FROM:	DATE OF DOCUMENT:	DATE RECEIVED	NO.:
The Babcock & Wilcox Co. 161 East 42nd Street, NYK 17 (R. H. Harrison)	7-10-62	7-12-62	7367

LTR.: X MEMO.: and Application REPORT: OTHER:

TO: LRA

ORIG.: 3 CC: 19 OTHER: plus 18 addl. of Report BAW-183

ACTION NECESSARY [X] NO ACTION NECESSARY [ ] CONCURRENCE [ ] COMMENT [ ] DATE ANSWERED: BY:

CLASSIF.: U POST OFFICE REG. NO.:

FILE CODE: 50-200 & 70-684

DESCRIPTION: (Must Be Unclassified)
Ltr. transmitting 6 enclosures as appl. for construction & oper of B&W Test Reactor (6 megawatts w/design for 12 MW oper) at B&W's Mt. Athos site near Lynchburg, Va., as part of Nuclear Development Center:

ENCLOSURES:
Attach. I - General Info (22 cys)
II - Facilities & Exp. (22)
III - Misc. Background ... (22)
IV - Report BAW-183--PH Report (40)
V - Financial Statement--(22)
VI - 1961 Annual Report (fin.) (22)

REFERRED TO	DATE	RECEIVED BY	DATE
S. Levine with: Suppl. File cy; 2 - Compliance; 4 - Extras; 1 ea OROO & TI	7-13-62	(assigned to you by Mr. Case)	
D. Nussbaumer with: File cy 70-684	7-13		
Mr. Nussbaumer:		Will SNM licenses have to be issued for storage only and "Auxiliary Systems" under Section 4 of Report BAW-183??? Reba	

REMARKS:
Mail Room Dist: 1 - Form 50-200
2 - OGC
1 - L. Huard & H. Steele
~~(NO CYS. SENT ISOTOPES FOR BYPROD Po-Be since Facility License will probably cover it because it is for startup)~~

U. S. ATOMIC ENERGY COMMISSION

MAIL CONTROL FORM FORM AEC-326S (8-60)

GOVERNMENT PRINTING OFFICE: 1961 — 613958

The Babcock & Wilcox Company
Established 1867
ATOMIC ENERGY DIVISION

CABLE ADDRESS
GLOVEBOXES

161 EAST 42nd STREET
NEW YORK 17, N. Y.

TELEPHONE
MURRAY HILL 7-6700

L&R File Copy

July 10, 1962

U. S. Atomic Energy Commission
Division of Licensing and Regulation
Washington 25, D. C.

Subject: Application for a (1) Facility Construction Permit;
(2) Class 104 License;
(3) Allocation of Special Nuclear Material;
(4) Special Nuclear Material License and
(5) Byproduct Material License

Gentlemen:

Pursuant to the Atomic Energy Act of 1954, as amended, and Title 10, C.F.R. Chapter 1, Part 50, "Licensing of Production and Utilization Facilities", The Babcock and Wilcox Company hereby makes application for a construction permit to construct a utilization facility hereinafter referred to as "BAWTR" (B&W Test Reactor).

The proposed facility, which is hereinafter more fully described, will be owned and operated by this applicant and will be located on a site in Campbell County, Virginia, which is also owned by this applicant.

Upon completion of construction of the proposed BAWTR in accordance with the terms and conditions of the aforementioned construction permit; upon the filing of any additional information needed to bring the original application up to date; upon filing of proof of financial protection and execution of an indemnification agreement as required by Section 170 of the Act and the Commission's regulations, and in conformity with the provisions of the Act and the rules and regulations of the Commission; this applicant requests the Commission to issue a Class 104 License to The Babcock and Wilcox Company pursuant to Section 104 (c) of the Act which license shall expire twenty (20) years from the date of the aforementioned construction permit.

The information required by Sections 50:33, 50:34, and 50:60 of the regulations in Title 10, Chapter 1, C.F.R., Part 50, is submitted herewith as Attachments:

I. General Information
II. Facilities and Experience
III. Educational Background and Experience of B&W Personnel
IV. BAWTR Summary Hazards Report — BAW-183
V. Financial Information
VI. B&W Financial Report for 1961

which are, by this reference, made a part of this application.

7367

U. S. Atomic Energy Commission
Division of Licensing and Regulation
Page Two
July 10, 1962

Additional information regarding allocation of Special Nuclear Material, Special Nuclear Material License and Byproduct Material License will be submitted later as a supplement to this application.

Except as specifically set forth in this application, no licenses have been issued or applied for in connection with the subject facility.

This application includes three (3) signed originals and nineteen (19) copies of the application plus eighteen (18) copies of Attachment ~~III~~ IV, Hazards Summary Report.

A copy of this letter application and one (1) copy of each of the attachments have been forwarded to Mr. Harvey A. Mitchell, Executive Secretary of Campbell County, Rustburg, Virginia, in accordance with the requirements of Section 2.101 (A) of the Atomic Energy Commission's regulations, 10 C.F.R., Part 2, "Rules of Practice".

Please advise if further information is required or if you wish to meet with our representatives to discuss this application.

IN WITNESS WHEREOF, The Babcock & Wilcox Company has caused its name to be hereunto signed and its corporate seal to be hereto affixed by its duly authorized officers this 10th day of July 1962.

THE BABCOCK & WILCOX COMPANY

R H Harrison
R. H. Harrison
General Manager

Attest:

E F Horgan
E. F. HORGAN, ASST. SECRETARY

Sworn to before me this 10th day of July 1962.

Emma S. Lombardo

EMMA S. LOMBARDO
NOTARY PUBLIC, State of New York
No. 60-2395000
Qualified in Westchester County
Certificate filed in New York County
Commission Expires March 30, 1963

50-200 + 70-68

EXTRACT FROM MINUTES OF A MEETING OF THE BOARD OF DIRECTORS OF THE BABCOCK & WILCOX COMPANY, HELD AT THE COMPANY'S OFFICE, 161 EAST 42ND STREET, NEW YORK, N. Y., ON WEDNESDAY, APRIL 25, 1962

On motion duly made, seconded and unanimously carried, it was

RESOLVED, that the following persons, and each of them be and they hereby are authorized to enter into contracts for and on behalf of the Company, with the Navy Department or any other department or agency of the United States of America, or any other purchasers, covering the sale of any of the Company's products or auxiliaries and accessories, or construction work in connection with the installation of such products, auxiliaries and accessories, and for and in the Company's name to sign and affix its corporate seal to any such contracts and to all bonds, obligations and undertakings required by any such departments, agencies or other purchasers, for the faithful performance of any such contracts:

M. Nielsen	President
A. E. Phin	Vice President
Carl Claus	Vice President
Boiler Division:	
L. S. Wilcoxson	Vice President
S. T. Mackenzie	Vice President
W. H. Rowand	Vice President
J. P. Craven	Vice President
G. W. Kessler	Vice President
J. D. Hitch, Jr.	Vice President
C. J. Odell	Manager, Pricing Section
G. H. Hodges	Manager, Marine Department
G. A. Mattucci	Assistant Manager, Marine Department
Tubular Products Division:	
W. J. Thomas	Vice President
J. S. Anderson	Vice President and General Manager
L. B. Wohlgemuth	General Sales Manager
W. H. Buley	Manager of Stainless Sales
J. A. Menster	Manager of Sales, Welded Tubing
K. W. Harris	Manager of General Sales - Milwaukee
Refractories Division:	
R. A. Barr	Vice President
R. P. Stuntz	Sales Manager
Atomic Energy Division:	
R. H. Harrison	General Manager
Research & Development Division:	
A. P. Taber	Vice President

- - - - - - - - 0 - - - - - - - -

I HEREBY CERTIFY that the above is a true and accurate transcript from the minutes of a meeting of the Board of Directors of The Babcock & Wilcox Company as recorded in the Minute Book of said Company.

Martin Victor

Martin Victor, Secretary

The Babcock & Wilcox Company
Established 1867
B&W
ATOMIC ENERGY DIVISION
CABLE ADDRESS GLOVEBOXES
TELEPHONE MURRAY HILL 7-6700
161 EAST 42nd STREET
NEW YORK 17, N. Y.

50-200 / 7-16-62

July 10, 1962

Mr. Harvey A. Mitchell
Executive Secretary of Campbell County
Rustburg, Virginia

Dear Mr. Mitchell:

By letter dated July 10, 1962, a copy of which and six (6) attachments thereto are enclosed herewith, The Babcock & Wilcox Company has filed an application with the United States Atomic Energy Commission for certain licenses relating to the construction, possession and operation of a Nuclear Development Center to be constructed, operated and owned by this Company.

In compliance with Section 2.101(A) of the Commission's regulation, 10 CFR, Part 2, "Rules of Practice", we are forwarding herewith a copy of the above mentioned letter application and the attachments thereto.

In order that our files may indicate that this letter and the attachments have been received by you, we will appreciate it if you will initial one (1) copy of this letter, indicating your receipt of that document and the attachments, and return it to Mr. G. W. Mitchell, The Babcock & Wilcox Company, 1201 Kemper Street, Lynchburg, Virginia.

Thank you for your assistance.

Sincerely,

R. H. Harrison
General Manager

RHH:GWM:acg

cc: Division of Licensing and Regulation - AEC

50-204
L&R File Copy 70-684 Suppl.

ATTACHMENT I

GENERAL INFORMATION

7367

1. Name of Applicant

The Babcock & Wilcox Company

2. Address of Principal Offices

161 E. 42nd Street
New York 17, New York

3. Description of Applicant's Business

A description of applicant's general business, experience and atomic energy are set out in Attachment II.

4. State of Incorporation

The Babcock & Wilcox Company is a corporation, organized and existing under the laws of the State of New Jersey.

5. Names, Addressees and Citizenship of Directors and Principal Officers

DIRECTORS	OFFICE	CITIZENSHIP
A. G. Pratt 345 Walnut Street Englewood, New Jersey	Chairman	United States
M. Nielsen 1035 Fifth Avenue New York 28, New York	President	United States
William O. Baker Spring Valley Road Morristown, New Jersey		United States
Walther H. Feldmann 150 E 69th Street New York 21, New York		United States
J. Roy Gordon 1115 Fifth Avenue New York 28, New York		Dominion of Canada
C. W. Middleton Birdwood Farms Route 6 Charlottesville, Virginia		United States

5. (Cont'd)

DIRECTORS	OFFICE	CITIZENSHIP
Stoddard M. Stevens 66 Windermere Terrace Short Hills, New Jersey		United States
W. J. Thomas 155 McKinley Road Beaver Falls, Pa.	Vice President	United States
B. A. Tompkins 770 Park Avenue New York 21, New York		United States
John C. Traphagen West Nyack, New York		United States
L. S. Wilcoxson 1977 Stockbridge Road Akron 13, Ohio	Vice President	United States

OTHER PRINCIPAL OFFICERS	OFFICE	CITIZENSHIP
R. A. Barr 867 River Road Fair Haven, New Jersey	Vice President	United States
Carl Claus 19 Paulding Drive Chappaqua, New York	Vice President	United States
Alan E. Phin 16 Herkimer Road Scarsdale, New York	Vice President	United States
A. P. Taber 1224 S. Union Avenue Alliance, Ohio	Vice President	United States
R. J. Cantwell 5 Cotswold Way Scarsdale, New York	Comptroller	United States
W. G. Dryden 19 Lebanon Road Scarsdale, New York	Treasurer	United States

5. (Cont'd)

OTHER PRINCIPAL OFFICERS	OFFICE	CITIZENSHIP
Martin Victor Locust Valley Long Island, New York	Secretary	United States

6. Ownership and Control of Applicant

The Babcock & Wilcox Company is not owned, controlled or dominated by an alien, a foreign corporation or a foreign government.

7. The Applicant has filed this application on its own behalf and is not acting as an agent or representative of another person or corporation.

8. Erection

The facility will be erected by the Applicant.

9. Financial Qualifications

Attached is a copy (Attachment V) of Applicant's annual financial report for the calendar year 1961.

10. Technical Qualifications of the Applicant

The applicant's products and experience are set forth in Attachment II and Attachment IV contains the educational background and experience of the personnel primarily responsible for the design, fabrication, erection and operation of the proposed facility.

11. Scheduled Dates for Completion of the Facility

Construction of the facility is scheduled to begin September 15, 1962. The earliest date for the completion of the facility is August 31, 1963. The latest date for the completion of the facility is December 31, 1964.

12. Access to Restricted Data

The applicant agrees that it 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 Commission on the character, associations, and loyalty of such individual, and the Commission shall have determined that permitting such person to have access to Restricted Data will not endanger the common defense and security.

50-200
70-68
L&R File Copy Suppl.

ATTACHMENT II

APPLICANT'S

FACILITIES

AND

EXPERIENCE

RECEIVED
JUL 1 1962
U.S. Atomic Energy Commission
DIVISION OF
LICENSING & REGULATION

7367

FACILITIES - B&W ATOMIC ENERGY DIVISION

The B&W Atomic Energy Division facilities consist of the following:

Offices	Location
Executive Offices Computer Center Engineering Offices	1201 Kemper Street Lynchburg, Virginia
Engineering Offices Experimental Prototype Development Facility	Mt. Athos (5 miles east of Lynchburg on James River)

The Computer Center includes an Electrodata 204 Digital Computer manufactured by the Burroughs Corporation. The unit is a decimal calculator with a speed of 200 instructions per second. Each of these instructions defines a ten digit arithmetic as logical operation.

Programs have been developed and are available for over 200 types of problems including multigroup, multiregion, core physics, reactor kinetics, shielding, engineering, stress work, hydraulic analysis and many others.

B&W recently put into operation a Pace 231R Analog Computer manufactured by Electronics Associates, Inc. This unit can handle a wide range of engineering problems. Individual control of all system integrators is available at the patch panel. As a result it is now possible to solve multi-stage type problems formerly beyond the capabilities of standard general purpose analog computers.

The Experimental Prototype Development Facility includes a machine shop, welding area, and mock-up area. The facility conducts developmental programs regarding mechanical handling equipment, fuel elements, control rods, etc. The shop has electronic equipment available to conduct electronic experiments, if required.

Further, the shop supplies engineering empirical data to the designer in support of conceptual designs. Thus, the shop is extremely useful to the designer in solving problems before shop fabrication and field operation.

FACILITIES - B&W RESEARCH AND DEVELOPMENT DIVISION

The Research and Development Division operates the Research Center in Alliance, Ohio. The Critical Experiment Laboratory in Lynchburg, Virginia is a part of that Division.

Research Center

The Research Center is located on an 18-acre fenced site, including 14 buildings that provide 136,000 sq. ft. of floor space. Complete facilities are provided to undertake research and development in the areas of chemistry, metallurgy, low-level radiochemistry and radiometallurgy, non-metallic materials, welding, nuclear engineering, heat transfer, fluid flow, stress analysis, electronics and theoretical analysis.

The Research Center staff has extensive experience in the complete design, fabrication and pre-testing of in-pile loops. The staff is well qualified to conduct aqueous as well as liquid metal corrosion and mass transport studies. Extensive experience is available in the testing of fuel elements - both individually and as bundles - for various simulated service conditions of thermal and pressure cycling, temperature differential tests, vibration, fatigue, collapse and bending tests.

Extensive experience and facilities are also available for determining pressure drops in both fluids and gases. Flow models are used successfully in solving problems in fluid mechanics. Special techniques have been developed that make it possible to obtain quantitative and qualitative results for both steady state and transient conditions.

Critical Experiment Laboratory

The Critical Experiment Laboratory is a complete facility which includes two critical experiment bays, the Lynchburg Pool Reactor (LPR), and their associated control rooms. The support laboratories within the facility include a health physics laboratory, chemical laboratory, counting room, and a physics laboratory where work related to, and in support of, the experimental programs is conducted.

Four research assemblies are in the Laboratory; three are critical assemblies, and the fourth is a pool reactor similar in concept to the Bulk Shielding Reactor at Oak Ridge National Laboratory.

B&W CRITICAL EXPERIMENT LABORATORY
(Lynchburg, Virginia)

CRITICAL EXPERIMENTS:	TYPE	PURPOSE	INITIAL OPERATION
Con Ed Throium Reactor	1. U_3O_8 extruded in plastic, thorium plates; Al plates in light water lattices - up to and including full size core.	Determine nuclear parameters of thorium - uranium light water system in connection with the design and fabrication of CETR power core.	March 1957
	2. ThO_2-UO_2, SS clad fuel pins in light water lattices - up to and including full size core.		January 1958
N. S. SAVANNAH (NMSR)	Slightly enriched UO_2, SS fuel pins in light water lattices - up to & including full size core. (Core I)	1. Determine nuclear parameters of slightly enriched uranium light water system in connection with the design and fabrication of the N. S. Savannah power core.	January 1958
		2. Criticality testing of fabricated Core I prior to installation	November 1959
	Core II	Criticality testing of fabricated Core II prior to installation.	November 1960
Liquid Metal Fuel Reactor (LMFR)	Fully enriched uranium foils attached to bismuth bars, canned in graphite in a graphite, dry, bed-type assembly - full size core.	Determine nuclear parameters of a uranium bismuth fueled, graphite moderated system in connection with the anticipated design fabrication and operation of the LMFRE.	September 1958

B&W CRITICAL EXPERIMENTS EXPERIENCE (Cont'd)

CRITICAL EXPERIMENTS:	TYPE	PURPOSE	INITIAL OPERATION
ThO_2-UO_2 Light Water Critical Experiments (TUPE)	Fully enriched Al clad ThO_2-UO_2 fuel pins in several metal-to-water ratios in light water lattices.	Measure basic physical properties of ThO_2-UO_2-H_2O systems and compare them with calculational models.	August 1959
Lynchburg Test Reactor (LTR)	MTR Type fuel elements in water, internally and externally reflected by beryllium.	To construct a critical assembly having basically the same geometrical features as a typical LTR core which could be used to check the validity of the two dimensional nuclear calculations required for the LTR itself; and to establish the neutron flux levels available in the irradiation spaces and the extent to which they are perturbed by other typical in-pile test loops.	January 1960
Spectral Shift Reactor - Basic Physics Program (SSCR)	Slightly enriched UO_2 in SS; slightly enriched UO_2-ThO_2 in Al clad; varying percentages of D_2O in H_2O.	Determine reduction in reactivity through relative increase in fertile absorptions caused by shift in reactor spectrum toward higher energies. Determine nuclear parameters relative to fertile and fissionable materials, cladding, and D_2O-H_2O relationship.	October 1960

EXPONENTIAL EXPERIMENTS:

Con Ed Thorium Reactor	ThO_2, SS clad fuel pins in light water lattices - temperatures up to 500F.	Determine variation of nuclear parameters with temperature in connection with design and fabrication of CETR power core.	January 1959

B&W CRITICAL EXPERIMENTS EXPERIENCE (Cont'd)

CRITICAL EXPERIMENTS:	TYPE	PURPOSE	INITIAL OPERATION
N. S. SAVANNAH Core I	Slightly enriched uranium, SS fuel pins in light water lattices - temperature up to 500 F.	Determine variation of nuclear parameters with temperature in connection with design and fabrication of N. S. SAVANNAH power core.	May 1959
N. S. SAVANNAH Core II	Same	Same	October 1959
Spectral Shift Control Reactor - Basic Physics Program	Same as above under Critical Experiments.	Same as above under Critical Experiments. Run cold exponentials for high D_2O concentrations; run hot exponentials under simulated operating conditions.	January 1961
Age Measurements Resonance Integral Measurements, etc.	Thorium in water lattices.	Determine basic physics parameters of thorium system.	Continuous
U-238 Resonance Integral Experiments (Euratom)	U-238 metal and U-238O_2 in water lattices.	Determine the variations in the effective resonance integral for U-238 metal and U-238 O_2 as a function of sample diameter using the static reactivity technique and the pile oscillator technique.	February 1961

FACILITIES - B&W BOILER DIVISION

In addition to the above facilities, the Boiler Division operates the Nuclear Facilities Plant at the Mt. Athos location (520 acre site) five miles east of Lynchburg. The Nuclear Facilities Plant is the country's first privately-owned fuel element fabrication plant and has a total of about 70,000 sq. ft. of floor space. This plant is fully-equipped and is now producing both research and power reactor fuel elemts, complete cores, and certain auxiliary equipment.

FACILITIES - B&W BOILER DIVISION (CONT'D)

Also, B&W has extensive shop facilities at the Barberton Works, Barberton, Ohio, for fabrication of pressure vessels and heavy structural components. Significant facts are set forth below:

PHYSICAL PLANT

101.3 acre site; 1,479,786 ft^2 of building area, including shops devoted exclusively to atomic energy products such as a "clean shop" and temperature controlled areas. These features plus X-Ray and other non-destructive examination techniques provide a high degree of quality control.

CONVENTIONAL EQUIPMENT

Drums, sections, headers, tube bending and assembly, castings, hollow forgings and process equipment.

ATOMIC ENERGY EQUIPMENT

Heavy components for all types of reactor systems. The following components have been fabricated (or are in process) for pressurized water reactors alone in excess of:

30 reactor pressure vessels
40 nuclear steam generators
10 pressurizers
Refueling Equipment for N. S. Savannah

B&W EXPERIENCE

B&W is deeply engaged in nearly every aspect of nuclear power development. Activities extend from the production of complete power, propulsion, and research reactors of varying sizes to the development of experimental units and the fabrication of component parts and fuel elements.

As a leading manufacturer of boiler equipment, B&W has specialized since 1867 in controlling heat and pressure for useful purposes; its boilers provide steam for the first central power station in this country in 1881. B&W's long record of specialized achievement in metallurgy, heat transfer, fluid flow, and pressure containment led to its selection as a supplier of heavy pressure vessels and heat exchangers for the wartime Manhattan Project.

When controlled fission became a reality, B&W rapidly broadened its interest to encompass every aspect of nuclear reactor development and manufacture. Diversified projects for the U. S. Navy, the U. S. Atomic Energy Commission, the USAEC National Laboratories, the U. S. Maritime Administration, utility companies, industrial groups, and foreign interests have provided B&W an additional wealth of information and experience.

Recognizing the expanding needs of the nuclear field, B&W established a separate Atomic Energy Division in April 1953 to handle contracts previously entered into by other divisions of the Company, and to handle all future atomic energy contracts including the development of new and improved reactors and associated components.

Since obtaining its first contracts for nuclear equipment for the Manhattan Project, B&W has executed many and varied contracts, private studies, and laboratory development related to reactor systems and components. Some of this work concerning reactor plant construction projects, conceptual design and evaluation, and related reactor development is briefly summarized on the following pages.

MAJOR B&W REACTOR PROJECTS

Consolidated Edison Thorium Reactor (CETR)

Purchaser: Consolidated Edison Company; New York, New York

Location: Indian Point Station; Buchanan, New York

B&W has supplied the reactor equipment for a nuclear steam electric generating station rated at 275 MWe from the reactor and 112 MWe from two oilfired superheaters. The reactor is a pressurized water, internal thorium converter and is the first unit to make use of thorium as the fertile material to supplement the base fuel which is highly enriched U-235.

B&W has the responsibility for the design of the complete reactor system and fabrication of the major components; for developing functional specifications of auxiliary systems and supporting equipment; and for conducting all required research and development.

Design has been completed; critical experiments and zero power tests on the power core have been completed; major components have been shipped; and construction work at the site is 100 per cent complete. Initial operation is expected in early 1962.

Liquid Metal Fuel Reactor Experiment (LMFRE)

Purchaser: USAEC (NYOO); New York, New York

The USAEC selected B&W to design, fabricate, and operate a Liquid Metal Fuel Reactor Experiment. The scope of work covered the design, development, fabrication, erection, andoperation of prototypes that would lead to large central-station units.

The LMFRE was a graphite moderated, liquid metal cooled reactor containing the nuclear fuel in solution in the liquid metal. This first unit was to be operated at a reactor power level of 5 MW.

All of the experimental plant parameters were established, and the design entered the final stages preparatory to fabrication. A critical experiment was designed, built, and operated. Extensive research and development work was accomplished with respect to the behavior of the liquid metal and graphite combination. Hot loops were built and operated at the B&W Research Center in Alliance, Ohio. A 4" liquid bismuth utility test loop was operated at Brookhaven National Laboratory. Development of remote maintenance equipment was executed.

However, the USAEC curtailed this program in 1960 based upon an evaluation of fluid fuel reactors.

Nuclear Merchant Ship Reactor (NMSR) for N. S. SAVANNAH

Purchaser: USAEC (NYOO); New York, New York

B&W has supplied the complete reactor plant, refueling equipment, steam turbine plant and reduction gears for ship propulsion. The reactor plant is a pressurized water unit using low enriched UO_2 as fuel and moderated and cooled by H_2O.

The N. S. Savannah, a combination passenger-cargo ship, is designed as a single-screw vessel 587 feet overall in length, 78 feet in beam, with a draft of 29'6" and a displacement of 21,800 tons.

Normal shaft horse power is 20,000 at a reactor heat generation of 63 MW with a maximum continuous rating of 22,000 shp.

B&W has the contract to design, fabricate, and supervise the installation and the testing of the nuclear power plant. Design and necessary critical experiments have been completed. Major components have been fabricated and installed in the containment vessel. Sea trials began in late 1961.

Subsidiary N. S. Savannah Contracts

As a logical outgrowth of the B&W NMSR Contract the following related contracts have been awarded to B&W:

1. N. S. Savannah Crew Training Program

   In 1958 the U. S. Maritime Administration engaged B&W to train the crew required to operate the reactor for the N. S. Savannah. The crew-training program was executed by the B&W Atomic Energy Division using NMSR design personnel in the classroom activities.

2. Todd Shipyards Corporation for U. S. Maritime Administration

   In 1959 B&W performed as a design consultant to Todd relative to the service barge, Atomic Servant, which Todd has constructed for the N. S. Savannah. Design assistance was provided relative to waste disposal and shielding requirements.

3. N. S. Savannah Upgrading Program

   B&W contracted with the USAEC to provide engineering services including design specifications and evaluation of certain components of the N. S. Savannah power plant. This study will provide information as to how the N. S. Savannah could be modified or changed to become a more economical and efficient means of nuclear marine propulsion.

4. Other

In addition B&W executed contracts with the USAEC relative to critical experiments on the second core for the N. S. Savannah and preparation of a safety report. Further, B&W has provided consultation services covering reactor equipment installation and testing on board the N. S. Savannah. These services are being provided to the USAEC and New York Shipbuilding Corporation.

Enrico Fermi Fast Breeder Reactor (APDA Project)

B&W has contributed significant engineering and research effort to the Atomic Power Development Associates' Fast Breeder Reactor project since its inception in 1951. In addition, B&W fabricated and erected a one-through sodium-to-water steam generator for test purposes in this program. Developmental fuel pins were fabricated in preparation for the design and fabrication of a complete core for the Power Reactor Development Company's Enrico Fermi Plant in Monroe, Michigan.

Gas Suspension Coolant Development Project

B&W conducted an experimental and theoretical investigation of the properties of gas suspension fluids as reactor coolants. Both Company- and Atomic Energy Commission- sponsored programs were conducted. Heat transfer and pressure drop correlations were obtained; reactor application studies, pump test experiments and graphite lubrication, fission product retention and off-gas experiments were conducted. The program demonstrated the advantages of gas suspension coolants for reactor systems.

Spectral Shift Control Reactor (SSCR) Concept

Purchaser: USAEC (NYOO); New York, New York

B&W is now conducting a Basic Physics Program to determine the reduction in reactivity through relative increase in fertile absorptions caused by shift in reactor spectrum toward higher energies. Nuclear parameters are being calculated relative to fertile and fissionable materials, cladding, and the D_2O-H_2O relationship. The experimental program includes slightly enriched UO_2 in stainless steel clad fuel elements and slightly enriched UO_2-ThO_2 aluminum clad fuel elements. The moderator consists of varying percentages of D_2O in H_2O which results in changes in the reactor flux spectrum and in the consequent neutron absorption characteristics.

Associated with the above Basic Physics Program is an SSCR Plant Study Program which B&W is also executing. This program involves engineering studies and economic analyses relating to an SSCR Power Plant.

Advanced Test Reactor (ATR) - Architect-Engineering Services

Purchaser: USAEC (IOO); Idaho Falls, Idaho

Location: National Reactor Testing Station; Arco, Idaho

In late 1960 B&W was awarded a subcontract covering nuclear design and development (including critical experiments) relative to the reactor complex and loops for ATR. Essentially, B&W has taken the conceptual design of the reactor and prepared engineering information for construction requirements.

The ATR, when in operation in 1965, will be the largest existing nuclear testing facility (250 MWt) with thermal fluxes of 1.5×10^{15}.

Research Reactors

B&W has developed a series of open-pool type research reactors ranging in size from 10 KW to 5000 KW. The basic design includes an MTR-type aluminum-uranium alloy plate fuel element, light-water moderator and coolant, and a wide range of experimental equipment.

1. FORD NUCLEAR REACTOR

Location:	University of Michigan Ann Arbor
Fuel:	Fully enriched Uranium
Thermal Power Rating:	1000 KW (future 5000 KW)
Operating Date:	September 19, 1957

2. UNIVERSITY OF SAO PAULO REACTOR

Location:	Sao Paulo, Brazil
Fuel:	20% enriched Uranium
Thermal Power Rating:	5000 KW
Operating Date:	September 16, 1957

MAJOR B&W REACTOR PROJECTS (Cont'd)

*3. KERNENERGIE GESELLSCHAFT REACTOR

Location:	Hamburg, Germany
Fuel:	20% enriched Uranium
Thermal Power Rating	5000 KW
Operating Date:	October 23, 1958

4. LYNCHBURG POOL REACTOR - B&W OWNED

Location:	B&W Critical Experiment Laboratory Lynchburg, Virginia
Fuel:	Fully enriched Uranium
Thermal Power Rating:	200 KW
Operating Date:	September 19, 1958

*5. CENTRAL AGENCY MILITARY ENERGY NUCLEAR (C.A.M.E.N.)

Location:	Livorno, Italy
Fuel:	Fully enriched Uranium
Thermal Power Rating:	5000 KW
Operating Date:	1961

*The Kernenergie Reactor is the largest installation of its type in the world. The C.A.M.E.N. Reactor will rival this claim upon completion.

50-208 7-20-68
Supl.

ATTACHMENT III

EDUCATIONAL BACKGROUND AND EXPERIENCE
OF APPLICANT'S PERSONNEL
PRIMARILY RESPONSIBLE FOR
DESIGN, ERECTION AND OPERATION
OF SUBJECT FACILITY

7367

RALPH P. GRIMES - MANAGER, NEW FACILITIES DEPARTMENT

B.S. Mechanical Engineering, Pennsylvania State College 1924

Mr. Grimes has over 38 years of diversified experience with the Babcock and Wilcox Company. He is presently manager of the department responsible for erection and field service for all facilities built by the Atomic Energy Division.

In previous assignments Mr. Grimes has been manager of the Contract Department of the Atomic Energy Division. In this capacity he had responsibility for administration of all Atomic Energy Division contracts and the responsibilities of efforts of other departments in meeting contractual commitments. He had direct project management responsibility for the propulsion System for the Nuclear Merchant Ship Savannah.

In prior assignments he has had extensive experience as an erection and service engineer on erection, start-up, and maintenance of all types of B&W equipment.

Member: Society of Naval Architects & Marine Engineers

LAWRENCE G. BARRETT - PROJECT MANAGER AND TECHNICAL MANAGER, NUCLEAR DEVELOPMENT CENTER

B.S. Physics, Trinity College 1951
Graduate, Oak Ridge School of Reactor Technology 1952

Mr. Barrett, in his present assignment, is responsible for coordination of all design and construction of the Nuclear Development Center. In previous assignments, he was responsible for the construction and operation of the Lynchburg Pool Reactor. He also supervised the construction of the Hot Exponential Facility connected with this reactor. He has had charge of special physics experiments performed in this facility as well as other experimental physics work performed by B&W.

In previous employment, Mr. Barrett was responsible for reactor physics calculations for the Submarine Intermediate Reactor and for nuclear testing during start-up. He was a specialist on techniques for measurement of reactor parameters. Mr. Barrett has also had extensive experience in industrial electronics.

Member: American Nuclear Society

Papers: Classified Physics Conference - 1955 - "Power Coefficient Measurements in SIR Mark A by Pile Oscillator Techniques".

ANS - 1957 - "Dynamic Techniques for the Measurements of Power Reactor Coefficients" (Co-Author).

ANS - 1958 - "Special Features of the Lynchburg Pool Reactor".

WILBERT C. GUMPRICH - MANAGER, ENGINEERING DEPARTMENT

B.S. Mechanical Engineering, Massachusetts Institute of Technology 1939
M.S. Marine Engineering, Massachusetts Institute of Technology 1940

Mr. Gumprich has over 22 years experience in engineering design and administration for steam generating purposes. He is presently manager of the Engineering Department of the Atomic Energy Division. In this capacity he is responsible for all engineering functions required for design and procurement of nuclear reactor systems and equipment.

In previous assignments he has had administrative and direct engineering responsibilities in reactor functional analysis, heat transfer, and thermal and fluid analysis for nuclear systems. Mr. Gumprich spent four years in an assignment at Argonne National Laboratory where he was analytical group leader for the STR Program. He was later assigned to the A.P.D.A. Project where he was technical coordinator being responsible for all technical activities of the project. During this assignment, he also acted as test engineer for the land base proto-type of the STR. Earlier assignments in the Company were all connected with conventional marine and submarine boilers.

Member: American Nuclear Society
Atomic Industrial Forum
American Society of Mechanical Engineers
American Standards Association

MILTON C. EDLUND - MANAGER, MARKETING DEPARTMENT

B.S. Physics, University of Michigan 1948
M.S. Physics, University of Michigan 1948
Passed general exams for Ph.D. in Theoretical Physics, Princeton University, 1952

Mr. Edlund is Manager of our Marketing Department with overall responsibility for development of new concepts in reactor design, preliminary design engineering, and all program planning and marketing activities of the Division. Prior to this assignment he has held several other management level assignments in the area of physics and mathematics with the Division.

Prior to his employment with B&W, he was at Oak Ridge National Laboratory where he was a lecturer at ORSORT and chief of the Reactor Physics Section.

Member: American Physical Society
American Nuclear Society

Patents: Nuclear Reactor and Method of Making Same
Nuclear Reactor and Method of Operating Same
Neutronic Reactor

Papers: 1952 "The Elements of Nuclear Reactor Theory"
1954 "Physics of the HRT Statics - ORNL - 1780"
1958 "Physics of Water Moderated Thorium Reactors"
1960 "The Spectral Shift Control Reactor"
1956 "Some Conclusions from 'LMFR' Design Study"
1959 "Comparative Value of Bred Fuels"
1960 "Pricing Bred Reactor Fuel"
1954 "An Harmonics Method Applied to D_2O Moderated Reactors"
1958 "Spectral Shift Control"
1957 "Analysis of Borax Experiments"

WILLIAM M. BREAZEALE - SENIOR TECHNICAL ADVISOR

B.S. Electrical Engineering, Rutgers University 1929
M.S. Electrical Engineering, Vanderbilt University 1933
PhD. Physics, University of Virginia 1936

Dr. Breazeale, in his present capacity as Senior Technical Advisor of the Division, has the responsibility of keeping management informed on technical matters. He also acts as a consultant to customers in the area of reactor selection, operation, and control. He has over 30 years experience in engineering and is a recognized authority in the field of research reactors. He was the first person licensed by the AEC as a Reactor Operator.

Before joining B&W, Dr. Breazeale was Professor of Nuclear Engineering at Pennsylvania State University where he designed and built their Pool Reactor. While with the AEC, he made major contributions in the area of Reactor Design, Control, and Instrumentation. He has also taught Electrical Engineering on the college level.

Member: ANS (Founder)
IRE
SNAME

Papers: Author or Co-Author of four textbooks.
Over twenty published papers in the nuclear field.

B. A. MONG - TECHNICAL ADVISOR, ENGINEERING DEPARTMENT

B. S. Mathematics, University of Akron, 1945
M. S. Physics, University of Akron, 1947
Oak Ridge School of Reactor Technology, 1953

Mr. Mong is presently assigned as Technical Advisor to the Manager of the Engineering Department. During more than ten years of experience in the nuclear engineering field he has participated in stress analysis of thermal shields; stress analysis of pressure vessels and reactor design.

In prior assignments Mr. Mong has acquired extensive experience in design and specification of Marine boiler parts.

Member: American Nuclear Society

Papers: Co-author of Design Report of a 10 MW Boiling Homogeneous Reactor - Oak Ridge, ORSORT

Presented Homogeneous Research Reactor paper at Research Reactor Conference at Oak Ridge

Presented Research Reactor paper at Atomic Industrial Forum, San Francisco

Co-author of article on Reactor Pressure Vessels - Nucleonics

Presented Nuclear Fuel Elements paper at Philadelphia Navy Yard, Naval Research Center, Jonesville, Pa., and Naval Bureau of Aeronautics, Washington, D. C.

Presented Reactor Types paper at American Society of Metals Regional Meeting, University of Akron.

EARL E. SCHOESSOW - ASSISTANT MANAGER, ENGINEERING DEPARTMENT

B.S. Mechanical Engineering, North Dakota State College, 1931
Graduate Studies, Purdue University

Mr. Schoessow is presently Assistant Manager of the Engineering Department responsible for all Physical Engineering. He has over 19 years experience in mechanical engineering directly relating to steam generation and power plant equipment. He is presently responsible for the engineering layout of reactor plants, systems design, and final specifications of all hardware items. He is responsible for contract procurement and inspection. In previous assignments, Mr. Schoessow has had direct design responsibilities for pressure vessels and other process equipment design; and has been in charge of systems and structural engineering in the Company's nuclear activities for over twelve years.

Member: American Society of Mechanical Engineers

Patents: Holds over ten patents on pressure vessels and heat exchangers.

JOHN F. MUMM - ASSISTANT MANAGER, ENGINEERING DEPARTMENT

B. S. Mechanical Engineering, Illinois Institute of Technology 1948
M. S. Mechanical Engineering, Illinois Institute of Technology 1954

Mr. Mumm is presently assigned as Assistant Manager for Functional Engineering in the Engineering Department. He is responsible for the functional design of plant and components and for fuel element and nuclear core design. In previous assignments he was responsible for all nuclear, thermal, and control aspects of the Consolidated Edison Thorium Reactor, the Nuclear Merchant Ship Savannah Reactor, and several other proposals and studies.

Mr. Mumm's employment previous to joining B&W was as a heat transfer engineer at Argonne National Laboratory. He was assigned to the Engineering Analysis Section and in this assignment made core components studies on the STR and the Nautilus. He also worked on conceptual design of advanced power plants for submarines and other new reactor concepts.

Member: American Nuclear Society

Papers: Co-Author of several reports in the design and evaluation of reactor systems.

Author - Argonne National Laboratory's report on Boiling Heat Transfer Method for Calculating Plutonium Production.

HOWARD F. DOBEL - CHIEF, PLANT DESIGN SECTION

B.S. Mechanical Engineering, Columbia University 1945
M.S. Mechanical Engineering, Columbia University 1961

Mr. Dobel has over 16 years experience directly related to power plant design. He is presently assigned as Chief of the Plant Design Section of the Engineering Department. In this assignment, he has responsibility for all overall system designs, electrical and instrumentation designs, and for the engineering activities connected with outside procurement for the systems. In previous assignments, he has been responsible for conceptual designs of the nuclear system for the Consolidated Edison Thorium Reactor. He has supervised the development of specifications of the primary cooling pumps and gate and check valves for this plant. He assisted in the design fabrication operation and analysis of the first high pressure burnout studies conducted at NTR.

Mr. Dobel has also had experience in designing fossil fuel, boiler systems, and in R&D activities connected with improving these designs.

HARVEY R. ROCK - CHIEF, MECHANICAL DESIGN SECTION

B.S. Aeronautical Engineering, University of Michigan 1944
Graduate Studies, University of Tennessee and University of Virginia

Mr. Rock is presently the Chief of the Mechanical Design Section of the Engineering Department. In this assignment, he is responsible for the detailed mechanical design of fuel elements, control rods, control rod drives, and other equipment for nuclear systems. In prior assignments he has had engineering and supervisory responsibility for design activities on fuel elements, control rod drives, and reactor test loops.

Mr. Rock's nuclear experience prior to joining B&W includes a special assignment to Oak Ridge in connection with an indirect cycle aircraft nuclear propulsion program. This work included design and development of out-pile and in-pile test loops and associated high temperature equipment. Non-nuclear assignments were in the area of design and R&D of high speed, high temperature bearings, seals, and special equipment.

Patents: Savannah Internals and Core

MELVIN F. SANKOVICH - CHIEF, THERMAL ANALYSIS

B.S. Mechanical Engineering, Ohio University 1951
M.S. (Candidate) Nuclear Engineering, Massachusetts Institute of Technology 1959

Mr. Sankovich, in his assignment as Section Chief of the Thermal Analysis Section of the Engineering Department, is responsible for all basic programs in the fields of heat transfer and fluid flow.

In previous assignments with B&W, Mr. Sankovich has been responsible responsible for revising and evaluating new and advanced ideas in the design of nuclear power plants and their components. He has also been responsible for design of fuel elements and design of test loops and facilities to confirm fuel element designs and operational characteristics.

Member: American Nuclear Society

Papers: Co-Author - "Thermal Buckling of Cruciform Control Rods" - Nuclear Science and Engineering, April 1959

Patent: Nuclear Merchant Ship Reactor Core Design

CARL E. THOMAS - CHIEF, OPERATIONAL ANALYSIS

B.S. Engineering Physics, University of Chattanooga 1954
Graduate Oak Ridge School of Reactor Technology 1955
Additional studies, University of Virginia, 1957

Mr. Thomas now has over seven years of experience directly related to the nuclear field. In his present assignment as Chief of the Operational Analysis Section of the Engineering Department, he is responsible for studies in reactor dynamics and control, performance analysis, and safeguards analysis. This group is responsible for establishing the basic operational plan for all reactor systems.

In prior assignments with B&W, Mr. Thomas supervised a study team that evaluated the breeding potential of the Liquidated Metal Fuel Reactor. He supervised the physics work on the LMFRE Project and had responsibility for reactor physics, reactor design, shielding, instrumentation, and associated research and development programs.

Member: American Physical Society

Papers: Author - The Liquid Metal Fuel Reactor - presented to joint meeting of Japan-U.S. Atomic Industrial Forum, Tokyo, Japan; May 1957

Co-author - The Effect of Slurry in Homogeneity on Nuclear Stability of Liquid Metal Fuel Reactors - presented to ANS meeting; December 1958

Co-author - A High Temperature Slurry Integral Reactor for Low Cost Power - presented to ANS meeting; October 1959

Author - The Liquid Metal Fuel Breeder Reactor - presented to the Physics of Breeding Conference, ANL; October 1959

RALPH A. WEBB, JR. - CHIEF, NUCLEAR ANALYSIS SECTION

B.S. Electrical Engineering, Tufts University 1955
M.S. Electrical Engineering, University of South Carolina 1955
Additional work, Johns Hopkins University

Mr. Webb, in his assignment as Section Chief of the Nuclear Analysis Section of the Engineering Department, supervises the physics of all reactor projects and establishes the methods and standards for all physics calculations. This section is also responsible for nuclear design of reactor systems.

Prior to joining B&W, Mr. Webb was chief physicist and physics group leader for other nuclear contractors. He has also had experience in starting and calibrating test reactors.

Member: American Nuclear Society

LOUIS R. WEISSERT - CHIEF, MATERIALS & CHEMISTRY SECTION

B.S. Mechanical Engineering, Stevens Institute of Technology 1951

Mr. Weissert, in his assignment as Chief of the Materials & Chemistry Section, is responsible for the development, testing, and specification of all materials. This Section is also responsible for developing and evaluating all irradiation testing programs. They develop fabrication methods and quality control for both nuclear and non-nuclear systems.

In previous assignments with B&W, Mr. Weissert has assisted in setting up a plant for fuel element manufacturing and developing fabrication methods for fuel elements. He has acted as consultant to Oak Ridge National Laboratory on heavy equipment fabrication and has supervised fabrication of components for nuclear reactors.

Member:	American Society of Mechanical Engineers American Nuclear Society (Charter) American Society of Testing and Materials
Papers:	American Nuclear Society - 1959 - "Fuel Element Manufacture" Mineral Industrial Journal - 1957 - "Fuel Element Manufacture"

GRANVILLE M. OLDS - SERVICE ENGINEER

B.S. Chemical Engineering, University of Oklahoma 1950
M.S. Chemical Engineering, University of Oklahoma 1951

Mr. Olds has over five years' experience in erection of nuclear equipment. This experience includes major responsibilities in the installation of the propulsion equipment for the N.S. SAVANNAH, as well as establishing and following test procedures. He was also Engineer-in-charge of a 4" Utility Liquid Metal Test Loop at BNL, as well as short assignments on other nuclear projects. Prior to this assignment, Mr. Olds was responsible for the start-up of over sixteen fossil fuel steam generation units.

Prior to joining B&W, Mr. Olds had six years' experience as a shift supervisor on operation and maintenance of a Hanford type reactor.

JOHN J. CONTARINO - CONSTRUCTION SUPERVISOR, NUCLEAR DEVELOPMENT CENTER

B.S. Civil Engineering, Purdue University 1950
Sales Engineering, City College of New York 1951

Mr. Contarino is the Engineer in charge of construction of the Nuclear Development Center. He has over twelve years' experience in engineering directly related to site construction work. Prior experience has included such major jobs as Erector for the Nuclear Facilities Plant, Resident Engineer on an office building project, Erector on nuclear submarine components, and Erector on several fossil fueled boiler projects.

Experience prior to joining B&W was also in the field of power plant installation.

DONALD A. ROSS - SERVICE ENGINEER

B.S. Mechanical Engineering, Clarkson College 1955
Graduate Studies, Nuclear Engineering, University of Michigan and North Carolina State College

Mr. Ross has had six years' experience in the checking, start-up and servicing of research and power reactors. He was a B&W Service representative during erection, testing and start-up of the power plant of the N.S. SAVANNAH. He participated in the training program for operating personnel for this reactor. Mr. Ross has participated in the start-up of several research reactors, the latest being the CAMEN reactor in Liverno, Italy.

Prior to joining B&W, Mr. Ross was employed in health physics work at Savannah River.

Member: American Society of Mechanical Engineering
American Nuclear Society
Pi Tau Sigma

ATTACHMENT V

FINANCIAL

INFORMATION

FINANCIAL INFORMATION

In compliance with the requirement in Section 50:33 that each application contain a statement regarding the Applicant's financial qualifications to engage in the proposed activities and to assume responsibility for the payment of Commission charges for special nuclear material, Applicant has attached its annual report for the calendar year 1961 as Attachment VI.

50-200
70-68
-Suppl.

From:

James O. Trudeau, Manager
Public Relations Services
Babcock & Wilcox
161 E. 42nd St., N.Y.C.
Phone: MU 7-6700
Home Phone: Norwalk, Conn., VO 6-6340

FOR IMMEDIATE RELEASE

Wednesday, April 25, 1962

BABCOCK & WILCOX REPORTS
HIGHER FIRST QUARTER EARNINGS
AT ANNUAL STOCKHOLDER MEETING

NEW YORK --- The Babcock & Wilcox Company reported today that 1962 first quarter earnings were approximately 12 per cent higher than those recorded in 1961.

President M. Nielsen told stockholders at the annual meeting here that income after taxes and minority interest was $5,266,000 or 85 cents per share. This compares with $4,720,000 or 76 cents per share earned in the first quarter of 1961.

Consolidated sales (shipments) in the period were $83,183,000 compared with $76,087,000 in the first quarter of 1961, an increase of approximately 9 per cent.

New orders totaled $80,825,000 and were more than 26 per cent higher than those received in the same quarter last year. The backlog of $330,669,000 at the end of the first quarter was about the same as it was at the beginning of the year, but was approximately 19 per cent higher than at the end of the same period in 1961.

Commenting on B&W's prospects for all of 1962, Mr. Nielsen noted that first quarter results "undoubtedly reflect some business which was placed with us at a time when it was uncertain as to whether or not there would be a steel strike. In other words, we have received and shipped orders in the first quarter that we normally would not have received and processed until later. We do not know the extent to which this will affect business during the remainder of the year.

"I would also like to caution you," Mr. Nielsen stated, "as I have in the past that in our type of business, with its mixture of products with long and short term deliveries, it is not possible to estimate the current year's total bookings, shipments and earnings simply by multiplying the first quarter by four. You are well aware, I am sure, that many other factors will have an effect on final performance for the year.

"While we fully realize that every year cannot continue to be the best, it appears that 1962 will be another good year from both management's and stockholder's points of view," he remarked.

Mr. Nielsen told the meeting, which was held in the Hotel Biltmore, that all divisions and subsidiaries, except the Atomic Energy division, are operating profitably.

"In atomic energy," he said, "we have been able to continue scientific and technical progress while substantially reducing our former large losses. Considering the type of pioneering operation which this is, we regard it as a good investment for the future."

(MORE)

Although applications of atomic energy are coming along, he emphasized that Babcock & Wilcox is "by no means" relaxing its efforts to improve the use of fossil fuels in steam generation.

He pointed out that as a result of its research and development work in that area, the Boiler division is now ready to market such equipment as a Cyclone Furnace to burn coal slurry from pipelines without drying, a coal pulverizer that will handle up to 50 tons of coal per hour which is more than twice the capacity of present B&W units and a burner that can fire three times as much coal, oil or gas per hour as previous B&W designs.

Noting that Babcock & Wilcox is engaged in the largest capital investment program in its history, Mr. Nielsen indicated that company funds will be adequate for that purpose without additional financing.

THE BABCOCK & WILCOX COMPANY, and subsidiaries, report for three months ended March 31, 1962, compared with the corresponding periods for 1961 and 1960:

	March 31 1962	March 31 1961	March 31 1960
Sales (Consolidated Shipments)	$83,183,000	$76,087,000	$78,412,000
Income (Before taxes on income and income applicable to minority interest)	11,451,000	10,192,000	10,222,000
Net Income	5,266,000	4,720,000	4,981,000
Earned per share (based on 6,183,313 shares issued)	$0.85	$0.76	$0.81

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# # #

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PRELIMINARY HAZARDS SUMMARY REPORT

Babcock & Wilcox Test Reactor

July, 1962

THE BABCOCK & WILCOX COMPANY

ATOMIC ENERGY DIVISION

BAW-183

PRELIMINARY HAZARDS SUMMARY REPORT

Babcock & Wilcox Test Reactor

July, 1962

Submitted to
THE UNITED STATES ATOMIC ENERGY COMMISSION
By

THE BABCOCK & WILCOX COMPANY
Atomic Energy Division
Lynchburg, Virginia

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1. INTRODUCTION

The Babcock & Wilcox Company (B&W) plans to construct and operate a Nuclear Test Reactor on the Company's nuclear facilities site at Mt. Athos near Lynchburg, Virginia. This preliminary Hazards Summary Report presents to the Atomic Energy Commission (AEC), Division of Licensing and Regulation, a description of the Nuclear Development Center, the characteristics of the proposed site, and an evaluation of the potential hazards associated with operating the test reactor at this location.

The Nuclear Development Center consists of a Nuclear Fuel Laboratory (NFL), the B&W Test Reactor (BAWTR), and B&W's Critical Experiment Laboratory (CEL). The CEL has operated since 1956; the NFL and BAWTR are sections of a new facility (hereinafter referred to as the Facility) yet to be built. The NFL will include a hot cell, Radioactive Materials Examination Laboratories, a Recycle Fuel Pilot Plant Area, a Machine Shop and Assembly Area, and a Feed Materials Building. The new Facility will develop and test improved nuclear fuel elements and establish methods for their fabrication. A large variety of fuel materials, including plutonium and U^{233} will be handled in the NFL hot cell.

Besides being used to conduct B&W fuel development programs, the Facility will also be available to others for irradiation space and the use of the associated laboratories. The development of economic nuclear power depends on the testing of various fuel element designs under reactor operating conditions, such as temperature, pressure, and power density. This information can be obtained only from actual operating experience, such as in a reactor loop experiment.

The main function of the NFL is to develop nuclear fuels and fuel elements as well as methods for their fabrication through the pilot stage.

The Radioactive Materials Laboratory section of the NFL provides a facility to prepare and examine specimens before and after irradiation.

In addition, metallurgical and radiochemical facilities will enable the performance of specialized experiments to evaluate further the pre- and post-irradiation characteristics of the test samples.

The BAWTR is a 6-MW pool-type reactor utilizing ETR-type fuel elements arrayed around a central thimble. A high-temperature water-cooled loop is inserted in this thimble to irradiate nuclear fuel under actual operating conditions. Other thimbles in the reflector are available to enable capsule-type irradiations of fuel samples and other materials. Though the reactor is designed for a one-loop, 6-MW initial operation, it may be extended to a two-loop, 12-MW operation at a later date. To allow for this possible expansion at this location, the hazards evaluations are made for the higher power level. The present application is to obtain a construction permit and an operating license for the reactor with a one-loop, 6-MW operation. When it is decided to increase the power, B&W will apply to the AEC Division of Licensing and Regulations to extend the operating power and the experimental capability.

The Nuclear Facilities Plant (NFP), which is located at the site, produces fuel elements, test specimens, and related component parts that will be utilized by the test reactor. The present CEL and the computing operation can be used advantageously to set up test programs and to evaluate safeguards and performance characteristics of the facility.

Since the main office of the B&W Atomic Energy Division is located 10 miles by highway from the site, the entire technical staff of the Division will be available for consultation on all matters relating to construction, safety, and operation of the proposed Facility. Figure 1.1 is an artist's conception of the Facility. Figures 1.2 and 1.3 are aerial photographs of the site.

Figure 1. 1. Artist Conception of New Facility

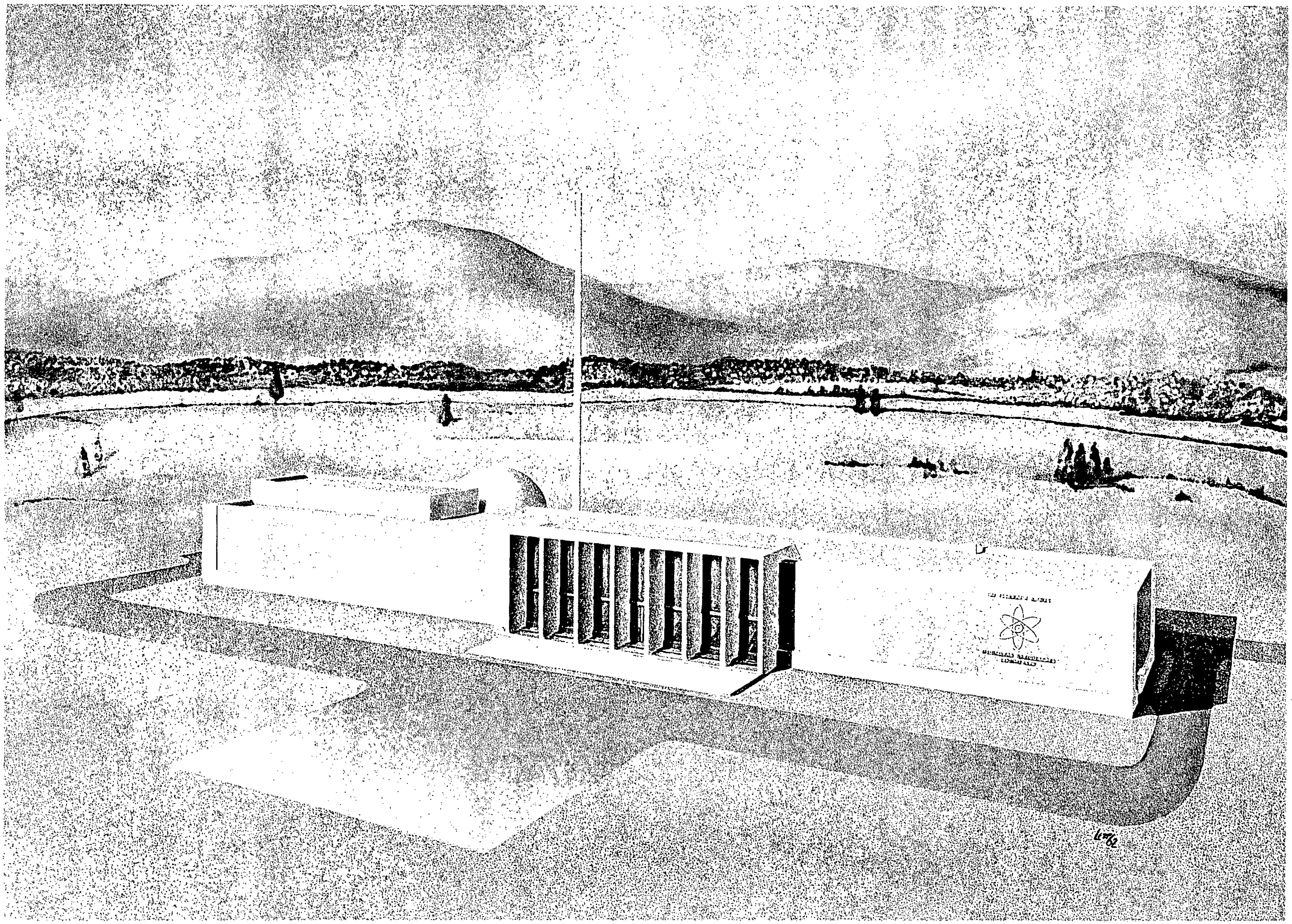


Figure 1.2. Aerial View of Mt. Athos Site Showing CEL
(Looking North)

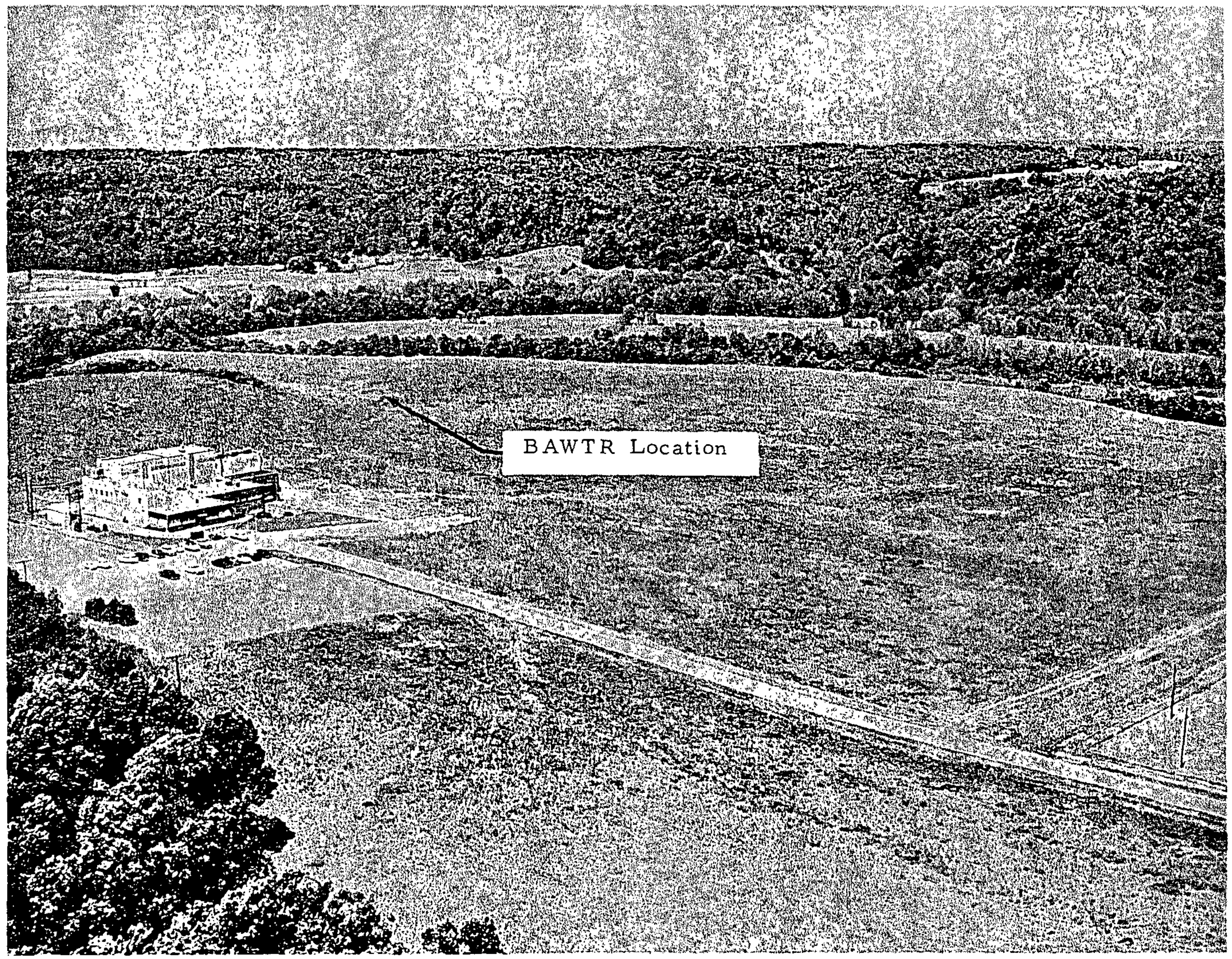
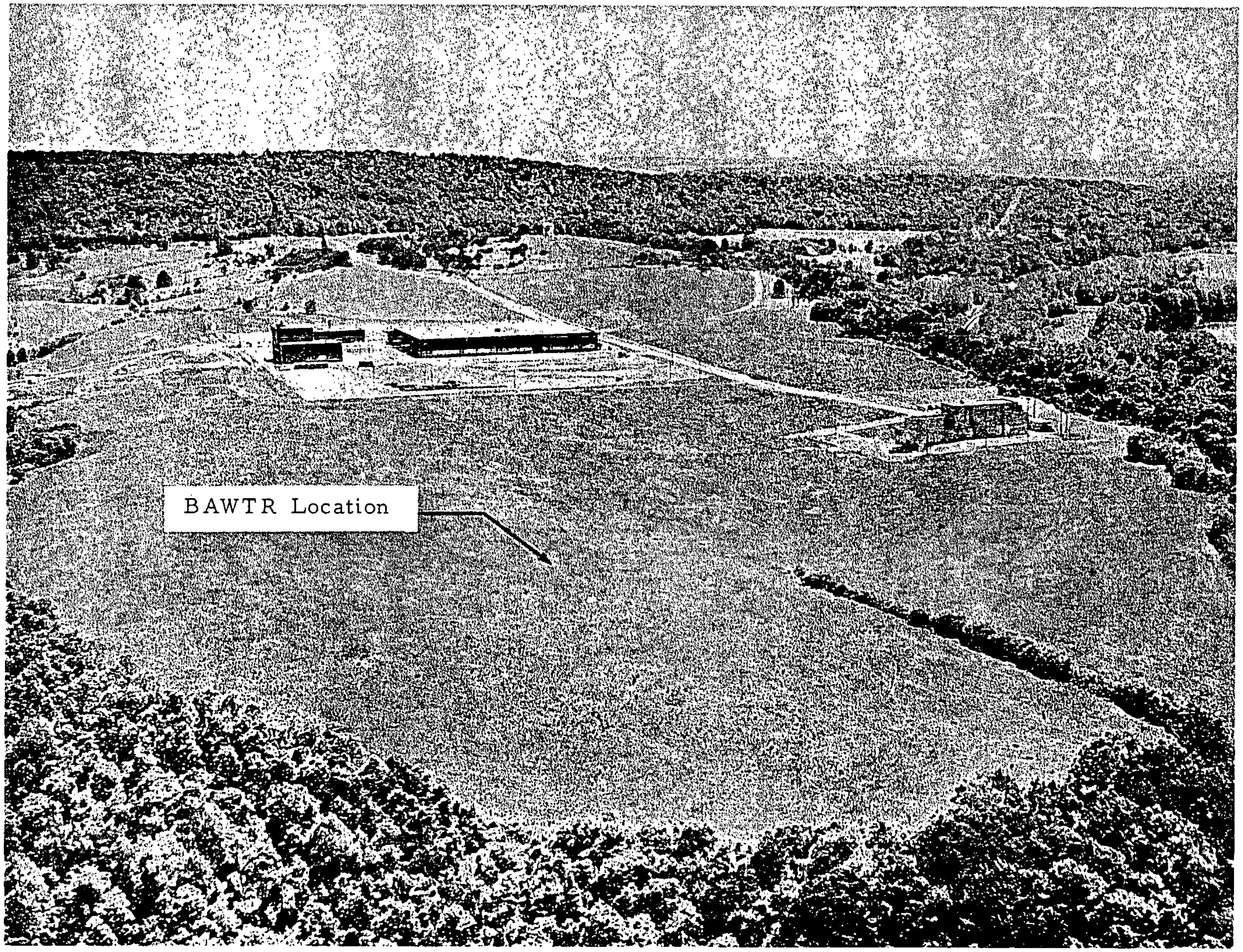


Figure 1.3. Aerial View of Mt. Athos Site Showing CEL and NFP (Looking Southeast)



2. SITE

2.1. Location

The Facility site is in Campbell County, Virginia about 4.25 air miles east of downtown Lynchburg, the largest population center in the area. The site is a 536-acre plateau bounded by the James River on three sides and Mt. Athos on the south and southwest. Figure 2.1 is a contour map of the area showing the site and its location with reference to Lynchburg. The nearest city limit boundary is 3 miles from the site.

The NFP and the CEL are also located on this site. The NFP is about 1700 feet southeast of the BAWTR and the CEL is about 500 feet south. Figure 2.2, the Mt. Athos site, shows the relation of the BAWTR to the other facilities on the B&W property and the site boundaries. The nearest distance to the river is about 900 feet south of the reactor. Since the river is not navigable at this location and is not used for recreational purposes, an additional 400 feet of exclusion distance is provided by the width of the river. This total distance of 1300 feet to the nearest off-site property is considered as the minimum exclusion distance for the BAWTR.

Other locations on the property were evaluated as possible sites for the reactor. However after careful consideration, this particular location was selected for the following reasons:

1. In case of an accident or release of activity at the NDC requiring the evacuation of the CEL and NFP, personnel could be evacuated in a direction away from the Facility.

2. The direction of the prevailing winds for the area are away from the other facilities of the site and the surrounding populated areas.

3. It is the highest spot on the site with enough space for a facility of this size.

4. It provides the greatest direct distance of any usable location to the NFP and the site boundary.

5. Drainage from the selected location is not directly into the river and requires a long period of time to reach the river. Other locations considered have surface drainage into the swamps near the river or directly into the river and the holdup time is considerably shorter.

The tracks of the James River Branch of the Chesapeake and Ohio Railroad follow the river around the site. The property and right-of-way on which the railroad tracks are located are owned by the Chesapeake and Ohio Railroad. At the closest approach, the tracks are about 820 feet west of the reactor; 1.6 miles of track are within a distance of 0.5 miles or less. Traffic on this line averages one train per hour, and there is no passenger service. It is possible for the railway traffic through the area to be rerouted to avoid passage through the site in case it is necessary. The emergency procedures for the Facility will require that the Chesapeake and Ohio Railroad be notified in case of an accident at the site. The railroad can then take the necessary steps to prevent its trains from crossing the site until radiation levels are reduced to acceptable levels.

U. S. Highway 460 passes about 2 miles to the south of the proposed location.

Future development and usage plans for the site are not known at the present time. It is likely that the fuel fabrication plant and the NDC will require some future expansion to keep up with the fuel requirements of the predicted reactor usage for this country. Any facilities added at the site will necessarily be a part of the nuclear development complex and, as such, will be under B&W supervision and control. These facilities would operate under the same emergency procedures and warning systems as the present and planned facilities for the site.

2. 2. General Characteristics

2. 2. 1. Topography

The proposed site lies in the valley of the James River on the eastern edge of the Blue Ridge Mountains. The general terrain is hilly with sheltered valleys. The Blue Ridge Mountains have peaks ranging as high as 4,000 feet. Mt. Athos, which borders the site on the southeast, has a peak of approximately 860 feet (see Fig. 2. 2).

2. 2. 2. Meteorology

2. 2. 2. 1. Weather Stations

Although only a limited amount of meteorological data has been collected at the proposed site, weather stations have been maintained in the Lynchburg area for more than 80 years. Originally, the weather station was located in downtown Lynchburg about 4.5 miles from the proposed reactor site. In November 1936, the present U. S. Weather Bureau station was established at the Preston Glenn Airport, 9.5 miles from the site of the proposed reactor. Instrument elevations at the airport vary from 950 to 1000 feet. Between the airport and the reactor site there is a range of mountains (Candler) with an average elevation of 1100 to 1200 feet.

2. 2. 2. 2. General Climatological Information

Weather in the Lynchburg area is influenced by two major air masses: the polar continental air mass and the gulf air maritime mass. The polar continental air mass, usually cold and dry, originates over the polar ice cap and northern Canada and predominates during the winter months. In the summer months, the usually warm, moist gulf maritime air mass, which originates near the equator, either moves into this area after sweeping up along the Atlantic seacoast or comes across the Appalachian Mountains after sweeping northward up the Mississippi Valley. These air masses explain the predominance of northwesterly and southwesterly winds in the Lynchburg area.

2. 2. 2. 3. Local Climatological Information

Because of its sheltered location east of the eastern ridge of the Blue Ridge Mountains, Lynchburg has a moderate climate. The average temperature is 38.9 F for the winter months and 75.4 F for the summer months. Precipitation throughout the year reaches a maximum in the summer months as a result of the gulf air mass. Most winter precipitation is rain although there are occasional snows.

Precipitation

Based on data collected at the U. S. Weather Station in Lynchburg from 1906 to 1961, 41 inches of rain, occurring in 124 days, may be expected in the Lynchburg area each year. Distributed by season, the record in inches is as follows:

Winter	10.0
Spring	10.6
Summer	11.7
Fall	8.8

Monthly averages for the period are given in Table 2-1. During the 55-year period, the maximum precipitation for one month was 14.87 inches, and the maximum for one 24-hour period was 7.59 inches (August 1928).

Table 2-1. Average Monthly Rainfalls

Lynchburg, Virginia
1906 - 1961

Month		Rainfall, in.
January		3. 37
February		2. 99
March		3. 59
April		3. 14
May		3. 45
June		3. 93
July		4. 08
August		4. 18
September		3. 35
October		3. 02
November		2. 54
December		3. 14
	Total	40. 78

Based on a record of 68 years, about 14 inches of snow may be expected annually. Snowfalls exceeding 1 inch may occur on four days of each year. The greatest recorded snowfall occurred in January 1922 when 22.7 inches fell in one month and 16.4 inches fell in one 24-hour period.

