during a loss of flow accident or a pipe rupture accident. Assurances that fuel element integrity will always be maintained cannot be given. The loop has been designed and operated as a fuel element development facility and as such it is capable of both detecting loss of fuel integrity and safely operating with ruptured or defective elements if this is desired. Throughout the design and operation of the facility, precautions have been taken to minimize the probability of fuel rupture but complete assurance cannot be given. Examples of these precautions and operational philosophy are given below. Since the simultaneous loss of loop and fuel integrity is a credible accident, detailed calculations on the specific thermal-hydraulic transient behavior of all test elements would not add substantially in assessing the hazards of loop operation. Provided below is a description of the mechanical accidents for a typical PWL fuel program. In the examples cited, fuel element integrity is maintained during the transients, which will be the case for many fuel test programs. Section 5.6.6.6 presents a description of the radiation exposures encountered subsequent to the simultaneous loss of fuel integrity and loop rupture accident. Although fuel element fission power will change these dosages slightly, these values are considered to be the maximum for all PWL fuel programs.

Loss of Electrical Power: Loss of the site power supply automatically scrams the reactor. The electrical power for the loop console (controls and instruments) and one main loop pump is supplied by the reactor 150 KW diesel-generator which is operated concurrently with the reactor. Operation of one loop pump is adequate to remove decay heat from the element during the shutdown transient.

Accidents involving simultaneous loss of both site power and emergency power are extremely unlikely. To date, this simultaneous failure has not occurred. If it should occur, the reactor would automatically scram and the PWL coolant flow would coast down. This accident is similar to the loop loss of flow accident described below; however, in this case (electrical failure) the reactor is scrammed while the loop is at full flow and the resulting element temperature transient is not severe.

PWL Loss of Primary Flow: The complete loss of primary flow in PWL is unlikely since there are two main pumps with separate power supplies. The pumps are operated outside of cavitation limits and the loop is instrumented (both alarms and automatic rundown) to detect cavitation-causing conditions. The main loop valves are either mechanically stopped to prevent complete closure or completely interlocked to prevent their use during pressurized loop operation. The loop is, of course, instrumented to detect a loss of flow transient and cause an automatic reactor scram. Section 5.6.6.2 and 5.6.7 describe the safety circuit set points. If the PWL does experience a complete loss of primary coolant flow accident, assurance that the integrity of all test elements will be maintained cannot be given. For high performance test elements such an accident, even with an automatic reactor scram, could cause burnout and possible rupture of the clad. This would cause release of some fission products to the main loop system; however, because loop integrity is maintained there would be no harm to the operators. The loop equipment is in the shielded cubicle which reduces the dose in the immediate area to less than 600 mr/hr. Operators would evacuate the building in five minutes or less. The resulting personnel exposure would be 50 mr or less even if the operator remained adjacent to the PWL cubicle for five minutes.

Loss of Secondary Flow: If secondary flow is lost to the main heat exchanger, the outlet temperature alarm would sound promptly indicating that the operator should take corrective action. If the operator fails to take any action or if the alarm fails to operate (several other temperature alarms would also sound) the reactor would be automatically rundown due to high facility tube outlet temperature. Loss of secondary cooling flow would not cause loss of fuel element integrity.

Loss of Pressure Control: Failure of any part of the pressurizer control system could cause system pressure to either increase or decrease. If it increases, the rate of pressure rise in the loop is limited by the power of the pressurizer heaters. If all heaters are on full (60 KW total), the pressure would increase at about 30 psi per minute - if no corrective action were taken. Normally these heaters are operated at about 30% of the available capacity. The operator would receive a high pressure alarm and if the pressure were allowed to increase further the safety circuit would cause an automatic reactor rundown at 100 psi above normal operating pressures. The loop relief valves are calibrated to operate at not greater than 1500 psig assuring that the loop cannot be operated at pressures above its code limit (1500 psig). Pressurizer failure causing low loop pressure would cause an automatic alarm, reactor rundown, and scram in that order.

Main Loop Rupture: The violent failure of the PWL is considered to be a very unlikely occurrence because failures in this type of equipment are normally not violent and do not result in a rapid blowdown of the loop. Nevertheless, in the accident considered here, it is assumed that the main loop piping severs in such a manner as to cause the entire loop contents to be expelled from the loop system in about one minute. Such an unlikely accident could lead to rupture of all the fuel rods in the loop experiment with subsequent transport of fission products from the loop via the severed pipe. The results presented here are for a fuel element which has been operated at 500 kilowatts for a sufficient time to build up equilibrium concentrations of fission products (about 500 days).

It is important to construct the sequence of events during a loop pipe and fuel rupture accident. The sequence of events and consequences are usually independent of the causes. In the event of a loop pipe rupture, the first indications would be noise from the cubicle and low loop pressure annunciators alarming, followed by an atuomatic reactor scram. If fuel clad integrity is lost as a result of this accident, fission products would be discharged through the severed pipe into the cubicle. The cubicle ventilation system would carry some of these fission products to the stack gas monitor which, in turn, would cause the reactor building isolation valves to close, isolating the building. These events would occur during the first thirty seconds following the pipe rupture. Upon isolation of the reactor enclosure, personnel would evacuate the building immediately. A conservative estimate of the time required for all personnel to clear the enclosure is five minutes.

Fuel element burnout is the most probable mechanism by which integrity of the cladding material could be lost, which in turn would lead to the release of fission products. Burnout would not occur simultaneously with pipe rupture since this accident would produce a short period of high mass flow (blow down) through the fuel element. The reactor would be automatically scrammed during this blow down period. Analysis has shown that burnout conditions are reached at the fuel element about the same time that the mass flow and pressure drop to zero - or after a substantial fraction of the loop water inventory has been expelled from the loop. The actual time of burnout, if it should occur, is dependent on several factors, primarily the location of the pipe rupture. In any event, the release of fission products to the cubicle would be delayed and could only occur after most of the loop coolant had been discharged. Since this delay is dependent on many factors, some of which are test variables, it has not been considered in these analyses.

In the determination of the amount of fission products which are released from the cubicle to personnel areas during this accident, the following steps have been considered:

1. Fuel operating conditions in the PWL may be sufficient to melt a portion of the fuel during steady state operation. Fission products can be released from either molten or non-molten fuel although the latter is insignificant in comparison to molten fuel releases. For the purpose of this analysis, fission products are released by evaporation from molten fuel and can escape from the element if the cladding is ruptured.

2. Typical fuel rods, operating at heat fluxes in excess of 1.0×10^6 Btu/hr-ft^2 in the PWL, would have less than 10% of their fuel molten during steady state operation. This value was determined by considering rod size, conductivities, surface temperatures, and axial power shapes.

3. Data on fission product release from molten fuel depends on the particular nuclide involved, and also on:
   a) the chemical form of the fuel
   b) the presence of steam, air or other gases during the accident
   c) the length of time the fuel is molten
   d) the extent of the fission product burden in the fuel
   e) the temperature of the melt

The information on fission product release is incomplete; the data points are scattered, and the results from various laboratories are not completely concordant. It is, however, believed that the release of fission products from molten fuel will not significantly exceed the following:

Table 5.6.6.1

Noble gases	100%
Volatiles (I, Br, Cs, Te, Se, Ru)	50%
Other	1%

4. Data on condensation of fission products on colder surfaces in the neighborhood of molten fuel have been obtained in one laboratory (MSA) and observed qualitatively in others. In the case of the PWL rupture, the loop piping and cubicle present a large cool surface on which products will condense. Considering the data available and the presence of the large cool surface areas available, it is believed that the following non-plate-out factors are conservative and are justified for determining the final release of airborne fission products:

Table 5.6.6.2

Fission Product Non-Plate-Out Factors

Noble gases	100%
Iodine and Bromine	50%
High Temperature Volatiles	30%
Other	30%

5. By comparing the enthalpy of the loop coolant at typical operating conditions (1000 psi and 500°F) and atmospheric pressure and 212°F (discharge conditions), it can be seen that there is about 300 Btu/lb of coolant available to make steam. Assuming that energy required to flash the liquid to steam is 700 Btu/lb (value at 1000 psi and 500°F) then about 40% of the loop coolant would flash to steam during this accident. This is an upper limit since the energy needed to flash liquid into steam is greater at lower pressures and temperatures. Also, the energy required to raise the temperature of the cubicle air to 212°F has been neglected. It is assumed that all noble gases released from the fuel become airborne in the cubicle and that other types of fission products become airborne only from the 40% of the loop coolant which flashes into steam.

6. In the case of the halogens and the other particulate fission products in solution with the water which flashes to steam, a certain fraction will not become airborne because of the decontamination factor. It has been found in a boiling water reactor, for example, that radioiodine has a decontamination factor of 10,000 to 100,000. The conditions of this accident and a boiling water reactor (vaporization of water) are similar although in the case of a loop blow down the vaporization of water occurs more rapidly. Therefore, an arbitrary reduction was assumed and a retention factor of 100 for halogens and particulates in the liquid phase (non-airborne) is conservative.

7. The liquid inventory in the loop is about 16 ft^3 or about 740 pounds of water. The volume of steam produced in the cubicle is about 8000 ft^3 at STP. The cubicle volume is 5000 ft^3. Although the cubicle walls do provide shielding (2 feet of high density concrete), the cubicle itself is not capable of containing internal pressure. Any additional volume expelled into the cubicle would cause an immediate equivalent volume of leakage. The expulsion of 8000 ft^3 of steam into the cubicle would, therefore, cause the leakage of 8000 ft^3 of a mixture of cubicle air and steam assuming no condensation of steam. The steam mixes uniformly with the cubicle air thus approximately 8/13 or 60% of the airborne fission products escape from the cubicle with the remaining 40% staying airborne in the cubicle at atmospheric pressure. It is important to note that as the steam enters the cubicle and contacts the relatively cool surfaces of the walls and equipment, some condensation is expected which will cause fallout (or rainout) of an additional amount of airborne fission products, particularly the halogens. This effect has not been considered in these analysis. A summary of the factors affecting the final amount of fission products which would become airborne in the reactor enclosure free volume is given in Table 5.6.6.3.

TABLE 5.6.6.3

	Noble	Halogen	Volatile	Other
Fraction of fuel molten (%)	10	10	10	10
Release from molten fuel (%)	100	50	50	1
Non-plate-out factor (%)	100	50	30	30
Liquid-steam release (%)	100	40	40	40
Decontamination factor (%)	100	1	1	1
Cubicle dilution and leakage (%)	60	60	60	60
Final fraction of total fission products available	$6x10^{-2}$	$6x10^{-5}$	$3.6x10^{-5}$	$7.2x10^{-7}$

Assuming equilibrium concentrations, the inventory of fission products in the test fuel elements can be calculated by knowing the isotopes involved, half-lives, decay schemes, decay products, and fission yields. These calculations have been carried out elsewhere and the results, listed as curies per kilowatt of loop power, are given in Table 5.6.6.4. The times listed in this table are the times after reactor scram.

TABLE 5.6.6.4

(Curies per KW)

	Time			
	0	2 min.	5 min.	10 min.
Noble gases	530	300	260	230
Halogens	400	270	250	230
Volatile Solids	530	390	310	260
Other	2500	2100	1900	1700

The amounts of each group of fission products which will actually become airborne outside the cubicle at a power level of 500 KW are (refer to Table 5.6.6.3 and 5.6.6.4):

TABLE 5.6.6.5

Curies

	0	2 min.	5 min.	10 min.
Noble gases	16,000	9,000	8,000	7,000
Halogens	12	8	8	7
Other	10	7	6	5

During the time immediately following the pipe rupture, the reactor enclosure vessel is isolated, as discussed above, causing the atmosphere in the building to become stagnant. Any fission products released from the PWL cubicle (located on the second floor) would, therefore, be concentrated in this immediate area. The free volume in the GETR reactor enclosure vessel is 230,000 ft^3 or 6.5×10^9 cc. The reactor building second floor accounts for about 15% of this volume. It is assumed that during the five minute evacuation time about 10% of the released airborne activity will spread to the first floor and 10% will also spread to the thrid floor - leaving 80% in the second floor area.

The actual time that a person in transit would spend in the second floor area is about one minute, with the remaining time (total evacuation time of 5 minutes) being spent either on the third floor, first floor, or basement. Table 5.6.6.6 lists the concentrations which would exist in the four main areas within the reactor enclosure area.

TABLE 5.6.6.6

(μc/cc)

	0	2 min.	5 min.	10 min.
Noble Gases				
3rd floor	.45	.25	.22	.20
2nd floor	12.8	7.2	6.4	5.6
1st floor	1.6	.9	.8	.7
Basement	0	0	0	0
Halogens				
3rd floor	.00034	.00022	.00022	.00020
2nd floor	.0096	.0064	.0064	.0056
1st floor	.0012	.0008	.0008	.0007
Basement	0	0	0	0
Other				
3rd floor	.00003	.00002	.00002	.00001
2nd floor	.0008	.00056	.00048	.0004
1st floor	.0001	.00007	.00006	.00005
Basement	0	0	0	0

The so-called "infinite cloud" geometry has been used to calculate the whole body dosage which assumes that the receptor is at the center of a hemisphere of inifinite radius. The effective energy from the decay of each gas or airborne contaminant is based on the total body being the critical organ. It is, of course, impossible to have an infinite cloud in the GETR containment vessel due to the physical space available. The GETR containment vessel is 66 feet in diameter, meaning that a person standing in the center of the third floor would be in a cloud 33 feet in radius. The evacuation routes and all other locations throughout the building are much closer to the containment vessel wall or shielding walls, which diminish the size of the cloud that the receptor is in. The maximum probable cloud that a receptor would be in is about 10 to 20 feet in radius. This will reduce the whole body dose to 10 to 20% of the infinite cloud value.

By use of the appropriate physical and radiological constants such as decay schemes, energy, and half-life, it is possible to determine the concentration of fission products required to give a receptor one Rem per hour if in a cloud 20 feet in radius. The resulting concentrations in microcuries per milliliter are given in Table 5.6.6.7

TABLE 5.6.6.7

(μc/cc for one Rem per hour)

	0	2 min.	5 min.	10 min.
Noble	$.15 \times 10^{-3}$	$.20 \times 10^{-3}$	$.22 \times 10^{-3}$	$.24 \times 10^{-3}$
Halogen	$.10 \times 10^{-3}$	$.14 \times 10^{-3}$	$.15 \times 10^{-3}$	$.16 \times 10^{-3}$
Other	$.32 \times 10^{-3}$	$.30 \times 10^{-3}$	$.28 \times 10^{-3}$	$.28 \times 10^{-3}$

To determine the whole body exposure to personnel in the reactor building, the concentrations given in Table 5.6.6.6 and the dose rates given in Table 5.6.6.7 were used and averaged over the time intervals. The maximum exposure would be received by the operators on the second floor, since they would be exposed to the fission products with the shortest decay time. We have assumed that the operator remains in the second floor area for one minute (being exposed in a cloud of fission products which have not been delayed in getting out) and further remains in the enclosure for four more minutes. The resulting whole body exposure would be 70 Rem

(50 Rem while on the second floor area and 20 Rem while in other areas). If the operator spent the first two minutes on the second floor (plus three minutes elsewhere) which is unlikely, his whole body dose would still be less than 100 Rem total. Operators on other floors of the building would receive much smaller whole body dosages.

The effect of the activity remaining in the cubicle as an additional source of direct radiation on the second floor can be neglected since it is several orders of magnitude less than the cloud dose. The cubicle is shielded with two feet of high density concrete.

The dose to the thyroid can be calculated by using the appropriate radiological constants and the halogen concentration data given in Table 5.6.6.6. The most significant halogens which will affect the thyroid glands are I-131, I-133 and I-135. The approximate percent of the halogen mixture for these three isotopes during the first five minutes after the accident are 10%, 20% and 20% respectively. (The other 50% of the halogens are biologically unimportant and have been neglected here.) Since the standard man inhales 10^5 cc of air in five minutes, with the total halogen concentrations given in Table 5.6.6.6 (using the residence times as stated above), the total amounts of I-131, I-133 and I-135 inhaled are 25 μc, 50 μc, and 50 μc respectively. Twenty-three percent of all iodine inhaled into the body is deposited in the thyroid, resulting in 6 μc of I-131, 12 μc of I-133 and 12 μc of I-135 deposited in the thyroid. The radiological factors to convert from microcuries deposited in the thyroid to Rads have been calculated to be 6.25 Rads/μc for I-131, 1.6 Rads/μc for I-133 and 0.4 Rads/μc for I-135. The maximum infinite dose to the thyroid is 40 Rads, 20 Rads, and 5 Rads respectively or a total of 65 Rads.

The above postulated radiation exposures were conservatively calculated as explained throughout this summary and the actual personnel exposure for this hypothetical accident would be lower than those indicated above.

5.6.7 Operational Limits

The following limits and requirements apply to PWL operation.

1. Each experimental program shall be reviewed and approved by the Manager - Reactor Technical Operation or his designated alternate prior to start of the program. All subsequent changes in operating conditions, loop equipment, fuel element design, etc., shall be reviewed in this same manner.

2. The maximum fuel element power level shall not exceed 500 KW.

5.7 Boiling Water Loop

5.7.1 Introduction

The GETR Boiling Water Loop (BWL) is a general purpose pool loop using water as the primary coolant. This loop is equipped to operate with either a boiling type fuel element or a non-boiling fuel element (Pressurized water system). Slight modifications of the equipment are necessary for conversion from boiling to non-boiling type operation. The BWL facility tube is the U-tube or hairpin type, located in the reactor pool (see Figure 5.1.1). The equipment and the loop control console are located on the third floor of the reactor building. The thermal neutron flux available in the BWL is about 5×10^{13} nv which, with the proper selection of fuel enrichment and design, will produce heat fluxes in the range of 500,000 Btu/hr-ft^2.

5.7.2 BWL Systems

The BWL is composed of several individual systems - integrated to form the entire facility. Each system is used at some stage of the experimental program although all systems are not operated concurrently. Presented here is a listing of the BWL systems, their purpose and function, location, instrumentation, and their relationship to the loop as a complete facility.

Main Loop System: The purpose of the main loop system is to circulate the primary coolant at the required program conditions past or through the fuel test element. All components associated with the full main loop flow are considered part of this system. These components are: the facility tube, the coupon station, the steam separator, the main heat exchanger, the main pumps, the heater, and the associated valves, piping and instrumentation. The main loop system and its components are shown schematically in Figure 5.7.1. A complete description of these components is given in Section 5.7.4. The main loop components are located in the reactor pool (the facility tube) and the BWL shielded cubicle on the third floor of the enclosure building.

Most of the BWL instrumentation is associated with the main loop system. The basic safety circuits to scram or rundown the reactor are a part of the instrumentation. Main loop conditions of flow, pressure and temperature are monitored at important locations in the system. In addition to serving as the safety circuit this instrumentation provides experiment data for post-operation analysis. The main loop operating console is adjacent to the shielded cubicle. A more complete description of the instrumentation is given in Section 5.7.4. Necessary cooling of the test fuel element is dependent on proper operation of the main loop. All other loop systems can be considered as supporting systems to the main loop system.

Pressurizer system: The purpose of the pressurizer system is to maintain the main loop at the desired pressure. This system consists of a pressurizer with built-in electrical heaters and water spray nozzles, a liquid level control, a vent condenser, and associated

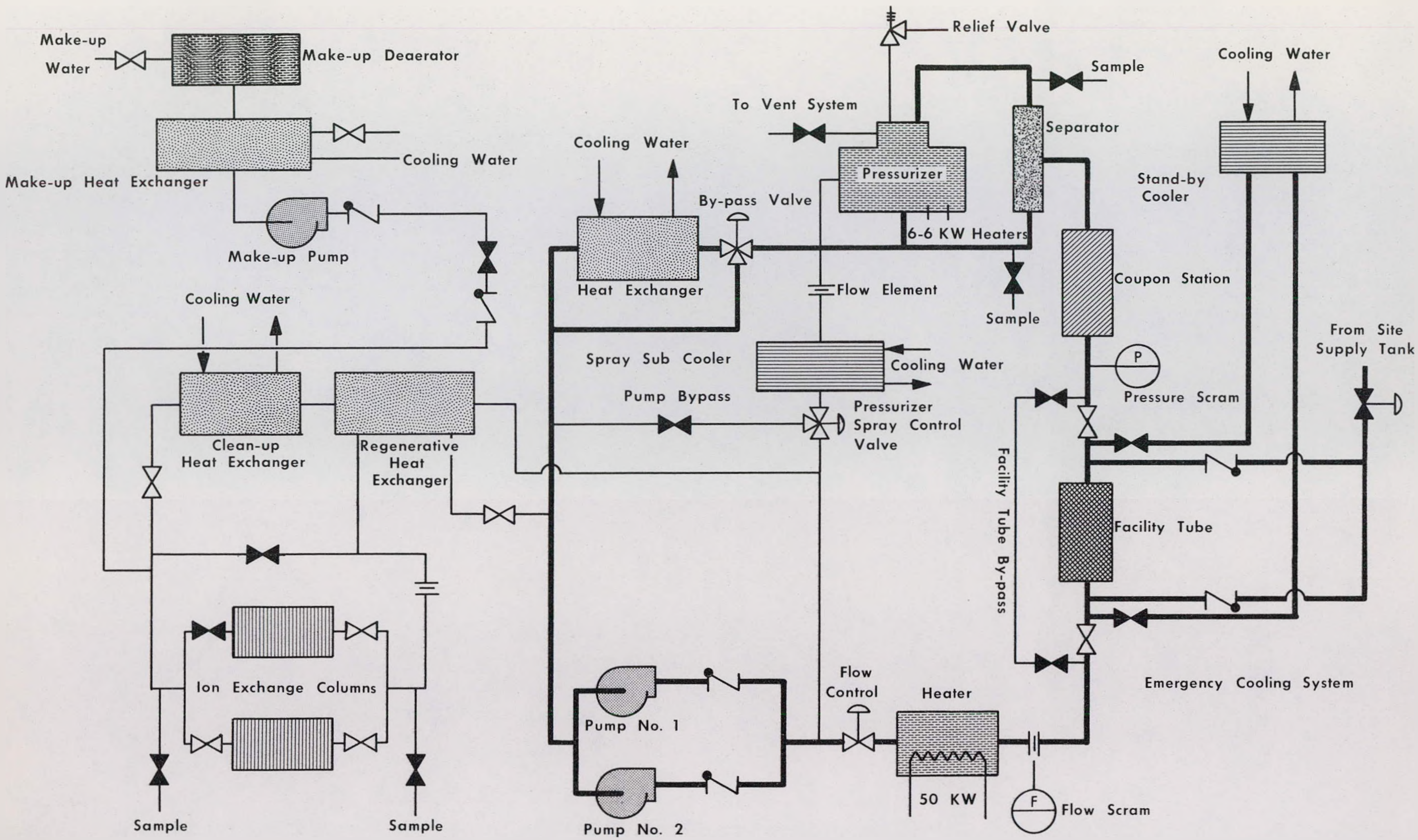

Figure 5.7.1 GETR BOILING WATER LOOP FLOW DIAGRAM

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valves and piping. This system is shown schematically in Figure 5.7.1. The pressurizer system operates by either boiling water in the pressurizer or by receiving steam from the steam separator, depending on whether or not boiling is taking place in the test section of the facility tube. In either case the steam is condensed at a controlled rate in the steam dome of the pressurizer by spray from the spray nozzles. The spray nozzles receive flow from the main pumps. Flow to the spray nozzles is regulated to maintain loop pressure automatically by a pressure control valve which receives a signal from the main loop pressure recorder controller. Noncondensable gases can be vented from the pressurizer to the vent condenser and then to storage tanks for subsequent release to the reactor stack exhaust system. The pressurizer system is located in the BWL cubicle. The instrumentation associated with this system includes a pressurizer liquid level indicator, pressure indicators, spray line flow indicator, and an automatic pressure control valve in the spray line. Pressure relief valves and a rupture disc are also parts of the pressurizer system.

Cleanup System: The purpose of the cleanup system is to provide a means for primary water chemistry control. About one gpm flows through the cleanup system as shown in Figure 5.7.1. The system consists of a regenerative heat exchanger, a heat exchanger, two ion exchanger columns, a sample line, and associated valves and piping. The cleanup system instrumentation includes temperature indicators, a conductivity indicator, flow indicators, and a radiation indicator. The cleanup system is located in the BWL cubicle.

Makeup System: The purpose of the makeup system is to add primary coolant to the loop as required during operation. This system consists of a makeup pump, a heat exchanger, a deaerator, and associated valves and piping. Demineralized water from the GETR is supplied to the deaerator tank. Makeup coolant from the deaerator passes through the heat exchanger and is pumped into the loop by the high pressure makeup pump. The makeup system is operated manually. Instrumentation for this system includes a level indicator, temperature indicator, and a pressure indicator. The BWL makeup system is located on the third floor of the reactor building and is connected to the Pressurized Water Loop makeup system. Either makeup system can be used for either loop.

Standby Cooling System: The purpose of the standby cooling system is to provide a means, in addition to the main loop, for cooling the fuel element or facility tube during certain periods of operation. The standby cooling system is rated at about 30 KW although gamma heat (with no fuel) or fuel decay heat amounts to only about 15 KW. This system is shown schematically in Figure 5.7.1. This system uses thermal head as the driving force and consists of a heat exchanger and associated valves and piping. With the exception of the facility tube, the standby cooling system is located in the BWL cubicle. There is no specific instrumentation in the standby cooling system although the main loop is instrumented to determine inlet and outlet facility tube temperatures during standby cooling.

Emergency Cooling System: The emergency cooling system for the BWL is similar to PWL emergency cooling system. It consists of a water supply line to both the inlet and outlet legs of the facility tube. Water from the site storage tank supplies coolant to these lines. The main supply line has an automatic valve and each supply line leg has a check valve (see Figure 5.7.1). In the event of a low loop pressure reactor scram, indicating a possible line rupture, the automatic valve opens. The site supply pressure is about 60 psi and the check valves would not open until the loop pressure had dropped to this value. At this time a long term supply of coolant is provided to cool the test fuel element. The equipment for this system is in the BWL cubicle.

5.7.3 BWL Operating Conditions

The BWL is a general purpose irradiation facility capable of producing a wide variety of test conditions and accommodating a wide variety of test fuel elements. The loop has been successfully operated as both a pressurized water system and a boiling water system - depending on the test program requirements. In either case, the fuel element is cooled by an upward flow of water through the test section of the facility tube. The coolant is supplied by the main loop (and the makeup system) as described in Section 5.7.2. The BWL cooling system is designed to remove up to 400 KW of thermal energy from the test section. Normally, test conditions in the BWL will be maintained to assure that the fuel element burnout ratio is 1.5 or greater at the conditions of low loop flow (scram point), low loop pressure (scram point), and the reactor over flux trip value (although for some tests a burnout ratio less than 1.5 may be permitted). It is, of course, quite unlikely that these three scram conditions would exist simultaneously; however, if they did occur, the burnout ratio at the clad hot spot would be 1.5 or greater. Limits on flow, pressure, and power are set accordingly to assure that the desired burnout ratio will be met. For example, the following are typical BWL test conditions: (Non-boiling)

TABLE 5.7.3

Element peak heat flux	300,000 Btu/hr-ft^2
Element power	140 KW
Loop pressure	1,300 psia
Loop flow	45 gpm
Test section inlet temp	510°F
Burnout ratio	4.0

For a typical BWL test program inlet coolant to the test element is held constant at 510°F. The temperature rise through the test section is 12°F with the outlet coolant being about 46° subcooled. From the test section the primary coolant passes through the main heat exchanger which lowers the temperature before the coolant enters the pumps. The main loop heater then raises the coolant temperature to 510°F to meet the test section inlet requirements. The loop differential pressure (at 45 gpm) is about 75 psi.

Standby cooling of the BWL test section is employed by closing the main loop isolation valves and opening the standby cooling system isolation valves. Coolant in this system flows by natural circulation. During this type of operation the coolant temperatures are about 120 to 130°F. The standby cooling system is not pressurized.

5.7.4 BWL Component Description

Fuel Test Element: The loop fuel element design is one of the most important variables of a test program. The BWL is a developmental facility and various types of fuel elements may be tested. Typical BWL fuel elements contain stainless steel clad uranium oxide, although other clad and fuel materials can be used. A typical fuel element contains fuel rods in a 3 by 3 array. Such fuel rods are about 36 inches long and one-half inch in diameter. Fuel enrichments are usually in the range of 2 to 5%. The fuel rods are supported in a shroud can which has provision to allow axial expansion of the fuel rods. The fuel element has a hold down rod to accurately position the fuel during operation (main coolant flow is from bottom to top through the fuel element). Instrumentation can be attached to BWL fuel elements with the leads penetrating the main loop through the facility tube loading head. Most BWL fuel elements irradiated to date have not been instrumented.

Facility Tube: The BWL facility tube is a "hairpin" type loop tube located in the reactor pool. Figure 5.1.1 shows a tube of this type in the reactor pool. The test section is the region of the facility tube adjacent to the reactor pressure vessel and contains the fuel test element. The primary coolant enters the facility tube from the main loop and flows down the outside leg away from the reactor pressure vessel. The coolant flows upward, through the fuel element, in the leg adjacent to the reactor pressure vessel and then back to the main loop. The facility tube is constructed with two concentric tubes which form a nitrogen filled annulus for thermal insulation between the two. The inner tube is designed to contain full system pressure and is coded for operating conditions of 1500 psig at 600°F. The test section is a 3 inch schedule 40 pipe with 3.068 inch nominal I.D. and 0.216 inch wall thickness. The remainder of the inner tube is 2-1/2 inch schedule 30 pipe. The material of the inner tube is type 347 stainless steel. The maximum combined stress for the BWL facility tube is below the maximum permissible stresses stated in the ASME code for this material. The outer tube is a 3-1/2 inch schedule 5 stainless steel pipe designed for 25 psig internal pressure at 200°F. Special flanges and light wire wrapped around the inner tube with a helical pitch maintain the annulus space between the two tubes. Packing glands seal the annulus between the two tubes to allow for differential expansion. Nitrogen at about one atmosphere gauge pressure in the annulus serves as insulation between loop coolant and the pool water. The facility tube test section is designed to accommodate a square fuel rod array up to 1.875 inches square and 46 inches long. A round fuel rod array can also be used. A "Marmon Conoseal" type access flange in the facility tube located above the test section permits loading and unloading test fuel elements during reactor shutdown. A 3/8 inch stainless steel hold-down rod running downward from the access flange to the test fuel element is used to hold the test fuel element in place. This hold-down rod is spring-loaded to supply a small axial compressive load to the test fuel element.

The facility tube is anchored to a support ring in the lower portion of the pool and is supported laterally from pool liner pads above the core centerline. On the support ring at the lower end of the facility tube a mechanism is installed which will allow slight adjustment of the facility tube position. This mechanism is actuated by a removable long handled tool from the reactor refueling bridge. The movement of the facility tube is in the radial direction. The facility tube can be moved a total of 3/8 inches in and 3/4 inches out from the "normal" position, which causes a change of about 37% in the thermal neutron flux. The longitudinal stress on the facility tube produced by these movements has been investigated. With highly conservative assumptions the stress on the facility tube is well below the allowable stress limits.