Wind Speed and Direction

Wind data were taken from the records of the U. S. Weather Bureau at Preston Glenn Airport and from observations made at the B&W CEL and the Mead Corporation Plant in Lynchburg. The Mead Corporation Plant is located on the James River just east of downtown Lynchburg between Lynchburg and the NDC site. The observations at the Mead Corporation Plant were taken every 15 minutes for about 1 year. The data indicate the direction from which the wind was blowing. Wind roses constructed from these data are shown in Figures 2. 3, 2. 4, and 2. 5.

The U. S. Weather Bureau data are a composite covering the years of 1914 through 1933 at downtown Lynchburg and 1936 through 1950 at the Preston Glenn Airport. Because the terrain between these sites is hilly, these data may not apply directly to the proposed test reactor site. However, data taken at the CEL from April 27, 1959 to April 26, 1962 indicate, as do the U. S. Weather Bureau and Mead Corporation observations, that winds in the Lynchburg area are predominantly from the west, the northwest, and the southwest. Since the data taken at the CEL is based on observations made during the daytime hours only, it may not be a true representation of the winds at the site. These observations were also made at variable hours throughout the day and only during the five working days each week. The information is used here only as an indication of general wind direction and velocities.

Table 2-2, which shows the frequency of surface wind directions at the Preston Glenn Airport Weather Station for the period of May 1953 to May 1958, indicates that winds may be expected from the east and southeast 8.85% of the time. Calms occurred 1.4% of the time.

Table 2-3, which shows the frequency of surface wind velocities by seasons for the period of January 1953 through December 1953, indicates that calms or light air prevailed 15.8% of the time, light or gentle breezes 68.9% of the time, and winds of greater velocity 15.3% of the time.

Generally, summer winds are from the south, southwest, and west; winter winds are from the west and northwest. For 37 years, the recorded wind velocity averaged 7.6 mph. The maximum recorded velocity of 62 mph occurred during two successive years, 1916 and 1917. Over long periods, winds of less than 4 mph are expected to occur 10 to 12% of the time.

Table 2-2. Hourly Surface Wind Observations

Preston Glenn Airport
May 1953 - May 1958

Direction	Frequency, %
N	14.45
NE	12.35
E	4.15
SE	4.70
S	18.2
SW	20.65
W	12.3
NW	11.8
Calm	1.4

Table 2-3. Frequency of Hourly Surface Wind Observations in Various Speed Groups

Preston Glenn Airport
January 1953 - December 1953

	Velocity, mph				
	0 to 3	4 to 12	13 to 24	25+	Average
			Frequency, %		
Winter	10.0	67.9	21.9	0.2	9.4
Spring	17.4	64.4	17.9	0.3	8.3
Summer	19.4	74.5	6.0	0.1	7.1
Fall	16.4	68.9	14.5	0.2	7.9
Annual average	15.8	68.9	15.1	0.2	8.2

Since wind directions during precipitation have not been recorded, it is assumed here that wind directions under this condition are the same as those recorded. Any prediction concerning washouts of air-borne contaminants must be a general statement based on this assumption.

Fog

Over a period of 43 years, heavy fogs (visibility less than 0.5 mile) occurred 21 days per year in the Lynchburg area. In 1953, there were 42 days during which fog occurred. It is estimated that only 2% of the fogs extends beyond mid-morning or noon.

Visibility

Table 2-4 gives the frequency of visibilities observed at the Preston Glenn Airport from January 1953 to January 1954.

Table 2-4. Frequency of Hourly Visibility Observations

Preston Glenn Airport
January 1953 - January 1954

Visibility, miles	Frequency, %
0-0.125	1.06
0.1875-0.375	0.87
0.5-0.75	1.13
1-2.5	2.59
3-6	11.71
7-15	61.93
20-30	10.14
35+	10.57

Ceiling

Table 2-5 gives the frequency of ceiling heights based on hourly observations at the Preston Glenn Airport from January 1953 to January 1954.

Table 2-5. Frequency of Hourly Ceiling Observations

Preston Glenn Airport
January 1953 - January 1954

Ceiling, ft	Frequency, %
0	0.42
100-200	2.48
300-400	1.75
500-900	2.49
1000-1900	2.67
2000-2900	2.69
3000-4900	5.67
5000-9500	8.10
9500+	73.73

2.2.2.4. Additional Meteorological Information

A large amount of meteorological data has been compiled for Lynchburg and the Preston Glenn Airport over a long period of time. The relatively small amount of data compiled at the site indicates a small variation in meteorological conditions from those listed previously. However, to establish more exact conditions at the site and to determine good values for stability parameters, diffusion coefficients, and wind velocities, a weather station will be located near the new facility. This station will consist of a tower with one or more locations where meteorological measurements will be taken. These values will be used to establish the necessary constants for future hazards evaluation studies. Besides furnishing basic meteorological data for the site, the station will also supply information for making controlled releases of radioactive gases from the stack and surrounding area if this type of operation is ever needed.

2.2.3. Geology

The bedrock formations in the area of the proposed test reactor consist chiefly of a thick mass of phyllite and quartzite. The phyllite, which makes up 95 to 97% of all the exposed rock along the bounds of the Mt. Athos site, is a dense, dark gray slaty rock with a submetallic sheen composed almost entirely of clay, fine silt, and finely divided mica flakes.

There is a continual slope from the facility locations to the river. This slope, steep for the first few hundred feet near the facilities, becomes more gradual nearer the river banks.

The reference elevation for the contour lines in Figure 2-1 is sea level. The contour lines for Figure 2-6, made from a private survey for B&W, uses as a reference elevation a concrete bench mark about midway between the CEL and the NFP. This point gives an elevation of 180.36 feet, about 567.91 feet above sea level as measured on the USGS contour map. (See Fig. 2-1.)

The phyllite bedrock exposures are of significance since they dip at steep angles (inclined to the southeast) and trend almost unvaryingly north 25 to 30 degrees east. Thus, both surface and substance drainage away from the plant site will not be westward toward the James River. The steep bluff rising above the railroad on the west side of the bend is bedrock which rises to the surface along the entire length of the meander neck. This rib of bedrock will deflect percolating influent seepage in an eastward direction.

The following quote is from a report by B. N. Cooper of Virginia Polytechnic Institute who studied the geology of the Mt. Athos site.

> "In summary, the writer wishes to reiterate his belief that no observed geological conditions are at all adverse to development of the plant site for the industrial purposes of the Company. Indeed, the site because of the geological conditions is almost ideal for control of effluent water, for building foundations, and in all other essential respects."

2.2.4. Hydrology

2.2.4.1. Surface Water Hydrology

The site of the proposed test reactor lies on the James River between two U. S. Geological Survey stream gaging stations. One station is located at Holcomb Rock, upstream of the site, the other is at Bent Creek, 17.5 miles downstream of the site. The water stage recorder at Bent Creek is 381.39 feet above sea level, at latitude 37 deg 32 minutes, longitude 78 deg 50 minutes. Records are available for the period of March 1925 to September 1954.

Figures 2.7 and 2.8 were prepared from the records covering the period of 1926 to 1953. Figure 2.7 shows the monthly mean discharges in cubic feet per second as a function of the frequency of observations less than a certain magnitude. In a normal distribution of sufficiently long record, the plot should be a straight line. The absence of the straight plot in the lower portion of the observation (500 cfs versus 4.4%) indicates a disturbing influence caused by irregular river flow at power plants and other dams above the gaging station. Besides regulating the flow of surface water, existing dams serve as a flood deterrent, although none of the dams were designed for flood control. (See Table 2-6.)

Table 2-6. Principal Dams in the James River Basin Above Lynchburg, Virginia

Location	River	Miles above the Lynchburg dam	Principal use
Reusens	James	3.45	Power
Holcomb Rock	James	11.90	Power
Colemans Falls	James	13.0	-
Big Island	James	18.37	Power
Bedford	James	21.58	Power
Cushaw	James	22.75	Power
Balcony Falls	James	27.60	Power
Lynchburg Reservoir	Pedlar	27.90	Water supply
Buena Vista	Maury	38.67	Power
Lyle	James	76.00	-
Covington	Jackson	106.82	Power

Although the U. S. Corps of Engineers has proposed additional dams above Lynchburg, their status is indefinite at present. The erection of more dams could change the runoff characteristics of the upper James River Basin but should not affect the water supply for the NDC.

Figure 2. 8, showing the recurrence intervals of annual flood peaks and the minimum daily discharges at Bent Creek for the years of 1926 to 1953, indicates that a flood peak of 115,000 cfs may occur one day in every 30 years. Such floods would not affect the NDC since the reactor location is at an elevation of 580 feet above sea level, approximately 100 feet above the normal level of the river.

Figure 2. 8 also indicates that the expected minimum flow is approximately 220 cfs, as recorded on October 31, 1931. A curve extension of this minimum flow indicates that the "bottom" has just about been reached so it is safe to conclude that the minimum river flow will not be less than 200 cfs at the site. This minimum flow is a factor of 100 greater than the NDC requirements. River water is used at the NDC only for cooling tower makeup, process water

makeup, and possibly for potable water makeup.

The nearest downstream dam of importance, Bosher Dam, about 130 river miles below Lynchburg, is used for diversion and power.

There are no major recreational areas downstream of the proposed site. Although individual game fishing is prevalent, commercial fishing is impossible in the James River for a long distance downstream from the site.

2.2.4.2. Ground Water Hydrology

Precipitation or a spill of water on the ground either percolates down through the soil, evaporates, or flows along the surface to streams.

Water that permeates the soil is intercepted by the roots of vegetation or continues downward to the zone of saturation, the upper surface of which is the water table. When the water reaches the zone of saturation, it creates a high point, or point of recharge, on the water table. Low points on the water table, such as springs, seeps, or wells, are points of discharge.

Ground water moves from points of recharge to points of discharge through interstices in the unconsolidated material and through fractures in the consolidated rocks.

Prominent along the western edge of the track immediately east of the wooded section shown in Figure 2.9 is a ridge of phyllite that runs from northwest to southeast and divides the surface water drainage. The consolidated material underlying the ridge generally slopes toward the river in a northeast direction. Ground water probably follows along the top of the consolidated material in the northeasterly direction. However, some of the ground water may follow the surface drainage pattern and flows southeast under the slight cut running almost the length of the site. (See Fig. 2.9). The consolidated rocks restrict the downward flow of ground water.

Small seepages along the Chesapeake and Ohio Railroad track on the site side of the track indicate the top of the ground water table at these points. The vertical distance from the seepages to the reactor is about 65 to 70 feet. The location of the seepages confirms the general direction of ground water flow.

In choosing a site for an installation like the BAWTR, the potential danger of contaminating surface and ground water with radioactive materials must be considered. Although contamination is controlled under normal conditions, standard operating regulations do not fully consider accidents or disasters. The course of travel of accidental radioactive spillage on the ground are postulated to follow the paths described in the following paragraph.

During periods of high precipitation or low temperature (frozen ground), the soil would not absorb all of the spill and some of the contaminant could flow over the surface as runoff. Generally, this runoff would be carried by natural drainage northwest or southeast toward the James River. However, two man-made obstacles prevent the immediate discharge of the runoff to the river: the road bed of the single-line railroad track and the abandoned James River and Kanawaha canal.

If the radioactive contaminant percolates into the soil, the probability of contaminating public or private water supplies would be remote since radioactive elements in the fluid probably would be absorbed in clay particles or removed by an ion exchange process.

If the radioactive fluid reaches ground water, it probably would move down the water table gradient. As previously indicated, the top of the ground water table includes the estimated water elevations in wells and the seepages along the railroad right-of-way. Contaminant discharge at these seepages would follow the course outlined for surface runoff, but the time delay between the accident and the appearance of contaminated seepage would be considerable.

Contaminated ground water not discharged as seepage probably would continue down the gradient and eventually discharge into the river since the time delay between the accident and this occurrence is long. The possibility of badly contaminating public or private ground water supplies is remote.

Any radioactive contaminants entering the surface water at the proposed site would follow one of several paths. When reasonably large activities are involved, the buildup of activity in aquatic animals and plants is possible. Concentration of activity in silts, sludges, slimes, and other bottom deposits also occurs. From past experience it appears unlikely that contaminants will be diluted and carried long distances.

2.2.4.3. Domestic Water Supplies Downstream

The nearest downstream community taking its domestic water supply from the James River is Scottsville, whose population was 396 in 1950. (See Fig. 2.10.) The water supply intake is approximately 60 miles downstream of Nine Mile Bridge. (See Fig. 2.9.) The U. S. Geological Survey maintains a gaging station nearby.

The Scottsville water treatment system consists of an intake, a small coagulation basin, a rapid sand filter, and a clearwell. Since the clearwell is small, a second reservoir on a nearby hill is needed.

In the event of a radioactive spill at the proposed site and in the unlikely event that the contaminated water reaches the river, domestic water consumers at Scottsville would be protected by several mechanisms:

1. Concentration of contaminants in aquatic animal and plant life.
2. Concentration of contaminants in sludges and slimes.
3. Dilution of contaminated fluid.
4. Treatment at the Scottsville plant.

The large volume of river water available for dilution makes it unlikely that the use of surface water downstream of the proposed site would be hazardous even in the unlikely event of contaminated water spillage.

2.2.5. Seismology

Table 2-7 summarizes the earthquake history of the Lynchburg area. The probability of an earthquake resulting in conditions too severe to be handled by emergency plans is remote.

Table 2-7. Earthquakes in Virginia

1833 - 1947

Date	Locality	Intensity, Rossi-Forel Scale	Area affected, sq mi
Summer, 1833	Central Virginia	Severe	-
Dec. 22, 1875	Arvonia	7	50,000
Jan. 2, 1885	Maryland and Virginia	4-5	3,500
Oct. 9, 1885	Variety Mills, Nelson County	6	20,000
May 3, 1897	Pulaski	7	150,000
May 31, 1897	Giles County	8-9	280,000
Oct. 21, 1897	Wytheville	5-6	20,000
Dec. 18, 1897	Ashland	5-6	7,500
Feb. 13, 1899	Lynchburg	5	30,000
Feb. 11, 1907	Arvonia	5-6	2,000
Aug. 23, 1908	Powhatan	5	450
Apr. 2, 1909	Virginia and West Virginia Maryland and Pennsylvania	5-6	2,500
May 8, 1910	Arvonia	5	350
Apr. 9, 1918	Luray	8	100,000
Sept. 15, 1919	Front Royal	6	-
July 15, 1921	Mevdota	5	-
June 10, 1927	New Canton and Arvonia	7	2,500
Dec. 26, 1929	Charlottesville	5-6	-

The following is a definition of the Rossi-Forel Scale.

1. Microseismic shock. Recorded by one or two seismographs of a given model.
2. Extremely feeble shock. Felt by a small number of persons at rest.
3. Very feeble shock. Felt by many persons at rest.
4. Feeble shock. Felt by persons in motion. Disturbance of movable objects, such as door and windows. May cause cracks in ceiling.
5. Moderate shock. Felt by everyone. Disturbance of furniture, beds, etc. Ringing of some bells.
6. Fairly strong shock. Awakening of those asleep. General ringing of bells. Oscillation of chandeliers. Stopping of clocks.
7. Strong shock. Overthrow of movable objects. Falling of plaster. Ringing of church bells. General panic. No damage to buildings.
8. Very strong shock. Falling of chimneys. Cracks in walls of buildings.
9. Extremely strong shock. Partial or total destruction of some buildings.
10. Shock of extreme intensity. Great diaster. Ruins. Disturbance of strata. Fissures in ground. Rocks fall from mountains.

2.3. Population Density

The largest center of population in the area surrounding the NDC site is Lynchburg, Virginia, approximately 4.25 air miles due west. To the east (the direction of prevailing winds), the population is predominantly rural. The nearest large concentration of population east of the site is Farmville, Virginia, about 40 miles away. Figure 2.11 and Tables 2.8 and 2.9 illustrate the rural character of the population east of the site.

An actual population count for the area was made in April 1962 by B&W. It includes all the area from the site out to a distance of 3 miles, the closest edge of the city limits of Lynchburg. This population study was made by individuals who actually counted the houses in each section. The number of homes was then multiplied by the average persons per family (reference 1960 census) to obtain the population density. The results of this survey are shown in Table 2.10.

Table 2-8. Inhabited Centers East of the NDC Site

Community	County	1950 Population	Direction from NDC	Miles from NDC
Amherst	Amherst	1,038	N	12
Appomattox	Appomattox	1,094	ESE	13
Buckingham	Buckingham	264	ENE	29
Farmville	Prince Edward	4,293 (1960)	ESE	33
Lovingston	Nelson	400 (estimated)	NNE	25
Pamplin City	Appomattox	370	SE	23
Scottsville	Albermarle	396	NE	41

Table 2-9. Population of Counties Around the NDC Site

County	Area, sq mi	Population in 1960	Average population density, persons/sq mi
Amherst	467	22,953	48
Appomattox	343	9,148	26
Buckingham	576	10,877	19
Campbell	530	32,958	62
Charlotte	478	13,368	28
Nelson	468	12,752	27
Prince Edward	357	14,121	39
City of Lynchburg	23	54,790	2,300

Table 2-10. Population Density

	Quadrant				
Distance, miles	NE	NW	SW	SE	Total
0 to 0.5	0	0	0	0	0
0.5 to 1	7	11	47	7	72
1 to 2	107	164	33	55	359
2 to 3	157	2300	602	226	3285
0.5 to 3	271	2475	682	288	2716

2.4. Conclusion

The considerations of site selection and evaluation for a nuclear test reactor are basically the same as those used to determine the proper characteristics of sites and plants for other industrial operations involving hazards. The prime difference lies in the more complex analysis of potential hazards and the intense public concern over the potential risk associated with radioactive materials.

This section has described in some detail the topography, meteorology, geology, hydrology, seismology, and culture (population density) of the region surrounding the NDC site. In each case, it is observed that the physical characteristics of the site are favorable for minimizing the risk associated with the operation of a nuclear test reactor. The gently rolling hills surrounding the site reduce the probability of any channeling of effluent from the stack which discharges above the peaks of most of the surrounding hills. Also, the major center of population in the area, Lynchburg, Virginia, lies in a direction opposite that of the prevailing winds. Any release of gaseous effluent probably would be carried over a region characterized largely by wooded growth or sparsely populated farmland. The geology and hydrology of the site provide an adequate water supply at all times and minimize the risks associated with an accidental spill of contaminated water. The history of this region is virtually free of earthquakes and severe storms.

Based on the physical characteristics of the NDC site and the absence of inherent physical disadvantages, it is concluded that the Mt. Athos site is suitable for the construction of the NDC.

Figure 2. 1. Topographic Map of Lynchburg Area

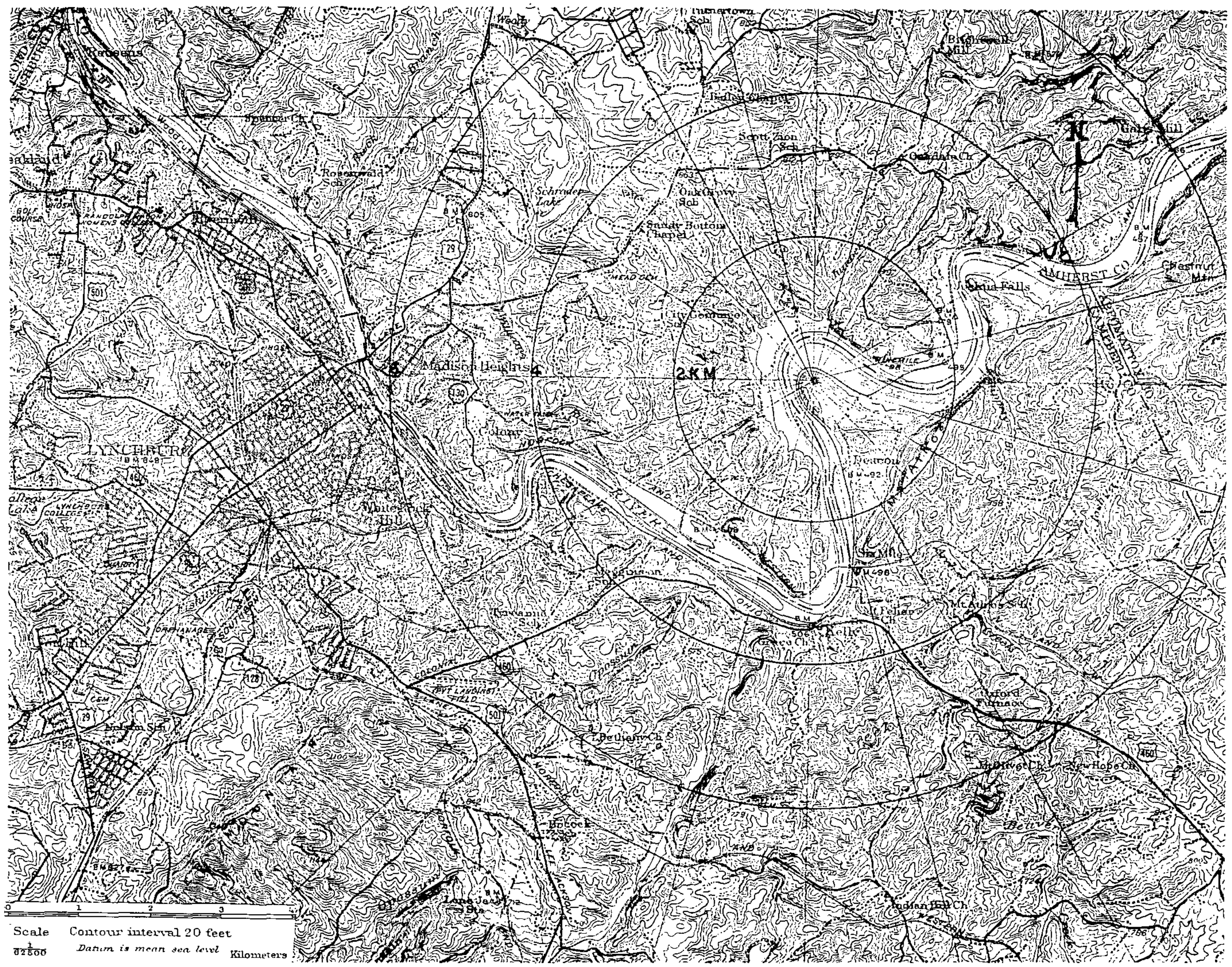


Figure 2. 2. Layout of the Mt. Athos Site

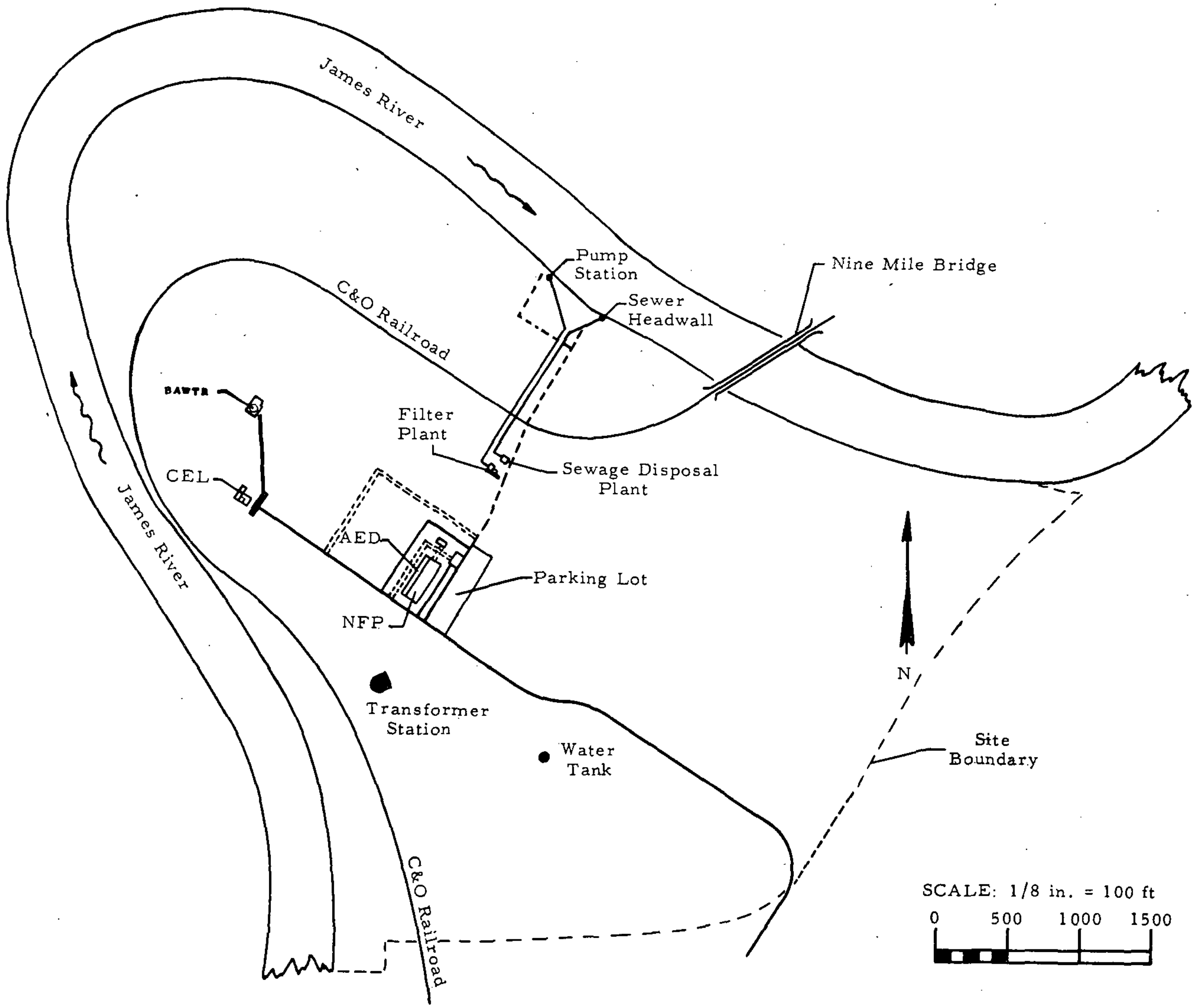
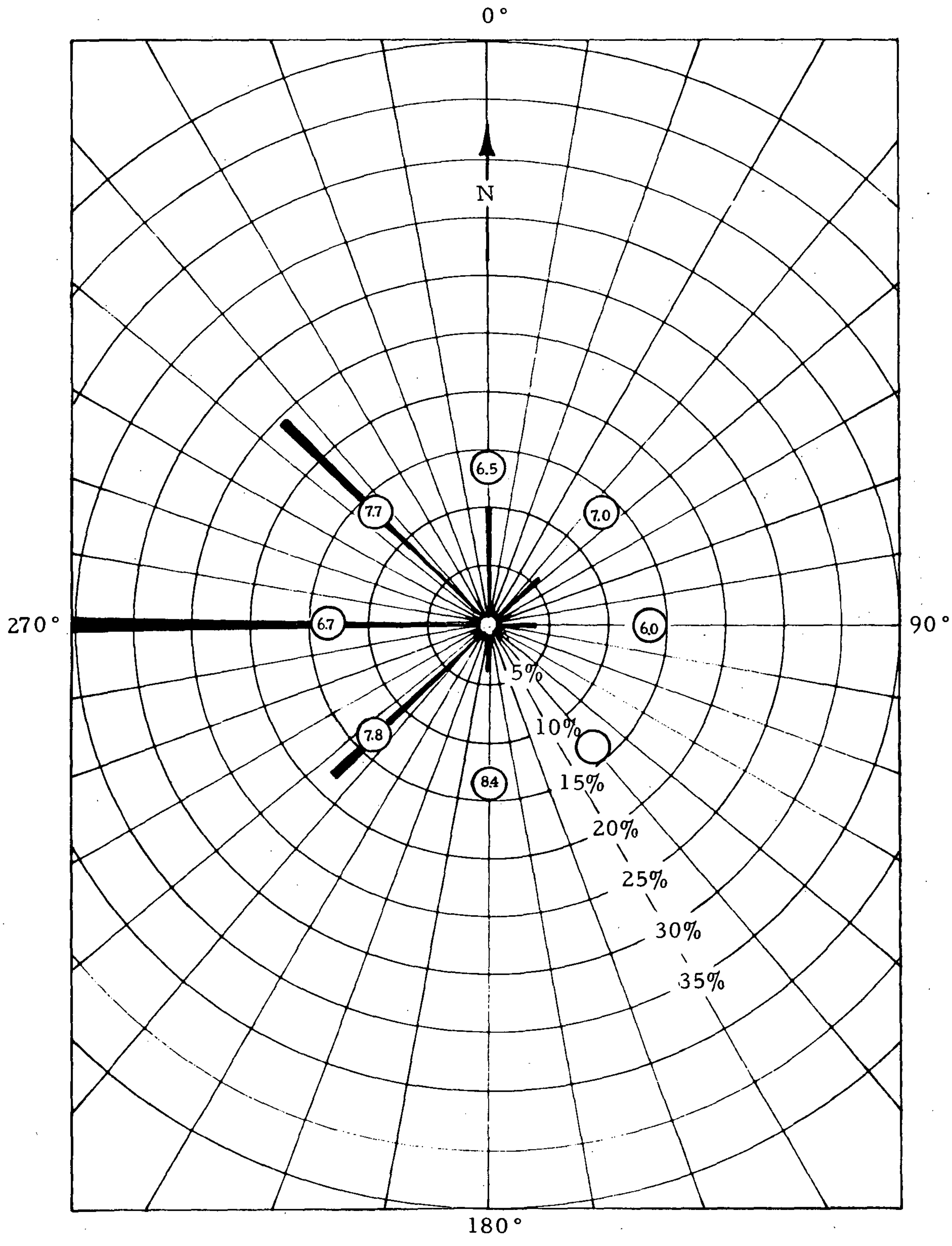
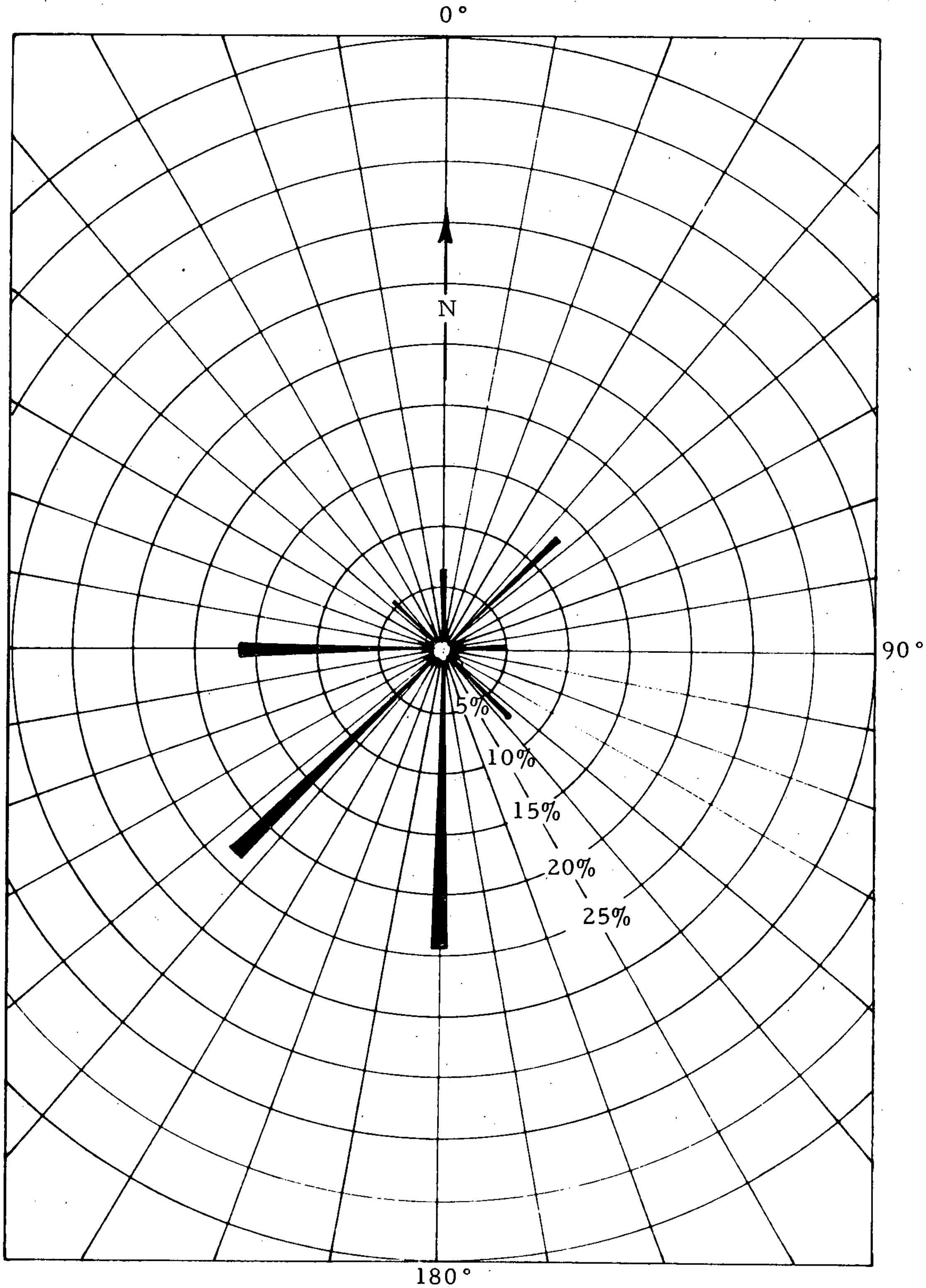


Figure 2. 3. Wind Rose for the City of Lynchburg



NOTE: Length of Bar Represents Probability of Wind Blowing from that Direction. Numbers in Circles Represent Average Wind Velocity in mph.

Figure 2.4. Wind Rose for the Critical Experiment Laboratory



NOTE: Length of Bar Represents Probability of Wind Blowing from that Direction. Numbers in Circles Represent Average Wind Velocity in mph.

Figure 2. 5. Wind Rose for the Mead Corporation
(James River Basin, Lynchburg, Virginia)

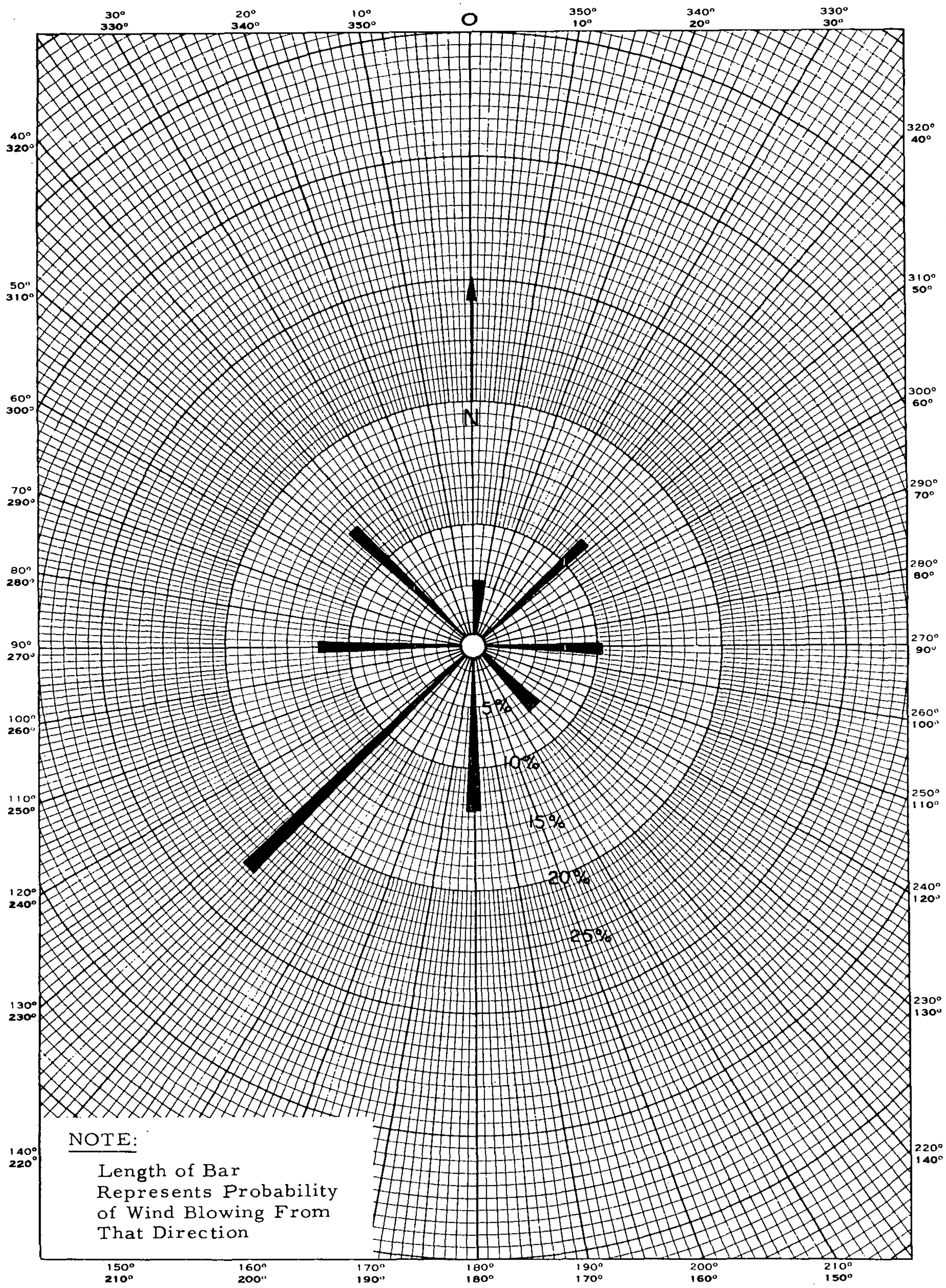


Figure 2.6. Topographic Map of B&W Area

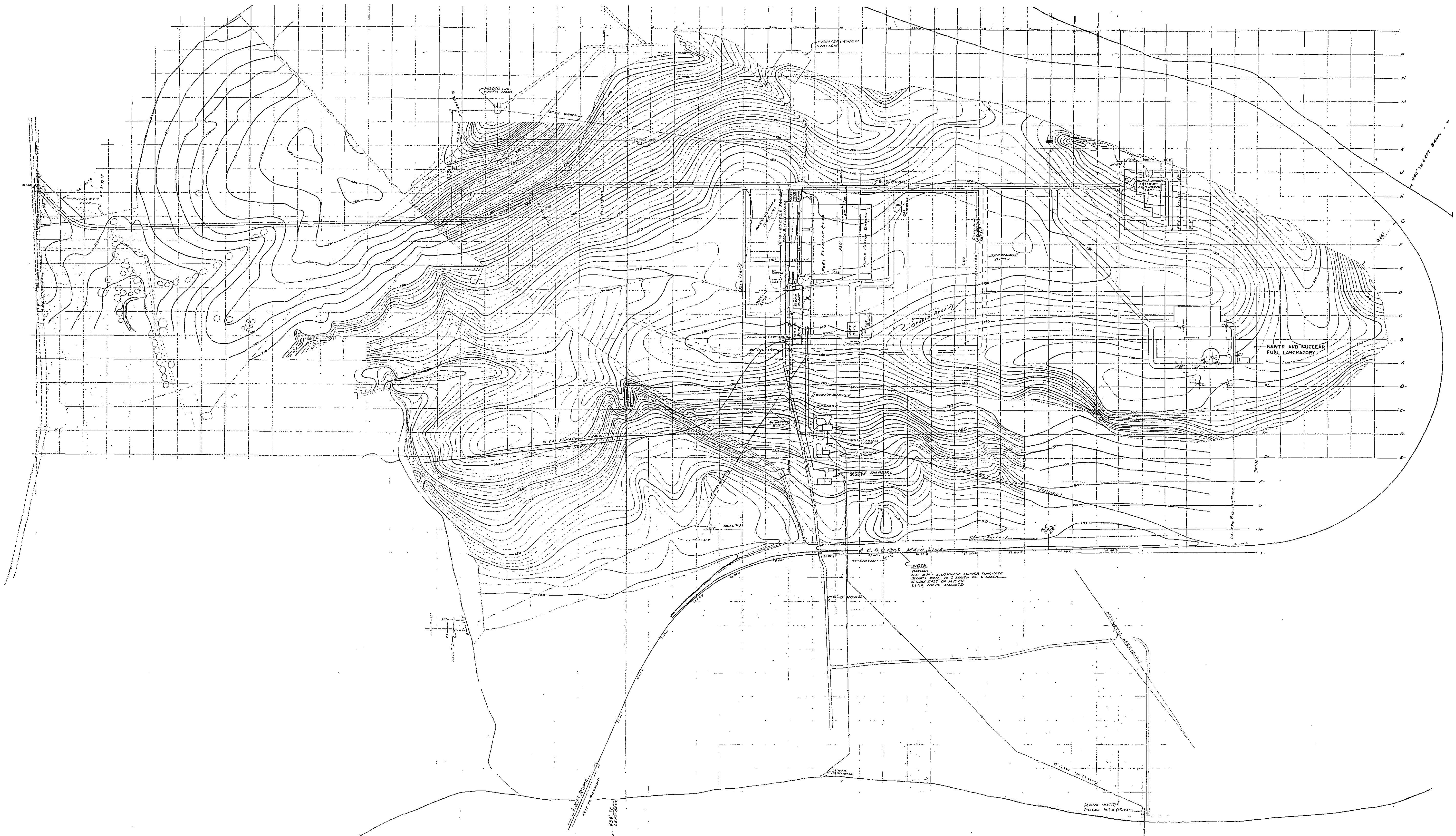


Figure 2.7. Runoff as Mean Monthly Discharge
(James River at Bent Creek, Virginia, 1926-1953)

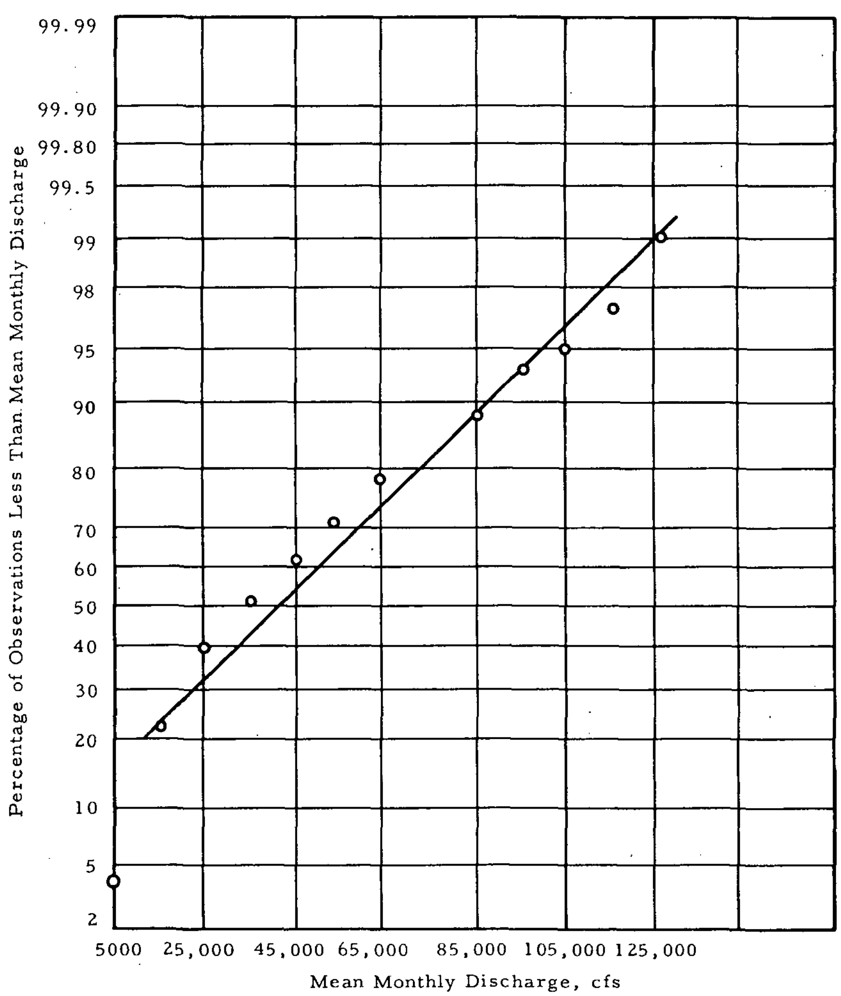


Figure 2.8. Annual Floods and Minimum Flow on the James River at Bent Creek, Virginia
(Drainage Area: 3671 sq mile Period, 1962-1953)

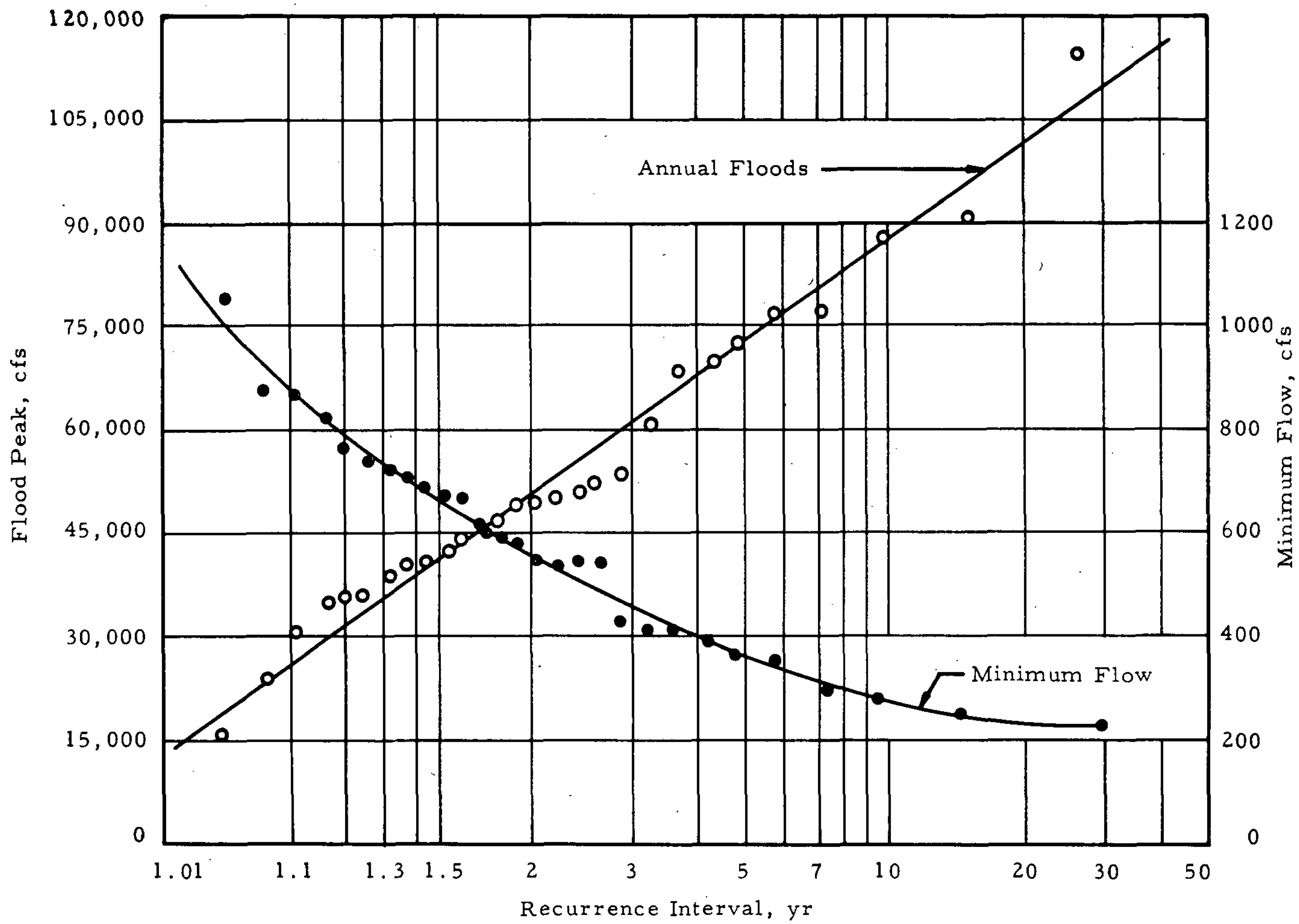


Figure 2. 9. Surface Drainage Pattern of BAWTR Site

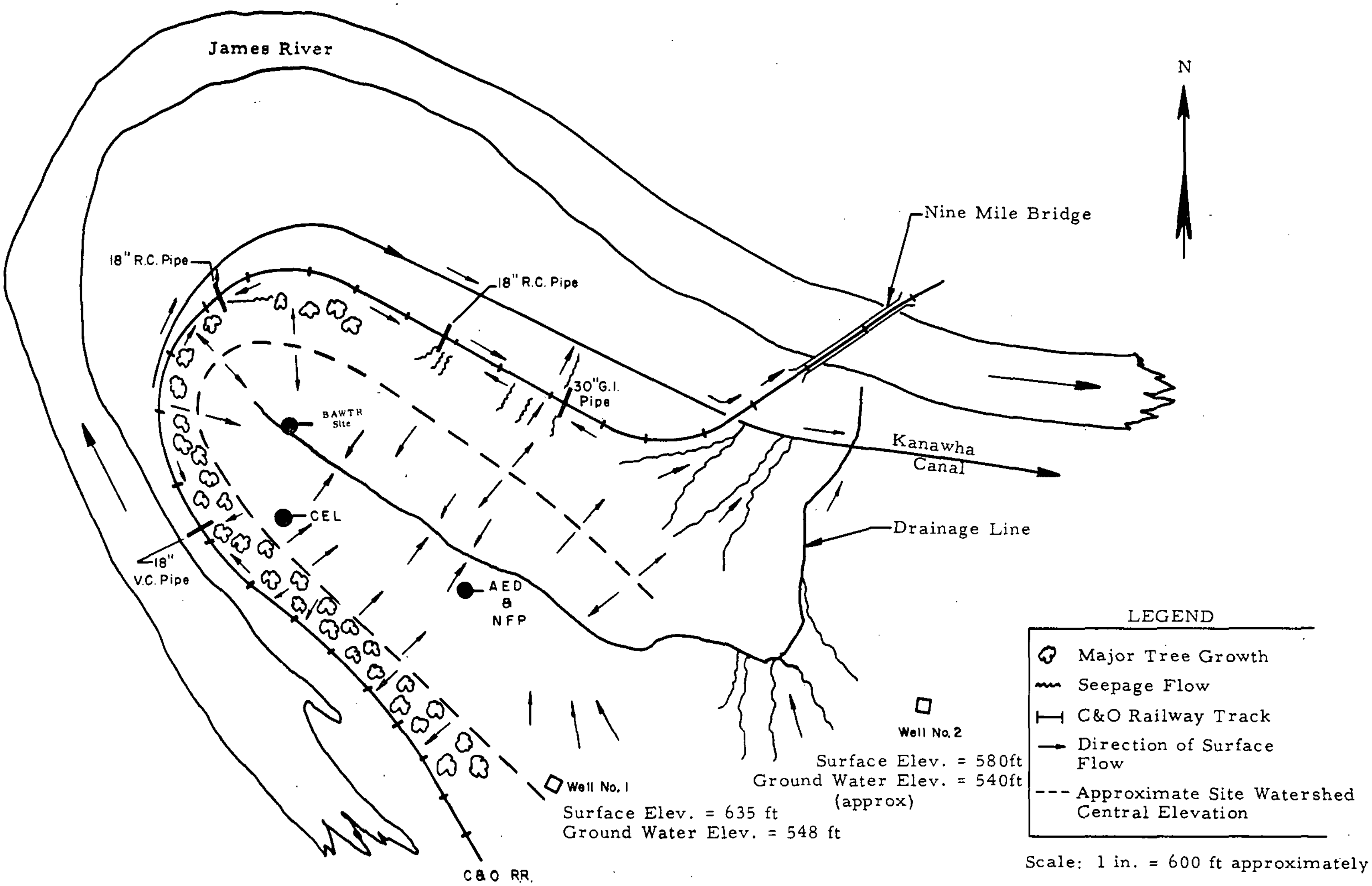


Figure 2.10. Populated Areas Cost of BAWTR Site

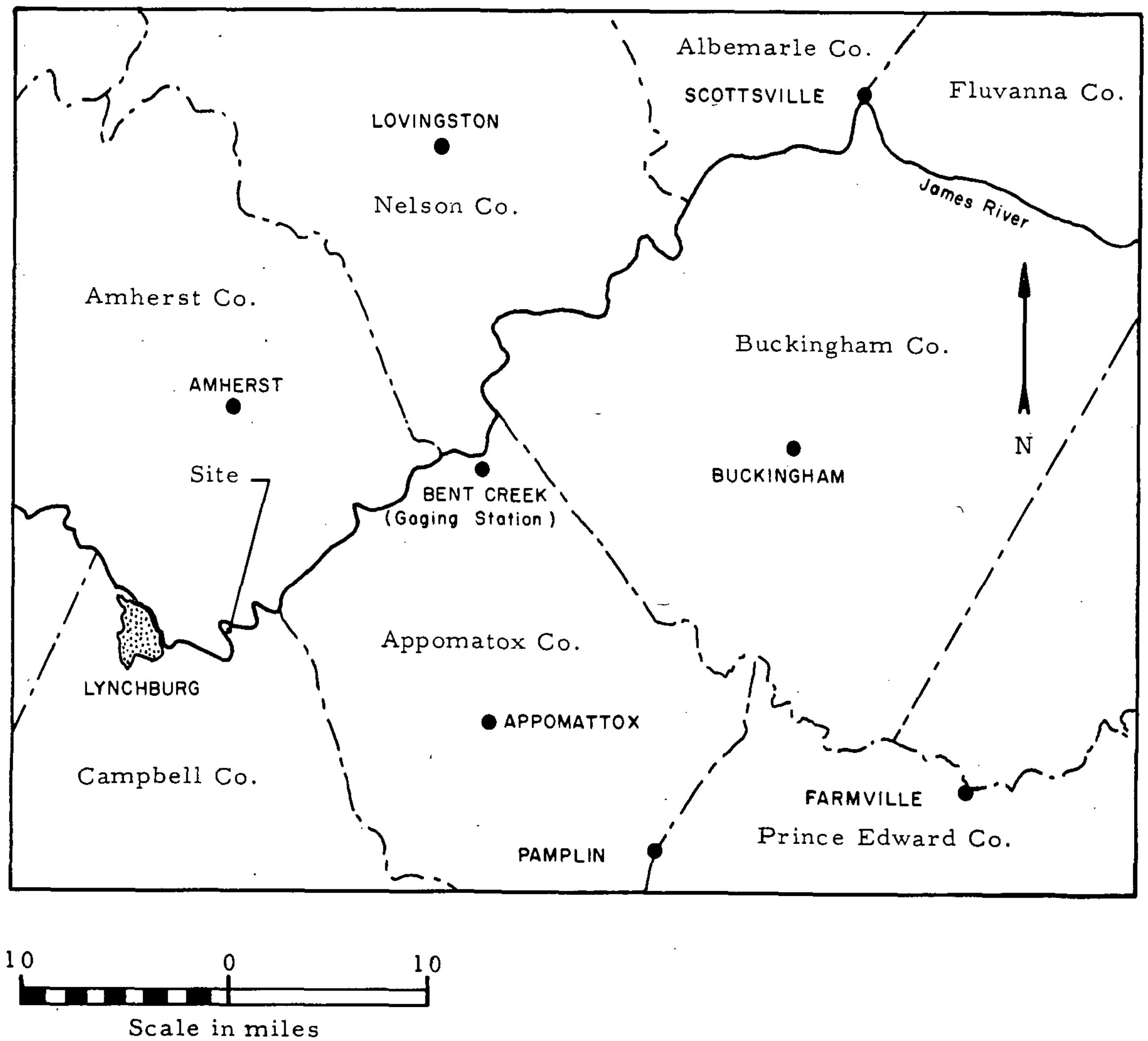
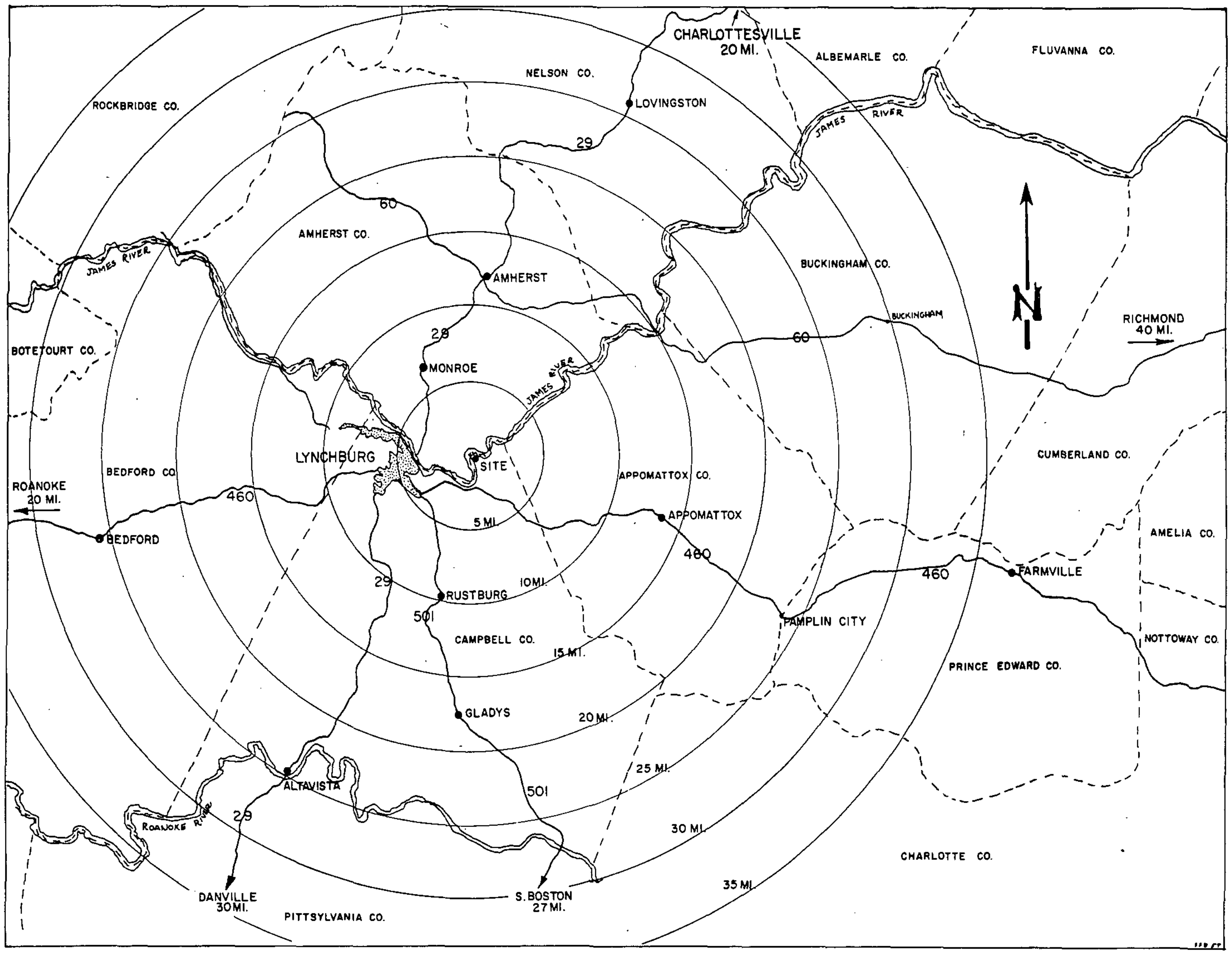


Figure 2.11. Map of Area Surrounding BAWTR Site



3. FACILITY DESCRIPTION

3.1 General

The Facility is functionally separated into two sections. One is the NFL, which includes the Radioactive Material Laboratories, a hot cell, and a Feed Materials Building. The other is the test reactor, BAWTR, which is an integral part of the same structure as that of the NFL. The functions to be carried on in each of the sections will compliment the total fuel development work of B&W's NDC.

Licensing of the NFL to permit operation with nuclear fuel and other radioactive isotopes will be under applicable requirements of the Code of Federal Regulations for such facilities. The NFL will be licensed separately from the BAWTR. All operation and handling of nuclear material in the NFL will be under strict operating procedures and will not affect the actual operation of the reactor.

Figures 3.1, 3.2, and 3.3 show the general arrangement of the Facility and the location of the reactor in reference to the NFL. It also shows the relative position to the other facilities at the site owned and operated by B&W. This arrangement of the NFL areas permits a consistent and easy scheme of operation for the total development center. Except in the case of extreme emergencies, the operation of the BAWTR would in no way affect operation of the CEL or the NFP.

3.1.1 Purpose and Use

The primary purpose of the NDC is to provide facilities to develope nuclear fuels. It provides space and equipment to fabricate, irradiate, examine, and separate the various components of most of the nuclear fuels being considered for use in present and future nuclear power plants. It will also provide irradiation and laboratory examination space for a variety of samples for chemical and metallurgical studies. It is not anticipated at the present time that fuels or other materials will be irradiated or otherwise operated to destruction. The Facility will be used in the commercial, government, and private programs to be carried out by B&W in its nuclear development work.

The Facility will be designed and equipped to handle a large variety of materials including irradiated fuels, such as U^{233} and plutonium. These will be handled under strict conditions, operating procedures, and limitations to prevent a hazard to the public. To carry out the planned program of nuclear fuel development, it is planned to have the plant completed and in operation by the end of 1963. It is planned to contruct and use the NFL previous to the operation of the test reactor. This will permit carrying out a number of fuel element development programs that do not depend on irradiation facilities. The following is a planned schedule for design, construction, and operation for the BAWTR.

BAWTR Construction Schedule

	Start	Completion
Design	4/62	10/62
Procurement	9/62	4/63
Construction	9/62	8/63
Testing	7/63	9/63
Startup and High Power Tests	9/63	12/63

3.1.2 Future Plans

The NDC is designed to test, under actual operating core conditions, a large variety of fuels and fuel elements to be used in nuclear power plants. Since the future of the nuclear power industry strongly depends on nuclear fuel costs, the program will include studies and evaluations of fuel cycles and the use of recycled fuels in reactors. A large portion of the work to be done at the Facility will be directed toward a good utilization of recycled fuels.

3.2 Reactor

3.2.1 General

The BAWTR is a pool reactor, moderated and cooled by light water. The core is made up of a square array of ETR-type fuel elements with the four central elements replaced by an experimental hole. Of the 12 remaining core positions, 9 are filled by 19-plate elements, one by a 10-plate element allowing space for the regulating rod, and 2 by 16 plate elements used as fuel-follower type safety rods. As the safety rods are withdrawn from the core, fuel is inserted. The core will operate at a power level of 6 MW and contains about 6.44 kg of U^{235} as fuel. The fuel cycle is 210 days at a 6-MW operation.

The core is reflected on four sides by a beryllium reflector. Eight shim rods are located in the reflector pieces on each side of the core (32 total). There are also experimental holes in beryllium reflector pieces to be used for capsule and sample irradiations. Figure 3.4 shows the reactor arrangement; Figure 3.5 shows a cross section of the core and reflector.

The fuel and reflector elements are surrounded by an aluminum shroud, supported at the bottom by a grid plate, and held in place during operation by the downward flow of the coolant water. The primary inlet plenum provides lateral support for the shim rod guides and also directs the flow downward through the core. Located below the core is a lower shroud and outlet header assembly that directs the flow of coolant water through the core to the circulating pump. The outlet header opens by force of gravity to permit a natural convection of the cooling water through the core in the event of loss of flow or reactor shutdown.

The design of the reactor is based on the use of mechanical equipment and design features similar to those used in existing reactors over a period of several years. It is not planned to make any experimental or developmental investigations on the components to be used for the BAWTR. Table 3-1 is a list of some of the important operating parameters of the BAWTR.

Table 3-1. BAWTR Operating Parameters

Reactor Power (no loops), MW	6.0
Maximum Power of Experiments, MW	0.75
Pressure at Top of Core, psig	9.0
Coolant Temperature, F	
Inlet	100
Outlet	112.5
Average Coolant Flow, fps	13
Coolant Flow, gpm	3800
Through Fuel Elements, gpm	3040
Through Reflector Elements, gpm	760
Surface Heat Flux, Btu/hr-ft^2	
Average	96,200
Maximum	288,600
Burnout	750,000
Maximum Fuel Plate Surface Temperature, F	206
Fuel Loading, kg of U^{235}	6.4
Metal-to-Water Ratio	0.65
Number of Fuel Elements	
Regular 19 plates	9
Control 16 plates	2
Control 10 plates	1
Reflector Pieces, Be	20
Control Rods	
Number of shim rods	32
Number of safety rods	2
Number of regulating rods	1
Regulating Rods	
Stroke length, in.	30
Withdrawal Speed, ipm	22.5
Speed of Insertion, ipm	22.5
Absorber section	Nickel
Length of Poison Section, in.	30

Table 3-1 (Cont'd)

Safety Rods	
Stroke Length, in.	34.25
Withdrawal Speed, ipm	5
Insertion Time, sec	Full Flow, 0.48 No Flow, 0.61
Absorber Section	Cd-Stainless Steel
Length of Poison Section, in.	31
Shim Rods	
Stroke Length, in.	30
Withdrawal Speed, ipm	18
Insertion Time, min	1.7
Absorber Section	Boron-Stainless Steel
Length of Absorber Section, in.	30
Active Core Height, in.	30

3.2.2 Core Structure and Support

The reactor and the associated parts of the reactor system are supported from the bottom of the reactor pool by four aluminum braces fastened to the pool floor. (See Fig. 3.4.) The four braces extend from the pool floor and are welded to the sides of the lower shroud. The lower shroud (Fig. 3.6) is a cylindrical aluminum tank smaller at the bottom to fit the outlet header. The cylindrical grid plate, in which the core and reflector are located, is fastened to the top of the lower shroud, and is made of thick aluminum. The upper shroud, which surrounds the core and reflector, is fastened to the top of the aluminum grid plate. This upper shroud is also made of aluminum, extends to the primary inlet plenum, and acts to direct the flow of coolant through the core and reflector. It is flattened on one side to permit an experiment to be placed close to the reactor core.

The aluminum primary inlet plenum, located around and above the core, functions to direct the flow of coolant to the top of the core. Open on the core side to direct most of the return coolant directly into the core, it also prevents a high level of activity at the pool surface by directing the large portion of activated water through the reactor core.

The primary inlet plenum is connected to four primary system risers extending from the primary distribution header. The risers are aluminum lines that supply cooling water to the reactor core through the inlet plenum and are fastened to the primary distribution header which is a circular aluminum pipe supported by aluminum brackets on the pool bottom. Cooling water is supplied to the distribution header from the aluminum water inlet line extending up above the core and through the pool wall.

The support for the BAWTR is designed with sufficient strength to support the reactor components and the maximum force generated by the downward flow of the coolant and control rod scram with an additional safety factor.

3.2.3 Fuel Elements

The BAWTR uses an ETR type fuel element. In the operating core there will be three variations of the element. Each type will use the same flat plates, but plate dimensions and the number in the types of elements will vary. The elements are described as the standard BAWTR element, regulating rod element, and the safety element. The features of each are given in the following sections. Figure 3.5 shows the core location of each type of fuel element.

3.2.3.1 Standard Fuel Element

The standard BAWTR fuel element (Fig. 3.7) is approximately 3 by 3 inches in cross section and 40 inches in total length. The active fuel section is 30 inches long. The element is made up of 19 flat fuel plates pinned into aluminum side plates. Fuel plates are fabricated of a 0.020-inch fuel layer of enriched uranium-aluminum and clad on each side with 0.015-inch thick aluminum. Each element contains approximately 593 grams of U^{235} and each core loading requires nine of these elements.

Support for the fuel elements in the core is provided by a lower end fitting attached to the fuel element section. This fitting is inserted into the holes in the grid plate to position and support the elements. To facilitate handling in and out of the core, a fitting is attached to the upper part of the element. Both the upper and lower end fittings are aluminum.

3.2.3.2 Regulating Rod Element

The regulating rod fuel element (Fig. 3.8) is located in one of the corner fuel positions of the core. It is approximately 3 by 3 inches in cross section, and the nine central fuel plates are removed to provide a slot for the regulating rod. The fuel plates are arranged five on each side of the element and are separated from the regulating rod by two 0.125-inch thick aluminum plates that create the channel for the regulating rod. The 10 fuel bearing plates contain approximately 311 grams of U^{235}. The fitting at the top of the element is designed to offer vertical support for the regulating rod drive mechanism.

3.2.3.3 Safety Elements

The fuel follower for the safety rod contains 16 fuel plates pinned into aluminum side plates. These elements are similar to a standard fuel element except they are 2.506 inches square. Two of the safety elements are used in each core assembly. Each element contains about 396 grams of U^{235}. Figure 3.9 shows the construction of the safety rod and the details of the fuel element section.

3.2.4 Reflector

The reflector elements (Fig. 3.10) are located around the periphery of the fuel elements. The 20 elements surrounding the core, fabricated of beryllium metal, are approximately 3 inches square and 30 inches long. An aluminum nozzle adapter is pinned to the lower end of each element and an aluminum top piece is pinned to the top. The nozzle adapter supports the element and directs the coolant flow through the element. The top pieces are for handling the elements in and out of the grid plate. The four corner elements each have a 1.625-inch ID capsule hole. The eight elements adjacent to the corner positions have a 1.125-inch capsule hole and a 0.563-inch hole for the shim rod. The remaining eight elements each contain three 0.563-inch diameter holes for shim rods and a 0.25-inch hole for cooling water flow.

3.2.5 Upper Shroud and Filler Blocks

Figure 3.11 is a cross section through the upper shroud. The side of the shroud facing toward the pool gate has been flattened to provide irradiation space. Large items to be irradiated may be brought into the pool through the gate and pushed out onto the transfer car bridge up to the side of the shroud. Aluminum filler blocks are used in the space between the beryllium reflector and the inside of the shroud wall.

Nuclear calculations indicate that aluminum filler blocks will reduce k_{eff} by approximately 1%, and the primary effect of this will reduce core life. These calculations also indicate that the reactivity change involved with the movement of a neutron absorbing material being irradiated from the outside of the shroud will be less than 0.2% δk.

3.2.6 Reactor Nuclear Instrumentation and Control

3.2.6.1 General

The Nuclear Instrumentation System (Figs. 3.12 A, B, C) includes a startup channel, two intermediate power channels, and two linear power channels. These function to perform the following:

1. Monitor the neutron flux level from source level through full power.
2. Provide rate of change (period) information from source level through full power.
3. Provide a signal to initiate regulating rod insertion when selected signals reach their respective set points.
4. Provide a signal to initiate scram when selected signals reach their respective scram set points.
5. Provide a signal for regulating rod control from 1% through full power.
6. Provide a convenient means for operation, annunciation, and indication of the reactor nuclear parameters.

Figure 3.13 shows the approximate range of detection of each channel.

The detectors are contained in aluminum instrumentation thimbles and are supported to reduce possible ground currents as well as microphonic and thermal noise in the detector. The thimbles extend vertically to the bridge where openings permit the removal and replacement of detectors. These thimbles are located approximately 2 feet from the edge of the reflector face in a full-power thermal flux level of about 3×10^9 nv. Surrounding each thimble is a layer of lead which reduces gamma radiation at the detector to about 10^5 r/hr.

Major equipment is located in the nuclear instrumentation cabinets and in the control console.

3.2.6.2 Instrumentation Description

The instrumentation system is divided into three functional groups: low-level startup instrumentation provides neutron flux and period information from source level to approximately 8×10^{-4} full power; the log N channel provides neutron flux and reactor period information, as well as inputs to the safety system, from approximately 1×10^{-4} full power to full power; linear power range instruments indicate the neutron level and provide operational information to the automatic control system and reactor safety system from 1.0×10^{-3} full power through full power. These three groups operate with overlapping ranges to prevent loss of information as the reactor is brought from startup to full power.

Startup Channel

The startup channel relies on the processing of individual pulses resulting from fissioning of the U^{235} atom coating in the startup fission chambers. Each detector pulse is transmitted through the preamplifier, which matches the impedance of the detector to that of the cable and transmits the pulses through the cable to the linear amplifier with maximum efficiency and minimum attenuation. The linear amplifier, composed of a high-gain, low-noise amplifier and discriminator, converts the preamplifier pulses to pulses of given magnitudes and widths.

The discriminator limits the linear amplifiers output to pulses from neutrons and eliminates pulses due to noise, gamma rays, and extraneous sources.

The logarithmic count rate circuit, through a series of integrators, converts the standard pulses from the linear amplifier to a signal proportional to the log of the pulse rate. The period or derivative amplifier differentiates the output of the count rate circuit and provides a signal proportional to the inverse reactor period.

Available from the count rate and period amplifier are indicating meters and outputs for a conventional strip chart recorder. The linear amplifier also will have a "scaler" output for use during critical experiments and reactor startup.

A Po-Be neutron source will be used for startup. After the initial high-power operation, sufficient photoneutrons from the beryllium reflector will serve as the startup source.

Intermediate Channel

The compensated ionization chambers supply a current ouput proportional to the neutron flux of the reactor. The current output of one of the compensated ionization chamber is amplified by a linear d-c amplifier. The amplified signal, which drives the linear power meter and the power recorder, is also fed to the automatic controller and the rod rundown control. This is used in the intermediate and power ranges.

The log N amplifier utilizes a log diode to convert the current signal from the chamber to a voltage proportional to the log of the neutron power. This power signal, indicated on a six-decade meter, can be recorded on the log N recorder. The log N amplifier output also goes to a period amplifier, which provides period information over the top six decades of reactor power. A period trip will be set to sound an alarm and initiate safety system action. This trip is designed to guard against rod withdrawal accidents over the source and intermediate ranges.

Safety Channel

Two ionization chambers are used to supply signals proportional to the power level into safety amplifiers. The output is a controlling signal to the magnet amplifiers. Scram action will de-energize the electromagnets and will release the safety rods.

3.2.6.3 Reactor Control

The reactor is controlled by 32 shim rods and a regulating rod that can be positioned manually. The regulating rod, in automatic control, can be positioned by the power comparator or driven into the core by automatic rundown.

Manual Control

Each control rod can be individually withdrawn or inserted by the operator. Also, four shim rods can be withdrawn as a group during startup to obtain the maximum shim rod reactivity change rate (approximately 0.5×10^{-3} $\delta k/k$ sec). Each rod has an individual position indicator and two limit switches: rod lower limit and rod upper limit.

Automatic Control

The automatic control is accomplished by comparing a set neutron level; that is, the power set point to an actual neutron power reading from the linear level channel. This is a simple "on-off" control system in which the regulating rod moves "in" or "out" according to power comparator demands.

Based on the differences between the power set point and the actual power, the controller utilizes proportional and integral control action to move the regulating rod and maintain the set power level. Regulating rod movement results when the error signal exceeds the controller's dead band. Since the integral portion of the system will drive the rods to reduce this error to zero, the neutron flux level should remain essentially constant.

An interlock will trip from automatic to manual control on indication of a scram.

3.2.6.4 Safety System

This system, primarily designed to protect the core from possible damage to the fuel elements, receives inputs from nuclear instrumentation, process instrumentation, test loops, and various auxiliary sources. If certain limits are exceeded, the system provides a signal to cause a regulating rod rundown or a scram.

During a rundown, the regulating rod is driven into the core at normal speed. A scram releases the safety rod-drive holding magnets allowing the safety rods to fall by gravity. Scram action also starts rod rundown on all shim rods.

Table 3-2 shows reactor scram conditions.

Table 3-2. Reactor Scram Conditions

Fast Scram (Electronic - fast acting)
- Short Period
- High Flux Level
- Safety System Malfunction

Slow Scram (Relay or other mechanisms - slower acting)
- Log N Period
- Log N Level
- Linear Level
- Loss of Primary Pump
- Loss of Primary Building Power
- Experiment or Loop
- Loss of Minimum Count Rate
- Loss of Recorder Power
- Manual Scram

Automatic Regulating Rod Rundown
- Period

3.2.7 Control Rods and Drives

3.2.7.1 Safety Rods and Drive Mechanisms

Figure 3.14 shows the arrangement of the various components associated with a BAWTR safety rod. The maximum over-all cross section dimension of this assembly is 3 × 3 inches. (This occurs at the guide tube.) Details of the fuel section, poison section, guide tube, and flow tubes are shown in Figure 3.9.

The guide tube extends from below the grid plate up through the reactor core to a point above the grid plate. At this point, the guide tube is attached to a shell assembly tube.

The safety element fuel section is attached to the poison section by a latch device. The top of the poison section is attached to the flow tube that is attached to a long connecting rod with a magnet seat at its upper end.

The electromagnet used for withdrawing the inner assembly of the fuel, poison, and the flow tube is located in the drive mechanism above the water line. An electric motor raises and lowers the electromagnet by an Acme screw. Details of the drive mechanism are shown in Figure 3.15. Limit switches provide an indication of rod position and magnet engaged and rod seated conditions. A selsyn transmitter permits remote indication of rod position.

The poison section of the safety rod is made up of a sandwich of cadmium within layers of stainless steel. The cadmium is attached to the inner stainless-steel tube by a metallizing process; that is, it is sprayed on.

3.2.7.2 Regulating Rod and Drive Mechanism

A nickel sleeve of an elliptical cross section is used as the regulating rod in the BAWTR. (See Fig. 3.5.) Figure 3.16 shows the details of the rod and drive mechanisms. The regulating rod drive mechanism design is similar to that of the safety rod drive mechanism. The principal difference is the speed of rod withdrawal and the fact that the regulating rod does not scram and therefore does not have an electromagnet.

The regulating rod drive mechanism receives lateral support from the reactor bridge. It is further supported through a long aluminum guide tube bolted to the top of the special regulating rod fuel element. With this support, it is impossible for a stuck regulating rod to withdraw a fuel element from the core.

3.2.7.3 Shim Rod and Drive Mechanisms

Thirty-two shim rods are used in the BAWTR core. By being located at the core reflector interface, these rods affect reactivity by parasitic absorption and also by changing reflector effectiveness. The rods are 0.5-inch OD solid boron stainless steel. Attached to each of these rods is an aluminum rod follower.

As shown in Figure 3.17, the shim rods are driven from below the core by flexible cables running within conduits. At the bridge above the pool, an electric drive motor engages the flexible cable through a drive gear. The shim rods do not drop on scram but begin driving in after a scram signal.

Shim rod guides are attached to the top of the reflector elements and receive lateral support from the primary inlet plenum.

3.2.8 Shielding

Sufficient biological shielding is provided to prevent personnel from being exposed to dose rates greater than those specified in 10 CFR, Part 20, for continuously occupied areas. In areas where the dose rates are higher than the specified rates, access time will be limited to assure that the maximum exposures never exceed the limits as given in 10 CFR, Part 20, for integrated doses.

Shielding around the sides of the core is provided by the pool water surrounding the reactor and the concrete walls of the pool. Most of the area surrounding the reactor is below ground level and is not accessible to personnel. Areas below ground level that can be entered will be closed and entry will not be permitted during reactor operation.

The principal biological shielding on the BAWTR consists of the pool water directly over the core. (See Fig. 3.4.) A depth of 21 feet over the core keeps the direct radiation well below tolerance levels. The N^{16} level is expected to be quite low since little mixing occurs between the primary coolant flow and the pool water proper. The major source of radioactivity at the surface of the pool arises from Na^{24} activity. At a power level of 12 MW, the dose rate at the pool surface is calculated to be less than 16 mr/hr.

Continuous recirculation of the pool water through a mixed bed ion exchange column removes much of the Na^{24} activity. This, however, causes a buildup in radioactivity at the demineralizer, and so concrete shielding is used in the primary equipment cell to permit access to the demineralizer valves.

The level of N^{16} activity in the primary coolant loop requires sufficient biological shielding to be used in the primary equipment cell to permit access to certain areas. Normal concrete will be used for shielding here.

The area within the containment over the experimental loop area is shielded from the core by the concrete floor and the concrete pool wall. Normal concrete is used in both places and the thickness is sized to reduce the dose rate above the floor to less than 7.5 mr/hr. The pool wall thickness is sufficient to permit access to the experimental loop area within a short period after reactor shutdown.

3.2.9 Experimental Facilities

The design of the BAWTR provides a large amount of irradiation space in and around the core of the reactor. Usable irradiation space for the reactor is given in the following list.

1. One 6 × 6-inch central irradiation hole.
2. Four 1.625-inch diameter holes in beryllium corner pieces.
3. Eight 1.125-inch diameter holes in beryllium reflector pieces.
4. The flat side of the upper shroud adjacent to core is accessible by the transfer car bridge.

The principal experimental facility is the 6 × 6 inch test hole for a test loop insertion. A cross section through a typical in-pile re-entrant-type thimble is shown in Figure 3.5. Section 5.1 gives a more detailed description of a typical pressurized water test loop. Also shown in Figure 3.5 are the twelve capsule holes running longitudinally down through the reflector elements.

Irradiation of large devices can be accomplished by using the flattened side of the upper shroud. Gamma irradiation experiments may be carried on in the storage pool using the spent fuel elements located in the storage racks. Both gamma and neutron irradiation may be performed by suspending materials into the water around the upper shroud. Rabbit tubes may be placed adjacent to the beryllium reflector or outside the upper shroud.

3.2.10 Transfer Canal and Storage Pool

The transfer canal connects the reactor pool with the hot cell. This canal is connected to the reactor pool through two removable gates, (Figs. 3.2 and 3.3) and to the hot cell through an 18-inch ID hatch in the cell floor. Thus, the canal water provides the shielding required during the transfer of samples to and from the test reactor. Also, the canal provides the shielding required when removing test elements from shipping coffins which may arrive from other sites. The two removable gates permit the independent draining of the reactor pool or storage pool for maintenance.

The canal also contains storage racks for spent reactor elements after they are used in the reactor. These racks support the elements so they are cooled by natural circulation. From the storage racks, the spent elements are transferred into a shipping coffin for shipment to a reprocessing plant. Other items requiring shielded storage will be held in the storage pool until disposal can be arranged.

3.2.11 Reactor Pool

The reactor is located in a pool of demineralized water about 12 feet in diameter and 30 feet deep. The pool, constructed of ordinary concrete, is lined with a liquid tile to waterproof the pool and prevent contact between water and concrete walls. The top of pool, at approximately ground level, is open to permit access to the reactor and experimental space. The primary coolant outlet penetrates the pool floor in the center of the pool directly beneath the core. Gutters are provided at the top of the pool for overflow. This water is pumped directly into the regeneration waste storage basins at the pits.

3.2.12 Containment

The reactor pool and out-pile loop space are completely enclosed in a low-leakage steel containment vessel about 33 feet in diameter and 65 feet high. The walls are 0.25 inch in nominal thickness. The heat exchanger for the primary coolant is outside the containment vessel. During operation, the transfer tunnel between the reactor pool and the storage pool is sealed by an aluminum gate. Access to the containment is provided through a personnel air lock. There is also a bolted plate at the rear of the structure to permit entry of large pieces of equipment. The shell will be designed and tested in accordance with the latest ASA document, "Proposed Standard for Design, Fabrication, and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors". Design calls for a leakage rate of the vessel to be less than 1% per day of the contained volume at a 5-psig internal pressure. The leakage rate is low enough to prevent exposures to people in Lynchburg above those specified in 10 CFR, Part 20, for emergencies resulting from a catastrophic accident to the BAWTR.

3.2.13 Exhaust Stack

A 150-foot high, 4-foot diameter steel stack will be built to provide for controlled release of the air drawn from various areas within the Facility. The stack will be supported on a concrete base and stabilized against wind forces by a system of corrosion resistant guy wires. The top of the stack will be necked down to provide an exit diameter of 2 feet. At an average wind velocity of 7.6 mph and flow rates in the range of 15,000 to 20,000 cfm, the effective height of the stack will be increased by 30 to 50 feet. Considering the topology of the site, this stack will permit effluent release at a height sufficient to assure against holdup within the local area.

3.2.14 Reactor Physics

3.2.14.1 General

To determine the nuclear characteristics of the BAWTR reactor, a large number of physics calculations were performed for the core. Physical geometry and dimensions of the core arrangement are as given in Figure 3.5 and Table 3-3. Most of the calculations were made by the PDQ code and the four-group diffusion theory. A mesh layout was chosen to describe as discrete regions most of the major core internals. The fuel plates and water gaps between the fuel plates were homogenized. However, the aluminum side plates of the fuel elements were described separately. All other water channels were shown separately as were the capsule holes and shim rods in the beryllium reflector. The typical experiment in the central thimble was homogenized, but the concentric structure around the experiment was shown in considerable detail. In each case, the operating condition for the core is the clean BAWTR operating at 6 MW and 120 F average temperature.

The experiment, as described in Section 5.1, is in the core at all times of operation. CETR-type fuel pins, enriched with U^{235}, are contained. Other comparisons are made by using an SSCR-type fuel pin that is 10%-enriched in U^{235}. Table 3-3 gives the reactor parameters used for the physics calculations.

3.2.14.2 BAWTR Power Distribution Factors

Calculations show that the maximum radial power factor occurs with the regulating rod of 0.8% δk value and 16 shim rods inserted in the core. This maximum radial peak-to-average power factor is 2.05. The maximum axial peak-to-average power factor is 1.46.

To facilitate thermal analysis, a plot of power-to-average BAWTR fuel power along the hottest homogenized fuel plate in the BAWTR is given in Figure. 3.18. This occurs with 16 shim rods and the regulator inserted. The hot spot along the plate coincides with the hot spot of the core. This power to average power is based on the PDQ analysis.

Table 3-3. Core Parameters for Physics Calculations

ETR Type Fuel Elements	
Full 19 plates	9
Safety Rod 16 plates	2
Regulating Rod 10 plates	1
Size, in.	3 × 3
Length of fuel, in.	30
Array (Central four elements removed)	4 × 4
Reflector, 3 × 3-inch Beryllium, pieces	20
Central Opening in Core, in.	6.06
Re-entrant Test Loop OD, in.	5.75
No. Shim Rod (0.5-in. OD)	32
Regulating Rod, 7/8 in. × 2 1/4 rods	1
No. Safety Rods (2 to 2.5 in.^{2})	2
No. Capsule Holes in Reflector (1.625 in. OD)	4
No. Capsule Holes in Reflector (1.125 in. OD)	8
Moderator and Coolant	H_2O
Reactor Power Level, MW	6
Average Operating Temperature, F	120
Operating Temperature of Typical Experiment, F	600
CETR-Type Fuel Composition	
U^{235} atom enrichment, %	4
Thorium, U^{235} ratio	24.5
Cladding	304 Stainless Steel
Loading, gm U^{235}/pin	6.3
SSCR-Type UO_2 Fuel Composition	
U^{235} wt% enrichment	10
U^{238} : U^{235} ratio	8
Cladding	Zr-2
Loading, gm U^{235}/pin	47.2
SSCR-Type ThO_2-UO_2 Fuel Composition	
U^{235} atom enrichment	10
Thorium: U^{235} ratio	9.1
Cladding	Zr-2
Loading, gm U^{235}/pin	43.9

3.2.14.3 Experiment Power Generation Rates

The experiment with CETR-type fuel will generate between 343 and 420 kw of power depending on the number of shim rods in the core. The BAWTR fuel produces 6 MW of power under all conditions.

The power and the power per pin per foot for the various fuel loadings investigated in the BAWTR experiment are reported in Table 3-4. The maximum power per pin per foot reported is the maximum power in the experiment which will occur in each case at the outside surface of the outer corner pins. This maximum power applies a 1.46 axial power factor together with the radial power factor of each experiment loading on the average power per pin per foot. These values reported in Table 3-4 are for the BAWTR core with no shim rods inserted; with shim rods inserted, these values will be 15 to 20% higher.

Table 3-4. BAWTR Experiment Power

Fuel Type	No. of Pins	Power in Experiment, kw	Average Power Per Pin Length, kw/pin-ft	Maximum Power Per Pin Length, kw/pin-ft
CETR	56	343.2	2.4	4.3
SSCR (10 wt% enriched in U^{235})	16	555.9	13.9	32.2
SSCR (10 wt% enriched in U^{235})	9	370.1	16.4	33.9
SSCR (ThO_2-UO_2 enriched to 10 atom% in U^{235})	16	501.3	12.5	29.1

3.2.14.4 Reactivity Control

A curve of the safety rod worth versus distance of insertion into the core is shown in Figure 3.19. This curve represents the relative effect of one or both of the safety rods.

Calculations indicate that a nickel regulating rod with a wall thickness of about 0.282 inches will provide 0.8% δk control when fully inserted. The results of reactivity control studies are summarized in Table 3-5.

Table 3-5. Reactivity Control in the BAWTR

Core Description	k_{eff}	δk_{eff} from no rod case
Clean operating, all rods out	1.1329	-
Clean operating, 32 shim rods inserted	0.9072	-0.2257
Clean operating, 16 shim rods inserted	0.9729	-0.1600
Clean operating, 8 shim rods inserted	1.0260	-0.1069
Clean operating, 1 safety rod inserted	1.0586	-0.0743
Clean operating, 2 safety rods inserted	1.0066	-0.1263
Clean operating, regulator inserted	1.1249	-0.0080

3.2.14.5 Reactivity Range of Experiments

Calculations were made to establish the maximum and minimum reactivity effects due to various experiments in the test region. The criticality value is summarized in Table 3-6. The maximum reactivity that could be expected in the BAWTR would result if BAWTR fuel is added to the central test region. The minimum reactivity expected will result if thorium pins are inserted into the test region.

3.2.14.6 Reactivity Effects of Experimental Accidents

Several accident conditions have been investigated. Two possible accidents calculated would result if the experimental fuel melts down and the experiment is either void or flooded with 120 F water. The results, summarized in Table 3-7, are compared with the minimum anticipated k_{eff}.

Table 3-6. BAWTR Criticality Values

	k_{eff}	δk_{eff}
Maximum Effective Multiplication (BAWTR fuel in experiment)	1.1705	+0.0376
Typical Effective Multiplication (CETR-type fuel in experiment)	1.1329	-0.0497
Minimum Effective Multiplication (Thorium pins in experiment)	1.0832	

Table 3-7. Experiment Accidents

Core Description	k_{eff}	δk_{eff}
Minimum Effective Multiplication (Thorium pins in experiment)	1.0832	-
Experiment Flooded with Water at 120 F (no pins in experiment)	1.0722	-0.011
Experiment Voided (no pins in experiment)	1.0892	+0.006

Another accident investigated was an explosion of the experiment under pressure with CETR fuel in the experiment. It is assumed that the explosion causes uniform displacement of the core internals in all directions from the center. There is no increase in k_{eff}. (See Fig. 3.20.)

3.2.14.7 Temperature and Void Coefficients

The BAWTR core temperature coefficient was calculated to be -1.4×10^{-4} $\delta k/F$ at 120 F. K_{eff} versus temperature is plotted in Figures 3.21 and 3.22.

The results of a temperature deficit investigation are summarized in Table 3-8. The k_{eff} increases in bringing the experiment from 68 F to 600 F due to the decrease in the poisoning effect of the water annulus being more important than the decrease of k_{∞} in the experiment fuel.

Table 3-8. BAWTR Temperature Deficit

Core Description	k_{eff}	δk_{eff}
Entire BAWTR at 68 F	1.1262	
		-0.0038
Central Experiment (CETR type fuel) at 68 F, remainder of core at operating temperature of 120 F	1.1224	
		+0.0105
Entire reactor at operating temperature, experiment at 600 F and remainder of core at 120 F	1.1329	

The BAWTR core void coefficient is -0.5 $\delta k/\delta\rho$(ρ in gm/cm^3) at a density of 0.9880 gm/cm^3. The maximum void coefficient of the test loop with CETR type fuel in the experiment was calculated to be $+1.5 \times 10^{-4}$ δk/% void at 0% void. A curve or k_{eff} versus % voiding of the experiment is shown in Figure 3.23. The maximum reactivity that could be added by partially voiding the experiment is +0.006 δk_{eff}, which results in going from 0% void to 60% void. The reactivity effect of complete voiding is -0.0163 δk_{eff}.

3.2.14.8 Flux Data

Average thermal fluxes from PDQ calculations are given in Table 3-9, and a plot of the thermal flux across the center line of the experiment when occupied by CETR-type fuel with 8 shim rods in the core is shown in Figure 3.24.

The average thermal neutron flux in the 6.06-inch squared center test region completely filled with aluminum is 1.86×10^{13} nvt. The fast-to-thermal flux ratio in the center test region when filled with Al is 8.3 to 1. The fast-to-thermal flux ratio in the CETR-type fuel in the experiment is 4.8 to 1. The fast flux includes all neutrons above 0.4 ev.

A representative one-dimensional thermal flux profile is shown in Figure 3.25 for the BAWTR with no rods inserted and CETR-type fuel in the experiment.

The average four-group radial fluxes in various BAWTR components are given in Table 3-10 for the BAWTR core operating with no rods inserted and Th-U, SSCR fuel in the experiment at 10 atom % enrichment in U^{235}. The fluxes are average axial values. An axial power factor of 1.46 should be applied to obtain average mid-plane flux levels in each of the materials.

Table 3-9. Average Thermal Fluxes in BAWTR $\times 10^{13}$

Flux Location in Core	Average Thermal Fluxes - neut/cm^2-sec		
	No rods	8 shim rods	16 shim rods
Experiment with CETR-type fuel	2.85	3.29	3.52
BAWTR fuel	2.08	2.05	2.04
1.125-in. ID Capsule Holes (Al filled)	2.99	2.70	2.73
1.625-in. ID Capsule Holes (Al filled)	3.07	2.90	2.96

3.2.14.9 Lifetime

Lifetime calculations indicate a life of 210 full power days with CETR-type fuel in the experiment. Figure 3.26 shows k_{eff} versus life. A mass balance is given in Table 3-11. It is estimated that the maximum reactivity condition will give a lifetime of 275 days; the minimum reactivity conditions will give a lifetime of 160 days. Reactivity balances for all three cases are given in Table 3-12.

Table 3-10. Four-Group Fluxes in BAWTR

Group	Energy Range, ev	Flux Value, nvt × 10^{13}				
		Experiment	BAWTR Fuel	Capsules, 1.125-in. ID	Capsules, 1.625-in. ID	Maximum flux at outer edge of Be reflector
1	0.9119×10^4	9.20	8.41	4.27	2.58	1.08
2	0.9119×10^4 to 0.50×10^2	3.57	2.59	1.73	1.27	0.63
3	0.503×10^2 to 0.4	2.50	2.03	1.29	0.98	0.62
4	0.4	1.63	2.08	2.96	3.03	4.95

Note: 1. Experiment is occupied with Th-U SSCR type fuel enriched to 10 atom % in U^{235}.

2. Capsules (both 1.125 in. ID and 1.625 in. ID) are filled with Al.

3. Core has no rods inserted.

Table 3-11. BAWTR Mass Balance (CETR-type fuel in experiment)

	Initial	Final
kg of U^{235} in BAWTR Fuel	6.442	5.208
Core Life at 6 MW		
days		210
MWD		1,260

Table 3-12. BAWTR Reactivity Balances

	δk_{eff}		
	Maximum Reactivity	Typical Reactivity	Minimum Reactivity
Equilibrium Xenon	0.0388	0.0388	0.0388
Equilibrium Samarium	.0095	.0095	.0095
Fission Products at End of Life	.0251	.0175	.0139
Maximum Reactivity Available for Lifetime and Experiment	.0971	.0671	.0210
	.1705	.1329	.0832
Control Rod Worth	.2257	.2257	.2257
Minimum Shutdown Reactivity (Shim rods only)	0.0552	0.0928	0.1425

3.2.14.10 Reactor Kinetics

Kinetics data for the BAWTR is given in Table 3-13.

Table 3-13. BAWTR Kinetic Data

Neutron Lifetime

ℓ, beginning of life 52.6×10^{-6} sec

ℓ, end of life 74.8×10^{-6} sec

Six-Group Neutron Delay Fractions with their associated Decay Constants

Group l	β_l	λ_i, sec^{-1}
1	0.000211	0.0124
2	.001400	0.0305
3	.001260	0.1114
4	.002530	0.3013
5	.000737	1.1360
6	.000267	3.0130
	0.006405	

Effective Total Delay Fraction

β_{eff}, beginning of life = (1.15) (0.006405) = 0.007366

β_{eff}, end of life = (1.05) (0.006405) = 0.006725

3.2.14.11 Summary of BAWTR Nuclear Characteristics

A summary of the nuclear characteristics of the BAWTR core with CETR-type fuel in the experiment is given in Table 3-14.

Table 3-14. BAWTR Nuclear Characteristics

Maximum radial peak-to-average power	2.05
Maximum axial peak-to-average power	1.46
Reactivity effects (clean core), k_{eff}	
All rods out	1.1329
32 shim rods in	0.9072
16 shim rods in	0.9729
8 shim rods in	1.0260
1 safety rod in	1.0586
2 safety rods in	1.0066
Voided experiment	1.1166
Regulator inserted	1.1249
Average thermal flux (CETR experiment in loop) $\times 10^{13}$ nvt	
Experiment (CETR fuel)	3.3
BAWTR fuel	2.0
1.12-in. capsule hole (Al filled)	2.7
1.625-in. capsule hole (Al filled)	2.9
Average core life, days	210
U^{235} core loading, kg	6.44
Reactivity Balance, δk_{eff}	
Equilibrium Xenon	0.0388
Equilibrium Samariam	.0095
Fission Products (end of life)	.0175
Lifetime and experiment (max.)	.0671
Total	0.1329
Control rod worth	.2257
Minimum shutdown (shims only)	0.0928

Table 3-14 (Cont'd)

Core temperature coefficient, 120 F	$-1.4 \times 10^{-4}\ \delta k/F$
Reactor fuel void coefficient, $0.988\ gm/cm^3$	$-\delta 0.5\ \delta k/\delta\rho$
Maximum positive void effect in CETR experiment (at 60% void)	$+0.006\ \delta k_{eff}$
Maximum positive void coefficient of CETR experiment (at 0% void)	$+1.5 \times 10^{-4}\ \delta k/\%$ void

3.2.15. Core Heat Transfer Analysis

The core heat transfer analysis is to assure that sufficient heat removal capability is provided to maintain coolant and core materials within safe temperature limits during reactor operation. The operating limitation for the reactor is fuel element failure as a consequence of burnout or breakdown of the heat removal mechanism. For the condition existing in this reactor, the correlation arrived at by Bernath is used. The value predicted by the correlation is reduced to 60% of the calculated value to account for uncertainties and scatter of the data.

The equation used is

$$q''_{Bo} = h_{Bo} \, (T_w - T_B)_{Bo} \, ,$$

where

$$h_{Bo} = \left[10{,}890 \left(\frac{D_e}{D_e + D_i} \right) + \frac{48V}{D_e^{0.6}} \right]$$

$$T_w = 32 - 97.2 \left(\frac{P}{P+15} \right) - 0.45\ V + 102.6 \ln P$$

D_e = equivalent diameter, ft

D_i = heated perimeter/π, ft

V = velocity, ft/sec

P = pressure, psia

The value used for design is

$$q''_{Bo\,(Design)} = 0.6\ q''_{Bo} = 750{,}000.$$

Besides the nuclear power distribution, the effect of manufacturing tolerances is accounted for in the design. It is assumed that the allowable deviations have a normal distribution about the nominal value. A most probable pessimistic deviation is calculated to obtain what is termed Engineering Design factors. Factors are calculated that apply to the fluid temperature rise ($F_{\delta T}$), the film temperature difference (F_{Θ}), and the heat flux ($F_{q''}$). For $F_{\delta T}$, consideration is given to variations in flow area and fuel content, to plenum and channel velocity distributions, and to power and inlet temperature regulation errors. For

F_Θ, consideration is given to variations in channel geometry, local fuel content and heat transfer area, and to plenum and channel velocity distributions and to power regulation errors. For $F_{q''}$, consideration is given to variations in local fuel content, heat transfer area, and power regulation error. The application of these factors gives a greater than 99% confidence level that the heat flux and surface temperature of the hot channel as calculated will not be exceeded.

Figure 3. 27 shows the various temperatures of the hot channel. The most probable peak surface temperature for this channel is 206 F. The maximum surface temperature expected with 105 F inlet fluid temperature is 248 F. The highest curve indicates the surface temperature at which local boiling begins. It is based on a correlation by Jens and Lottes.

Considering the $F_{q''}$, the maximum heat flux of the hot channel should not exceed 350,000 Btu/hr-ft^2 during operation. Since the calculated burnout heat flux is 750,000 Btu/hr-ft^2, the minimum ratio of burnout heat flux to maximum operating heat flux is greater than 2.

It is informative to compare the operating characteristics of the BAWTR with the ORR, an operating reactor whose characteristics are close to those of the BAWTR. A comparison of significant parameters is shown in Table 3-15.

Table 3-15 indicates that the thermal design characteristics of the two reactors are comparable, and no difficulty should be experienced in operating the BAWTR.

3. 2. 16. Stability

Operation of SPERT, BORAX, MTR, ETR, ORR, and many other reactors of this type have demonstrated the inherent nuclear stability of reactors of the BAWTR-type up to power densities much greater than planned for the reactor. With the amount of information available, there appears to be no reason for concern about the stability of the BAWTR. As long as the power level is no greater than the design levels, no effort will be made to determine stability transfer functions or make other extensive investigations in relation to reactor nuclear stability.

Table 3-15. BAWTR and ORR Thermal Comparison

	BAWTR	ORR
Reactor power, MW	6	20
Number of fuel elements	12	24 to 28
Average power per element, MW/element	0.5	0.83 to 0.715
Operating pressure at core inlet, psig	9	20
Velocity in fuel element, ft/sec	13	20
Nominal inlet temperature, F	100	120
Temperature rise, F	12	11
Power ratio	3.0 (design)	2.45 (measured)
Average heat flux Btu/hr-ft^2	96,200	163,000 at 24 elements 140,000 at 28 elements
Maximum heat flux Btu/hr-ft^2	288,000	400,000
Bernath burnout heat flux	750,000	1,160,000
Ratio of $\frac{\text{burnout flux}}{\text{maximum flux}}$	2.6	2.9

3.3. Nuclear Fuel Laboratory

3.3.1. General

The NFL, adjacent to the reactor containment structure, includes all areas other than the operations offices, control rooms, and small physics laboratory. The building is a two story structure with most of the second floor used as offices and the reactor control room. The first floor is utilized primarily for shops, laboratories, change rooms, and equipment handling rooms. The building, of masonry construction using pre-stressed concrete frame, is 312 feet long and 82 feet wide. It is a conventional laboratory building and will be constructed under standard regulations for buildings of this type in this section of the country. Floor plans and arrangements of the Facility are shown in Figures 3.1, 3.2, and 3.3.

The NFL is designed and equipped to:

1. Investigate and develop core fabrication processes.
2. Develop and test core materials and components.
3. Investigate and develop radiochemical techniques.
4. Examine and analyze irradiated materials and components from operating nuclear devices.

The areas for examination of irradiated materials and devices are located adjacent to the test reactor. The area for development of core fabrication processes is located at the opposite end of the center and is separated by general purpose laboratories. A brief description of these laboratories and the fuel fabrication area follows.

3.3.1.1. Radiochemistry Laboratory

This laboratory will provide facilities and shielding for chemical experiments and analytical work with materials of millicurie-level radioactivity.

1. Analysis of irradiated fuel element segments (elements that have been tested in-pile and elements that have been irradiated as a result of testing) for fission products and nuclear fuel concentrations to arrive at a calculation of the percentage of fuel burnup.

2. Dissolution and aliquotting of hot samples for fission product analysis.

3. Coolant chemistry, including

Mechanisms of activity buildup
Corrosion characteristics
Fission product transport studies
Solubility of fission products
Purity control.

4. Development of a failed-fuel-element detection system for Company sponsored reactors, such as the Universal Pressure Reactor.

5. Activation analysis for very low levels of impurities in irradiated specimens.

6. Development of new radiochemical analytical techniques including adaptation, both microchemical and instrumental, of normal analytical methods to remote work.

3.3.1.2. Spectrographic Laboratory

Initially, the spectrographic laboratory will include a Time-of-Flight Mass Spectrograph for the analysis of gases from fuel samples and from test loop operation. This instrument, also to be used for the determination of fuel burnup through analysis of the uranium content, will be of an advanced design to achieve the resolution required for isotopic analysis.

For the future, space has been provided for an emission spectrograph and an X-ray diffraction apparatus. The advanced design of the former will permit the analysis of impurities and other surface contamination found on irradiated specimens. It will also be used in the analysis of nuclear fuels.

3.3.1.3. Chemistry Laboratory

Work to be performed in this laboratory includes general and analytical procedures for non-radioactive and non-irradiated materials, and water chemistry analyses for the reactor test loop.

3.3.1.4. Tracer Laboratory

This laboratory will provide facilities for carrying out chemical experiments and analytical work with materials having very low levels of radioactivity. Such materials cannot be handled in a regular radiochemistry laboratory because of the higher activity levels in such a lab and the possibility for contamination of samples with tracer level activity. Handling of low (tracer) level activity in the analytical laboratory is also undesirable, since spills and contamination will cause a shutdown and work delay. It is a feasible to conduct tracer level analytical work at an off-site location due to the short radioactive decay times involved and costs involved in shipment.

The main functions of the tracer laboratory will be:

1. Analytical services for the Feed Materials Building where control of a continuous process, such as sol-gel oxide production, requires rapid, accurate analyses.

2. Quality control of raw materials for feed preparation utilizing natural activity of such materials as thorium or induced activity (activation analysis).

3. Qualify control on feed material product.

4. Low level contamination studies and service for health physics.

3.3.1.5. Metallurgy Laboratory

The metallurgy laboratory is equipped to provide the usual examination services on a macroscopic and microscopic scale and to determine mechanical properties and effects of heat treatment. Space is also provided for the conduct of out-pile experiments preparatory to the design and conduct of in-pile tests. Both structural and fuel bearing materials (unirradiated) will be handled in this laboratory in small quantities.

3.3.1.6. Photographic Processing Laboratory

This room is equipped with conventional equipment to process, print, and enlarge the film produced by the chemical, metallurgical, and hot cell operations.

3.3.1.7. Counting Room

This room will contain the necessary electronic equipment for counting various radioactive samples prepared in the Radiochemistry Laboratory, and automatic scanning of the irradiated wires and foils used in the test reactor flux mapping and experimental irradiations.

3.3.1.8. Feed Materials Building

The Feed Materials Building will house all operations involved in the production of urania, thoria, and urania-thoria. It is designed to produce the individual or mixed oxides by the ORNL sol-gel process. Although "cold" material can be handled, the main objective is the utilization of U^{233}, as uranyl nitrate solution, from Thorex or other processing plants. Consequently, an adequately shielded glove-box line will be provided to enclose and contain activity associated with U^{233}. Also, two cleanup units will be provided in separate, shielded enclosures, one an ion-exchange setup, the other a solvent extraction unit. These will be used to clean up incoming uranyl nitrate solutions and to redissolve scrap produced in the process and separate uranium from thorium. The oxide product will be sized, weighed, packaged, and, after package decontamination, transferred to the Oxide Area of the NFL.

The building will have its own ventilating system to discharge air and gases into the main stack after filtering and monitoring. Air flow direction will be from cold to hot areas (inside of glove boxes containing U^{233}), through absolute filters, then through the duct work to the stack. Also, air volume will remove heat from within those glove boxes enclosing dryers and furnaces. Kilogram quantities of uranium and thorium oxides will be handled in this building subject to criticality limitations.

3.3.1.9. Oxide Area

The Oxide Area will house all operations involving the handling of oxide fuel and fabrication of fuel rods. Area layout and containment is based on future use in the fabrication of recycled oxide fuel. Oxide will be received and loaded into fuel rods and will be mechanically compacted. The rod end caps are then welded and leak checked. In the case of fabricating, the "cold" fuel assembly will be carried out in the external assembly area. For assembling recycled fuel, such as U^{233}which emits gamma rays, a water-filled canal is provided at the end of the oxide area. This canal will serve also for storage of completed fuel rods and test elements. For U^{233} fabrication work, a special glove box line must be designed and constructed and existing equipment must be modified to fit. Ventilation from the area passes through an absolute filter before being discharged. A special duct system is provided for connection to individual glove boxes at a later date. A negative pressure with respect to the rest of the laboratory will be maintained within the area. Personnel egress from this area is controlled through health physics monitored hot and cold change areas.

Kilogram quantities of uranium and thorium oxides will be handled both in bulk form and in fuel rods and prototype elements. In all cases for U^{235} and U^{233}, batch sizes and configurations will be governed by criticality considerations.

3.3.1.10. Assembly and Machine Shop Area

The Assembly and Machine Shop Area will provide machine shop services for the entire Facility and will provide space for the assembly and inspection of core components on a pilot line basis. At one end of the area is located a small pickling facility and a pit for autoclaves to corrosion test fuel rods and elements prior to use and test.

3.3.1.11. Non-Destructive Testing Laboratory

The Non-Destructive Testing Laboratory services both the oxide and Machine Shop Areas in providing NDT service. It will be equipped with the equipment required to confirm the integrity and density characteristics of oxide fuel rods, such as X-ray, Ultrasonics, Helium Leak Detection, Gamma-scanning and liquid penetrants.

3.3.2. Hot Cell

The hot cell part of the NFL is located in the NFL building. The cell is designed to permit handling sources as high as 10^6 curies of 1.5 Mev gammas. The inside dimensions of the cell are 8 feet deep by 15 feet long by 12 feet high. It is shielded on all sides by a minimum of 4 feet of ilmenite concrete (density approximately 210 lb/ft^3), and viewing is provided by two zinc bromide windows about 4 feet thick. The dose rates at the surface of the shielded operating areas will not exceed 2.5 mr/hr. The cell will be equipped with light and heavy manipulators and experimental equipment to examine fuels, fuel rods, and fuel elements. Materials and components for examination can be introduced to the cell from the canal through a hatchway or directly from casks through the isolation area.

The necessary services provided for operation of the cells include the entrance canal, decontamination area, crane room, and equipment repair room. Access to these areas will be controlled through a monitored hot and cold change area. Similar control is provided for the Radiochemistry Laboratory previously mentioned. The hot cell will be used to perform the following operations:

1. Cut open capsules, fuel pins, and other small containers of radioactive material.
2. Visual and photographic examination of components and material cross sections on a macroscopic scale.
3. Disassembly of rod bundles.
4. Dimensional measurements of elements and rods.
5. Monitoring and scanning of gamma radiation from rods.
6. Fission gas collection and gamma analysis.
7. Wet chemical analysis for fuel burnup determination.
8. Preparation of gas and fuel samples for analysis in the radiochemical and spectrographic laboratories.
9. Preparation and examination of metallographic specimens.
10. Microhardness determinations.

The hot cell will be large enough so that power reactor fuel elements may be examined in the cell after service.

3.3.3. Use and Capability

The main function of the NFL will be to develop and test improved nuclear fuels and core components and to establish methods for their fabrication through the pilot stage. The Laboratory will be designed to develop materials, components and processes involving both virgin and recycled fuels. Initially, operations will center around rod-type oxide fuel for power reactors. Using the services of the BAWTR, these materials and components will be irradiation tested and subsequently examined in the NFL hot cells. Future operations may involve using new components, processes, and fuels. The design of the Facility is such that most fuels and components under investigation at the present time can safely and conveniently be handled in the NFL.

Figure 3.1. Plot Plan

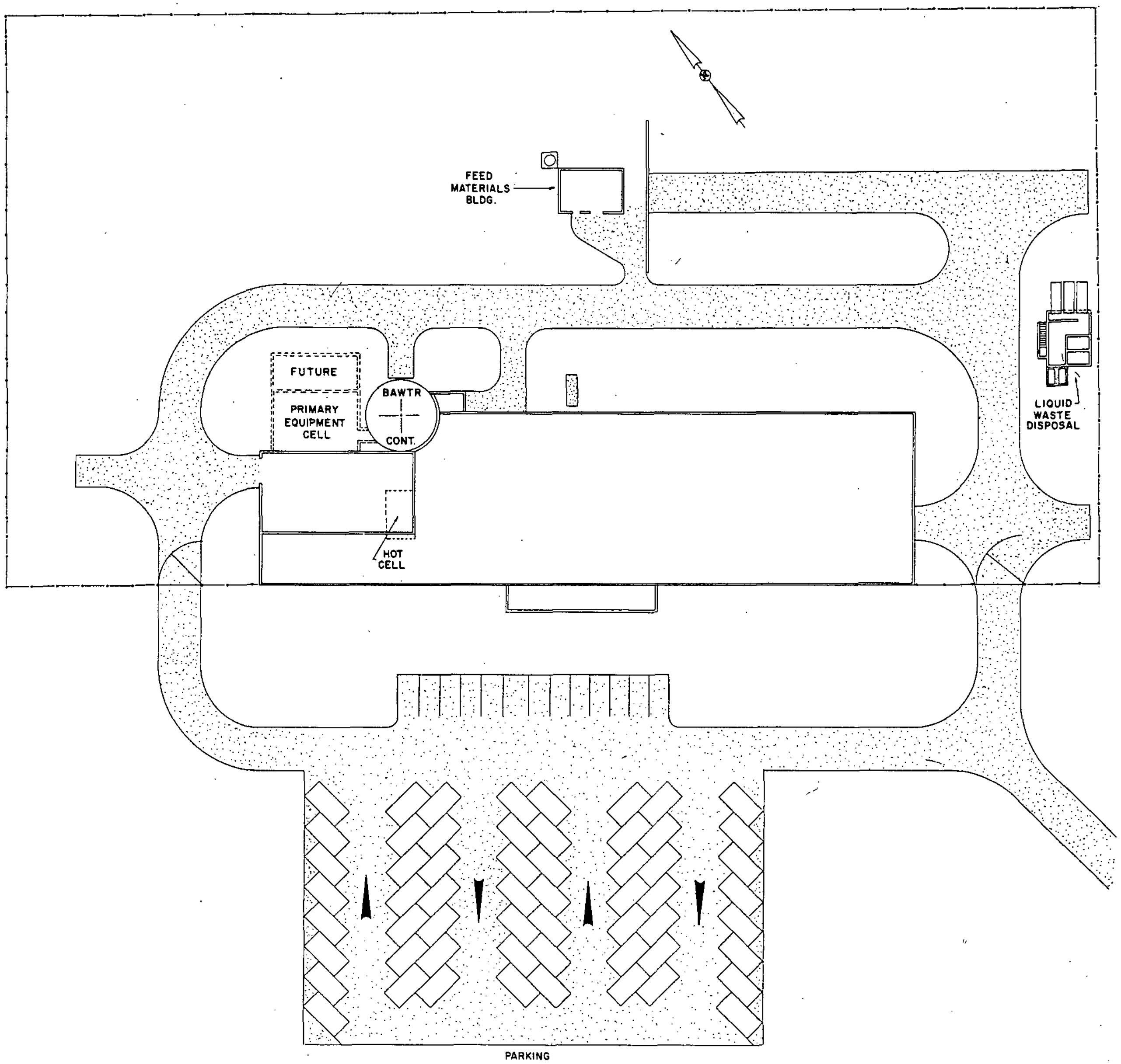


Figure 3.2. New Facility Floor Plans

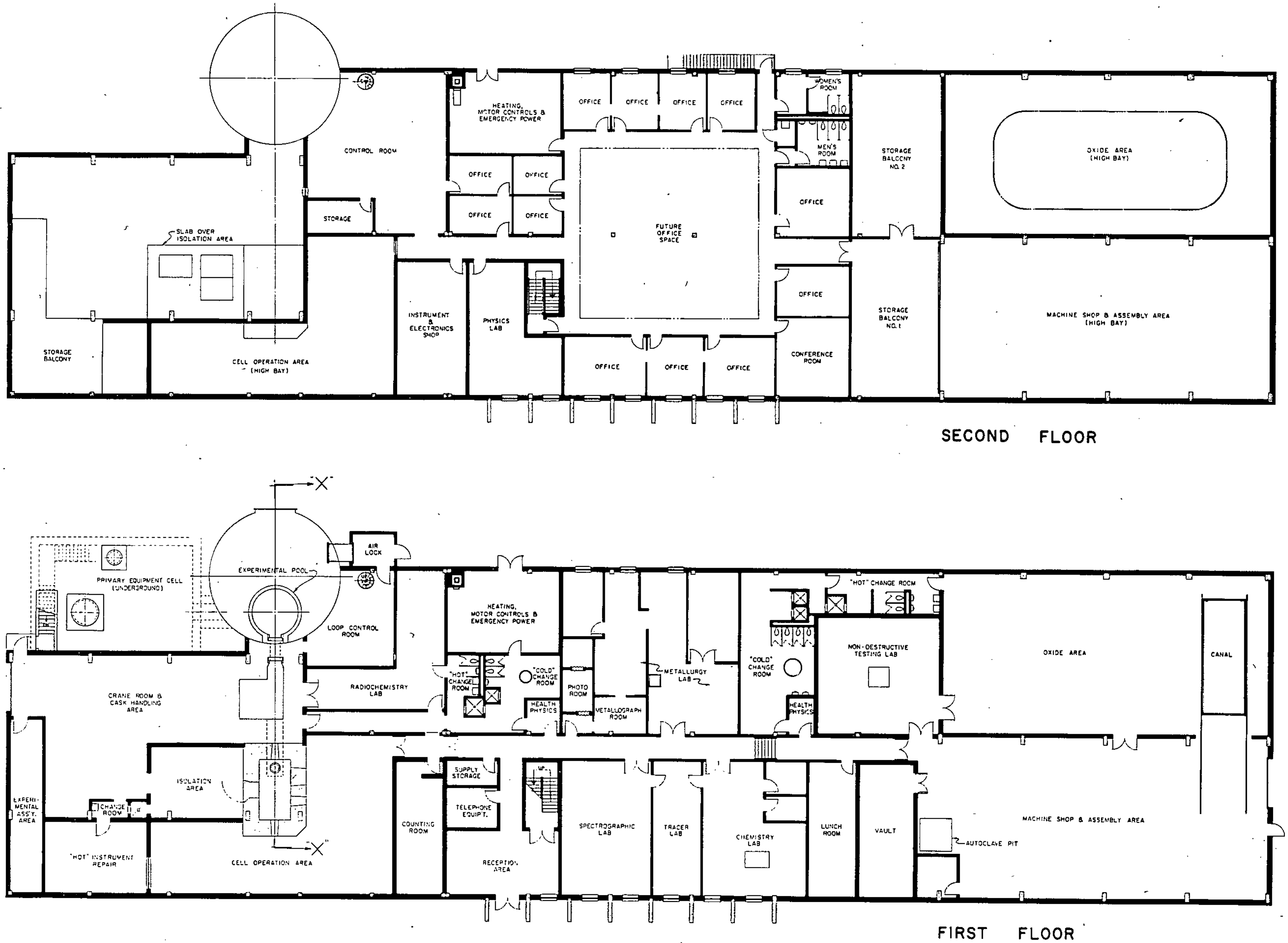
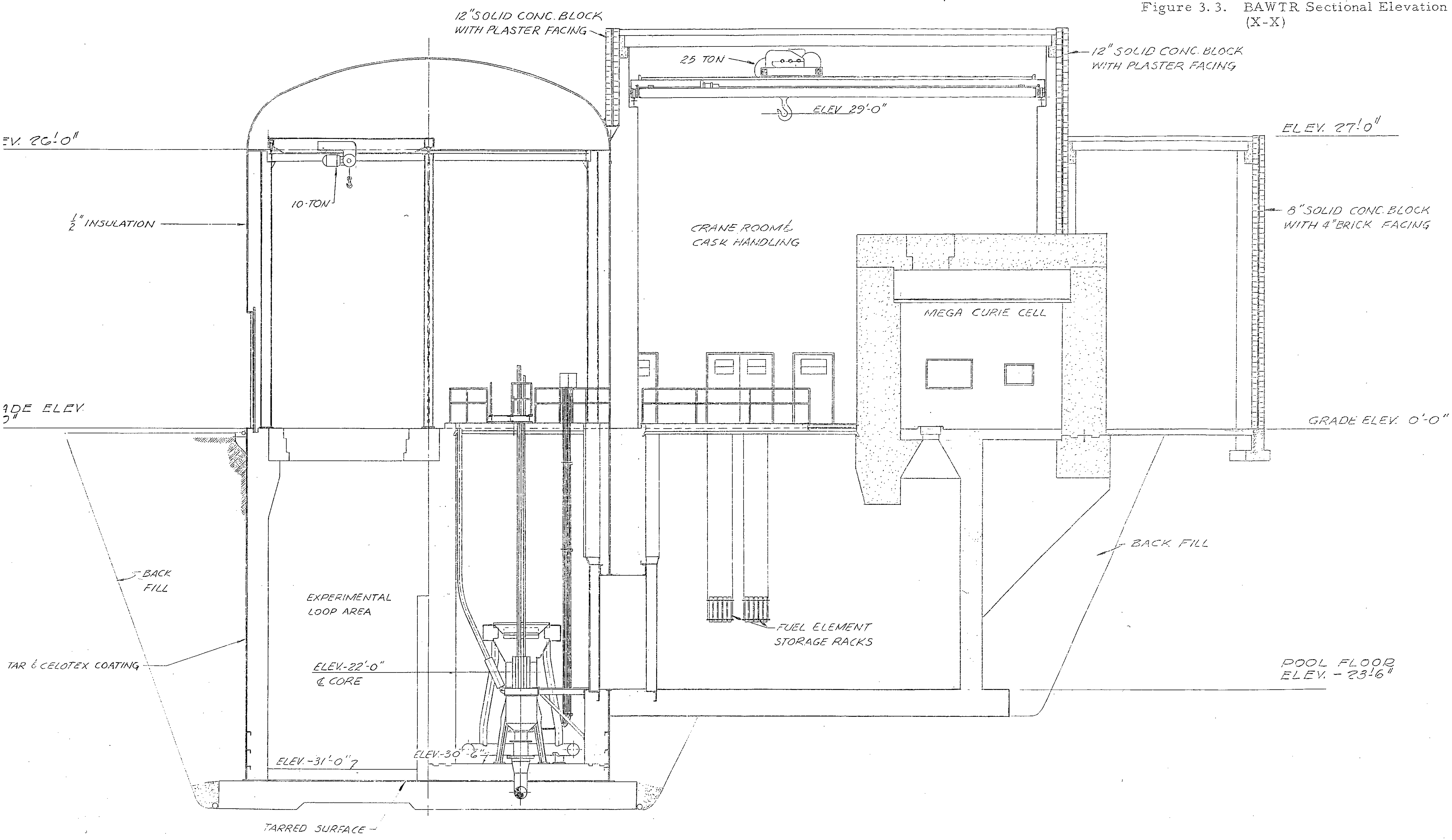
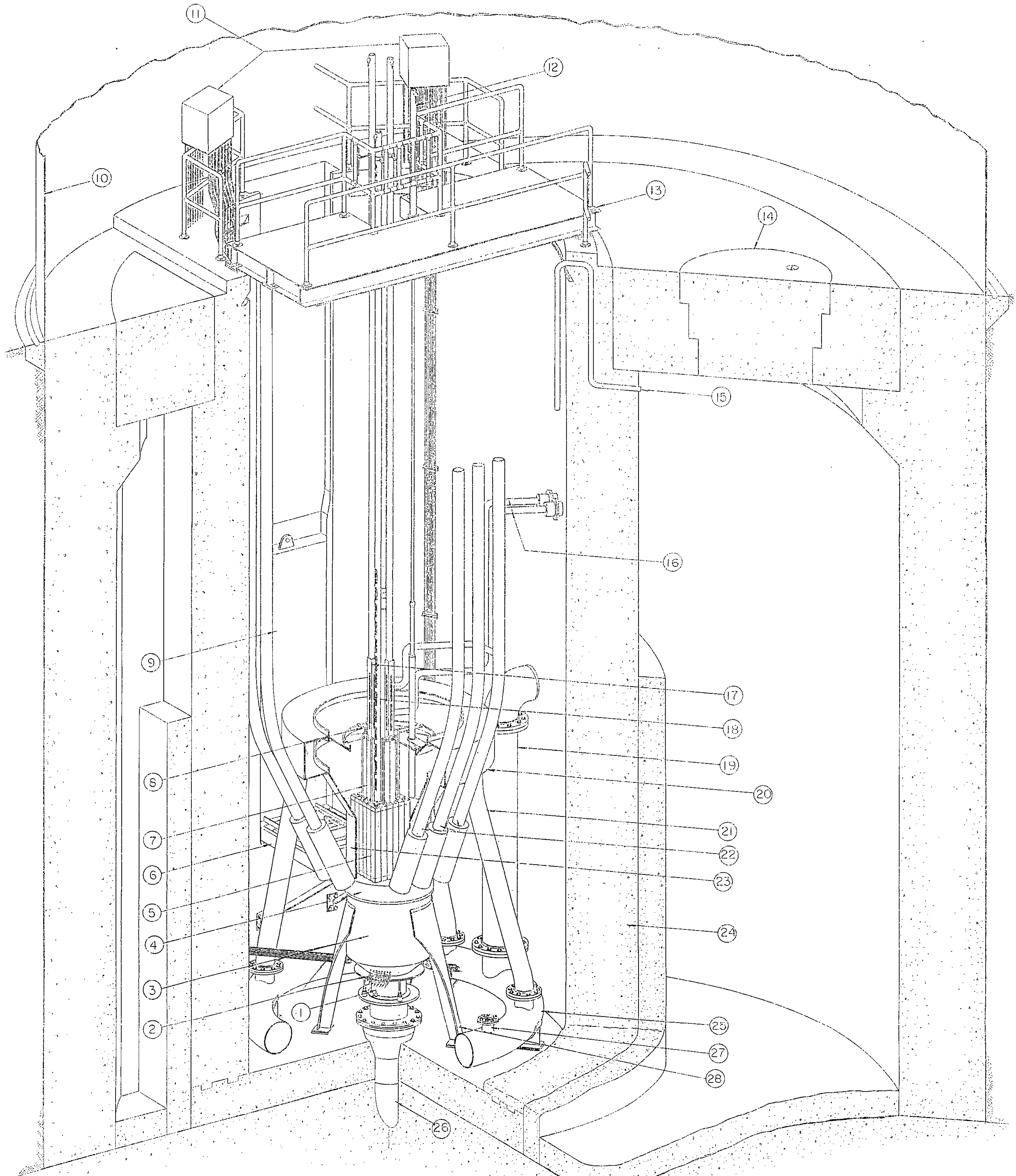


Figure 3.3. BAWTR Sectional Elevation (X-X)



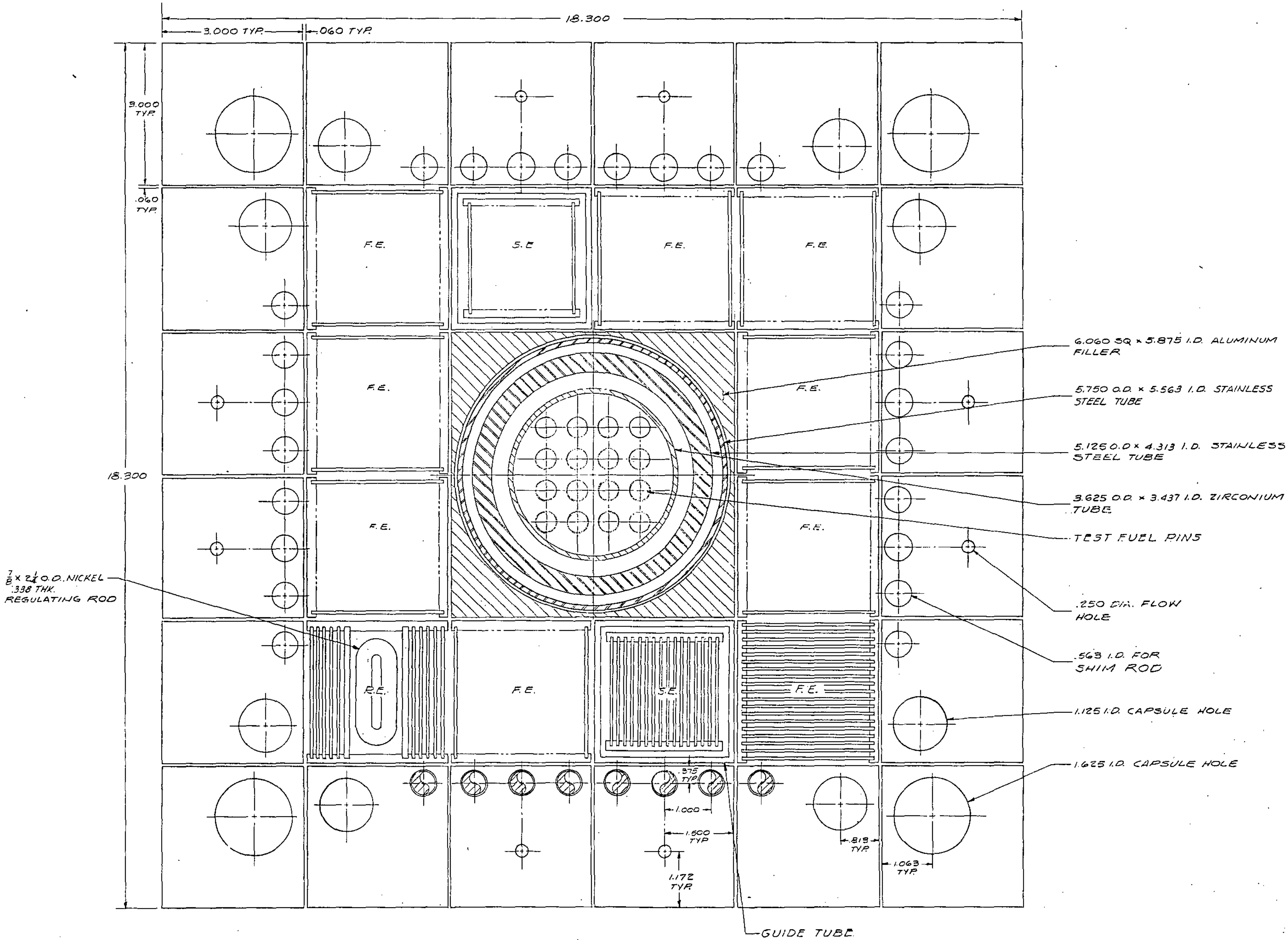
SECTIONAL ELEVATION "X-X"



1 PRIMARY OUTLET HEADER
2 SHIM ROD DRIVE CABLE CONDUIT
3 LOWER SHROUD
4 GRID PLATE
5 REACTOR CORE
6 TRANSFER CAR BRIDGE
7 SHIM ROD GUIDE
8 SHIM ROD GUIDE SUPPORT
9 POOL GATE
10 CONTAINMENT SHELL
11 SHIM ROD DRIVES
12 FISSION CHAMBER DRIVE
13 BRIDGE
14 ACCESS PLUG
15 VAPOR SUPPRESSION TUBE
16 HIGH PRESSURE LOOP INLET & OUTLET
17 SAFETY ROD DRIVE
18 REGULATING ROD DRIVE
19 PRIMARY WATER INLET
20 PRIMARY INLET PLENUM
21 PRIMARY SYSTEM RISER
22 ION CHAMBER CONTAINERS
23 UPPER SHROUD
24 POOL WALL
25 INLET DISTRIBUTION HEADER
26 PRIMARY PIPING
27 POOL RETURN
28 CORE SUPPORT LEGS

Figure 3.4. BAWTR General Arrangement

Figure 3. 5. Core Layout



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"Figure 3.6 BAWTR Layout Vertical Section"

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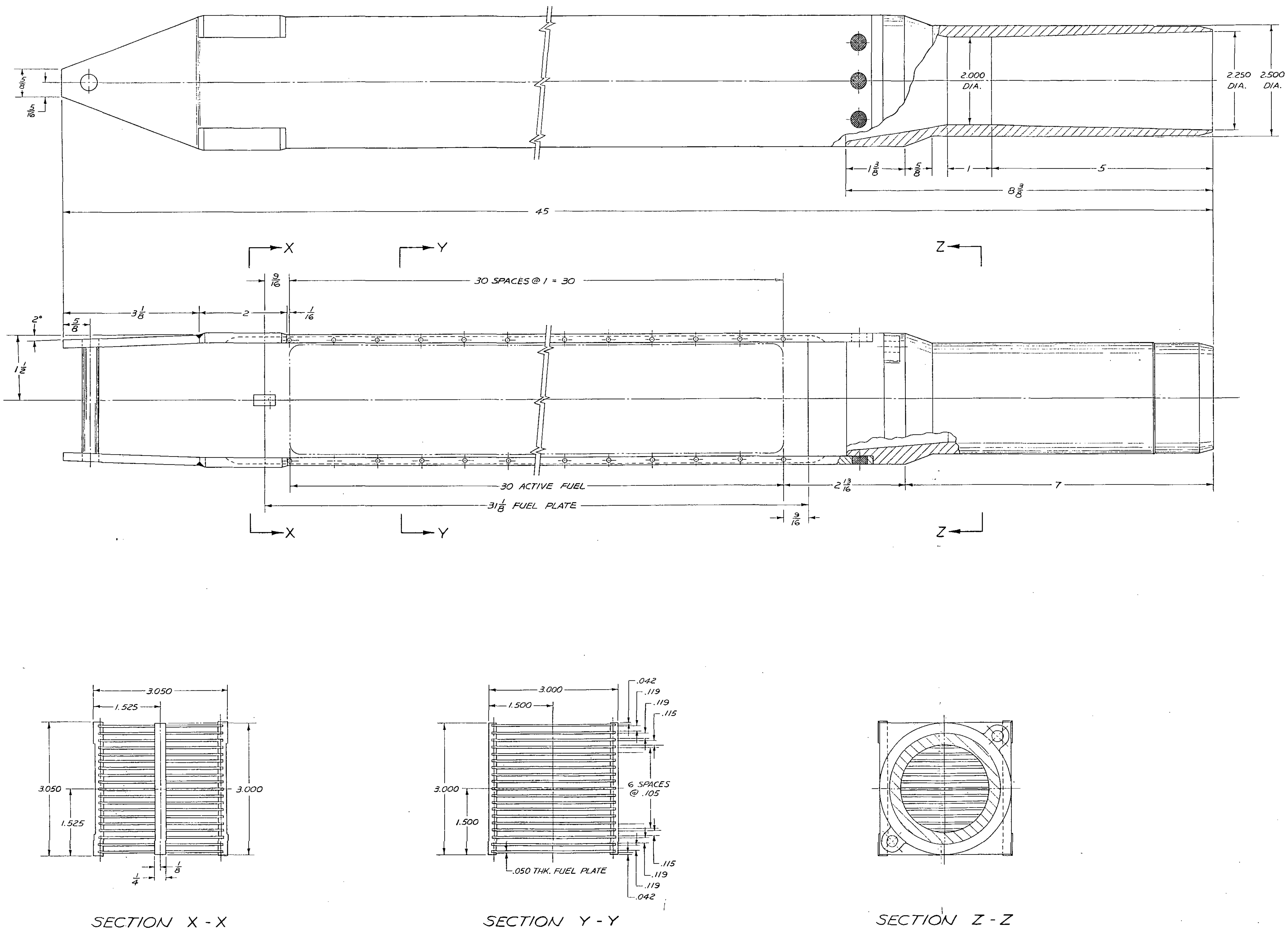
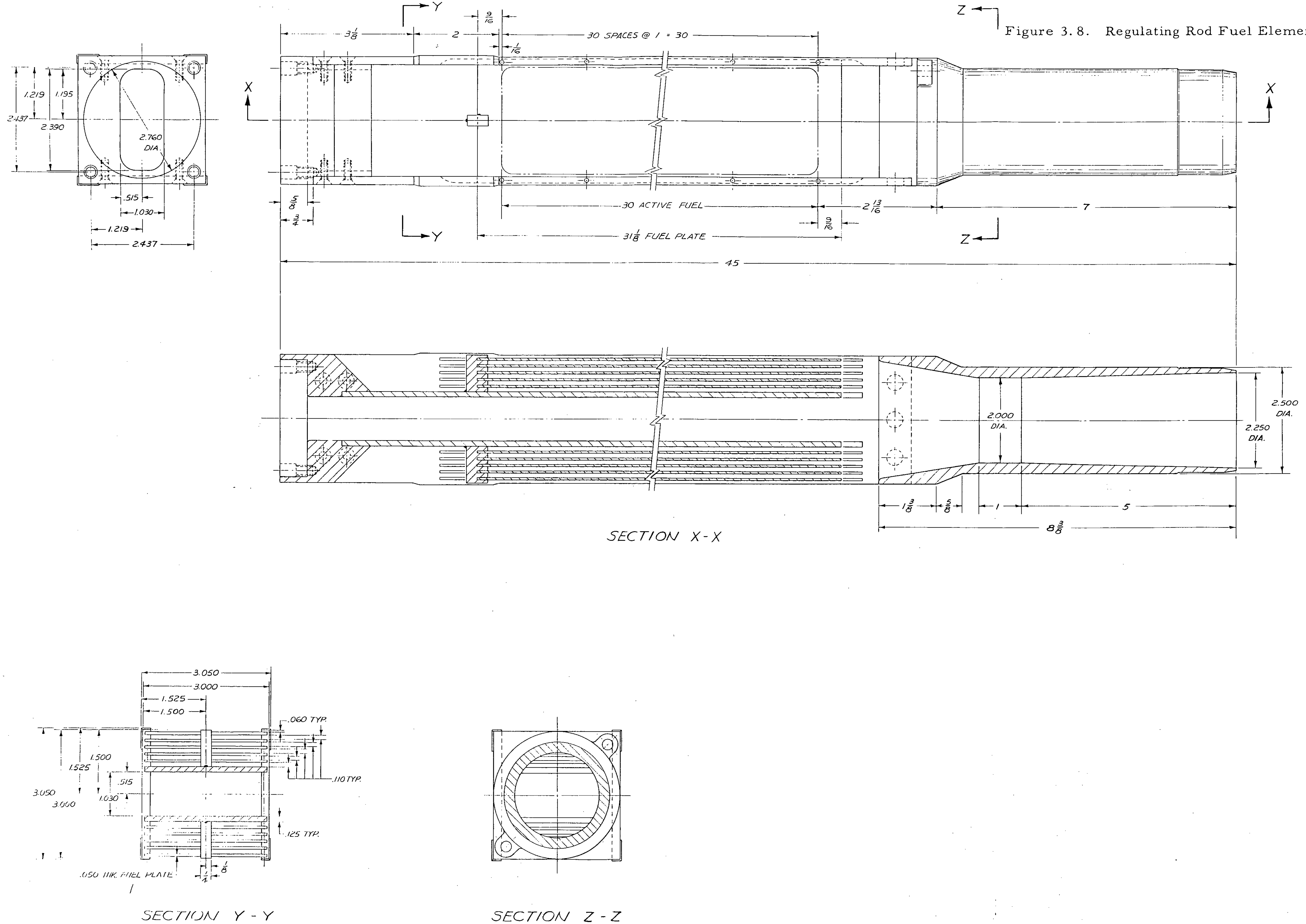

2.000 DIA.
2.250 DIA.
2.500 DIA.
45
30 SPACES @ 1 = 30
30 ACTIVE FUEL
31 1/8 FUEL PLATE
.050 THK. FUEL PLATE
6 SPACES @ .105
SECTION X - X
SECTION Y - Y
SECTION Z - Z

Figure 3.8. Regulating Rod Fuel Element



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“Figure 3.9 Safety Element and Guide Tube Details”

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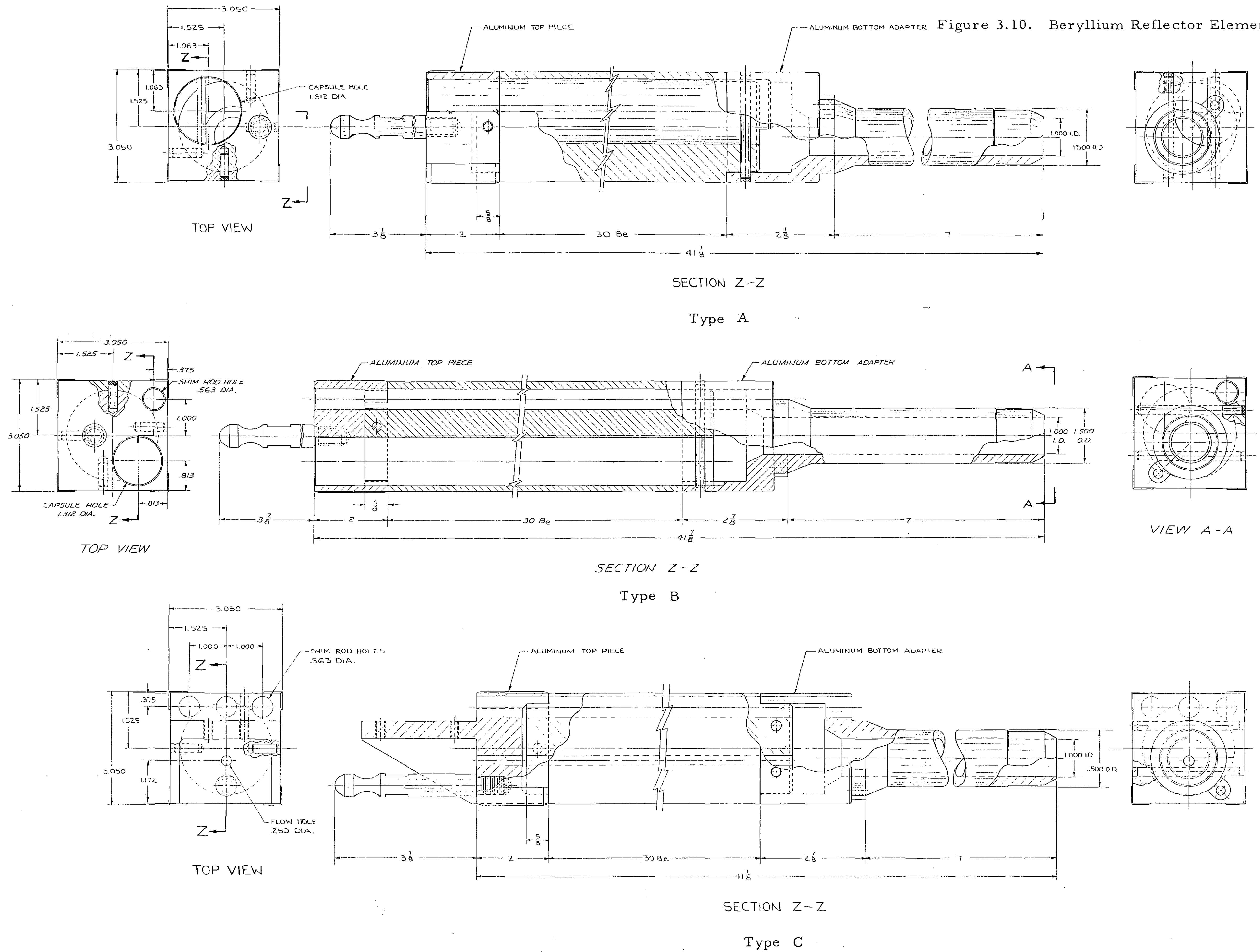