<u>The Shutter or Window</u>: The test section portion of the facility tube may be equipped with either a neutron shutter, a window, or no such appendages, depending on the requirements of the test being performed in the loop. The shutter forms a concentric tube surrounding the facility tube test section and is constructed of cadmium silver alloy completely enclosed by aluminum cladding. The length of the shutter is 36 inches which will shadow the fuel bearing portion of most test pieces, when in the down position. The shutter can be raised or lowered by means of a hydraulic piston. The stroke of the piston is 38 inches. The neutron shutter is designed to reduce the fuel test piece power to approximately 40 percent when in the down position. The window is a device to displace pool water in the area between

the reactor pressure vessel and the facility tube test section and thereby increase the neutron flux to the test piece. A common material used in the window is aluminum. The window, like the shutter, may be actuated by a hydraulic piston. The window and the shutter may be disassembled and assembled remotely on the facility tube test section.

The Coupon Station: The BWL coupon station is a section of 2-inch schedule 80 stainless steel pipe equipped with an autoclave type head through which test coupons may be inserted or removed. The useable test length of the coupon station is about 4 feet.

Steam Separator: The steam separator is six inches in diameter and about forty-two inches long. Coolant from the test section enters the side of the unit. Separated steam (if present in the coolant) leaves from the top and water leaves from the bottom of the unit. For boiling type operation, steam from the separator is piped to the pressurizer where it is condensed as part of the pressure regulation system. During non-boiling loop operation this steam line is blocked. Figure 5.7.1 shows this line open. The steam separator is constructed from type 316 stainless steel and is designed for 1500 psig and 600°F service.

Pressurizer: The pressurizer is a vessel twenty-four inches in diameter and about seventy inches long. The wall thickness is 1-1/4 inch (stainless steel clad). The design operating conditions are 1500 psig and 600°F. The pressurizer has a 36 KW electrical heater. The spray line in this vessel is a 1/2 inch line with four spray nozzles. The pressurizer is equipped with pressure, liquid level, and temperature measuring instrumentation. The spray line has a flow indicator. Noncondensable gases can be vented from the pressurizer steam dome to a vent condenser and then to hold up tanks for subsequent venting to the reactor stack. The main loop pressure relief valves, set at 1500 psig, are also part of the pressurizer.

Main Heat Exchanger: The main heat exchanger is a shell and coiled tube type with a capacity of 484,000 Btu/hr. The tube design conditions are 1500 psig and 600°F. The materials of construction are

304 stainless steel. A three-way valve ahead of this heat exchanger can be used to automatically regulate pump suction temperature by bypassing a portion of the primary coolant around the heat exchanger. This valve receives a control signal from a temperature recorder controller downstream from the main heat exchanger.

Main Pumps: The BWL has two centrifugal canned rotor type pumps installed in parallel to circulate primary coolant. For some test programs one pump may be adequate to satisfy cooling requirements although both pumps are usually operated concurrently. Each pump is rated at 215 feet of head at 25 gpm. The wetted parts are stainless steel. The design pressure rating is 1500 psig. Check valves on the discharge side of each pump prevent reverse flow in the event one pump stops. If BWL is operated with only one pump, then the pipe stubs for the removed pump are capped off.

Main Heater: The main heater is a 304 type stainless steel pipe 8 inches in diameter and about 4-1/2 feet long in which six "Calrod" units are installed. The heater may be either automatically or manually controlled. All primary piping throughout the main loop is type 304 stainless steel unless otherwise stated above.

Standby Cooling System: The basic component of this system is the heat exchanger, which is a shell and coiled tube type. This heat exchanger is designed and coded to operate at 1500 psig and 600°F although it is normally operated at atmospheric pressure and about 130°F. The heat exchanger and all valves and associated piping are stainless steel.

Cleanup System: The basic components in this system are the regenerative heat exchanger, the heat exchanger, the ion exchange columns, and the instrumentation. The flow through this system is about one gpm. The regenerative heat exchanger is a stainless steel unit designed for 1500 psig and 600°F operation. The other heat exchanger is also a stainless type with the same design conditions. The outlet temperature of this exchanger is about 100°F. There are two shielded ion exchange columns in this system which are three inches in diameter and 36 inches long. These units are made of stainless steel. The instrumentation includes an activity monitor, flow indicator, temperature indicators, and a sampling station.

Makeup System: The makeup system consists of a deaerator tank, a heat exchanger, and a makeup pump, plus associated valves and piping. The deaerator tank is thirty inches in diameter and sixty inches long with an electrical heater installed. The tank receives demineralized water from the reactor supply. The heat exchanger is a small "Heliflow" unit which cools the makeup water. The makeup pump is a positive displacement type with a 1200 psig discharge pressure. The capacity of this pump is 5.5 gph. All parts in this system are stainless steel.

Instrumentation: The BWL, being a general purpose irradiation facility, is equipped with instrumentation to assure safe operation and to provide operational data for the experimenter. The type of instrumentation and data required for both safety and the experimenter may change depending on the requirements of the test program. Listed below are the basic safety circuit, the experimental data instrumentation, and the control systems for the loop.

BWL Safety Circuit: The safety circuit contains all the loop instrumentation which can cause an automatic reactor scram or rundown in the event of loop operating abnormalities. In all cases scram and rundown signals are preceeded by alarm signals. On many occasions this warning permits the loop operator to take corrective action before any danger exists and the reactor is scrammed. Reactor scrams are also preceeded by rundowns. The BWL safety circuit consists of the following items:

	Alarm	Rundown	Scram
1. Main Loop Flow	Yes	Yes	Yes
2. Main Loop Pressure (Low)	Yes	Yes	Yes
3. Main Loop Pressure (High)	Yes	Yes	No

The scram circuits are "2 out of 3" coincidence circuits which requires three transmitters for each function. For example, on loop pressure there are three pressure transmitters and two of the three must trip to cause a reactor scram. This type of circuit has been used successfully at GETR for over four years. The rundown instrumentation does not have the "2 out of 3" circuit. The relative location of the safety circuit sensing elements is shown on Figure 5.7.1, the loop piping schematic drawing. The set points and safety criteria are presented in Section 5.7.6 and 5.7.7.

General Experimental Instrumentation: The BWL is a well instrumented facility for obtaining performance data for post irradiation evaluation. The general classes of data produced are: flow, temperature, pressure, radioactivity, and tank level. Some of these instruments are a part of the safety circuit mentioned above. Flow is measured in the main loop (safety circuit), the cleanup system, the spray line to the pressurizer, and in the secondary cooling water system to the various heat exchangers. In addition, differential pressure, which can be related to fluid flow, is measured across the main pumps and across the facility tube. There are many locations throughout the BWL for measuring temperatures. In the main loop, inlet and outlet temperatures for the facility tube, main heat exchanger, main pumps and heater are all measured. The temperature in the pressurizer, cleanup, makeup and standby cooling systems are also measured. The temperature of the secondary water cooling system is also measured. For specific test programs the fuel element may be instrumented to obtain additional data. Pressures are measured at the facility tube inlet and outlet, in the pressurizer, and in the makeup system. In addition, differential pressures across the main pumps and facility tube are measured. Radioactivity of the loop coolant is measured in the cleanup system. In addition, ionization chambers located in the main cubicle and at the loop control console are provided for personnel protection. Liquid level indicators are provided for the pressurizer tank and the deaerator tank.

These are the basic experiment instructions. Additions or deletions may be made to accommodate specific test needs.

Loop Controls: In addition to the automatic actions associated with the safety circuit and alarm system, the BWL has automatic controls for coolant temperatures at the heat exchanger outlet and main heater outlet. Main loop pressure is automatically controlled by the pressurizer system and main loop flow is automatically controlled. The temperature in the cleanup system is also controlled automatically. Manual control backup is provided for all automatic loop controls and may be used interchangeably with automatic control.

5.7.5 BWL Operating Procedures

A manual of BWL operating procedures is provided. This manual is prepared by the reactor organization with assistance from the design group and safeguards personnel as required. Upon completion of the manual, it is reviewed by the operations group, safeguards personnel, and it is subject to review by the Vallecitos Laboratory Safeguards Group. Operating procedures are to some extent dependent on the test program requirements. Prior to initiation of a new test program, the manual is reviewed and changes made as needed. Listed below are typical items covered by the manual.

1.0 BWL Description
- 1.1 General Description and Figures
- 1.2 Component and System Description (approximately 20 items)
- 1.3 Valve List
- 1.4 Instrumentation Summary
- Figures and Tables

2.0 Operating Procedures
- 2.1 Startup (check lists and tables)
- 2.2 Normal Operation (levels, data)
- 2.3 Shutdown

3.0 Emergency Procedures
- 3.1 General Standards
- 3.2 Alarm Causes and Responses
- 3.3 Emergency Shutdown Procedures

4.0 Safety and Radiation Precautions

APPENDIX
- A. Drawings and Diagrams
- B. Tables
- C. Instrument Correction and Calibration Data

Procedures of this general type have been used for the BWL over the past four years. Changes and additions have been made as needed based on experience and use of the manual. Special tests, such as changes in loop water chemistry or sampling frequency, are always reviewed and approved prior to their enactment.

5.7.6 BWL Safety Analysis

5.7.6.1 Review: All test programs to be performed in the BWL irradiation facility are reviewed and approved for operation prior to initiation of the program. Typical items considered by these reviews are: (a) performance - the element power, heat flux, desired operating conditions, etc.; (b) limits - establishment of safety circuit set points for the program; (c) instrumentation - review of the loop instrumentation, make changes when required; (d) fabrication - review the construction details to insure high quality equipment; (e) inspection - physically inspect loop equipment, instrumentation, and the fuel test pieces; (f) equipment changes - determine all equipment changes which may be necessary; (g) establish a BWL Operating Standard (See Section 4); (h) review the instruction manual; (i) audit performance of the test program periodically.

Often as part of the review process the experimental program (and loop changes, if required) is discussed with the Vallecitos Laboratory Safeguards Group.

5.7.6.2 Reactivity: The reactivity effect of the BWL, loaded with a typical fuel element, is less than 0.2% ΔK.

5.7.6.3 Cooling Experiments: The Boiling Water Loop fuel element is cooled by an upward flow of water through the test section of the facility tube. Depending on the specific test objectives, boiling may be permitted over a portion of the fuel test piece. The coolant is supplied by the main loop and an independent makeup system as described in Section 5.7.2. The cooling system is designed to remove up to 400 KW of thermal energy from the test section. For most tests the fuel element is designed to operate with a burnout ratio of 1.5 or greater under steady state conditions although for some tests lower burnout ratios may be permitted. Limits on flow, pressure, and power are set accordingly to insure that the desired burnout ratio will be met.

A second method of cooling the BWL fuel element is use of the standby cooling system (see Section 5.7.2). This system may be used to remove decay heat from a test element after shutdown of the reactor, or the system may be used to remove gamma heat from an unloaded facility tube during operation of the reactor. The main loop can be operated in place of the standby cooling system to perform either of these functions, although the standby cooling system frees the main loop for maintenance or decontamination, if necessary. The BWL standby cooling system is instrumented to determine facility tube inlet and outlet water temperatures. The bulk outlet water temperature during standby operation (either removing fuel element decay heat or gamma heat from an unloaded facility tube) is about 130°F which indicates that boiling does not occur in the test section. Although the BWL standby cooling system is designed for 1500 psi operation, it is operated at or near atmospheric pressure.

The BWL has an emergency cooling system to cool a fuel element in the event of an accident. This system is described in Section 5.7.2 and Section 5.7.6.5.

5.7.6.4 Reactor Startup Accident Effects on BWL

The effects of a reactor startup on the BWL facility are the same as those described for the Pressurized Water Loop (PWL) in Section 5.6.6.4.

5.7.6.5 BWL Mechanical Accidents

In the design and operation of the BWL every precaution has been taken to prevent unplanned or accidental occurrences of any type. Loop experimental test programs are evaluated prior to their initiation to determine the credible accidents and what steps must be taken to minimize their occurrence. This evaluation also includes the consequences of these accidents should they occur. Of particular significance is the evaluation of the steady state thermal burnout ratio and the instrument set points for alarm and automatic reactor scram. Certain mechanical accidents could lead to loss of fuel element integrity. For example, a test program involving fuel elements with a large thermal time constant and a high peak heat flux may experience thermal burnout of the element during a loss of flow accident or a pipe rupture accident. Assurances that fuel element integrity will always be maintained cannot be given. The loop has been designed and operated as a fuel element development facility and as such it is capable of both detecting loss of fuel integrity and safely operating with defective fuel elements. Since the simultaneous loss of loop and fuel integrity is a credible accident, detailed calculations on the specific thermal-hydraulic transient behavior of all test elements would not add substantially in assessing the hazards of loop operation.

In many respects there is a similarity of the Pressurized Water Loop (PWL) and the BWL. For example, the main loop systems, pressurizer systems, makeup-cleanup systems, emergency and standby cooling systems are practically identical for each facility. The operational limits are also similar for these two loops. It follows therefore that the mechanical and operational accidents and the resulting hazards to the operators are also similar for the two facilities. One of the basic indicators used to assess the degree of hazards is the fuel element fission product inventory, which is assumed to be proportional to the steady state power level. In the case of BWL the limit on power level is 400 KW (500 KW for PWL). With only a few minor wording variations, the Mechanical Accidents for PWL (Section 5.6.6.5) are applicable to BWL as well. Such accidents involving BWL would result in lesser hazards to operators than if the

PWL was involved because of the differences in fission product inventory, which is assumed to be proportional to the steady state power level.

5.7.7 BWL Operational Limits

The following limits and requirements apply to BWL operation:

1. Each experimental program shall be reviewed and approved by the Manager-RTO or his designated alternate prior to start of the program. All subsequent changes in operating conditions, loop equipment, fuel element design, etc., shall be reviewed in this same manner.

2. The maximum fuel element power level shall not exceed 400 KW.

5.8 The Nitrogen Loop

5.8.1 General

The Nitrogen Loop is a general purpose gas loop installed in the GETR. Although this loop could be adapted to operate with an in-core facility tube, it is more convenient to operate the loop with facility tubes located in the pool (due to the greater space available in the pool). Physically the loop is located in the reactor building basement and the second floor, with the facility tubes located in the reactor pool. Figure 5.8.1 shows the location of the major loop components in the reactor building. The Nitrogen Loop has components installed to permit up to five independent facility tubes to be operated concurrently in the reactor pool. Currently, only two pool facility tubes are installed although five tubes have been operated in the past. The Nitrogen Loop can operate with different gases as the primary coolant. For example, the loop was originally operated with helium, later modified and nitrogen was used as the primary coolant which has been subsequently changed to air. The Nitrogen Loop has been in operation, with short period of down time for modifications, since the reactor commenced power operation in 1959.

5.8.2 Nitrogen Loop Description

The primary loop components and systems are shown in Figure 5.8.2. The main loop consists of compressors, a regenerative heat exchanger,

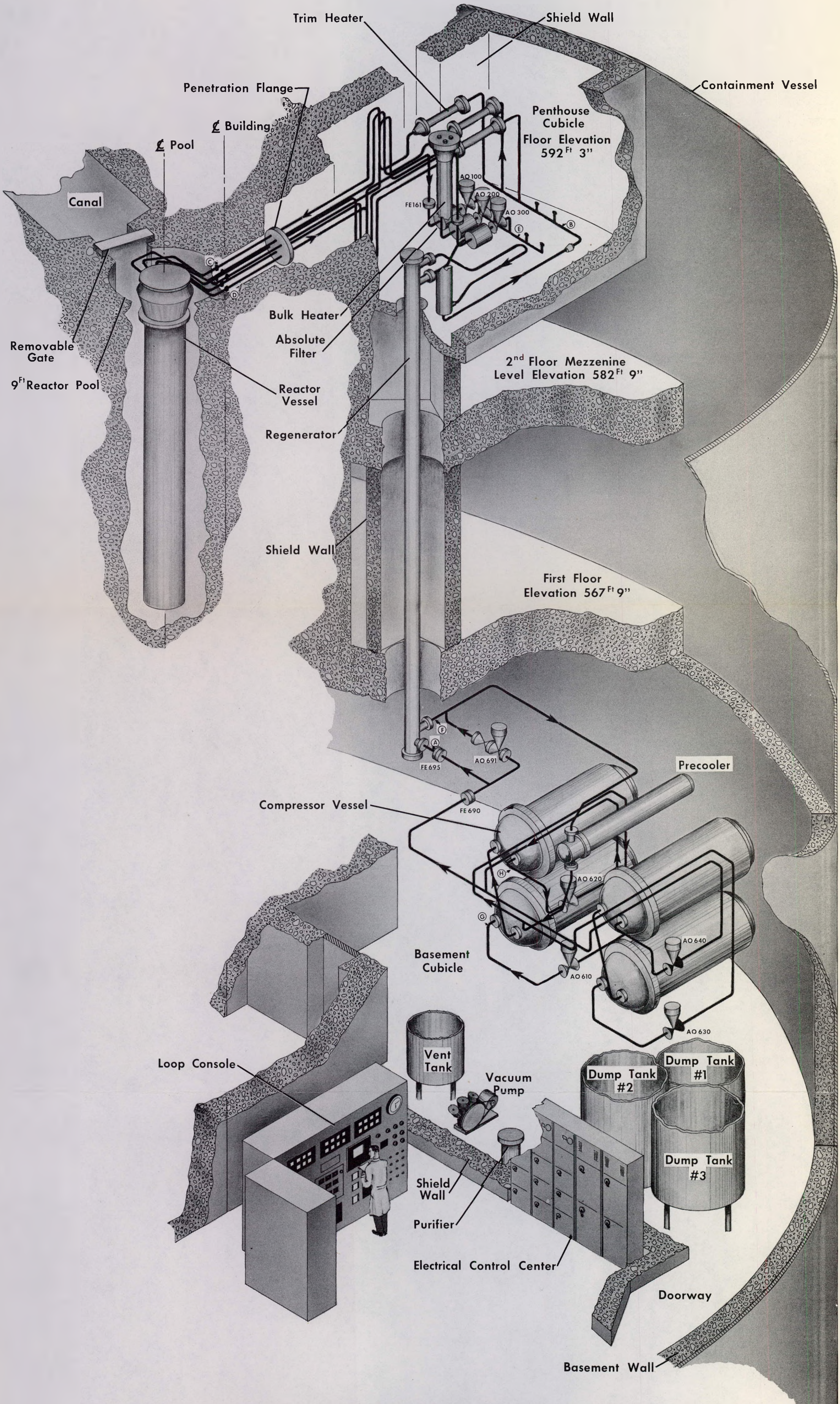

Figure 5.8.1 NITROGEN LOOP INSTALLATION
(Secondary Piping Systems Not Shown)

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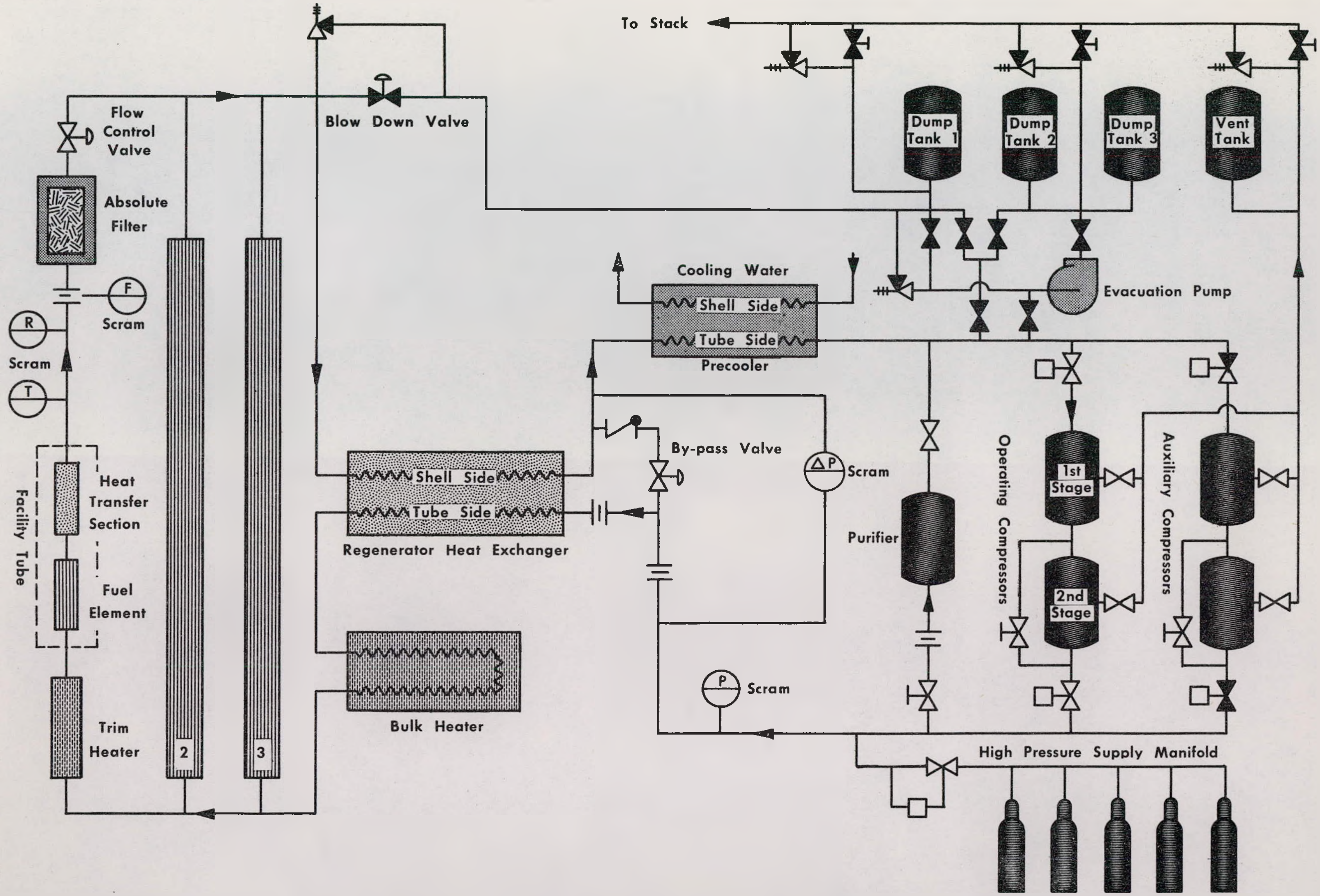

Figure 5.8.2 GETR NITROGEN LOOP FLOW DIAGRAM

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electrical heaters, the facility tubes (containing the fuel test pieces), filters and valves, a pre-cooler and associated valves and piping. The auxiliary systems include the makeup system, the purification system, the vent system, and the emergency cooling system.

Compressors: The coolant flow is provided by axial-flow rotary helical compressors, designed to deliver about 9700 pounds of nitrogen (or air) per hour. Two compressors in series provide the necessary head and flow for loop operation under normal operating conditions, with a portion of this flow being bypassed around the facility tubes (see Figure 5.8.2). Two other compressors installed in parallel with the operating units are held in standby in case of failure or breakdown of the operating units. The loop may be operated without the standby compressors in an operative condition. The compressors and motor assemblies are housed in individual containment vessels. The containment vessels operate at approximately loop pressure and are equipped with coolers to remove excess heat produced by the electric motors. A vent line leading to the vent tank is attached to each containment vessel. Primary coolant is continuously vented through this system from the compressor containment vessels to the vent tank to sweep contaminants out of these vessels to the vent tank. The vent lines are equipped with flow meters and a sampling station. Each compressor has two lube oil reservoirs equipped with sight glasses to determine the oil level. A 40 horsepower electric motor inside each containment vessel drives the compressors. Pressure in the containment vessel is maintained below compressor suction pressure to insure coolant leakage is always away from the primary system. Typical design performance for two compressors in series is as follows:

	Stage I	Stage II
RPM	1180	1180
Intake Pressure	288 psia	308 psia
Compression Ratio	1.070	1.062
Discharge Pressure	308 psia	327 psia
Intake Temperature	120°F	143°F
Discharge Temperature	143°F	162°F
Brake Horsepower	19.7	18.6
Intake CFM	118	120
Blow Back CFM	---	10
Flow lbs/hr	9700	9700

An electrical interlock of the motor controllers assures that both stages start simultaneously. The two compressors are not balanced for volume flow, thus requiring a bleed or blow back of about 10 CFM from the second stage output to the interstage piping. An automatic transfer system is provided to start and valve-in the standby compressors (should this system be armed). Switch over from one bank of compressors to the other cannot be performed without causing a low loop flow reactor scram, thus there are no safety requirements for the standby compressors to always be operative and held as a ready backup. Both compressor banks are supplied by the normal electrical supply system.

Regenerative Heat Exchanger: The gas to gas regenerative heat exchanger is a 32-foot long unit which extends from the mezzanine floor to the reactor building basement in a special shield. The shell of the heat exchanger is 8-inch pipe. The exchanger contains 60 tubes with a heat transfer area of 290 square feet. Half-moon baffles are used in the shell side to provide cross-flow. On the tube side the gas enters at $160^{\circ}F$ and leaves at $680^{\circ}F$. On the shell side of the heat exchanger the gas enters at $820^{\circ}F$ and leaves at $290^{\circ}F$. Operating pressure will be 318 psia on the tube side and 295 psia on the shell side. With these conditions the exchanger will transfer about 6.25×10^{5} Btu/hr with three facility tubes in operation. The total flow through the three facility tubes will always flow through the regenerative heat exchanger. The heat exchanger is built to conform with ASME Code requirements and is Code stamped.

Facility Tubes and Fuel Assemblies: Up to five facility tubes, side by side, occupy a sector of the pool space adjacent to the reactor pressure vessel on the canal side. The facility tubes are the hairpin type as shown in Figure 5.8.3. Access flanges are located at approximately the same level as the top of the reactor pressure vessel.

The fuel assemblies are located in the facility tube leg adjacent to the reactor pressure vessel and centered at the reactor core mid-plane. The fuel assembly is supported by mechanical stops at the lower end of the facility tube. Typically, a 3/16-inch stainless steel fuel element lifting cable is attached to the facility tube access flange. This cable is used for insertion and removal of the fuel element and also acts as a thermocouple lead support. The coolant flow is from top to bottom over the fuel assembly.

A typical fuel assembly consists of 18 fuel pins contained in a shroud tube. The fuel pins contain enriched UO_2 in the form of pellets and are clad in Hastelloy X, 0.030 inches thick. The outside diameter of the fuel pin is 0.241 inches. The outside diameter of the fuel test assembly is 1.75 inches and it is 33 inches in length. The active fuel length is 22 inches. Space is provided above the active fuel, in each fuel pin, for gas expansion.

The fuel pins are positioned and supported in the test assembly by use of Hastelloy-X spiders. A connecting rod extending up the center of the test assembly maintains the proper spacing between the spiders. The lower ends of the individual fuel pins slide in guide holes in the lower spider to permit axial expansion. The fuel pins are equally spaced in relation to each other over the entire length. A spiral wrap of 0.040 inch Hastelloy-X wire around each fuel pin maintains this spacing.

The fuel assemblies are equipped with thermocouples to determine inlet and outlet gas temperatures. Some fuel assemblies may be equipped with additional thermocouples to measure clad and fuel temperatures. These data are used for experiment evaluation and are not required for loop safety.

Experiments are loaded and unloaded from the facility tubes during a reactor outage in the following manner: the water level in the reactor pool is lowered below the tube access flanges, the access flange is removed, the test assembly is lowered into place, the access flange is replaced and tightened and the instrument leads are connected. For removal of an irradiated test element the access flange is removed from the facility tube and the reactor missile shield placed over the pool. A grappling tool is lowered through a hole in the missile shield and attached to the fuel element lifting cable. The complete assembly is then pulled into a cask positioned over the hole in the missile shield.

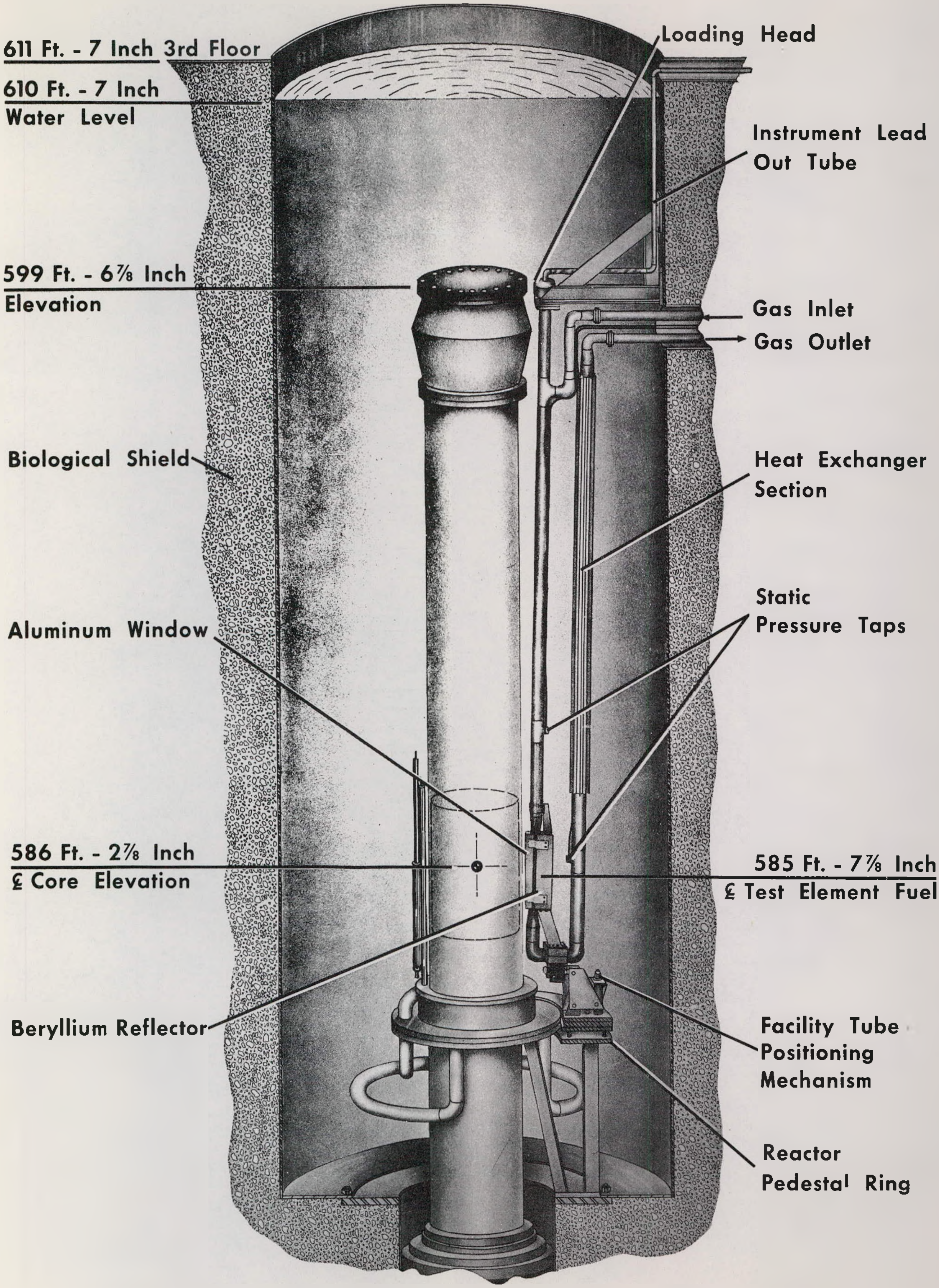

Figure 5.8.3 NITROGEN LOOP IN-POOL FACILITY TUBE

Each of the facility tubes may be operated with independent flow and temperature conditions. The small shielded cubicle outside the biological shield on the mezzanine floor contains the common inlet and outlet headers, individual flow control valves, heaters, filters, and control instrumentation. The inlet gas from the supply header first flows through the facility tube trim heaters for the final adjustment of inlet temperature. The inlet gas then enters the reactor pool through a flange penetration in the biological shield to the facility tube leg adjacent to the reactor pressure vessel. The coolant flow is in the downward direction over the fuel test piece where the temperature is increased as the energy is transferred from the fuel pins. The gas then flows up the outlet leg of the facility tube in the reactor pool.