Figure 3.10. Beryllium Reflector Element

Figure 3.11. Upper Shroud and Filler Blocks

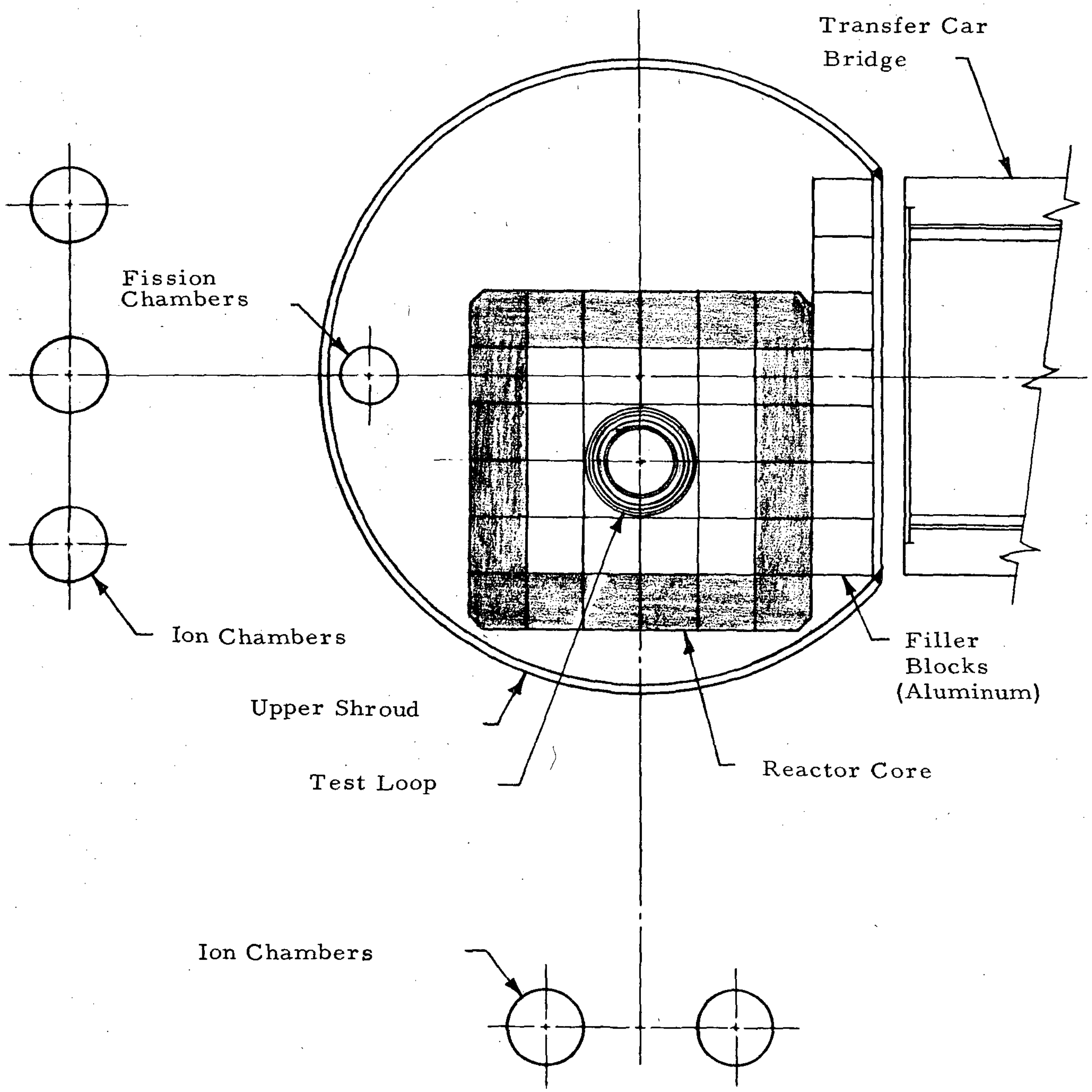
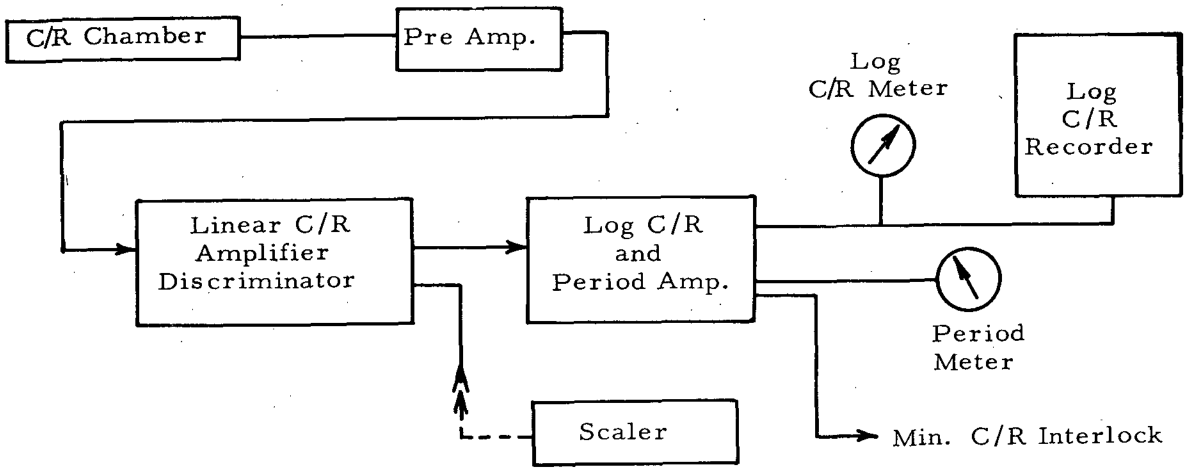
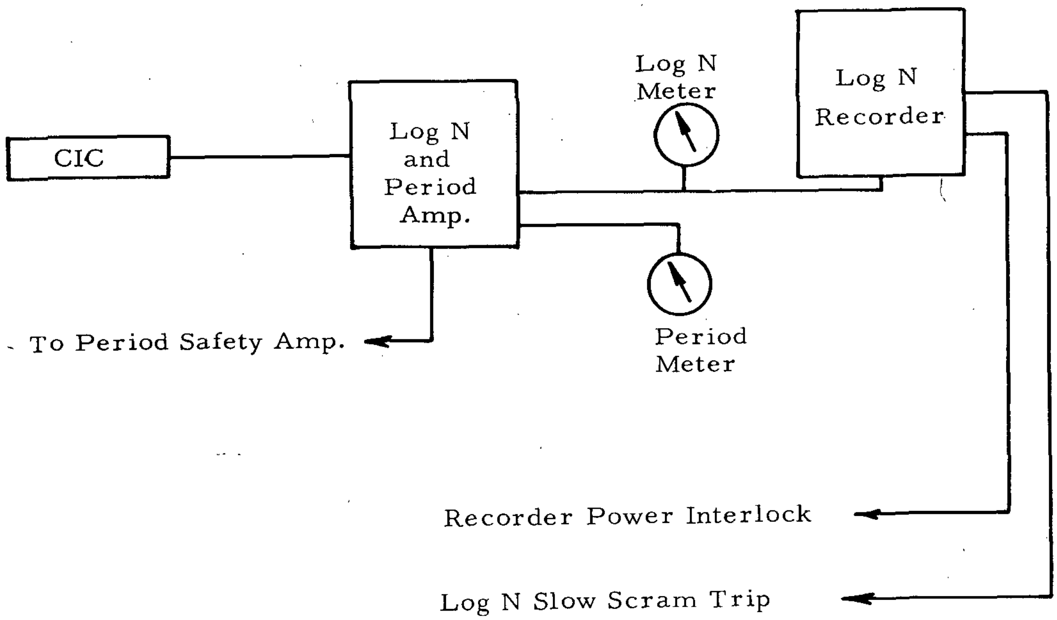


Figure 3.12 A. Nuclear Instrumentation — Log N Channel and Startup Channel



Start-Up Channel



Log N Channel

Figure 3.12 B. Nuclear Instrumentation — Linear Level and Automatic Control

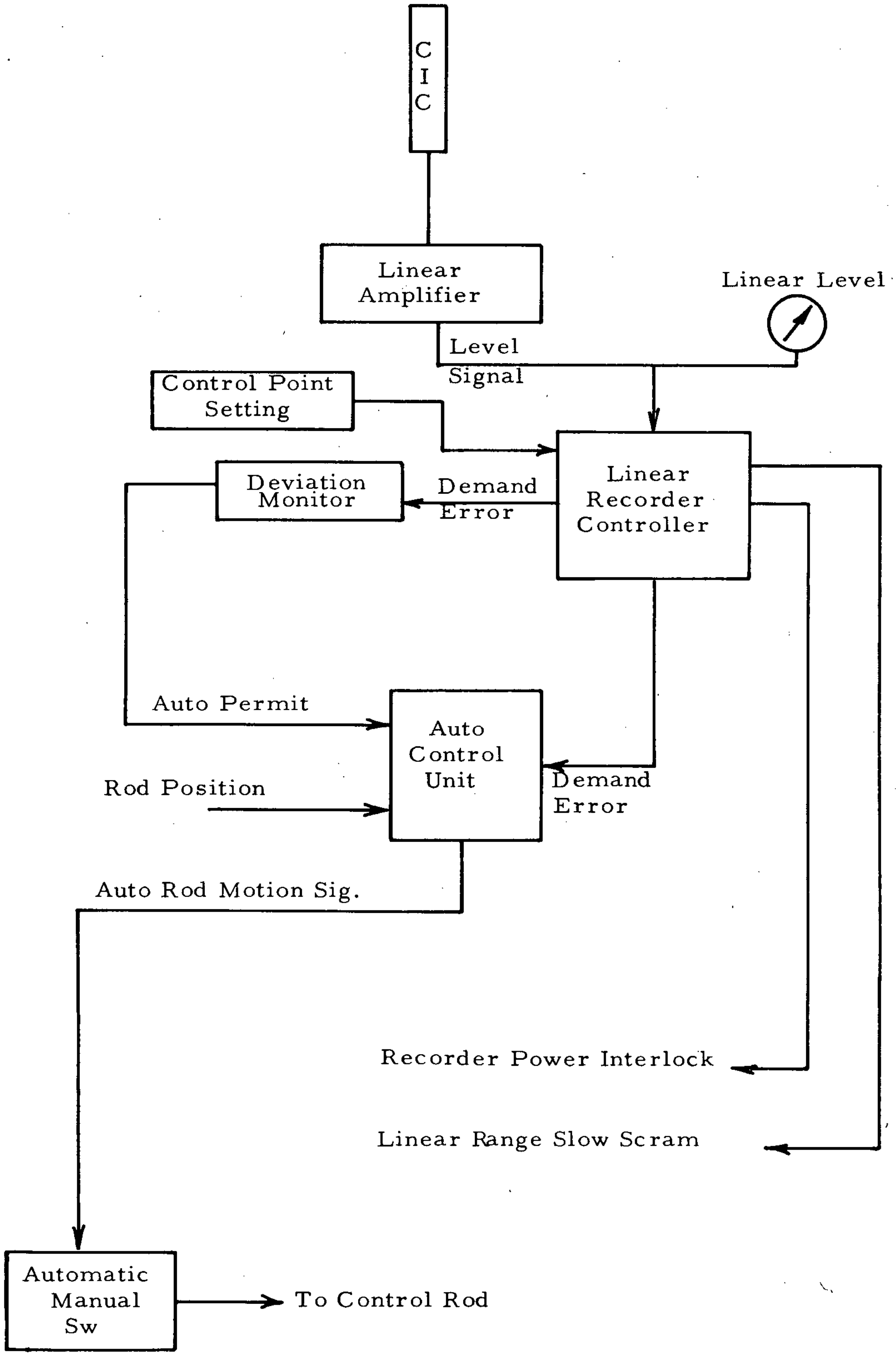


Figure 3.12 C. Nuclear Instrumentation — Safety System

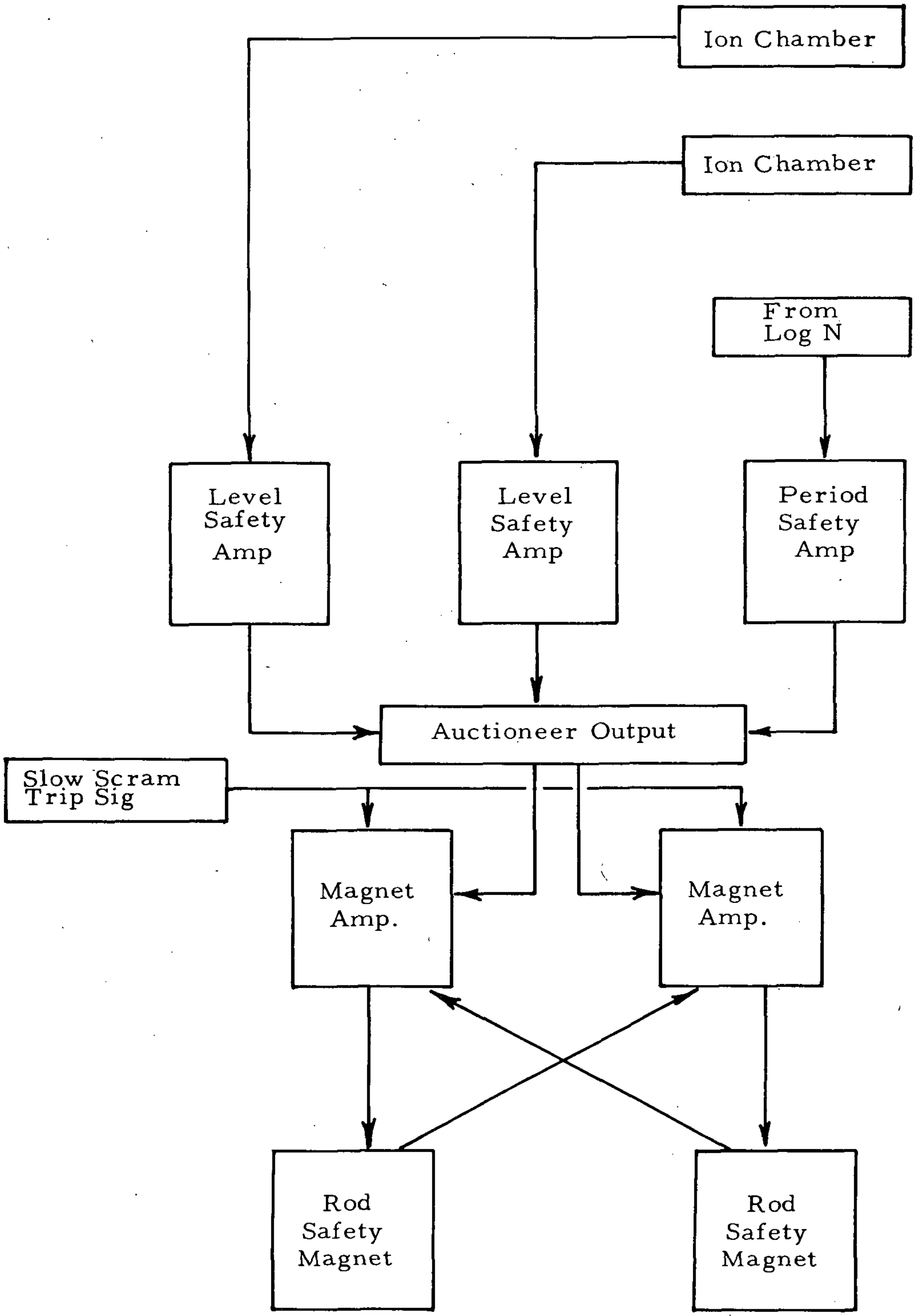


Figure 3.13. Nuclear Instrumentation — Channel Ranges

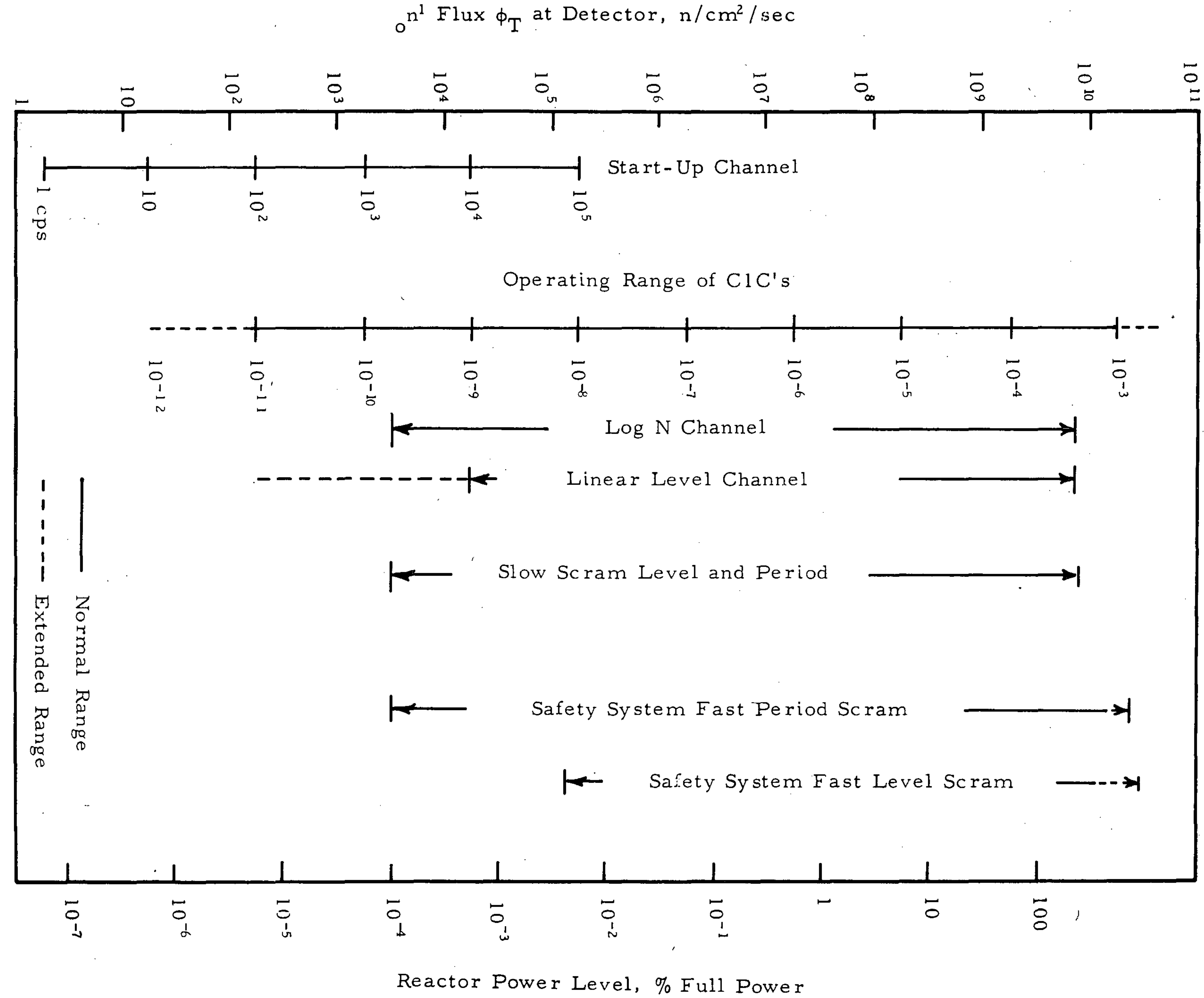
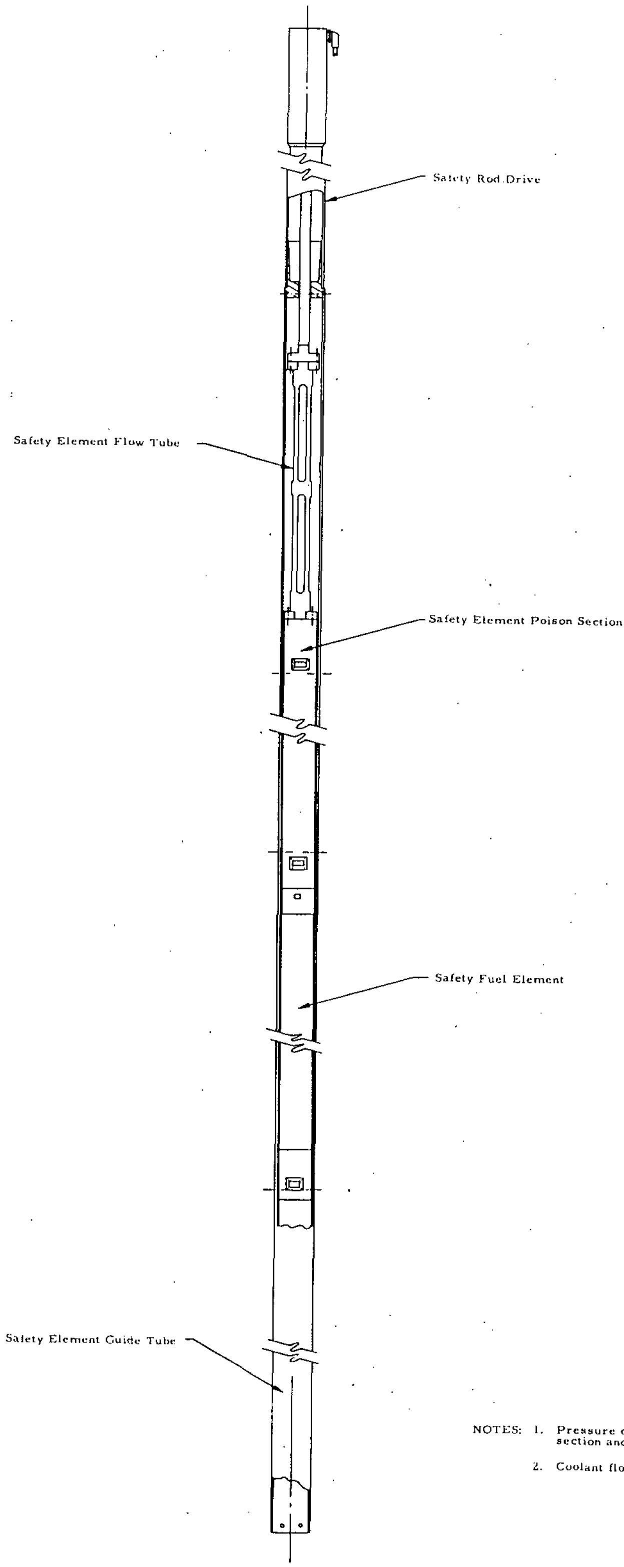


Figure 3.14. Safety Element



NOTES:
1. Pressure drop thru flow tube, poison section and element 3-25 psi.
2. Coolant flow 102,000 #/hr.

Figure 3.15. Safety Rod Drive

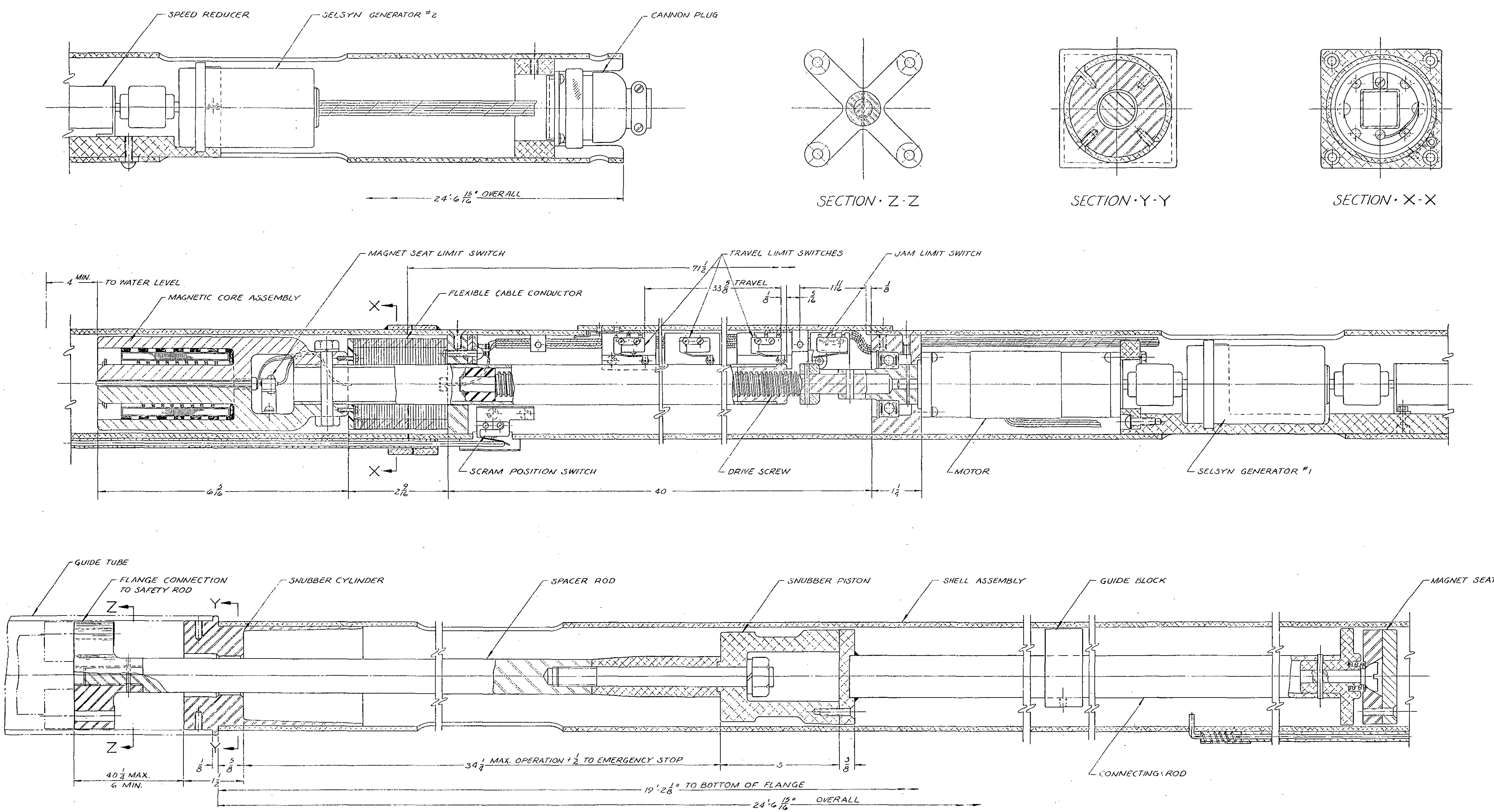


Figure 3.16. Regulating Rod and Drive Unit

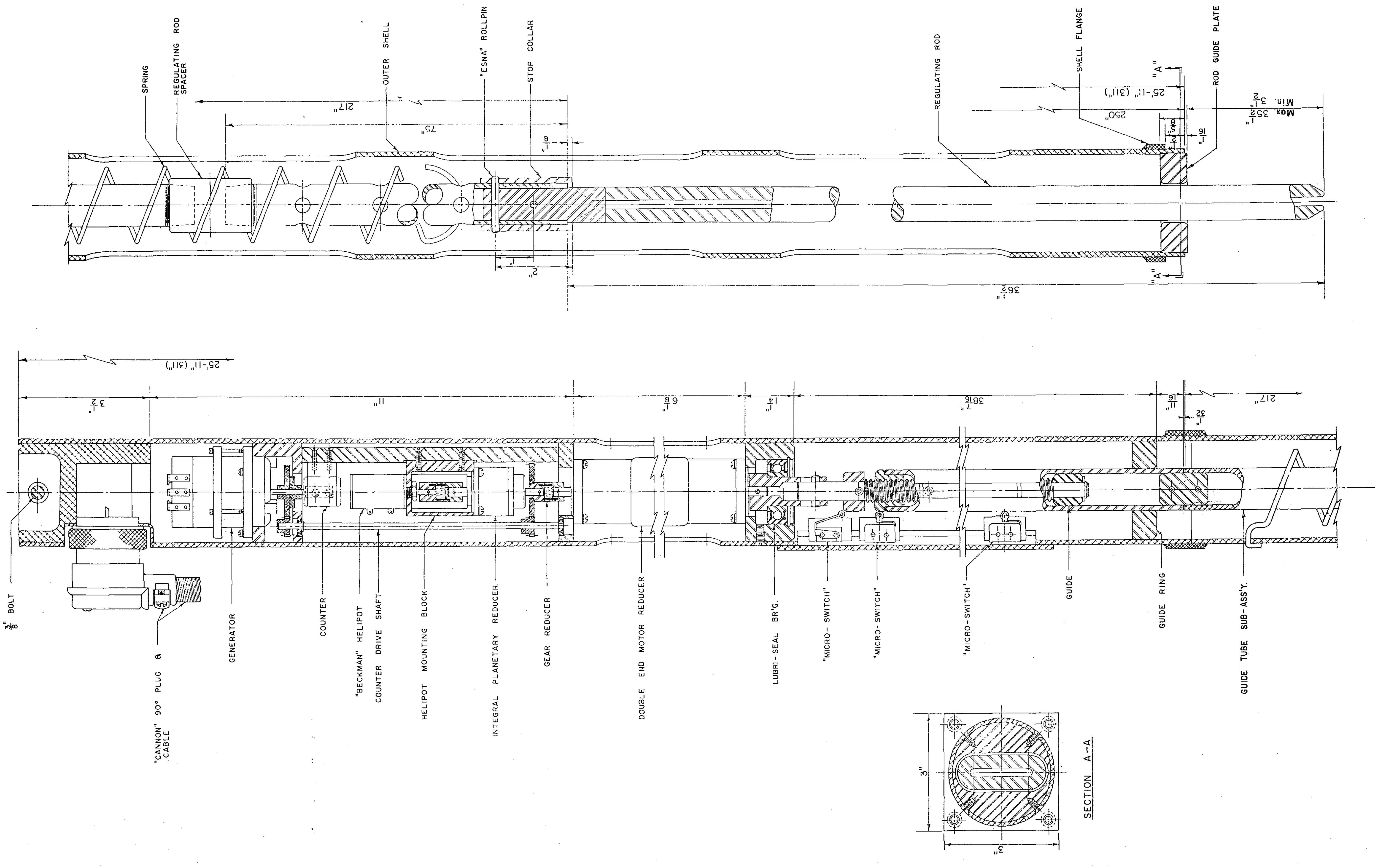


Figure 3.17. Shim Rod and Drive

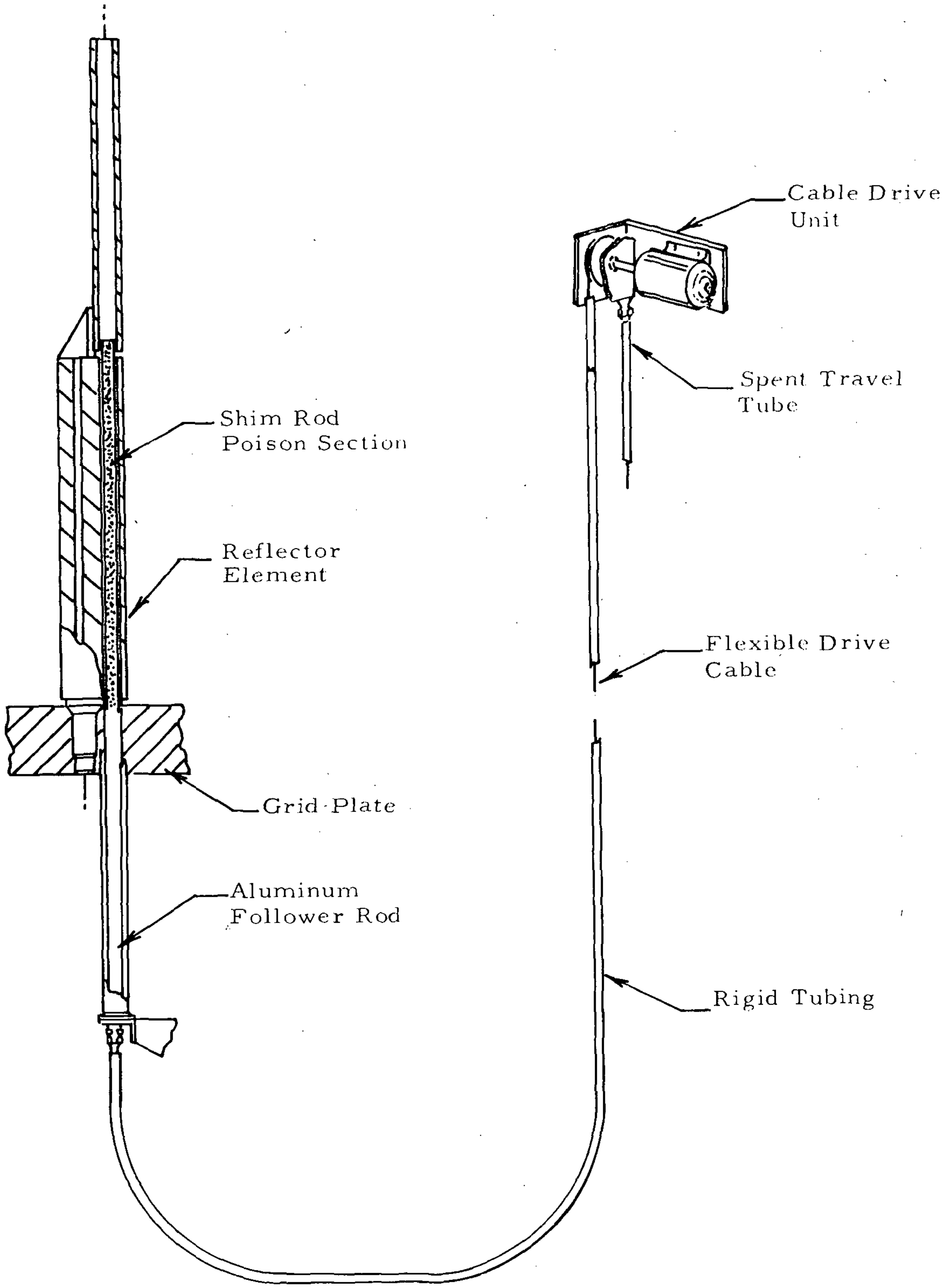


Figure 3.18. P/P_{avg} Along the Hottest Fuel Plate

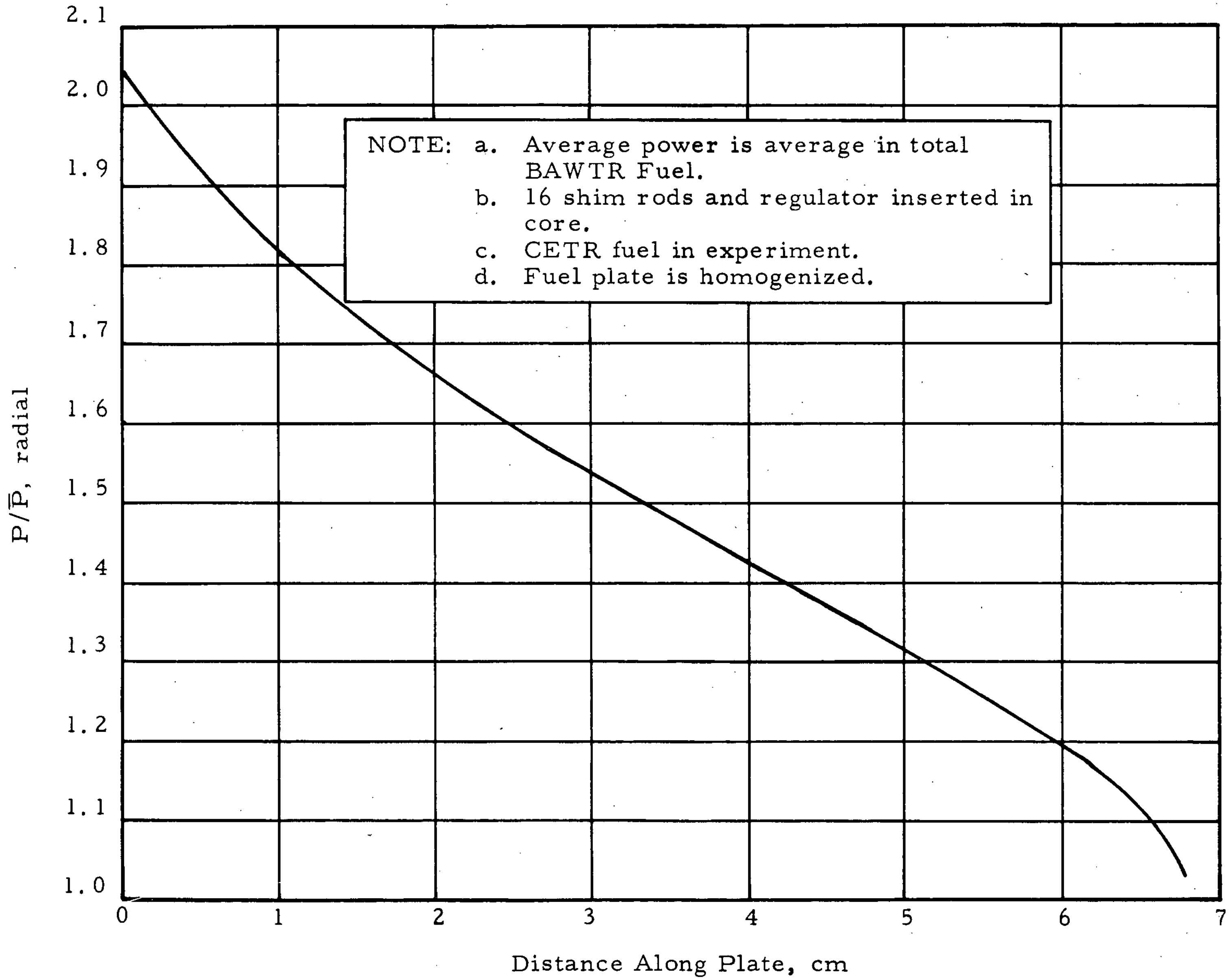


Figure 3.19. Safety Rod Worth Vs Percentage Insertion

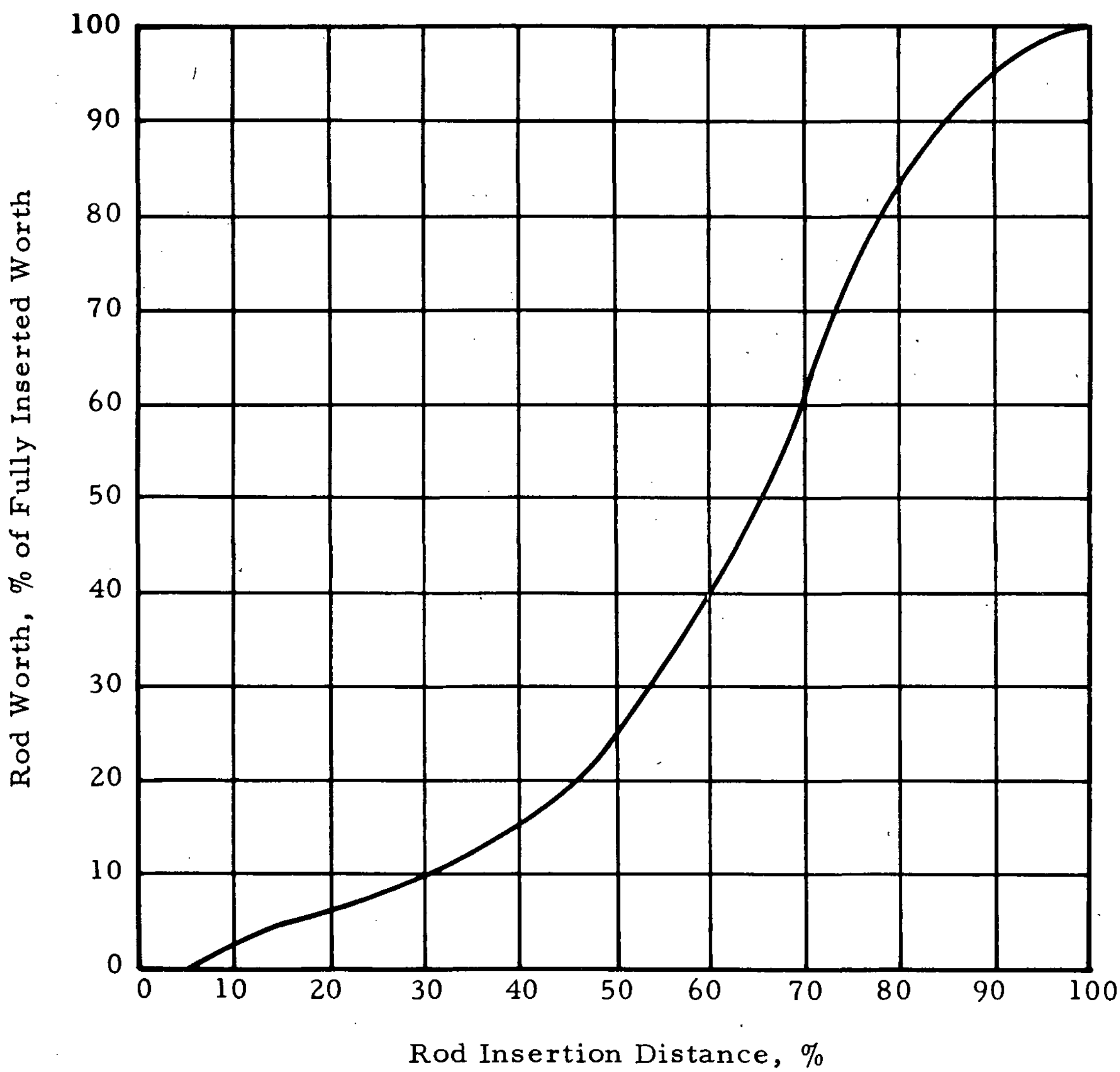


Figure 3.20. k_{eff} Vs Core Displacement

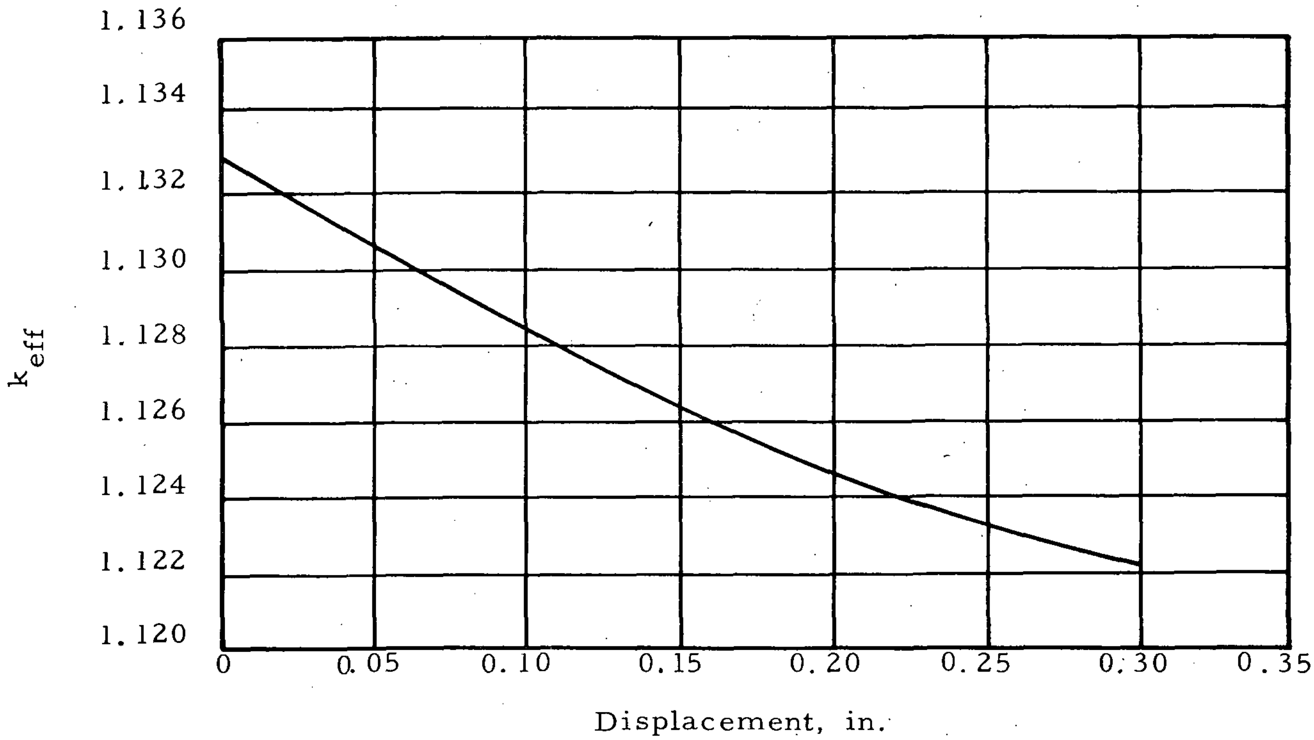


Figure 3.21. k_{eff} Vs Core Temperature

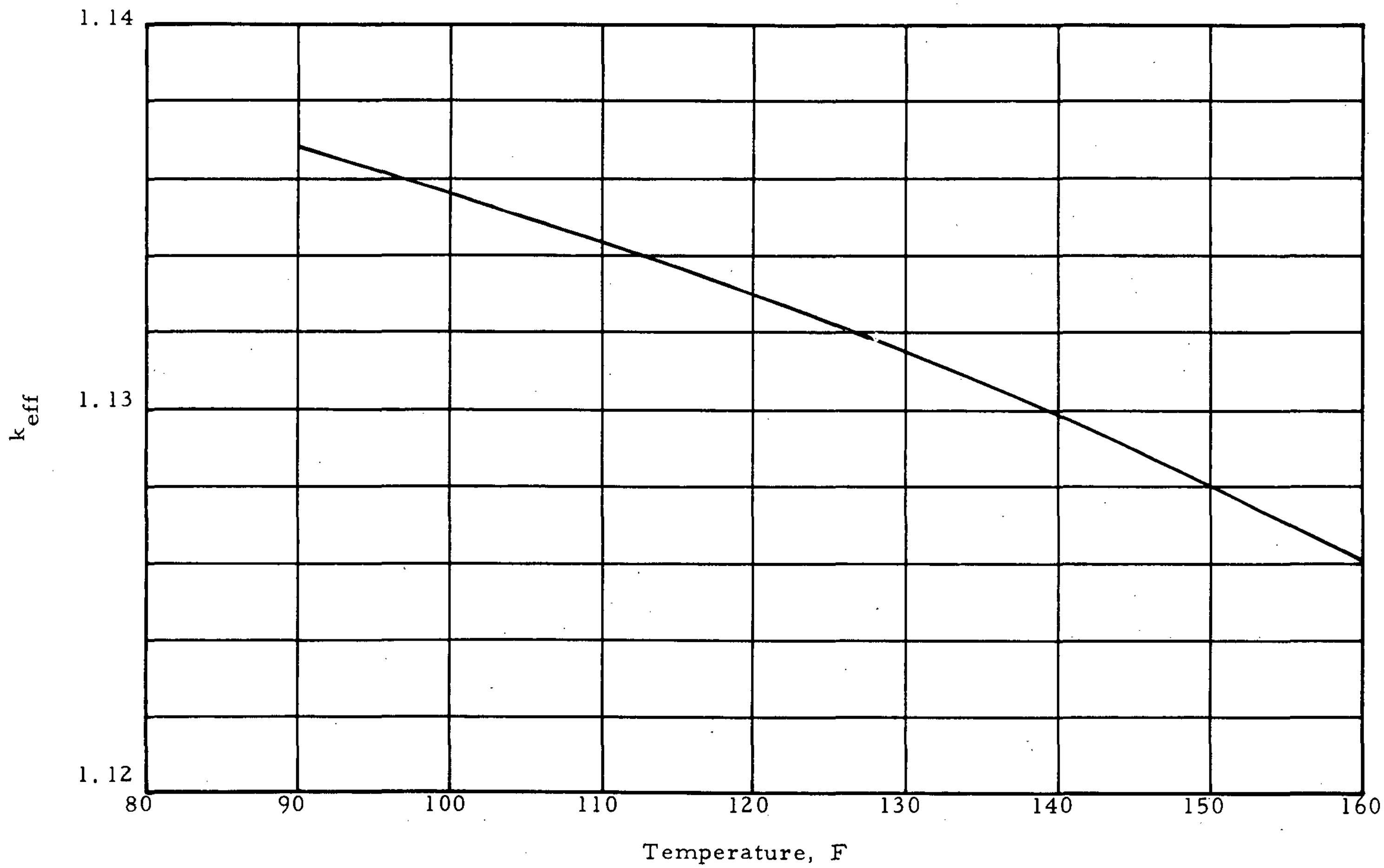


Figure 3.22. k_{eff} Vs Annulus Temperature

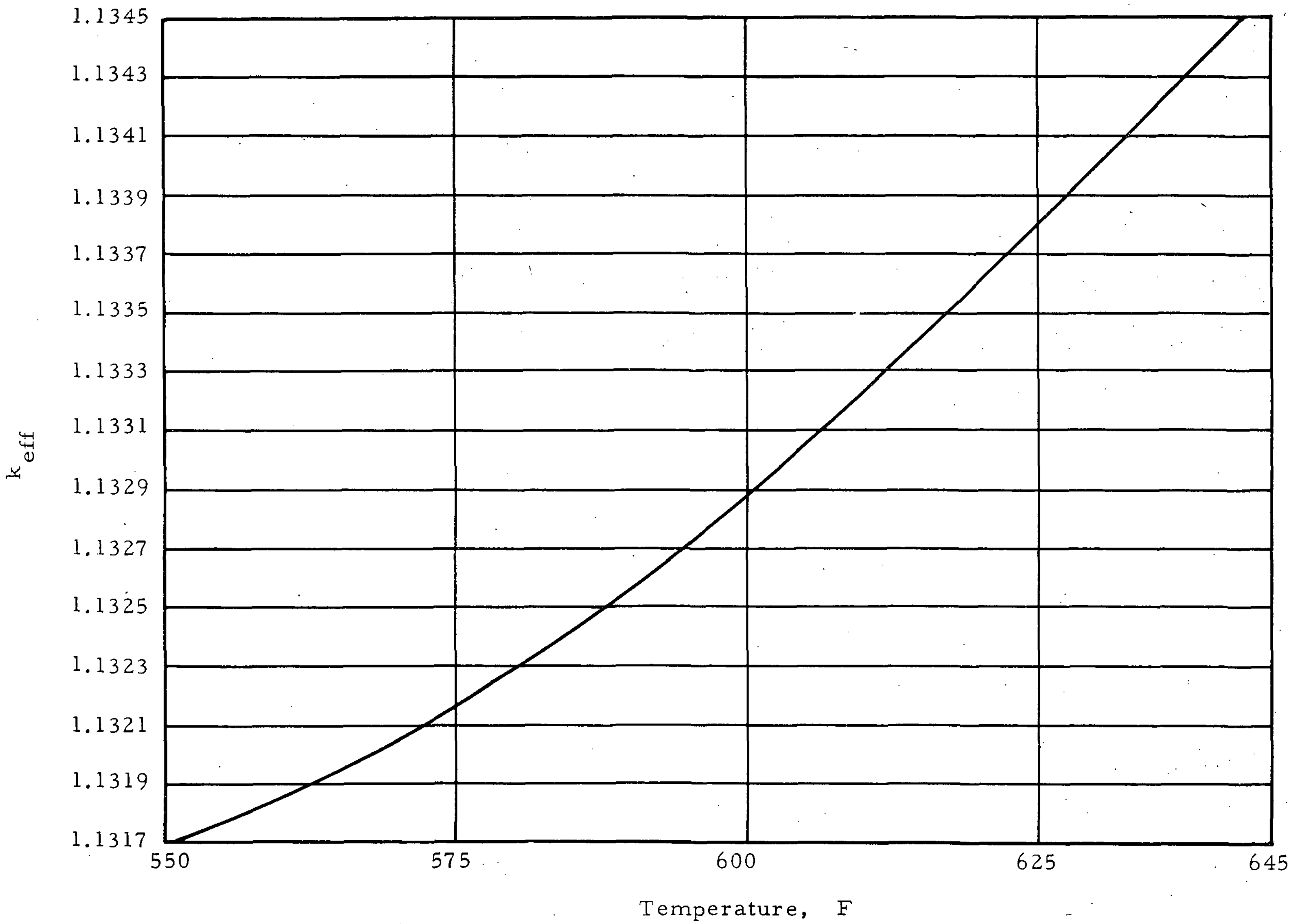


Figure 3.23. k_{eff} Vs Percentage Void in Experiment (CETR Type Fuel in Experiment)

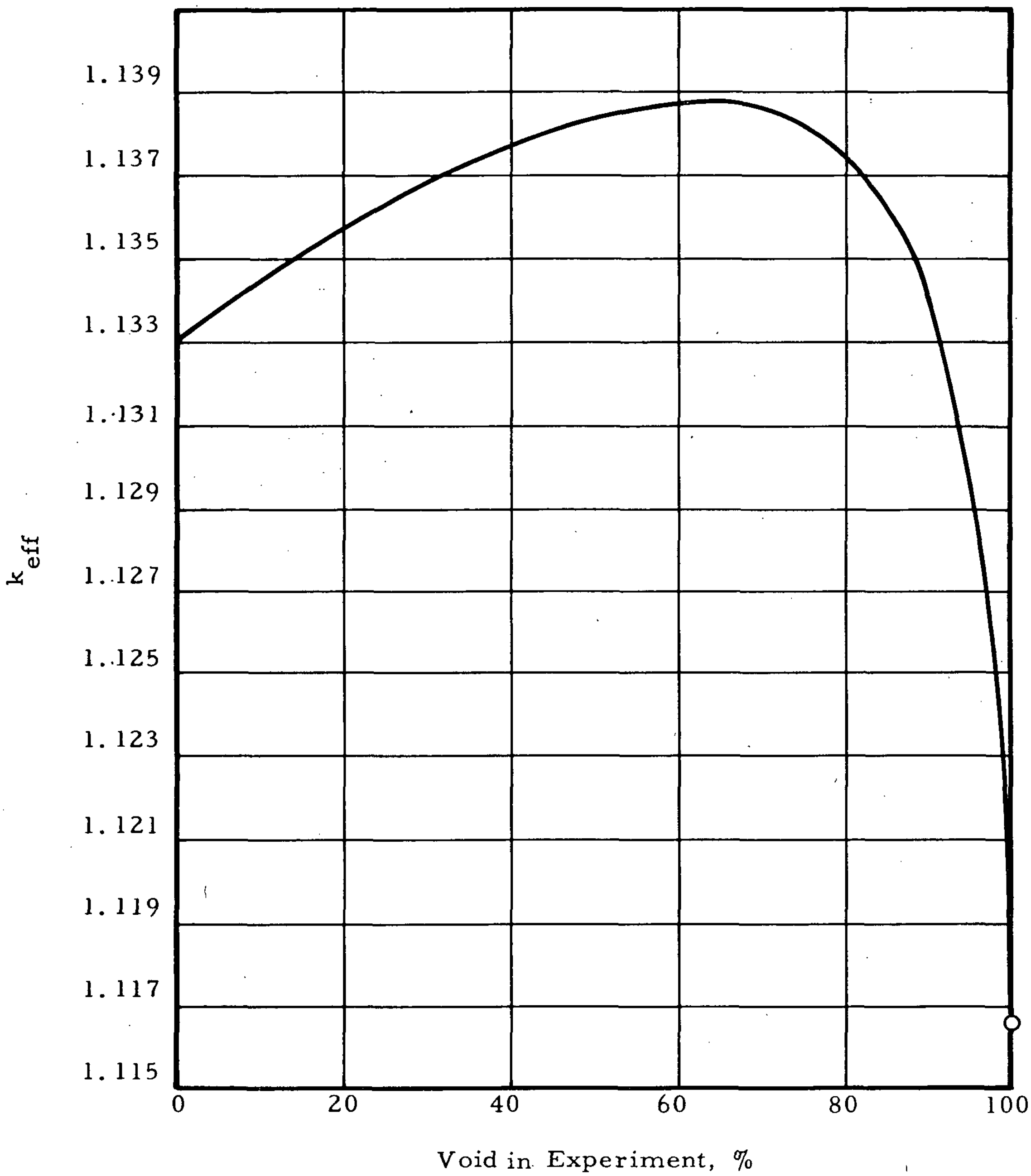


Figure 3. 24. Thermal Neutron Flux Vs Distance Across Experiment
(CETR Type Fuel in Experiment and 8 Shim Rods in Core)

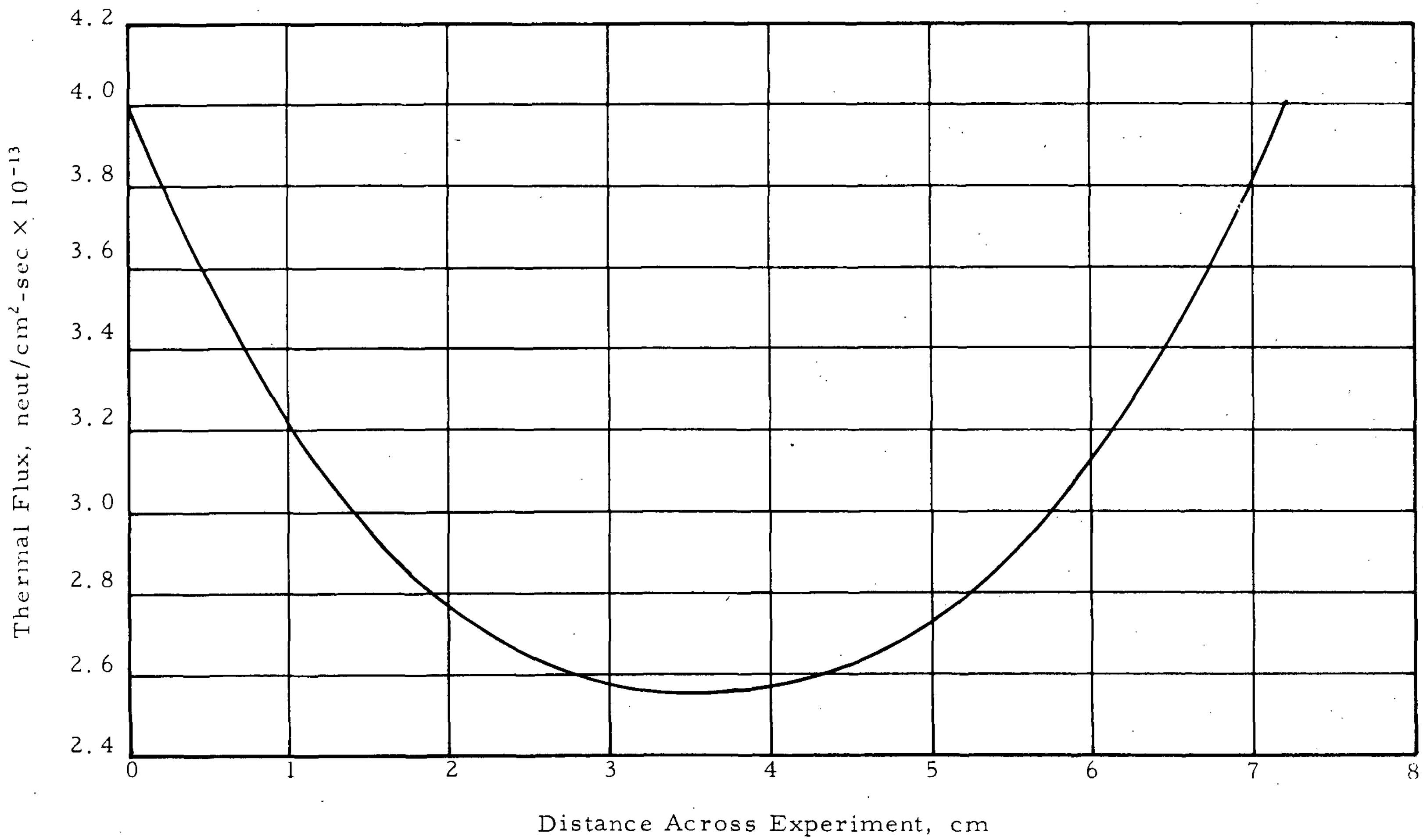


Figure 3. 25. Radial Thermal Flux Profile

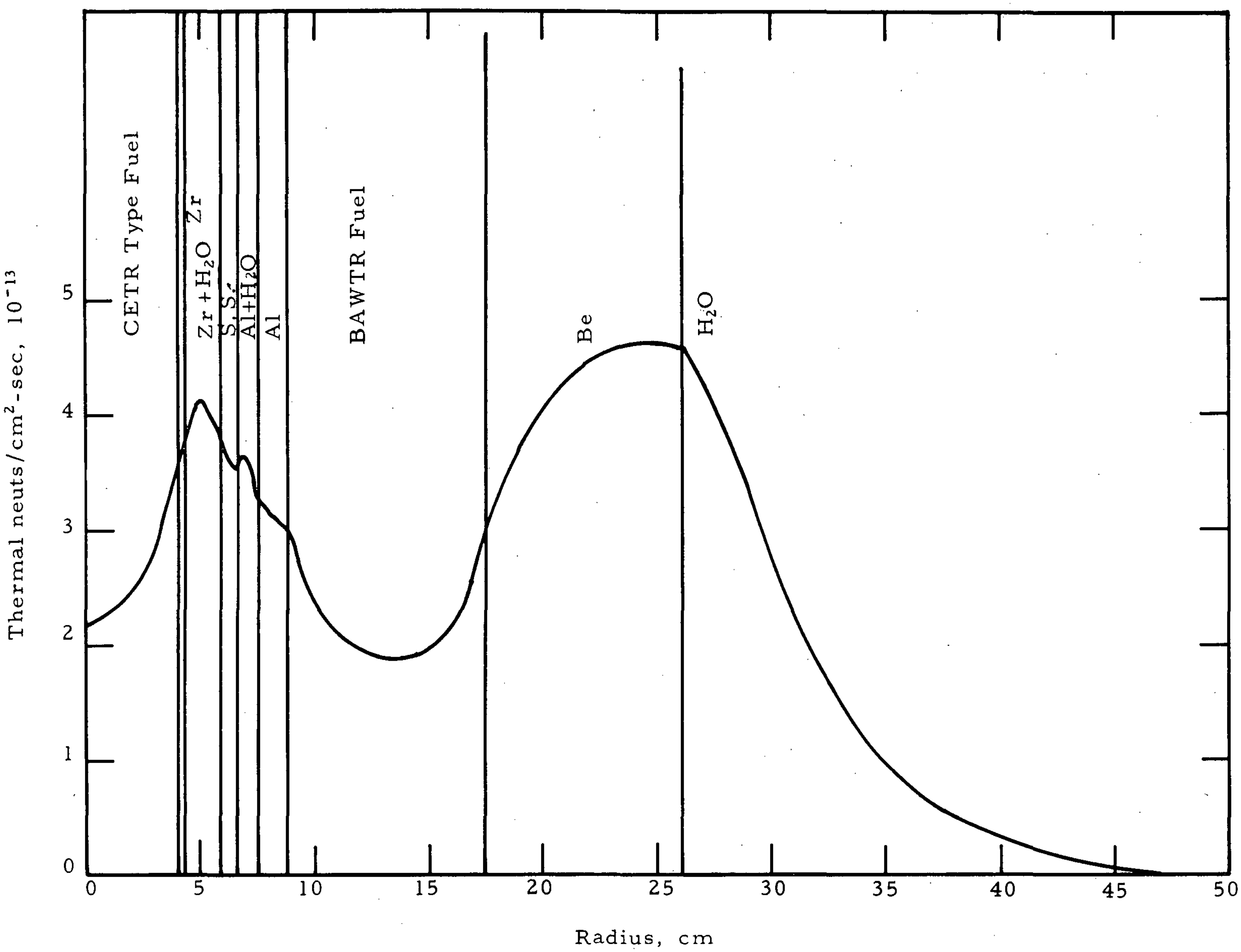


Figure 3.26. k_{eff} Vs Core Lifetime
(CETR Type Fuel in Experiment)

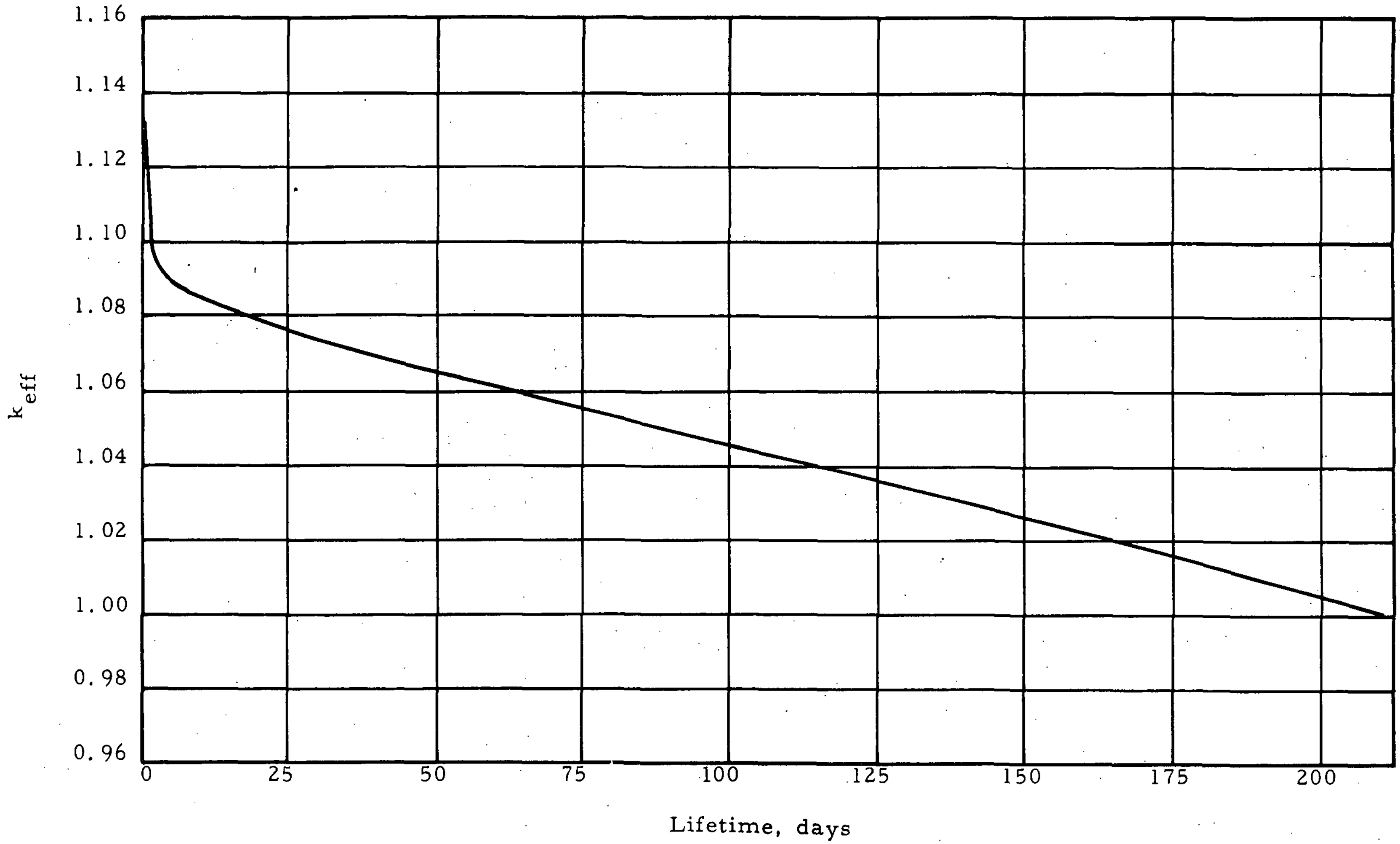
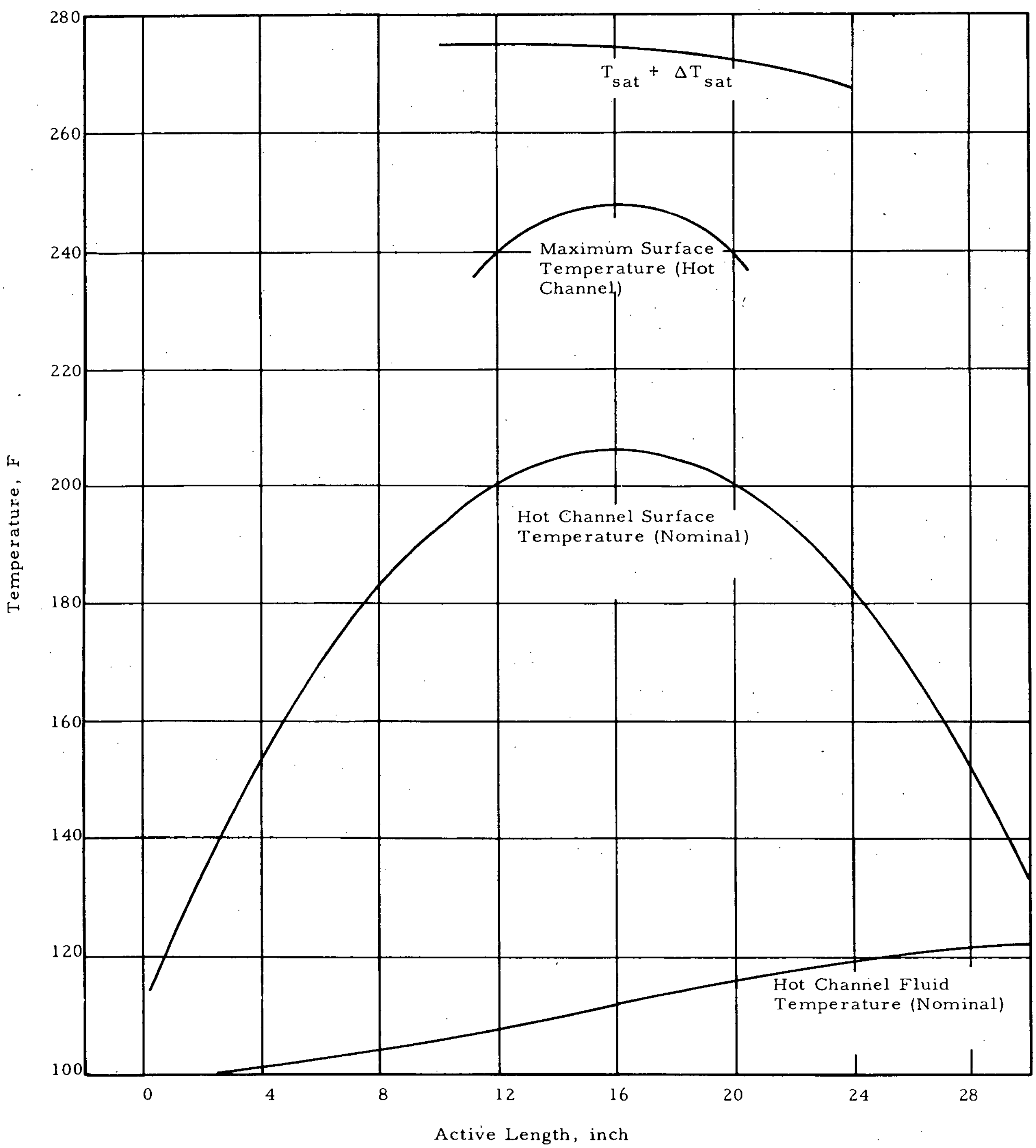


Figure 3. 27. Fuel Element Axial Temperature Profile



4. AUXILIARY SYSTEMS

4.1. Primary Cooling System

The reactor core, beryllium reflectors, core shroud, and water surrounding the core is cooled by circulation of the pool water through a heat exchanger. The demineralized pool water is pumped down through the core and reflector, through lower shroud and outlet header, then through the primary pump and shell side of the exchanger back into the pool through a distribution header located above the reactor core. A side stream of 25 gpm is continuously run through the water purification system to maintain cooling water purity. Figure 4.1 is a flow diagram of the reactor cooling systems.

The outlet header (Fig. 3.6), located directly beneath the reactor core, directs flow through the core during the operation of the pumps and permits natural convection cooling when forced cooling is not used. The outlet header, under loss of forced circulation flow, drops down and permits the core to be cooled by natural convection. The header is normally held up by the pressure loss through the core, and is interlocked with the primary pump so that the pump cannot be started with the header down. It is also interlocked with the reactor control system so that the reactor cannot be operated above 100 kw with the header in the down position. The header may be manually closed at reactor startup.

The primary piping is 16-inch OD schedule 40 aluminum. The lines penetrate the containment shell and are sealed and tested to maintain the integrity of the shell. The coolant exit line connects to the bottom of the suction header and to the intake side of the primary pump. The inlet line enters the pool above the reactor core where it then drops in the pool and is connected to a distribution header. The header and primary inlet plenum permit an even distribution of water above the reactor core to prevent a turbulence in the pool water which would release a large amount of dissolved gas into the containment vessel.

To prevent syphoning of the water from the pool in case of an accident, the exit line will have a stop valve located just inside the primary equipment cell. Syphoning from the inlet line is prevented by a loop in the line extending above the core. Therefore, loss of pool water from syphoning is prevented. The valves and flanges used in the primary system are of class 150-lb aluminum or stainless steel. The design pressure for these is about 180 psi at 150 F. The water is circulated through the primary system at a rate of 3800 gpm by the primary coolant pump. The pump is designed to supply 3800 gpm coolant water at 42.5 psi and is operated by a 125-hp electric motor which takes its power from the plant electric system. There is no auxiliary or emergency power to the pump water; on loss of electrical power or coolant flow, the reactor automatically scrams. Following normal pump coastdown, it is cooled by natural convection circulation of the pool water.

The heat exchanger is a shell and tube type fabricated of aluminum. It provides cooling capacity for the reactor and experiments, and is designed to withstand an interval pressure of 85 psig. The flow of coolant through the heat exchanger is measured and indicated on the control room console. Low flow is indicated audibly and visually in the control room. Both the inlet and outlet temperature of the primary water to the heat exchanger is measured and indicated in the control room. Audible and visual alarms alert the operator of a failure of the primary coolant flow or system cooling ability. In case of loss of coolant, the reactor is automatically scrammed and the outlet header opens to permit the core to be cooled by natural convection.

The pool water purification system consists of an activated carbon (charcoal) filter and two mixed bed demineralizers. The system is used to maintain the purity of the coolant water in the pool by recirculating 25 gpm continuously through the system and back into the primary coolant. It is also used for purification of makeup water and fill water for the pool. Normal additive water is obtained from the building demineralized water supply. The charcoal filter removes dissolved gases, colloidal material (algae, rust, etc.) and undissolved solids from the pool water. The mixed bed demineralizers remove the ionic material remaining after a passage through the charcoal filter. There are two mixed bed demineralizers in the system. Normally, only one of these will be used, but the other is always available in case of failure or to provide for additional capacity.

The radioactive material in the water will consist largely of A^{41}, N^{16}, and Na^{24}. The Na^{24} will originate from two sources, Na^{23} (n, γ) Na^{24} and Al^{27} (n, α) Na^{24}. Most of the Na^{24}, which decays with a 15-hour half-life, is removed from the water by the primary system demineralizers. Much of the Na^{24} decays on the bed during the collection period. That remaining is stored in the retention basins for further decay following the regeneration of the demineralizers. The N^{16} is produced by the reaction, 0^{16} (n, p) N^{16}. The N^{16}, which has a 7.4-second half-life, is not removed from the pool water. The pool return header is designed to reduce the pool turbulence which, in turn, reduces the concentration of N^{16} near the surface of the pool. The A^{41} is produced by the reaction, A^{40} (n, γ) A^{41}. It is not removed from the water except for the escape from the surface of the water.

The water for the pool and primary system is normally taken from the plant water supply obtained from the James River. Table 4-1 is a chemical analysis of the plant water from the river before and after filtration.

Table 4-2 gives the design and operating information for the primary cooling system.

Table 4-1. Chemical Analysis of James River Water Before and After Filtration

Date sample collected 8-31-55
Date sample analyzed 9-9-55
Sample taken from Nine-Mile Bridge

Substance	Symbol	ppm as Ca CO_1 Before filtration[a]	After filtration[b]
CATIONS			
Calcium	C^{++}	54	54
Magnesium	Mg^{++}	20	20
Sodium	Na^{++}	21	38
Hydrogen-Acidity	H^{+}	0	0
Total		95	112
ANIONS			
Bicarbonate	HCO_3^-	69	66
Carbonate	CO_3^-	0	0
Hydroxide	OH^-	0	0
Phosphate	PO_4^-	0	0
Chloride	CI^-	7	7
Sulfate	SO_4^-	19	39
Nitrate	NO_3^-	0	0
Total		95	112
Total Hardness		74	74
Alkalinity A (Methyl Orange)		69	66
Alkalinity B (Phenolphthalein)		0	0
Non-Carbonate Hardness		5	8
Sodium Alkalinity		0	0
		ppm	ppm
Carbon dioxide	CO_2	11	22
Silica	SIO_2	6.8	6.8
Iron	Fe	0.8	0.8
Turbidity		35	0.2
Color		60	
pH		7.1	6.8

Table 4-1 (Cont'd)

Notes:

(a) Raw James River water

(b) Coagulated precipitator effluent addition of the following chemicals before filtration:

Alum	0.28 lb/1000 gal H_2O
Soda ash	0.14 lb/1000 gal H_2O
Clay	0.28 lb/1000 gal H_2O

Table 4-2. Primary Cooling System Design and Operating Information

Operating conditions	
Reactor power output, MW	6
Cooling water flow rate, gpm	3800
Core inlet pressure, psig	9
Core outlet pressure, psig	2.7
Core inlet temperature, F	100
Coolant (core) exit temperature, F	112.5
Flow rate through fuel, ft/sec	13
Max. fuel plate surface temperature, F	(Design 248) 206
Saturation temperature at hot spot, F	230
Saturation temperature at outlet pressure, F	219.8
Heat exchanger pressure drop, psi	11.8
Inlet temperature to Hx, F	112.5
Outlet temperature from Hx, F	100
Pump power, hp	125
Pool water temperature outside core, F	
Normal	100
Maximum	110
Design conditions	
Pump design flow rate, gpm	3800
Values and flange pressure, psig	180
Heat exchanger pressure, shell, psi	85
Heat exchanger pressure, tubes, psi	85
Length of heat exchanger, ft	18.5
Diameter of heat exchanger, ft	3-5/6
Number of tubes	1744
Size of tubes	
OD, in.	0.75
BWG	16
Length, in.	168
Size of pump motor, hp	125
Material of heat exchanger (shell and tubes)	Aluminum
Pump housing material	Stainless Steel

Table 4-2 (Cont'd)

Piping	Aluminum or stainless steel
Coolant water chemistry	
Total solids (maximum), ppm	2.0
Chlorides (maximum), ppm	1.0
Cu, Hg, Fe (maximum), ppm	0.01
pH range, pH	4.5-6.5

4.2. Secondary Cooling System

The secondary cooling system (Fig. 4.1) removes the heat generated in the primary water by the reactor and experiments in the heat exchanger and dissipates it in the cooling tower. Water from the cooling tower basin is pumped through the heat exchanger and returned to the tower. The heat produced from the reactor and experiments is then rejected to the atmosphere as the water passes down through the cooling tower.

The water is pumped through the tube side of the heat exchanger to the top of the cooling tower. It then flows down the cooling tower into the basin where it is pumped to the intake side of the pump.

The secondary pump has a capacity of 3220 gpm and circulates the secondary water through the system at this rate. It is driven by a 125-hp electric motor which is supplied electrical power from the plant electrical system. In case of a pump or power failure, the reactor would be shut down and operated only on convection cooling.

Flow of secondary coolant through the heat exchanger is measured and indicated in the control room where there is also an indication of heat exchanger inlet and outlet temperature. Gate valves, similar to those of the primary system, are provided in the primary equipment cell to permit isolation of the components of the secondary cooling system for replacement or repair. The cooling tower is of a conventional design. Water is cooled in the tower by forced circulation of the air as the water moves down through the tower.

Initial quantities of water and makeup water are taken directly from the plant water supply. The secondary system water quality is controlled by blowdown and the addition of a mixture of sodium chromate and sodium polyphosphate glass. This material is fed to the system by passing the makeup water over solid balls of the chemicals held in a feeder. Due to the low solubility of the glass, the makeup water dissolves only a few parts per million of the chemical.

The design characteristics and operating information for the secondary cooling system are given in Table 4-3.

Table 4-3. Secondary Cooling System

Heat removal capability, MW	6.7
Cooling water flow, gpm	3200
Pump power, hp	125
Cooling water temperature from tower, F	82
Heat exchanger inlet temperature, F	82
Temperature rise across He, F	14.3
Heat exchanger pressure drop, psi	13.6
Water evaporation, gpm	46.0

4.3. Plant Water

The plant water will be broken down into the following components: (1) potable water, (2) process water, and (3) building demineralized water. Figure 4.2 shows the plant water purification system.

4.3.1. Potable Water

The water required for personal usage will be taken from wells at various locations on B&W property. This water is not processed or treated in any way. It will supply the fountains, sinks, showers, and laboratories throughout the Facility. Figure 2.9 shows the location of the wells.

4.3.2. Process Water

The process water taken from the James River is treated, filtered, and stored in storage tanks. Two 155,000-gallon tanks, one existing and one proposed, provide sufficient storage to assure an adequate supply to the building. The tanks are located south of the NFL.

This water will be used for: (1) makeup to the cooling tower, (2) supplying the building demineralized water equipment, (3) supplying the building toilets, and (4) backup emergency cooling.

4. 3. 3. Demineralized Water

The Facility is served by three demineralizer units or packages. Each package will contain at least one mixed bed demineralizer, one carbon filter, and the necessary equipment to regenerate them. Two of the units are maintained as cold processing water containing very low levels of radioactive material; the third unit is normally used with the primary system. However, if either the storage pool and canal or the assembly canal become contaminated, the primary system demineralizer will be used to clean it up. Figure 4. 2 shows the interconnection between the purification equipment to accomplish the clean up operation.

4. 3. 3. 1. Primary System Purification

The primary system purification is described under the primary system.

4. 3. 3. 2. Building Demineralized Supply

The demineralizers serving the Facility are capable of water purification at the rate of 10 gpm. The system also contains a 100-gallon supply tank for storage. The effluent has a maximum concentration of 2 ppm total dissolved solids. The makeup water for the reactor, test loop, storage pool and canal, assembly canal, and chemistry laboratory is taken from the storage tank. The inlet water to the package comes from the process water system.

When the package is not used to supply building demineralized water, it is used to purify water in the assembly canal. During this type of operation, water is recirculated through the demineralizer and filter at the rate of 10 gpm.

4. 3. 3. 3. Storage Pool and Canal

The water in the storage pool and canal is recirculated through the activated carbon filter and demineralizer at the rate of 10 gpm.

4.4. Waste Disposal

The disposal of radioactive waste involves general operating problems of collection, classification, treatment, storage, and disposition. These materials are described in a form of gaseous, liquid, or solid wastes. The handling methods, retention facilities, and disposal methods are discussed for each type.

4.4.1. Gaseous Wastes

Quantities of gaseous wastes will be produced in the containment vessel, high pressure loops, hot cell laboratories, and oxide areas. It is not possible at present to estimate the amount of gases that will be produced in the Facility. However, the design of the Facility radioactive gas disposal system permits control of the gas released from the Facility. Control is accomplished by filtration, holdup and decay, and monitoring of any gases discharged from the Facility. In no case will be level of activity exceed those defined in 10 CFR, Part 20. A diagram of the radioactive gas disposal system is included in the ventilation system diagram, Figure 4.5.

Radioactive gases present in the containment vessel air will be from the pool water disassociation and entrained air. This waste will consist of A^{41}, N^{16}, H_2, and O_2. The escape of these gases from the pool water is greatly reduced by the pool return header which reduces the turbulence of the water and, therefore, the rate at which the gases are discharged to the air. The activity of the gas in the containment vessel is monitored and then discharged through a prefilter and an absolute filter to the stack at a controlled rate of release. Personnel will not be permitted to enter the containment if the concentration of gas in the building gives a dose rate higher than those of 10 CFR, Part 20.

All types of activated wastes are expected to be generated in the high pressure loops. The volume of gases that may be found in the loops is low due to the low inventory. Normally this gas will be monitored, diluted, filtered, and discharged up the stack at or below permissible concentrations. Facilities are provided to compress and bottle gaseous wastes having such a high activity level that they cannot be discharged through the normal off-gas system. The high level bottled gas will be stored to permit decay to an acceptable level or will be shipped off site for permanent disposal.

The gaseous waste from the hot cell is discharged to the atmosphere through a prefilter, an absolute filter, an activated charcoal filter, and a monitor. The high level gaseous waste collected inside the hot cell dry boxes is monitored and either bottled and handled as described or discharged at a controlled rate through the hot cell ventilation system to the effluent stack.

The gaseous wastes from the feed materials building, radiochemistry laboratory, and oxide area are from the dry boxes and hoods. These gases are passed through a prefilter, an absolute filter, and monitored prior to discharge up the stack. Gas bottling facilities can also be used to control the hazards from high activity of these gases.

As described previously, all radioactive gases not bottled for off-site shipment will be discharged to the atmosphere through the stack at such a rate that no hazard will result to the populated areas. The stack effluent gases will be monitored by the following systems:

1. A gross beta-gamma radiation detector (G-M Tube).
2. A particulate activities monitoring system.
3. A radioactive monitoring system.
4. A gaseous activity monitoring system.

The G-M tube, positioned in the stack so that the effluent gas flows past it, is used to detect gross beta-gamma activity. The particulate monitoring system consists of a monitored filter; the radioiodine monitoring system consists of a monitored charcoal absorption bed; the gaseous activity monitoring system consists of a monitored spherical volume.

A side stream from the stack is first passed through the particulate monitoring system, then through the radioiodine monitoring system, and finally through the gaseous activity monitoring system. The activity levels detected in each of these systems will be 10^{-7} $\mu c/cm^3$ — the lowest limit obtainable in the gaseous activity monitoring system. However, in the particulate and radioiodine systems, it will be possible to lower this limit if desired since an integrated sample is being monitored in each of these systems.

4.4.2. Liquid Wastes

Activated liquid wastes will originate from the primary purification system, pool or canal, high pressure loops, the hot cell, and the laboratories. The arrangement of the liquid waste disposal system for the NDC is shown in Figure 4.3, and the fluid flow diagram is given in Figure 4.4. Large quantities of liquid wastes are not expected from the operation of the Facility, but provisions are made to permit handling of any amount that could possibly be expected to occur from any source.

The liquid waste system is located underground at the west end of the NFL. It consists of two 3600-gallon regenerant storage basins, two collection tanks of 300 gallons capacity each, and three storage tanks with a capacity of 2000 gallons each. It also provides sufficient pumps, pipe, and sampling arrangements to transfer and evaluate the wastes.

The effluent from the regeneration of the primary purification equipment is routed to storage basins. There are two basins, each capable of handling the fluid resulting from the regeneration of one mixed bed ion exchange unit. The regeneration effluent is stored in the basins for the decay of the Na^{24} removed from the primary water. Following the decay period allotted for the decay of Na^{24}, the basin fluid is monitored to determine the disposal method. The waste may be pumped into the sanitary sewerage only after the acitivity level is below the concentrations defined in 10 CFR, Part 20.

Before the pool or canal are drained, the water activity level is reduced to a value less than maximum permissible concentrating as defined in 10 CFR, Part 20 for non-occupational exposure. The water is drained into the sanitary sewerage system for disposal.

The liquid waste from the high pressure loops is monitored and transferred to the proper storage tank in the liquid waste system where it is either stored for decay until it may be pumped into the plant sewer or is collected for shipment in shielded tank trucks for processing and disposal at an off-site facility.

Many levels of waste are expected in the hot cell. The extremely high level liquid waste is collected in glass cartons protected by metal containers. This waste is accumulated and shipped off site for permanent disposal. Other liquid wastes are collected along with the lab waste in one of two 100-gallon collection tanks where it is monitored and transferred to the proper storage tank at the pits.

The laboratory drainage is collected in the 300-gallon collection tanks where it is monitored and neutralized. The activity level determines the storage tank into which it is transferred. One of the storage tanks receives low level waste between 10 and 1000 times MPC as defined in the 10 CFR, Part 20 for non-occupational exposure. The second tank is used to store wastes which exceed 1000 times MPC as defined in 10 CFR, Part 20. The third tank, for emergency storage, is used in those instances following an accident or spill and will not be used during normal operation.

4.4.3. Solid Wastes

All solid wastes are collected in shielded containers and shipped to off-site burial grounds for permanent disposal. Rags, wipers, and miscellaneous laboratory equipment will be packed in flexible bags. Contaminated filters, glassware, scrap metal, tools, etc., are collected in metal containers. Spent ion exchange resins are sluiced into shielded drums. Solidified wastes are collected and prepared for off-site shipment in shielded containers. Fuel element end pieces and experimental equipment are retained in shielded storage until the accumulation is sufficient for economical off-site shipment.

4.5. Sewerage

The Facility sanitary sewer will be tied into the existing plant sewerage system servicing the CEL and NFP. (See Fig. 2.) The system is large enough to handle all wastes from the site. The sewerage from the plants is piped to the sewerage treatment plant north of the NFP. At this point, sewerage is treated and charged into the James River just above Nine-Mile Bridge.

There are no potable water intakes for a distance of 60 miles downstream of the sewer headwall. Sewerage is carefully monitored before being discharged into the plant sewerage system, but is not monitored again prior to discharge to the river. Storm drainage and blowdown from the cooling tower are piped to the adjoining flats and disposed of by evaporation or soaking into the ground. Again, this source of water is not contaminated by radioactivity.

4.6. Ventilatng System

The office portion of the building, including the control room, will be air conditioned.

The potentially contaminated gases are routed to the stack, where they will be monitored before being exhausted. A blower near the base of the stack will be used to maintain a slight negative pressure throughout the collection system. This fan also maintains a negative pressure to all potentially hot laboratories and hoods during those periods of time following power failure or source blower failure. The line diagram, Figure 4.5, shows the exhaust and intakes for the building other than for the air conditioner.

4.6.1. Containment Vessel

The containment temperature and humidity is controlled by an internal recirculation system. A blower recirculates the containment air through temperature and humidity control equipment. This air is picked up near the pool and is returned throughout the containment vessel.

An air purge line is used to ventilate the containment vessel prior to entering. The flow may be as high as 2400 cfm. This purge line is interlocked with a monitor which will not permit purge when the activity exceeds a preset limit. For those periods of time when purge is not permissible, the air clean-up equipment installed inside the vessel will be used to remove the particulate material and the radioactive iodine. This equipment consists of a pre-filter, an absolute filter, and a charcoal filter. These filters will be used in conjunction with the containment temperature and humidity control equipment.

Connections are provided in the external purge system for the containment so that clean-up action can be taken if the internal clean-up system fails to operate.

4.6.2. Radiochemistry Laboratory

Air conditioned air enters the rooms, but there will be no return. The hood blowers will exhaust the air from the room. When the hoods are not in use, the blowers will continue to exhaust air, but the door should be down or closed. The hoods are exhausted through a pre-filter, an absolute filter, a blower, a one-way damper into the common duct going to the stack.

The perchloric acid hood is an exception since it will discharge directly to the roof through the shortest run possible. At no point will the duct be run horizontally or such a way that pockets may exist. The exhauster will be the last equipment the fumes encounter before leaving the building. A wash-down connection will be included near the fan suction. There will be no filter in the perchloric acid hood exhaust.

4.6.3. Cell Operations Area

The cell operation area will be air conditioned. The duct system is designed to maintain the area at a pressure that is positive with respect to the hot cell and adjoining areas. However, it must be at a lower pressure than in hall way and counting room.

There is a manometer indication and alarm on the hot cell to measure loss of hot cell negative pressure. (See Section 4.4.6.)

4.6.4. Assembly and Crane Bay Area

The crane bay and assembly area will have ventilation air forced into them.

4.6.5. Cell Isolation Area and Repair Room

Air will be removed from these areas at a rate which will maintain a 0.1 to 0.2 inch negative pressure between them and the adjoining areas. The exhauster (blower) will discharge into the stack duct. Manometers will be used to measure the negative pressure and an indicator light in the health physic office will indicate the loss of the negative pressures. A prefilter and an absolute filter will be installed between the intake and the blower.

4.6.6. Hot Cell

The hot cell will be maintained at a negative pressure of 0.6 to 1 inch of water when the cell door is closed. A manometer with an indicator light in the health physics office will alert the health physics and personnel in the cell operations area. When the door is open, the air flow through the door will be held to a minimum velocity of 100 linear feet per minute. The air leaving the cell will pass through a prefilter, an absolute filter, a blower and a one-way damper before entering the stack discharge duct. This blower will be placed on the emergency generator.

Extremely high-level gases originating in the hot cell will be compressed and stored in gas bottles.

4.6.7. Oxide Area

The oxide area is divided into three rooms. Ventilation ducts from each room contain an air flow damper before entering a common duct. This type of control permits an air flow regulation from lower contaminated to higher contaminated areas. The dampers will be located on the storage balcony. A common prefilter, absolute filter, blower, and a one-way damper serve the area. The air leaving the damper enters the common duct going to the stack. An indicator on the filter unit will indicate the pressure loss across the filter and will indicate when the pressure drop decreases to a set minimum by a light in the health physic office (this light indicates the loss of the blower). The air movement throughout the area will be held low.

The two hoods are provided with a common discharge duct with a damper for each hood. They use a common prefilter, an absolute filter, a blower, and a one-way damper between the hoods and a common duct going to the stack. The filters and blower will be installed on the wall above the hoods. Connectors for the portable glove box are provided ahead of the filters.

A tee connection to the hood discharge down stream of the one-way damper provides the necessary connector for the future glove box line to be installed along the back wall. The blowers, filters, etc., will be installed with the glove box line.

4.6.8. Machine Shop

Air will be blown into the machine shop; therefore, this clean area will be at a pressure exceeding that of the adjoining areas. The pickling area and welding hood will contain an outside exhaust.

4.6.9. Non-Destructive Test Laboratory

This laboratory is air conditioned. The return duct should be designed to return less air than is received to establish a positive pressure with respect to the oxide area. This positive pressure will be maintained when the oxide area is at atmospheric pressure. A manometer with indicator lights in the health physics office will indicate loss of the positive pressure.

4.6.10. Change Rooms

The change rooms are air conditioned with the inlet into the cold change room. The discharge is taken from the hot change room. A manometer indicator light in the health physics offices will indicate the loss of a negative pressure between the adjoining hot areas and the change rooms.

4.6.11. Metallurgy Laboratories

These laboratories are air conditioned. The return duct should be sized to return less air than the room receives. The dark room contains a kitchen-type exhaust with a duct to rear wall. Excess air delivered to the room will leave through this duct even though the exhauster is off.

The hoods are exhausted through empty filter boxes and a one-way damper into a common duct going to the stack.

4.6.12. Hall

The hall receives air from the air conditioning unit, but there is no return. This air will leave through doors opening into the hall.

4.6.13. Analytical Chemistry Laboratory

Air conditioned air enters the rooms, but there will be no return. The hood blowers will exhaust the air from the room. When the hoods are not in use, the blowers will continue to exhaust air. The hoods are exhausted through an empty filter box, a blower, and a one-way damper into the common duct going to the stack.

The perchloric acid hood is an exception since it will discharge directly to the roof through the shortest run possible. At no point will the duct be run horizontally or such a way that pockets may exist. The exhauster will be the last equipment the fumes encounter before leaving the building. A wash-down connection will be included near the fan suction. There will be no filter in the perchloric acid hood exhaust.

The glove box discharges into the common stack duct through a one-way damper. These boxes contain a blower and filter.

4.6.14. Tracer Laboratory

This laboratory is treated the same as the analytical chemistry laboratory except it contains no perchloric acid hood.

4.6.15. Spectrographic Laboratory

This laboratory is air conditioned with constant temperature and humidity control. The hood is exhausted to the common stack duct through an empty filter box, a blower, and a one-way damper. A second hood connection will be installed to the lab next to the tracer laboratory wall.

4.6.16. Fan Room and Stack

A large blower will be installed near the feed materials building. This fan will take suction on the common stack duct leaving the main building causing the duct to be held under a slight negative pressure. This blower would also provide any dilution air deemed necessary. Also, on a power failure to the main building, this blower would maintain a negative pressure to all potentially hot labs and hoods (it may replace the requirement that individual hood fans be connected to the emergency power supply).

4.7. Electrical System

4.7.1. Function

This system supplies and controls electrical power to the reactor during emergency as well as normal operation. The system consists of these power supplies:

1. 440 v, three-phase normal supply for motors rated from 0.5 to 150 hp.
2. 115/230 v, single-phase normal supply for instrumentation, control, lights, and motors rated below 0.5 hp.
3. Uninterruptable supply to maintain power to critical equipment.

4.7.2. Electrical Distribution System

Figure 4.6 is a diagram of the electrical power supply system.

The present facility is served by 12 kv from an Appalachian Power Company 33/12-kv substation comprised of one 3750-kva, three-phase, 33/12-kv transformer and a related switching structure.

4.7.3. Uninterruptable Power Supply

A 100-kw uninterruptable power supply is provided by a synchronous alternator which floats on the line as a motor until line voltage drops, then transfers to a diesel power and picks up its assigned load without interruption. A large flywheel permits the uninterrupted change over to the diesel. This unit, operating in conjunction with the uninterruptable power supply control center, provides uninterrupted power to the vital plant equipment and instrumentation. This power supply will be located indoors to insure reliability.

4.8. Effluent Control

The method of handling and disposing each type of waste or effluent from the Facility is described in Section 4.4. Since there will be some quantities of materials with a high level of activity from the Facility, the necessity for careful control of all radioactive material is given great consideration. There will be no effluent discharged from the plant that will expose the off-site population to concentrations of radioactive isotopes greater than those specified in 10 CFR, Part 20 from routine plant operation; in no case will levels exceed those of 10 CFR, Part 100 due to an accident. This is accomplished by proper monitoring, filtering, holdup, storage, discharge, or other methods of handling the radioactive materials.

4.9. Process Instrumentation

Besides the nuclear instrumentation and other monitoring systems for the Facility, process monitoring instrumentation is provided to supply operational information for the various systems (the coolant systems in particular). These monitoring systems are designed to activate certain alarms and indicators that give an indication of systems not operating within their design limitations. Table 4.4 lists the important monitors for the process systems.

Instrumentation associated with the reactor operation is located in the primary and secondary coolant and the primary purification systems. The primary and secondary flow rates are measured with differential pressure cells across the heat exchanger. The differential pressure measurements are converted to pneumatic signals and transmitted to the non-nuclear instrumentation console. At the console, the pneumatic signals are converted to a flow indication displayed on two 4.5-inch round face indicators.

The inlet and outlet temperatures are measured on the primary and secondary side of the heat exchanger. Gas-filled bulbs are used to detect the temperature. The measurements are converted to a pneumatic signal in the transmitters which forward the temperature signals to the non-nuclear instrumentation console. The heat exchanger inlet and outlet temperatures for the primary side are displayed on a single 4.5-inch round face duplex indicator.

Conductivity measurements are made in line between the activated carbon filter and mixed bed demineralizers. These measurements are used to determine the necessity for regenerating the mixed bed demineralizers.

The motor start-stop buttons and indicating lights are also located on the non-nuclear instrumentation console. These include the primary pump motor, the purification pump motor, the secondary pump motor, and cooling tower fan motors.

A three-point annunciator is provided with the non-nuclear instrumentation console. One point alarms low primary flow, a second point alarms high heat exchanger outlet temperature, and the third point is a spare. These audible and visual alarms alert the operator to the conditions affecting the reactor cooling.

Table 4-4. Monitors for Process Systems

System	Monitor	Location	Indication	Signal
Primary Water	Flow, Temperature	Primary Equipment Cell	Yes	Yes
Secondary Water	Flow, Temperature	Primary Equipment Cell	Yes	No
Pool Water	Temperature	Containment	Yes	No
Purification	Conductivity	Primary Equipment Cell	Yes	No
Hot Drainage	Level	Waste Pit	No	Yes
Experimental Cooling	Flow, Temperature, Pressure	Containment	Yes	Yes
Ventilation Air	Flow, Pressure	Various	Yes	Yes
Gas Cleanup	Flow	Containment	Yes	No

4.10. Area Monitoring

A fixed radiation monitoring system, similar to the tracer laboratory model RMI-103, will be installed within the reactor operations area. The system will consist of a minimum of ten points monitoring stations with normal operating ranges of 0.05 mr/hr to 1 hr gamma. All stations will alarm and read out in the reactor control room.

Detector stations outside the containment will be equipped with local alarms to warn personnel in the area of unexpected radiation fields. In addition to the control room readout, a repeated level reading will be displayed in the Reactor Operations Health Physics office.

The areas covered by the fixed area monitoring system are:

1. Reactor Containment
2. Storage Pool Area
3. Crane Bay and Adjoining Facilities
4. Primary Machinery Room
5. Hot Cell Operations Area

Besides the fixed system there will be three mobile air monitor carts similar to Nuclear Measurements Corporation's (NMC) model AM-2A and three semi-portable gamma area monitors similar to NMC gamma alarm system model GA-2A. These units will not be assigned to fixed locations, but will be placed, under the direction of the Health Physics Department, in those areas where such monitoring facilities are required or desirable.

The air monitor carts and semi-portable gamma area monitors are designed for continuous duty operating from the 115-volt building power. Each unit is complete with a level readout instrument and an audible-visual alarm system.

Outside the reactor operations area, but included within the Facility, will be the independent operation of the Oxide area disassociated from the reactor. A semi-fixed radiation monitoring system will be installed in the Oxide area employing stations similar to NMC gamma alarm system model GA-2A.

The units will be wall mounted, but may be moved as changing operating conditions warrant. Each unit is self-contained, and operates from the 115-volt building power. The operating range will be from 0.05 mr/hr to 1 r/hr. A local level readout and a visual and audible alarm are provided.

Besides these active systems, a passive system of radiation monitoring will be employed throughout the building. The passive system employes film packets of the same type and nature as used for personnel film badges. These film packets will be attached to the building in fixed locations and will be developed and read by an outside laboratory along with the personnel film badges.

The Facility will be equipped with a stack gas monitoring system. (See Section 4.4.).

4.11. Communications

The building is serviced with a telephone system. In addition, a two-way intercom system is used between the reactor control room and associated areas (containment, loop control room, storage canal, health physic office, etc.).

Other facilities in this area are connected to the NDC by the telephone service. Information concerning accidents or hazardous conditions will be passed to the other areas by telephone.

The building will be equipped with two alarm systems, a fire alarm, and a building evacuation alarm. Pull boxes will be located at vital points throughout the plant. Each alarm system will be unique to avoid any possible confusion.

4.12. Utilities for Experiments

The reactor core is designed around a test facility, approximately 6 inches square, and will accommodate an experiment somewhat smaller than the 6 inches. As shown in Figure 3.5, the typical experiment will occupy about 3.5 inches of this space. The remainder will be filled with the steel tubes and other parts of the experiment. The corners will be filled with aluminum filler blocks. The experiment will be designed for a maximum power of 25 kw per foot and will be capable of handling water pressure up to 2500 psig and temperatures to 670°F.

Cooling for the central experiment is provided by circulating water in the closed loop of the experiment. A heat exchanger will be installed in the loop circulating system and may utilize either the primary or secondary cooling water to remove the heat from the loop coolant. Figure 4.7 is a flow diagram of a typical loop using the primary coolant water to carry the heat away from the loop coolant. In either case, the heat generated by an experiment is eventually released to the atmosphere through the cooling tower.

The test loop is designed so that emergency cooling is provided by natural circulating of the loop water. In case of a loop cooling system failure, the reactor is automatically scrammed and the loop may be cooled by natural circulation. The present layout provides a differential height of 10 feet or greater for the heat sink above the heat source. Secondly, a steam generator is used to remove heat from the loop. This steam generator contains sufficient secondary water to remove decay heat for approximately one hour with no feedwater. The steam removed from the steam generator is condensed in the pool water. The building demineralized water supplies makeup water for the loop. Figure 4.8 shows the kw heat produced as a function of δT across the core. It indicates that the experiment is sufficiently cooled by convection cooling following reactor shutdown.

Electrical power to operate circulating pumps, instrumentation, and other loop equipment is supplied by the normal plant electric supply. Emergency electrical power is furnished to the essential instrumentation, pumps and other vital mechanisms by the plant emergency power system to assure that there is always an electrical power supply to these vital pieces of equipment. Other facilities, such as pressure regulation, temperature control, flow variation, water purification, and cover gas, are supplied to the experimental setup as needed for operation. Instrumentation for the experiment is also connected into the reactor in such a manner that failure or malfunctioning of the experiment will automatically reduce the power of the reactor and will cause a scram if necessary.

Special cooling systems are not provided for experiments in the capsule holes; they are cooled by circulation of the pool water through the facilities. Instrumentation to the samples will be provided as desired or necessary to permit monitoring and data taking for samples.

4.13. Environmental Monitoring

Plans are not yet complete for the environmental monitoring program to be carried out at the site. However, a program will be used to monitor the area to determine the buildup of activation. Pre-operational surveys will be conducted to determine the background level of radiation and contamination. These surveys will be performed on and off the site particularly in the direction of the prevailing wind (eastern quadrant). This survey will be made to determine the radioactive background. However, it is expected that it will show only the variation of radiation produced by weapons tests. Gross beta activity concentrations and an analysis for at least one long-lived radioactive isotope (probably Sr^{90}) will be performed. No iodine determinations will be performed in the pre-operational program. These measurements will give a normal for the start of the operational surveys. Iodine monitoring will be included in the operational phase.

Figure 4.1. Coolant and Purification Systems

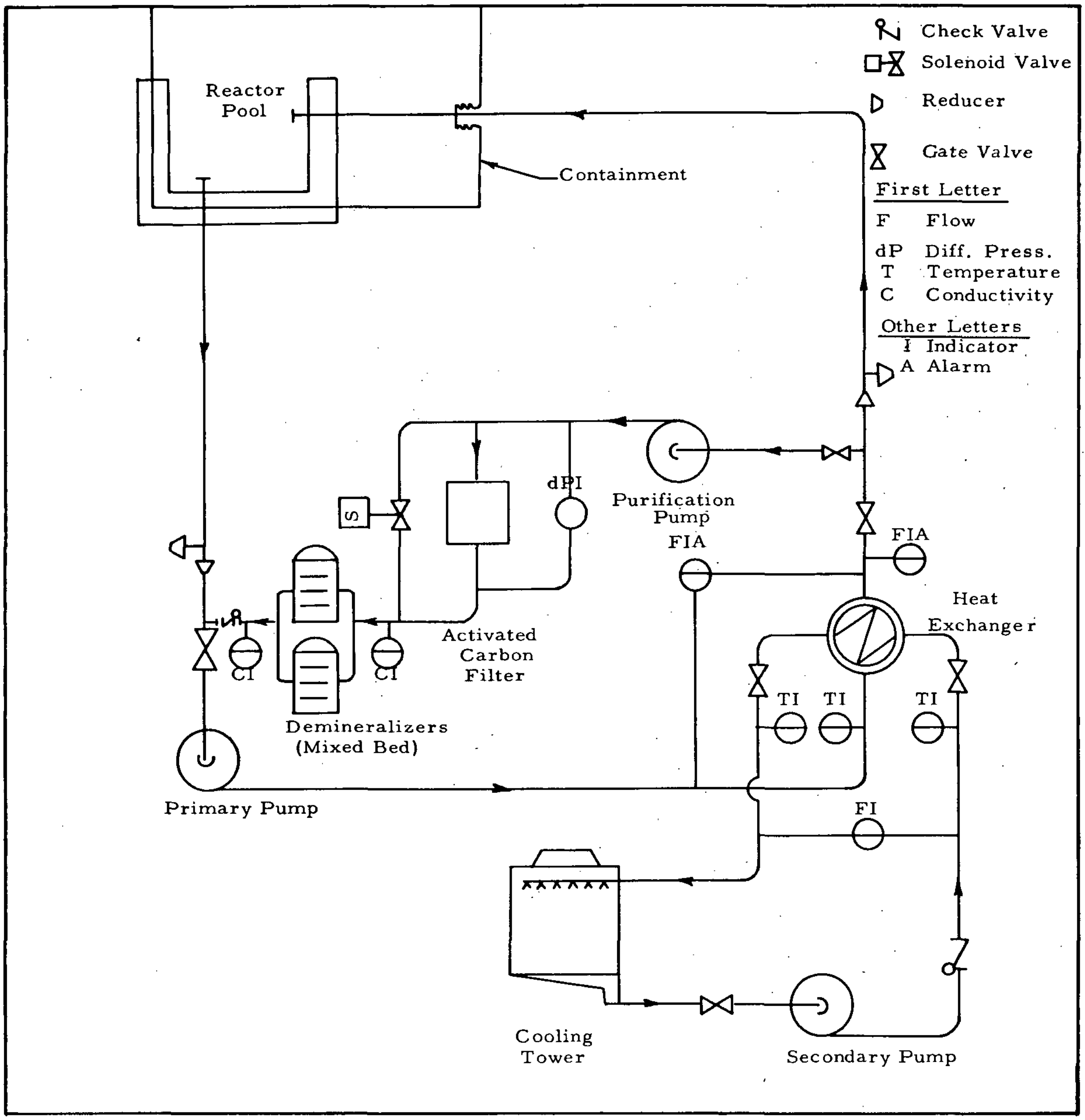


Figure 4. 2. Demineralizers Systems

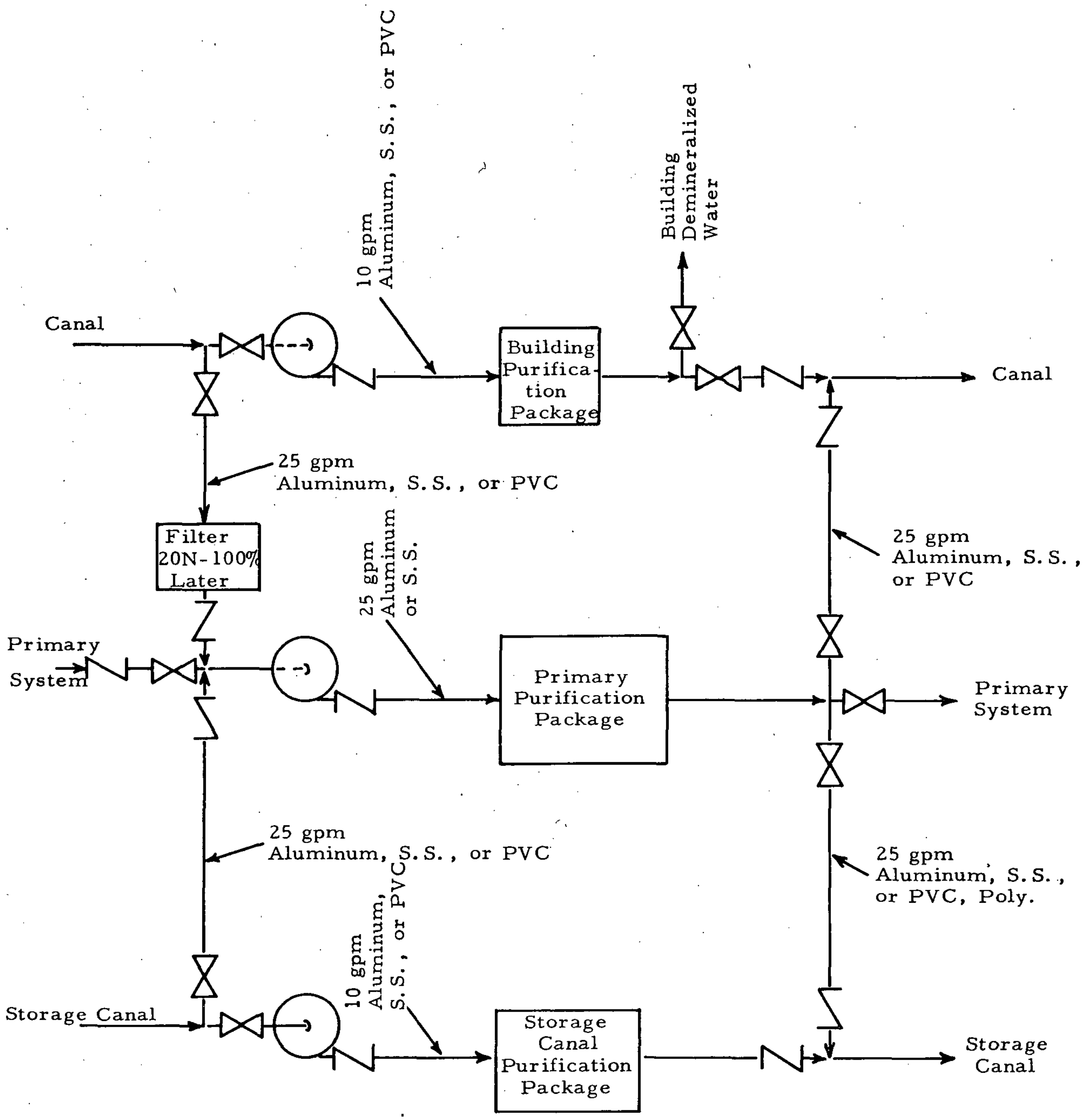


Figure 4. 3. Liquid Waste Disposal System

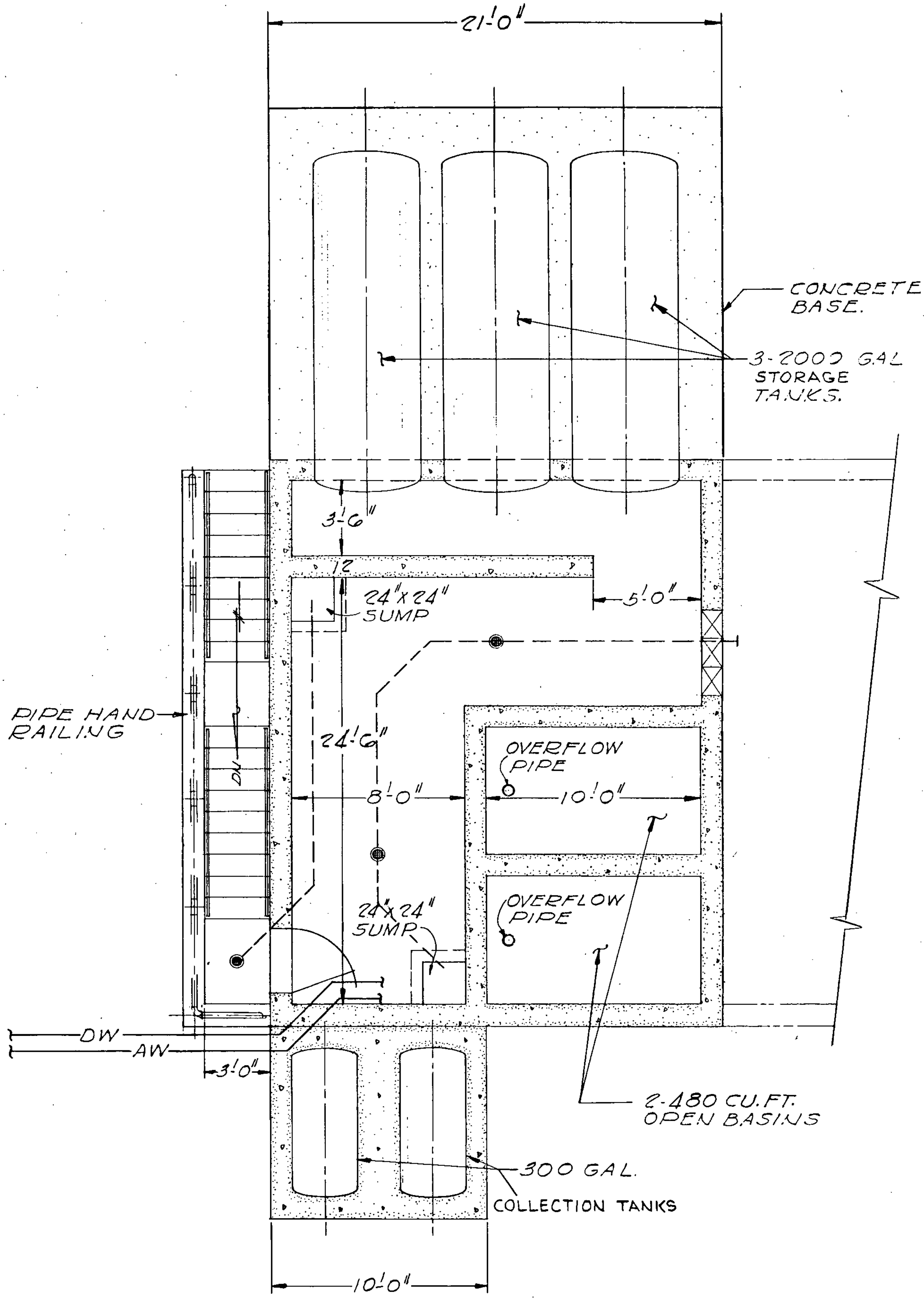


Figure 4. 4. Liquid Waste Disposal System — Flow Diagram

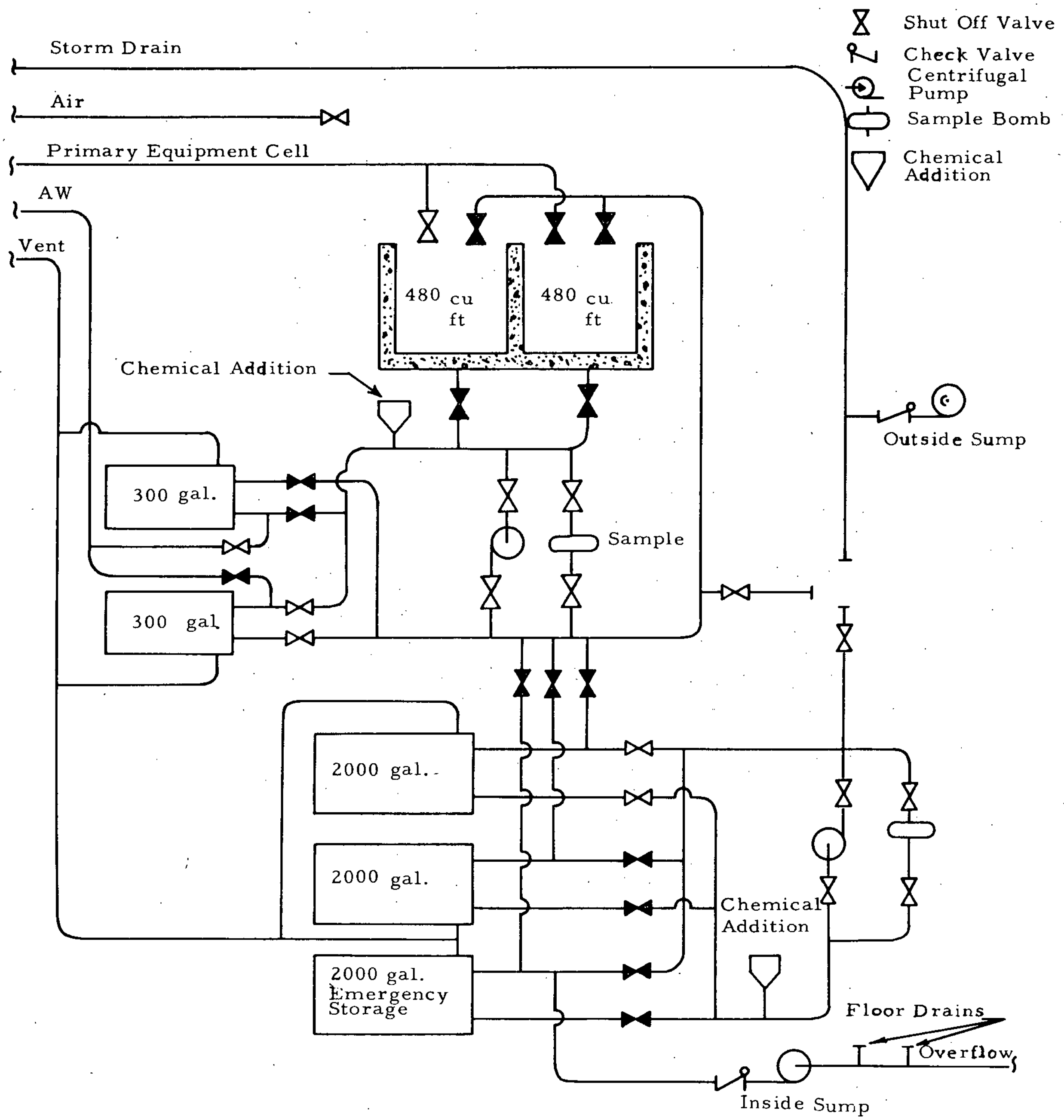


Figure 4. 5. Building Ventilation and Gaseous Waste Disposal System

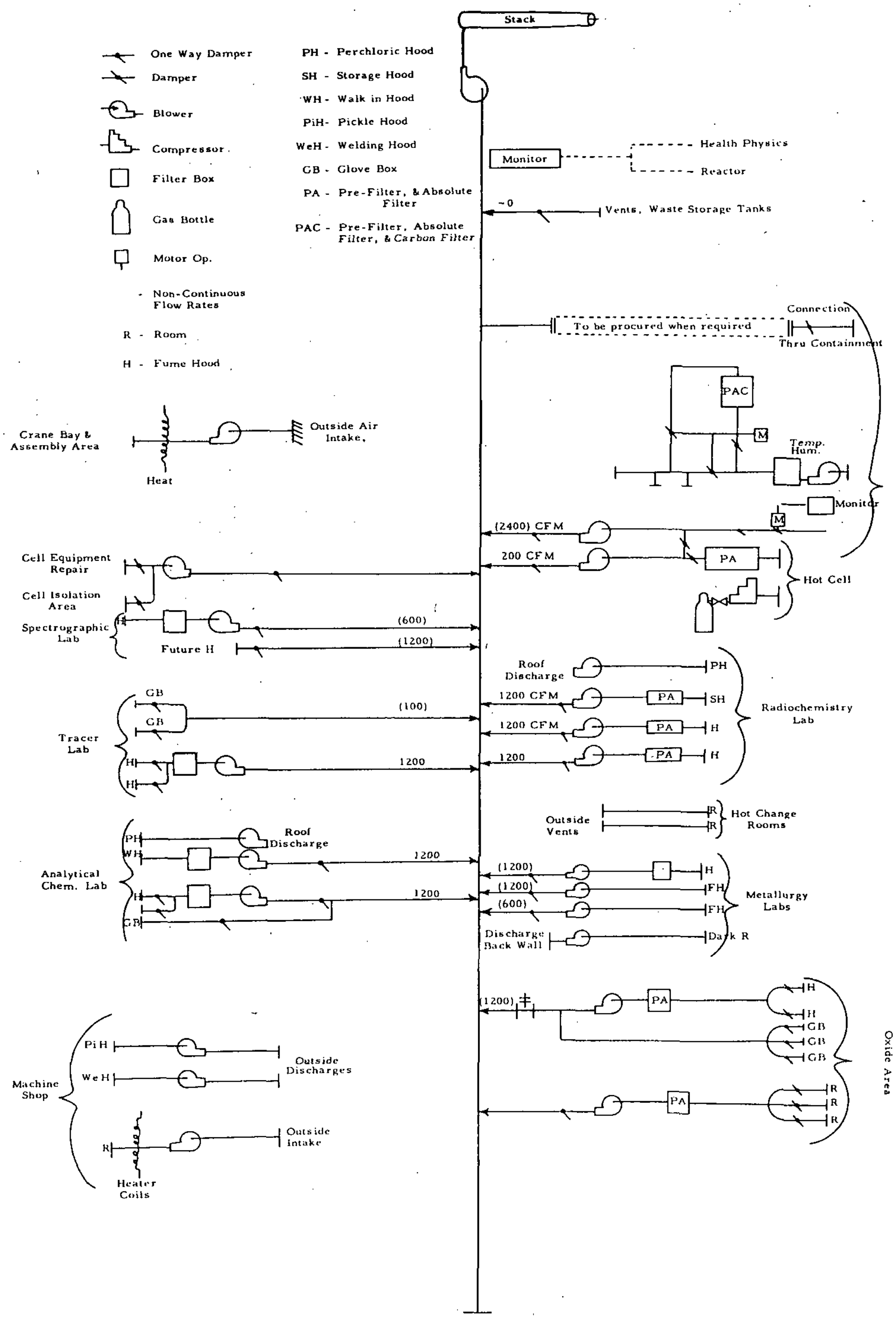


Figure 4.6. Electrical Riser Diagram

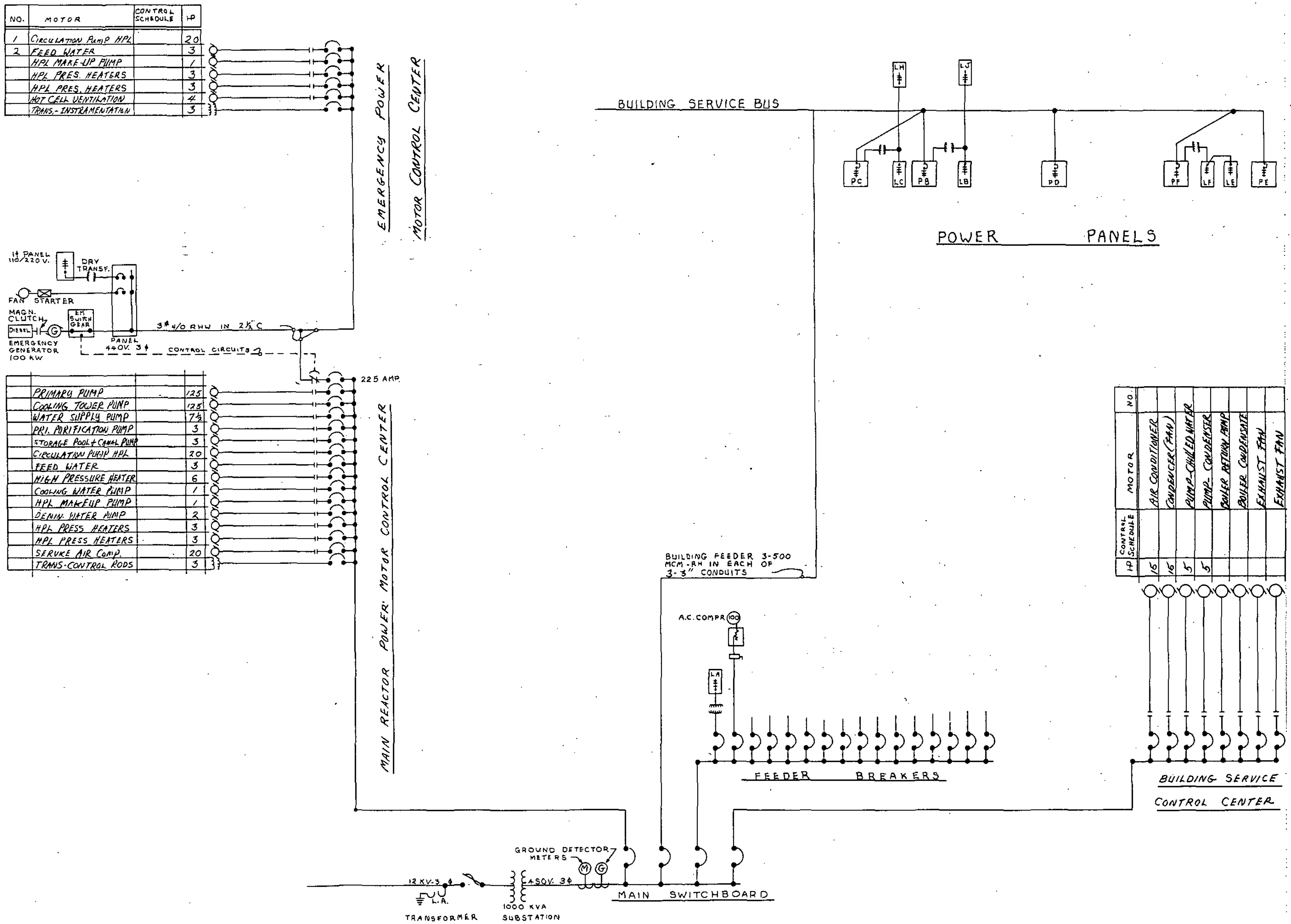
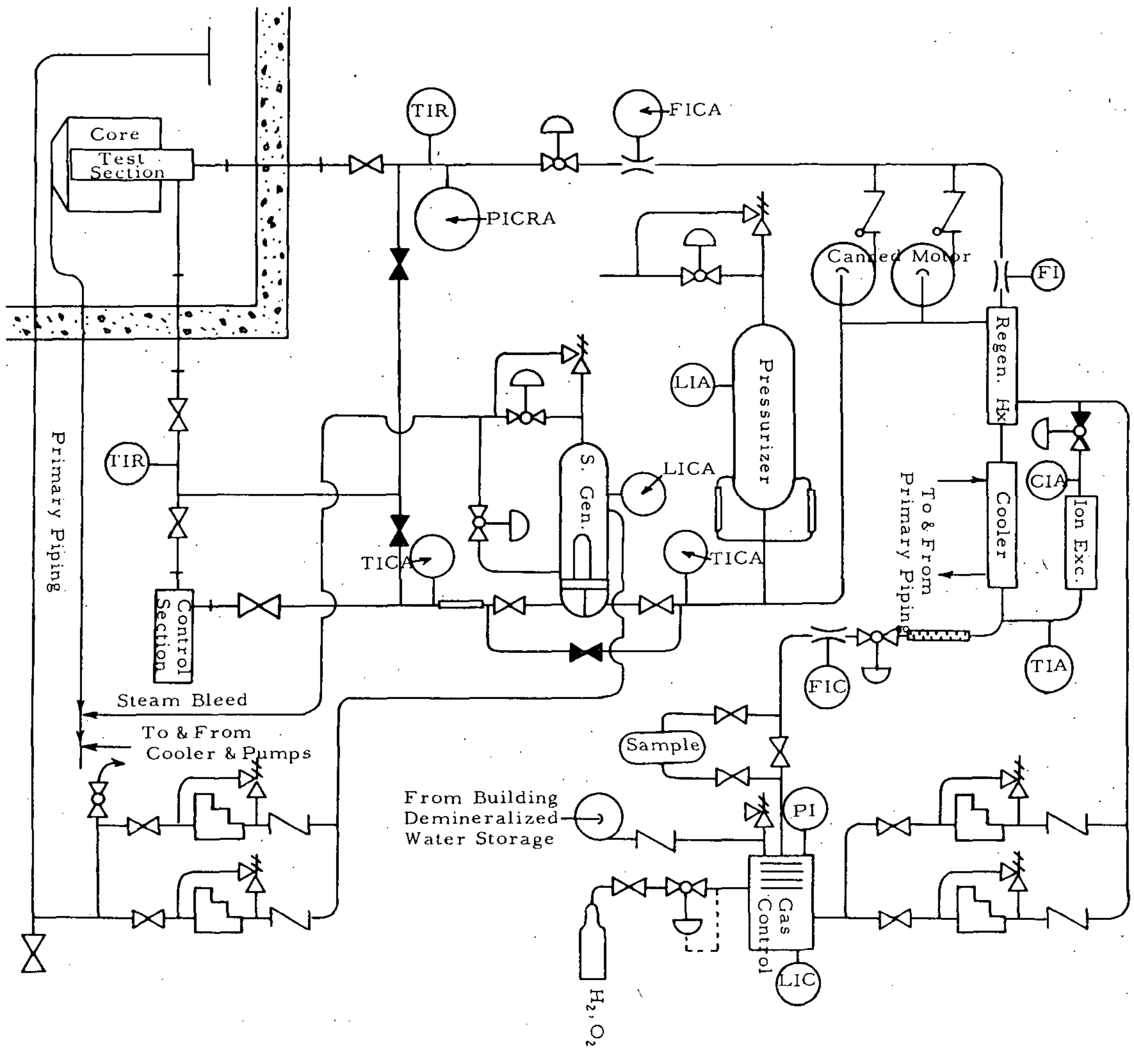


Figure 4.7. Test Loop Flow Diagram



First Letters

F - Flow
T - Temp
L - Level
P - Pressure
C - Conductivity

Other Letters

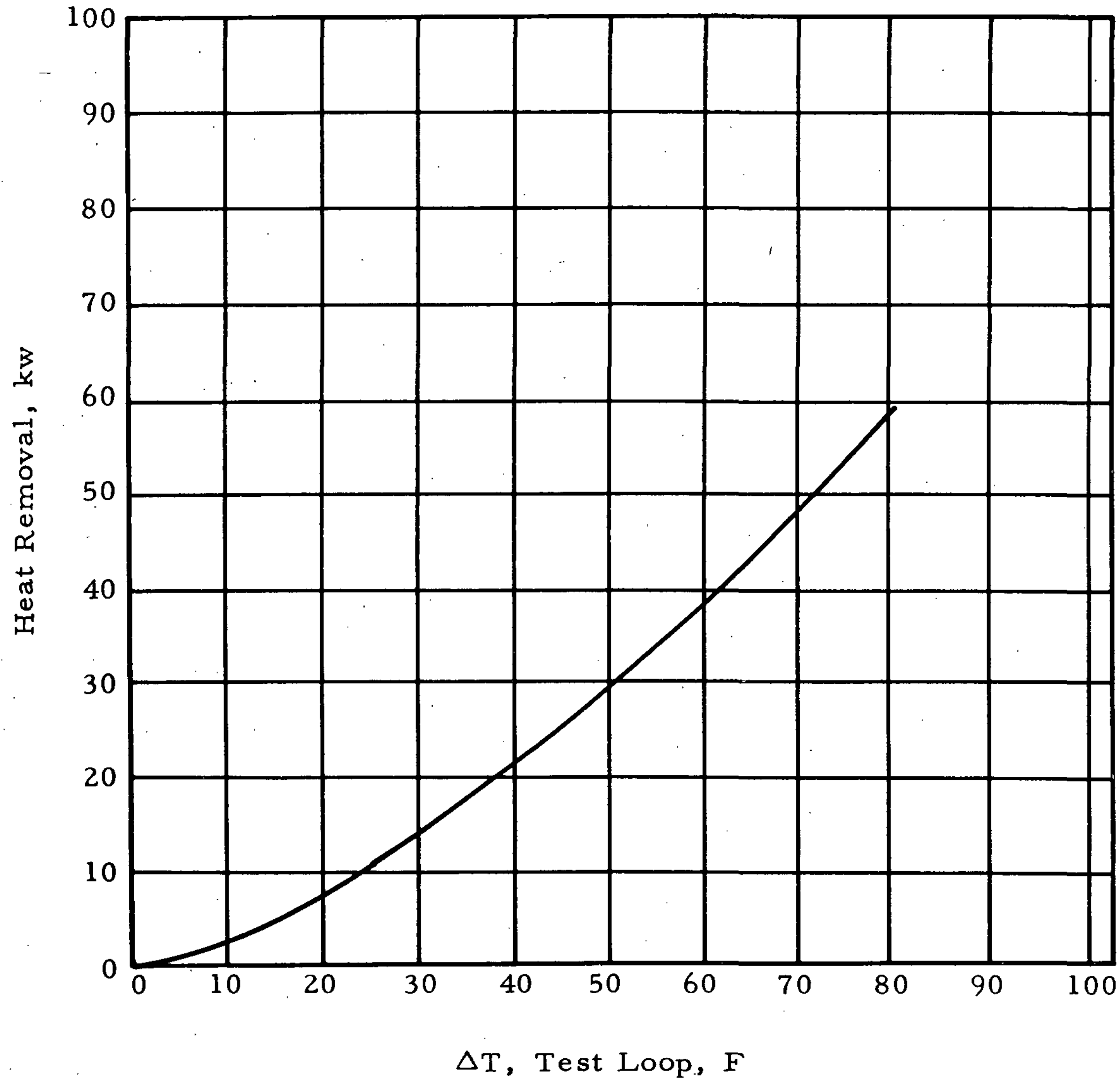
A - Alarm
C - Controller
I - Indicator
R - Recorder

Relief Valve
Centrifugal Pump
Check Valve
Gate Valve
Globe Valve

Pneumatic Operator
Pressure Drop Orifices
Positive Displacement Pump

Figure 4.8. Heat Removal Vs ΔT for Test Loop

Natural convection cooling with a 10-foot elevation differential between heat source and heat sink.



5. EXPERIMENTAL CAPABILITY

5.1. Types of Experiments

Irradiation experiments to be conducted in the BAWTR range from simple capsules to complicated environmental capsules and test loops. The BAWTR design provides for an in-pile test loop tube to be centrally located within the core. Experimental holes are provided at the corners of the core in the reflector pieces for capsule experiments. Also, experiments may be irradiated in the pool outside of and against the reflector pieces.

5.1.1. Capsules

Irradiation capsules vary from simple static environmental types to complicated lead types. Within a single static capsule, if the material to be irradiated is non-corrosive, the capsule may be of an open design where the specimen is directly cooled by the primary coolant water. Most capsules will consist of a cannister containing the specimen. Liquid metal may be used to provide better heat transfer from specimen to cannister wall.

For lead capsule types, externally supplied heating or cooling along with sensing and control instrumentation may be added to perform experiments at elevated temperatures. Other environmental conditions will be monitored and controlled so that the specimen is exposed to various gaseous atmospheres at elevated pressures, etc., and to vary thermal barrier conditions, or measure or modify the magnitude and spectrum of the neutron flux.

In a representative capsule experiment, the samples would be supported in a sealed steel cannister filled with NaK and instrumented with thermocouples. A small heating element may be wrapped around the cannister and the assembly will be inserted into a large cannister. A gas-filled annulus may be retained between the two cannister walls for thermal insulation. The leads extending from the outer cannister are sheathed by a flexible tube until they reach the top of the pool. The sheath tube may be wet or dry and slightly pressurized. The leads are then taken to the control and monitoring instruments.

5.1.2. Loop Experiments

A loop consists of the in-pile tube and the out-pile central control equipment. The latter is located in the shielded loop equipment pit with controls and instrumentation located outside the containment vessel. The loop system is designed so that a specimen can be subjected to dynamic environmental conditions differing greatly from those of the BAWTR core. The in-pile tube allows not only different conditions, but also the flexibility of varying conditions during the course of the irradiation. The experiment may consist of single specimens or simulated fuel element assemblies. These experiments may be instrumented with thermocouples, flow orifices, sampling lines, and wires for determining the flux profile to provide more exact data than is provided by loop instrumentation.

The in-pile portion of the loop will be a top-entering, re-entrant tube (Fig. 5.1). The inlet and outlet coolant lines will leave the pool through pool wall penetrations into the loop equipment pit below the pool water surface. The in-pile tube will normally remain in place while the irradiated samples are removed through a closure located below the pool water level. The water height above the closure is to provide adequate shielding during irradiated sample removal. The irradiated samples may be transferred to the canal by the underwater pool gate for subsequent insertion into the hot cell complex for examination. Figure 4.7 is a flow diagram for a typical loop experiment to be irradiated in the BAWTR.

5.1.3. Typical Experiment

In a typical experiment, the sample (Fig. 5.2) would consist of a bundle of fuel pins in a square or triangular lattice. The fuel pins would be affixed together by grid assemblies at four equidistance positions over the 2.5-foot length. Thermocouple will be attached to provide temperatures at critical points of the bundle. Tubes may be provided upstream and downstream of critical flow baffles for differential pressure measurements. The complete bundle assembly including instrumentation would be contained within flow director to restrict the water flow as required by thermal and hydraulic considerations.

All instrumentation leads would penetrate the loop pressure vessel at the closure port by high temperature-pressure fittings as required. The leads are then taken to the monitoring and control instruments as required.

5.2. Reactivity Effects of Experiments

One of the primary safety features of the BAWTR is the limitation on the amount of available excess reactivity that can be inserted into the reactor by any operational error, equipment failure, or experiment failure. This feature is accomplished by limiting the available excess in the control rods and the total reactivity designed in the experiments. The control limitation is described in Section 3.