A diagram of the in-pool portion of a facility tube is shown in Figure 5.8.3. A portion of the heat picked up by the coolant in the test section is transferred to the pool by means of a 9-foot long finned cooling section in the outlet leg of the facility tube. All in-pool piping (with the exception of the fin collar section) is insulated to maintain gas temperatures and prevent pool boiling on the outside of the piping. The finned cooling section in the reactor pool has a large surface area to prevent pool boiling. The exit gas enters the mezzanine cubicle through the inlet gas flange penetration in the biological shield. The individual facility tube outlet gas instrumentation in the cubicle measures radiation level, gas temperature, and flow. The outlet gas lines are also equipped with absolute filters and automatic flow control valves. A sample station from which gas samples may be obtained from the individual facility tubes is located near the mezzanine cubicle.

The facility tubes are supported at the top by an A-frame bracket which is attached to the periphery of the reactor pool. The bottom of each facility tube is guided by individually adjustable foot assemblies attached to the lower reactor pedestal ring. The facility tubes may be positioned radially with respect to the reactor core to adjust the power production in the fuel assemblies. The foot assemblies are actuated by an operator with a grappling tool. A total movement of $\pm$ 1-3/8 inches from the "normal" position is obtainable.

This movement will alter the thermal neutron flux in the test assemblies by about ± 50%.

Aluminum and beryllium filler pieces, surrounding the facility tube section, may be used. These filler pieces, shown in Figure 5.8.3, increase the specific power of the fuel test element to achieve the desired heat flux. The reactivity effect of two loop fuel elements on the reactor core has been examined and found to be less than 0.2% $\Delta k/k$.

Pre-Cooler: The gas pre-cooler is used to control the inlet temperature of the gas to the compressors. The gas-to-water heat exchanger has U-shaped tubes to allow for differential expansion between the tubes and shell. The unit has a surface area of 182 square feet. It has a capacity of 850,000 Btu per hour but is only required to transfer about 280,000 Btu per hour at normal operating conditions. Normal inlet water temperature of the shell side is about 85°F and the outlet temperature is about 100°F. The tube side has an inlet gas temperature of about 190°F and an outlet temperature of less than 120°F. The operating pressure is 292 psia. The over-all length of the heat exchanger is about 10 feet, and the diameter is 14 inches. Sixty-one U tubes make up the tube side of the exchanger. Water flow through the pre-cooler is regulated by a temperature controller to maintain constant gas temperature at the compressor intake. In addition, the water flow is measured and a low-flow alarm provided. High discharge water temperature will also actuate an alarm. The pre-cooler is fabricated in compliance with the ASME Code and applicable nuclear Code cases. The pre-cooler is located in the basement cubicle between the two compressor assemblies.

Gas Supply System: A gas supply system is provided to deliver gas at a constant pressure to the loop. This system is used to make up system pressure from leakage and the venting operation. The make-up system also supplies gas for emergency cooling. The supply system regulating valves maintain the operating pressure within ± 5 psi. Gas can be supplied from either conventional cylinders or from commercial gas tube trailer systems manifolded to the loop inlet.

An emergency or standby manifold of 5 standard gas cylinders is always held in reserve.

Purification System: An absolute type filter is located in each facility tube return leg. The filter is a cartridge type and can be removed from the line for storage and disposal. These filters are designed to remove micron size particles from the gas. A purifier unit, located in the basement equipment cubicle, is provided in a bypass line between the compressor discharge and intake. A dessicant material for removal of water vapor and impurities is used in this purifier. The volume of dessicant contained in the purifier is approximately 2 cubic feet. With the absolute filters taking out particles down to micron size and the gas purifier continually bleeding off a portion of the loop for purification, the system coolant is maintained at a high level of purity.

Dump Tanks and Vacuum Pump: The Nitrogen Loop is equipped with three 180 cubic foot dump tanks, located in the basement equipment cubicle. These tanks provide storage for removal of contaminated gas from the main loop and excess gas injected into the loop during blow down (emergency cooling).

When it becomes necessary to empty the loop (during shutdown), the gas temperature is allowed to drop to less than 200°F and the loop is isolated from the gas supply. A manually operated valve is opened to admit the system gas to the dump tanks.

The emergency cooling system is designed to operate for 10 minutes. The volume of gas discharged into the three dump tanks during the 10 minute blowdown is about 150 cubic feet, or about the volume of one tank. The gas is retained in the tanks at less than 100 psi until the radiation level permits gradual venting to the stack. A safety valve on the tanks prevents overpressure and is sized to take the full capacity of the 1/2 inch inlet line if the loop is inadvertently dumped. The safety valve exhausts to the stack through the stack isolation valve.

A two-horsepower standard roughing-type vacuum pump is located in one of the lines leading to a dump tank. This vacuum pump will serve two purposes. First, it is used to evacuate air from the system before introducing coolant gas in the charging operation. The system may have to be purged several times to get the contaminants down to an acceptable level. Secondly, the vacuum pump is used to pump contaminated gas from the system if the need arises. The pump will be used only after the coolant pressure has been reduced to atmospheric.

Vent Tank: A vent tank receives gas through flow control valves located on the vent lines from each operating motor-compressor containment vessel during normal operation. The vent tank is provided for the purpose of retaining the gas in case of abnormal activity or contamination. The gas is held at approximately 50 psig and subsequently exhausted to the stack. A safety valve is located on this tank to protect against overpressure. The vent tank, like the dump tanks and all other pressure vessels in the system, is fabricated in accordance with ASME Code for Unfired Pressure Vessels. The vent tank has a volume of approximately 70 cubic feet.

Emergency Cooling System: The Nitrogen Loop includes a blow down system for emergency cooling. This system will be used only when unplanned circumstances require it. The system consists of the gas supply system plus the five bottles being maintained in reserve. The emergency cooling system will operate automatically upon loss of flow after the reactor scrams. The emergency cooling system has sufficient capacity to cool the fuel elements after a loss of flow until decay heat generation is low enough to maintain safe test element temperatures by natural convection and heat losses through the loop. The blow-down valve shown in Figure 5.8.2 opens on loss of loop flow (which also scrams the reactor). This valve will close automatically 10 minutes after initiation of the emergency cooling. The flow will be approximately 100 pounds per hour for each facility tube. The clad temperatures reached during this accident are discussed in Section 5.8.4.

Control and Instrumentation: All instrumentation needed for the loop is centralized at the local control panel located in the experiment area. See Figure 5.8.1. Those critical parameters which are capable of causing a reactor scram are repeated and recorded in the reactor control room.

An annunciator in the control panel indicates the source of the trouble. Critical parameters actuate both alarms and reactor scrams. Table 5.8.4 lists the various parameters being measured, the location of the recorders, and the levels of alarms and scram settings for typical tests.

The loop system is fully instrumented to measure radiation levels, coolant temperatures, flow rates, and pressures. All points of interest throughout the system will furnish data on recorders, thus providing information for post-operational analyses. Suitable instrumentation is provided for reactor shutdown should a potentially serious situation be indicated. Reactor scram will be initiated by high exit gas temperature at the mezzanine cubicle, high activity in the exit gas piping, low flow in the facility tube piping, high pressure differential across the by-pass line, and low compressor discharge pressure. The system is laid out in such a way that all parameters concerned with the operation of the system and the contained experiments are under the direct control of the loop system operator. Facility tube gas temperature, mass flow, loop system pressure, and differential pressure are automatically regulated as long as conditions stay within the normal operating range and, consequently, do not usually change enough to require operator intervention. Should a transient occur, the operator will be able to make suitable system adjustments to bring about system equilibrium conditions. Fast-acting transients involving temperature and flow excursions will cause an automatic scram of the reactor when these parameters exceed pre-set limits.

Gas mass flow to each facility tube is controlled by one of four thermocouples measuring exit gas temperature in the mezzanine cubicle. Thermocouple failure will drive the recorder controller upscale opening the flow control valve. Manually switching to another thermocouple corrects the trouble. Alternate thermocouples are used for reactor scram on high exit gas temperature. Inlet gas temperatures is controlled by one of the two inlet gas temperature thermocouples. This temperature controls electrical power to the trim heater through a saturable reactor. Thermocouple failure drives the recorder controller upscale, shutting off the trim heater. Manual switching to the remaining couple corrects the trouble. High inlet and outlet gas temperatures also will actuate alarms.

Fuel element thermocouples for measuring clad and fuel temperatures will be provided on some test pieces. This instrumentation is not used directly for loop system control or reactor scram. The data obtained will be used primarily for monitoring fuel element performance and post-operational analysis.

Exit gas flow and radioactivity from each facility tube will be recorded. Flow dropping below preset levels will first cause an alarm and then an automatic reactor scram. High radiation levels also will alarm at a preset level and cause an automatic reactor scram at a higher set point.

Pressure, temperature, and flow at the compressor discharge will be recorded. Low pressure alarm and automatic scram instrumentation are provided. The main loop bypass valve will be controlled by the pressure differential across the loop from the cold side regenerator inlet to the hot side exit. This bypass valve automatically maintains a constant compressor load.

Water flow through the precooler will be regulated by a temperature controller to maintain constant gas temperature at a compressor intake. In addition, the water flow will be indicated and low flow will cause an alarm. High discharge water temperature also will cause an alarm.

Additional temperatures, pressures, and flows will be measured to provide information required for efficient operation of the loop. Pressure switches on the nitrogen supply system will actuate alarms on low pressure.

<u>Shielding</u>: All of the radioactive portions of the test facility are

shielded. The major portion of the loop and its equipment is located within a shielded cubicle on the basement floor of the containment building. A portion of the equipment is located in a separate cubicle on the mezzanine floor. A shield is provided around the regenerative heat exchanger between the mezzanine floor and the basement. The shielding walls are concrete blocks, sized to limit radiation to acceptable levels. On the basement floor one wall of the cubicle is formed by the reactor biological shield; the second wall of the cubicle is the reactor containment vessel.

An exhaust system provides a continuous flow of air into the equipment cubicle from the containment building and out of the reactor containment building through an absolute filter to the reactor building stack. This flow is approximately 600 cubic feet per minute. The pressure inside the shielded cubicle will be maintained at approximately 1/4-inch of water less than the pressure outside the cubicle. The radiation in the vicinity of the local control and power panels will be reduced to acceptable levels. The loop piping between the reactor pool and the shielded cubicle on the mezzanine is in the biological shield of the reactor.

5.8.3 Operating Parameters and Procedures

The parameters described in this section and in Figure 5.8.4 should be considered as typical steady-state values for most tests. Flow rates in pounds per hour, temperatures, and pressures for the coolant are shown at various locations throughout the system. The table below summarizes the basic parameters of the system for typical fuel assemblies:

Number of facility tubes	3
Active fuel length of assemblies	22 inches
Maximum test space diameter	1.76 inches
Average cladding temperature	1500°F
Maximum cladding temperature	1820°F
Cladding material	Hastelloy X
Cladding thickness	0.030 inches
Fuel pin diameter	0.241 inches
Number of fuel pins per assembly	18
Fuel material	UO_2
Fuel enrichment	93%
Thermal neutron flux at test section midplane	10^{13}
Average heat flux (Btu/hr-ft^2)	216,000

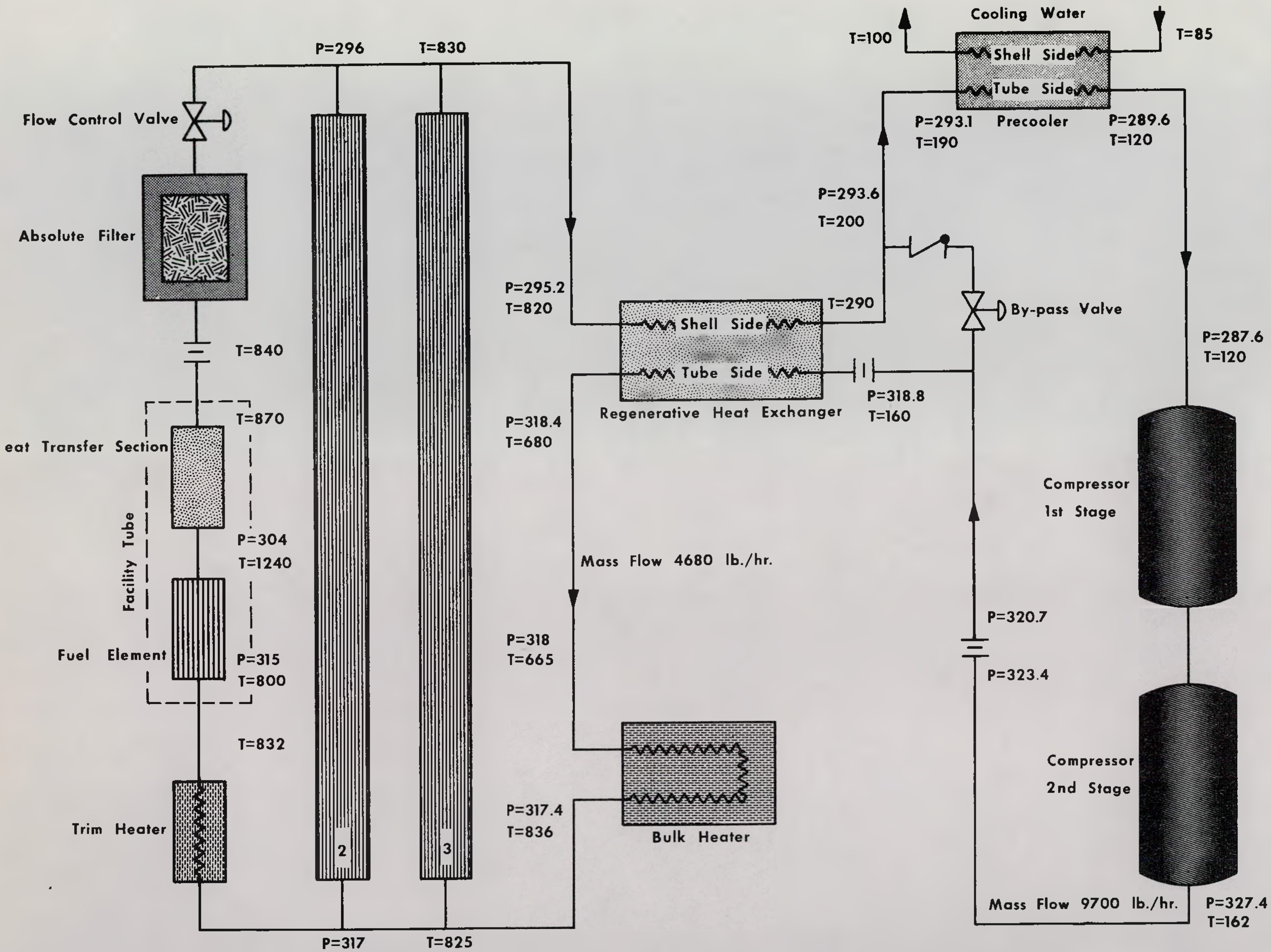

Figure 5.8.4. TYPICAL NITROGEN LOOP OPERATING CONDITIONS

Maximum heat flux (Btu/hr-ft^2)	256,000
Coolant	Nitrogen or Air
Normal operating pressure	315 psia
Test section inlet temperature	800°F
Test section outlet temperature	1300°F
Coolant flow per facility tube	1560 lbs/hr
Heat production per facility tube	55 KW

In accordance with the laws of California, the Nitrogen Loop was not operated until approved by the Division of Industrial Safety. This approval required the submittal of partial data reports which indicated that each loop component conformed with the appropriate section of the ASME Code and all applicable approved code cases such as 1270N and 1273N. Each portion of the loop was designed for the maximum pressure and temperature which could be attained.

5.8.4 Nitrogen Loop Safety Analysis

5.8.4.1 Loop Coolant

The Nitrogen Loop is designed to operate with either helium, nitrogen, air or a mixture of these gases as the primary coolant. The basic analyses presented below is for nitrogen or air, or a combination of the two.

The physical properties of nitrogen and air at system conditions (800°F and 300 psi) are essentially identical. These properties are listed below:

	Nitrogen	Air
Specific Heat Btu/lb	0.266	0.263
Ratio of Specific Heats Cp/Cv = K	1.40	1.40
Density (ratio to nitrogen at the same conditions)	1.0	1.03
Viscosity (centipoises)	0.0332	0.0323

The only difference in loop operation for use of nitrogen or air might be a decrease in rated flow of approximately 2% (for air operation) to compensate for the increased density and decreased specific heat to maintain the rated gas stream temperatures. No changes in loop alarm or scram set points or other conditions of operation would be required.

The effect of increasing the oxygen content in the coolant on the oxidation corrosion of loop components at high temperatures has been evaluated. The following table lists the loop components which operate at elevated temperatures, their materials of construction, the maximum operational temperatures, and the safe operating temperature limits in high temperature oxidizing gases (temperature at which scaling is negligible for the expected life of the components).

		Max. Operating Temperature	Safe Operating Temperature
(a)	Low alloy and carbon steels		1000°F
	1. Absolute filter bodies	900°F	
	2. Regenerative heat exchanger (shell and tube sides)	900°F	
	3. Control valve bodies	900°F	
(b)	Austenitic stainless steels		1650°F
	1. Piping	900°F	
	2. Trim heaters	900°F	
	3. In-pool piping and facility tubes	400°F	
(c)	Inconel		2000°F
	1. Bulk heater-cone and flanges	800°F	
(d)	Nichrome		1800°F
	1. All heater elements	1200°F	
(e)	Hastelloy X		2300°F
	1. Fuel element clad	1820°F	

As shown in this table, the high temperature loop components will operate at temperatures well below the safe operating temperature limits, to prevent damage due to oxidation corrosion. All other loop components (which are also composed of the above materials) operate at sufficiently low temperature to neglect any oxidation corrosion effects from the dry gas stream.

The induced radioactivity in the primary coolant is higher with atmospheric air than with nitrogen due to the increased argon content. The argon content increases, at most, by a factor of about 20 (500 ppm to about 10,000 ppm). The other activation products in the primary coolant remain essentially unchanged in comparison to the argon activity.

The following table presents the calculated loop coolant activities for (1) nitrogen coolant (99.5% nitrogen, 0.5% oxygen) using extremely conservative assumptions and (2) atmospheric air (highest argon content of all loop coolants proposed) using actual loop parameters.

CALCULATED LOOP COOLANT ACTIVITIES

($\mu c/cm^3$)

Isotope	Nitrogen Coolant (1)	Atmospheric Air (2)
Argon-41	0.120	0.345
Carbon-14	0.0025	0.00195
Nitrogen-13	0.00022	0.00017
Nitrogen-16	0.00022	0.00017
Xenon-132	1×10^{-5}	1×10^{-5}
Xenon-134	3.8×10^{-6}	3.8×10^{-6}
Krypton-85	3.3×10^{-6}	3.3×10^{-6}
Totals	0.123	0.354

The design calculations of coolant activity and dose rate used several conservative assumptions. Two of these assumptions are (1) three in-pile facility tubes (versus the two which are installed) and (2) a conservative estimate of the total loop gas volume (the loop volume used in the original calculation was 20 ft^3 and the actual value is 30 ft^3). Additional calculations, performed after the loop was operational, show that if the above-mentioned assumptions are corrected to the actual loop system, the dose rates measured can be confirmed.

The coolant activities presented in column (2) above are calculated using the actual loop system volumes and the highest argon content expected. Typical steady state dose rates from the loop system operating with atmospheric air are:

1. Surface of compressor vessels 40 mr/hr
2. Surface of regenerative heat exchanger shield 8 mr/hr
3. Loop radioactivity monitor readings 114 mr/hr

The above values are based on a steady state coolant activity of 0.354 $\mu c/cm^3$ (Column (2) above).

Loop components are located in shielded cubicles which are provided with radiation monitoring equipment. The operating console for the loop is located outside the basement cubicle and personnel are not required in the cubicles for loop operation. The cubicles are designed to provide personnel protection for activities several orders of magnitude greater than those present.

In addition to the continuous venting of the compressor vessels to the vent tank, the loop system leaks a small amount of primary coolant (as detected by makeup requirements to the loop). Primary coolant leaks are not easily detected by leak testing techniques although a continued effort is being made to locate and repair any leaks. Leakage from the loop is confined to the loop equipment cubicles, mentioned above, which are ventilated and exhaust directly to the reactor stack. The contribution of the coolant leakage leaking to the permissible stack discharge limit would be about 9% assuming 0.354 $\mu c/cm^3$ and no decay of radioactive constituents. To date no detectable change in the reactor stack discharge activities have been noticed as a result of the loop operation with either air or nitrogen coolant.

There is a possibility that if the fuel clad should fail, further oxidization of the UO_2 fuel will occur. Such an accident would be exothermic and has been evaluated to determine the energy release. This accident would cause the loop instrumentation to scram the reactor due to high radioactivity in the loop coolant. There are about 1210 grams of UO_2 in two typical fuel elements in the loop. The oxidization reaction converts UO_2 to U_3O_8 producing about 93 calories per gram of UO_2 converted. Thus a total of 112.5 kilo-calories could be produced if all the UO_2 were oxidized. A conservative assumption is that it would take from 20 to 30 minutes for this reaction to go to completion (complete oxidation of all of the fuel). If it is assumed that the reaction goes to completion in 20 minutes, the total energy produced would be less than .4 KW or about 0.3% of steady state power. It is not credible to assume that all fuel pins in both elements (36 pins total) would rupture simultaneously, but should this remote accident occur the energy produced would be small in comparison to the loop operational power, or the decay power after reactor scram.

The possibility of an oil-air explosion has been evaluated and found to be extremely small. The operating compressors, lubricated with Shell-Tellus-69, produce some oil vapor. This oil vapor is removed from the loop system by continuously venting the compressor containment vessels.

Primary loop coolant leaks from the compressors to their containment vessels (which operate at about 30 psi less than loop pressure) sweeping oil vapor away from the circulating primary coolant. An oil vapor concentration of 7000 ppm (in air) is required to propagate a flame front. Such concentrations cannot be obtained in the primary loop coolant due to the compressor vent system. Tight control of the primary coolant constituents is maintained and the hydrocarbon content, plus other impurities, is measured by sampling the loop gas. Primary coolant gas analysis has shown less than 200 ppm hydrocarbons. Further, the coolant purification system which operates continuously will remove hydrocarbons from the primary gas. The purification system flow is measured and equipped with a low flow alarm to insure flow through the unit. The system is periodically checked for effectiveness, particularly for the removal of hydrocarbons.

The possibility of formation of nitric acid in the Nitrogen Loop exists although the probability of accumulating quantities capable of damaging the experiment or equipment is small. An analysis has been performed to determine the rate of formation and the equilibrium concentrations of nitrogen fixation products in the loop. This analysis indicates that nitric acid formation will not constitute a safety problem in the loop. Examination of the loop after extended operation (with air as the primary coolant) has shown that this conclusion is correct.

5.8.4.2 Nitrogen Loop - Mechanical Accidents

In the evaluation of potential accidents for the Nitrogen Loop, the prevailing conditions are assumed to be pressure at the fuel element of 315 psia, (compression discharge pressure of 327 psia), fuel element power of 60 kw (maximum power for the majority of the fuel elements), flow of 1560 lbs/hr, and maximum clad temperature of 1820°F.

The general categories of accidents discussed are loss of coolant flow, loss of electrical power, loss of secondary cooling, component malfunction, operator error, and mechanical accidents. These are considered in turn in the following sections. Following these, the radiation levels and release of coolant and fission products are discussed.

5.8.4.3 Loss of Coolant Flow

A loss of coolant flow could occur from several causes, the principal ones being compressor failure, system rupture, and loss of electrical power. The results of such an accident will be considered here without stipulating a particular cause since the cause has little effect on the results.

The failure of one compressor bank will result in the automatic change-over to the alternate compressor bank (if operative) when the flow drops to a predetermined value. A manual change-over to the alternate compressor bank is provided also. The total loss of compressor circulation produces a decrease of flow from 100% to essentially no flow in less than one second which, in essence, is an instantaneous loss of flow in the main loop. This accident would cause an increase in fuel element temperature and, therefore, has been analyzed in detail. The loop instrumentation would detect the low flow condition and initiate an automatic reactor scram when the flow dropped to 70% of normal. At about the same time, the emergency blowdown valve would open supplying about 300 pounds per hour of nitrogen through the facility tubes to the dump tank. The primary coolant flow drops from 100% to 6-1/2% in less than one second. The emergency cooling system then maintains a flow of about 6-1/2% through the facility tube for the next 10 minutes. The reactor power would drop to about 6% of normal as a result of scram one second after the start of the loss of flow transient.

The effect of the loss of flow accident is an increase in fuel clad temperature of about 250°F above the maximum steady state temperature of 1820°F during the first four seconds after the reactor scram. The maximum transient clad temperature produced by the accident would be about 2070°F. The fuel element temperatures reach the maximum in about four seconds and drop to less than 1600°F at the end of one minute. The results of this accident are shown in Figure 5.8.5. The figure presents reactor power, facility tube flow and the maximum clad temperature as a function of time after the loss of flow.

The above is based on a fuel element power of 60 KW; however, because this is the most severe accident, the results have been evaluated for a fuel element power of 65 KW. The corresponding fuel element temperature transient is essentially the same as the 60 KW fuel tests, except that the maximum clad temperature produced is 2095°F.

Test data for typical fuel elements, clad with Hastelloy-X, indicate that slight deformation of the fuel pins may occur above 2100°F but no melting or loss of element integrity is expected at temperatures below 2300°F. It is concluded that the fuel clad temperatures reached during a loss of flow accident are safe and will not cause melting of the cladding.

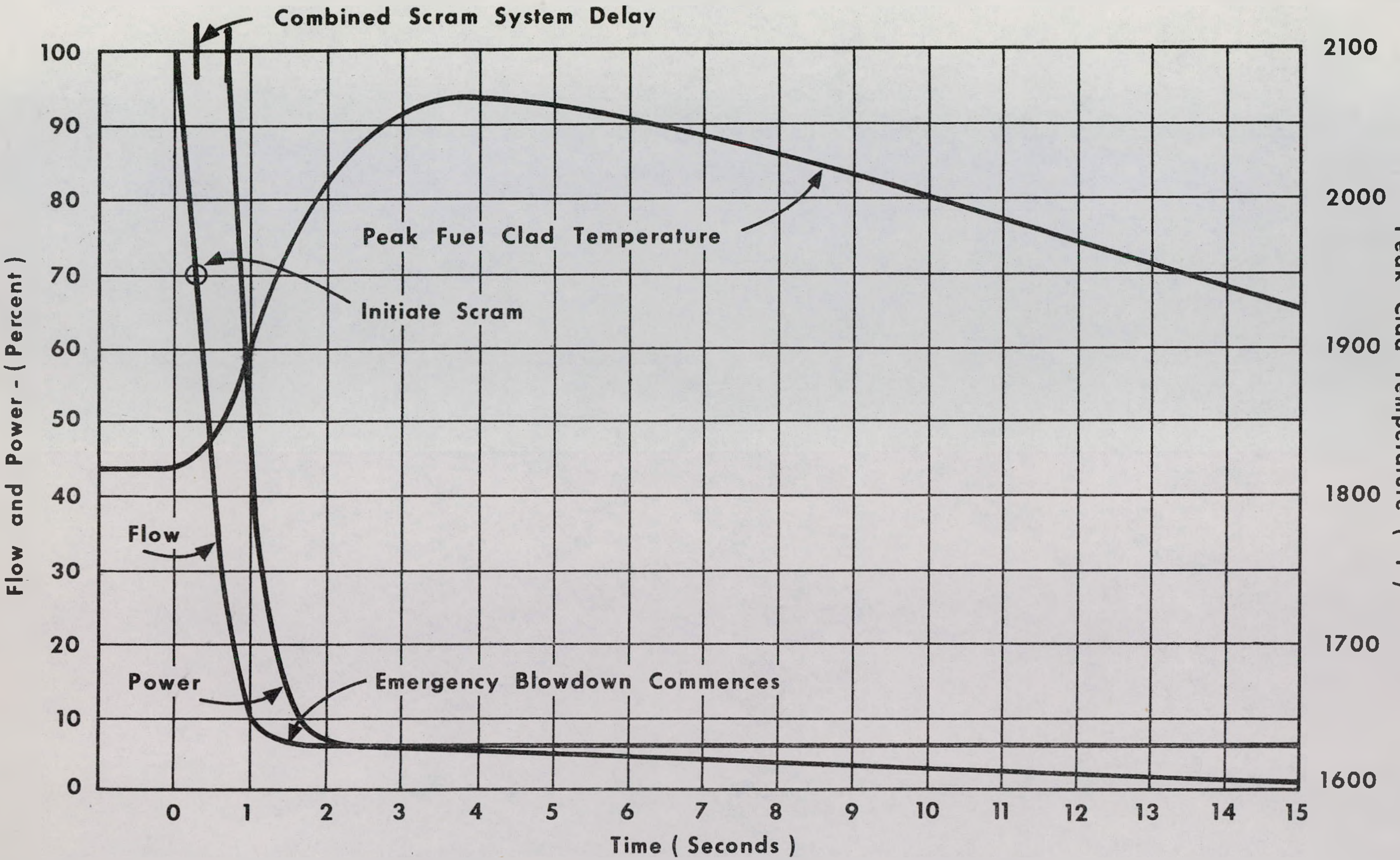

Figure 5.8.5 NITROGEN LOOP LOSS OF FLOW TRANSIENT

VAL 1445

5.8.4.4 Loss of Electrical Power

In case of loss of normal electrical power the reactor will scram immediately and the operating compressors will stop. Loss of normal electrical power would produce a loss of flow transient similar to that described in Section 5.8.4.3 but would result in lower transient clad temperature because the reactor scram would occur at the same time that the operating compressors stopped.

All valves operated by electro-pneumatic transducers in the loop fail safe or are powered by the emergency diesel generator to permit the loop coolant to flow through the facility tubes and also to operate the emergency cooling system.

5.8.4.5 Component Malfunctions

The Nitrogen Loop is provided with sufficient instrumentation and safety circuits to alert the operator to equipment malfunctions or to automatically scram the reactor when required. In general, most malfunctions will not cause a dangerous situation immediately, and the loop operator can take corrective action or have the reactor scrammed. Those conditions which cause immediate danger, such as stoppage of primary flow, are detected by instrumentation and cause an automatic reactor scram. Typical examples of equipment malfunction to the Nitrogen Loop are outlined below.