The experiments for the reactor will be designed so that reactivity greater than 1.5% $\delta k/k$ cannot be added to the reactor as a result of failure or malfunction of the experiment. Prior to operating an experiment, the reactivity effect of the experiment on the reactor will be calculated by conventional computational methods. Prior to operation, this value will be checked by an actual measurement of the amount of reactivity in the experiment.

The experiment will not be operated if the available excess reactivity from experimental failure exceeds 1.5% $\delta k/k$.

Calculations made for some types of experiments that could be operated in the central facility of BAWTR indicate that there are areas over which a positive temperature or void coefficient may occur in the experiment. Over the range of temperatures and voids possible in the experiment, the contribution of positive reactivity is less than 1.5% $\delta k/k$. For all experiments considered for irradiation in BAWTR, core temperature coefficient adds at least a factor of 5 greater in negative reactivity per degree temperature than the positive effects of an experiment. Therefore, any rise in temperature of equal magnitude in core and experiment results in a negative reactivity effect.

5.3. Experimental Handling

Experiments initiated by B&W will generally be fabricated and assembled within the NDC complex. Experiments by others will be fabricated and assembled to BAWTR requirements. Prior to the initiation of insertion procedures, all experiments must meet certain acceptance criteria and receive the BAWTR Safeguards Committee's approval.

Prior to a scheduled BAWTR shutdown, the experiment is moved into the containment staging area. Monitoring and control instrumentation are setup in the assigned space, and the necessary connecting lines and wires are readied at the assigned location at the edge of the pool.

After shutdown and the removal of terminating experiments, the new experiments are inserted. A loop experiment would be inserted into the in-pile thimble through the top closure assembly and positioned by extensions of the flow director and bundle assembly. The positioning extensions would enclose and protect the instrumentation leads extending from the test bundle against the turbulent water flow at the inlet and outlet loop coolant line connections. The instrumentation leads would penetrate the closure by using the proper high temperature, high pressure fittings. The leads then extend upward through the pool water to the side of the pool and are connected to the interconnecting leads to the monitoring and control instruments.

Non-lead capsule-type experiments would be inserted into the irradiation position in support baskets. With this arrangement, almost any size capsule can be handled and positioned. A non-lead capsule located within one of the reflector positions can be no larger than the hole size less spacing for the basket and the required water coolant channel. A lead capsule may be inserted allowing for only the water coolant channel provided additional clearance for the leads is not required. Such an experiment will employ the leads as a handling device.

On the termination of an experiment, loop, or capsule, the instrumentation leads, if any, will be severed at the poolside and water proofed if required. A capsule then would be removed from the irradiation position and passed under water through the open pool and canal gates into the canal for storage or insertion into the high level hot cell through the canal flow plug.

Discharge of a loop experiment starts with the shutdown of the loop after an adequate cooling period. During and after loop shutdown, the test would be monitored so that overheating will be prevented. After severing and protecting the instrumentation leads, the top closure is removed followed by the experiment removal. The experiment is transferred to the canal for storage after an inspection as described for the capsule.

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"Figure 5.1 Proposed Pressurized Water Test Loop"

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D-03

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THAT CAN BE VIEWED AT THE RECORD TITLED:

"Figure 5.2 Proposed Pressurized "Water Test Assembly"

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D-04

6. ORGANIZATION AND PROCEDURES

6.1 Organization

Figure 6.1 shows the B&W organization with particular reference to the BAWTR. The Director of the Nuclear Development Center reports directly to the Vice President of the Research & Development Division.

A Safeguard Committee will advise the Director of the NDC with regard to safety and technical soundness of changes or proposals under consideration for the NDC. The Safeguard Committee will act as an independent group to review operation and changes to the Facility. This Committee will be made up of senior scientists from the Research and Development Division and the AED. On matters of safety, this group will advise the cognizant NDC Managers, and their reports will also go to the General Manager of AED in Lynchburg. They will review any matters in which a safety problem is involved including experiments, operating procedures, and facility changes.

Staffing of the facility has not yet started. The key staff members will be recruited from people who have had considerable practical experience with the everyday problems of reactor operation. The key staff members will train the additional operators and supervisors during the initial performance testing of the reactor. These trainees will also prepare procedure and control manuals to be used in plant operation. A training officer on the staff will study the educational and training needs of the organization and will make arrangements for the required instructions.

Reporting to the BAWTR Manager are two sections: one involved principally with reactor operations, and the other with project engineering for experiments being performed within the BAWTR. The first section is devoted exclusively to reactor operations and experimental observations. The operations chief will have four shift supervisors reporting to him. Each shift supervisor and the operators reporting to him will be licensed for BAWTR operation. The responsibilities of the operations section will be to perform all maintenance required to keep the BAWTR at peak performance. All fuel cycle work associated with the BAWTR core will be handled by this section. This will include ordering of new fuel, and unloading and disposition of spent fuel elements. Taking of all data during any irradiation period will be accomplished by this section. This includes both reactor operational data and experimental data. The operations section will also be responsible for the performance of any special critical experiments or experiments designed to measure reactivity effects of various devices to be ultimately tested at high power.

A Project Engineer from the Project Engineering Section will be assigned to each irradiation experiment being conducted within the BAWTR. In practice, one project engineer may have several projects under his jurisdiction. Responsibilities of a Project Engineer will depend on the type of project involved. The principal variation is the manner in which he participates in the design of the experiment as well as in the actual performance evaluation of the experiment. In every experiment, a Project Engineer must satisfy the BAWTR Operations Chief and the Safeguards Committee that the experiment will be carried out safely. The Project Engineer will have the responsibility of coordinating all design and fabrication work associated with each experiment. He will also coordinate nuclear and thermal analysis needed in optimizing irradiation experiments. He will also prepare hazards summary reports as needed for the purpose of obtaining BAWTR license amendments. He will work with the BAWTR Operations Chief to establish the necessary operating procedures and data taking procedures to be employed during irradiation cycles for his assigned project. He will participate in coordinating all operational efforts during the period of transition to a shutdown wherein an irradiated loop is removed or parts of it removed and his new experiment is installed. After irradiation, he will participate in the transfer operations to remove the unit to a hot cell or for transport to an outside area by a cask. In order that the Project Engineer may more fully understand and apply the safety aspects of the reactor to the design of experiments, he will be required to spend time within the BAWTR Operations Section.

6.2 Policies and Procedures

It is not possible to describe operation procedures at this time, but they will be contained in the application for an operating license for the BAWTR. Procedures will be established to cover mode of operation, reactor operation, laboratory operation, experiment handling, waste disposal facility modification, fuel handling, health and safety, and emergencies. Reactor operating procedures will include fuel handling and loading, reactor pre-startup, startup, operation, recovery from scram, shutdown emergency, abnormal operating, and maintenance schedules. These will follow definite written procedures including step-by-step check lists. These procedures will enable the reactor operators to know and follow the proper method and limitations of reactor operations so that the operation will be followed in an orderly and safe manner. The supervisor in charge will have the responsibility to initiate, follow, and verify the compliance of the operations as called for in the procedure.

6.3 Maintenance

The operating cycle of the reactor will vary over a period of 150 to 240 days. The length of the cycle will depend on the type of experiments to be performed and the limitation on the reactivity of the core. This cycle will not be rigidly followed and may vary if the major experiments require a different time of irradiation. Normal maintenance for the facility will be done as is needed under the cognizance of the BAWTR Operations Chief. Major maintenance or changes to the reactor will require that the reactor be shutdown, and generally will be done during a refueling period. It will be possible to shut the reactor down at any time to do necessary maintenance work.

Maintenance will be done under strict regulations and with approval of the Health Physics group. During the performance of work in any contaminated area, the Health Physics group will be responsible for monitoring and assuming safe working conditions for the maintenance workers and assuring that proper protective clothing is worn when needed. Applicable health safety procedures will be followed during all maintenance work.

6.4 Radiation Protection

All individuals working with radioactive materials and sources of ionizing radiation or who may be exposed to significant levels of external radiation will be trained in the proper use of portable survey instruments. Routine monitoring for exposure or contamination control will be accomplished by the individual or group directly concerned whenever feasible. In instances where the safety of personnel involved critically depends on proper techniques or operating procedures, a work permit system will be utilized and Health and Safety personnel will be actively involved in the work situation.

Personnel monitoring equipment consists of portal monitors, hand and shoe counters, and count rate meters. The latter two instruments, equipped with survey probes, will be located at strategic points throughout the facility for personnel contamination surveys.

6.4.1 Personnel Monitoring Devices

All individuals admitted to the BAWTR will be issued and required to wear personnel monitoring devices. Personnel assigned to work in areas where significant exposures are possible but not probable will wear a beta-gamma film badge. Personnel assigned to work with sources of ionizing radiation or assigned to areas where potentially high levels of radiation are probable will wear film badges and pocket ionization chambers.

6.4.2 Bioassay Program

All individuals assigned to work with radioactive materials or to work in hazardous areas where potentially high concentrations of airborne radioactivity may be encountered or where significant contamination of the person or work area is possible will be required to submit biological specimens. Specimens may be collected on a non-routine as well as on a routine basis. Where iodine exposures are suspected, thyroid uptake measurements will be made with a scintillation counting system.

6.4.3 Personnel Decontamination

The decontamination of personnel will be accomplished as quickly as possible. Emphasis will be placed on self-help. Mild hand soaps with water will be used as the initial step in the decontamination process. All injuries involving contamination or potential contamination or injuries occuring in controlled zones will be reported to the Health and Safety Officer with referral to a physician as indicated.

6.4.4 Medical Examinations

All B&W personnel on an assigned status at BAWTR will be given routine physical examinations by physicians at least once a year. These examinations will be directed toward determining the normal or assumed pre-irradiation condition of each individual; special emphasis will be placed on the detection of any abnormalities that might later be associated with radiation exposures.

6.4.5 Personnel Exposure Records

Exposure records will be maintained on all individuals for whom personnel monitoring is required.

6.4.6 Personnel Protection

Protective clothing will be worn by individuals working with radioactive materials, contaminated materials or items of equipment, or while working in zones where significant radioactive contamination is likely.

Also, respiratory protection and other standard safety items will be available for use in hazardous areas. A safe work permit system will be used to control specific jobs where unusual or abnormal hazards are likely to be encountered or where close coordination of work affecting other operations is necessary.

Protective clothing such as coveralls, lab coats, shoe covers, surgeons caps, gloves, etc., will be available for routine use in controlled zones. Specific items of protective clothing required in each controlled area will be posted at all entrances. Additional items of protective clothing or safety gear will be specified by the Health and Safety group as required.

Protective clothing or other apparel and safety equipment will be monitored at established transition points when going from zones of higher contamination to zones of lower potential contamination. Garments contaminated to levels exceeding that permissible for the lower level zones will be exchanged for clean items at the transition point. Instructions will be given on the proper use of protective clothing.

6.4.7 Portable Monitors

Portable direct-reading radiation survey instruments of the Geiger-Muller and ionization type, portable and fixed air sampling equipment, and counting room equipment including gas flow, proportional, scintillation, and end-window GM counting systems are available for Health and Safety assays.

6.5 Security

Physical security of the facility is important as a safety measure and for the protection of classified information. NDC physical security will be made up of the following.

6.5.1 Physical Barriers

The NDC will be surrounded by permanent security fencing; the front wall of the main building will act as part of the front border of fencing. The fencing will include two locked vehicular access gates. The front door to the main building will be locked and of heavy duty construction. The windows along the front of the main building will be of the "permanently closed" design with heavy glass construction.

6.5.2 Security of Matter in Storage

Security matter other than SSNM will be stored in a heavy-duty steel safe-type filing cabinet. SSNM will be stored in a vault-type room with an automatic alarm system.

6.5.3 Protective Alarms

A central station alarm system will be used as an aid to the full-time armed guard force. In addition to the vault alarm, all doors to buildings will have complete circuit type contact alarms and all alarms will be connected to an alternate source of power.

6.5.4 Protective Lighting

The protective lighting system will illuminate all perimeter fencing with overlapping light distribution. The front face of the main building will be floodlighted. No lights that directly or indirectly use gas will be used, and all protective lighting will be connected to an alternate source of power.

6.5.5 Personnel Identification

A pass-badge exchange system will be employed at the front door of the main building during normal working hours and by the full-time guard force at other times.

6.5.6 Guard Force

The NFP Guard Force will perform all of the functions appropriate to the proper maintenance of NDC security, such as routine patrols as well as a physical inspection of the facility once every hour during other-than-normal working hours. The guards will monitor the central station alarm system on a full-time basis and maintain a constant check on employees and visitors entering and leaving the area.

6.6 Incidents

It is not easy to predict a large portion of incidents or accidents which could occur to a facility such as the one under consideration. However, it is possible to classify most of the accidents under general categories, such as fire, explosion, activity release, etc., and provide emergency procedures to safely handle each type of accident. To safely handle incidents, an Emergency Control Staff, under the direction of the Health Physics Department, will be established to function in case of an accident at the plant. When an emergency is shown to be localized and small in magnitude, the Emergency Coordinator on duty may delegate the control of the emergency to the supervisor in whose area the emergency has occurred. In case of a major incident requiring extensive emergency action, the Emergency Control Staff will be responsible for the resulting action to be taken.

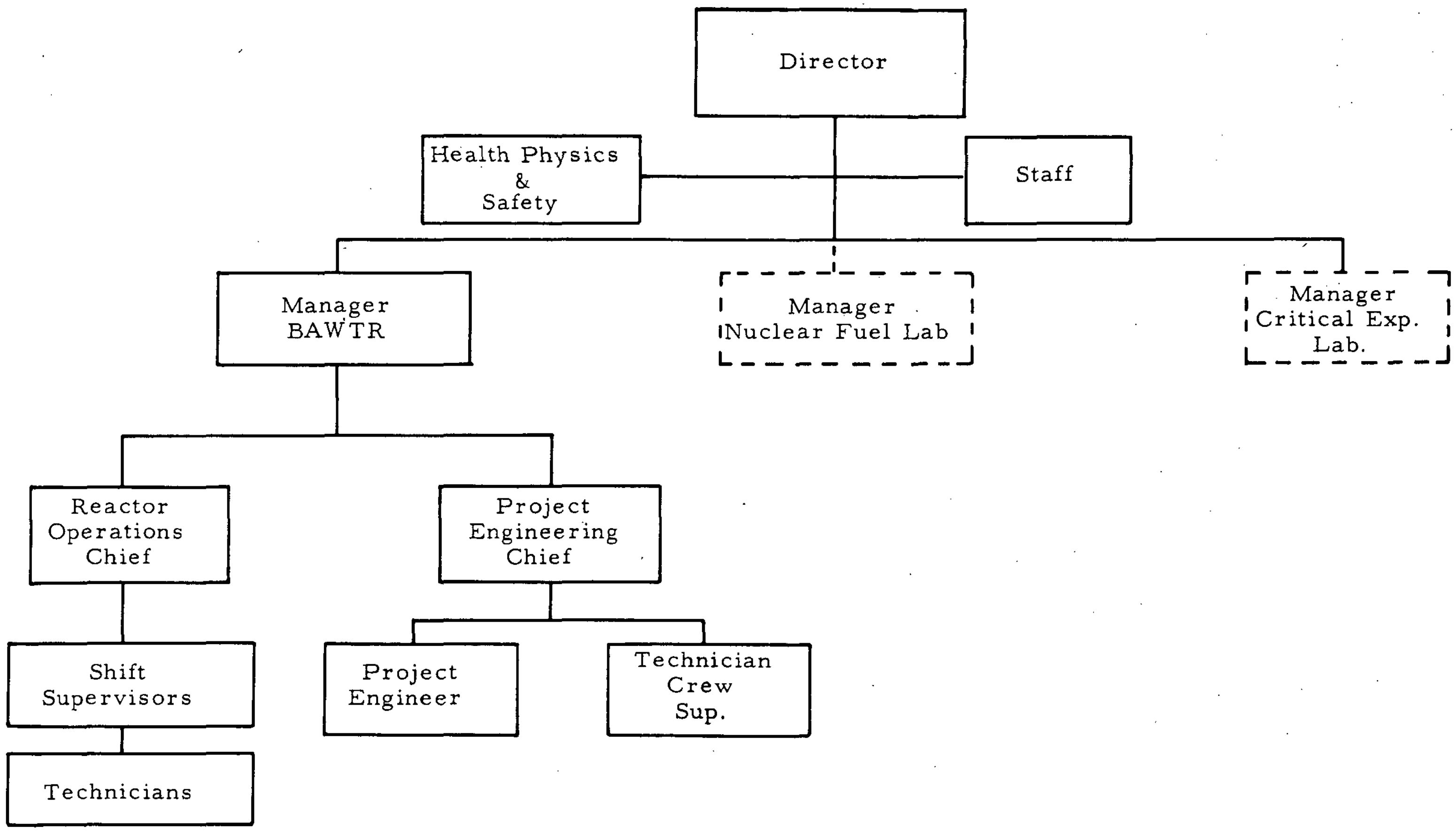
6.7 Evacuation

Evacuation routes will be established and rehearsed as required by the nature of the hazard and the location. Special attention will be given to areas of limited access or where procedural methods are required (air lock doors, etc.). As evacuation routes are established or changed, signs will be posted.

The signal for evacuation of the site or fenced area will be normally given by the Emergency Coordinator. In the event that the alarm is given without his authority, he must be notified immediately. On hearing the evacuation alarm signal, the reactor and experiments will be shut and secured. Non-essential equipment and equipment representing a hazard if left operating unattended will also be shut down. Personnel will immediately leave the fenced area in an orderly manner along the predetermined routes. All building exterior doors will be left closed but not locked.

Site evacuation will proceed with all personnel proceeding to the parking lot entrance. All drivers will report with their cars to the parking lot entrance to receive passengers as directed by the Security Guard. The drivers will be instructed what route to take to an off-site reassembly area. Evacuation of the area may be initiated by either the local RAMS alarm or a vocal command over the paging system.

Figure 6.1. NDC Organization Chart



7. ACCIDENT ANALYSIS

The initial operation of the BAWTR will be at a power level of 6 MW with one experimental loop located in the core. Future plans are to extend the core size to accommodate two experimental loops and to raise the operating power level to 12 MW. The exact time this power increase is to be made has not been decided, but the power increase has been considered in the discussion of the major accidents that could happen to the reactor. The hazards analysis of the reactor at the 12-MW power level shows that it can be operated safely at the site and in no case will the public be exposed to a dose as great as 300 rem to the thyroid or to a 25-rem whole body dose due to an accident with the reactor.

7.1. Nuclear Fuel Laboratory

The NFL, as described in Section 3.3, will be licensed and operated under applicable Federal authority. Chemical, mechanical, and spectrographic examination of irradiated and non-irradiated nuclear fuels and other material samples will be made in the laboratories of the NFL. Assembly, dissassembly, and examinations will be made of fuel elements utilizing new, spent, and recycled fuels. Provisions are made for collecting, monitoring, and analyzing samples of gas and fuel in the radiochemical and spectrographic laboratories.

All fuel materials will be stored and handled in such a manner that a criticality accident will not result. These precautions are presently followed in the NFP, which is operating at the site and is licensed under the same fuel handling restrictions and authority as the NFL will be in reference to nuclear fuels.

The only other possibility for an accident at the NFL is the release of quantities of activated material to the building areas. These releases could be in the form of gases, liquids, or solids. However, the handling of any radioactive material will be done only in a manner and location that presents no hazard to the operating people or the public.

In case of a release of gas into areas of the hot cell or laboratories, the design of these areas is such that the gas is confined to a small area and would be contained or discharged up the stack in quantities presenting no problem or radiation above those of 10 CFR, Part 20. It is conceivable that a situation could arise in which it becomes necessary to evacuate certain areas of the NFL. This condition would be detected and annunciated by the radiation detection system at locations in the buildings, and an orderly and safe evacuation would be made.

The buildings and handling facilities of the laboratories are designed so that any liquid spill will be confined to the area in which the material is being handled and would not be spread to the remainder of the building. In each area, liquid collection and storage capacity is supplied for handling any radioactive spill through the normal liquid waste disposal system. Here, as in the gaseous release of wastes, high levels of activity would be detected in any area and evacuation procedures would be carried out.

The release of solid radioactive wastes into any area is considered somewhat remote. If this release occurred,it would necessitate an instigation of emergency procedures and could possibly require the evacuation of certain areas of the NFL with cleanup procedures being used to remove any activity. However, the laboratory will be equipped to safely handle any such occurrence. In no event will a release of activated wastes present a problem or hazard to the public.

7.2. Reactor

7.2.1. Safety Features

Many features of a reactor of the BAWTR type naturally act to prevent serious damage to the reactor core in case of intentional or accidental reactivity insertion. In addition to the inherent protection in this type of reactor, there are many additional safety factors and conservative design features built into the BAWTR. Some of the important protective features are given in the following paragraphs.

7.2.1.1. Negative Temperature and Void Coefficients of Reactivity

The reactor will always be loaded in such a manner that the over-all temperature and void coefficient of the core will be negative and will tend to decrease reactivity due to a power rise in the core from any cause. For any significant changes in the core or experiments, a redetermination of the temperature coefficient of reactivity will be made to assure that it has a negative effect on the core. In general, this will be checked by noting control rod position in the core as an approach to operating power is made. However, prior to the actual start-up of the reactor with core or experiment changes, this value will be first determined by analytical methods.

7.2.1.2. Low Available Excess Reactivity

The amount of available excess reactivity that may be inserted into the core at any time will be limited to 1.5% as a range insertion at a rate of no greater than $5 \times 10^{-4} \frac{\delta k}{k}$ or a 0.6% step-increase. This will be accomplished by reactor design and experiments to prevent greater amount from being available at any time. Much of this will be done by using a reliable and safe system of interlocks on the control rods and drive systems. Also, careful design of experiments, the reactor core, and the associated systems are provided.

Experiments performed at SPERT, BORAX, and other light-water moderated reactors indicate the inherent safety of this type of reactor from excursions with a limited amount of available excess reactivity.

7.2.1.3. Control System

The large amount of reactivity control built into the shim and safety rods gives assurance of protection against a single failure allowing the insertion of enough reactivity to damage the reactor. This is assured since the shim rods alone have sufficient control to completely shut the reactor down in case of necessity. This is also true of the safety rods, which always remain in a cocked position above the core during reactor operation. At any time during a reactor cycle, the reactivity available in the core is far less than can be met by the stuck rod criteria for either of the safety rods. An additional feature of the control system is the speed with which any rod or group of rods can be withdrawn from the reactor. This feature mechanically limits the rate of reactivity insertion by control rod removal to a value low enough to prevent any serious core damage by continuous rod withdrawal.

7.2.1.4. Adequate Reactor Cooling

In case of a failure of the primary coolant system from the loss of the primary pumps or by a rupture of the primary coolant lines, there is a large quantity of emergency cooling water available. This is first supplied by the water stored in the reactor pool. In addition to this, the plant water supply is stored at such a level that it can easily be supplied to the reactor if necessary. It is doubtful that any accident or emergency condition would ever require use of the plant water. Ample cooling to the core is always available from the water contained in the reactor pool.

7.2.1.5. Site

One of the best safety features of the BAWTR is its site. As is demonstrated in the hypothetical accident, using 10 CFR, Part 100 conditions for the 12-MW BAWTR, the site is suitable for a reactor with a higher power level factor of 9. The natural features of the site, such as rivers and mountains, prevent a significant buildup in population within a radius of about 2 miles from the BAWTR location. Also, the low population zone presently is a distance of 3 miles from the location, the exclusion distance is about 1300 feet, and the nearest population center is a minimum of 3 miles from the site. For a 12-MW reactor, 10 CFR, Part 100 suggests that the exclusion area be 475 feet, the low population zone be 0.56 mile and the population center be 0.75 miles from the reactor. Any spilled wastes or effluent from the plant have a great distance to travel and a long time factor prior to reaching any populated areas.

7.2.1.6. Reactor Designs

The mechanical and hydraulic features of the reactor are designed conservatively. Operating parameters of the BAWTR, including heat transfer and hydraulic characteristics, are comparable to those of existing reactors that have been operated for long periods of time. Of particular significance is the successful operation of reactors with similar fuel elements, such as MTR and ETR at NRTS and ORR at ORNL. Most of the mechanical components to be used in the BAWTR have been in extended use previously in operating reactors and will require no testing or development programs. Reliability of these components has been demonstrated over long periods of actual service in operating reactors.

7.2.1.7. Experiments

The largest safety factor of the experiments will be the reactivity limitations imposed upon the design of any loop type or large capsule placed in the reactor. They will be designed so that any failure will not cause a reactivity increase greater than can be handled by the inherent safety characteristics of the reactor and the safety system. They will also be designed so that three degrees of containment, fuel cladding or capsule, experimental tube walls, and the containment vessel must be breached to release any fission products to the atmosphere. If the first two restrictions are penetrated, the resulting pressure and steam that could result from failure is vented through the pool water which would remove all the steam formed and most of the irradiated products produced in the experimental loop. In the event of a rupture of the out-pile test loop equipment, the steam released will be diverted through pipes back into the pool a few feet below the surface of the pool, and the steam will be condensed and will not be released to the containment.

7.2.1.8. Containment

The containment vessel for the reactor is a steel shell of a type and construction similar to those used and successfully tested for leakage in many large reactors in the United States. It is not to be used as a pressure vessel since there is no pressure buildup in the vessel as a result of any accident to core or experiments. This is a considerable safety factor for the BAWTR, and the integrity of the vessel would certainly lower, by many factors, the calculated results of the hypothetical accident considered in the accident analysis.

7.2.1.9. Experienced Personnel

It is not possible to evaluate the contribution to reactor safety that is a direct result of the experience of people who operate the reactor. However, it is significant that the CEL has been operated since March 21, 1957 without incident. This operation has involved the running of the Lynchburg Pool Reactor (LPR) and about eight critical experiments on a large assortment of core types and sizes. Much of the experience gained in the operation of the CEL reactor will be used to operate the BAWTR safely.

7.2.2. Electrical Power Failure

If there is a failure of the electrical supply to the primary pump, a flow coastdown begins, and at a flow of approximately 1100 gpm (in 3 seconds), the lower header at the bottom of the core falls open. A signal from the pump scrams the reactor and the safety rods start to insert in 0.12 seconds after the power loss. At the time the lower header drops, the power level is about 10% of the 6-MW operating power. Thermal calculations indicate that natural convection cooling in the BAWTR core is adequate to remove the heat produced from a 0.9-MW power level. Thus, at the time forced coolant circulation ends, the reactor power level is lower than the level which can be safely accommodated by natural circulation. Figure 7.1 shows the coolant flow coastdown, reactor power, and the coolant flow at which local boiling occurs for the loss of the pump power accident. Figure 7.1 also shows that convection cooling is sufficient to cool the core following a reactor power failure and will result in no melting of the fuel plates.

During a complete electrical power failure, the safety rod electromagnets release at the same time that power is lost to the pump. This condition is less severe than the case described above since the power will decay faster than is shown in Figure 7.1. The emergency generator comes on the line and supplies electrical power for the nuclear instrumentation and other essential parts of the plant and experiments.

The reactor core will be protected with proper instrumentation at all times by the emergency power system. There will be sufficient emergency power available to supply facilities for experimental loops that cannot tolerate an extended period without coolant flow. Thus, power failure will not result in damage to the core or experiment.

7.2.3. Primary System Failure

The primary cooling system may fail by primary line rupture, the lower header inadvertently opening, heat exchanger failure, and other accidents of a similar nature including electrical failures considered previously. If a return coolant line is ruptured, water may continue to be pumped from the reactor pool. When the water level reaches 20 feet above the core, the low water alarm would automatically sound in the control room. The reactor would then be scrammed and the coolant pump stopped. Emergency procedures would call for procedures to further loss of water from the pool. These procedures would include closing the valves which are provided in the system to isolate the primary system outside the pool.

It is not expected that the lower header located at the bottom of the core could inadvertently be opened during reactor operation. However, if this should happen, the reactor would immediately be scrammed by the provided interlocks. The primary pump continues to operate and the flow of coolant through the core is reduced to approximately 20% of full flow. The reactor power drops to 10% in 0.5 seconds. Therefore, adequate cooling is available to cool the core and no damage will result.

A heat exchanger failure may result in the release of a small quantity of activated primary coolant into the secondary system. This would cause the radiation monitor for the secondary system to alarm in the area and the control room. The reactor would then be manually scrammed with the primary coolant flow stopped. The levels of the monitors are set to alarm at a level which would not cause any hazard to operating personnel. It would be necessary then to pump the contaminated secondary cooling water to the holdup tanks for examination and final disposition. A sufficient capacity is provided to permit holdup of the contents of the secondary system.

7.2.4. Secondary System Failure

A normal loss of the secondary cooling system activates alarms in the control room. As a result, the reactor would be scrammed and would operate on convection cooling with no difficulty or problems arising. In the unlikely event that the reactor continued to operate after a loss of secondary cooling, the resulting temperature effects would shut the reactor down prior to any core damage due to overheating.

On the loss of secondary cooling which failed to scram the reactor, the pool water temperature would continue to rise. The pool water temperature would increase about 75 F which add about 1.5% negative reactivity to the core. This is sufficient to completely shut the reactor down by compensating for the reactivity available in the control rods. Therefore, the inherent temperature characteristics of the core tend to prevent core damage due to the loss of secondary cooling.

7.2.5. Loss of Pool Water

Precautions and reactor design prevent the loss of large amounts of water from the pool. This is accomplished by a valve in the exit water line which is closed on the loss of coolant flow and an anti-syphon design for the inlet water line to prevent syphon of pool water in case of line failure. Water level indicators on the pool act to scram the reactor when the level of the pool drops to 1 foot below the normal operating level. This instrumentation is also backed up by the area alarms which scram the reactor due to high irradiation levels in the containment vessel. These alarms will also stop the primary pump to prevent pool water from being pumped from the reactor.

The walls of the pool are reinforced concrete lined with a plastic coating. There is no conceivable way in which a large break could occur in the pool walls that would allow large quantities of pool water to escape in a short period of time. If a fracture or crack should be made in the pool wall, it is possible that a small leak could occur. This would result in the release of some quantity of activated water into the containment vessel where it is treated as liquid waste. This type of accident would not result in pool drainage because any leak of this type could be compensated by plant water which can be added to the pool at a rate of 100 gpm for practically an infinite period of time. A complete and immediate loss of pool water is not considered a realistic accident for this reactor, and with precautions taken, no hazard is presented to the public from pool leakage.

7.2.6. Loss of Instrument Air

The primary and secondary pressure transducers and most of the non-nuclear instrumentation for the plant will be operated by compressed air. On low air pressure or with a complete loss of compressed air, alarm signal will be actuated. The reactor coolant flow is not affected by the loss of instrument air. The nuclear instrumentation does not utilize compressed air and will monitor the nuclear behavior of the core at all times.

7.2.7. Fuel Element Failure

A fuel element cladding failure can release both volatile and non-volatile fission products to the primary coolant system. These products would increase the activity levels of the coolant and the gaseous wastes.

When the activity level in the gaseous wastes disposal system exceeds a preset level, the reactor will be shut down automatically and the damaged fuel element will be removed. Building monitors will also detect the increased activity in the building, and appropriate alarms and signals are given in the control room. This allows the operator to scram the reactor to prevent a more serious contamination. Cleanup of the water can be accomplished by circulation through the demineralizers and by removal of some of the contaminated water to the holdup tanks for later disposal. A containment cask will be placed around an element to confine the fission product release. Procedures for handling a damaged fuel element will assure that the public will not be exposed to concentrations greater than those specified in 10 CFR, Part 20.

All areas are monitored to assure that the dose rate from fission products released to the primary coolant does not constitute a direct radiation hazard to personnel during operation. Shielding in all normally accessible working areas is thick enough so that operating personnel would not be exposed to high dose rates from the primary coolant loop or the heat exchanger as a result of a fuel element failure.

7.2.8. Start-Up Accident

Primary protection against a start-up accident is provided by a period trip which is effective from 10^{-4} to full power. Additional protection is provided by the high flux level trip.

Figure 7.2 shows the peak power level for a start-up accident which occurs at various reactivity addition rates. Only the high flux level trip is assumed to be effective. Figure 7.2 also shows the total energy released in the core while the power level is above the scram level. If it is conservatively assumed that all heat generated in the fuel at the hot spot remains in the fuel, a total energy release of 5.5 MW/sec will just raise the hot spot to the melting temperature. Using this conservative assumption for the energy release limit, reactivity addition rates below 5.4×10^{-4} δk/sec will not cause core damage.

In the BAWTR, the maximum worth of grouped shim rods is about 1.6% $\delta k/k$. Withdrawing a 1.6% $\delta k/k$ shim rod group at its maximum speed and at the position of greatest differential worth results in a reactivity addition rate of no greater than 5×10^{-4} $\delta k/k/sec$. With this reactivity addition rate, the core is protected from damage even if the period trip is inoperative.

7.2.9. Cold Water Accident

The possibility of a serious cold water accident in the BAWTR is non-existent since the water supply for reactor cooling is only 12.5 F cooler than the exit water from the core. There is no other available source of a significant quantity of cold or cooler water that could be injected into the reactor to produce a cold water accident.

The only plausible accident of this type is a failure, during operation, of the lower header which admits the pool water to the core. The pool water temperature is normally about 15 F below the average exit water from the core. If it entered the core at this lower temperature, only 0.21% reactivity would be added. However, if this happened, the reactor would be automatically scrammed. The lower header interlock and the excess reactivity control of either safety rod is sufficient to compensate for a much higher step insertion of reactivity with no damage to the core. The cold water accident does not present a significant hazard to the BAWTR.

7.2.10. Excess Fuel Addition

All normal fuel additions and installation of experiments will be made under conditions in which all control rods are fully inserted and the reactor shut down. These operations will be performed under strict operating procedures to prevent mishandling of these components. The design of the components, fuel, experiments, and reflector assures that there can be no inadvertent interchange of these parts and an accident cannot result in a faulty location. Once the reactor is in operation, it is not possible to place fuel into the core or reflector and or to add reactivity in this fashion. This is prevented by the closely-packed lattice arrangement of the core.

The only foreseeable accident caused by an addition of excess fuel results from dropping or deliberately placing an additional fuel element on top of the core. This accident was investigated by calculating the amount of reactivity added to the core by placing one element across the top of the core and observing the effect on the core.

If the element could be positioned across the top of the core, it would add less than 0.5% reactivity to the core at normal operating conditions. This would immediately scram the reactor due to a power increase, and no core damage would result.

7.2.11. Experimental Failure

Experience in operating test reactors, such as MTR, ORR, ETR and others, has demonstrated that in essentially all experimental failures, the only hazard has been a release of activated products to the reactor building. These releases have caused a short-time evacuation of the buildings in a large number of cases, but have resulted in no property or personal damage of any consequences. It is not anticipated that such experimental releases would necessitate an evacuation of the Facility because the containment would be sealed to contain any accidental release. High level radiation alarms may occur which would scram the reactor and would necessitate decontamination action to be taken, but high dose rates would not be received by personnel.

A high pressure loop rupture which could result in a disarrangement of the fuel elements and a changed core configuration is considered unlikely. If such a rupture occurred in the core, it would probably move one or more fuel elements from a section of the core rather than affect the total core. This would result in an automatic scram of the reactor from the experiment loss, and at least one control rod would fall into the core giving sufficient control to completely shut the reactor down.

An even more unlikely failure is the complete explosion of a core section of the high pressure loop which moves all the core in an outward direction. The effects of such an accident were investigated by assuming that the fuel and surrounding reflector are moved out from the center under the force of the explosion in a cylindrical pattern. The results of the reactivity change versus distance that the core is displaced from its original position are shown in Figure 3.20. Here, it is assumed that the safety rods fail to function, but the shim rods, which are driven by flexible cables, are not moved from their position. The effect of this accident is to shut the reactor down by loss of reactivity.

7.2.12. Failures Causing Reactivity Additions

To prevent the addition of large amounts of reactivity to the core from the failure of experiments or components of the reactor system, the available excess reactivity that can be added to the reactor is limited to 1.5% slow ramp increase or 0.6% step increase. This limitation is achieved by proper reactor design, control over experimental reactivity, and design of the reactor instrumentation and control system.

If the reactor power level should increase beyond the set limits of the control system, the safety trip will limit the power overshoot by a safety rod scram at 120% of full power. The rods will drop into the core under the force of gravity in 0.48 seconds. Figure 7.3 shows the maximum power that would occur for a given step increase of reactivity before the safety rods could reduce the power or shut the reactor down. Figure 7.3 also shows the total energy released in the core while the power level is above the scram level. If it is conservatively assumed that all heat generated in the fuel at the hot spot remains in the fuel, a total energy release in the core of 5.5 MW/sec will just raise the hot spot to the melting temperature. It is seen from Figure 7.3 that a step reactivity addition of 0.66% is required to release the 5.5 MW/sec conservatively assumed as a safe limit. The 5.5-MW/sec limit is conservative because it assumes no heat release from the fuel at the hot spot. Even if the fuel plates were steam blanketed, some heat would be transferred from the fuel plate. Thus, a step reactivity addition up to 0.66% δk will not cause serious damage to the core.

7.2.13. Maximum Credible Accident

In this report, an analysis has been made of the accidents resulting from operational errors, mechanical failures, and various reactivity additions to the BAWTR. In each case, it has been demonstrated that with reasonable precautions and the operating limitation on the reactor there is no unnecessary hazard presented to the public or any serious damage to the reactor core. The operating personnel are also adequately protected by a system of alarms and interlocks which give warning of any condition that could present a radiation hazard. Operating procedures will require the evacuation of any area before dose rates reach an unacceptable level for a working area. No serious results are expected from the operating incidents, but consideration is given to a credible accident which may have serious consequences. This accident, considered the maximum credible accident for the BAWTR, is a failure of the central experimental loop which releases the fission products of the irradiated experiment. This type of accident could be brought about by a loss of pressure or coolant flow in the experimental loop which would fail to scram the reactor. It could also result from a pressure buildup in the loop which results in an explosive failure of the section of the loop in the reactor core.

The loop is designed so that a failure of any section results in steam or fission products released being discharged into the reactor pool. This feature condenses the steam and prevents a pressure buildup in the containment vessel. It also acts to greatly reduce the amount of fission products entering the vessel by absorbing a large portion of them in the pool water. It is not clear whether or not any solid fission products would ever escape the pool water. If so, they would be entrapped inside the containment vessel. If the location of the loop failure is outside the reactor core, no damage would result to the core due to the amount of cooling available for the core. If the temperature in the core did continue to increase, it would be shut down without damage by the negative temperature coefficient of reactivity. If the experiment explodes in the core, the resulting effects are considered in Section 7.2.11. It appears that no additional hazard would result from core damage.

For the analysis of the maximum credible accident, it is assumed that the typical experiment, as described in Section 5.1, has been operating in the reactor continuously for a period of 175 days with the reactor power level of 6 MW. The fuel in the experiment is enriched UO_2 mixed with ThO_2. The average power per fuel pin is 17.9 kw/ft, which is a total power in the experiment of 716 kw. The experiment fails, and as a result, 100% of the noble gases, 50% of the halogens, and 1% of the solids are released into the containment vessel. This represents about 15% of the gross fission products in the experiment. Of the iodines released to the containment vessel, it is assumed that only 50% of these escape to the atmosphere.

There is no pressure buildup in the containment vessel due to the release. However, the leakage from the vessel is 0.1% per day of the contained volume. The "average worst" dispersion conditions and the wind speed and direction do not change over the entire period of the release. Calculations were made of the doses due to iodine inhalation, strontium inhalation, and direct irradiation from the fission products contained in the containment shell. Decay of the products is considered to occur only during the time until release from the vessel and not during transit to the individual receiving the dose.

Calculations of the buildup and decay of individual isotopes in the fission product chains in the test sample were calculated by conventional electronic computing methods. The doses at various distances from the containment were calculated by conventional methods described in AECU-3066, "Meteorology and Atomic Energy". A review of the results of the isotopes present in the experiment at the time of release indicate the thyroid due to iodine inhalation is the controlling contribution.

Table 7-1 shows the calculated doses at various locations from the reactor. Figures 7.4, 7.5, and 7.6 are plots of the doses as a function of distance from the containment shell following an experimental failure which releases a large quantity of the fission products produced in an experiment. These calculations show that the public and the operating personnel are adequately protected and will not be exposed to emergency doses greater than those suggested in 10 CFR, Part 100 as a result of the maximum credible accident to the BAWTR. It is therefore concluded that the BAWTR can be safely operated at the site described and will present no undue hazard to the public.

Table 7-1. Doses (Rem) Due to Maximum Credible Accident

	CEL	NFP	900-ft River exclusion	1300-ft River exclusion	0.5 mile
Thyroid dose, iodine					
Exclusion, 2 hr	5.10	0.77	2.07	1.18	0.44
Low population, 30 days	3.00 (2)	4.75 (1)	1.28 (2)	7.30 (1)	2.45 (1)
Bone dose, SR					
Exclusion, 2 hr	2.85 (-3)	4.35 (-4)	1.20 (-3)	6.90 (-4)	2.25 (-4)
Low population, 1 week	1.55 (-1)	2.38 (-2)	6.70 (-2)	3.75 (-2)	1.22 (-2)
Direct irradiation, 15% of total					
Exclusion, 2 hr	6.40	0.15	1.25	0.35	1.35 (-2)
Low population, 30 days	1.75 (1)	2.85 (-1)	3.25	0.88	3.40 (-2)

7. 2. 14. Hypothetical Accident

The maximum credible accident previously described for the reactor is considered to be the maximum accident with any degree of credibility that could ever happen to the BAWTR with the operating and design limitations that will be in force. To investigate the consequences of a complete core failure, a hypothetical accident is chosen. It is hypothetical because there are no multiple failures or sequence of events resulting in an accident of this magnitude. This analysis is made primarily to demonstrate that the reactor is well within the limits as suggested or specified in various publications on site guides and criteria (especially 10 CFR, Part 100).

In the hypothetical accident, it is assumed that the reactor has been operating at a power level of 12 MW for a period of 600 days. This period was used to obtain the concentrations of all isotopes except Sr^{89} and Sr^{90}. For these, it was assumed that the core had operated only one cycle of 150 days. An accident occurs to the core causing a complete core meltdown which makes available the release of 100% of the noble gases, 50% of the halogens, and 1% of the solids of the fission product inventory. This represents about 15% of the gross fission product activity in the core at this time.

Fifty % of the iodine released from the core is absorbed on the containment vessel walls. This does not take into account any absorption of iodine by the pool water. A metal-water reaction for the accident is not considered to occur due to the difficulty in reacing the temperature and metal dispersion conditions necessary for such a reaction. There is no buildup of pressure within the containment vessel, and it is assumed that the leakage rate is 0.1% per day of the contained volume.

Environmental effects of the hypothetical accident are summarized in Table 7-2. The calculations supporting these results are included in Appendix B. These calculations show that there is no exposure up to 300 rem to the thyroid or 25 rem whole body to the public for an infinite exposure or to the operating personnel up to 2 hours exposure with proper evacuation procedures. These calculations were made under the assumption that the strontium utilized the same mode of release as the gases. This assumption was used because of the lack of a more defined model for strontium escape. Calculations were not made for the BAWTR for rainout and fallout doses. These studies have been made for the LTR which was not constructed at the site. They show that the rainout and fallout exposures are a factor of 100 to 1000 below the doses calculated for all the iodine isotopes. The results of the calculated doses are shown in Figures 7.7, 7.8 and 7.9. It is believed that the public is adequately protected from the results of a hypothetical accident to the BAWTR.

Table 7-2. Doses (Rem) Due to Hypothetical Accident

	CEL	NFP	900 ft River exclusion	1300 ft River exclusion	0.5 mile
Thyroid dose, iodine					
Exclusion, 2 hr	75.0	10.5	28.2	16.5	0.56
Low population, 30 days	3850	595	1500	935	300
Bone dose, Strontium					
Exclusion, 2 hr	5.3 (-2)	7.8 (-3)	2.1 (-2)	1.2 (-2)	4.1 (-3)
Low population, 1 week	3.1 (0)	4.6 (-1)	1.2 (0)	7.1 (-1)	2.3 (-3)
Direct irradiation, 15% of total					
Exclusion, 2 hr	103.0	1.92	19.5	5.6	0.24
Low population, 30 days	290.0	4.70	51.0	14.5	0.65

Figure 7.1. Flow Coastdown Accident

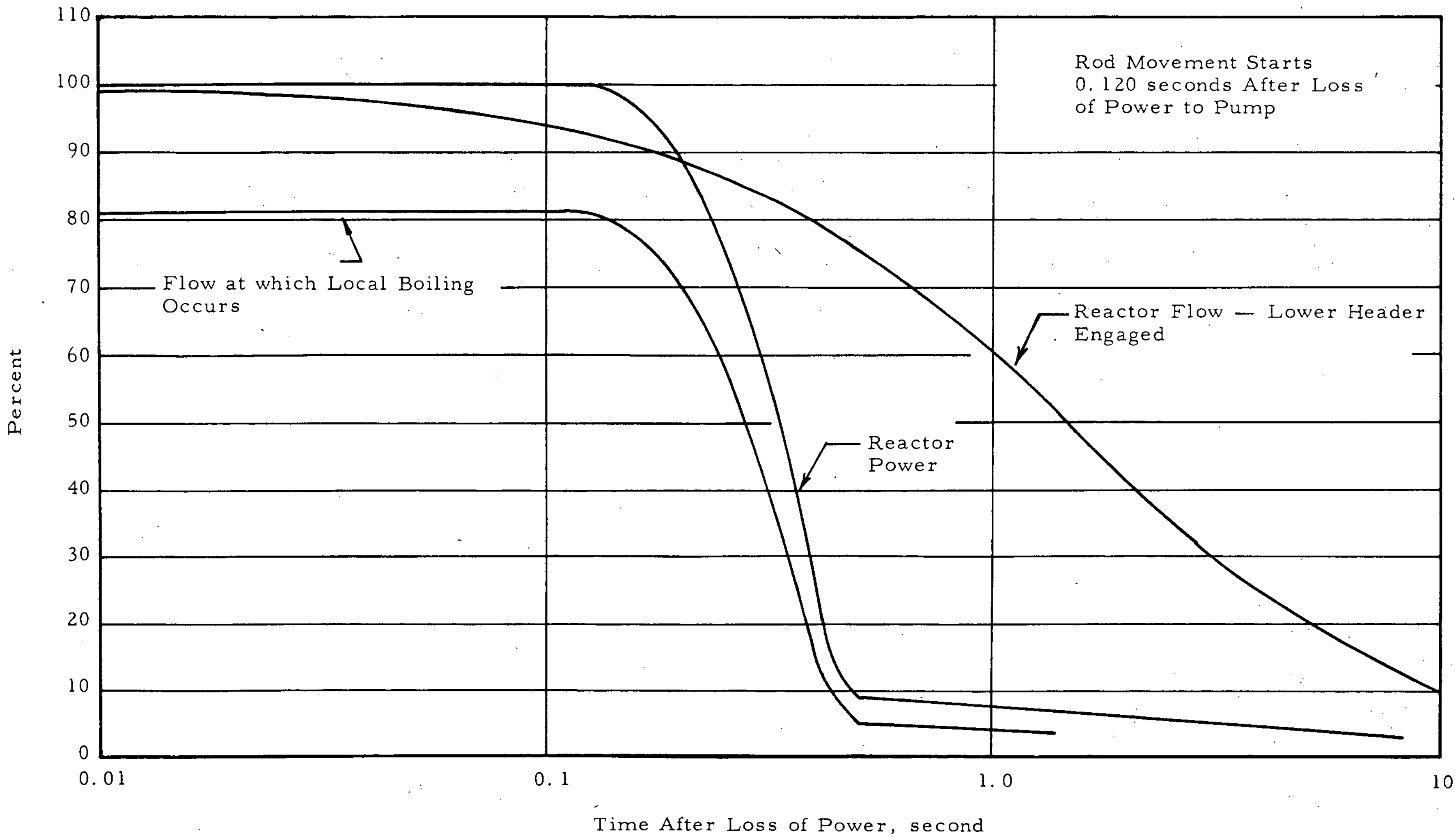


Figure 7. 2. Startup Accident

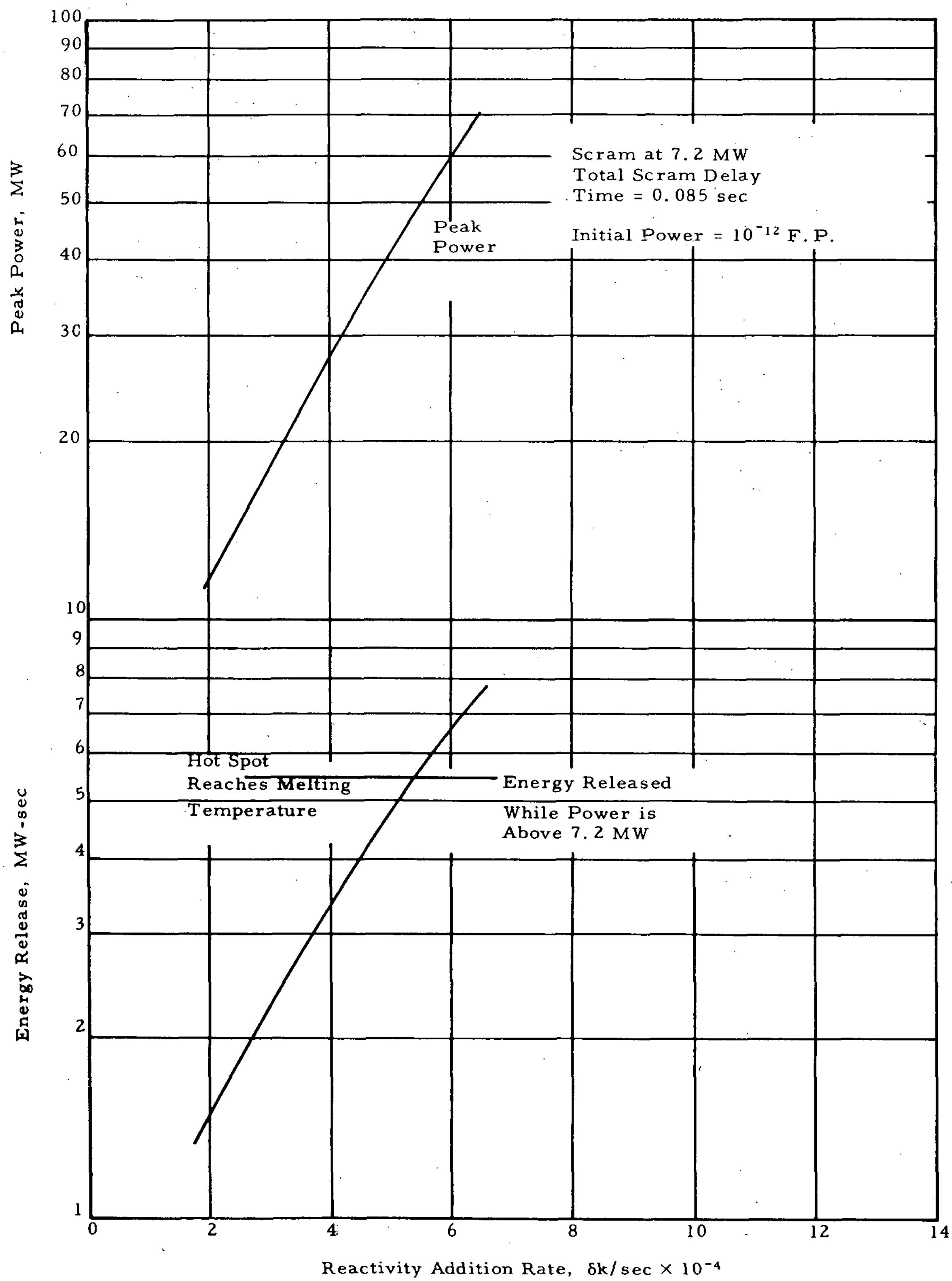


Figure 7. 3. Step Reactivity Addition

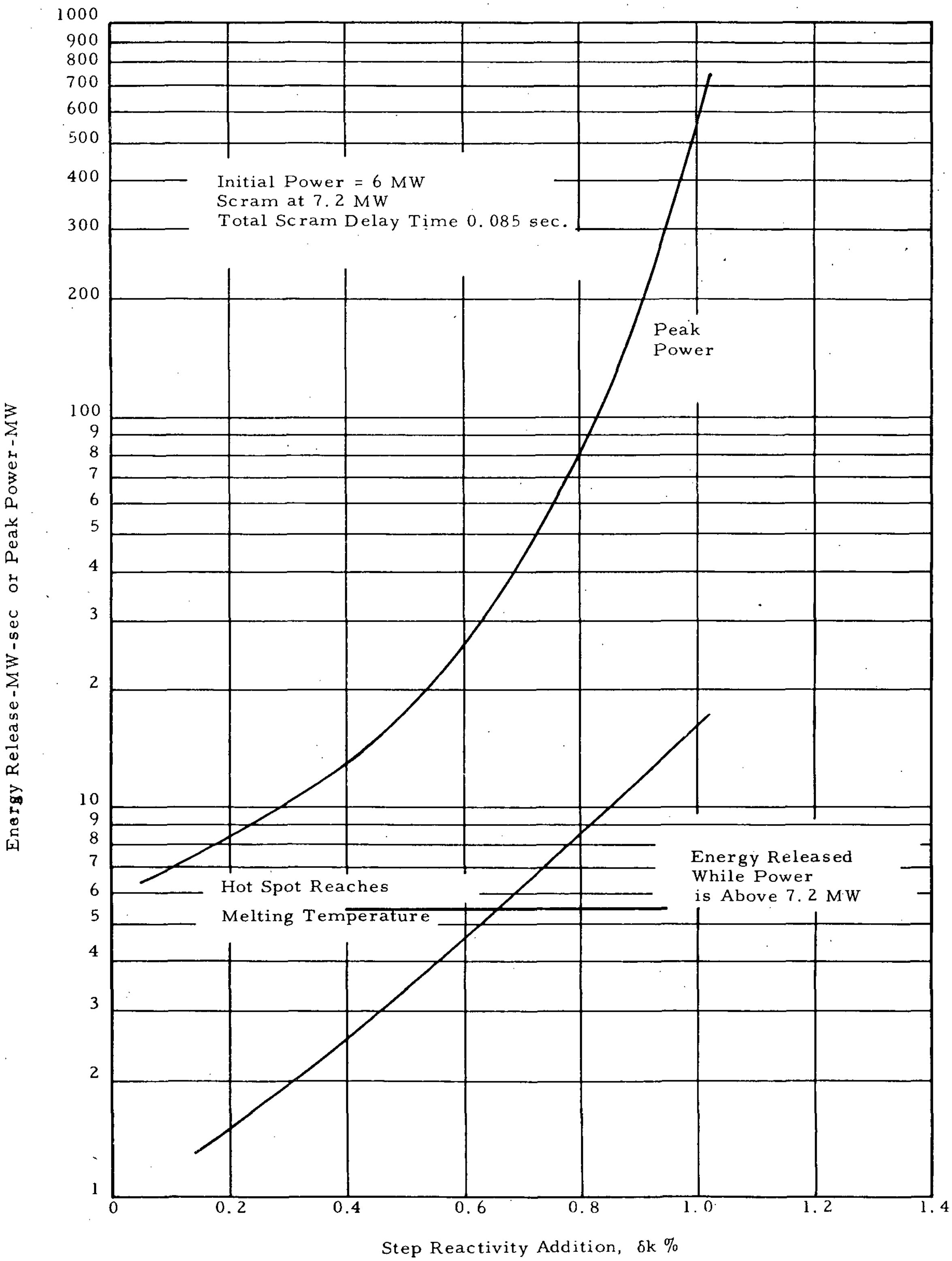


Figure 7. 4. Thyroid Dose Due to Iodine Inhalation, Maximum Credible Accident

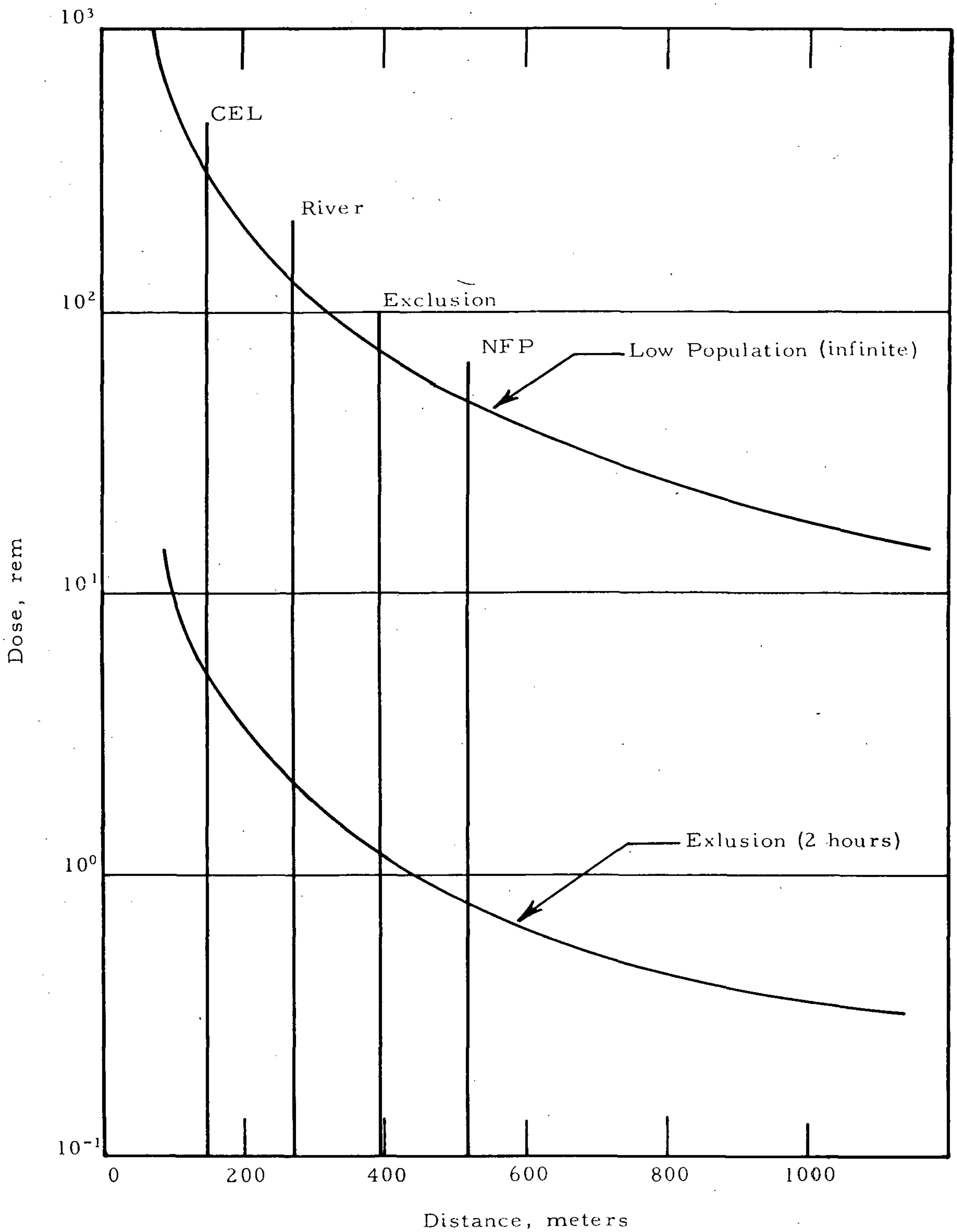


Figure 7. 5. Bone Dose Due to Strontium Inhalation, Maximum Credible Accident

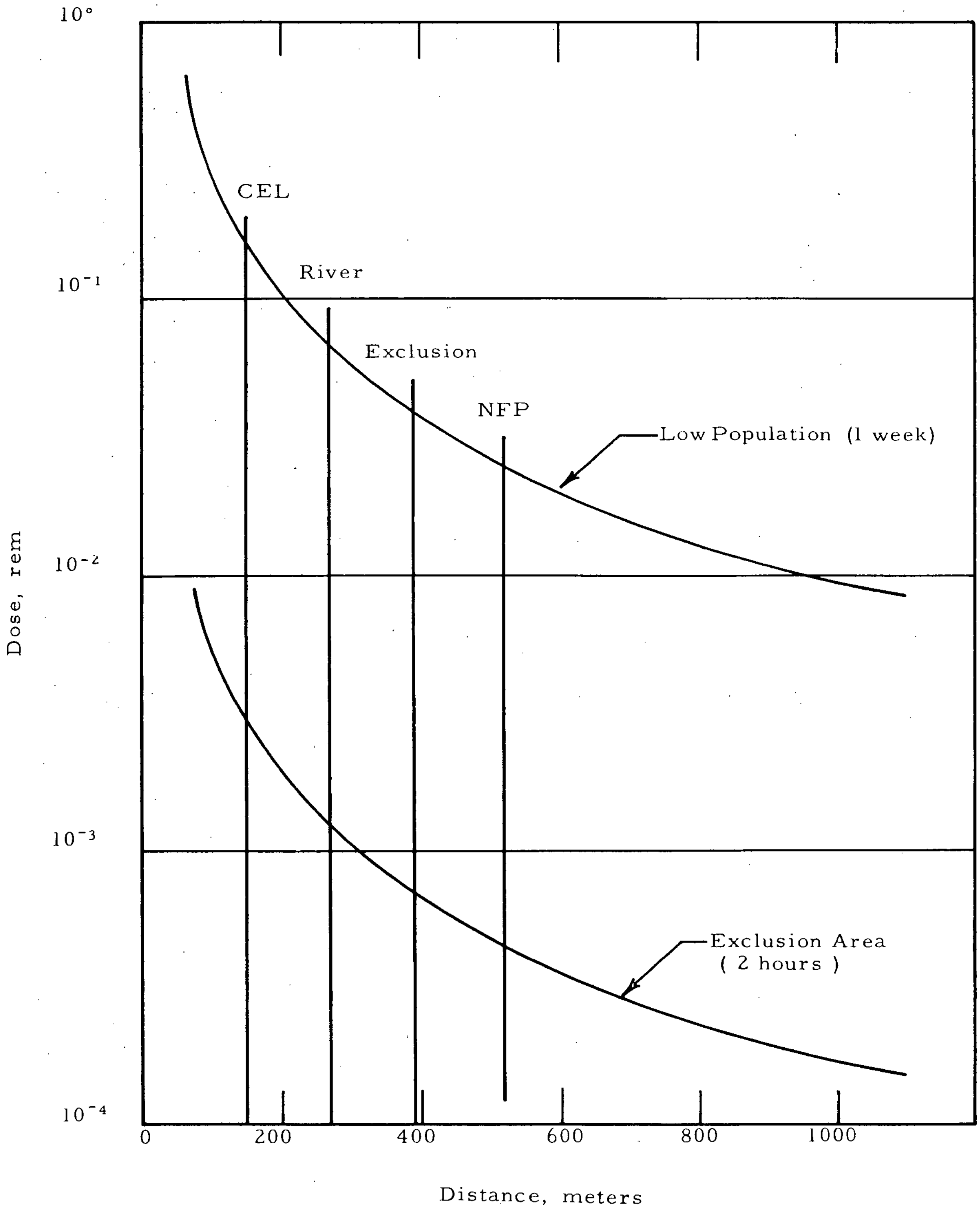


Figure 7.6. Body Dose Due to Direct Irradiation, Maximum Credible Accident

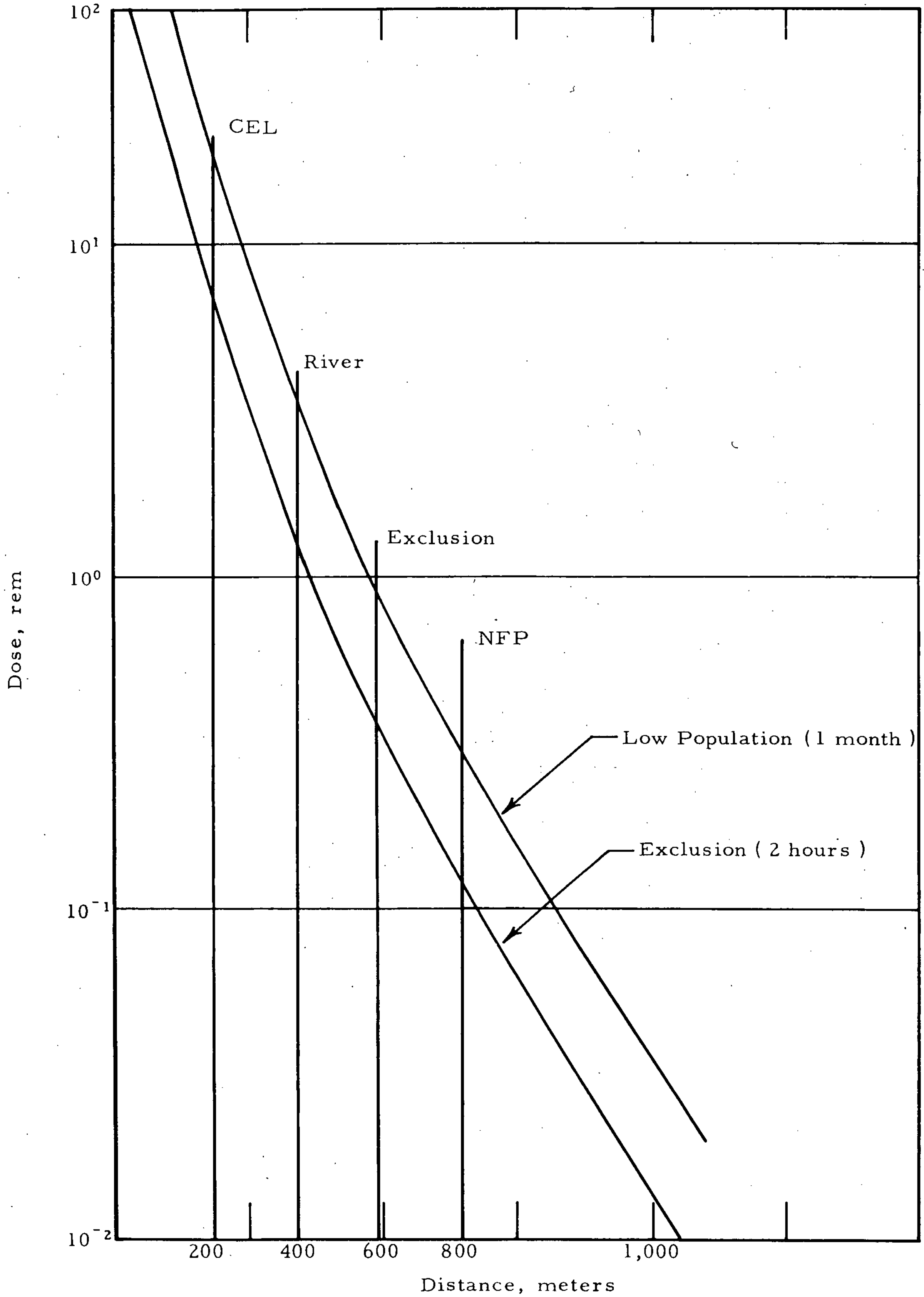


Figure 7.7. Thyroid Dose Due to Iodine Inhalation, Hypothetical Accident

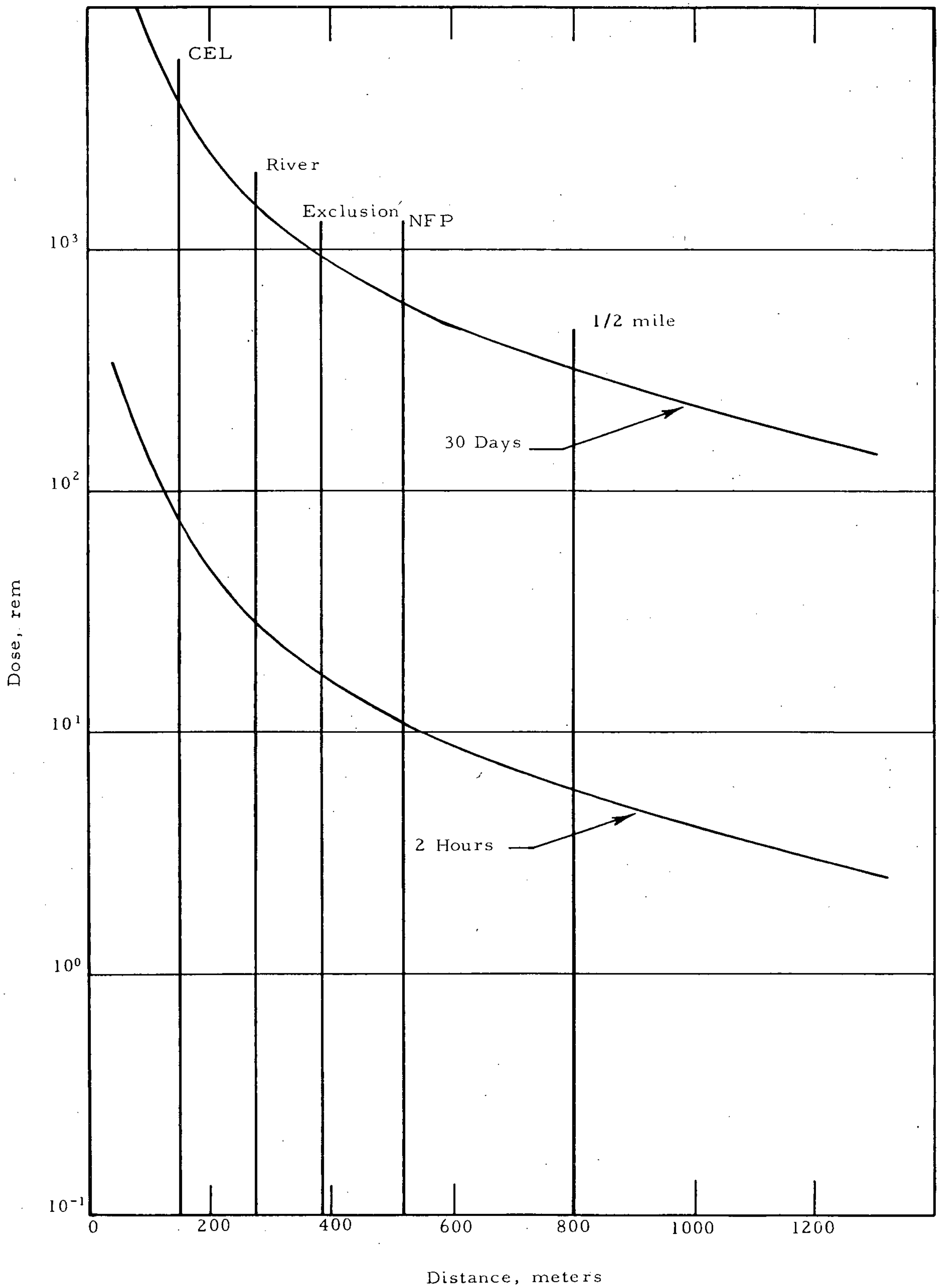


Figure 7.8. Bone Dose Due to Strontium Inhalation, Hypothetical Accident

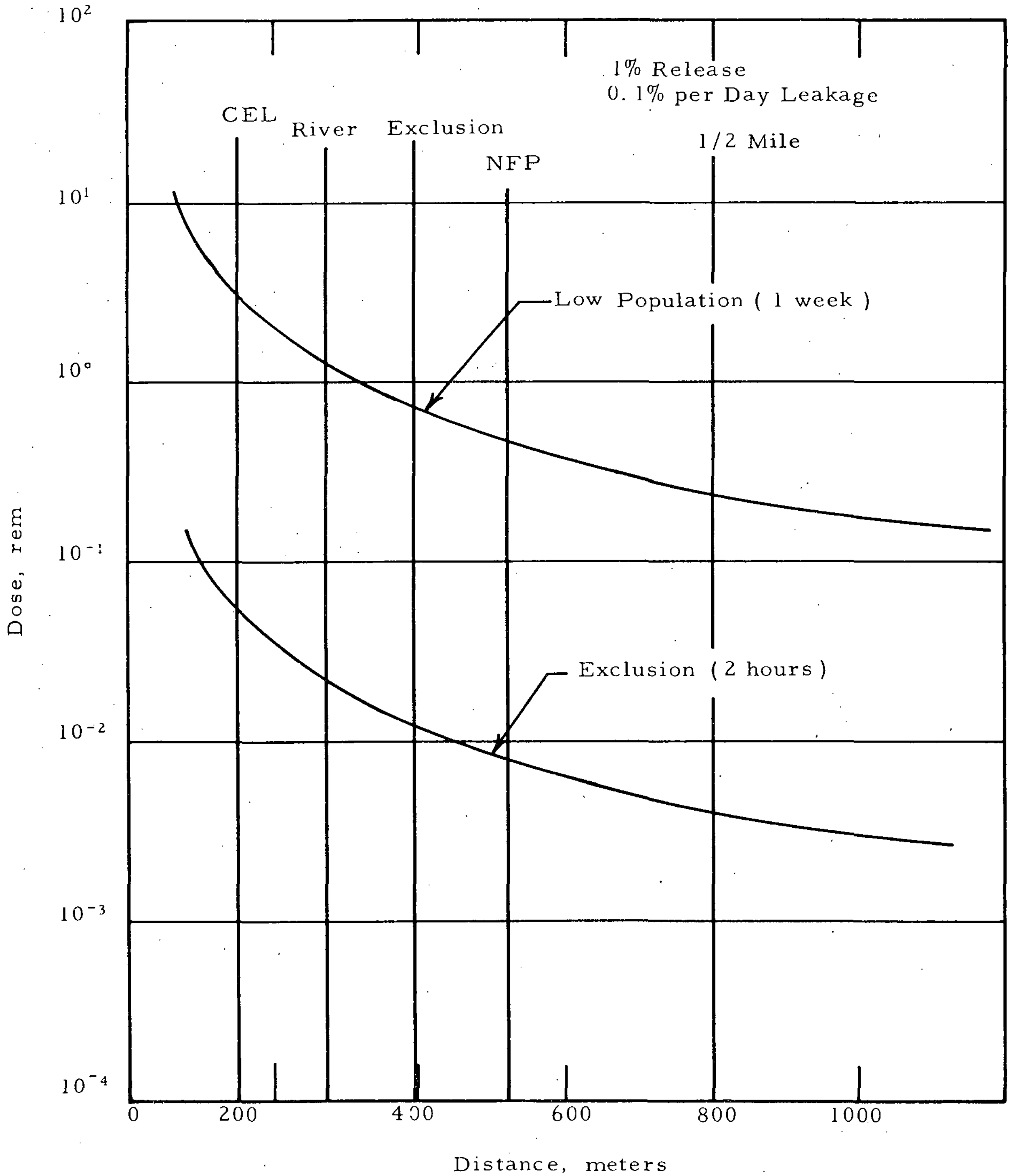
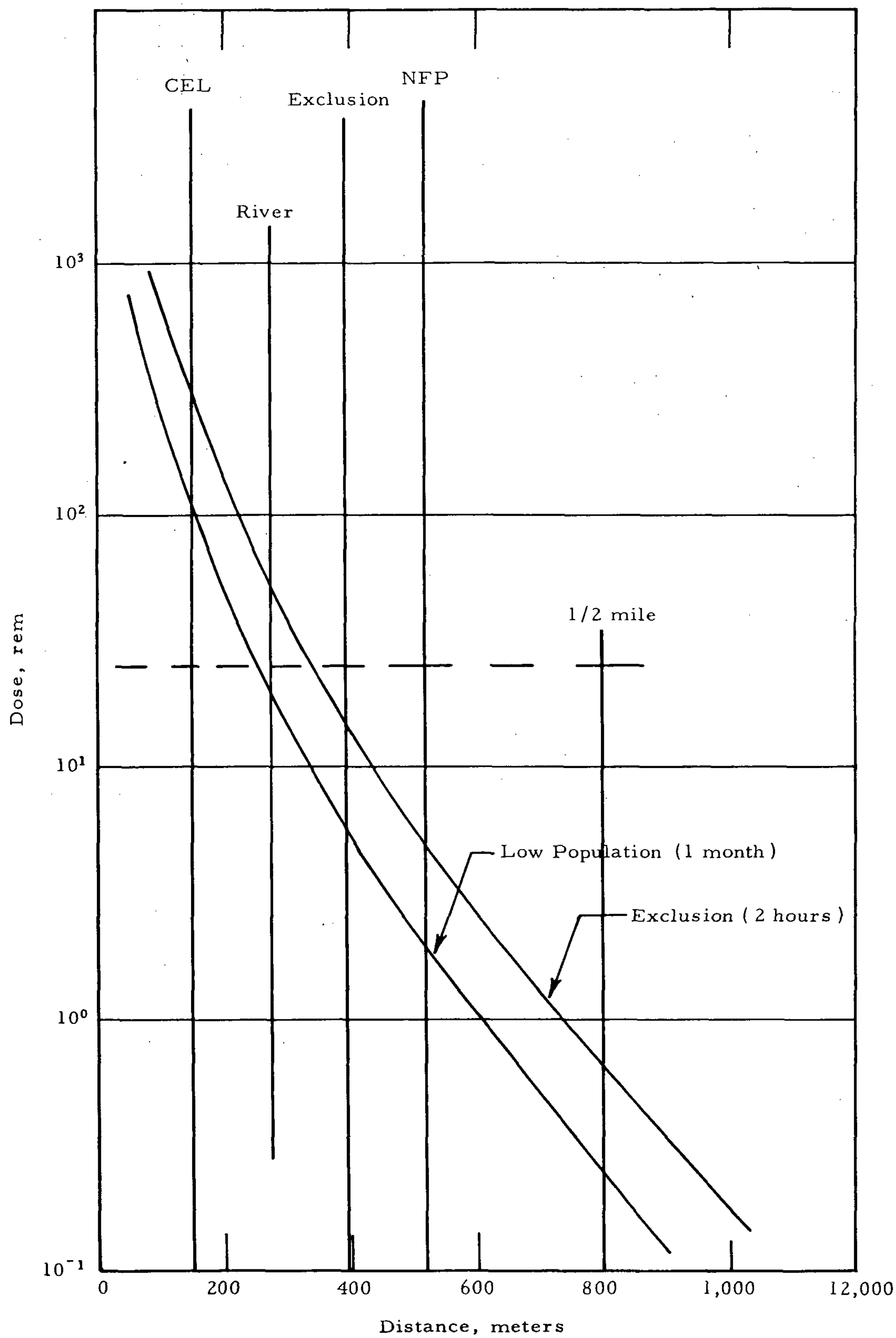


Figure 7. 9. Body Dose Due to Direct Irradiation, Hypothetical Accident



APPENDIX A

U. S. DEPARTMENT OF COMMERCE
WEATHER BUREAU

LOCAL CLIMATOLOGICAL DATA

WITH COMPARATIVE DATA

1961

LYNCHBURG, VIRGINIA

NARRATIVE CLIMATOLOGICAL SUMMARY

Lynchburg is situated in the valley of the James River, and on the eastern edge of the Blue Ridge Mountains. The terrain is definitely hilly, with sheltered valleys which are visited by early autumn and late spring frosts. However, the average frost free season is 204 days. The average climate is usually a pleasant one, being neither too hot in summer, which has an average temperature of 75.4°, nor too cold in winter, which has an average temperature of 38.9°. Rainfall is fairly evenly distributed throughout the year, but there is a distinct summertime maximum of rainfall, occasioned by afternoon thunderstorms, of which there are an average of 35 per year. Most of the wintertime precipitation falls as rain with the approach of low pressure troughs, although there are occasional heavy snows, which total an average of 14.3 inches per year.