An open pressure relief valve or blowdown valve will allow about 200 pounds per hour of nitrogen to escape to the dump tanks. The makeup system would maintain system pressure for about 30 minutes, should this condition continue. The operator would be alerted by an increase in dump tank pressure of about 3-1/2 psi per minute. In addition, the low gas supply pressure alarm will actuate before the makeup supply is exhausted.

This accident allows adequate time for operator corrections to be made or shutdown of the reactor before any damage to the system can occur.

An open high-pressure-supply regulating valve in the supply manifold would dump large quantities of gas into the main loop system causing the pressure relief valves in the main loop and the compressor-motor containment vessels to open at a system pressure of 400 psi. These relief valves vent to the dump tanks. In this event the operator would be warned by an increase in dump tank pressure, a high primary system pressure alarm, and low supply pressure alarm. Again this accident allows the operator to make corrections prior to requiring a reactor shutdown.

The main loop bypass valve is designed to fail closed to provide the maximum flow available to the facility tubes in the event of power failure. This valve is automatically controlled to maintain a constant pressure differential across the compressors. If the main loop bypass valve should fail open or is inadvertently opened, most of the primary loop coolant would bypass the facility tubes. About 500 pounds of gas per hour would still pass through each fuel test section if the bypass valve were fully open. The result would be a sudden drop in flow at the facility tube which would cause an automatic reactor scram and automatic opening of the loop blowdown valves. Malfunction of the bypass valve would result in a loss of flow accident less severe than the total loss of flow accident described in Section 5.8.4.3 because of the higher facility tube flow available during the period immediately after scram.

All loop instruments causing reactor scram are equipped with three individual circuits for each parameter being monitored. Two coincident signals from an individual parameter are required to scram the reactor. Failure of one individual safety circuit will produce a signal (to scram) but the reactor will not scram until two coincident signals are received. Failure of one instrument also causes an alarm notifying the operator of the trouble. The loop scram instrumentation is listed in Table 5.8.4.

Typical examples of instrumentation failures which could possibly occur are:

A. Test section inlet-gas-temperature thermocouple failure shuts off the facility tube trim heater. This situation can be corrected by manually switching to a duplicate standby thermocouple.

B. Test section outlet-gas-temperature thermocouple failure opens the individual facility tube flow control valve. Corrections can be made by manually switching to a duplicate standby thermocouple. Three exit gas thermocouples are normally used in the scram circuit so that failure of one (up scale burnout) would cause an alarm but not scram the reactor. Failure of a second thermocouple would result in a scram.

C. The main loop flow instrumentation is fail-safe. Failure of main loop flow instrumentation or low flow causes an automatic compressor change-over at a pre-set value. If main loop flow drops to a second set point (low-low flow) the loop blowdown valve opens and the by-pass valve closes, both automatically. At 70% of normal flow, the reactor will scram.

D. The instrumentation provided on some fuel test elements will be used for informational purposes and will not be used in the safety circuits. Failure of this type of instrumentation will not affect the safety of the loop system.

5.8.4.6 Operator Errors

During normal operation of the Nitrogen Loop the operator manually controls the bulk gas heater temperature. This heater (and the individual trim heaters) is automatically shut off on low system flow and low system pressure. All other loop controls are automatically operated. In addition, manual backup control of compressor change-over,

TABLE 5.8.4

Nitrogen Loop Instrument Table and Setpoints For Typical Tests

Parameter	Recorder Location		Alarm		Scram	Units
	Loop Console	Reactor Control Room	High	Low		
1. Compressor Discharge Pressure			335	280	250	psig
2. Facility Tube Flow	X	X	---	1320	1090	lbs/hr
3. Facility Tube Exit Gas Temp. (at pool penetration)	X	X	875	---	900	°F
4. Primary Coolant Activity	X	X	500	---	1000	mr/hr
5. By-Pass Δ P	X	X	40	17	50	psi
6. Supply gas pressure	X	---	---	700	---	psig
7. Compressor Intake Gas Temp.	X	---	140	---	---	°F
8. Facility Tube Inlet Gas Temp.	X	---	840	---	---	°F
9. Pre-cooler Secondary Outlet Temp.	X	---	120	---	---	°F
10. Pre-cooler Secondary Outlet Flow	X	---	---	50	---	%
11. Dump Tank Radiation	X	---	100	---	---	mr/hr
12. Main Loop Flow	X	---	---	80	---	%

facility tube flow control valves, and the main loop bypass valve is also available. The facility tube flow control valve has a 20% flow area stop to prevent this valve from being completely closed. All important temperatures, flows, pressures, and radiation levels in the loop are recorded. Alarms are set to warn the operator if operating parameters exceed pre-set limits.

If alarms are unnoticed backup protection on all critical parameters is provided by automatic reactor scrams. Reactor scrams are, in all cases, preceded by alarms.

5.8.4.7 Loop System Piping Rupture

All loop system components, including piping, valves, vessels, heat exchangers, etc., are designed and constructed in accordance with applicable ASME Code requirements. Overpressure is protected against by relief valves in the main loop and in the motor-compressor vessels. The complete system was pneumatically pressure-tested prior to startup to assure system integrity and leak tightness.

Rapid loss of system pressure is possible only in the event of a loop rupture followed by a rapid coolant discharge from the rupture. The loop has been designed to prevent the primary coolant from being discharged in both directions from the fuel elements, following a rupture. In all rupture accidents, part of the primary coolant inventory is forced past the fuel elements. The loop system was analyzed and the worst case was found to be a line rupture involving a complete shear of the loop piping in the area of the facility tube inlet line or in the facility tube inlet manifold. The assumption used to select this location as the worst case is that it permits the fuel cladding to reach the highest temperature during the ensuing blow-down period. In the event of a line rupture in the facility tube inlet section, the flow past the test fuel element immediately reverses in direction and exhausts from the end of pipe. The reactor scrams from low facility tube flow approximately

0.10 seconds after the pipe ruptures. The flow is normally about 0.4 pounds per second, but during loop rupture and blowdown it increases to about 0.6 pounds per second in the first half second after the rupture and then decays to zero in about 3 seconds. During the short period of high mass flow, the fuel element is subcooled due to the increased flow rate. Transient analysis of this accident indicates that the fuel clad will not reach the melting point. The maximum steady state clad temperature is about 1820°F. Two seconds after the pipe rupture the maximum clad temperature is reduced to about 1700°F. The clad temperature will then rise, due to reduced flow and decay heat, and reaches a maximum temperature of about 1860°F, seven seconds after the rupture occurs. This temperature is lower than that attained in the loss of flow accident described in Section 5.8.4.3.

The fuel element would be subjected to an upward force if the inlet facility tube line ruptured. If the fuel element was displaced upwards in the facility tube it could not be blown out of the loop because the pipe turns in the facility tube (see Figure 5.8.3). Fuel elements would not be damaged in this accident because the shroud tube prevents any mechanical damage to the fuel pins. If the facility tube ruptures on the outlet leg, the results of the accident would be similar to those discussed above. A rupture at this location would permit the emergency cooling system to pass gas over the element. (The blowdown emergency cooling system would discharge out the end of the pipe directly for the rupture case cited above.) The blowdown system would, in this case, limit the maximum fuel clad temperature to less than 1850°F. The fuel element would not be discharged out the end of the pipe due to the mechanical stops at the lower end of the facility tube, holding the element in position.

The compressors act as a check valve to prevent coolant blowdown (after a pipe rupture) in either direction through the compressors. System rupture in the area of the compressors would subject the fuel elements to a longer blowdown period than that discussed above for loop rupture in the supply manifold. The result would be less serious in this case. The main loop bypass line is equipped with an automatic flow control valve and a check valve. In the event of a rupture in the bypass line or in the check valve, the flow control valve will close (on a "high-high" differential pressure signal). If the flow control valve ruptures, the check valve will close. In either case, the bypass line will be blocked and thus force a portion of the loop inventory past the fuel elements. The resulting loop blowdown would continue for a longer period than the blowdown for the inlet manifold rupture case and the maximum clad temperatures would be lower.

5.8.4.8 Radiation Levels

The situation in which a fuel pin rupture occurs, allowing release of fission products to the system, has been considered. A fuel cladding rupture might occur if two fuel pins were deformed or warped to the extent that they were touching, producing over-heating and clad rupture.

The activity levels in the loop for this occurrence have been calculated with the following assumptions:

1. Rupture of two fuel pins.
2. Operation for the infinite length of time at full element power before rupture occurs.
3. Fuel melting does not occur.
4. Release of 10% of all gaseous fission products to the loop coolant.

On the basis of these assumptions, the loop coolant would contain about 9.4 millicuries per cubic centimeter of gaseous fission products immediately after shutdown. This would be reduced by a factor of ten after two hours. The highest radiation field immediately after the incident would be, at most, 450 mr/hr, occurring at the surface of the regenerative heat exchanger shielding. This will drop off rapidly and the exposure at all other parts of the system will be substantially less.

The Nitrogen Loop is fully instrumented to detect radioactivity in the loop system and in the equipment cubicle. Alarms are provided to warn operating personnel of abnormal activities and an automatic reactor scram will be initiated if high radiation levels are encountered.

5.9 High Temperature Helium Loop

5.9.1 Introduction and Summary

This section presents the description and accident analysis for the High Temperature Helium Loop. The first two years of operation of this facility at design power were very successful. The most significant result, other than establishing system reliability and operating characteristics, was that the actual quantity of fission products in the system was over-estimated in the original hazards evaluation by about five orders of magnitude. This result has, of course, simplified operation of the loop but it also proves that the original design specifications and estimates of the hazards were much too conservative. In an effort to retain the original design and hazards analysis philosophy (and its proven conservatism), the accident evaluation section for the High Temperature Helium Loop has, with only a few exceptions, not been changed. The description of the loop has been modified to include the minor changes made and to present actual performance data.

The High Temperature Helium Loop is designed to test graphite clad fuel assemblies under environmental conditions.

The facility consists of a single fuel assembly, a closed loop helium gas circulating system with provision for sampling fission products released from the fuel, a fission product trapping system and containment cubicles with a ventilation system containing filters. A diagram showing the principal components of the fuel and a typical heat balance diagram is shown in Figures 5.9.1 and 5.9.2. The position of the facility tube in the reactor is shown in Figure 5.9.3. Figure

5.9.4 is a diagram of the loop piping and instrumentation. Actual operating parameters representative of the first test are shown in Table 5.9.1.

The facility is designed to irradiate fuel specimens in helium at about 350 psia with a thermal neutron flux slightly higher than 10^{13}nv generating about 76 kw of fission and gamma power. The resulting heat flux will be near 150,000 Btu/hr/ft^2 and the maximum fuel temperature will be about 3500°F.

Information is being obtained regarding:

a. The physical stability of a prototype fuel assembly under irradiation.

b. Fission product release from the fuel.

c. Fission product deposition in loop components.

d. Fission product trapping system.

During normal operation, fission product contamination in the main loop and sampling system is expected. Multiple containment has, therefore, been provided to offer maximum protection. All in-pool equipment is located within an aluminum vacuum vessel and all other equipment in contact with the primary coolant is located in a steel lined shielded cubicle. Details of the components and their arrangement are given in the following sections.

5.9.2 Design and Test Requirements

The test facility was built in accordance with code requirements which are applicable to fabrication and construction. The loop and components were designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, for lethal substances and applicable nuclear code cases. Components of the loop which operate under helium pressure were mass spectrometer leak tested to a minimum sensitivity of 5×10^{-8} atmospheric cc/sec. The components were fabricated so that no leaks were detected at the sensitivity selected. During the shop fabrication phase, the out-of-pool portion of the main loop was pre-assembled and operated at temperature and pressure to verify system performance and leak tightness. The design objective was to reduce the leakage of the system to less

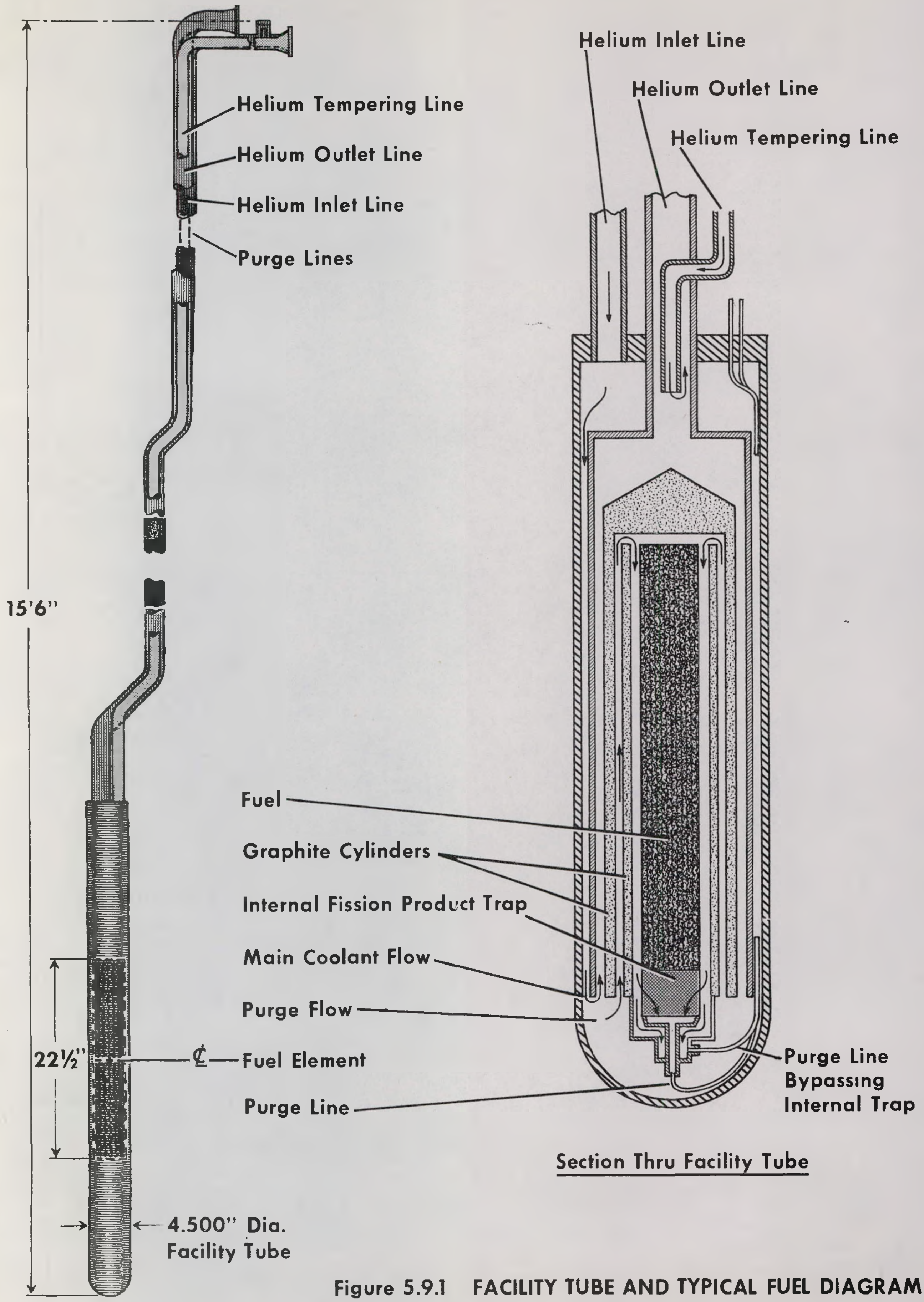

Figure 5.9.1 FACILITY TUBE AND TYPICAL FUEL DIAGRAM

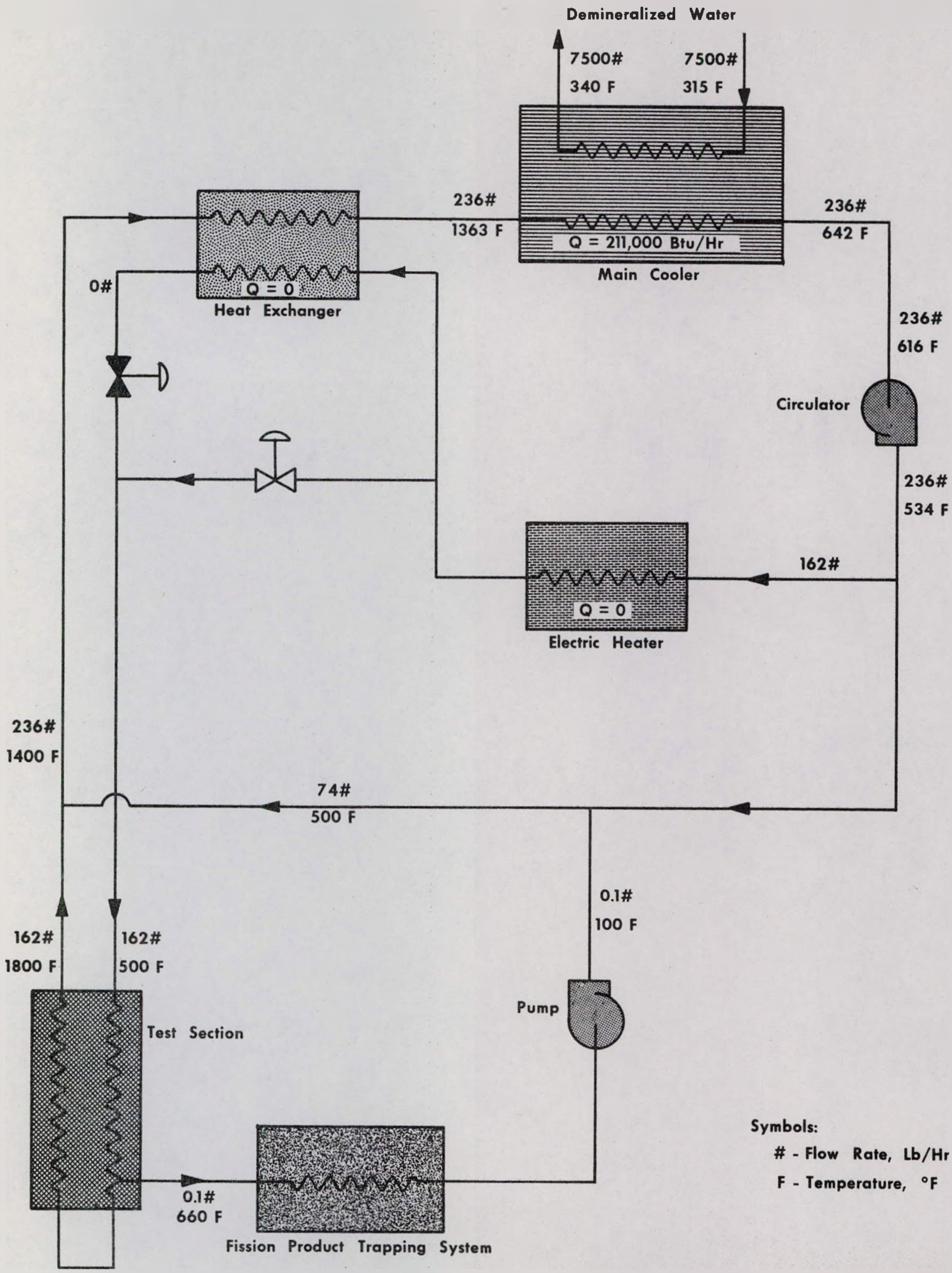

Figure 5.9.2 TYPICAL HEAT BALANCE DIAGRAM

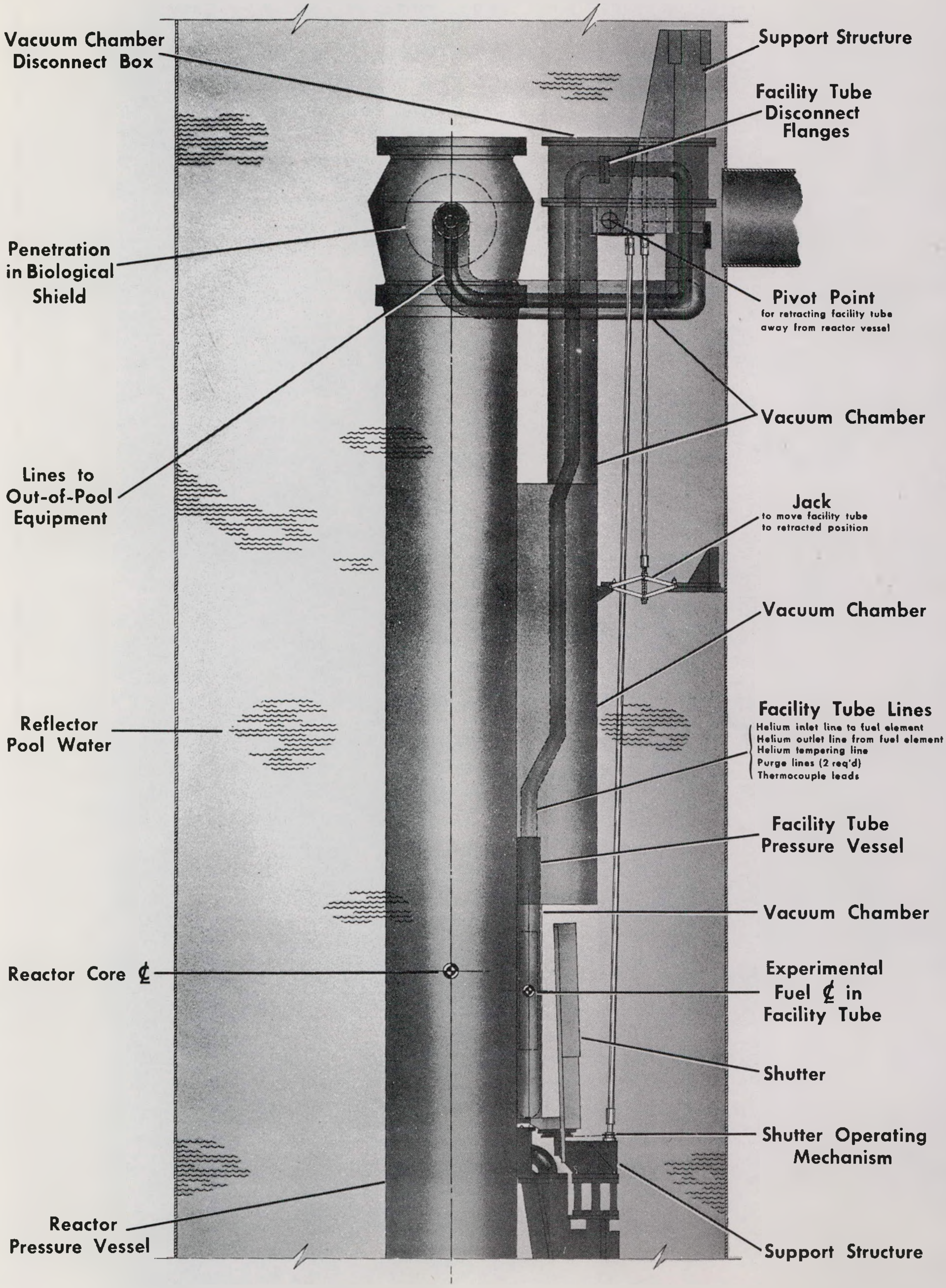

Figure 5.9.3 IN-POOL ARRANGEMENT

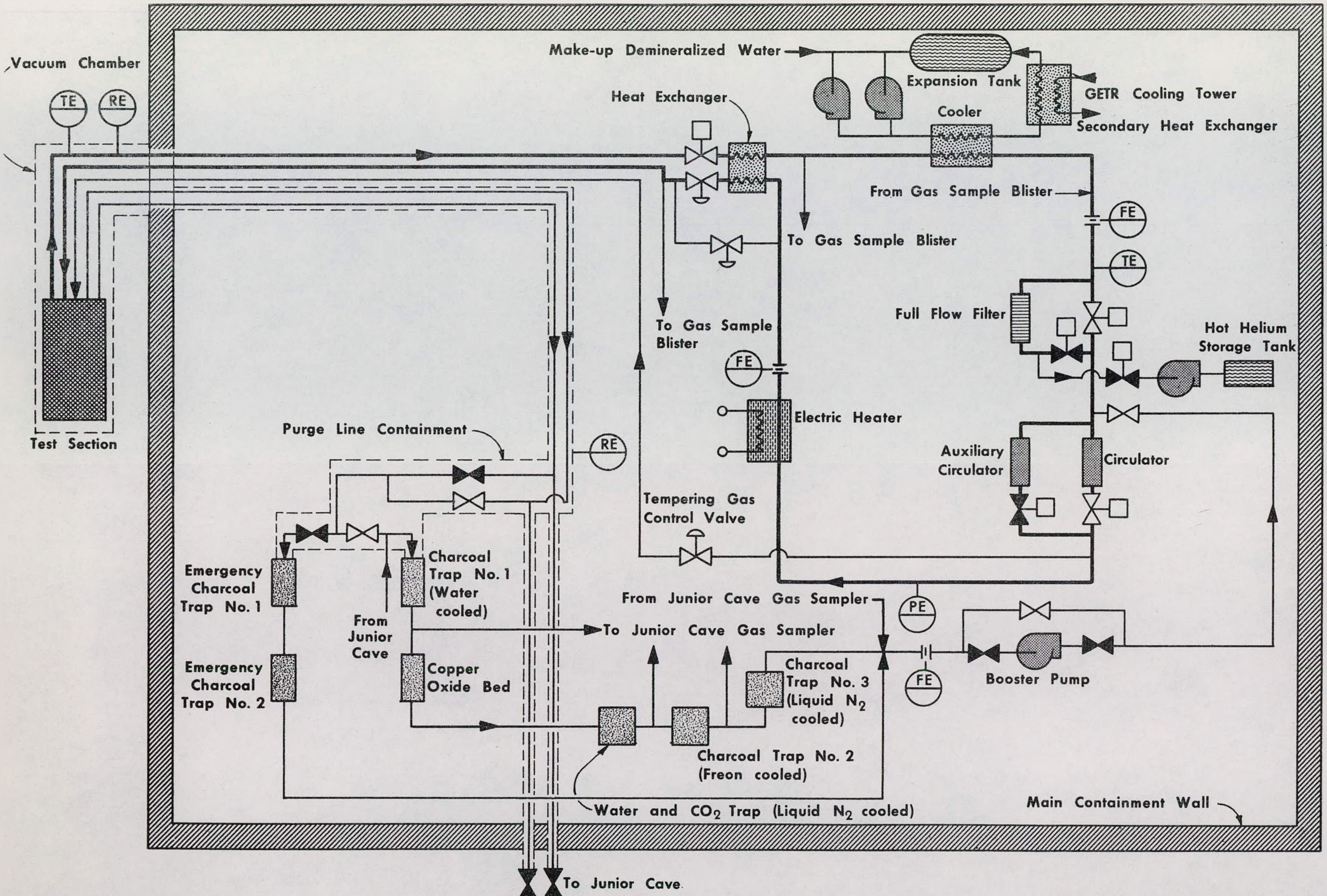

Figure 5.9.4 PRINCIPAL COMPONENTS OF THE LOOP PIPING AND INSTRUMENTATION

TABLE 5.9.1

TYPICAL OPERATING PARAMETERS

Function	Typical Operation	Alarm	Scram
Main Loop			
Facility Tube Flow (lbs/hr)	250	230	200
Facility Tube Exit Gas Temperature (°F)	1300	1440	1480
Facility Tube ΔP (psi)	8	12	-
Loop Pressure High (psia)	350	400	-
Loop Pressure Low (psia)	350	340	300
Internal Fuel Temperature (°F) (Maximum)	2700	-	-
Exit Pipe Temperature (°F)	1200	1300	-
Automatic Transfer to Aux. Circulator		Indicated by light	
Gas Temperature at Circulator Intake (°F)	400	620	660
Hot Helium Storage Tank Pressure (psig)	0	500	-
Auxiliary Systems			
Cooling Water Flow to Circulators (gal/hr)	120	100	-
Vacuum Chamber Pressure (microns)	100	1000	-
Cooling Water Temperature (°F)	340	360	-
Cooling Water Flow to Cooler (lbs/hr)	7500	95%	-
Low Pressure Holdup Tank Pressure (psia)	14.7	15	-
Fission Product Trapping System			
Outlet Gas Temperature from FPTS (°F)	-100	-50	-
Outlet Gas Temperature from Emergency Trap(°F)	-300	-250	-
Pressure in Purge Line Secondary Containment (psia)	13	17	-

than 1 x 10^{-5} atmospheric cc/sec, as measured by a mass spectrometer. At the reactor site the total loop was assembled and operated at temperature and pressure, using a heater and bypass line as a final check of system performance before significant power was generated in the test fuel.

<u>In-Pool Equipment</u>: The vacuum chamber contains the in-pool portion of the test facility which consists essentially of the facility tube, disconnect flanges and the piping connecting the facility tube with the portion of the system within the main cubicle. The general arrangement of the in-pool equipment is shown in Figure 5.9.3.

The facility tube contains the experimental fuel element and is positioned within the vacuum chamber vertically in the reactor pool. Piping within the vacuum chamber connected to the remainder of the loop outside the pool consists of the line bringing helium coolant to the fuel element, the line returning the heated helium to the heat exchangers in the main cubicle, a tempering gas line to bring helium from the circulator exit to a tempering chamber within the facility tube, and two purge lines for carrying fission products to the trapping and purification system.

The purpose of the vacuum chamber is to provide a thermal barrier between the pool water and the facility tube assembly and piping containing the flowing helium. It also provides secondary containment for radioactive fission products in case of leakage from the in-pool portion of the loop. The vacuum chamber extends up to the main cubicle at the pool penetration flange in the reactor biological shielding. The atmosphere of the vacuum chamber and the main cubicle are not directly connected.

The vacuum chamber is made of 6061-T6 aluminum. An enlargement in the chamber is provided in the portion near the reactor vessel head to contain the piping disconnects. A flanged plate permits access to the thermocouple and piping disconnects and allows removal of the facility tube. The vacuum chamber is designed to withstand the peak pressure expected if the system lines should rupture within the chamber. A four-inch rupture disk on the chamber rated at 15 psig releases the loop gas to the main cubicle in case of the rupture.

The vacuum chamber is flange-connected to the penetration flange in the pool liner within the biological shield penetration. The penetration connection is made of stainless steel.

Shielding is provided internally and externally to minimize radiation upward from the fuel and the lines carrying fission products.

During normal loop operation the facility tube is located in close proximity to the reactor vessel to take advantage of the high neutron flux. A means is provided to retract the vacuum chamber in the core region up to five inches from the reactor vessel for the case when it is desired to operate the reactor with the loop facility shut down. The jacking mechanism providing the movement is shown on Figure 5.9.3. In the retracted position a shutter consisting of lead, steel, and cadmium shields the fuel in the facility tube to minimize heat generation. During this time helium is provided in the vacuum chamber annulus assuring the necessary conductance for removal of the small amount of heat still being generated.

A means is provided to flood the vacuum chamber with water up to a pre-determined level during the refueling operation. Auxiliary systems connected to the vacuum chamber include the evacuation system and the emergency cooling system.

The facility tube assembly consists of a pressure vessel containing the fuel assembly and flow and thermal baffles; coolant inlet and outlet piping; tempering gas piping; purge gas piping; thermocouples; disconnect joints; and internal shielding.

A radial cross section of a typical fuel assembly is shown in Figure 5.9.5. The assembly consists of fuel compacts located in a graphite cylinder within a second graphite cylinder; graphite end plugs, heat insulators, connections, and structure; an internal fission product trap; purge lines; and thermocouples. Typical fuel compact consists of a central core of boron dispersed in graphite and an annular cylinder of U^{235} and Th^{232} dispersed in graphite. Heat is transferred to the helium coolant at the surface of the outer cylinder and the annular space between the inner and outer graphite cylinders is swept by helium gas to purge the fission products entering this space. The helium purge gas enters the outer annulus at the plenum in the lower end of the element, flows upward between the two cylinders to an upper plenum and downward between the fuel compacts and the inner cylinder. At the lower end the purge gas flows through an internal fission product trap or through a trap bypass purge line. The two purge lines are connected to the fission product trapping system in the main cubicle.

Experimental evidence indicates that there is a very high internal trap efficiency. Under normal operation the bypass line will be isolated by valves. The efficiency of the internal trap has been tested by periodic use of the bypass line.

The baffle separates the downward and upward coolant flow and virtually eliminates radiative heat transfer between the graphite and the pressure vessel. Stagnant helium between the baffle walls minimizes heat transfer between the two flowing helium streams. The baffle consists of an outer wall of stainless steel and two walls of Nichrome V. The weight of the fuel assembly is carried by the outer wall of the baffle.

The facility tube pressure vessel is made of type 321 stainless steel. It is designed for a pressure of 400 psig at a temperature of 1000°F. The upper end of the facility tube assembly consists of the helium coolant inlet pipe, outlet pipe, tempering gas mixing chamber and line, two purge lines, thermocouple ring, piping and thermocouple disconnects, and steel shielding. The tempering gas is brought from the circulator discharge to mix with the outlet helium flow for the purpose of decreasing the temperature of helium leaving the facility tube assembly at 1400°F maximum. Thermocouples are provided to measure temperatures of the fuel compact internals, helium at test section inlet, fuel element inlet, and fuel element outlet. Six thermocouples are attached to a thermocouple ring on the outlet pipe. Steel shielding is provided in the upper region of the facility tube to minimize radiation in a vertical direction above the fuel.

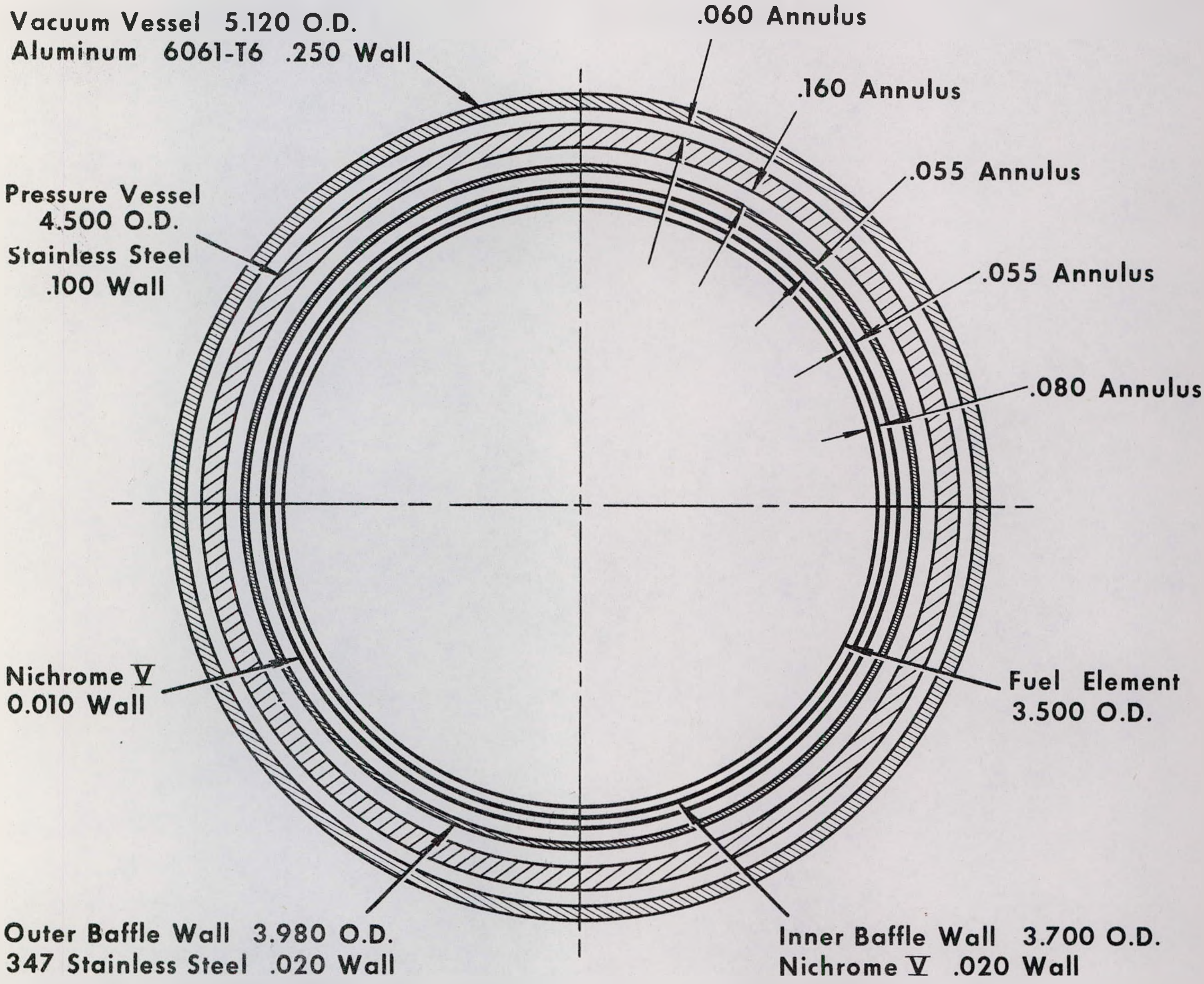

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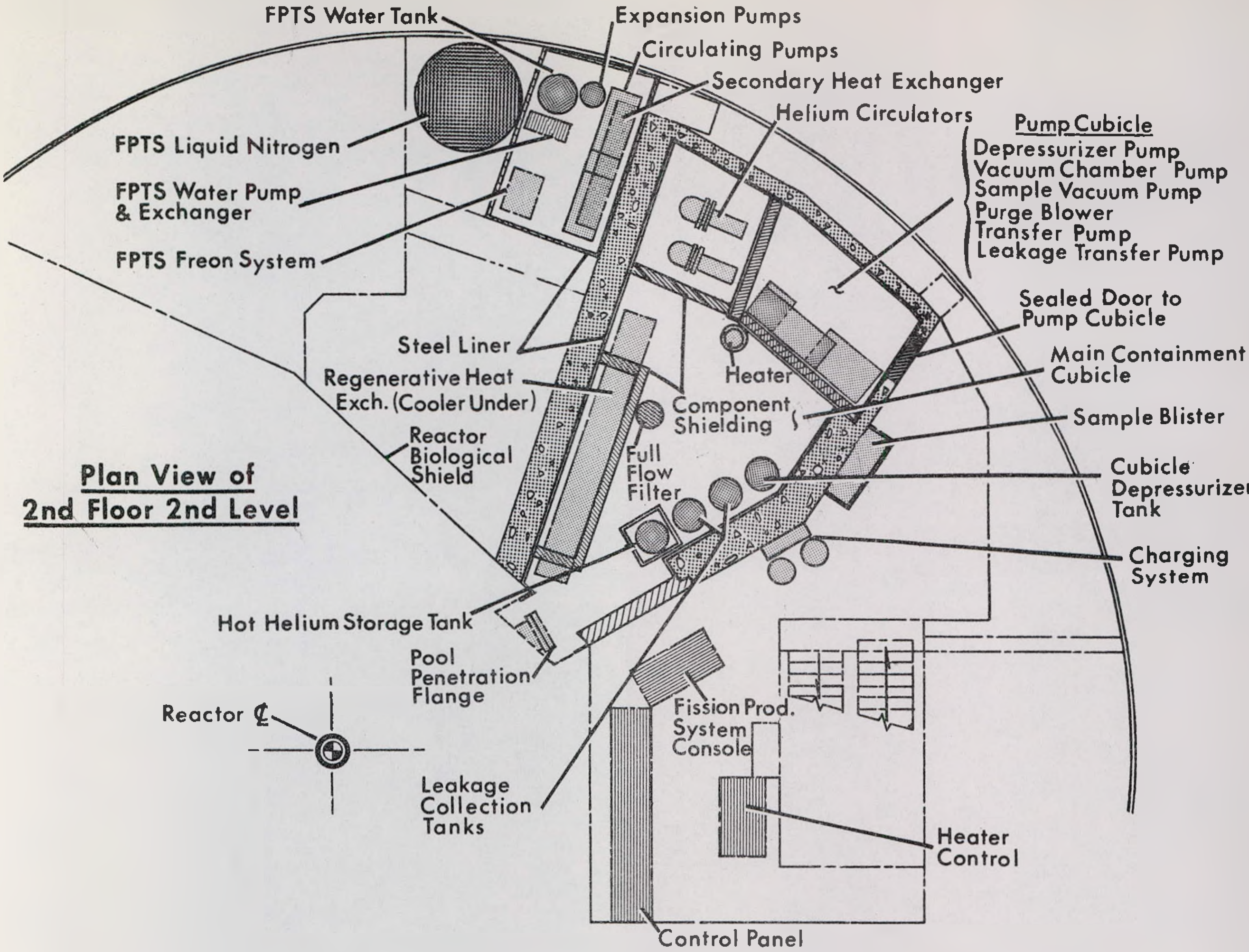
Figure 5.9.5 SECTION THRU TYPICAL TEST ELEMENT

The disconnects for the five lines connecting the facility tube to the remainder of the system are Marman type flanges utilizing double seals. The space between the seals on each flange is evacuated. Leakage, if any, is directed to the leakage collection system.

Out-of-Pool Equipment: The containment cubicles, gas circulating system, fission product trapping and purification system and supporting equipment are located on the second floor of the GETR reactor building. The equipment location and cubicle design are shown in Figure 5.9.6. Details of the out-of-pool equipment are discussed below.

Main Cubicle: The main cubicle is positioned on the second floor as shown on Figure 5.9.6. The cubicle is completely enclosed in concrete and is lined inside with steel plate which forms secondary containment. The cubicle is divided into two levels with the fission product trapping system equipment located on the first level and the primary loop and associated equipment located on the second level. The second level is divided into two compartments, the larger of which is open to the lower level and contains the major loop components, such as the circulators, heat exchanger, cooler, full flow filter, hot helium storage tank, leakage collection hold up tanks, depressurizer tank, cubicle cooling system, ventilation blower and main loop piping, valves and instrumentation. The other compartment on the second level is the pump room which contains the main transfer pump, leakage transfer pump, vacuum chamber pump, purge containment blower, fission product trapping system vacuum pump, and depressurizer compressor. The pump room is provided to permit access to the pumps for periodic inspection and maintenance. It is shielded from the components in the other compartment and is also surrounded by shielding. Access is through a door at the operating mezzanine level.

Access to the lower level is through an air lock. Access to the upper level, exclusive of the pump room, is by ladder from the lower level. Local shielding is provided around the circulators, cooler, full flow filter, hot helium storage tanks, and leakage hold up tanks to minimize radiation to personnel entering the cubicle for inspection and maintenance during accessible periods.



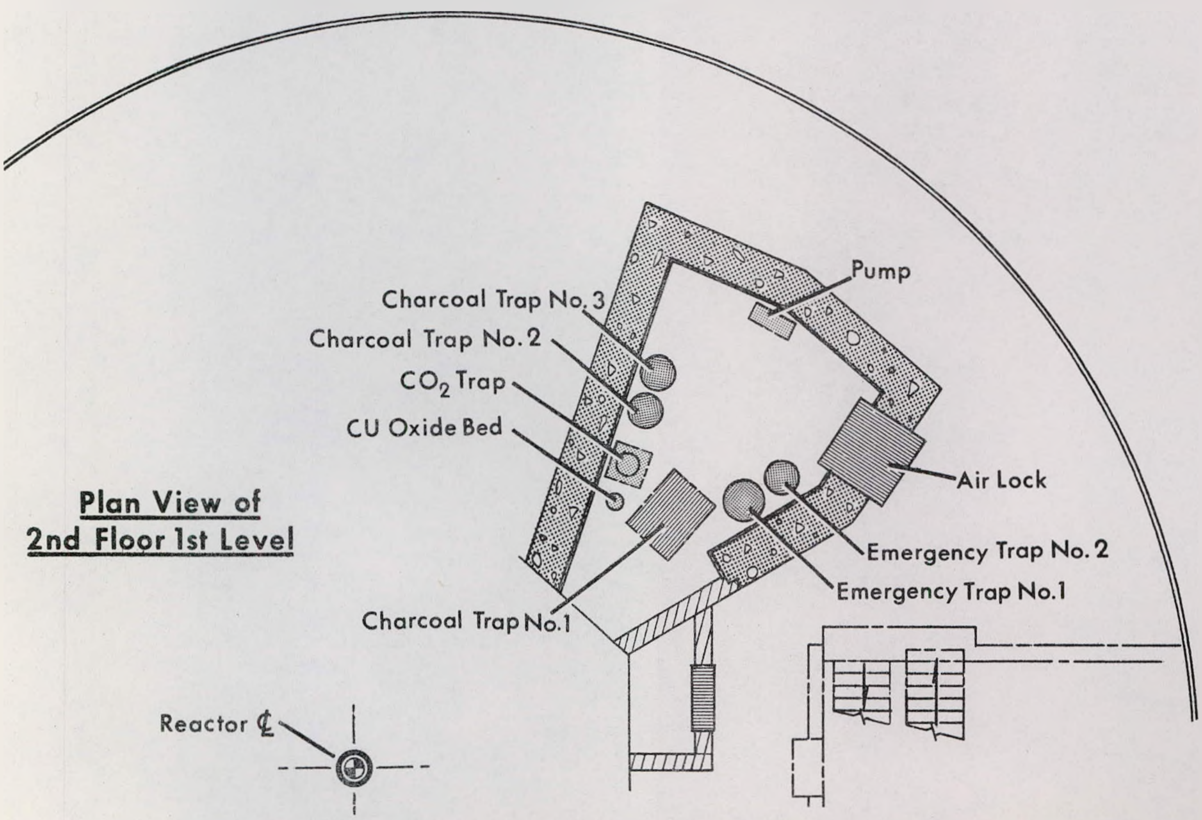


Figure 5.9.6 OUT-OF-POOL EQUIPMENT

The cubicle normally operates under a slight negative pressure to minimize out-leakage.

Main Cooler: The primary loop main cooler is a single wall gas-to-water type (approximately 7 ft. by 1-1/2 by 6 ft.) located along the north wall in the upper level of the cubicle. The purpose of the main cooler is to transfer heat from the hot helium gas to a closed demineralized secondary water system which, in turn, transfers heat to the cooling tower water through a secondary heat exchanger. The cooler is designed to operate at 1400°F maximum helium inlet temperature without local boiling on the water side. The maximum capacity of the heat exchanger is 328,000 Btu/hr.

Hot Helium Storage Tank: A storage tank is provided which will be used to contain the loop contents during maintenance and in case of excessive loop leakage to minimize loss of helium to the cubicle. A schematic of the system is shown in Figure 5.9.4 and Figure 5.9.6 shows the location of major system components. The tank, with a capacity of 10 cubic feet, is designed, fabricated, and tested in accordance with the ASME Boiler and Pressure Vessel Code for 600 psig at 650°F. A pump is used to transfer helium to and from the tank. Since the main loop volume is about 10 cubic feet, the tank pressure resulting from transferring helium at operating conditions without using the transfer pump is about 175 psig. The transfer pump is sized to reduce the main loop to atmospheric pressure with the helium at operating conditions which will result in a maximum tank pressure of about 400 psig.

Full Flow Filter: A full flow filter is located on a by-pass line upstream of the circulators. It has been used to remove fragments from the main loop stream to protect the circulator bearings. The unit is located on the second level of the main cubicle. It is approximately 15 inches in diameter and 2 feet long, including shielding.

Circulators: Two circulators are normally installed to pump the helium coolant around the closed loop although the loop may be operated with only one circulator installed. One circulator provides the necessary flow to cool the fuel element. Two types of helium circulators have been used; the gas-bearing type and the ball bearing type. Each type is cooled by water flow through the circulator housing. The housings are leak-tested such that gas leakage is not detectable with a helium mass spectrometer.

Design performance conditions are:

	Gas Bearing	Ball Bearing
1. Inlet Temperature (max)	650°F	600°F
2. Operating Pressure (max)	700 psig	400 psig
3. Circulator differential pressure	10 psi	10 psi
4. Mass flow	540 lbs/hr	250 lbs/hr

Vacuum and Leakage Collection System: A system is provided to collect and contain leakage from points in the system most likely to leak radioactive gas. Figure 5.9.7 shows a schematic of the vacuum and leakage control system. The main evacuation pump serves several purposes. During normal operation this pump is used to maintain a high vacuum on the vacuum chamber to minimize heat losses to the reactor pool water. The same pump also maintains a vacuum on the space between the seals of the facility tube disconnect flanges. Leakage, if any, into the vacuum chamber is directed through a moisture detector, cold trap and vacuum pump to the low pressure hold-up tank. Leakage from the disconnect interseals is also directed to the tank through the same pump. A slight negative pressure is maintained on the purge line containment shell by the purge containment vacuum pump. The sample lines to the junior cave and sample blister are also evacuated by separate pumps to purge the lines of gas which may be radioactive. The gas from the interseals, and purge line containment discharges through a filter into the 10 cubic foot pressure hold-up tank. If and when the pressure in this tank reaches 15 psia or over, a leakage transfer pump transfers the gas to the five cubic foot high pressure hold-up tank. The gas is stored here until it becomes necessary to relieve the excess gas. This gas is exhausted through an iodine trap and filter ~~to the~~ main cubicle. Radiation detectors are included on the system to monitor activity in the various lines. The main loop may also be evacuated by the main evacuation pump if required during shutdown.

Emergency Helium Cooling System: The emergency helium cooling system

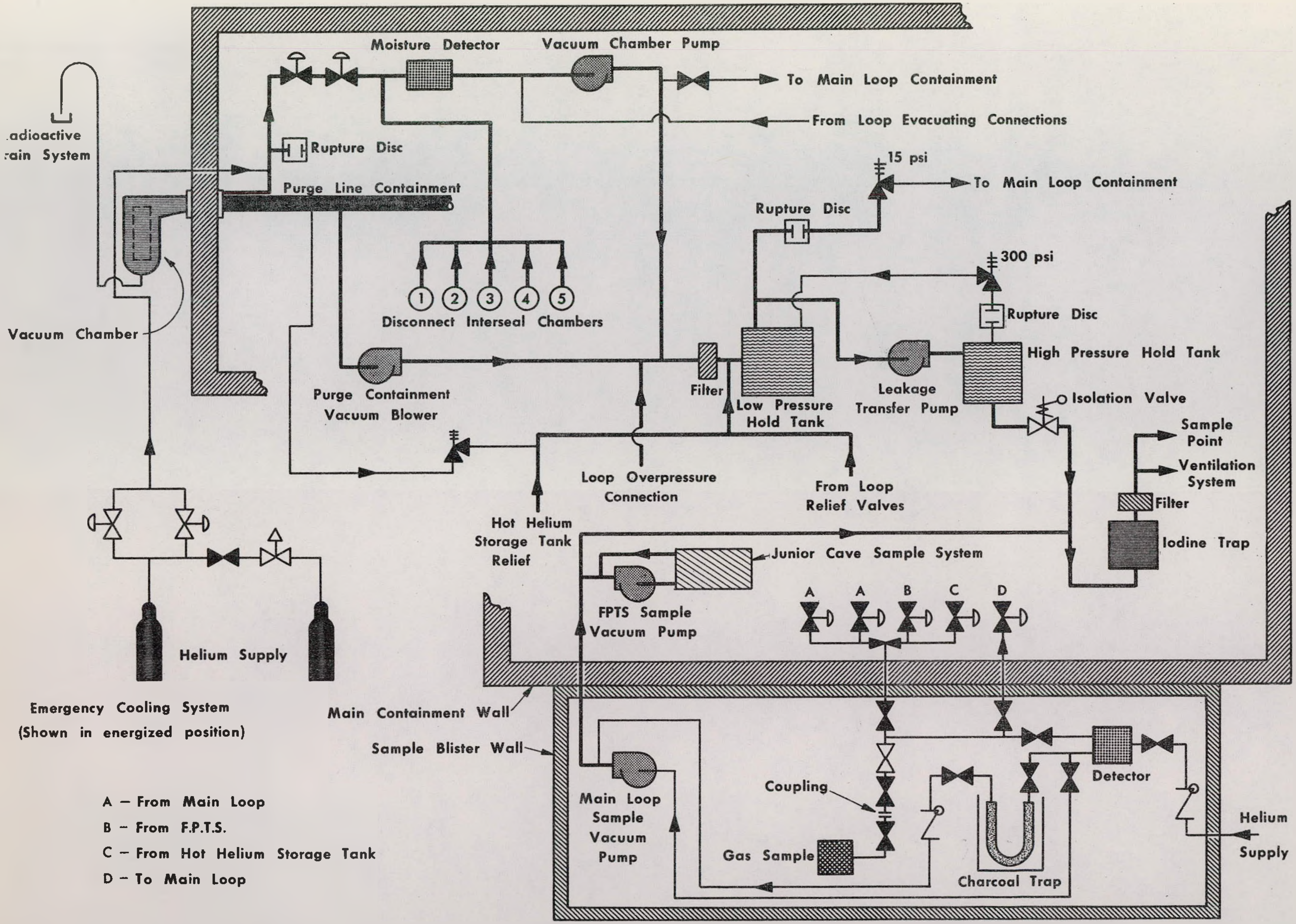


Figure 5.9.7 PRINCIPAL COMPONENTS OF THE VACUUM AND LEAKAGE COLLECTION SYSTEM AND EMERGENCY COOLING SYSTEM

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is shown on Figure 5.9.7. It provides a means of admitting helium to the vacuum annulus of the vacuum chamber to increase heat loss from the fuel element to the pool. The system is put into operation in case of low loop flow and/or low loop pressure. A cylindrical tank in the system is kept charged with sufficient helium to feed the annulus upon demand. Double valves are used to increase reliability. A time delay of up to one minute may be added to this system if there are two operating circulators in the loop.

Electric Heater: A 50 kw capacity electric heater is located downstream of the circulators (see Figure 5.9.4) to control the return gas temperature. The heat output is automatically regulated by a saturable core reactor controlled by the helium temperature downstream of the heat exchanger on the main loop coolant return line to the fuel element. The control equipment for the heater will be located outside the cubicle on the control mezzanine.

Heat Exchanger: The fuel element inlet temperature conditions require for the experiment are higher than the maximum permissible operating temperature of the gas circulator. In order to minimize the loop heat losses, a regenerative heat exchanger is located in the loop as shown in Figure 5.9.4. The heat exchanger is designed, fabricated and tested in accordance with ASME Boiler and Pressure Vessel Code for the maximum loop temperature and pressure conditions. The heat exchanger is sized to exchange 235,000 Btu/hr. A full flow bypass around this heat exchanger is provided.

Cubicle Ventilation System: Located within the cubicle is a ventilation system designed to maintain a negative pressure on the main cubicle, junior cave, sample blister, pump room, and auxiliary equipment containment area, and to isolate the containment system in the event of high activity release from the loop. The ventilation system includes a mechanical blower located in the main cubicle, two high efficiency air filters, an iodine trap, radiation monitors, and associated piping as shown in Figure 5.9.8. The mechanical blower takes suction from the main cubicle through a high efficiency filter and discharges to the stack through an iodine trap and the second high efficiency filter. Air flow into the main cubicle is from the GETR containment through the auxiliary containment, junior cave, pump room, and sample blister. Valves, located on all inlets, will close on a signal of high radiation or high pressure in the main cubicle. The valves on the interconnecting lines between the various units being ventilated will open in case the pressure in the units becomes higher than the main cubicle pressure.

Auxiliary Equipment Containment: Some auxiliary systems which do not contain radioactive materials are located in an unshielded steel lined cubicle located on a mezzanine outside of the main cubicle as shown in Figure 5.9.6. Systems located here are those which serve the main loop but are separated from the radioactive main loop coolant by the primary containment. For example, the equipment located here includes the main loop cooling demineralized water system, the fission product trap water coolant system, and the fission product trap freon cooling system. In the unlikely case of leakage of radioactive gas to these systems, this auxiliary containment provides extra precaution against leakage of radioactive gas to the GETR containment building.

Fission Product Trapping and Purification System: The fission product trapping and purification system (FPTS) is provided to remove radioactive fission products and gaseous impurities from the fuel element purge stream and to return the essentially clean helium to the main coolant stream. Figure 5.9.4 shows a schematic of the system. It consists of a series of adsorbent beds and filters, operated at various temperatures, through which the purge stream is passed. The equipment is located on the lower level of the main cubicle and is arranged approximately as shown in Figure 5.9.6.

The main components are:

a. Charcoal Trap No. 1 is water-cooled and contained in a lead cask about nine inches thick. The trap is designed to remove the majority of fission products other than noble gases.

b. Charcoal Trap No. 2, which is included to reduce the concentration of noble gases, is freon-cooled to -40°F and shielded with six inches of lead.

c. A copper-oxide bed with electric heater and lead brick shielding is included to convert carbon monoxide in the coolant to CO_2 and hydrogen to H_2O.

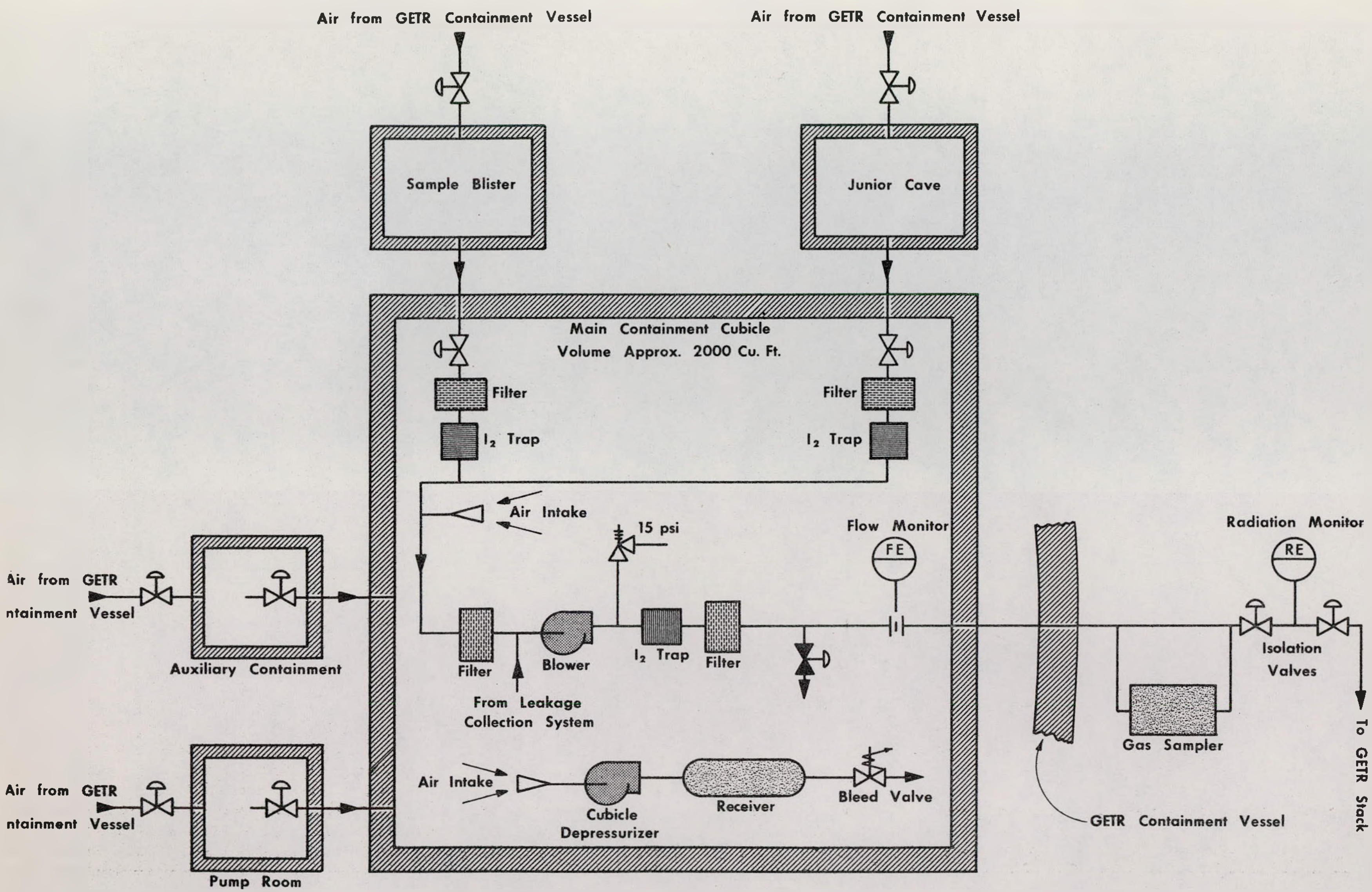

Figure 5.9.8 PRINCIPAL COMPONENTS OF THE VENTILATION SYSTEM

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d. A CO_2-H_2O Trap removes CO_2 and H_2O formed in the previous trap. It is shielded with lead bricks.

e. Charcoal Trap No. 3 is cooled by liquid nitrogen to -320°F and shielded with six inches of lead. The trap is designed to remove the majority of the remaining noble gases not trapped by Charcoal Trap No. 2.

f. Emergency Trap No. 1, which is used as a stand-by trap for Charcoal Trap No. 1 is water-cooled and shielded with nine inches of lead.

g. Emergency Trap No. 2 is liquid nitrogen cooled and shielded by six inches of lead. It will be used as a stand-by trap for Charcoal Traps No. 2 and 3.

h. A Booster Pump, which can be used to adjust the flow rate through the purge lines to the desired value, is located downstream from the traps. Circulation of purge gas is accomplished by taking advantage of the pressure drop across the main loop. The booster pump is used only when the pressure differential is not sufficiently great to obtain the desired flow rate.

The purge lines between the pool penetration flange and the Charcoal Trap No. 1 and Emergency Trap No. 1 are contained in a secondary containment shell to minimize the possibility of excessive leakage to the main cubicle. The purge lines are also connected to the junior cave trapping and sampling systems. The water and freon cooling systems for the FPTS are located in the auxiliary containment cubicle. The cooling water system for the FPTS consists of a water tank, cooler, circulating pump, piping, valves and instrumentation. Demineralized water is used to transfer heat from the traps to cooling tower water by means of the cooler. The system serves to cool Charcoal Trap No. 1. The freon system cools Charcoal Trap No. 2 to about -40°F. The liquid nitrogen system is used to cool the CO_2-H_2O trap, Charcoal Trap No. 3, and Emergency Charcoal Trap No. 2. The temperature of the liquid nitrogen cooled traps is reduced to approximately -320°F. The liquid nitrogen tank is approximately 9 feet high by 5 feet in diameter and is located outside of the cubicles to reduce the probability of overpressurization of a cubicle in case of a nitrogen leak. Isolation valves and radiation detectors are provided to isolate the internal system in case of leakage of radioactive products into the nitrogen system. The control panel for the fission product trapping system is located adjacent to the control panel for the main system on the operating mezzanine. Substitutions and changes in the FPTS are made as part of the experimental program.