Spring really makes itself felt in March, which shows an eight degree advance over February's temperature, and autumn comes in rapidly in October, which shows an eleven degree drop below September. The approaching autumn season is especially felt when Lynchburg is under the influence of a high pressure system in New England with an accompanying ridge of high pressure southward just east of the Appalachians. This brings two to three days of cloudy cool weather, with high humidity, and even drizzle, with its attendant discomfort. In mid-winter however, when the pressure systems push even further south, after the passage of a cold front, dry invigorating air, with clear skies, is the rule in Lynchburg. There are occasional snow showers under these conditions, but the mountains to the immediate west act as a barrier and sheltering agent from high winds.

The mountains also act as a barrier from extremely cold weather, temperatures have fallen below zero only on 20 days in 76 years, and hundred degree heat is almost as rare, although this mark has been exceeded in the months of May through September. Ninety degree weather is reached only on an average of 32 times per year.

Great variation in temperature is quite frequently noted during clear, still nights in the winter months. On some such nights, differences of as much as 10 to 15° occur between the low valleys and the higher terrain.

LATITUDE 37° 20' N
LONGITUDE 79° 12' W
ELEVATION (ground) 947 Feet

METEOROLOGICAL DATA FOR THE CURRENT YEAR

LYNCHBURG, VIRGINIA
PRESTON GLENN AIRPORT
1961

Month	Temperature								Precipitation						Relative humidity				Wind							Number of days										
	Averages			Extremes								Snow, Sleet				EST	EST	EST			Fastest mile					Sunrise to sunset							Temperatures			
																																	Maximum		Minimum	
	Daily maximum	Daily minimum	Monthly	Highest	Date	Lowest	Date	Degree days	Total	Greatest in 24 hrs.	Date	Total	Greatest in 24 hrs.	Date		7:00 a.m.	1:00 p.m.	7:00 p.m.	Average hourly speed	Prevailing direction	Speed	Direction	Date	Percent of possible sunshine	Average sky cover sunrise to sunset	Clear	Partly cloudy	Cloudy	Precipitation .01 inch or more	Snow, Sleet 1.0 inch or more	Thunderstorms	Heavy fog	90° and above	32° and below	32° and below	0° and below
JAN	41.9	22.9	32.4	58	11+	7	22	1005	1.06	.30	19	8.8	4.4	19		67	50	56	8.4		34	W	24	71	4.4	16	6	9	7	2	0	0	0	7	25	0
FEB	49.4	31.4	40.4	73	24	11	3	681	5.23	1.30	18	8.1	4.3	7- 8		79	61	69	8.6		50	S	25	43	7.3	7	3	18	11	3	1	3	0	2	14	0
MAR	59.3	39.3	49.3	79	7+	26	18	489	4.15	.98	21-22	T	T	21+		75	52	58	9.7		32	S	6	55	6.7	7	7	17	14	0	0	2	0	0	6	0
APR	60.2	41.4	50.8	84	25+	32	5	439	2.72	1.34	9-10	T	T	3		71	47	53	10.5		43	S	16	57	6.3	9	8	13	11	0	2	0	0	0	1	0
MAY	72.1	50.3	61.2	84	8	39	28	137	3.06	.93	1	.0	.0			81	51	58	8.4		30	S	25	67	5.6	10	10	11	1	0	6	3	0	0	0	0
JUN	80.9	60.4	70.7	92	2	47	16	17	6.12	3.04	20-21	.0	.0			90	58	71	7.2		29	S	9	63	6.7	4	12	14	1	0	7	4	1	0	0	0
JUL	85.4	65.4	75.4	93	31	53	10	0	2.28	.74	15-16	.0	.0			86	57	69	6.6		26	SW	18	71	5.6	6	18	7	12	0	12	3	7	0	0	0
AUG	83.8	65.9	74.9	90	29+	57	15	2	4.12	1.18	6- 7	.0	.0			92	61	74	6.6		30	SW	7	60	6.5	5	12	14	12	0	7	6	2	0	0	0
SEP	81.2	61.2	71.2	92	3	46	16	37	4.05	1.89	20	.0	.0			90	57	75	7.2		26	NE	7	74	3.8	18	5	7	9	0	5	6	2	0	0	0
OCT	70.5	46.3	58.4	81	13	33	28	208	4.55	3.30	20-21	.0	.0			88	50	68	7.1		34	NE	21	79	3.1	21	4	6	5	0	1	3	0	0	0	0
NOV	57.8	41.3	49.6	80	5	24	22	476	2.46	1.07	6- 7	.6	.6	19		79	57	68	7.9		29	N	28	55	5.7	11	6	13	11	0	1	5	0	0	9	0
DEC	45.3	29.7	37.5	60	3	14	30	846	4.89	1.50	9-10	2.1	1.2	23-24		74	58	68	8.0		34	SW	13	45	6.5	9	4	18	12	0	0	4	0	3	20	0
YEAR	65.7	46.3	56.0	93	JUL 31	7	JAN 22	4337	44.69	3.30	OCT 20-21	19.6	4.4	JAN 19		81	55	66	8.0		50	S	FEB 25	62	5.7	123	95	147	132	5	42	39	12	12	75	0

NORMALS, MEANS, AND EXTREMES

Month	Temperature								Precipitation												Relative humidity				Wind							Mean number of days									
	Normal			Extremes												Snow, Sleet					EST	EST	EST	EST			Fastest mile					Sunrise to sunset							Temperatures			
																																							Max.		Min.	
	Daily maximum	Daily minimum	Monthly	Record highest	Year	Record lowest	Year	Normal degree days	Normal total	Maximum monthly	Year	Minimum monthly	Year	Maximum in 24 hrs.	Year	Mean total	Maximum monthly	Year	Maximum in 24 hrs.	Year	1:00 a.m.	7:00 a.m.	1:00 p.m.	7:00 p.m.	Mean hourly speed	Prevailing direction	Speed	Direction	Year	Pct. of possible sunshine	Mean sky cover sunrise to sunset	Clear	Partly cloudy	Cloudy	Precipitation .01 inch or more	Snow, Sleet 1.0 inch or more	Thunderstorms	Heavy fog	90° and above	32° and below	32° and below	0° and below
(a)	(b)	(b)	(b)	17		17		(b)	(b)	17		17		17		17	17		17		9	17	17	17	17	9	17	17		17	17	17	17	17	17	17	17	17	17	17	17	17
J	46.6	28.7	37.7	76	1952	6	1959	846	3.43	4.45	1949	.76	1956	2.13	1948+	4.2	12.1	1948	14.5	1948	75	75	57	65	8.9	SW	38	S	1959+	49	6.4	8	8	15	12	1	*	5	0	3	21	0
F	48.9	29.5	39.2	79	1948	2	1958	722	2.48	5.23	1961	.64	1947	1.66	1955	3.9	14.0	1960	12.8	1960	69	74	53	59	9.2	SW	50	S	1961	55	6.2	8	7	13	10	1	*	4	0	2	17	0
M	57.1	35.7	46.4	87	1945	11	1960	584	3.57	6.69	1948	1.11	1957	2.43	1949	3.7	24.9	1960	12.1	1960	70	73	50	57	9.5	SW	43	S	1950	57	6.1	8	10	13	12	1	1	2	0	1	12	0
A	66.9	44.0	55.5	91	1960+	27	1950	289	3.13	4.66	1951	2.22	1954	2.45	1945	.2	2.8	1957	2.8	1957	71	78	46	55	9.6	SW	43	S	1961	60	6.1	8	9	13	11	*	3	3	*	0	2	0
M	76.4	53.5	65.0	92	1956+	32	1947	82	3.09	6.02	1960	1.36	1957	3.47	1960	.0	.0		.0		80	78	52	63	8.2	SW	56	N	1958	63	6.1	8	10	13	12	0	7	4	1	0	*	0
J	83.4	62.2	72.8	100	1945	40	1945	5	3.88	8.50	1949	1.65	1956	3.23	1949	.0	.0		.0		84	81	54	66	7.3	SW	56	SW	1951	67	5.7	8	12	10	11	0	8	3	6	0	0	0
J	86.1	65.9	76.0	103	1954	53	1961+	0	4.15	7.92	1950	2.05	1957	2.65	1956	.0	.0		.0		87	84	56	70	7.0	SW	43	NW	1954+	63	5.9	8	12	11	12	0	10	3	10	0	0	0
A	84.1	64.2	74.2	98	1953+	48	1946	0	4.35	11.36	1952	1.97	1956	3.49	1950	.0	.0		.0		89	88	57	72	6.5	N	46	NE	1952	61	5.8	8	12	11	11	0	8	4	7	0	0	0
S	79.2	58.4	68.8	101	1954	37	1956	49	3.10	7.48	1957	1.47	1954	3.10	1957	.0	.0		.0		87	88	57	73	7.3	N	40	NE	1956	61	5.5	10	9	11	9	0	4	4	2	0	0	0
O	69.1	46.4	57.8	93	1951	23	1952	236	2.87	5.89	1954	.90	1945	4.98	1954	T	T	1957+	T	1957+	81	85	52	68	7.7	N	41	N	1954	61	4.9	13	7	11	8	0	1	3	*	0	1	0
N	57.3	37.2	47.3	83	1950	11	1949	531	2.37	5.10	1951	.90	1960	2.65	1951	.5	2.9	1953	2.9	1953	74	77	52	63	8.2	SW	43	NW	1949	56	5.5	11	7	12	9	*	1	4	0	*	10	0
D	47.8	30.0	38.9	77	1956	8	1957+	809	3.15	6.74	1948	.61	1955	3.03	1948	2.4	8.4	1945	6.2	1950+	73	75	55	64	8.1	SW	45	SE	1950	53	5.9	11	6	14	9	1	*	4	0	3	19	0
YR	66.9	46.3	56.6	103	JUL 1954	2	FEB 1958	4153	39.57	11.36	AUG 1952	.61	DEC 1955	4.98	OCT 1954	14.9	24.9	MAR 1960	14.5	JAN 1948	78	80	53	65	8.1	SW	56	SW	JUN 1951+	59	5.8	107	110	148	125	5	44	42	28	8	81	0

The normals published in this issue are derived from 1921-1950 data. Effective with January 1962 data, new climatological standard normals based on the period 1931-1960 will be used. These new normals are available in "Decennial Census of U. S. Climate; Monthly Normals of Temperature, Precipitation and Degree Days Virginia, Climatography of the United States No. 81 - 38."

Means and extremes in the above table are from existing or comparable location (s). Annual extremes have been exceeded at prior locations as follows: Highest temperature 106 in July 1936; lowest temperature -7 in January 1912; maximum monthly precipitation 14.87 in August 1928; minimum monthly precipitation .03 in November 1890; maximum precipitation in 24 hours 7.59 in August 1928; maximum snowfall in 24 hours 16.4 in January 1922; fastest mile wind 62 W in April 1917+.

REFERENCE NOTES

(a) Length of record, years.
(b) Normal values are based on the period 1921-1950, and are means adjusted to represent observations taken at the present standard location.
* Less than one half.
\- No record.
† Airport data.
‡ City Office data.
\+ Also on earlier dates, months, or years.
T Trace, an amount too small to measure.
Wind direction to 8 compass points only.

Unless otherwise indicated, dimensional units used in this bulletin are: temperature in degrees F; precipitation, including snowfall, in inches; wind movement in miles per hour; and relative humidity in percent. Monthly degree day totals are the sums of the negative departures of average daily temperatures from 65°F.

Sky cover is expressed in a range of 0 for no clouds or obscuring phenomena to 10 for complete sky cover. The number of clear days is based on average cloudiness 0-3; partly cloudy days 4-7; and cloudy days 8-10 tenths.

Sleet and hail were included in snowfall tables from July 1948 through December 1955, snow and sleet only prior to and after that period.

Below zero temperatures are preceded by a minus sign.

Heavy fog in the Normals, Means and Extremes table includes fog referred to at various times in the past as "dense" and "thick". The upper visibility limit for heavy fog is 1/4 mile.

Mean values at the bottom of the Average Temperature and Total Precipitation tables are long-term means based on the entire period of record beginning in 1875% Values have not been corrected for changes in instrument location listed in the Station Location table.

Data for earlier years may be obtained by contacting the Weather Bureau office in the city for which this summary was issued.

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% Average Temperature Table, 1875. Total Precipitation Table, 1872.

AVERAGE TEMPERATURE

Year	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	An'l
1906	42.0	38.4	41.2	58.6	65.2	74.6	75.6	77.5	72.4	55.4	48.6	40.6	57.5
1907	43.1	35.8	52.6	49.8	62.2	67.6	77.1	74.0	70.8	53.9	45.2	40.0	56.0
1908	36.5	34.1	51.6	58.6	65.6	71.6	77.0	73.6	66.6	57.2	49.4	40.0	56.8
1909	41.2	45.2	45.0	57.2	64.2	75.4	74.2	73.6	66.7	53.9	52.5	34.2	56.9
1910	37.2	36.8	55.0	57.7	62.7	69.6	77.2	73.3	71.0	60.4	42.4	33.7	56.4
1911	42.0	41.2	45.0	51.7	69.2	74.8	77.4	76.8	72.5	59.2	43.5	43.2	58.0
1912	29.0	34.4	45.0	59.0	66.2	71.6	77.2	75.2	72.0	59.4	47.4	41.4	56.5
1913	45.2	40.2	50.2	57.4	66.0	73.3	78.6	75.1	67.5	59.7	49.9	43.5	58.9
1914	42.0	35.2	42.6	56.7	68.4	76.6	76.0	76.3	66.7	60.9	46.6	35.0	56.9
1915	38.4	42.2	41.2	59.8	65.4	71.6	76.2	74.2	70.6	60.8	48.1	38.0	57.2
1916	43.2	38.8	44.6	54.7	69.0	71.0	76.4	75.3	66.8	57.7	48.4	38.6	57.0
1917	39.4	36.9	46.6	57.4	60.7	72.8	76.4	75.4	64.9	53.0	45.2	29.0	54.8
1918	27.0	41.2	52.3	53.3	70.0	71.2	72.8	77.6	64.3	61.0	46.8	42.9	56.7
1919	41.0	39.8	49.4	55.6	65.0	73.4	77.2	73.9	69.8	65.4	48.2	36.1	57.9
1920	34.4	35.4	47.4	54.3	61.8	71.9	74.8	73.7	69.5	60.8	46.4	40.2	55.9
1921	39.0	41.4	57.1	59.1	63.2	74.3	78.8	74.5	75.5	57.6	50.6	42.3	59.4
1922	35.1	42.0	48.8	57.8	67.2	74.8	76.4	72.8	70.9	60.6	48.4	42.5	58.1
1923	39.8	37.8	47.9	55.8	63.8	75.6	76.5	74.8	70.0	56.6	46.1	47.4	57.7
1924	36.2	37.8	45.5	54.1	61.4	72.1	74.0	76.0	64.0	58.4	47.7	40.2	55.6
1925	37.4	46.6	49.2	59.6	61.3	77.5	78.2	74.0	74.6	52.8	45.6	39.4	58.0
1926	38.0	42.8	42.5	54.4	66.4	71.1	77.8	77.8	71.8	59.2	44.0	37.9	57.0
1927	38.4	47.0	51.2	55.4	66.2	70.4	75.6	70.8	71.4	60.4	51.6	40.1	58.2
1928	39.2	40.6	47.5	53.2	64.4	71.8	77.8	77.4	64.8	60.1	49.8	40.6	57.3
1929	37.2	38.2	52.8	59.6	65.3	72.6	76.0	73.2	70.0	55.8	48.0	40.9	57.5
1930	39.2	45.8	45.4	56.3	68.4	72.7	80.5	75.4	74.5	55.6	46.0	36.5	58.0
1931	40.5	42.0	42.2	55.0	65.0	74.5	80.2	74.6	73.8	61.0	54.9	47.3	59.2
1932	48.2	47.8	43.4	56.2	66.0	74.0	79.2	77.8	69.8	58.8	45.1	41.2	58.9
1933	45.4	40.5	46.4	56.4	69.7	76.0	76.5	75.6	75.5	57.4	46.4	43.6	59.1
1934	41.2	29.8	43.0	57.0	67.2	76.8	81.8	76.4	71.8	57.8	49.6	39.0	57.6
1935	36.2	40.2	52.2	54.3	64.2	73.5	78.3	76.8	68.5	59.0	51.6	32.4	57.3
1936	32.4	34.6	51.4	54.0	68.9	73.6	79.8	77.8	71.6	60.8	46.2	40.9	57.7
1937	45.3	39.6	45.2	56.0	66.4	75.9	75.8	76.4	66.3	55.3	46.0	39.2	57.3
1938	38.1	45.2	53.6	58.6	65.8	71.3	77.3	78.3	68.9	59.5	50.6	40.0	58.9
1939	41.9	44.7	50.0	55.2	68.2	77.0	75.6	76.4	72.0	60.2	47.2	42.4	59.2
1940	28.2	40.4	44.6	54.4	66.4	75.4	76.2	73.6	67.4	57.9	49.3	44.0	56.5
1941	37.6	36.2	42.8	62.2	68.4	73.6	78.6	77.6	72.8	65.3	50.3	44.2	59.1
1942	37.6	36.9	49.6	60.6	69.4	76.0	79.5	75.0	70.6	60.4	49.9	37.0	58.5
1943	41.1	41.4	46.4	54.4	68.6	79.4	77.9	78.2	67.4	57.8	48.0	39.9	58.4
1944	40.8	41.4	45.2	56.0	71.6	76.4	77.2	74.6	69.2	56.4	46.4	33.5	57.4
1945	34.0	40.0	56.6	59.2	62.1	72.8	75.8	73.8	71.4	57.0	48.0	32.6	56.9
1946	37.2	40.8	53.0	57.3	63.2	71.6	73.8	70.9	68.1	59.2	51.8	42.6	57.5
1947	41.7	31.0	38.4	57.4	65.4	70.8	73.3	77.1	68.4	61.9	44.2	38.4	55.7
1948	31.4	39.2	49.6	58.0	64.9	72.8	76.7	74.2	67.2	55.0	50.8	40.9	56.7
1949	45.0	45.4	46.9	54.2	65.4	73.7	77.8	74.2	65.2	61.7	47.6	40.6	58.2
1950	47.8	39.6	42.8	53.9	64.9	72.6	73.3	72.8	66.3	60.5	44.6	34.6	56.2
1951	37.8	39.3	45.4	55.3	64.4	72.9	76.8	75.5	68.9	60.7	42.0	39.2	56.5
1952	42.1	41.4	45.3	56.7	65.1	76.9	77.7	73.4	66.5	54.2	48.2	38.8	57.2
1953	41.2	42.2	47.2	55.5	70.3	73.8	77.6	75.7	68.5	61.1	48.0	39.3	58.4
1954	37.2	43.8	45.4	60.4	60.8	73.0	76.5	75.2	72.6	59.9	43.8	37.1	57.1
1955	35.3	39.3	48.4	59.6	67.0	67.7	78.5	76.0	68.4	57.5	45.0	34.1	56.4
1956	36.4	41.7	45.3	54.1	64.5	72.1	76.1	74.8	66.3	58.4	45.6	47.7	56.9
1957	35.5	43.0	46.4	59.8	66.4	73.2	75.8	73.6	68.8	53.3	48.6	41.3	57.1
1958	33.5	31.6	41.2	55.9	63.3	70.9	77.3	73.7	67.8	56.8	50.2	33.7	54.6
1959	36.0	40.4	45.5	57.9	68.0	73.5	75.8	77.7	69.4	59.1	45.5	41.5	57.5
1960	37.8	36.8	34.7	60.1	63.0	72.4	75.6	76.6	68.9	58.3	48.5	33.0	55.5
1961	32.4	40.4	49.3	50.8	61.2	70.7	75.4	74.9	71.2	58.4	49.6	37.5	56.0
RECORD MEAN TEMP	37.9	39.5	46.9	56.3	65.9	73.5	77.2	75.4	69.4	58.3	47.5	39.3	57.2
MAX	46.7	48.9	57.0	67.4	77.0	83.9	87.3	85.2	79.7	69.3	57.6	48.1	67.3
MIN	29.0	30.1	36.7	45.1	54.7	63.0	67.0	65.6	59.1	47.3	37.4	30.4	47.1

TOTAL PRECIPITATION

Year	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	Annual
1906	4.08	1.14	4.93	2.47	3.18	3.21	4.10	4.84	2.34	7.15	2.27	2.92	42.63
1907	0.98	0.78	2.50	3.43	3.30	7.00	2.63	3.21	10.53	0.60	4.47	4.45	43.88
1908	4.37	2.85	2.51	2.96	3.86	4.66	4.49	6.55	2.22	3.52	1.18	3.54	42.71
1909	1.36	3.09	2.37	2.38	4.58	4.95	1.85	3.20	2.65	1.46	0.52	2.79	31.20
1910	3.44	3.58	0.36	5.87	2.78	6.59	4.19	2.32	3.19	5.09	0.87	2.00	40.28
1911	4.19	1.23	2.12	4.21	0.66	2.82	5.53	5.73	0.85	3.87	3.22	4.65	39.08
1912	1.85	3.36	8.54	1.89	4.95	2.35	3.64	1.28	6.98	0.96	2.72	2.36	40.88
1913	1.91	1.99	5.50	3.60	4.76	2.88	1.53	2.40	2.44	3.30	3.32	2.56	36.19
1914	3.00	2.77	2.24	1.70	0.59	2.21	4.53	2.60	0.67	4.42	2.31	4.68	31.72
1915	3.86	2.88	1.14	0.87	1.99	4.16	3.05	5.45	3.26	3.21	1.66	2.37	33.90
1916	0.99	2.92	1.32	1.94	5.13	6.28	9.76	2.69	2.55	2.22	1.52	2.32	39.64
1917	2.69	1.66	4.97	3.10	2.21	5.17	2.97	3.53	1.96	2.24	0.25	1.70	32.45
1918	4.42	0.53	2.41	4.97	2.15	2.91	3.75	2.91	2.61	0.95	1.26	2.59	31.46
1919	4.11	2.23	3.02	2.18	3.64	7.61	5.21	3.03	0.47	2.65	2.48	1.96	38.59
1920	1.64	4.03	2.82	3.53	0.79	5.12	4.82	6.76	4.51	0.10	7.14	2.34	43.60
1921	2.60	2.60	1.75	2.76	6.15	1.85	3.56	0.83	1.71	2.45	1.65	1.01	28.92
1922	3.90	3.32	7.50	1.53	4.37	3.37	2.25	1.18	1.42	4.19	0.19	3.42	36.64
1923	2.25	2.38	5.91	2.71	1.66	2.12	2.52	3.44	2.84	1.50	1.98	2.72	32.03
1924	3.37	1.87	2.77	3.35	5.16	3.58	3.74	4.80	4.69	2.48	1.45	3.34	40.60
1925	3.40	1.04	1.06	2.61	1.34	2.77	3.56	0.56	0.43	4.08	2.68	2.86	26.39
1926	3.89	3.59	2.52	1.82	0.44	1.13	2.71	1.67	1.59	3.16	3.43	4.99	30.94
1927	1.09	3.14	1.29	3.99	1.42	1.87	5.90	4.27	0.31	6.51	1.91	3.88	35.58
1928	2.53	2.40	1.93	4.55	0.66	2.98	3.55	14.87	6.01	0.65	0.79	1.02	41.94
1929	2.38	2.75	3.22	5.76	3.06	5.00	2.21	3.99	2.20	5.21	2.82	1.87	40.47
1930	2.77	0.95	2.26	1.45	1.75	2.57	0.90	0.42	0.90	0.62	2.44	2.80	19.83
1931	1.48	1.81	3.75	3.11	5.93	2.43	5.37	5.16	0.73	0.67	0.45	2.74	33.63
1932	4.29	1.25	4.80	1.01	1.38	2.54	0.75	1.70	2.22	8.86	4.63	3.85	37.28
1933	2.73	2.17	2.72	4.41	4.26	2.61	4.30	3.50	0.35	1.45	1.25	3.86	33.61
1934	1.92	3.48	5.75	3.52	4.09	4.21	3.04	4.69	7.37	1.69	8.11	2.99	48.86
1935	5.27	3.18	5.70	4.78	3.26	8.53	6.76	3.80	7.53	2.18	5.05	2.60	58.64
1936	9.46	3.99	7.48	4.54	1.23	4.64	3.07	5.95	4.64	3.93	0.66	5.35	54.94
1937	8.49	2.82	1.63	5.07	2.93	6.24	5.97	4.89	2.61	7.91	2.14	1.03	51.73
1938	2.99	1.16	3.16	1.60	4.13	8.25	5.73	3.24	1.87	0.77	2.70	4.18	39.78
1939	2.92	4.99	3.21	2.80	1.33	3.88	6.05	7.66	1.21	2.73	1.89	2.11	40.78
1940	2.86	3.02	2.07	4.26	2.73	2.33	5.45	9.72	1.14	0.91	4.40	3.02	41.91
1941	2.38	0.75	1.47	3.77	0.52	4.73	8.70	1.12	0.09	1.09	1.28	4.34	30.24
1942	2.77	2.08	3.25	0.79	6.02	6.14	2.02	10.36	5.30	7.02	1.51	3.93	51.19
1943	4.18	2.04	4.51	2.61	4.47	4.48	2.51	2.27	1.57	1.30	0.75	2.93	33.62
1944	2.39	5.03	5.77	2.90	1.98	2.23	4.21	1.89	10.87	2.01	2.03	1.96	43.27
1945	2.37	2.76	1.96	3.98	3.16	2.72	6.26	3.01	6.90	0.90	3.33	3.99	41.34
1946	2.66	2.54	2.80	2.36	4.08	2.51	5.27	3.34	2.66	1.41	1.75	3.38	34.76
1947	4.20	0.64	3.06	2.93	2.99	4.15	4.04	3.48	2.96	3.13	4.13	0.89	36.60
1948	4.30	2.69	6.69	2.84	4.13	5.11	2.22	8.26	2.07	2.61	4.86	6.74	52.52
1949	4.45	2.50	3.49	3.91	3.84	8.50	4.07	4.03	2.68	2.83	1.20	2.31	43.81
1950	2.69	2.12	3.63	2.23	4.36	2.88	7.92	6.45	6.27	1.87	1.61	4.37	46.40
1951	1.79	2.31	3.49	4.66	2.08	7.26	2.58	2.46	1.56	1.66	5.10	4.82	39.77
1952	4.37	2.69	5.04	3.60	3.24	3.32	5.42	11.36	1.79	1.14	4.85	3.12	49.95
1953	3.41	2.89	4.82	3.26	3.04	4.78	2.46	2.77	3.11	3.97	1.00	4.86	40.37
1954	3.84	1.54	3.14	2.22	3.40	2.72	2.32	2.00	1.47	5.89	3.59	3.04	35.17
1955	1.33	4.36	4.37	2.79	3.66	4.92	3.68	6.17	2.15	0.95	1.51	0.61	36.50
1956	0.76	3.64	2.52	3.22	2.37	1.65	6.40	1.97	3.67	3.20	2.26	2.88	34.43
1957	3.99	4.31	1.11	3.77	1.36	5.30	2.05	2.40	7.48	1.93	4.36	3.09	41.15
1958	3.32	2.80	3.57	3.06	5.45	2.40	4.72	4.45	1.49	1.72	1.38	3.58	37.94
1959	2.01	1.57	3.40	4.26	1.57	1.85	4.58	4.76	4.91	3.61	2.63	2.49	37.62
1960	2.39	4.64	3.25	2.72	6.02	2.15	2.33	3.22	4.26	2.26	0.90	1.77	35.91
1961	1.06	5.23	4.15	2.72	3.06	6.12	2.28	4.12	4.05	4.55	2.46	4.89	44.69
RECORD MEAN	3.35	3.01	3.60	3.14	3.44	3.95	4.06	4.18	3.35	3.03	2.54	3.16	40.81

MONTHLY AND SEASONAL DEGREE DAYS

Season	July	Aug.	Sept.	Oct.	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May	June	Total
1907-1908	0	0	19	354	597	770	895	893	423	219	131	12	4313
1908-1909	0	18	39	243	467	774	738	556	619	259	113	0	3826
1909-1910	0	0	55	346	375	954	863	790	332	229	142	39	4125
1910-1911	3	0	18	183	674	969	717	663	616	406	66	0	4315
1911-1912	0	3	10	192	644	676	1117	893	722	198	72	23	4550
1912-1913	0	0	29	201	526	732	615	694	464	248	56	35	3600
1913-1914	0	1	59	193	497	667	711	834	698	279	58	3	4000
1914-1915	2	1	74	163	552	932	826	637	736	228	75	32	4258
1915-1916	0	7	52	174	504	837	612	759	631	319	26	1	3922
1916-1917	0	1	55	266	501	820	791	786	568	261	192	7	4248
1917-1918	0	0	89	375	592	1123	1179	668	401	335	35	9	4806
1918-1919	2	2	85	159	543	685	748	705	486	285	85	5	3790
1919-1920	0	1	28	123	510	894	948	858	546	330	138	14	4390
1920-1921	0	0	13	161	557	769	807	659	281	198	127	2	3574
1921-1922	0	2	0	237	438	705	925	644	504	238	34	0	3727
1922-1923	1	1	14	176	498	694	783	764	533	289	9	11	3773
1923-1924	0	6	35	262	567	645	892	790	606	230	141	25	4199
1924-1925	0	0	77	220	519	768	855	517	488	236	185	0	3865
1925-1926	0	2	12	393	584	795	835	625	696	336	54	28	4360
1926-1927	1	1	22	221	632	838	826	503	448	328	83	33	3936
1927-1928	1	14	28	273	398	777	796	706	541	363	98	9	4004
1928-1929	0	0	111	203	458	757	864	751	403	224	97	24	3892
1929-1930	0	0	45	329	518	747	800	543	604	292	30	5	3913
1930-1931	0	3	1	307	571	886	758	645	708	301	91	6	4276
1931-1932	0	11	19	175	308	551	525	497	669	270	73	1	3099
1932-1933	0	0	30	201	594	736	607	688	580	264	35	14	3749
1933-1934	1	1	0	251	558	463	742	986	686	252	58	0	3998
1934-1935	0	3	2	235	458	805	892	694	399	336	89	0	3913
1935-1936	0	0	17	213	408	1008	1012	881	425	356	28	10	4358
1936-1937	0	0	27	169	564	742	611	707	613	286	63	0	3782
1937-1938	0	0	56	315	570	800	833	556	374	240	73	7	3824
1938-1939	0	0	36	191	438	776	714	567	479	317	92	1	3611
1939-1940	4	0	8	216	536	702	1140	712	635	327	66	0	4346
1940-1941	3	1	67	236	472	650	848	806	690	165	72	11	4021
1941-1942	0	0	10	126	441	647	851	789	478	187	18	0	3547
1942-1943	0	5	60	156	459	865	741	663	577	339	57	0	3922
1943-1944	1	0	66	241	508	778	759	681	619	279	24	1	3957
1944-1945	0	3	37	281	559	980	962	696	302	215	162	48	4244
1945-1946	0	0	19	251	511	1004	862	677	374	256	107	12	4073
1946-1947	0	14	32	196	400	694	720	955	825	244	108	13	4201
1947-1948	0	0	99	130	628	828	1042	746	492	236	58	3	4262
1948-1949	0	0	52	314	428	745	619	550	566	324	75	2	3675
1949-1950	0	6	85	142	523	753	529	705	682	344	71	0	3840
1950-1951	0	0	75	154	613	936	835	712	600	305	82	1	4313
1951-1952	0	0	33	179	682	791	701	678	602	278	65	0	4009
1952-1953	0	4	42	338	498	803	727	635	545	288	12	3	3895
1953-1954	0	0	40	152	504	791	857	588	601	182	201	10	3926
1954-1955	0	0	9	256	630	857	913	711	504	176	50	25	4131
1955-1956	0	0	21	243	595	950	883	667	606	356	108	37	4466
1956-1957	0	5	92	216	575	527	912	610	569	238	69	17	3830
1957-1958	0	6	65	355	488	728	969	929	730	289	100	7	4666
1958-1959	0	0	34	264	444	964	891	683	595	230	53	11	4169
1959-1960	0	0	37	250	577	722	837	809	933	206	146	1	4518
1960-1961	0	0	29	226	492	989	1005	681	489	439	137	17	4504
1961-1962	0	2	37	208	476	846							

MONTHLY AND SEASONAL SNOWFALL

LYNCHBURG, VIRGINIA
PRESTON GLENN AIRPORT
1961

Season	July	Aug.	Sept.	Oct.	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May	June	Total
1905-1906	0	0	0	0	T	4.3	1.1	4.8	5.2	0	0	0	15.4
1906-1907	0	0	0	0	T	0.4	1.5	5.5	1.8	3.9	0	0	13.1
1907-1908	0	0	0	0	T	T	0.8	12.1	5.3	0	0	0	18.2
1908-1909	0	0	0	0	3.5	10.9	0.1	1.8	4.5	0	0	0	20.8
1909-1910	0	0	0	0	T	5.0	3.5	2.7	1.2	0	0	0	12.4
1910-1911	0	0	0	0	T	5.5	5.8	0.4	3.9	0	0	0	15.6
1911-1912	0	0	0	0	T	0	5.5	3.4	9.6	0	0	0	18.5
1912-1913	0	0	0	0	1.0	2.5	1.3	2.1	0	0	0	0	6.9
1913-1914	0	0	0	0	T	1.5	11.0	10.7	0	0	0	0	23.2
1914-1915	0	0	0	0	0	6.2	2.2	0	3.1	2.3	0	0	13.8
1915-1916	0	0	0	0	0	2.7	1.7	3.6	1.1	1.1	0	0	10.2
1916-1917	0	0	0	0	T	5.3	0.1	0.7	0.1	0.4	0	0	6.6
1917-1918	0	0	0	T	T	12.1	19.6	1.3	0	2.4	0	0	35.4
1918-1919	0	0	0	0	0	T	T	1.6	0	0	0	0	1.6
1919-1920	0	0	0	0	0	1.4	1.0	2.9	0.3	T	0	0	5.6
1920-1921	0	0	0	0	0	0	11.9	1.3	0	0	0	0	13.2
1921-1922	0	0	0	0	0	0	22.7	4.7	0.4	0	0	0	27.8
1922-1923	0	0	0	0	T	T	2.7	5.1	0	0	T	0	7.8
1923-1924	0	0	0	0	0	0.6	0.8	1.3	5.2	T	0	0	7.9
1924-1925	0	0	0	0	0.7	0	2.2	0	T	0	0	0	2.9
1925-1926	0	0	0	2.9	0	0	5.5	1.5	4.7	1.7	0	0	16.3
1926-1927	0	0	0	0	T	0	2.1	2.4	2.7	T	0	0	7.2
1927-1928	0	0	0	0	0	0.6	3.4	0.3	2.2	0	0	0	6.5
1928-1929	0	0	0	0	0	0	0.9	7.6	0	0	0	0	8.5
1929-1930	0	0	0	0	3.5	4.9	12.0	0.2	T	0	0	0	20.6
1930-1931	0	0	0	T	T	7.8	1.4	0	4.3	0	0	0	13.5
1931-1932	0	0	0	0	0	0	0	0.5	0.8	0	0	0	1.3
1932-1933	0	0	0	0	0	11.9	3.1	5.9	T	0	0	0	20.9
1933-1934	0	0	0	0	T	T	T	16.2	1.1	T	0	0	17.3
1934-1935	0	0	0	0	0	4.0	2.6	0.1	T	T	0	0	6.7

Season	July	Aug.	Sept.	Oct.	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May	June	Total
1935-1936	0	0	0	0	1.0	7.8	2.0	10.4	3.0	T	0	0	24.2
1936-1937	0	0	0	0	T	1.0	0	5.3	3.0	T	0	0	9.3
1937-1938	0	0	0	0	0	1.9	1.2	T	T	0	0	0	3.1
1938-1939	0	0	0	0	3.0	1.0	3.3	T	T	0	0	0	7.3
1939-1940	0	0	0	0	1.2	5.9	18.5	T	0.9	T	0	0	26.5
1940-1941	0	0	0	T	0	0	1.5	1.4	6.2	0	0	0	9.1
1941-1942	0	0	0	0	0	0	2.3	0.4	2.5	0	0	0	5.2
1942-1943	0	0	0	0	0	2.8	1.3	0.4	3.3	T	0	0	7.8
1943-1944	0	0	0	0	0.4	2.2	2.1	4.0	1.6	T	0	0	10.3
1944-1945	0	0	0	0	T	1.7	1.4	0.2	0	0	0	0	3.3
1945-1946	0	0	0	0	T	8.4	3.7	2.0	0.5	0	0	0	14.6
1946-1947	0	0	0	0	0	1.0	0	7.4	19.1	0	0	0	27.5
1947-1948	0	0	0	0	0.7	0.2	12.1	13.0	T	0	0	0	26.0
1948-1949	0	0	0	0	T	4.1	4.1	2.4	T	T	0	0	10.6
1949-1950	0	0	0	0	T	1.5	T	1.6	5.6	T	0	0	8.7
1950-1951	0	0	0	0	T	8.2	1.7	1.3	T	0	0	0	11.2
1951-1952	0	0	0	0	0	3.6	1.6	2.3	0.4	T	0	0	7.9
1952-1953	0	0	0	0	1.4	1.1	T	0	6.1	T	0	0	8.6
1953-1954	0	0	0	0	2.9	T	4.6	T	T	T	0	0	7.5
1954-1955	0	0	0	0	2.5	1.5	10.1	0.4	T	0	0	0	14.5
1955-1956	0	0	0	0	0.4	1.1	2.1	0.2	0.7	T	0	0	4.5
1956-1957	0	0	0	0	T	T	4.6	0.8	0	2.8	0	0	8.2
1957-1958	0	0	0	T	T	1.4	3.3	13.0	6.3	0	0	0	24.0
1958-1959	0	0	0	0	T	4.1	7.8	T	T	0.6	0	0	12.5
1959-1960	0	0	0	0	T	0.7	5.6	14.0	24.9	0	0	0	45.2
1960-1961	0	0	0	0	0	1.2	8.8	8.1	T	T	0	0	18.1
1961-1962	0	0	0	0	0.6	2.1							

The horizontal lines drawn on the Average Temperature, Total Precipitation, Monthly and Seasonal Degree Days, and Monthly and Seasonal Snowfall tables separate the data according to station location (see Station Location table). Data for the period 11-15-36 through 7-26-44, when both City and Airport offices were in operation, are for the City Office location. Airport Data thereafter.

STATION LOCATION

Location	Occupied from	Occupied to	Airline distance and direction from previous location	Latitude	Longitude	Elevation above Sea level: Ground	Sea level: Actual barometer elevation (H_p)	Ground: Wind instruments	Extreme thermometers	Psychrometer	Telepsychrometer	Tipping bucket rain gage	Weighing rain gage	8" rain gage			REMARKS
						CITY OFFICE											
11th and Main Streets	5-24-71	6- 2-71		37° 25'N	79° 09'W	-	#645.3			-				-			U.S. Army Signal Corps. #Elevations not thought to be reliable.
8th and Court	6- 2-71	9- 1-73	1500 ft. NW	37° 25'N	79° 09'W	-	#730.3		-	-				-			U.S. Army Signal Corps. #Elevations not thought to be reliable.
135 Main Street	9- 1-73	5- 1-88	2000 ft.NNW	37° 25'N	79° 09'W	#624	#646.8	%69	31	31				50			U.S. Army Signal Corps. #Elevations not thought to be reliable.
Virginia Building 1001 Main Street	5- 1-88	4- 1-90	3000 ft. SE	37° 25'N	79° 09'W	-	#653.3	%73	67	67				57			U.S. Army Signal Corps. #Elevations not thought to be reliable.
Law Building 807 Main Street	4- 1-90	12-23-13	700 ft. NW	37° 25'N	79° 09'W	614	680.7	%88	83	83				77			
Peoples National Bank. Bldg. 801-803 Main St.	12-23-13	7-26-33	30 ft. NW	37° 25'N	79° 09'W	614	756	188	148	148		142		142			Wind velocity probably affected by bluffs to south.
309 Essex Street	7-26-33	11- 1-36	1.4 mi. NW	37° 25'N	79° 09'W	-	756.41		5	5				3			
Peoples National Bank Bldg. 801-803 Main St.	11- 1-36	12-22-37	1.4 mi. SE	37° 25'N	79° 09'W	614	756	184	148	148		142		142			
Peoples National Bank Bldg. 801-803 Main St.	12-22-37	7-26-44	-	37° 25'N	79° 09'W	614	756	184	144	144		142		142			Shelter moved to offset effects of a ventilator and chimneys near by.
						AIRPORT STATION											
Preston Glenn Airport 6.5 Mi. SSW of Post Office.	11-15-36	7-26-44		37° 20'N	79° 12'W	*947	*959	* 15		* 10							*All elevations approximate. Airport observations by CAA personnel.
Preston Glenn Airport 6.5 Mi. SSW of Post Office.	7-26-44	Present	6.7 Mi. SSW	37° 20'N	79° 12'W	947	955	50	5	5		3		3			Pibals taken from 8-1-44 to 12-10-47 then transferred to Roanoke. Airport observations by Weather Bureau personnel.

% Wind record prior to December 23, 1913 thought to be inaccurate due to high bluffs south of the exposure. All wind data prior to 12-23-13 omitted from summarizations according to orders from Central Office issued in 1939.

USCOMM-WB-Asheville, N. C. --- 1/19/62 --- 600

APPENDIX B

1. HYPOTHETICAL ACCIDENT

1.1 Considerations

Experiments performed with SPERT, BORAX, and other pool-type reactors show that no core damage occurs due to reactivity insertions if the available excess reactivity is kept to a low value. From these experiments, assurance is obtained that the likelihood of a major reactor excursion is almost non-existent. However, since the complete mechanism for a nuclear excursion or melting of a core is not understood, the results of a catastrophic accident are considered for the BAWTR. In this reactor, there are many features that prevent damage to the reactor from mechanical failures or accidents. Some of these are:

1. Negative temperature and void coefficient of the core tend to decrease reactivity in case of a power excursion.
2. Only a small amount of reactivity may be inserted into the core by any conceivable method.
3. There is an extensive amount of control built into the safety rods which are free to fall into the core.
4. There is a large amount of available cooling water in the pool in case of any type of accident. This water would also absorb a large portion of the products released from the core in an accident involving core meltdown.
5. There is provision for convection cooling for the core in case of electrical or mechanical failures.
6. The control system is designed to prevent a large ramp increase in reactivity from being added to the core.

7. Fission product release to the atmosphere must occur after penetrating three barriers, the fuel cladding, pool water absorption, and the containment building.

8. Coolant pipe rupture causes no loss of cooling water to the reactor.

For the hypothetical accident, it is assumed that the reactor and typical experiment have been operating for a period of 600 days at a power level of 12 MW (saturation). An accident occurs causing the melting of the core and the release of fission products. The following assumptions are used for the dose determinations at various locations:

1. The accident releases to the reactor building 100% of the noble gases, 50% of the halogens, and 1% of the solids in the fission product inventory. This represents about 15% of the gross fission product activity.

2. Fifty % of the iodine released from the core will be absorbed on the vessel walls. This does not take into account the absorption of iodine in the pool water.

3. The leakage rate to the atmosphere is 0.1% per day of the containment volume.

4. Wind direction and velocity or difussion parameters do not change during the course of the release.

5. Atmospheric dispersion is assumed to occur under "maximum average" meteorological conditions as given in the appendix to 10 CFR, Part 100.

6. Cloud depletion does not occur as dose is received.

7. Fission products are assumed to decay during the time in the containment but not during transit to persons receiving the dose.

8. The containment shell remains intact other than the leakage of 0.1% per day.

1.2 Inhalation Dose Calculations

The methods used to compute the doses to critical organs are the simplifications of Sutton's formulas as given in "Meteorology and Atomic Energy" (AECU 3066). The thyroid dose for the method of release is the controlling factor. Although the methods are general, the results of the calculations are specifically for the iodine isotopes.

Consideration is given to a person located at a distance (d, meters) downwind of the source (containment) for a period of time, T. No attempt was made to improve on existing methods or calculating doses because of the present state of the art.

The concentration of isotopes in the core was calculated, using existing computer codes, for operation at 12 MW for a period of 600 days. The amount of the isotopes present in the core at this time is given in Table B-1.

Table B-1. Inventory 600 Days at 12 MW

Isotope	Curies/MW	Curies, C	Decay Constant (sec^{-1})
I^{131}	2.5 (4)*	3.0 (5)	9.96 (-7)
I^{132}	3.7 (4)	4.5 (5)	8.26 (-5)
I^{133}	5.3 (4)	6.4 (5)	9.20 (-6)
I^{134}	6.1 (4)	7.4 (5)	2.20 (-4)
I^{135}	4.6 (4)	5.6 (5)	2.86 (-5)
$Xe^{131}m$	2.8 (2)	3.4 (3)	6.67 (-7)
$Xe^{133}m$	1.3 (3)	1.5 (4)	3.51 (-6)
Xe^{133}	5.2 (4)	6.4 (5)	1.52 (-6)
$Xe^{135}m$	1.5 (4)	1.8 (5)	7.45 (-4)
Xe^{135}	5.2 (4)	6.2 (5)	2.11 (-5)
$Kr^{83}m$	3.8 (3)	4.5 (4)	1.01 (-4)
$Kr^{85}m$	1.1 (4)	1.3 (5)	4.42 (-5)
Kr^{87}	2.5 (4)	3.0 (5)	1.45 (-4)
Kr^{88}	2.8 (4)	3.4 (5)	6.95 (-5)
Sr^{89}	3.8 (4)	4.5 (5)	1.57 (-7)
Sr^{90}	1.9 (3)	2.3 (4)	7.88 (-9)

* (4) = 10^4

The amount of the isotope, Q_t, released from the reactor building during the time interval, T, is given by

$$Q_t(\text{curies}) = F\ C\ \ell \int_0^T e^{-(\ell+\lambda)\,t}dt$$

where

F = fraction released from the building, for iodine $F = 0.25$

C = Curies in Table B-1

ℓ = leakage rate of containment

ℓ - 0.1%/day = 1.157 (−8)%/sec

λ = rate of decay in Table B-1

$$Q_t = \frac{FC\ell}{\ell+\lambda}\left[1-e^{-(\ell+\lambda)\,T}\right]$$

$$Q_\infty = \frac{FC\ell}{\ell+\lambda}$$

The amount of iodine isotopes released from the reactor building is given in Table B-2.

Table B-2. Curies Iodine Isotopes Released from Containment

	$\ell+\lambda$	$\frac{\ell}{\ell+\lambda}$	$Q_{2\ hr}$	Q_∞
I^{131}	1.01 (−6)	1.14 (−2)	0.63 (1)	0.86 (3)
I^{132}	8.26 (−5)	1.40 (−4)	0.72 (1)	1.60 (1)
I^{133}	9.22 (−6)	1.26 (−3)	1.30 (1)	1.99 (2)
I^{134}	2.20 (−4)	5.26 (−5)	7.71 (0)	9.69 (0)
I^{135}	2.86 (−5)	4.04 (−4)	1.05 (1)	5.64 (1)

The amount of radioactivity inhaled at ground level from the radioactive cloud at a distance, d (meters), from the containment vessel is given by

$$A_T(\text{curies}) = \frac{2RQ_T}{\pi\, C_y C_z \bar{\mu}\, d^{(2-n)}},$$

where

R = breathing rate, m/sec
= 10 m^3/8 hr (exclusion) = 3.47 (–4)m^3/sec
= 20 m^3/24 hr (populated) = 2.32 (–4)m^3/sec

C_y = 0.4 (diffusion coefficient)

C_z = 0.07 (diffusion coefficient)

$\bar{\mu}$ = 1.0 m/sec (wind velocity)

n = 0.5

$$A_T = 7.89(-3)\frac{Q_T}{d^{1.5}} \quad \text{(exclusion area)}$$

$$= 5.27(-3)\frac{Q_T}{d^{1.5}} \quad \text{(low population zone)}$$

Table B-3 gives the curies inhaled at the exclusion area and in low populated areas as a function of distance from the containment.

The dose (Rad) to the critical organ after a long period of time due to inhalation is given by

$$D_\infty = \frac{8.54\times10(2)\ A_T\, f_a\, \bar{E}\, T_e}{m},$$

where

m = mass of organ, gm

f_a = fraction deposited in the body of the inhaled air

$\bar{E}$ = energy (Mev) absorbed per disintegration

T_e = half life for isotope in body, sec

Table B-3. Curies of Iodine Isotopes Inhaled

Distance, m	100	200	500	1,000	10,000
			Exclusion, 2 hr		
I^{131}	4.96 (-5)	1.75 (-5)	4.44 (-6)	1.57 (-6)	4.96 (-8)
I^{132}	5.77 (-5)	2.04 (-5)	5.16 (-6)	1.82 (-6)	5.77 (-8)
I^{133}	9.68 (-5)	3.42 (-5)	8.66 (-6)	3.06 (-6)	9.68 (-8)
I^{134}	6.13 (-5)	2.16 (-5)	5.46 (-6)	1.93 (-6)	6.10 (-8)
I^{135}	8.29 (-5)	2.93 (-5)	7.41 (-6)	2.62 (-6)	8.29 (-8)
			Low Population Zone (∞ time)		
I^{131}	45.42 (-4)	16.10 (-4)	40.63 (-5)	14.36 (-5)	45.42 (-7)
I^{132}	0.85 (-4)	0.30 (-4)	0.76 (-5)	0.27 (-5)	0.85 (-7)
I^{133}	10.51 (-4)	3.72 (-4)	9.40 (-5)	3.32 (-5)	10.51 (-7)
I^{134}	0.51 (-4)	0.18 (-4)	0.46 (-5)	0.16 (-5)	0.51 (-7)
I^{135}	2.98 (-4)	1.05 (-4)	2.66 (-5)	0.94 (-5)	2.98 (-7)

Doses due to iodine inhalation versus distance from containment are given in Table B-4.

These results indicate that, under inverse weather conditions, the doses at the populated areas will not exceed 300 rem to the thyroid due to the specified release of iodine to the atmosphere.

Table B-4. Dose (Rad) to the Thyroid from Iodine Inhalation

Distance, m	100	200	500	1,000	10,000
			Exclusion, 2 hr		
I^{131}	7.34 (1)	2.64 (1)	6.56 (0)	2.32 (0)	7.34 (-2)
I^{132}	0.31 (1)	0.11 (1)	0.28 (0)	0.98 (0)	0.31 (-2)
I^{133}	3.87 (1)	1.37 (1)	3.46 (0)	1.22 (0)	3.87 (-2)
I^{134}	0.15 (1)	0.05 (1)	0.14 (0)	0.05 (0)	0.15 (-2)
I^{135}	1.03 (1)	0.36 (1)	0.92 (0)	0.32 (0)	1.03 (-2)
Total	12.80 (1)	4.54 (1)	11.36 (0)	4.02 (0)	12.70 (-2)

Table B-4 (Cont'd)

Distance, m	100	200	500	1,000	10,000
			Low Population (∞ time)		
I^{131}	67.23 (2)	23.82 (2)	60.13 (1)	21.26 (1)	67.23 (-1)
I^{132}	0.05 (2)	0.02 (2)	0.04 (1)	0.01 (1)	0.05 (-1)
I^{133}	4.20 (2)	1.49 (2)	3.76 (1)	1.33 (1)	4.20 (-1)
I^{134}	0.01 (2)	0.00 (2)	0.01 (1)	0.00 (1)	0.01 (-1)
I^{135}	0.37 (2)	0.13 (2)	0.33 (1)	0.12 (1)	0.37 (-1)
Total	71.86 (2)	25.48 (2)	64.27 (1)	22.72 (1)	71.86 (-1)

1.3 Direct Dose from Containment

After a long period of reactor operation the delayed gamma photon spectrum in the reactor core is represented by,

$$N(E) = \ .0\ e^{-1.1E}/\text{mev-fiss}$$

where E is energy of the photons in mev

Operation of the 12 MW reactor requires

$$3.125 \times 10^{10}\ \frac{\text{fiss}}{\text{watt-sec}} \times 1.2 \times 10^{7}\ \text{watts} =$$

$$3.75 \times 10^{17}\ \text{fiss/sec}$$

Then the γ photons in energy ranges are represented by

$$\gamma(E) = 2.25 \times 10^{18} \int_{E_i}^{E_{i+1}} e^{-1.1E} dE\ \ \gamma/\text{sec}$$

The source strengths are then as given in Table B-5.

Table B-5. Volumetric Source Strength

γ energy, mev	S_o, γ/sec	$\frac{S_o}{V} = S_v$, γ/cm³-sec
0.35	5.15 (17)	3.27 (8)
0.75	5.57 (17)	3.54 (8)
1.35	2.97 (17)	1.88 (8)
1.65	1.81 (17)	1.15 (8)
2.10	1.01 (17)	0.64 (8)
2.55	0.65 (17)	0.41 (8)

$V = 1.575 \times 10^9 \text{cm}^3 = 5.565 \times 10^4 \text{ft}^3$ (volume of containment)

The dose-rate at a distance (a) from a cylinder is given by

$$D\left(R/hr\right) = \frac{C(E)\, S_v(E)\, B(E)\, R^2}{2\,(a+z)} \int_0^{\Theta} e^{-b \sec \Theta}\, d\Theta ,$$

where

C(E) = conversion factor, $\frac{\gamma}{\text{cm}^2\text{-sec}}$ to $\frac{R}{hr}$

$S_v(E)$ = Volumetric Source Strength, $\frac{\gamma}{\text{cm}^2\text{-sec}}$

B(E) = Dose buildup factors

R = Radius of containment, cm (16.5 feet)

a = Distance from source to receptor, cm

z = Self shielding factor of source, cm (used as = 0 for calculations)

b = $\mu_i\, t_i$ i (dimensionless)

μ_i = linear coefficient of absorption

t_i = thickness of shields

Θ = $\tan^{-1} H/2a$

H = height of source, cm (65 feet)

The dose rate in the air at a distance from the containment with no shielding is given by

$$D\left({}^{R}/hr\right) = \sum_{E} \frac{C(E)\ S_v(E)\ B(E)\ R^2}{2a} f(\Theta, \mu a),$$

where

$$\mu a = \mu \text{ (linear absorption coefficient of air)} \times a.$$

$$f(\Theta, \mu a) = \int_0^{\Theta} e^{-\mu a \sec\Theta} d\Theta$$

Table B-6 gives the buildup factors and the values of $f(\Theta, \mu a)$ used for the calculations.

Table B-6. Buildup Factors and f $(\Theta, \mu a)$

	B(E)			
	a = 100 m	a = 200 m	a = 500 m	a = 700 m
0.35	2.33	14.55	37.50	66.50
0.75	1.83	4.80	11.00	18.00
1.35	1.80	4.00	4.40	6.65
1.65	1.70	3.75	4.15	6.00
2.10	1.65	2.35	2.45	4.26
2.55	1.65	2.00	3.50	4.40
	$f(\Theta, \mu a)$			
0.35	2.55 (−2)	0.70 (−3)	0.43 (−4)	0.03 (−4)
0.75	3.80 (−2)	2.00 (−3)	2.80 (−4)	0.43 (−4)
1.35	4.80 (−2)	4.20 (−3)	7.50 (−4)	1.65 (−4)
1.65	5.00 (−2)	4.30 (−3)	8.60 (−4)	2.00 (−4)
2.10	5.10 (−2)	5.40 (−3)	13.80 (−4)	3.00 (−4)
2.55	5.25 (−2)	6.70 (−3)	16.50 (−4)	4.50 (−4)

The calculated dose rates at peak concentrations are given in Table B-7.

Table B-7. Dose Rate ($^R/hr$) for Peak Concentration

E	C(E)	(air) cm^{-1}	a 100	300	a, m 500	700
0.35	0.72 (−6)	1.20 (−4)	1.77 (2)	1.01 (1)	0.95	0.08
0.75	1.50 (−6)	0.83 (−4)	4.65 (2)	2.15 (1)	4.13	0.73
1.35	2.45 (−6)	0.63 (−4)	5.04 (2)	3.27 (1)	3.85	0.92
1.65	2.85 (−6)	0.60 (−4)	3.52 (2)	2.22 (1)	2.95	0.71
2.10	3.35 (−6)	0.53 (−4)	2.27 (2)	1.14 (1)	1.83	0.49
2.55	3.85 (−6)	0.47 (−4)	1.73 (2)	0.89 (1)	2.31	0.55
Total (100% Release)		D_p =	18.98 (2)	10.68 (1)	16.02	3.48
Total (15% Release)		D_p =	284.96	16.04	2.39	0.52

The following equation is used to obtain dose-rates at time, T, after the peak concentration.

$$D_T \ (^R/hr) = \sum_E \frac{S_T(E)}{S_o(E)} \ D_P(E).$$

Table B-8 shows the dose-rates at times after the peak.

Table B-8. Dose Rate ($^R/hr$) at Times After Peak Concentration

(15% Release)

T, days	100 m	300 m	500 m	700 m
1/4	25.04	1.46	2.06 (−1)	4.43 (−2)
1.0	21.92	1.29	1.82 (−1)	3.89 (−2)
4.0	15.24	0.88	1.28 (−1)	2.68 (−2)
10.0	9.32	0.52	0.79 (−1)	1.62 (−2)
30.0	4.10	0.22	0.35 (−1)	0.69 (−2)

The decay constant averaged over the first 6 hours is calculated to be 0.40. The dose is expressed by

$$D_T(\text{Rem}) = \int_o^T D_o e^{-0.4t} dt = \frac{D_o}{0.4} e^{-0.4T} .$$

The integrated doses were then calculated by integrating the dose rate at various distances over the periods of time up to 2 hours for the exclusion area and 30 days for the low population zones. The results are given in Table B-9.

Table B-9. Doses (Rem) due to Direct Irradiation

Distance, m	100	300	500	700
For 2 hr	261.0	15.0	2.20	0.48
For 30 days	668.2	37.8	5.6	1.2

1961
Annual Report

The Babcock & Wilcox Company

Our 94th Year of Helping Industry Serve America

The Babcock & Wilcox Company

Incorporated in 1881 from a partnership formed in 1867

Directors

A. G. PRATT, *Chairman*
L. M. CURRIE
WALTHER H. FELDMANN
J. ROY GORDON
C. W. MIDDLETON
M. NIELSEN
STODDARD M. STEVENS
W. J. THOMAS
B. A. TOMPKINS
JOHN C. TRAPHAGEN
L. S. WILCOXSON

Officers

M. NIELSEN, *President*
J. S. ANDERSON, *Vice President*
R. A. BARR, *Vice President*
CARL CLAUS, *Vice President*
J. P. CRAVEN, *Vice President*
L. M. CURRIE, *Vice President*
J. D. HITCH, JR., *Vice President*
G. W. KESSLER, *Vice President*
S. T. MACKENZIE, *Vice President*
ALAN E. PHIN, *Vice President*
W. H. ROWAND, *Vice President*
A. P. TABER, *Vice President*
W. J. THOMAS, *Vice President*
L. S. WILCOXSON, *Vice President*
R. J. CANTWELL, *Comptroller*
W. G. DRYDEN, *Treasurer*
MARTIN VICTOR, *Secretary*
O. R. CARPENTER, *Assistant Vice President*
N. K. FELDMAN, *Assistant Comptroller*
E. F. HORGAN, *Assistant Secretary*

General Offices	161 East 42nd Street, New York 17, New York
Stock Transfer Agent	Bankers Trust Company, 16 Wall Street, New York 15, New York
Registrar	The Bank of New York, 48 Wall Street, New York 15, New York
General Counsel	Sullivan & Cromwell
Auditors	Price Waterhouse & Co.

Contents

The Babcock & Wilcox Company

Eighty-first

Annual Report

for the year ended December 31

1961

1961 Highlights

	1961	1960
Per Share Issued[1]		
Net income for the year	$ 3.27	$ 2.88
Cash dividends declared	1.45	1.25
Stockholders' equity	31.23	29.39
In Thousands of Dollars		
Orders received	$361,285	$301,146
Unfilled orders	333,027	291,095
Sales (on percentage of completion method for long-term contracts)	319,353	310,999
Income before income taxes and minority interest	43,122	36,500
U. S. and foreign taxes on income	22,580	18,380
Net income for the year	20,241	17,817
Cash dividends declared	8,849	7,632
Expenditures for property, plant and equipment	6,188	6,105
Depreciation and amortization	8,673	8,543
Payrolls and employee benefits	127,123	128,227
Average number of employees during year	18,600	19,500
Number of stockholders at year end	16,597	18,377
Number of shares issued at year end	6,183,313	6,183,313

[1] Based on the shares issued, including shares held in the Treasury for the Stock Option Plan.

The President's Letter

To the Stockholders of
The Babcock & Wilcox Company:

Net income for 1961 of $20,241,000 or $3.27 per share was the highest in the Company's history. It exceeded the previous record, established in 1960, by 13.6 per cent even though sales (shipments) increased only 2.7 per cent.

At its meeting in November 1961, the Board of Directors increased the dividend from an annual rate of $1.40 to $1.60 per share. This is the third successive year that the cash dividend has been increased.

B&W profit ratios have been improving during the past three years. Operating profit as a per cent of sales reached 13.1 per cent in 1961, the highest achieved during the past decade. This has been brought about by sustained efforts to improve operating efficiency and as a result of the large capital investments made in recent years.

Programs to insure future progress are being strengthened and accelerated. During 1961, the Board authorized approximately $28,000,000 for capital projects to be completed during 1962 and 1963. This is the largest amount appropriated in any one year in B&W history. The Company's research and development program is also being intensified to speed the development of new and better products and manufacturing processes.

The moderate improvement in general business conditions during 1961 is reflected in a higher volume of new orders as compared with 1960. Since these new orders exceeded shipments, the backlog of $333,027,000 at the end of 1961 was 14.4 per cent higher than the backlog at the end of 1960.

Previous annual reports have described the diversification of the Company's products and markets. In this year's report, we show a cross-section of B&W products and their applications, with particular emphasis on products of more recent design.

The progress the Company has made would not have been possible without the cooperation of its many customers, the teamwork of its employees, and the support of its stockholders. Speaking for the Board, I wish to express our appreciation to all who are contributing to these efforts.