Demineralized Water System: The demineralized water system for the main loop provides cooling for the main cooler and the junior cave. The system consists essentially of an expansion tank, two circulating pumps, secondary heat exchanger, stainless steel piping and valves, and instrumentation. Heat is transferred to the cooling tower water system by the secondary heat exchanger. One circulating pump is a stand-by unit. One pump receives power from the normal power supply, the other from the emergency power supply. Flow rate is manually adjusted. Flow rate and temperature are monitored and alarms are provided.

Junior Cave: The Junior Cave is located outside the main cubicle on the second floor adjacent to the reactor biological shielding. Its over-all size is approximately 6 feet long by 6 feet high by 2 feet, 9 inches deep.

Within, means are provided for sampling the purge gas stream both as it leaves the fuel assembly and after it has passed through various traps within the FPTS. The cave is a shielded containment vessel constructed with the same leak tightness requirements as the main cubicle containment structure. Penetrations are provided on the cubicle face of the cave for admission of sample connections directly from the purge lines and connection to sample points at the several locations in the Fission Product Trapping System. Penetrations are provided in the opposite face of the cave for utilities and service lines. These include: electrical power and instrumentation signal lines, water cooling and liquid nitrogen cooling lines, and a helium supply line. Vacuum and ventilation to the cave are also provided by connections from the main cubicle. The cave is designed for a total fission product source activity of 1000 curies. To reduce the radiation field to within working limits, the structure is shielded with one inch of steel and 9 inches of lead on the top, left and front faces. The remaining faces of the cave are shielded by the reactor biological shielding and the main cubicle shielding. All of the required operations within the cave are performed by a claw-type manipulator. The control mechanism for the equipment is located on the outside front face of the cave and is easily accessible to one operator. The manipulator is designed to operate valves for control of gas flow and to handle and remove the test traps. Observation into the cave is provided for by a high-density lead-glass window of shielding effectiveness equivalent to the cave structure. The window is of sufficient size to give adequate view to the entire working volume of the cave. After the sampling, the test traps and samples are removed from the cave structure. The traps are removed in a cask through an air lock located in the bottom of the cave. Samples are collected in a semi-evacuated sample vial located in an unshielded sample blister exterior to the cave structure. Instrumentation within the cave will include provisions for measurement of local gamma activity in the purge line, and gross activity within the cave area. This latter instrumentation establishes the cut-off point for sampling operations because of a high radiation field.

Sampling System: The sampling system is shown schematically on Figure 5.9.4. Samples are handled in a sample blister located adjacent to the main cubicle shield wall on the operating mezzanine (Figure 5.9.6). The blister contains stop valves on the piping leading to and from the system sample points, a gas chromatograph detector, a liquid-nitrogen-cooled charcoal trap, a point for connecting a sample container and miscellaneous piping and valves.

The blister is a steel enclosure normally kept under a slight negative pressure to insure in-leakage. A sealed view-window and glove ports are provided to permit handling of equipment within the blister. A pass chamber is also provided which can be flushed to minimize the probability of internal air escaping to the GETR containment when samples or equipment are passed in and out of the blister. The sampling equipment is connected to the helium loop at several sample take-off points and one sample return point. Air-operated on-off take-off valves are provided within the main cubicle. Lines are manifolded to limit the number of lines penetrating the main cubicle wall to the blister. Stop valves on the lines within the blister are used to control flow. Samples can be passed through the gas chromatograph for direct reading of impurity content, or samples can be taken by connecting the sample container at a connection provided for that purpose. While loop gas is recirculating through the sample equipment to purge the sample system prior to taking a sample, the sample system is under loop pressure. After isolation of the system by closing the valves, the gas pressure is reduced to atmospheric by bleeding excess gas through the nitrogen-cooled cold trap to remove the fission products. Samples are taken with the system at atmospheric pressure. The sample container is normally under vacuum before filling and, therefore, draws in a known volume of gas from the sample system. The sample system is connected to a clean helium supply and a vacuum pump which discharges to the leakage collection system. These auxiliary systems are used to purge the sampling system of undesirable gases.

Helium-Charging System: The charging system, which is located outside the main cubicle is designed for manual operation and supplies high-purity helium for filling and purging of the main loop and subsequent main loop make-up requirements. Through suitable valving, including a back flow check valve, high purity helium from this charging system can be supplied when required to the junior cave, sample blister, interseal disconnect chambers, and Charcoal Trap No. 1 of the fission product trapping system. The helium is supplied from one cylinder. Two pressure regulators are used. The first pressure regulator, located upstream of the purifier, maintains a minimum back pressure of 350 psig on the helium purifier. A second pressure regulator downstream of the purifier provides pressure control of 0-400 psi to allow loop filling and make-up at various pressures. Each regulator is equipped with pressure indicators to indicate upstream and downstream pressure. A helium supply line is equipped with three valves to allow venting of the air during helium bottle replacement to be vented, assuring an air-free helium supply to the purifier. A pressure switch and low pressure alarm indicates abnormal pressure conditions in the bottle. The helium purifier consists of a 10 micron filter, a molecular sieve cartridge for water vapor removal and an oil vapor cartridge. This unit is designed to limit the impurities in the helium to the following values:

Water Vapor ------------	less than 10 ppm/volume
Hydrocarbons------------	less than 10 ppm/weight
Solid particle size ----	less than 6 microns

Depressurizer System: A system is provided to reduce cubicle pressure in case of a major line rupture, which may raise the main cubicle pressure as high as 1.75 psig. This system consists of a compressor, high pressure receiver (300 psi) and the necessary piping, valves, and instrumentation. On a signal of cubicle pressure above atmospheric (about one inch of water), the compressor will take suction from the cubicle atmosphere and pump air into the receiver until the cubicle pressure drops to essentially atmospheric pressure. The receiver is sized to accommodate a volume of air equivalent to the main loop contents at standard conditions of temperature and pressure. This system is capable of reducing cubicle pressure from 1.80 psig to zero psig in about six minutes.

Control and Instrumentation: All instrumentation needed for the loop is centralized at the local control panel in the experiment area. Critical parameters are repeated and recorded in the reactor control room. An annunciator on the control room panel indicates loop troubles. Additional annunciators on the local panel indicate the source of the trouble. Critical parameters have two levels of action: alarm and scram (or rundown) of the reactor. Tables 5.9.1, 5.9.2, and 5.9.3 list the various parameters being measured, indicate the location of the recorder, and the levels of alarm, scram or rundown settings. The loop is fully instrumented to measure required fuel and coolant temperatures, flow rates, and pressures. All points of interest throughout the system will furnish data to recorders, thus providing information for post-operational analyses. Suitable instrumentation is provided for reactor shutdown should a potentially serious condition be indicated. Reactor scram will be initiated by high exit gas temperature, high circulator intake gas temperature, low flow or pressure of main loop gas. In addition, a reactor rundown will be initiated by activity monitors located on the main loop and purge lines. Intermediate levels of these parameters and other warning signals will be annunciated. The general system is laid out in such a way that the system and the experiment instruments are under the direct control of the loop system operator. Should a transient occur, the operator will be able to make suitable system adjustments to bring about a system equilibrium condition. Fast-acting transients involving temperature, pressure or flow excursions will cause an automatic reactor scram by the experiment when these parameters have exceeded pre-set limits. A list of loop and fission product trapping system controls is given in Table 5.9.2. Primary heat exchanger water flow will be metered and low flow will cause an alarm. High discharge water temperature will also operate an annunciator. A separate continuous monitoring system as described in Section 5.9.7 is provided to determine cubicle exhaust gas activity before release to the GETR stack.

Given in Table 5.9.3 is a list of controls and indicators located at the reactor console. The controls and instruments allow the reactor operator, in case of an emergency, to introduce emergency cooling which will increase the heat loss from the experiments, thereby reducing the gas and fuel assembly temperature. If the loop fails to respond, the gas coolant may be transferred to the hot helium storage tank. This action will reduce the loop pressure and automatically cause a reactor scram.

Shielding: The shielding design assumptions are that a large percentage of fission products escape from the system to the main cubicle and that large quantities of fission products are plated out in various sections of the loop. Operation has shown that the majority of fission products are retained by the fuel element. Details of the assumptions and results of the analysis are given below. The in-pool shielding was based on the maintenance requirements of the reactor and the requirements for direct work on the facility. In both cases, the pool water level was assumed to be about 6 inches below the top of the reactor pressure vessel.

The assumptions used in the shielding calculation were:

a. 0.05 curie/cm^3 activity in the main loop

b. 50 curie/cm^3 activity in the purge stream

c. 0.2% of the potential loop plate out occurs in the region of the disconnect joints ($\sim$ 190 curies).

d. Direct beaming of radiation from the fuel up the evacuated secondary containment.

e. Maximum contribution from all loop components to a man on the vessel head is 100 mr/hr.

TABLE 5.9.2

LOOP CONTROLS

Main Panel Indicators

(a) Test section temperatures (fuel)
(b) Test section outlet gas temperature
(c) Primary loop gas temperatures
(d) Heat exchanger (cooler) water temperature
(e) Facility tube differential pressure
(f) Facility tube gas flow
(g) Circulator gas flow
(h) Primary loop pressure
(i) Hot helium storage tank pressure
(j) Vacuum annulus pressure and moisture content
(k) Leakage collection system pressures
(l) Primary loop, FPTS, and leakage collection system gas activity

Main Panel Controls

(a) Primary loop heater power (automatic)
(b) Primary loop valves (manual)
(c) Auxiliary valves (manual)
(d) Blowers and pumps (manual)
(e) Emergency cooling valves (automatic)

FPTS Panel Indicators

(a) System gas temperatures
(b) Charcoal trap cooling water temperatures
(c) System gas flow
(d) Junior cave gas sample flow
(e) System gas activity

FPTS Panel Controls

(a) Copper-oxide bed heater power (automatic)

TABLE 5.9.2 (Cont'd)

Activity Monitors: FPTS PANEL	Activity Monitors: MAIN LOOP PANEL	Application
	IR	Gas outlet from facility tube (γ)
	IR	Heat exchanger (γ)
	IR	Cooler (γ)
	IR	Circulator No. 1 (γ)
	IR	Circulator No. 2 (γ)
	IR	Electric heater (γ)
	IR	Purge line containment exhaust (γ)
	IR	Junior cave sample exhaust & bleed (γ)
I	R	FPTS purge line (γ)
I	R	FPTS downstream of Trap No. 1 (γ)
I	R	FPTS downstream of booster pump (γ)
I	R	Auxiliary cubicle room monitor (β γ)
I	R	Cubicle room monitor (β γ)
	IR	Sample blister sample exhaust & bleed (γ)
	IR	Holdup tank discharge to cubicle (γ)
I	R	Vacuum chamber pump discharge (β γ)
I	R	Junior cave working area (β γ)
I	R	Junior cave process (γ)
I	R	Sample blister area monitor (β γ)
	I	Liquid nitrogen system (γ)
	I	Facility tube disconnect flange (γ)

I = Indicate

R = Record

TABLE 5.9.3

LOOP CONTROLS AND INDICATORS AT THE REACTOR CONSOLE

Indicators

(a) Test section outlet gas temperature

(b) Circulator inlet gas temperature

(c) Facility tube differential pressure

(d) Facility tube gas flow

(e) Circulator gas flow

(f) Primary loop pressure

(g) Test section outlet gas activity

Controls

(a) Reactor scram and rundown bypass (for use with planned shutdown only)

(b) Initiate emergency cooling

(c) Initiate gas transfer to hot helium storage tank

(1) Open pump discharge valve and pump bypass valve
(2) Close pump bypass valve and start pump

(d) Manual control of cubicle isolation valve (Push button override to be used only during GETR building isolation)

Activity Monitors

Reactor Panel	Application
IR	Gas outlet from facility tube (γ)
IR	FPTS Purge line (γ)
R*	Cubicle discharge to stack (β γ)

R* = Record on reactor plant instr.

The shielding required was calculated to be about 5 inches of lead in certain locations. A shielded radiation detector may be located above the disconnect flanges and calibrated to indicate the dose level at shutdown.

The cubicle shielding thickness requirements were based on a loop power of 100 kw operating for about three years. The resulting equilibrium fission product activity for potential helium contaminants having half-lives greater than 10 minutes is about 1 x 10^5 curies. The following is a list of potential contaminants:

a. Xe, Kr
b. I, Br
c. Se, Sr, Y, Sn, Sb, Te, Cs, Be, La, Sm

Other fission products are either non-diffusion or reactive with graphite, thus never leaving the fuel compacts. The upper level of the main cubicle is shielded with 1-1/2 feet of ferrophosphorous concrete. The concrete density is about five grams/cc. The resulting dose rate at the surface would be about 200 mr/hr. The distribution of contaminants within the cubicle is assumed to be about 75% of the equilibrium noble gases, 10% of the halides, and 10% of the others. For shielding calculations of the lower level of the main cubicle, it is assumed that 100% of all equilibrium fission products with half-lives greater than 10 minutes are distributed uniformly in the cubicle. The dose rate through 1-1/2 feet of ferrophosphorous concrete would be about 300 mr/hr. Under normal operation the fission products will be retained within the main loop piping, in the fission product trapping system piping, and in the individually shielded FPTS traps. The dose rate calculations for normal operation are based on reactor rundown activity levels of 0.05 curie/cc in the main loop lines and 50 curie/cc in the fission product trapping system lines.

Each trap of the fission product trapping system is individually shielded. The maximum estimated radioactivity of the trapped fission products and shield thickness is given below.

Trap	Gamma Activity in curies at 100 kw	Inches of Lead Shielding
Charcoal 1	60,000 mixed fission products	9
Emergency 1	60,000 mixed fission products	9
Charcoal 2	33,000 Xe and Kr	6
Charcoal 3	33,000 Xe and Kr	6
Emergency 2	33,000 Xe and Kr	6

From the above assumptions the calculated activity level is sufficiently low to allow occupation by operating personnel without exceeding dose limits given in part 20 of the Code of Federal Regulations.

Shielding thicknesses in the junior cave (~ 1" of steel and 9" of lead) are designed to safely handle a 1000-curie sample. If samples of greater activity are taken internal shielding will be added.

5.9.3 Physics

A fuel loading of 195 grams 93% U^{235} was used for the first experiment in the loop. This loading resulted in a fission power generation rate of about 65 kw (thermal) in the experiment. The approximate average thermal neutron flux in the experiment is about 2.0×10^{13} nv. This value was measured by the central wire of a nuclear mock-up for a 50-gram loading. The vertical (axial) neutron flux distribution is based on a measurement from the nuclear mock-up. The measurement was taken at the start of a GETR fuel cycle (no xenon present) with all control rods inserted 17 inches into the top portion of the 36-inch height core. If the power level generated by the experiment is higher than desired, it may be adjusted downward by moving the experiment laterally outward from the pressure vessel.

To reduce the power generation in the experiment with the GETR operating at rated power, provision has been made to move the fuel assembly five inches laterally outward from the pressure vessel. In this "shutdown" position, the experiment can be shielded from neutrons and gammas by the use of a shutter. The mechanical arrangement of the shutter is shown in Figure 5.9.3. With the reactor operating at 60 Mw thermal power, the fission power in the experiment adjusted to 65 kw, the predicted thermal power generation rates in the experiment for the operating and shutdown positions are given in the following table:

	Operating	Retracted
Power (thermal neutron fission)	60 kw	3
Power (epithermal neutron fission)	5 kw	1
Gamma Heating (core gammas)	11 kw	1
Decay Heat	Included w/fission	4*
Total	76 kw	9 kw

* Decay heat following 65 kw fission power generation, for irradiation times in excess of 50 hours. One hour following a reduction in fission power level from 65 kw to 5 kw or less, this value falls off to about 0.5 kw.

The experiment in the shutdown position is expected to generate 12% of the operating power after a long irradiation time. The effect of moving the experiment from its normal operating location to the shutdown position is expected to decrease the core reactivity by less than 0.17% $\Delta k/k$. The effect of inadvertant flooding of the gas and insulating chamber of the experiment with water is expected to affect the core reactivity much less than 0.1% $\Delta k/k$.

5.9.4 Heat Transfer

The design of most fuel assembly tubes incorporates re-entrant type construction wherein the coolant passes downward through an outer annular section of the pressure vessel and returns upward through the center to provide coolant to the fuel element. (See Figure 5.9.1.) Assuming the total power generated in the test section is 76 kw and the weight flow in the test region is 160 lbs/hr, the maximum fuel temperature occurs approximately 12 inches from the lower end of the fuel. A radial temperature distribution at that point is given in Figure 5.9.9. The maximum temperature effect of the flux variation across the fuel is shown in the figure and represents a difference of about 200°F. The maximum baffle temperature occurs near the hot (exit) end of the fuel. The inner nichrome V baffle reaches a temperature of about 1450°F, the center baffle maximum temperature is about 1200°F, and the outer baffle maximum temperature is

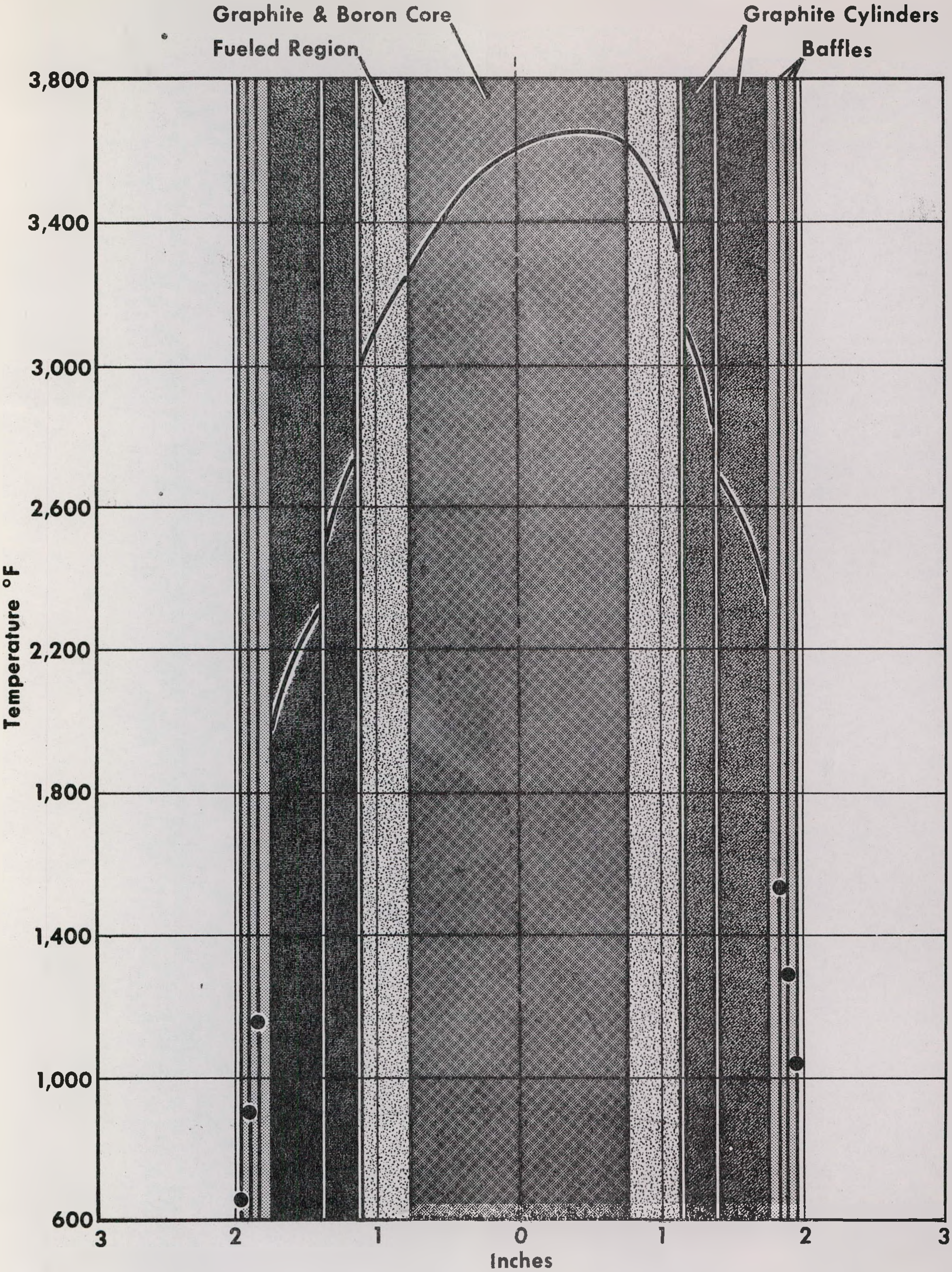

Figure 5.9.9 RADIAL TEMPERATURE DISTRIBUTION

about 900°F. Under conditions where the normal heat removal capacity of the primary cooling system is reduced or lost completely, the heat generated in the test fuel element is reduced in an emergency by a reactor scram initiated from a loss of flow or a loss of pressure.

When the loop is shut down, i.e., no coolant flow, and the reactor is still operating, the test section is shielded with the lead-cadmium shutter shown in Figure 5.9.3. The heat still being generated in the test section in the shutdown case by either fission product decay or neutron and gamma leakage through the shield is removed by heat loss to the reactor pool water. Heat transfer is by radiation and by conduction through the walls of the test section. To increase the heat loss, the vacuum in the facility tube that normally insulates the pressure vessel is broken by the injection of helium from an accumulator. This constitutes the emergency cooling system. Parallel sets of valves between the accumulator and vacuum vessel are provided to minimize the effect of valve failure.

Helium injection into the vacuum vessel is provided under the following conditions:

a. Loss of flow (automatic)
b. Loss of pressure (automatic)
c. Excessive gas temperatures (manual)
d. Loop shutdown with reactor in operation (manual)

The first three cases are accompanied by reactor scram. The last case requires a reactor shutdown to move the test section away from the reactor and to place the shutter in position. Calculations show that the heat generation is highest for the last case; therefore, if the temperatures are acceptable for this case, it can be concluded that overheating will not occur for the other cases. The following analysis is for this case.

The assumptions used in analyzing the temperature distribution in the test section for the case where the loop is shut down, the facility tube retracted, the shield in place, and the reactor in operation, were:

a. The flow of heat is radial only.
b. Heat transfer is by radiation and conduction, i.e., no heat transfer by convection.

c. The temperature drops occur between the metallic shells; the temperature drop in the metal is negligible.

d. The heat generation is independent of azimuthal orientation in each plane normal to the axis of the test section.

e. The peaking factor is 2.1 which is the value for a fresh core.

f. The test section is displaced radially a distance of 4 inches away from its normal operating position.

g. The gamma and fission heating is reduced by a lead and cadmium shield.

Physics calculations show that the fission heat generated in the test fuel with the reactor at full power is reduced to approximately 12 percent by the combination of displacing the test section and interposing the shutter. The gamma radiation heating is likewise attenuated from an average value of 0.8 watts/grams to about 0.06 watts/gram. The total heat generated in the test section is about 5.5 kw.

The radial temperature distribution in the various components of the test section is calculated to be as follows:

Fuel maximum temperature	2110°F
Inner baffle (nichrome V)	1915°F
Middle baffle (nichrome V)	1710°F
Outer baffle (321 S.S.)	1480°F
Pressure vessel (321 S.S.)	915°F
Facility tube (aluminum)	230°F

These temperatures are higher than those attained in practice because of the conservative nature of the assumptions. For example, axial conduction will decrease the peak temperatures somewhat, and it is expected that some natural convection will occur.

In the emergency cooling situations where the reactor is scrammed, the temperatures will be lower than calculated above. The high thermal capacity of the fuel element precludes any rapid rise in temperature. It takes approximately 90 seconds after scram for the temperature in the fuel to equilibrate. At this time the residual heating has decayed to approximately 3.5 percent of full power. This is lower than the heating rates considered above.

5.9.5 Operating Procedures

The High Temperature Helium Loop operating procedures are contained in the loop operating instruction book. These procedures are prepared by the reactor organization with assistance from the design group and safeguards personnel as required. These procedures are reviewed by the reactor operating group, safeguards personnel, and they are subject to review by the Vallecitos Laboratory Safeguards Group. This loop has been in operation for over one year using these procedures. Changes and corrections have been made as required. Listed below is the outline for the loop operating procedures.

1. General Description of Program
    - 1.1 Purpose of Test
    - 1.2 Fuel Assembly
    - 1.3 Summary of Purposes of Test

2. Loop Description and Principles of Operation
    - 2.1 General Location
    - 2.2 Main Loop Systems
    - 2.3 Fission Product Trapping Systems
    - 2.4 Auxiliary Systems - Schematic and Description
    - 2.5 Instrument and Electrical

3. Normal Operating Procedures
    - 3.1 Startup
    - 3.2 Test Conditions
    - 3.3 Shut Down

4. Emergency Procedures
    - 4.1 Alarm Conditions - Trouble Tabulation
    - 4.2 Emergency Cooling
    - 4.3 Emergency Shut Down
    - 4.4 Leakage
    - 4.5 High Activity Shutdown

5. Health and Safety
    - 5.1 Loop Operating Standards
    - 5.2 Shutdown and Maintenance

5.9.6 Disposal of Radioactive Materials

Fuel Handling: At the conclusion of an experiment the helium in the main loop is transferred to the hot helium storage tank and the system is purged with clean helium. The facility tube is then removed from the vacuum chamber by lowering the water in the pool, flooding the vacuum chamber to a point just below the bottom of the disconnect box, and disconnecting the facility tube flanges and instrumentation. The open ends of the flanges are sealed, the level of the water in the pool is raised, and the facility tube is removed from the vacuum chamber to the GETR canal. After the facility tube is held in the canal for an appropriate time, depending on the length of irradiation time, the piping attached to the facility tube pressure vessel is filled with a thermosetting plastic and cut under water. The parts are then transferred in a fuel transfer cask from the reactor to those authorized to receive. Fuel transfers from the loop, using this technique, have been made.

Trap Handling: The traps are designed with individual shielding to remain in place during the entire life of an experimental fuel assembly. At the conclusion of the experiment, the trap contents can be sealed within the trap by closing the inlet and outlet lines. The entire assembly, including the shielding, can be transferred to those authorized to receive it. The estimated radioactivity of the trapped fission products and shield thickness is given below.

Trap	Gamma Activity (curies)	Shielding Thickness (inches of Pb)
Charcoal 1	9000 mixed fission products	9
Emergency 1	9000 mixed fission products	9
Charcoal 2	5000 Xe and Kr	6
Charcoal 3	5000 Xe and Kr	6
Emergency 2	5000 Xe and Kr	6

Liquid Waste Disposal: Under normal operating conditions no liquid wastes will be produced by the loop. Provisions are available, however, to transfer water that could become contaminated from leaks in the cooling systems to a retention tank.

Gaseous Waste Disposal: Gaseous wastes for the GETR are discharged to the atmosphere through a stack 95 feet high, equipped with an "absolute" filter and radiation monitor. The maximum permissible release rate for the GETR stack is 6000 μc/sec. All gaseous wastes from the loop are collected in the cubicle as described in Section 5.9.1. The cubicle exhaust system shown in Figure 5.9.8 passes the gas through an absolute filter, an activated charcoal trap with a 99.998% efficiency for iodine removal and a second absolute filter before exhausting to the stack. Any gases released are, therefore, primarily noble gases. A radiation monitoring system located in the cubicle exhaust is set to close a cubicle isolation valve, isolating the cubicle, when the activity released reaches 5000 μc/sec of noble gases. Experience to date has shown that the maximum release of fission products from the filtering system is less than .001 μc/sec or essentially non-measurable. To insure that this limit is not exceeded, however, a sampling system with an absolute filter and iodine trap can be used to sample the air leaving the loop exhaust system.

The cubicle exhaust will bypass the GETR Stack Radiation Monitoring System. A diagram of the arrangement is shown in Figure 5.9.10. A signal for a building isolation from the GETR stack radiation monitoring system will cause both the GETR and cubicle exhaust isolation valves to close.

5.9.7 Accident Evaluation

Introduction: Potential hazards in the operation and maintenance of the gas loop have been examined as well as results of major system component failure. A discussion of the results of this evaluation is given in subsequent paragraphs. It is shown that the reactor can be scrammed and the loop safely shut down and cooled following an operation or system failure. The loop maximum credible accident is also presented.

Containment Design: The entire gas loop facility is located within the GETR containment building which constitutes the final containment. The primary containment is the loop piping. Secondary containment is provided by the aluminum vacuum chamber for the in-pool portion of the loop and by the main cubicle, a shielded steel-lined compartment, for the out-of-pool portion of the loop. In addition, unshielded containment is provided for such auxiliary systems as the demineralized water system, freon cooling system, and leakage collection systems in order to minimize the consequences of failure of main loop components which could release radioactive gas to these auxiliary systems. A multiple failure sequence is required, therefore, to release contaminated helium coolant to the GETR containment vessel. A general arrangement showing location of equipment within the cubicles is presented in Figure 5.9.6. Figure 5.9.3 shows the in-pool portion of the loop.

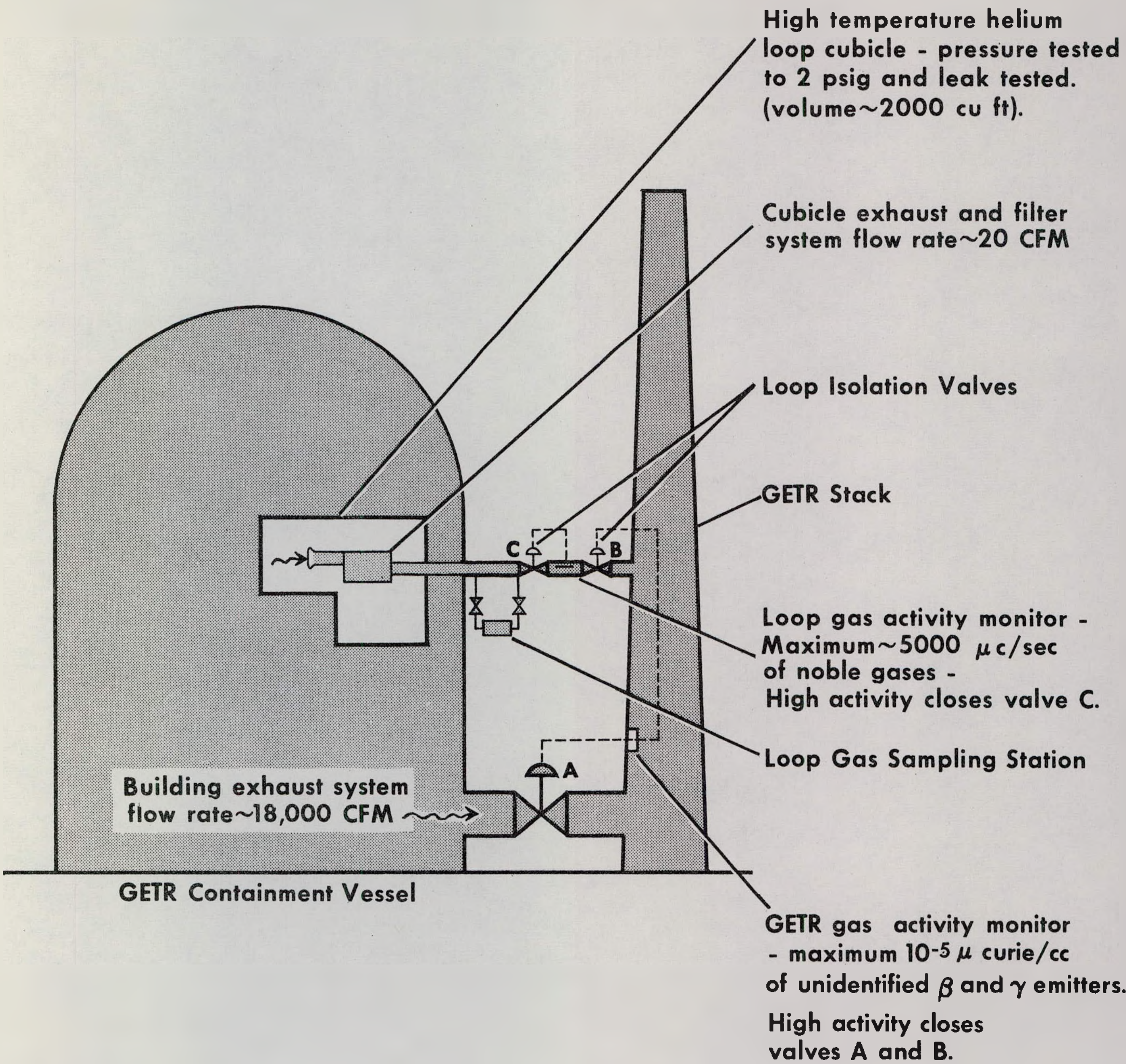

Figure 5.9.10 DIAGRAM OF LOOP AND GETR GAS RELEASE SYSTEM

The main cubicle and the auxiliary containment cubicle are constructed for minimum leakage. The units are pressure-tested at 2 psig and the maximum leakage rate is less than 300% of cubicle volume per day. The units normally operate at a slight negative pressure to ensure in-leakage from the GETR containment building, rather than out-leakage to the building. The air in the two cubicles is exhausted by a common blower system to the stack of the GETR containment vessel.

The primary containment, the loop itself, is designed, fabricated, and inspected in accordance with the ASME Codes and the requirements of the State of California. In addition, the design of the primary containment is based upon the philosophy of essentially all-welded construction with leaktightness verified by mass spectrometer leak testing equipment.

As an additional precaution, the purge lines of the fission product trapping system from the cubicle penetration to the first trap are contained within a secondary containment shell to minimize the possibility of leakage from this potentially highly radioactive source to the main cubicle. Because it is impossible to obtain an absolutely leaktight system, a Vacuum and Leakage Collection System is provided to collect and hold leakage from areas of potentially high activity. Figure 5.9.7, a schematic of this system, shows that a vacuum is maintained on the vacuum chamber, the disconnect flange interseals, and the secondary containment around the purge lines. Relief valve discharge from the main loop purge line containment shell and the hot helium storage tank also is directed into the leakage collection system to minimize main cubicle contamination. The junior cave and sample blister are provided to allow removal of samples from the main loop and the fission product trapping system as shown in the schematic diagram Figure 5.9.4. Each unit is provided with an air lock to permit removal of samples and with interlocks on the valving to prevent inadvertant release of loop contents. During operation, both units are held at a slight negative pressure and both units contain radiation monitors.

<u>Analysis of Accidents</u>: The multiple containment has been designed to limit the consequences of any type of system accident, less than

a maximum credible accident, to energy release or release of radioactive contaminants to the secondary containment, thus minimizing the possible contamination of the GETR containment vessel. During normal operation, all system leaks are collected in the vacuum and leakage collection system or the cubicle and eventually exhausted through the cubicle ventilation system as described in Section 5.9.1. Several system failures which are believed to represent the worst conditions have been selected for analysis and are presented below.

The following conservative assumptions have been made in order to evaluate the consequences of hypothetical loop accidents:

1. Total power generated in the fuel assembly is 113 Kw.
2. The activity in the main loop has reached the shutdown initiating condition of 0.05 curies/cc.
3. The activity in the purge line to the first trap has reached the shutdown initiating condition of 50 curies/cc.
4. The loop gas pressure is 400 psia.

Loss of Coolant: Several modes of failure resulting in a loss of coolant will lead to similar transient temperatures in the fuel element and facility tube pressure vessel. System failures were examined, such as loss of secondary flow to the heat exchangers, failure of the circulator, inadvertent closure of line valves and line rupture. All loss of flow accidents will be subject to the same corrective action.

Although the probability is very slight, for the purposes of evaluating the severity of such an incident, it is assumed that the return line from the in-pool facility tube in the cubicle is instantaneously severed and displaced, resulting in minimum flow resistance to gas escaping from the loop. As a result of this hypothetical failure, the following events will occur:

1. In approximately 0.20 second the contents of the main loop and fission product trapping system are released to the cubicle.
2. A temperature transient is in progress.
3. A reactor scram and emergency cooling to the loop fuel element is initiated by a signal from loss of flow and/or pressure and in five seconds the power in the test section is reduced to less than 10 kw.

4. Circulators are shut down and the circulator valves closed on a signal of very low pressure.
5. Valves in the cubicle exhaust system isolate the cubicle on a signal of high radioactivity.

The maximum rate of increase of temperature of the fuel is ~ 10°F/sec and that of the facility tube pressure vessel and baffles is ~ 3°F/sec, even if it is assumed that loop power continues at 113 kw and no heat is removed. During the five seconds required to initiate emergency cooling following reactor scram, the fuel temperature will have increased approximately 50°F and the facility tube pressure vessel and baffles will have increased approximately 15°F. It should be noted that a time delay as large a 10 seconds will not seriously damage any loop in-pool component.

Once emergency cooling is initiated and the reactor is scrammed, the rate of loss of heat is greater than the rate of heat input. The temperatures thus decrease until the system temperatures equilibrate.

The fuel assembly has, therefore, sufficient heat capacity, with a wide margin of safety, to avoid over-temperatures of loop in-pool components even in the extreme case of complete loss of coolant. Since the rate of absorption of fission products in the FPTS traps is a function of temperature, (increasing with decreasing temperatures), fission products will not desorb as a result of sudden depressurization. The fuel compact is surrounded by a double graphite barrier capable of withstanding the pressure difference resulting from a sudden depressurization, and even if the fuel element should break up from out-gassing during a sudden change in pressure, the resulting particulate matter would be prevented from reaching the cubicle by the multiple graphite barrier. The activity released to the cubicle is limited, therefore, to the activity contained within the piping of the main loop and fission product trapping system. The pressure in the cubicle is estimated to reach a maximum of 1.9 psig. A pump and tank (cubicle depressurizer shown in Figure 5.9.8) located within the cubicle are designed to start automatically on a signal of high cubicle pressure (slightly above atmospheric), restoring cubicle pressure to near atmospheric in about six minutes, thus minimizing release of fission products to the building. The leakage from the cubicle will be held to less than 300% of the cubicle volume per day at maximum pressure which will result in a release of about four curies to the GETR containment building in the first 6 minutes. If it is assumed that the operators evacuate the building in 5 minutes, and 1% of the fission products other than noble gases is iodine dispersed in 1000 cubic feet, the maximum dose received is estimated to be less than one R from external radiation and 2.5 rem from iodine inhalation. All other organs receive less than 2 rem. From the above discussion, it can be seen that even in the extreme case of a complete line severance, the system is automatically shut down and sufficient time is available for the reactor operation personnel to safely evacuate the building. It should also be observed that this major hypothetical accident will result in a negligible increase in the release of activity external to the reactor containment building.

Failure of the Main Cooler: The main cooler is a single wall gas-to-water type described in Section 5.9.1. A closed, demineralized secondary water system transfer heat through a secondary heat exchanger which, in turn, transfers heat to the cooling tower water.

For radioactive contaminants to escape to the cooling tower water system, a failure in the main cooler and a failure of the secondary heat exchanger would be required. Loss of circulation of the secondary water causes the temperature of the gas entering the circulation to increase which, in turn, causes alarms to sound and a reactor scram.

The rate of increase of temperature with time of the piping walls, assuming all the heat produced in the test section is transferred uniformly to the 600 lb. of piping is $\sim$ 1.5°F/sec. The reactor scram and emergency cooling occur in 5 sec. The maximum increase in temperature of the piping walls would be $\sim$ 8°F. The heat capacity of the gas and heat loss from the piping were neglected in this evaluation. The high heat capacity of the loop piping, together with the fast automatic corrective action (reactor scram in less than 5 seconds), provides a wide margin of safety for loss of secondary coolant accidents.

<u>Failure of Circulators</u>: Normally two circulators are provided with an automatic transfer from one to the other on a signal of low flow, reducing the probability that both circulators will simultaneously be inoperative. The failure of both circulators results in an accident of less severity that the pipe rupture accident described above. The circulators are designed in accordance with ASME code to withstand 400 psig pressure at 600°F. As already shown, the reactor will be scrammed and the fuel assembly cooled by the emergency cooling system before a significant increase in the fuel temperature occurs.

<u>Inadvertent Closure of Line Valves</u>: Valves in the main loop include bypass valves for the full flow filter, by-pass valves for the auxiliary circulator, temperature mixing valves, and heat exchanger bypass valves. Appropriate interlocks are included wherever possible to prevent incorrect selection of valves during loop operation. The temperature mixing valves, however, must be continuously variable to provide adequate loop temperature control. It is possible, by operation error, to close both valves simultaneously. The same sequence of events will occur as those given in this hypothetical "loss of coolant" accident except the fission products will be retained within the main loop. The rise in temperature of the fuel and associated parts during the few seconds intervening between a sudden flow stoppage and the instigation of emergency cooling is minimized by the natural heat capacity of the assembly. The resulting maximum temperatures will be essentiallythe same as that discussed above.

The "Inadvertent Valve Closure" is included here to point out that a near instantaneous flow stoppage will result in no more than an operating inconvenience because the high heat capacity of the fuel assembly provides excellent protection against any sudden change in heat transfer characteristics in the fuel region.

<u>Failure of the Purge Line Ahead of the First Trap</u>: Fission products produced in the fuel are purged by a directed stream of main coolant gas. The purge gas containing fission products passes through a trap within the fuel and a series of external traps located within the main cubicle and returns to the main coolant stream. The main purge line from the fuel element to the first external trap contains

all fission products swept from the fuel region not trapped in the internal fission product trap. A second purge line, normally not used, bypasses the fission product trap within the fuel element and provides a fresh fission product source. The gas flowing in this line can be highly contaminated. Both lines are contained within an additional containment shell extending from the main cubicle penetration flange to the first trap. The shell is held at a reduced pressure relative to the cubicle by the vacuum pump. Leakage into the containment shell is pumped through a filter to the 10 cu. ft. hold-up tank shown in Figure 5.9.7. If a failure in either line occurs, the system contents will be emptied in less than 1-1/2 minutes to this secondary containment. The gas will flow through the transfer piping containing the filter to the hold-up tank. The hold-up tank rupture disc will relieve to the main cubicle at 20 psig. The cubicle exhaust system will isolate on a signal of high radioactivity if the stack monitor setting is exceeded.

If the containment shell surrounding the purge line to the first trap also failed, the contaminated gas is still retained in the main cubicle. The leakage collection system is designed primarily to reduce the probability and extent of cubicle contamination from potential small leak sources, thus reducing the total stack activity released during steady-state operation.

The time to empty the coolant from the system is much greater than that considered in the hypothetical "loss of flow" accident, resulting in negligible increase in fuel assembly temperatures.

In the unlikely event that both purge lines and the containing shell break, the long time required to reach the maximum cubicle pressure and the relatively rapid cubicle pump-down time will result in less activity released to the GETR containment building than that discussed above. A purge line which breaks beyond the first external trap is expected to release primarily long-lived noble gases because a backflow check valve at the exit of the purge system will cause the gas contained in the primary system to pass the first trap. The gas contained in the purge lines and traps downstream from the break and ahead of the check valve will also contain primarily noble gases.

Those gases which have not been condensed by the cold trap are expected to be released. In any case, a break downstream of the first trap is of a lesser magnitude with respect to fission product release. For this reason, the cubicle together with the cubicle vacuum and leakage collection system provides a very reliable control of fission products which may be released from the fuel.

<u>Loss of Coolant to the First External Fission Product Trap</u>: The purge lines leading from the fuel element are discharged in charcoal trap No. 1, located within the main cubicle. The trap is cooled by a demineralized water circulating system which transfers heat from the trap to a heat exchanger cooled on the secondary side by water from the cooling tower. A large heat reservoir is provided by the water in the cooling supply tank and the water within the trap. A 3-gpm circulating pump is provided, but natural circulation will prevent serious overheating in the event of a pump failure. In addition, an alarm is provided for high coolant temperature. In the unlikely event that the cooling line should rupture, the coolant from the cooling supply tank would be lost. The tank is located above the trap so that a line rupture will not drain the water in the trap. It is estimated that three hours would be available after a loss of flow warning occurred for loop operating personnel to take corrective action before overheating could occur in the walls of the trap vessel. It should be noted that this cooling system also supplies cooling water to emergency charcoal trap No. 1. Under normal operation this trap will be isolated from the main system. If this trap was in operation during a cooling water failure, the same conditions would exist as described above.

<u>Loss of Refrigerant to Trap No. 2</u>: Trap No. 2 is cooled to -40°F by a refrigeration system using freon as the coolant. The first trap removes all the fission products escaping from the fuel except the noble gases. Since there is virtually no fission products other than noble gases in trap No. 2 and since the noble gases are trapped without refrigeration, there is virtually no decay heat from fission products. It is not probable, therefore, that a structural failure of the trap would occur as a result of a loss of refrigerant. In any case, a trap failure would be similar to a line break downstream of this first trap which was discussed above.

Loss of Liquid Nitrogen: Liquid nitrogen is supplied to the CO_2 - H_2O trap, charcoal trap No. 3, and emergency charcoal trap No. 2. Loss of liquid nitrogen would allow slight quantities of water, CO_2, and the noble gas fission products to be returned to the main loop. When the main loop gas activity reaches a pre-selected value, the reactor will automatically be started on a "run down".

Loss of Power: Loss of normal power supply will automatically scram the reactor. The reactor emergency power supply, a 150-kw diesel-generator, will have been operating at partial capacity and will supply the required electrical power to the loop. Failure of the emergency power to pick up the loop load will result in negligible increase of fuel assembly temperature because the reactor scram will be accomplished in less than half the time of that described above in "Loss of Coolant" and the coolant is not lost, providing additional heat capacity to the system. The emergency cooling system is designed to flood the vacuum chamber with helium from a loss of power or a loss of instrumentation.

Loss of Instrument Air: All pneumatic valves in the main loop are designed to maintain loop flow without instrument air. If, however, loss of instrument air should cause a main stream valve to close, the effects would be the same as that discussed above in "Inadvertent Closure of Line Valves". It is possible that the valves leading to the hot helium storage tank, although normally closed, could open. The loop contents would be relieved to the tank and the same sequence of events would occur as described previously in "Loss of Coolant" but with less severity because the time to scram the reactor would be approximately the same, while the time to exhaust the loop would be greater. Helium make-up valves and valves on the sampling system are all manually operated. Failure of automatic controlled valves will, therefore, not release loop contents. All valves in the ventilation system fail closed and all valves in the emergency cooling system fail open.

Failure of Rupture Discs on Main Loop: Two separate discs are located on the main loop. The main loop is relieved in the event of overpressurization through the rupture discs into the 10 cu. ft.

hold-up tank which is part of the vacuum and leakage collection system discussed in Section 5.9.1. The same sequence of events would occur as described above in "Loss of Coolant" except the loop contents would not be discharged to the cubicle.

Failure of Main Loop Piping in the Facility Tube: The consequence of the main loop piping failure in the facility tube would be to fill the vacuum chamber with the loop contents. The vessel is designed for the resulting static pressure of 100 psig. A rupture disc is provided, however, that will relieve the pressure at 15 psig to the cubicle resulting in the same degree of contamination of the cubicle and GETR containment building as discussed in "Loss of Coolant". As discussed earlier, when a reactor scram has occurred and emergency cooling has been provided, the temperature starts to decrease. In this case the accident provides, in effect, emergency cooling. The temperature rise of the in-pool components would, therefore, be less than that discussed in "Loss of Coolant".

Failure of the Emergency Cooling System: An emergency cooling system has been designed to flood the vacuum chamber with helium on a signal of low flow and low pressure, increasing the conductivity sufficiently to remove the heat during an emergency condition such as inadvertent closure of line valves. In order to improve reliability of the system, a helium supply tank with a low pressure warning device and two independent valves in parallel are provided. Warning lights on the loop panel and reactor control panel warn that a low pressure exists in the vacuum vessel after a loop-initiated reactor scram, indicating that the emergency cooling system has failed to operate. The rate of increase of temperature of the loop pressure vessel is relatively slow so that sufficient time will be available for the operator to relieve the loop pressure to the hot helium storage tank before danger of loop pressure vessel failure would become imminent. This action may be initiated from either the loop control panel or the reactor console.

Maximum Credible Accident: The maximum credible accident of the loop is a complete circumferential failure of the facility tube pressure vessel adjacent to the core. It is assumed that the 0.250-inch thick aluminum vacuum chamber also fails, allowing the entire fuel assembly to be discharged to the reactor pool. Even in this very unlikely event, the fission products released to the containment vessel by the loop is only $\sim 0.3\%$ of that released in a GETR reactor maximum credible accident, assuming the same percentage of fission products are released. These assumptions are:

1. Operation at power for a sufficient time to establish radioactive equilibrium.
2. 100% of the noble gases and halogen fission products, plus 30% of the solid fission products (total release of 51%) are vaporized instantaneously and immediately mixed uniformly in the containment vessel atmosphere.

The maximum possible reactivity change resulting from flooding the entire vacuum vessel and test assembly, assuming the fuel remains in place, is less than $0.1\% \; \Delta k/k$.

The maximum thrust that could be developed by the vessel as a result of discharging the loop coolant is estimated to be less than 400 pounds. This force applied to the 3/4 inch aluminum wall of the reactor vessel is not expected to cause significant damage. Forces developed as a result of displacement of pool water during the hypothetical accident are considered negligible, since the time required to discharge the helium under water is greater than 1/2 second and the physical arrangement would cause the discharged gas to be broken up into relatively small bubbles. The reaction between water and the fuel element is endothermic and thus the element would be rapidly cooled.

SECTION 6

RADIOACTIVE WASTE AND ENVIRONMENTAL SURVEYS

6.1 Introduction

Radioactive wastes which result from operation of the GETR and the environmental monitoring program which verifies the effectiveness of methods and procedures used to control the release of radiation material are discussed in this section.

6.2 Solid Wastes

The majority of solid wastes consist of demineralizer resins, activated or contaminated reactor or experimental components, filters, and the materials which become contaminated during the course of activities in the facilities' radioactive material areas. Based on experience to date, it is estimated that 6,000 cubic feet of solid waste per year will be transferred from the facility.

Resins and slurries are normally transferred to an 1,800 gallon underground storage tank for decay. They may be concentrated, solidified, or diluted, depending upon the degree and type of contamination, prior to being transferred for disposal. Other solid waste will be transferred for disposal following temporary or decay storage if necessary.

6.3 Liquid Waste

All liquid waste from the laboratory is collected and analyzed prior to release. Disposal of GETR liquid waste is described below.

6.3.1 Non-Contaminated Liquid

Liquids which are not normally contaminated by use at the GETR are piped to one of several 50,000 gallon retention basins at the laboratory chemical treatment plant. Laboratory sewage is collected and treated on site, and the effluents are also transferred to the chemical treatment plant. These liquids are analyzed prior to ground release. If the analysis determines that the liquids are unsuitable for release as specified by federal, state and local regulations, the liquids will be transferred to a waste disposal contractor, allowed to decay, or treated to acceptable levels. The acceptable release concentrations for radioactivity are 10^{-7} μc/ml of beta-gamma activity, 10^{-8} μc/ml of alpha activity or the values specified in 10CFR Part 20, Appendix B, when analysis demonstrates the absence of the isotopes specified in that document. This method of operation has been reviewed and approved by the California State Regional Water Pollution Control Board. Their most recent public hearing on this system was held October 19, 1961.

6.3.2 Contaminated Liquids

Only liquids of low radioactive content have resulted from normal operation. Liquids which are, or are suspected to be, contaminated are routed to one of three 25,000 gallon retention tanks which provide adequate capacity for the maximum expected volume of liquid wastes. Two of the tanks usually provide storage capacity for water of a purity suitable for demineralizing and re-use. The third tank is generally used for water which is not suitable for re-use. These tanks may be used for decay storage as necessary. An additional 20,000 gallon tank is available for above ground storage if necessary. The capacity of this system is adjusted as necessary to provide for anticipated waste quantities. The reactor will not be operated unless there is at least 25,000 gallons of storage capacity available.

Approximately 80,000 gallons of liquid waste a year has been generated at the facility. All but approximately 20,000 gallons per year is treated and re-used. Subsequent handling of these wastes may include their disposal as non-contaminated liquid, further storage at other site facilities, concentration or dilution, and transfer for disposal as radioactive waste depending upon the degree and type of contamination.

6.4 Gaseous Waste

Gaseous effluent wastes which result from operation of the GETR are collected by the facilities' ventilation system and released through a 95-foot high stack. The sources of gaseous effluents, the ventilation system, monitoring system, and release rates are described in this section.

6.4.1 Ventilation System

The building ventilation system is described in Section 2.11 and is illustrated in Figure 2.22. Ventilation systems associated with experimental facilities are described in Section 5.

6.4.2 Sources

The significant sources of gaseous radioactive effluent are loop vents and containment cubicles, reactor vents, the reactor pool, and the experimental capsule and beam port irradiation facilities.

Hold-up tank systems for and vents for the relief of radiogases are associated with the loops as required. The loops also use cubicle containment for isolation of gases which could leak from out-of-pile process systems. Negative pressures are maintained within these cubicles by separate ventilation systems, providing substantial dilution of the radiogas concentrations. Loop vents and cubicle ventilation effluents are exhausted through the stack.

Reactor vents include those used to relieve gas buildup in the primary coolant system, to provide pressure equalization in the retention and storage tanks, and to relieve the reactor pressurizer and demineralizer systems. Effluents are exhausted through the stack.

Effluents in the reactor pool result from neutron activation of dissolved gases, releases from experiments in the capsule header facility, opening the primary system to the pool, an emergency cooling trip actuation, and the fill and flush operation. These gases are removed by air flow sweeping the top of the pool to a ventilation duct leading to the stack.

The experimental capsule hold-up tank system is used to contain and monitor gases released by in-core and capsule header experiments. Individual experiment vent lines are connected to a common manifold which leads to storage tanks. Activity in the tanks is constantly monitored, and, in the event of a capsule failure, the tanks isolate to prevent high gaseous release to the containment ventilation system. The tanks are purged to the stack following adequate dilution and decay.

The forward compartment of the beam port facility is purged of water by nitrogen under pressure. The trace amounts of argon associated with the nitrogen and activated during operation are vented to the stack when the compartment is refilled with water at the end of the run. Controlled purging satisfactorily maintains low release rates during this operation.

Operation of experiments in the shuttle and trail cable irradiation facilities occasionally result in the release of small amounts of radiogases which are not significant in relation to other sources.

6.4.3 Monitoring System

Effluents from the building ventilation system are monitored, when they reach the stack, by the stack monitor and an ionization chamber. An effluent sampling station can be used to collect samples for isotopic analysis. The monitoring of effluents from experimental facilities and isolation of these facilities are described in Section 5.

The stack monitor system collects the stack effluents at a point approximately 20 feet from the base of the stack which is well above the inlet ducts to the stack. A vacuum pump continuously draws effluents through the sampling line, particulate filter, and gas detector. The sample stream is discharged from the gas monitor back to the stack. The gas monitor consists of a shielded vessel which contains a canned scintillation crystal - photomultiplier combination sensitive to gamma radiations only. Output pulses are fed to preamplifier discriminator, an amplifier, and a ratemeter. This equipment is located in a small building at the base of the stack. Readout from the ratemeter is recorded in the control room. The recorder is a two-point unit indicating gas and particulate activity. An adjustable alarm circuit is connected to the ratemeter, producing an isolation trip signal when the count rate reaches a calibrated level. A high stack gas alarm is provided by an adjustable contact on the recorder mechanism that is set to trip at 10% of the isolation level. The monitor design allows collection of an effluent sample during an isolation.

An activated charcoal cartridge will collect iodine from the stack air if present. The cartridge is periodically analyzed for iodine content.

The significant factors affecting the response time of the monitor are the holdup times in the input piping, and sampling volume, and the time constant of the ratemeter. Measurements have shown that there is a 2-second delay between a step concentration increase at the stack inlet and a corresponding increase at the detector crystal. A ratemeter time constant of 10 seconds is used during normal operation. The over-all accuracy of the monitor is better than plus or minum 15% of the indicated count rate. Sensitivity of the monitor, defined as the minimum quantity producing a detectable count rate above background was experimentally determined to be 6.3×10^{-7} μc/cc for A-41, 3.6×10^{-4} μc/cc for Kr-85, and 4×10^{-6} μc/cc for Xe-133.

Gas monitor calibrations have demonstrated that the detector will respond to all gamma emitting fission gases in the range of 0.03 to 1.29 Mev. The detector sensitivity is adequate up to an energy of approximately 2.5 Mev.

The particulate monitoring portion of the stack monitor collects the airborne particulates from the sample stream on a slow moving filter and continuously monitors radioactivity from the trapped particulates. Signals from the detector are fed through a ratemeter for amplification and counting to a multi-point recorder in the control room. The particulate monitor is sensitive to activity as low as 10^{-9} μc/cc.

The performance of the monitor and isolation valves is tested weekly by exposing the gas detector to a calibrated gamma source.

An ionization chamber is mounted inside the containment vessel on the main exhaust duct leading to the building isolation valve. It is connected to a ratemeter, alarm circuit in the reactor control room, and building isolation circuit. This isolation function is intended as a backup to the primary stack monitor. Calibration of the trip point and response of the equipment is checked each operating cycle using a calibrated source. The response time of this alarm circuit is approximately 1 second and is dependent upon gamma intensity. Accuracy of the reading is better than plus or minus 15%.

6.4.4 Release of Gaseous Waste

The release of radiogases from the stack will normally be limited to less than 6,000 μc/sec during steady state operation. If the activity of the stack effluent exceeds ten times this amount, the enclosure ventilation dampers and isolation valves close automatically to seal the containment vessel.

Special care is also exercised to minimize the release of iodines and long-lived particulate materials which might deposit on the surrounding terrain. Operation of the reactor and its auxiliaries is regulated as necessary to prevent an average release of more than 0.1 μc/sec of I-131 or 4 μc/sec of long-lived particulate material. Integrated samples of stack effluent are collected on a continuous basis and analyzed for these constituents at least once each week. In addition, a trip on the particulate monitor is used to close the enclosure ventilation dampers and isolation valves in the event the release of long-lived particulate activity reaches approximately 40 μc/sec. The monitor measures total particulate activity and cannot discriminate between short and long-lived components. Therefore, the trip is set at an appropriate factor above the normal indicated value as determined by comparing the analysis results of a composite long-lived sample with the average particulate activity indicated by the monitor.

These stack release limits were established at a sufficiently low value to assure that the average annual concentration of radiogases and particulates beyond the site perimeter do not exceed the maximum permissible concentration as specified in the applicable section of 10CFR-20. Table 6.1 shows the average composition of the GETR stack exhaust and the maximum annual offsite concentration of these constituents assuming continuous release at a rate of 6,000 μc/sec. The approximate distribution of radioisotopes has been verified by analysing stack gas samples.

Dilution of the effluent was determined using methods described in VBWR License Application Amendment No. 48, License No. DPR-1, Docket 50-18. Specifically, diffusion was evaluated for moderately stable and unstable atmospheric conditions with wind speeds of 1, 5, and 10 meters per second.

Typical seasonal variations of wind direction and velocity were determined from the Vallecitos wind records for the period from Feb. 1, 1962 to Jan 31, 1963. These data are shown on the wind roses of Figure 7.7. Since the effectiveness of the relatively short GETR stack was not known, the diffusion calculations did not take credit for elevated release, but conservatively assumed ground level release. No credit was taken for depletion of the plume concentration as a result of ground deposition.

As indicated in Table 6.1, argon 41 is the limiting constituent of the gaseous effluent from the GETR. Control of this constituent to prevent an offsite concentration of one-half of the MPC, also limits the offsite concentration of all other constituents to less than 0.1 of their respective MPC values. Since neither iodine nor particulate material will be present in concentrations greater than approximately one-thousandth of the maximum permissible concentration for breathing air, human intake through other methods such as ingestion of contaminated food or milk will be insignificant.

Table 6.1

Average Composition of GETR Stack Exhaust

Based upon total stack release rate of 6,000 μc/sec

Constituent	Percent of Mixture	Release Rate (μc/sec)	Max. Average Annual Offsite Concentration (μc/cc)	Annual Offsite Concentration as % of MPC
A-41	40	2000	2×10^{-8}	50
Xe-133	32	2000	2×10^{-8}	7
Xe-135	15	1000	1×10^{-8}	10
Kr-88	10	500	5×10^{-9}	14
Kr-85M	3	200	2×10^{-9}	0.7
Particulate	.0005	0.03	3×10^{-13}	0.1
I-131	.00002	0.001	1×10^{-14}	0.003

6.5 Waste Packaging and Transfer

High level wastes are promptly transferred after packaging and labeling to site waste facilities authorized by material licenses. Low level waste may be temporarily stored at the GETR site until transferred to a waste disposal contractor or for additional storage at site waste facilities. Wastes containers are inspected and surveyed to assure that they meet all applicable specifications and regulations governing the packaging, labeling, and shipment of radioactive waste. Final waste disposal will be accomplished by the Commission, its authorized contractors, or licensed waste disposal contractors.

6.6 Environmental Monitoring

Environmental monitoring is performed both on and off site to verify the control of radioactive material. The monitoring programs determine radiation levels in the environs and the radioactive contents in water, soil, vegetation, and air.

6.6.1 Gamma Monitoring Stations

Over thirty gamma monitoring stations are in operation on the site. A few additional stations are in operation in the surrounding areas. Each station usually contains ionization chambers with ranges of 0-10 mr and 0-200 mr, and a film pack. The site stations have been located 360 degrees around and at varied distances from the reactor areas. For this purpose the site has been divided into sixteen 22.5 degree sectors. Each sector contains one to six stations depending upon the predominant wind directions.

6.6.2 Water Sampling

Streams in the area adjacent to the site, streams leaving the site, and waste water effluent discharge are sampled and the samples analyzed for uranium and beta-gamma and alpha activity.

Surface waters and ground waters are sampled bi-monthly. All surface streams leaving the site, key streams in the vicinity of the site, and strategically located wells on the site proper are sampled for the usual chemical and mineral constituents as well as for radioactivity.