M. Nielsen.
President

New York, N.Y.
March 1, 1962

Financial Review

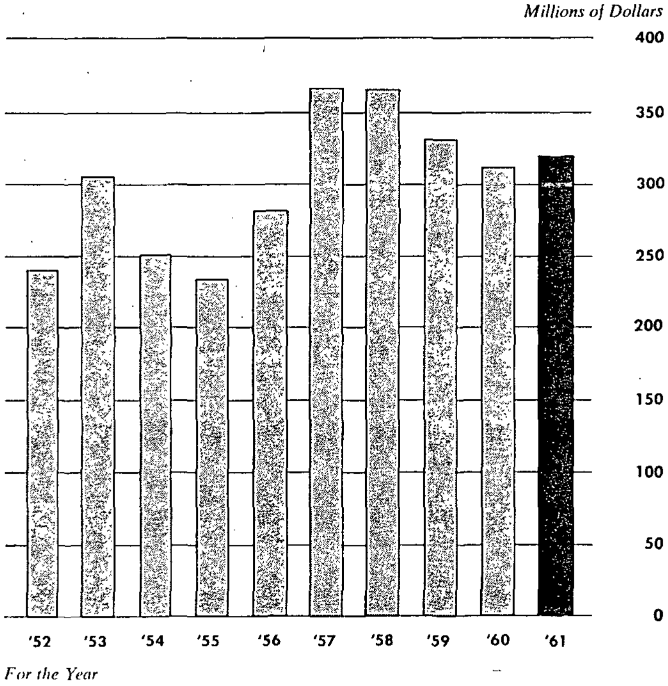
The figures given in this report result from a consolidation of the accounts of the Company and its subsidiaries, Bailey Meter Company, Diamond Power Specialty Corporation, and A. M. Lockett & Co., Ltd.

Sales (Shipments), Income and Dividends

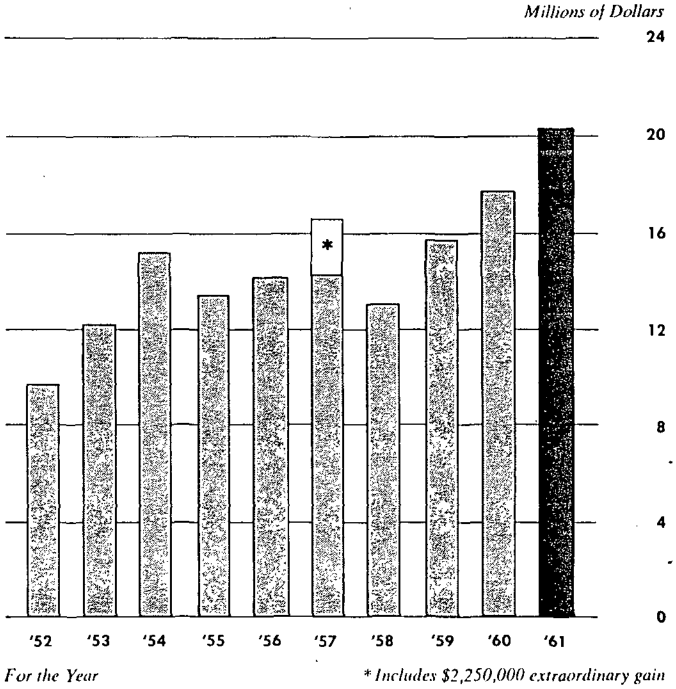
	1961	1960
Sales (shipments)	$319,353,000	$310,999,000
Income from operations . .	41,959,000	35,707,000
Net income . . .	20,241,000	17,817,000
Shares issued . .	6,183,313	6,183,313
Earned per share .	$3.27	$2.88

Operating efficiency increased still further in 1961. As a result, a 2.7 per cent increase in sales (shipments) produced a 17.5 per cent increase in income from operations.

Sales (Shipments)



Net Income



* Includes $2,250,000 extraordinary gain

Dividends were paid and declared during 1961 as follows:

Paid in 1961

Amount Paid	Date Paid	Date of Record
$ .35	Jan. 4	Dec. 9, 1960
.35	April 3	Mar. 10
.35	July 3	June 9
.35	Oct. 2	Sept. 11

$1.40 per share total paid

Declared in 1961

Amount Declared	Date Paid	Date Declared
$ .35	April 3	Feb. 23
.35	July 3	May 25
.35	Oct. 2	Aug. 31
.40	Jan. 2, 1962	Nov. 30 *

$1.45 per share total declared

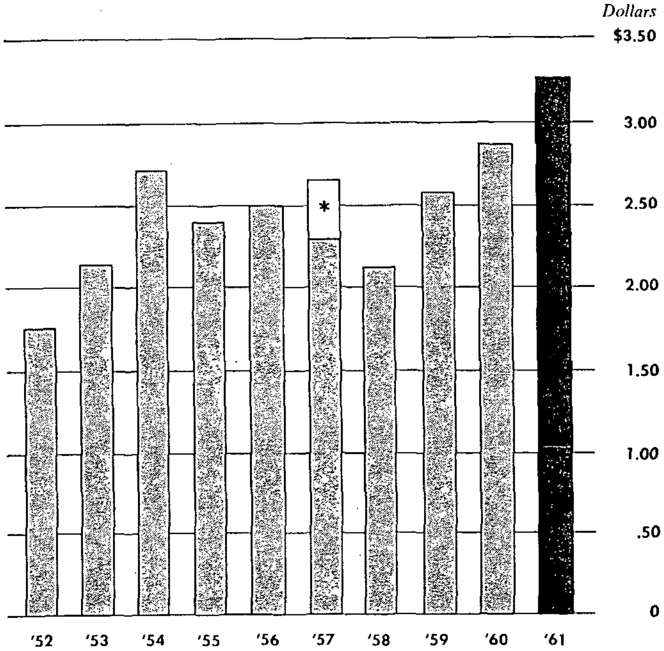
* *To holders of record on Dec. 11*

Financial Position

The Company's financial position was further strengthened during the past year. Cash and marketable securities increased from $61,570,000 to $66,660,000. At the year end they exceeded total current liabilities of $50,206,000 by $16,454,000. These funds, together with expected cash flow, should be sufficient to finance the business until at least the end of 1963.

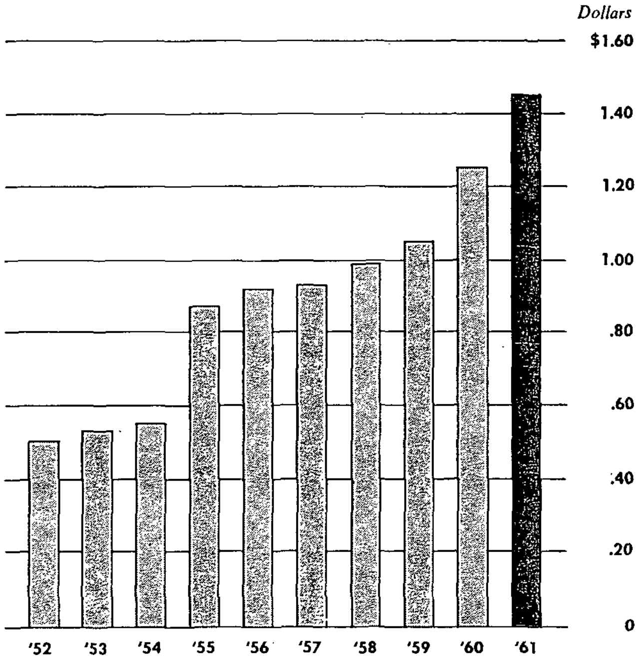
On June 1, 1961, the Company paid the first installment of $2,200,000 on its long-term debt. It also exercised its option to pay an additional $2,200,000 without penalty. As a result, the total debt was reduced from $32,000,000 to $27,600,000.

Earnings per Share



For the Year — ** Includes 36 cents per share extraordinary gain*

Cash Dividends Declared per Share



For the Year

Orders and Backlog

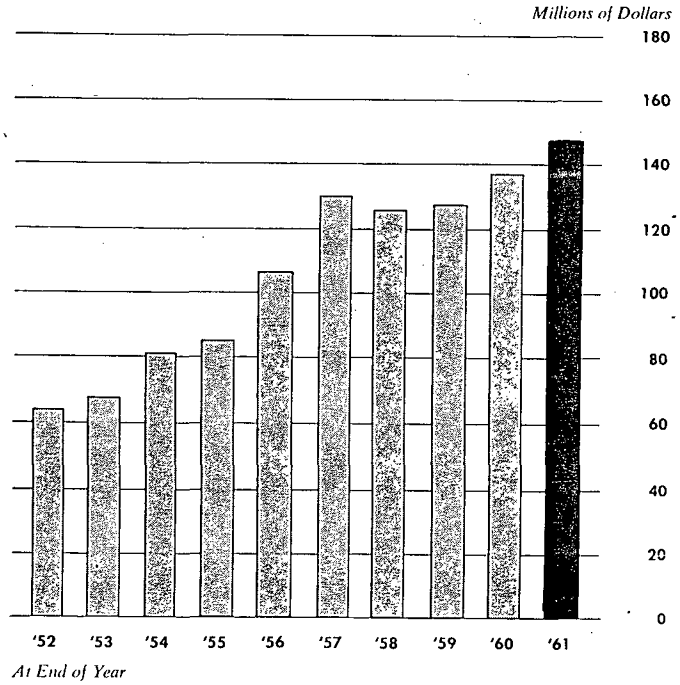
New orders received during 1961 totaled $361,285,000, up from the $301,146,000 booked in 1960. Because new orders exceeded sales (shipments), unfilled orders increased from $291,095,000 at the end of 1960 to $333,027,000 at the end of 1961. A large part of the backlog is for steam generating equipment to be shipped in 1963 or later.

Property, Plant and Equipment

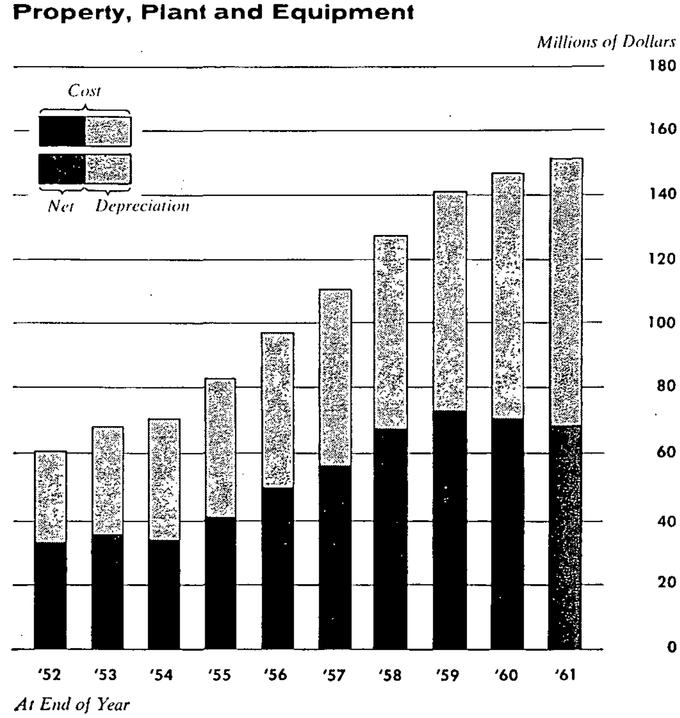
During 1961 the Company spent $6,188,000 for property, plant and equipment. At the year end, $25,628,000 authorized by the Board remained to be expended. This is the largest program of capital projects in the history of the Company. All of it is scheduled for expenditure in 1962 and 1963.

In the Boiler Division, the major expenditures will be for new machine tools and facilities to improve efficiency and lower costs. In the Tubular Products Division, the largest projects are additional steel melting capacity and modern rolling mill facilities. Research and Development, Refractories, Atomic Energy, Bailey Meter and Diamond Power are also included in this capital program.

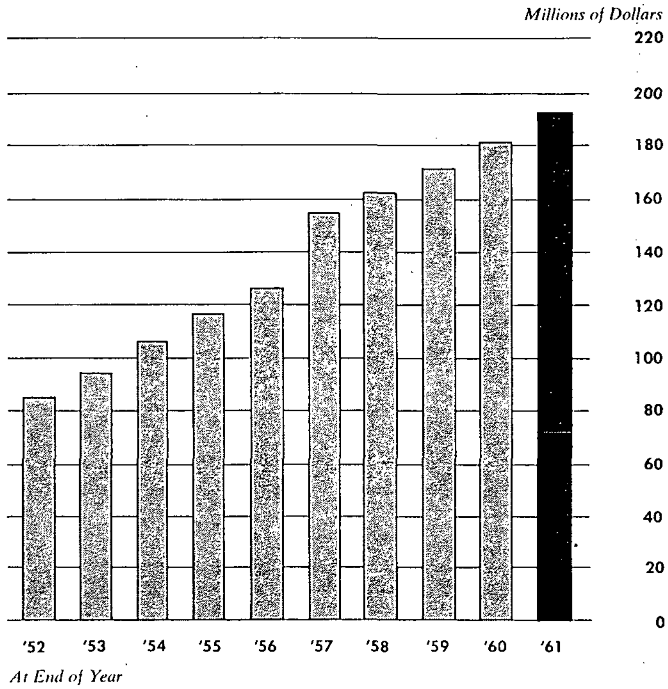
Working Capital



Property, Plant and Equipment



Stockholders' Equity



Employees

The number of active employees during 1961 averaged 18,600. At the end of December 961 retired employees were receiving pensions, an increase during the year of 102. The cost of payrolls and employee benefits, including Company contributions to pension funds, totaled $127,123,000. On December 31 B&W employees were covered by a total of $154,560,000 of group life insurance.

Under formalized pension plans approved by the stockholders in 1955, actuarially determined payments totaling $4,455,000 were made to the pension and retirement funds in 1961.

The changes in the funds are summarized below:

Total assets on January 1, 1961	$19,481,000
Contributions by Company during 1961	4,455,000
Income from pension fund investments during 1961 .	795,000
Less pensions paid during 1961	(1,327,000)
Total assets on December 31, 1961 . . .	$23,404,000

Following is the status of the Company's Stock Option Plan approved by the stockholders in 1958. It authorized granting of options to key employees to purchase up to 150,000 shares of Babcock & Wilcox capital stock at prices not less than 95 per cent of the market value at the time of granting. In accordance with the plan, the number of shares was increased to 151,500, after the one per cent stock dividend was declared in 1958.

Options were outstanding on January 1, 1961 for .	83,927 shares
Options were exercised during 1961 for	7,671 shares
Options were outstanding December 31, 1961 for a total of	76,256 shares

At the end of December, 62,119 shares remained for the granting of options.

Because only previously issued shares must be used to meet the stock options granted, the Company acquired sufficient shares on the open market for that purpose. The cost of shares acquired for options outstanding on December 31, 1961 was $2,542,915. This amount is included in Prepaid Expenses and Other Assets in the Statement of Financial Condition. If all outstanding options are exercised, $2,405,651 will be returned to the Company.

1961 Income and Where It Went

	Our Income Was
From products shipped to customers	$319,353,000
From investment income	2,283,000
Total	**$321,636,000**

	We Paid Out and Provided	
For materials and services purchased	$138,624,000	43.1%
For wages, salaries and employee benefits	127,123,000	39.5
For taxes	25,554,000	8.0
For depreciation of plant and equipment	8,673,000	2.7
For interest on money borrowed	1,120,000	0.3
For minority interest in subsidiary company	301,000	0.1
For cash dividends to stockholders	8,849,000	2.8
For future growth (retained earnings)	11,392,000	3.5
Total	**$321,636,000**	**100.0%**

Consolidated Statement of Income and Retained Earnings

for the Calendar Year

	1961	1960
Sales (on percentage of completion method for long-term contracts)	$319,352,917	$310,999,299
Costs and expenses:		
Materials, payrolls, services and taxes other than U. S. and foreign taxes on income	268,720,966	266,750,115
Depreciation of plants and equipment (includes amortization $1,246,261 in 1961 and $1,323,116 in 1960)	8,672,924	8,542,657
	277,393,890	275,292,772
Income from operations	41,959,027	35,706,527
Income from investments and marketable securities	2,282,931	2,011,217
Interest expense	(1,119,850)	(1,217,500)
	43,122,108	36,500,244
U. S. and foreign taxes on income	22,580,000	18,380,000
	20,542,108	18,120,244
Income applicable to minority interest	(301,586)	(302,650)
Net income for the year	20,240,522	17,817,594
Cash dividends declared	8,849,285	7,631,916
Remainder added to retained earnings	11,391,237	10,185,678
Retained earnings at beginning of year	99,405,814	89,220,136
Retained earnings at end of year (see note below)	$110,797,051	$ 99,405,814

NOTE: Under note agreements, retained earnings in the sum of $66,079,000 on December 31, 1961 were unrestricted for the payment of dividends.

Consolidated Statement of Financial Condition

	AT DECEMBER 31 1961	1960
CURRENT ASSETS		
Cash	$ 10,771,086	$ 12,887,100
Marketable securities, at cost (approximate market)	55,889,154	48,682,491
Accounts receivable	54,218,097	53,133,067
Unbilled shipments and installations, at contract prices	17,452,803	15,579,953
Inventories, at lower of cost or market	72,150,592	66,188,164
Advance payments on contracts	(13,752,233)	(13,205,440)
TOTAL CURRENT ASSETS	196,729,499	183,265,335
CURRENT LIABILITIES		
Portion of long-term debt due within one year	2,200,000	2,200,000
Accounts payable	9,816,887	11,191,654
Accrued liabilities	17,562,902	15,911,264
Cash dividends payable	2,442,823	2,135,065
Provision for additional costs on contracts	2,000,000	1,603,107
U. S. and foreign income taxes	16,183,191	13,059,618
TOTAL CURRENT LIABILITIES	50,205,803	46,100,708
WORKING CAPITAL	146,523,696	137,164,627
INVESTMENTS, at cost	1,044,417	723,120
PROPERTY, PLANT AND EQUIPMENT (see page 11)	68,406,298	70,924,940
PREPAID EXPENSES AND OTHER ASSETS (includes cost of shares of the Company's stock acquired for stock options — 76,256 shares in 1961; 83,927 shares in 1960 — see page 7)	3,929,735	4,096,486
PATENTS	1	1
WORKING CAPITAL AND OTHER ASSETS	219,904,147	212,909,174
DEDUCT		
Long-term debt — notes payable (3⅝% and 3⅞%) due in annual installments, June 1, 1964-1974	25,400,000	29,800,000
Minority interest in subsidiary company	1,375,669	1,371,933
NET ASSETS	$193,128,478	$181,737,241
REPRESENTED BY		
Capital stock, par value $9 per share; authorized 9,000,000 shares; issued 6,183,313 shares (including $26,681,610 excess over par value — capital surplus)	$ 82,331,427	$ 82,331,427
Retained earnings (see note page 9)	110,797,051	99,405,814
	$193,128,478	$181,737,241

Consolidated Statement of Property, Plant and Equipment

	AT DECEMBER 31	
	1961	**1960**
COST		
At beginning of year	$146,140,091	$140,587,265
Expenditures during year	6,188,038	6,105,390
Retired or sold during year	(810,331)	(552,564)
At end of year	151,517,798	146,140,091
DEPRECIATION AND AMORTIZATION		
At beginning of year	75,215,151	67,196,871
Charged to operations during year	8,672,924	8,542,657
Accumulated on property retired or sold during year	(776,575)	(524,377)
At end of year	83,111,500	75,215,151
Net book value	$ 68,406,298	$ 70,924,940

Opinion of Independent Public Accountants

TO THE BOARD OF DIRECTORS OF
THE BABCOCK & WILCOX COMPANY:

In our opinion, the accompanying statements present fairly the consolidated financial position of The Babcock & Wilcox Company and its subsidiaries at December 31, 1961 and the results of their operations for the year, in conformity with generally accepted accounting principles applied on a basis consistent with that of the preceding year. Our examination of these statements was made in accordance with generally accepted auditing standards and accordingly included such tests of the accounting records and such other auditing procedures as we considered necessary in the circumstances.

PRICE WATERHOUSE & CO.

New York, N. Y.
February 28, 1962

Ten Year Comparison of Income and Retained Earnings

(in thousands of dollars)

	1961
Sales (on percentage of completion method for long-term contracts)	$ 319,353
COSTS AND EXPENSES:	
Materials, payrolls, services and taxes other than U. S. and foreign taxes on income	268,721
Depreciation and amortization	8,673
	277,394
Income from operations	41,959
Income from investments and marketable securities	2,283
Interest expense	(1,120)
	43,122
U. S. and foreign taxes on income	22,580
	20,542
Income applicable to minority interest	(301)
Provision for pension reserve prior to funded plan	
Extraordinary gain less related taxes	
Net income for the year	20,241
DIVIDENDS DECLARED:	
Cash	8,849
Stock – 1% in 1958, 4% in 1957 and 1956, 5% in all other years indicated	
	8,849
Remainder added to retained earnings	11,392
Retained earnings at beginning of year	99,405
Transfer to capital stock	
Transfer from pension reserve	
Retained earnings at end of year	$ 110,797

Statistics

Income from operations before taxes – % of sales	13.1%
Net income for year – % net worth first of year	11.1%
Orders received – thousands of dollars	$ 361,285
Unfilled orders – thousands of dollars	$ 333,027
Shares issued during year: Stock dividend	
Stock split	
Stock sold	
Shares issued end of year	6,183,313
Per Share issued (adjusted for stock dividends and stock splits – 6,183,313 shares in 1957-1961 inclusive and 5,621,194 in previous years)	
Income for year	**$ 3.27**
Cash dividends declared	**1.45**
Stockholders' equity	**31.23**

Note: Figures for 1952 have been restated to include subsidiaries not consolidated in the published report for that year.

1960	1959	1958	1957	1956	1955	1954	1953	1952
$ 310,999	$ 332,071	$ 365,941	$ 366,081	$ 281,485	$ 233,291	$ 250,471	$ 305,746	$ 241,267
266,749	291,058	329,221	326,375	244,890	197,006	214,539	273,701	213,510
8,543	7,650	7,013	6,779	6,699	6,249	5,234	4,410	3,289
275,292	298,708	336,234	333,154	251,589	203,255	219,773	278,111	216,799
35,707	33,363	29,707	32,927	29,896	30,036	30,698	27,635	24,468
2,011	1,509	462	386	562	515	737	206	436
(1,218)	(1,218)	(1,218)	(1,238)	(854)	(354)	(580)	(891)	(397)
36,500	33,654	28,951	32,075	29,604	30,197	30,855	26,950	24,507
18,380	17,430	15,640	17,630	15,320	16,520	15,470	14,190	13,750
18,120	16,224	13,311	14,445	14,284	13,677	15,385	12,760	10,757
(303)	(286)	(276)	(239)	(203)	(190)	(173)	(699)	(898)
								(50)
			2,250					
17,817	15,938	13,035	16,456	14,081	13,487	15,212	12,061	9,809
7,632	6,435	6,107	5,753	5,146	4,901	3,112	2,963	2,822
		1,993	7,000	8,760	7,470	4,500	2,810	2,353
7,632	6,435	8,100	12,753	13,906	12,371	7,612	5,773	5,175
10,185	9,503	4,935	3,703	175	1,116	7,600	6,288	4,634
89,220	79,717	74,782	71,079	70,984	67,618	60,018	53,730	49,096
				(80)				
					2,250			
$ 99,405	$ 89,220	$ 79,717	$ 74,782	$ 71,079	$ 70,984	$ 67,618	$ 60,018	$ 53,730

11.5%	10.0%	8.1%	9.0%	10.6%	12.9%	12.2%	9.0%	10.1%
10.4%	9.8%	8.4%	13.0%	12.0%	12.7%	16.1%	14.2%	12.5%
$ 301,146	$ 395,540	$ 236,201	$ 306,072	$ 495,257	$ 317,283	$ 192,199	$ 157,255	$ 227,563
$ 291,095	$ 300,948	$ 237,479	$ 367,219	$ 427,228	$ 213,456	$ 129,464	$ 187,736	$ 336,227
	61,220	235,465	205,826	81,677	77,787	74,083	70,555	
				3,430,436				
			535,148					
6,183,313	6,183,313	6,122,093	5,886,628	5,145,654	1,633,541	1,555,754	1,481,671	1,411,116
$ 2.88	**$ 2.58**	**$ 2.11**	**$ 2.66**	**$ 2.50**	**$ 2.40**	**$ 2.71**	**$ 2.15**	**$ 1.75**
1.25	**1.05**	**.99**	**.93**	**.92**	**.87**	**.55**	**.53**	**.50**
29.39	**27.74**	**26.21**	**25.09**	**22.44**	**20.85**	**18.92**	**16.77**	**15.15**

Ten Year Comparison of Financial Condition

	1961
CURRENT ASSETS	
Cash	$ 10,771
Marketable securities	55,889
Accounts receivable	54,218
Unbilled shipments and installations	17,453
Inventories	72,151
Advance payments on contracts	(13,752)
TOTAL CURRENT ASSETS	196,730
CURRENT LIABILITIES	
Notes payable	2,200
Accounts payable	9,817
Accrued liabilities	17,563
Cash dividends payable	2,443
Provision for additional costs on contracts	2,000
U. S. and foreign income taxes	16,183
TOTAL CURRENT LIABILITIES	50,206
WORKING CAPITAL	146,524
INVESTMENTS	1,044
PENSION RESERVE INVESTMENTS	
PROPERTY, PLANT AND EQUIPMENT	68,406
PREPAID EXPENSES AND OTHER ASSETS	3,930
WORKING CAPITAL AND OTHER ASSETS	219,904
DEDUCT:	
Long-term debt	25,400
Reserve for pensions prior to funded plan	
Minority interest in subsidiary companies	1,375
NET ASSETS	$193,129
REPRESENTED BY	
Capital including capital surplus	$ 82,332
Stock dividends paid January following year	
Retained earnings	110,797
	$193,129

Ten Year Comparison of Property, Plant and Equipment

COST	
At beginning of year	$146,140
Expenditures during year	6,188
Retired or sold during year	(810)
At end of year	151,518
DEPRECIATION AND AMORTIZATION	
At beginning of year	75,215
Charged to operations during year	8,673
Accumulated on property retired or sold during year	(776)
At end of year	83,112
NET BOOK VALUE	$ 68,406

Note: Figures for 1952 have been restated to include subsidiaries not consolidated in the published report for that year.

(at December 31st in thousands of dollars)

1960	1959	1958	1957	1956	1955	1954	1953	1952
$ 12,887	$ 9,531	$ 11,764	$ 12,873	$ 14,578	$ 14,184	$ 20,121	$ 9,062	$ 6,785
48,682	29,987	11,686	4,729	1,999	8,805	7,268	1,550	1,950
53,133	59,726	52,514	45,686	39,394	34,198	39,277	43,886	42,560
15,580	21,688	29,533	20,200	10,161	10,925	17,758	20,638	17,597
66,188	63,357	76,939	104,504	91,486	56,996	40,975	56,902	56,768
(13,205)	(10,608)	(11,226)	(12,168)	(9,427)	(2,232)	(2,649)	(2,069)	(13,887)
183,265	173,681	171,210	175,824	148,191	122,876	122,750	129,969	111,773
2,200							20,500	10,000
11,192	12,678	12,991	10,838	12,270	9,394	11,413	12,833	11,240
15,911	16,950	15,436	14,922	12,186	9,045	9,002	10,922	9,085
2,135	1,832	1,520	1,472	1,286	1,225	778	741	706
1,603	2,611	3,913	3,315	1,762	2,803	4,000	2,500	2,500
13,060	12,157	12,312	14,975	13,867	15,109	16,588	15,560	14,770
46,101	46,228	46,172	45,522	41,371	37,576	41,781	63,056	48,301
137,164	127,453	125,038	130,302	106,820	85,300	80,969	66,913	63,472
723	604	604	604	1,600	1,600	1,603	2,177	2,194
						2,250	2,250	2,250
70,925	73,390	67,867	56,741	49,830	40,657	34,289	35,816	32,748
4,097	3,472	1,890	734	1,058	748	566	397	505
212,909	204,919	195,399	188,381	159,308	128,305	119,677	107,553	101,169
29,800	32,000	32,000	32,000	32,000	10,000	10,000	10,000	10,000
						2,250	2,250	2,250
1,372	1,367	1,350	1,261	1,158	1,091	1,049	1,025	3,739
$181,737	$171,552	$162,049	$155,120	$126,150	$117,214	$106,378	$ 94,278	$ 85,180
$ 82,332	$ 82,332	$ 80,339	$ 73,338	$ 46,311	$ 38,760	$ 34,260	$ 31,450	$ 29,097
		1,993	7,000	8,760	7,470	4,500	2,810	2,353
99,405	89,220	79,717	74,782	71,079	70,984	67,618	60,018	53,730
$181,737	$171,552	$162,049	$155,120	$126,150	$117,214	$106,378	$ 94,278	$ 85,180

(in thousands of dollars)

$140,587	$127,800	$110,291	$ 97,311	$ 82,215	$ 70,117	$ 67,177	$ 60,183	$ 50,893
6,105	13,183	18,162	13,708	15,921	12,640	3,792	7,493	9,684
(552)	(396)	(653)	(728)	(825)	(542)	(852)	(499)	(394)
146,140	140,587	127,800	110,291	97,311	82,215	70,117	67,177	60,183
67,197	59,933	53,550	47,481	41,558	35,828	31,361	27,435	24,529
8,543	7,650	7,013	6,779	6,699	6,249	5,234	4,410	3,289
(525)	(386)	(630)	(710)	(776)	(519)	(767)	(484)	(383)
75,215	67,197	59,933	53,550	47,481	41,558	35,828	31,361	27,435
$ 70,925	$ 73,390	$ 67,867	$ 56,741	$ 49,830	$ 40,657	$ 34,289	$ 35,816	$ 32,748

Product and Market Developments

When the Babcock & Wilcox partnership was formed in 1867 it had only one major product: a water-tube boiler that was a significant technological advance over the then prevalent fire-tube boiler.

Today, as a result of continuing product improvement, diversification and manufacturing integration, the Company has several divisions and subsidiaries supplying a wide range of products and services to American industry.

Growth through these methods continues. Some current examples are:

Bailey Meter Company's development of economical systems for automating power plants and industrial processes.

Diamond Power Specialty Corporation's extension of its foreign coverage by acquiring sales and manufacturing facilities abroad.

The Tubular Products Division's modernization of its steel-making facilities and diversification into new products such as extrusions and welding fittings.

The Refractories Division's new product development in high-temperature firebrick and ceramic fibers.

The Boiler Division's development of the Universal Pressure Boiler System, which makes possible a higher level of efficiency in generating electricity.

The Atomic Energy Division's development of a new concept in pressurized water reactors offering substantial savings in capital and total energy-producing costs of atomic power plants.

These are representative of the many new and improved products developed by all B&W operating divisions and subsidiaries. American industry is accelerating its search for better ways to serve its customers. B&W, which serves so much of industry, also is stepping up the pace of its research and development activity to insure that it can continue to keep ahead of the needs of its customers as well as to develop new products and markets.

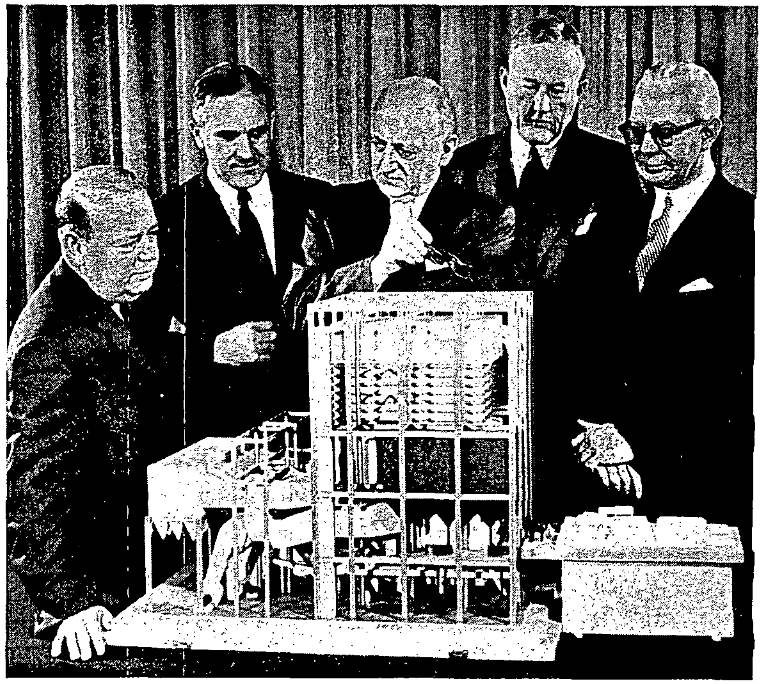
An advanced concept in steam generation: the Universal Pressure Boiler System

A major new product resulting from B&W's long-range research and development program is the Universal Pressure Boiler System.

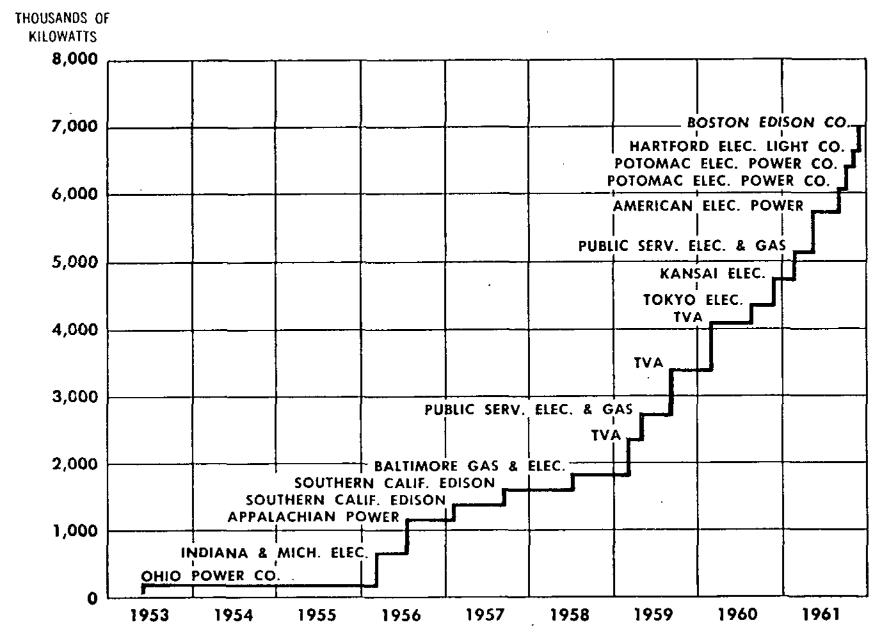
The UP Boiler makes it possible to operate at exceedingly high pressures and thus assures high levels of plant efficiency. It also makes possible significant savings in construction and operating costs for plants operating at lower pressures.

It is a true once-through boiler system – in effect, one long tube. Water enters at one end, absorbs heat, changes to steam, becomes superheated and then leaves the other end at the desired temperature and pressure. The boiler drum, the heaviest single component of the natural circulation boiler, has been eliminated.

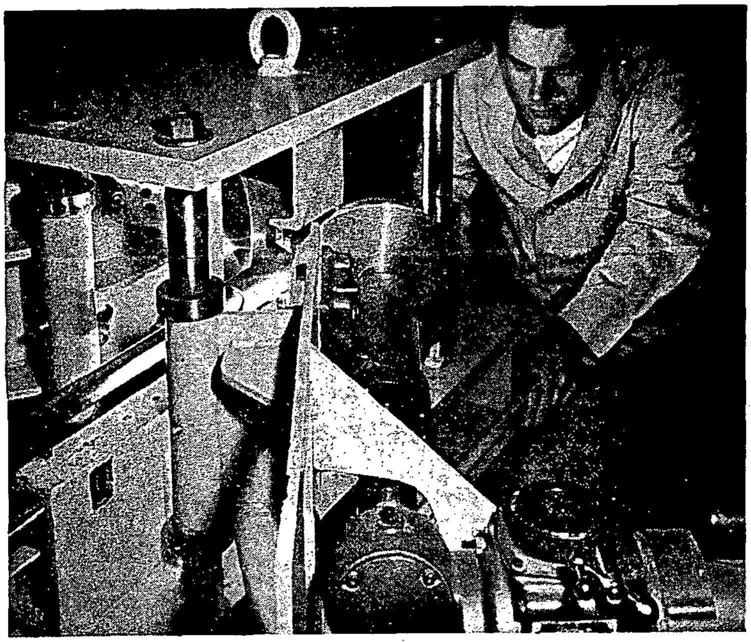
Eighteen UP Boilers, some more than 20 stories high, are installed or on order. The plants containing these boilers represent an investment by utility companies of approximately one billion dollars. When completed and operating at capacity, the plants will generate more than seven million kilowatts of electricity each hour – enough to supply the entire present electrical needs of all six New England states.



Boiler Division management is shown with a model of the Universal Pressure Boiler System, a B&W development which makes possible a higher level of efficiency in generating electricity.



Orders for Universal Pressure Boiler Systems now total 18 units or 7,010,000 kilowatts



Among the scores of research and development programs underway at B&W's Research Center is this fundamental study of the piercing of billets to make seamless tubes.

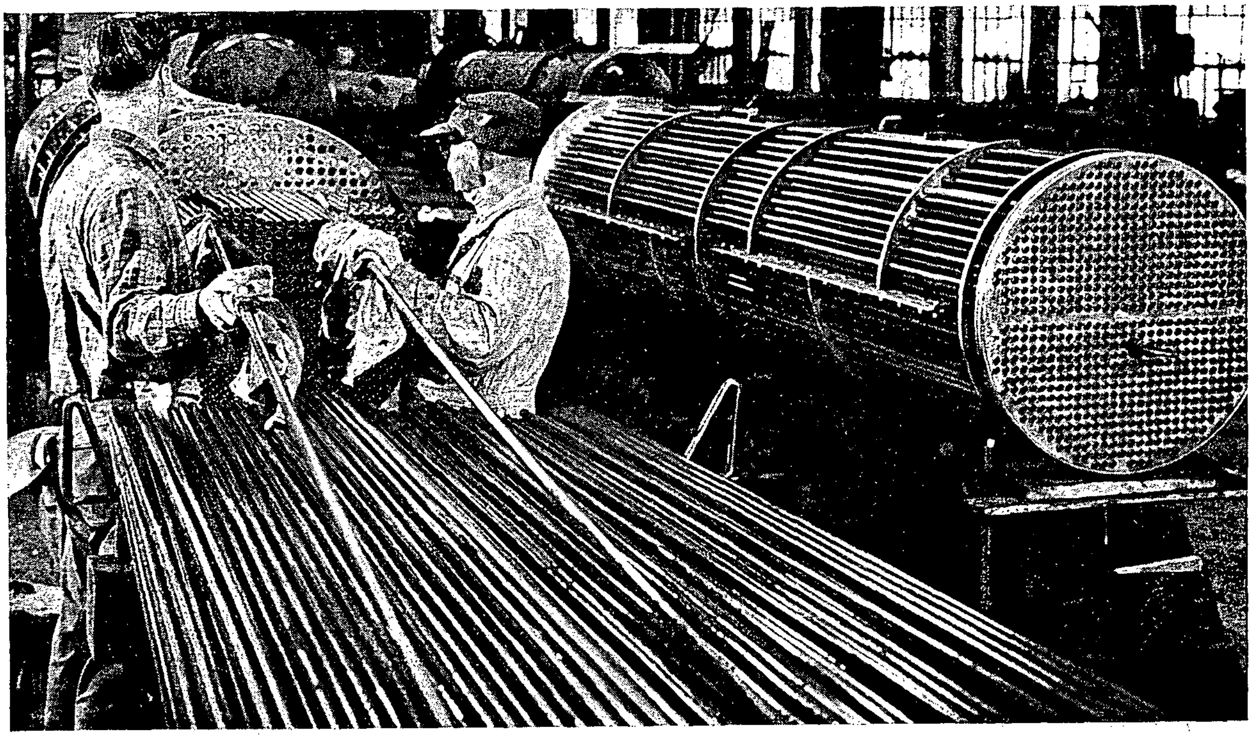
There are several major reasons for the success of this new product and for B&W's belief that it will receive increasing acceptance by the utility industry. The UP boiler costs the customer less in installed capital cost than other types of similar capacity. Fuel and maintenance costs also are measurably lower, and the UP requires less downtime for routine maintenance and repairs, thus assuring the customer of higher availability.

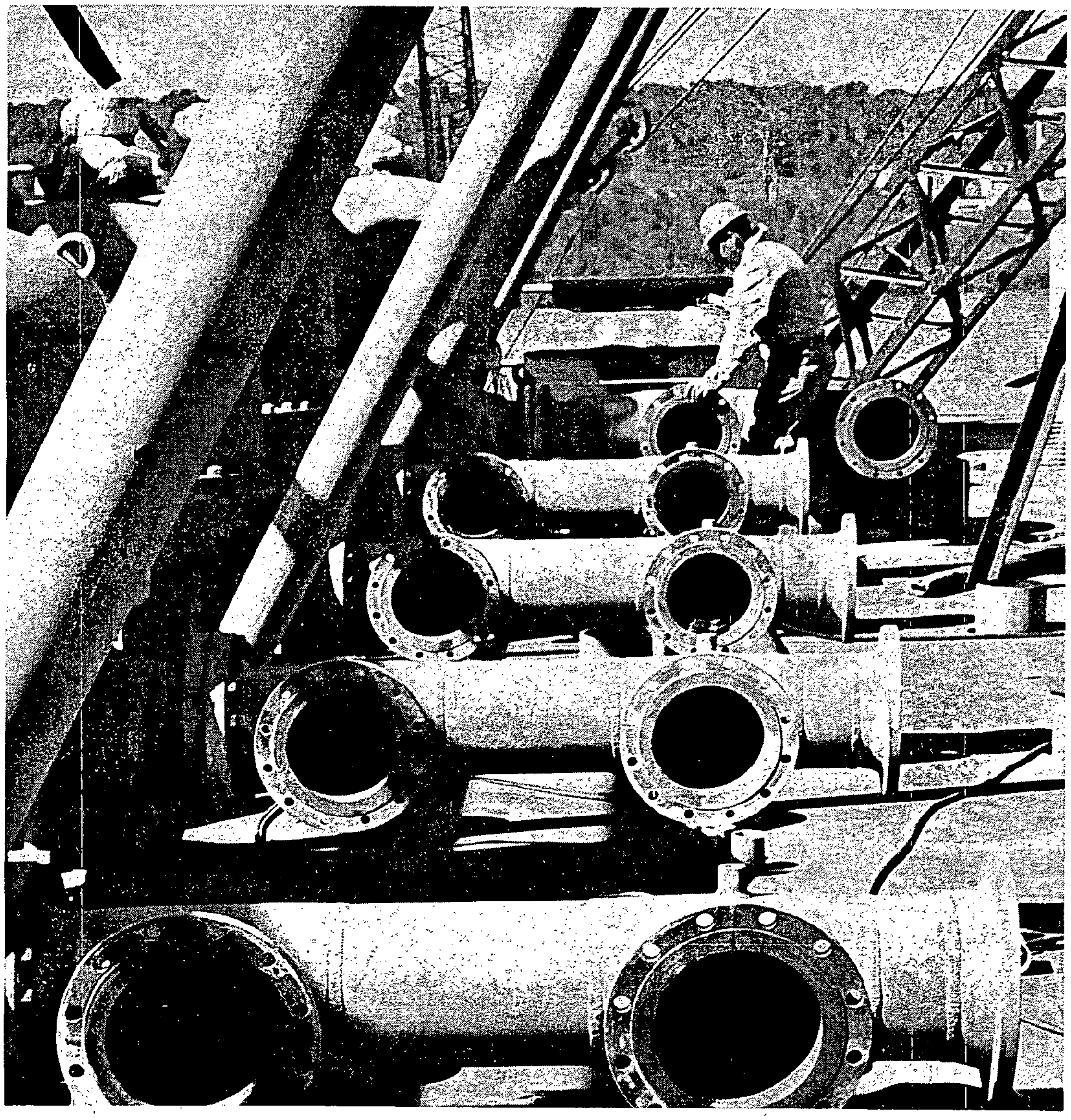
Sales of both B&W utility and industrial boilers in the last several years have been considerably less cyclical than formerly. Indications are that this trend in boiler sales will continue.

New tubing products expand markets

The Company has pioneered improvements in seamless mechanical and pressure tubing and in the processes to make them. Both fundamental and applied research studies on tubing are continuing under the

Your Company makes both welded and seamless tubing of carbon, alloy and stainless steel. Most industries use tubing in one form or another, such as in this heat exchanger for the petroleum industry.





Welding fittings are used in many different applications, such as in the piping systems on off-shore oil drilling platforms. The Company is continuing to improve its position in this field.

joint efforts of the Tubular Products and Research and Development Divisions.

Sales of B&W's welded heat exchanger and condenser tubing are increasing steadily. Lowered cost and improved quality have expanded markets for this tubing and the companion "Lectrosonic" welded hydraulic line tubing among manufacturers of machine tools, farm implements, road building equipment, materials handling equipment and hydraulic components and controls.

Since the new B&W welding fittings plant was placed in production in October 1960, the Company has steadily improved its position in this highly competitive field. As the only integrated manufacturer of fittings in the United States, B&W is able to supply matched components to meet exacting standards for all parts of carbon and alloy piping systems.

Bailey Meter advances automation

Bailey Meter has developed an advanced concept to automate power plants and industrial processes. Known as the Bailey 700 System, this electronic digital system performs data processing, performance computing, fault monitoring and program control.

The Bailey system offers a step-by-step approach to automation that requires only step-by-step commitment, yet provides benefits along the way. For example, an initial partial installation will provide many of the benefits of complete automation but with correspondingly smaller investment. Decision to take each succeeding step can be made after there is evidence that it would be justified both on the basis of cost and function.

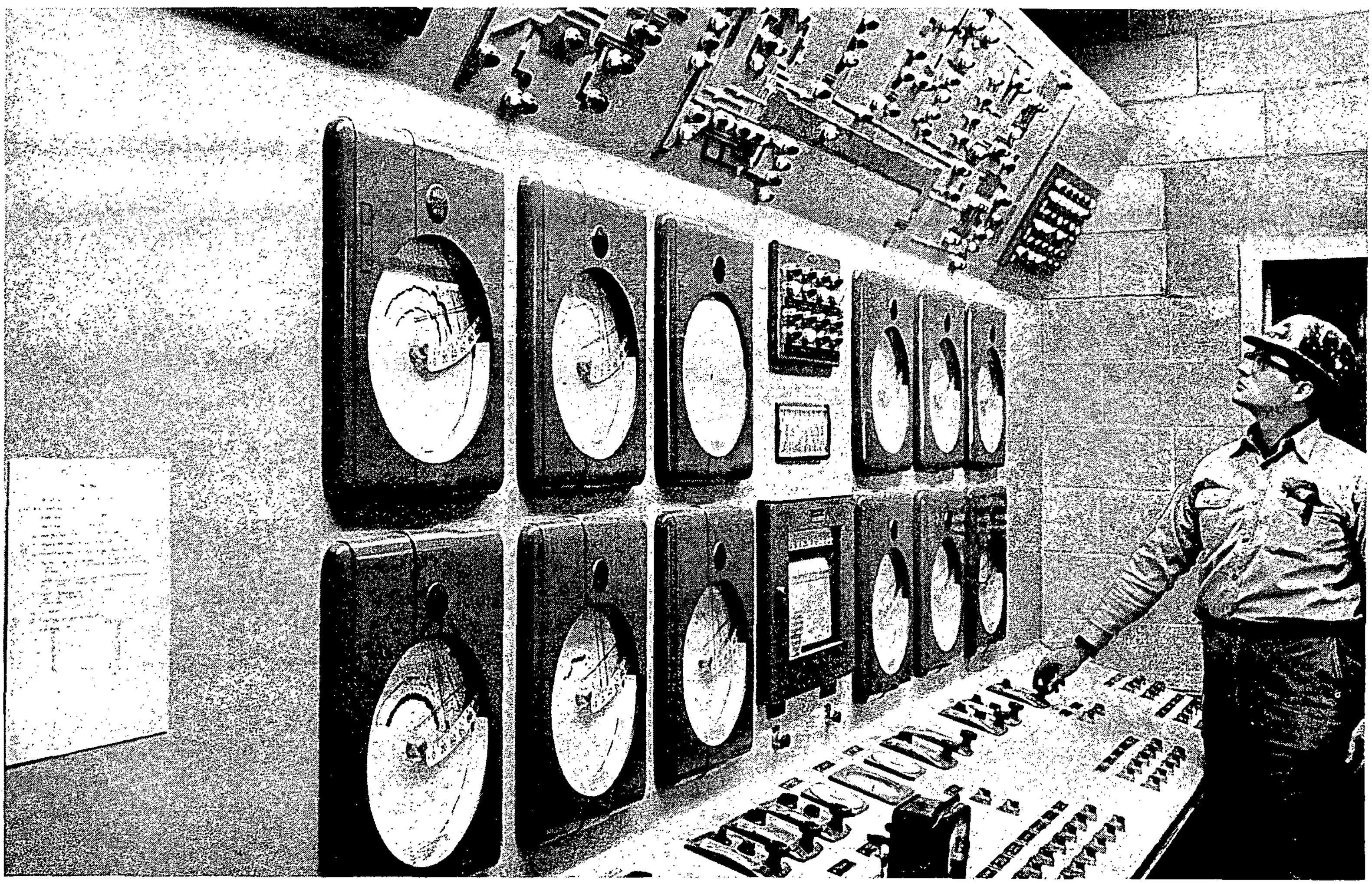
A typical application of the Bailey 700 System is on two B&W boilers recently installed for a southwestern utility. Here plant operators receive simplified data from a Bailey 750 information system. The ability to achieve uniform and safe furnace operation is increased by a Bailey 760 burner light-off system. More than 50 of the new 700 Systems have already been purchased.

Until about six years ago, most of Bailey Meter's products were built for electric power utilities. Today Bailey has developed much broader market opportunities. Sales have been steadily increasing to such industries as iron and steel, chemical, cement, non-ferrous metals, pulp and paper, and ceramics.

New refractory products introduced

New refractory products, coupled with efficient materials handling methods and improved processes for making brick semi-automatically, are increasing the profit potential of our Refractories Division.

Two recently introduced types of high-temperature firebrick have increased B&W's ability to meet



the refractory requirements of the steel and glass industries. These firebrick are being used in electric furnace roofs, reheating furnaces, metal mixers and other severe-service applications.

Another new product, a refractory castable, was designed for fast repair of blast furnace linings. The castable is sprayed rapidly in place by a pneumatic gun, dries rock hard, and quickly becomes strong when exposed to high temperatures. Substantial savings are made by reducing downtime for repairs to a few days and by extending the life of the lining.

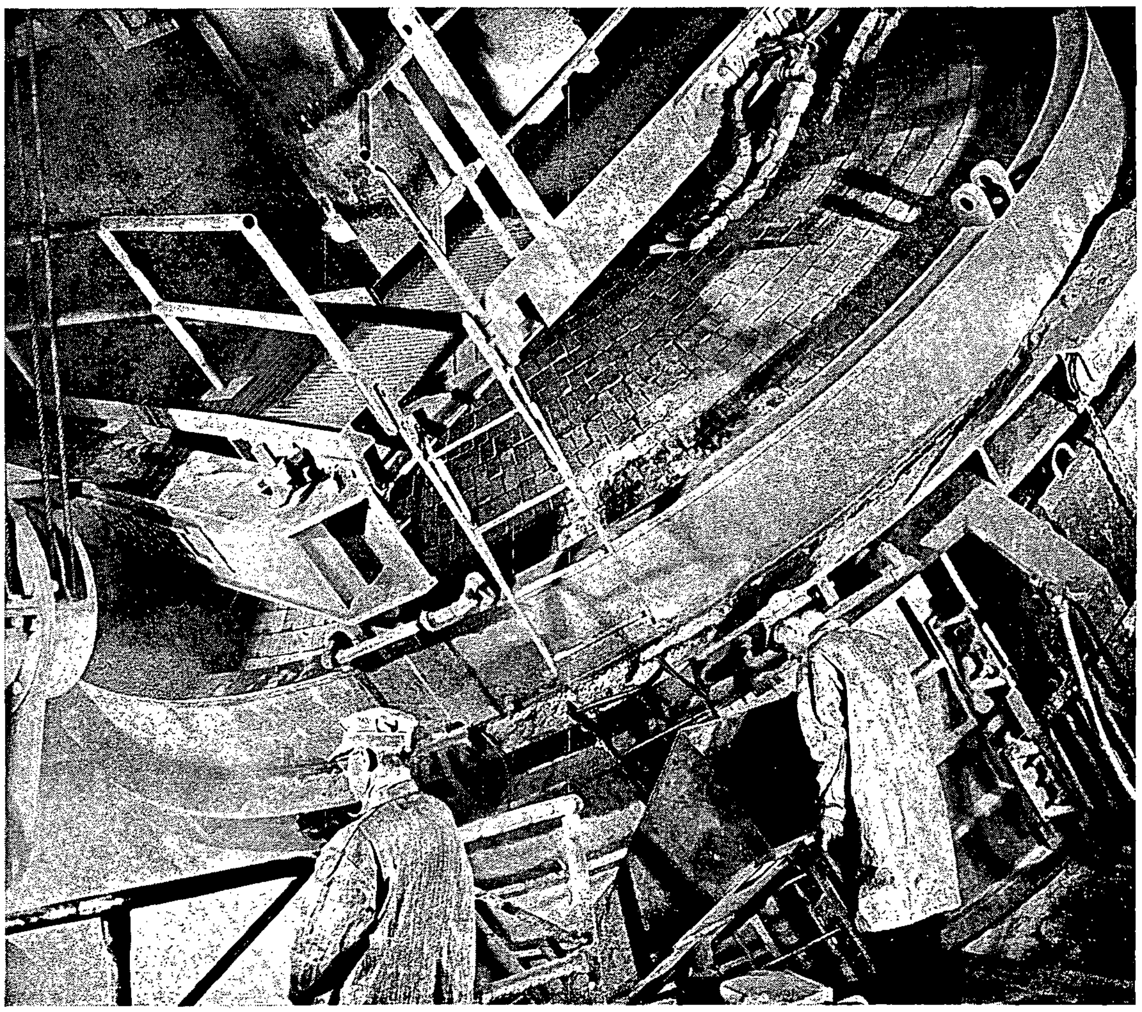
Diamond Power expands markets

During 1961 Diamond Power Specialty Corp. added more than ten new or improved products to its line. Also, it purchased outright a small English company which will manufacture Diamond Power products for Great Britain and the British Commonwealth as well as the European Continent. This foreign facility will help to increase Diamond Power's sales in these markets.

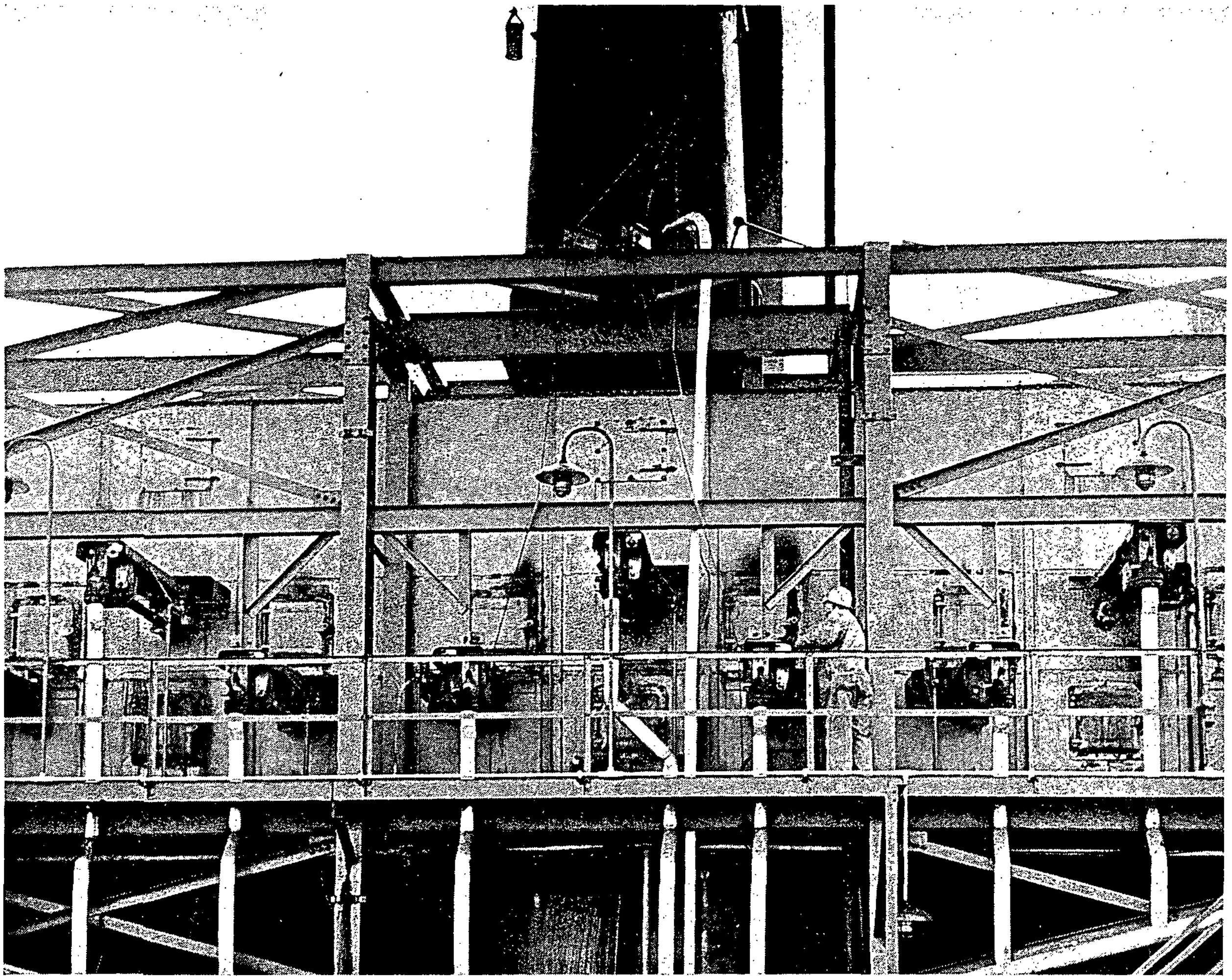
Principal new Diamond products include an extra-long retractable blower for cleaning boilers, new blowers that need to be lubricated only once a year under normal operating conditions, and new all-weather industrial television camera units.

It is now feasible for boiler designers to increase the width of furnaces because Diamond's new extra-long blowers, mounted on each side of a furnace, can economically clean an area up to 90 feet wide. The new minimum-lubrication blowers save maintenance time and cost for their users.

◀ *Bailey control system shown here simplifies operation in a modern cement plant.*



▶ *Durability of B&W's new high-temperature firebrick was proved by extensive use in the Tubular Products Division's electric-arc furnace at Koppel. This new type of brick is now widely used in the steel industry.*



Diamond Power's principal product line is boiler and furnace cleaning equipment, such as the seven retractable soot blowers installed in this refinery. Among Diamond Power's newer products is a combination industrial TV camera and power unit.

A combination industrial television camera and power unit, known as the Hawk 401, was placed on the market in August. This rugged and economical equipment gives the viewer a clear picture over a wide range of lighting conditions without frequent manual adjustments.

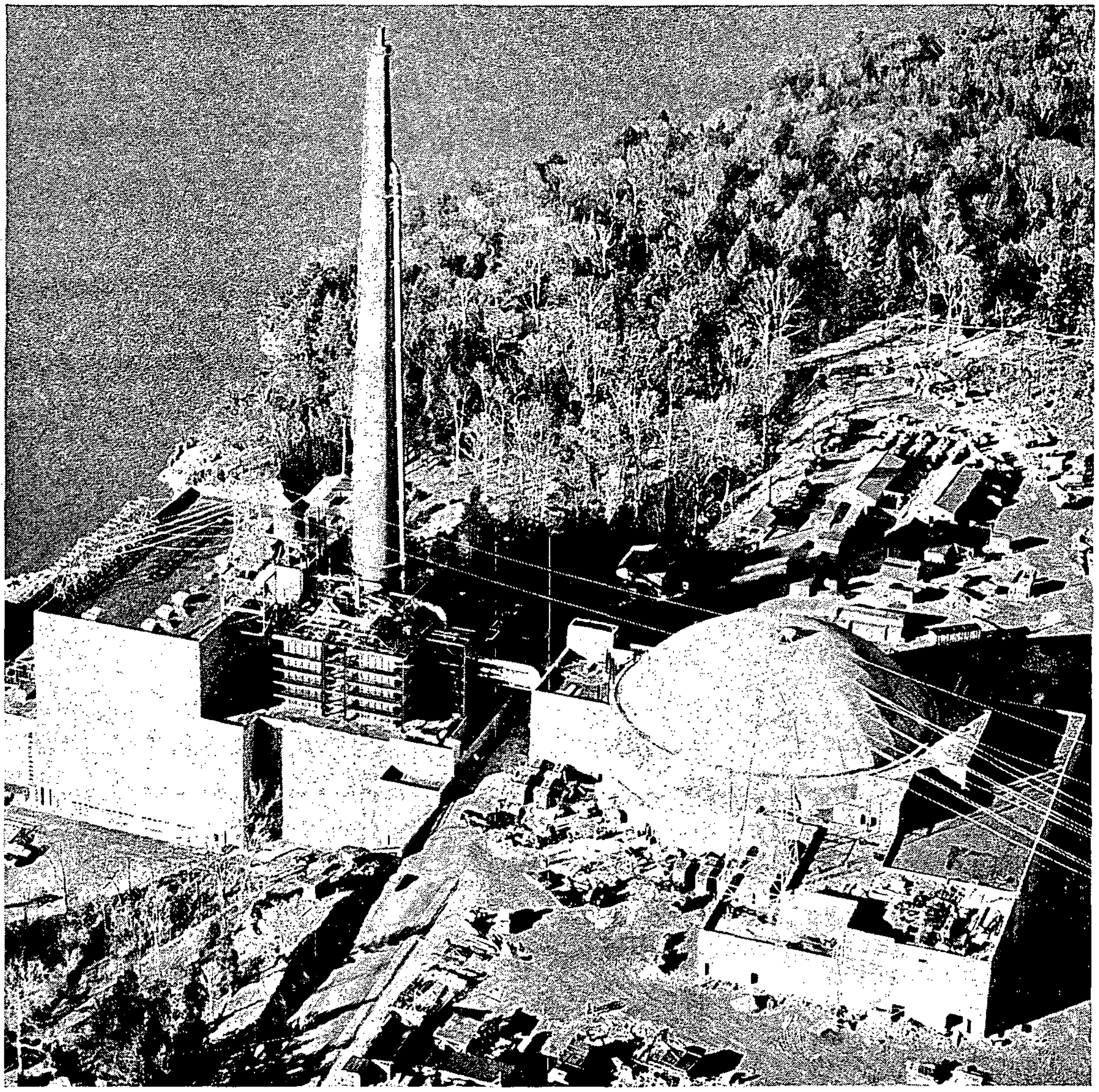
B&W a leader in nuclear energy

B&W continues to be one of the world leaders in the peaceful applications of nuclear energy. B&W offers its design and engineering know-how and sells nuclear reactors, components and fuel elements here and abroad. Major current projects include Consolidated Edison's Indian Point nuclear power plant and the Nuclear Ship Savannah, the world's first atom-powered merchant ship.

An historic milestone was reached on December 21, 1961, when the Savannah's atomic reactor went critical for the first time aboard ship. The entire system will undergo a period of testing before the vessel sets out on her sea trials.

The B&W-developed pressurized water reactor in the $100,000,000 Indian Point power station takes the place of a furnace in a conventional power plant. This reactor is the first to have a core which converts thorium into fissionable uranium, thus opening the way to utilizing the nation's thorium reserves for power production. The core includes in excess of a ton of highly enriched uranium oxide and close to 18 tons of thorium oxide.

The Company is now ready to build and sell nuclear reactors in the 400,000 kilowatt range employing a new concept of control developed by B&W scientists and engineers. Known as the Spectral Shift Control Reactor, it is an advanced design of the pressurized-water type. Control is achieved by varying the ratio of light to heavy water serving as moderator-coolant. This reactor will permit production of electricity at competitive prices in areas where fossil-fuel costs are high.



An important B&W nuclear energy project is Consolidated Edison's atomic power plant at Indian Point, N. Y. B&W is a major factor in the nuclear field, selling its design and engineering know-how in addition to equipment such as nuclear reactors, components and fuel elements.

The Babcock & Wilcox Company

General Offices: New York, N. Y.
Sales Offices in principal cities.

Facilities

Boiler Division

Alliance, Ohio
Barberton, Ohio
Brunswick, Georgia
Lynchburg, Virginia
Paris, Texas
West Point, Mississippi
Wilmington, North Carolina

Engineering Branch Offices
Cleveland, Ohio
St. Petersburg, Florida
Scranton, Pennsylvania

Atomic Energy Division

Lynchburg, Virginia

Tubular Products Division

Alliance, Ohio
Beaver Falls, Pennsylvania
Milwaukee, Wisconsin

Refractories Division

Augusta, Georgia

Research and Development Division

Alliance, Ohio

Bailey Meter Company

Cleveland, Ohio
Wickliffe, Ohio
Montreal, Canada

Diamond Power Specialty Corporation

Lancaster, Ohio
Southampton, England

Products

STEAM GENERATING EQUIPMENT

For Power, Industrial Process
and Heating, and for Marine Service
Water Tube Boilers
Water Cooled Furnaces
Superheaters and Reheaters
Economizers and Airheaters
Pulverized Coal Equipment
Cyclone Furnaces, Stokers
Oil, Gas and Multi-Fuel Burners
Metering and Control Apparatus
Power Plant Information Systems
and Performance Monitors
Soot Blowing Systems and Water Gauges

TUBULAR PRODUCTS

Stainless, Alloy, or Carbon Steel
Tubes and Pipe for all types of Pressure
and Mechanical Applications
Extrusions – Tubular and Solid Shapes
Welding Fittings and Flanges
Seamless Rolled Rings

ATOMIC ENERGY

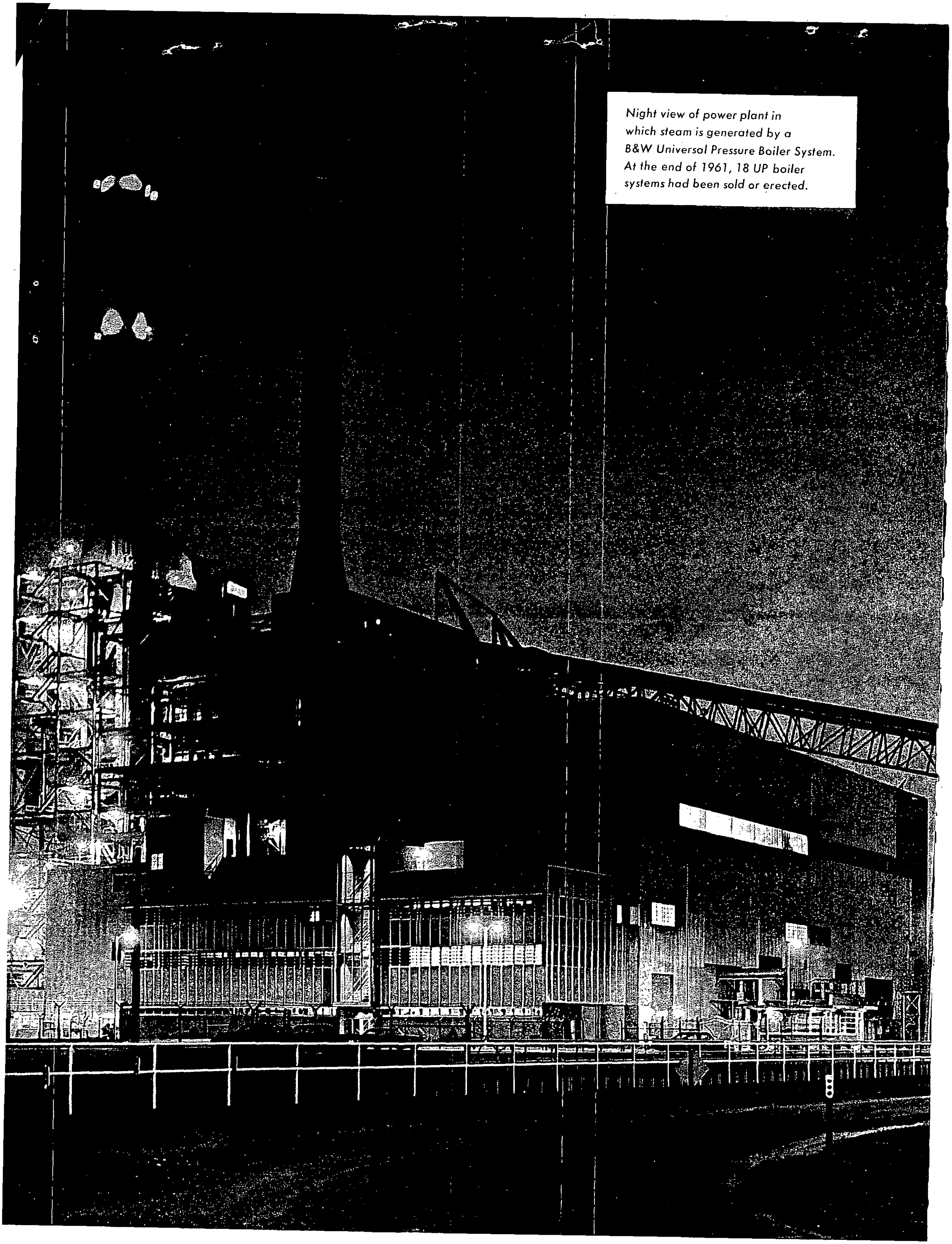
Complete Nuclear Power Systems
and Components
Reactors, Cores and Vessels
Fuel Elements
Heat Exchangers
Controls and Instrumentation
Test Reactors
Critical Experiments
Research and Economic Studies

REFRACTORIES

For Furnaces and Heat Treating Processes
Insulating Firebrick
Special Firebrick
Silicon Carbide
Plastics, Mortars, Castables
Ramming Mixes, Calcines
Ceramic Fiber

SPECIAL EQUIPMENT

Hollow Forgings
Alloy Castings
Pressure Vessels
Heat Exchangers
Automatic Process Control Systems
Recovery Processes for the Pulp Industry
Pulverizers for Cement Materials,
Rock Products and Ores
Industrial (Closed Circuit) Television



Night view of power plant in which steam is generated by a B&W Universal Pressure Boiler System. At the end of 1961, 18 UP boiler systems had been sold or erected.

1961

Annual Report