6.6.3 Soil and Vegetation

Soil and vegetation samples from the site and offsite locations are collected and analyzed periodically. The program includes the collection of samples from stream bottoms, marine flora, plankton, cattle fodder, established sampling holes, plants, and surface soil.

The soil samples are analyzed for uranium and gross beta-gamma and alpha activity. The vegetation samples are analyzed for uranium, I-131, and gross beta-gamma and alpha activity.

6.6.4 Air Monitoring

Four environmental monitoring stations, placed approximately 90° apart, monitor the air on the site. Each station contains a lead shielded geiger tube assembly. The air is drawn directly from the atmosphere, through the filter paper, and exhausted back to atmosphere. The buildup of beta-gamma activity on the filter paper is monitored by the G-M tube and the signal picked up by a logarithmically calibrated count rate meter, the output of which is recorded on a strip chart recorder. Each station also contains an unshielded, thin wall G-M tube, which records in the same manner as the particulate monitor. The filter paper component detects particulate matter in the atmosphere while the open G-M tube component detects background radiation and all other beta-gamma radioactivity. Additionally, the filter papers are removed weekly, counted, and the radioactivity concentrations recorded.

Data from these stations may be correlated with data obtained from the site meteorological station which consists of sufficient components to record wind direction, speed, turbulence, gustiness, temperature, humidity, lapse rate, and precipitation. Wind roses developed from data collected at the station are shown in Figure 7.7.

6.6.5 Results

Environmental survey results demonstrate that the natural background radiation level has not increased over the several years of reactor operation. Levels of the order of 100 times normal are measured during fallout periods following weapons testing.

Survey results have been routinely exchanged with local nuclear installations and provided to the AEC Mare Island Naval Shipyard, Naval Radiological Defense Laboratory, San Francisco Bay Region Water Polution Board, U. S. Public Health Service, California Department of Public Health, Lawrence Radiation Laboratory, San Jose City Health Department, Radiation Detection Company, AEC San Francisco Operations Office, Santa Clara Health Department, Stanford University, and the local newspapers.

SECTION 7

THE SITE

7.1 Introduction

The General Electric Test Reactor is located on the 1594 acre Vallecitos Atomic Laboratory site in Alameda County, California as shown in Figure 7.1. The site location, surrounding area, site characteristics, facilities, and activities are described in this Section.

7.2 Site Location

The Vallecitos Atomic Laboratory is situated near the center of Pleasanton Township. The site location is circled on the U. S. Geological Survey map shown in Figure 7.2 The Laboratory is east of San Francisco Bay, approximately thirty-five (35) air miles east southeast of San Francisco and twenty (20) air miles north of San Jose. The nearest towns are Pleasanton, Livermore, and Sunol.

7.3 Surrounding Area and Population

7.3.1 Geography

The Laboratory is located on the north side of the Vallecitos Valley which is approximately two miles long and one mile wide with major axis running east-northeast and west-southwest. The valley is at an elevation of approximately 500 feet and is surrounded by barren mountains and rolling hills which rise to elevations of 700 feet above the general site elevation as illustrated in Figure 7.3.

The land immediately adjacent to the site as well as land to the north, south and west is devoted to agriculture and cattle raising. The area to the east and southeast is largely waste land and is sparsely covered with scrub trees and wild grass.

7.3.2 Population Centers

The residential population density in the immediate vicinity is very low. There has been no significant population growth or change in the use of the surrounding area within a three mile radius of the

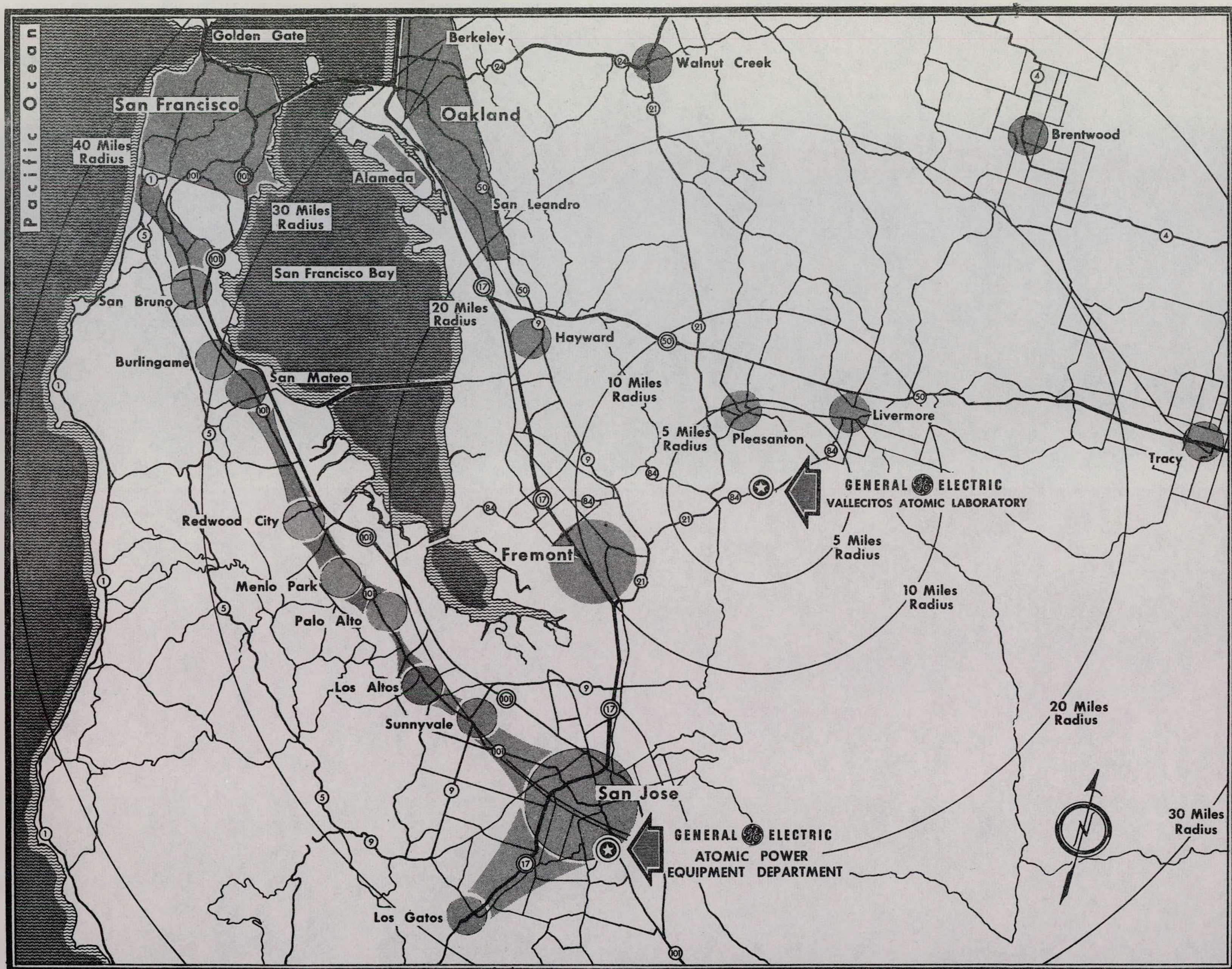

Figure 7.1 BAY AREA MAP


SANTA RITA
AMADOR
VALLEY
LIVERMORE
VALLEY
VALLE DE SAN JOSE
(SUNOL & BERNAL)
VALLE DE SAN JOSE
(SUNOL & BERNAL)
MISSION
FREMONT
Pleasanton
Irvington District
Niles District

Figure 7.2

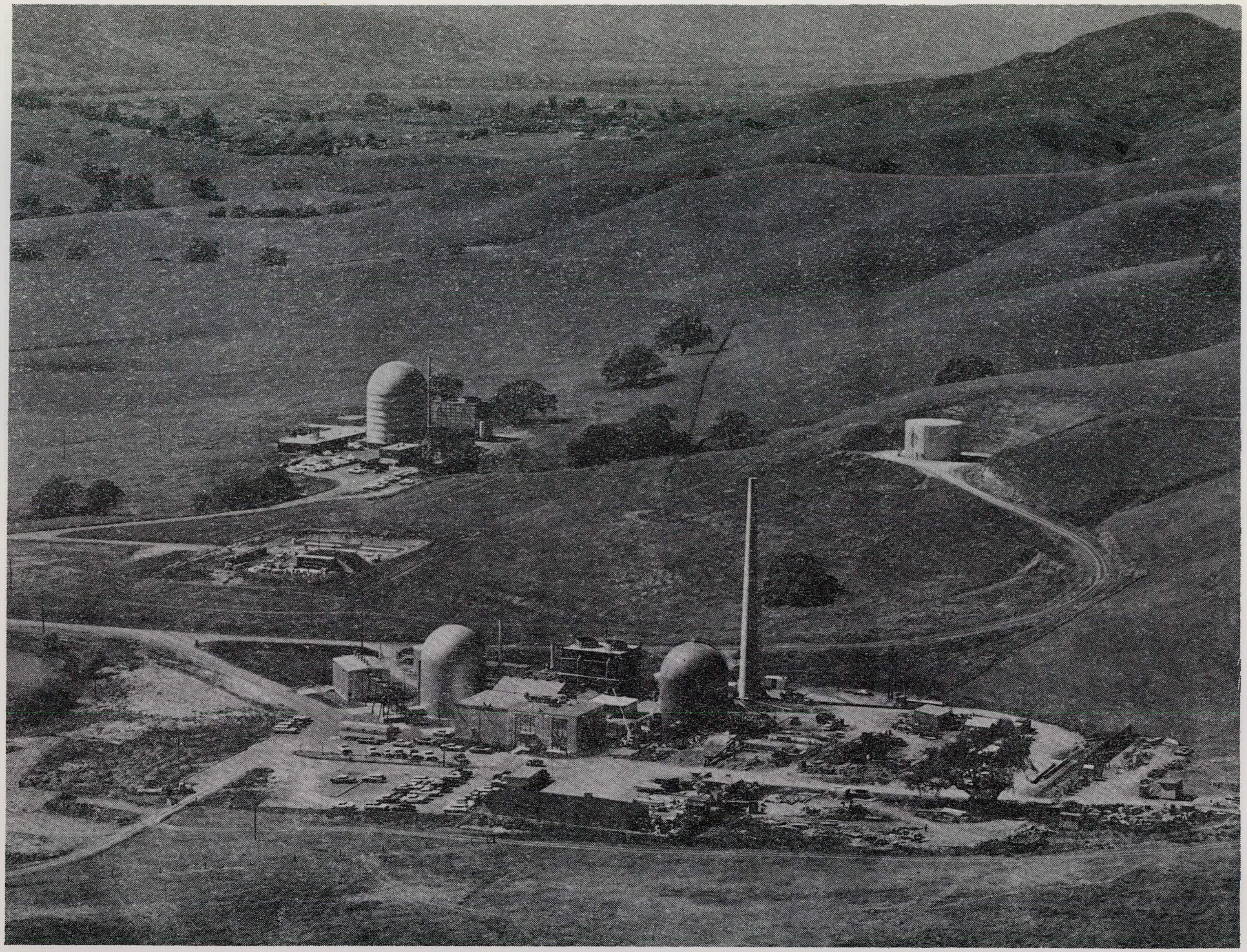

Figure 7.3

Figure 7.4

GETR since its construction and none is anticipated. Minor land development has occurred at distances of two and one-half to three miles to the west and northwest associated with the expansion of the towns of Pleasanton and Sunol, and the unincorporated area of Happy Valley. Population within a ten mile radius of the site is estimated to be 34,000, while that within a twenty mile radius is estimated to be 400,000.

The largest population center within a ten mile radius is Livermore with a population of approximately 16,000 and located six miles to the northeast beyond a 1200 foot mountain range. It is the principal center of trade and shipping for the local agricultural activities. The second largest town is Pleasanton, a farm center of 4,200 population, located over the hills four miles northwest of the site. A United States Veterans Administration Hospital with a population of approximately 1,000 is located four air miles to the east.

Located to the southwest at a distance of approximately twelve miles are a group of towns which recently incorporated into a single city called Fremont. Fremont has a population of approximately 44,000, borders the eastern shore of San Francisco Bay, and is separated from the site by a mountain range which rises 1200 feet above Vallecitos Valley. Also to the southwest and beginning at a distance of 20 miles from the Laboratory are the suburbs of San Jose, which has a population of over 200,000. Beginning 15 miles to the northwest is the city of Hayward which has a population of approximately 73,000, and at greater distances to the northwest are the large cities of San Francisco, Oakland, Berkeley, and their residential suburbs. Somewhat over 20 miles to the west is the San Francisco peninsula with its continuous string of suburbs along the west shore of the Bay. Population figures are based on the 1960 Bureau of Census population report.

7.3.3 Commerce and Industry

The economy of the area is based on agriculture and cattle raising. Commercial activity within a 15 mile radius is limited largely to the type of retail businesses necessary to support the farm community. It is not until such towns as Hayward and San Lorenzo are reached that a suburban type of community is encountered. Approximately 7 miles to the northeast is the Livermore site of the University of California Radiation Laboratory. Camp Parks and Alameda County Prison Farm are located about 3 miles north of Pleasanton near the towns of Dublin and San Ramon.

There is very little industry within a 20 mile radius of the site. The towns of Livermore and Pleasanton contain a small amount of light industry, but can in no way be considered industrial centers. The city of San Jose to the south, 20 miles distant, and Oakland and San Francisco, 30 and 35 miles respectively, to the northwest are the major industrial centers in the vicinity.

The Southern Pacific and Western Pacific Railroads lie about two miles west of the site and pass through the towns of Livermore and Pleasanton. Sidings on both lines are situated about three miles from the site.

7.4 The Laboratory Site

The Laboratory site is shown on Figure 7.4 and described in this Section.

7.4.1 Topography

Approximately one-quarter of the site in the southwestern corner is gently sloping or slightly rolling terrain. The remainder consists mostly of the southwestern slope of a ridge serrated by several small canyons or draws. The site is on the north side of Vallecitos Road which is a two lane paved highway. A topographical map of the site is shown in Figure 7.5.

7.4.2 Security

Vallecitos Atomic Laboratory property contained within fencing at the perimeter of the site, shown outlined in Figure 7.5, is considered the site exclusion and restricted area. There are three gates in the perimeter fencing. The main gate is guarded at all times to control personnel entrance and exit. The other gates are kept locked, with appropriate procedural control of keys. Inner site

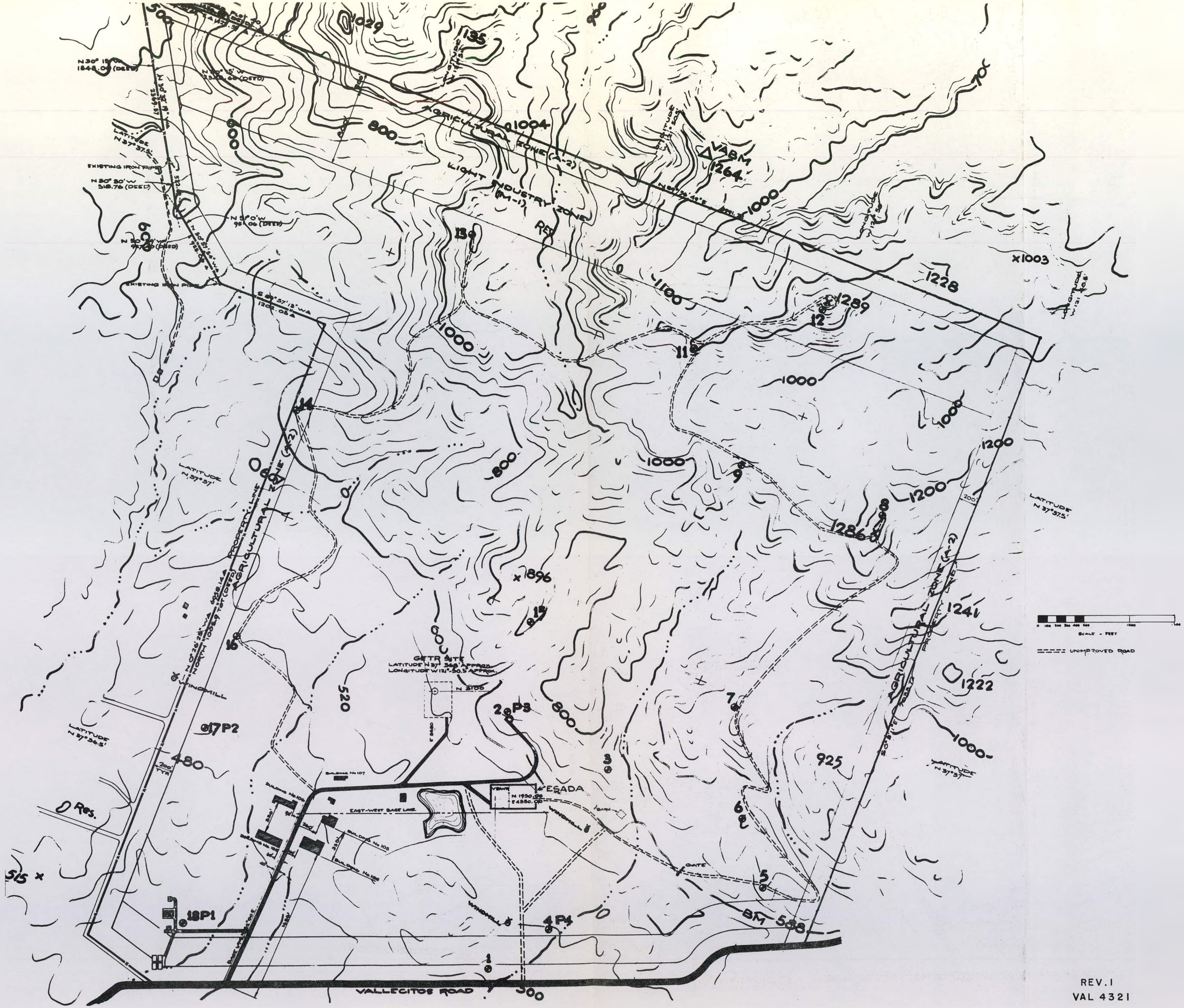

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Figure 7.5.

fences have been erected and isolate the Laboratory operating facilities from the property leased for grazing.

The GETR is located 2800 feet from the nearest site boundary and 3,000 feet from Vallecitos Road. Access to the test reactor facility is also restricted and is limited to those personnel having a valid interest in the operation and service of the Laboratory facilities.

7.4.3 Water Supply

Water is supplied to the Laboratory from Hetch-Hetchy Aqueduc which provides water to the city of San Francisco. A 14-inch line 25,000 feet long has been installed from that aqueduct to the site. The pumps presently installed have a capacity of 1,000,000 gallons per day. The ultimate capacity of the pump house and pipe line is over 3,000,000 gallons per day, although the Laboratory is presently authorized to withdraw up to 2,000,000 gallons per day. A 500,000 gallon storage tank is provided on the Laboratory site. As shown in Figure 7.6, it is located on the hillside above GETR at a distance of 1500 feet.

7.4.4 Fire Protection

The designs of GETR facilities make maximum use of non-combustible structural material. A six inch fire loop surrounds the facility with hydrants on all four sides. This loop is supplied from the 500,000 gallon storage tank described in Section 7.4.3. One hundred thousand gallons are reserved for fire protection. Other fire precautions include appointment and training of a fire brigade, availability of fire fighting equipment for use throughout the Laboratory, and installation of a fire alarm network and communication system.

If no source of power is available, raw water from the fire protection system may be flushed into the reactor pool, then drained off to the retention tanks to maintain the pool levels.

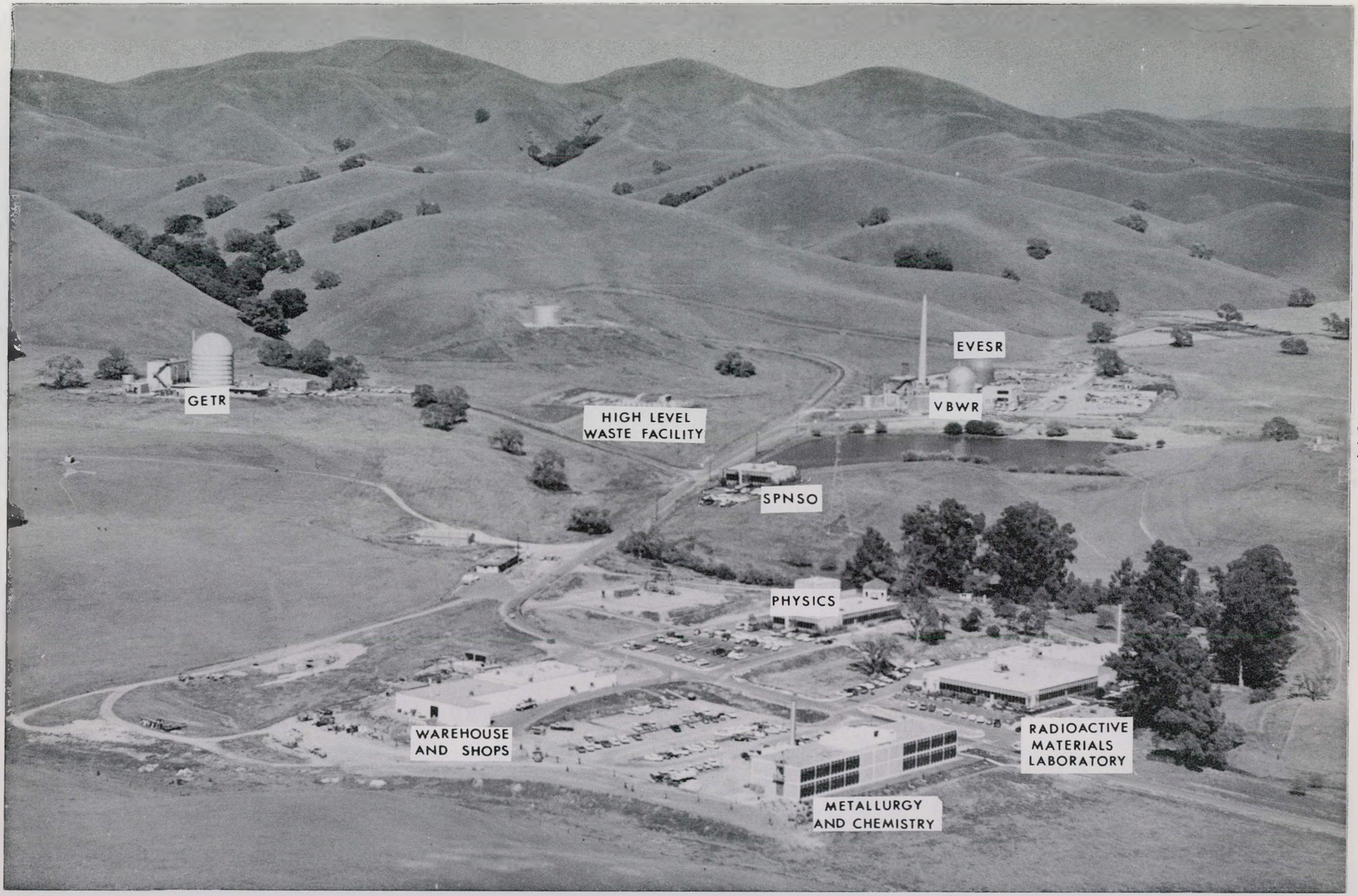

Figure 7.6. GENERAL ELECTRIC VALLECITOS LABORATORY

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7.4.5 Electrical Power

Electrical power is supplied to the main Laboratory substation by the Pacific Gas and Electric Company, from where it is distributed to other site facilities. It is supplied to the primaries of the GETR substation at 12 Kv, 3 phase.

7.4.6 Laboratory Facilities and Activities

Facilities currently located on the Laboratory site are shown in Figure 7.6. Approximately 1500 acres of the site are leased for raising feed crops and cattle grazing. The main Laboratory buildings are located approximately 1700 feet north of the Vallecitos Road. The GETR is located approximately 1700 feet northeast of these buildings and 1200 feet northwest of the VBWR and EVESR.

Building 102 houses the Radioactive Materials Laboratory and Nuclear Material Laboratories where research and development activities, and irradiation studies and services are performed. Building 103 Laboratories are used for chemistry, metallurgy, and ceramics research, development and analytical activities. The Physics Building 105 houses the Nuclear Test Reactor and Critical Experiment Facility. Building 106 contains maintenance and development shops and warehouse facilities. The SPNSO Building houses facilities for development of special purpose nuclear systems. The High Level Waste Facility (HLWF) is located to conveniently service all nuclear facilities. The VBWR and EVESR are developmental boiling water and superheat reactors. Administrative offices are provided in Building 107 and at each facility described. The liquid waste chemical treatment plant and sewage treatment plant are located in the southwest corner of the site as shown in Figure 7.5.

7.5 Site Characteristics

7.5.1 Hydrology

The hydrology of the site has been studied by Joseph F. Poland, District Geologist, Ground Water Branch, U.S. Geological Survey, Department of the Interior, from the point of view of the paths traveled by materials dispersed near the reactors (up hill from the other Laboratory facilities) until they reach points of diversion of surface water or pumping of underground water. The full text of the report appears in Appendix A of SG-VAL-2, Third Edition, General Electric

Vallecitos Boiling Water Reactor Final Hazards Summary Report which is part of Docket No. 50-18.

7.5.2 Meteorology

A meterological study has been made of the Laboratory site by M. Neiburger of the University of California at Los Angeles. His report, entitled "Preliminary Report on Expected Meteorological Conditions at the Proposed General Electric Research Laboratory in Vallecitos Valley, California", dated October 21, 1955, appears in Appendix B, SG-VAL-2, Docket No. 50-18. This report is based in part on a report prepared by the Scientific Services Division, U. S. Weather Bureau, dated May, 1952, entitled "Expected Meteorological Conditions for the Livermore Research Laboratory of the Atomic Energy Commission" and appears in Appendix C, SG-VAL-2, Docket No. 50-18. Seasonal wind roses for spring, summer, and autumn 1962 and winter of 1962 - 1963 are plotted on Figure 7.7 from data obtained at the meteorological station located on site.

There is no likelihood of major flooding of the Vallecitos Atomic Laboatory. There is, however, some possibility of flash floods resulting from heavy rainfall and resultant runoff from the surrounding hills.

Violent storms are infrequent in this area. The main consequence of such storms would be the interruption of power service from the Pacific Gas and Electric system. To eliminate this hazard, all reactor electrical circuits requiring uninterrupted power for safe and reliable operation are fed by a diesel-generator at the site. This unit will be in continuous service thus making the facility coolant system independent of outside electric power. The reactor is automatically shut down in the event normal power is lost. The containment vessel is designed for winds of 75 mph, which is greater than any wind velocities recorded in this locality.

7.5.3 Seismology

A seismographic study of the Laboratory site has been made by Perry Byerly, Seismologist, and Jack F. Evernden, Geologist, of the University of California at Berkeley. Their report is included in Appendix D, SG-VAL-2, Third Edition, Docket 50-18. The author of

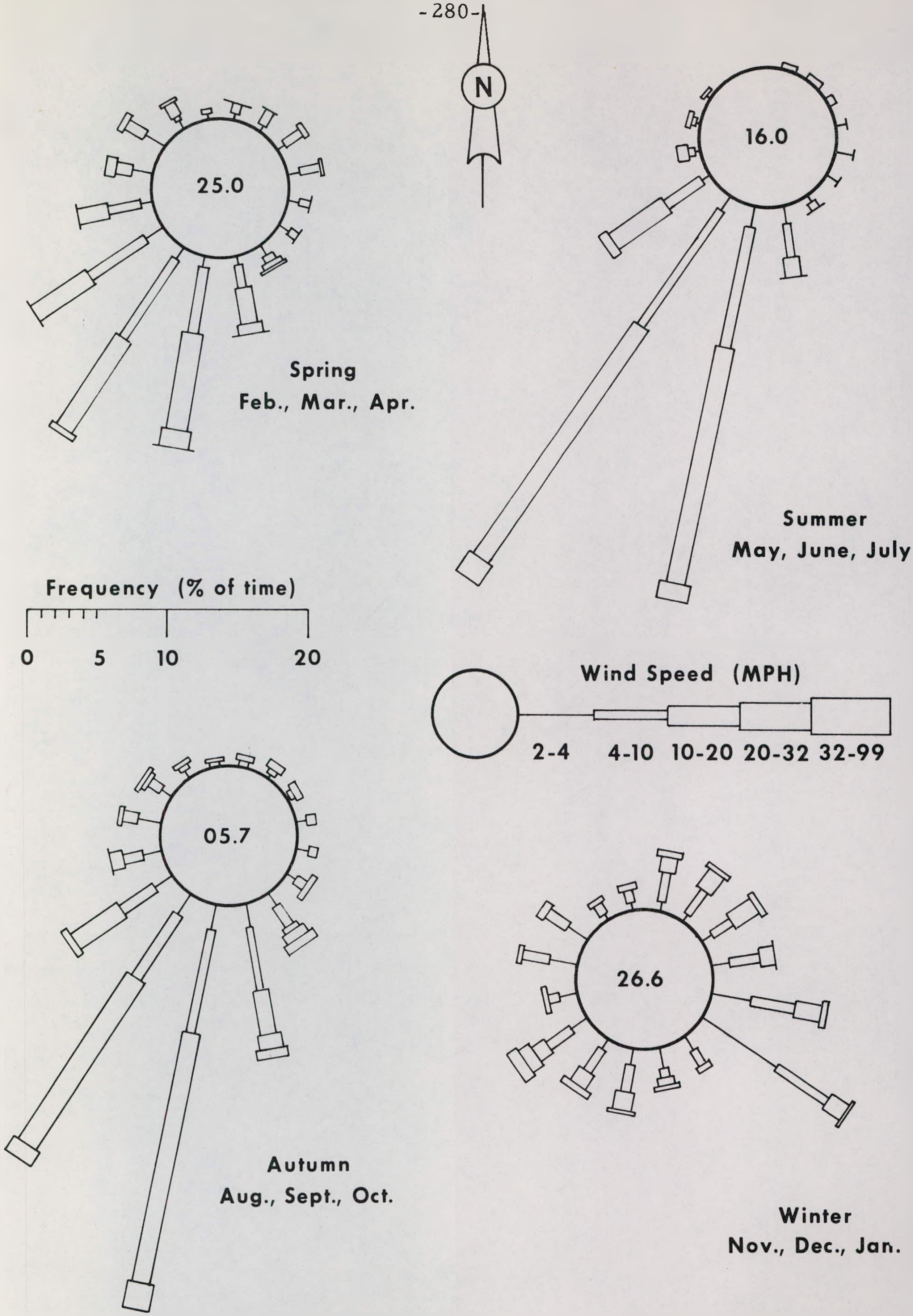

Figure 7.7. SEASONAL WIND ROSES VALLECITOS ATOMIC LABORATORY
February 1, - January 31, 1963

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This report concluded that the entire east bay area should be considered as a seismically active area, but that the Williams fault in the vicinity of the Laboratory is less dangerous than other nearby faults. The reactor enclosure and plant were designed and constructed to conform with Uniform Building Zone 3, Earthquake Code, which applies to the San Francisco Bay Area. A seismoscope is provided and instrumented to shut down the reactor in the event of significant seismic activity.