TID-8518(2) Book 2 CIVILIAN POWER REACTOR PROGRAM • PART III • 1959 • United States Atomic Energy Commission

ATOMIC ENERGY COMMISSION ★ UNITED STATES OF AMERICA ★

CIVILIAN POWER REACTOR PROGRAM

9/36

PART III

TID–8518(2)
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Status Report on Pressurized Water Reactors as of 1959

UNITED STATES ATOMIC ENERGY COMMISSION

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Status Report on Pressurized Water Reactors as of 1959

Published: 1960

UNITED STATES ATOMIC ENERGY COMMISSION

For sale by the Superintendent of Documents, U.S. Government Printing Office
Washington 25, D.C. - Price $1.25

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I. INTRODUCTION AND SUMMARY

A. INTRODUCTION

The U.S. Atomic Energy Commission has undertaken a study of various reactor concepts to develop a comprehensive plan for a 10-year civilian power reactor program.

This study reports the status of pressurized water reactors on a technical, economic, and operating experience basis.

B. CONCEPT DESCRIPTION AND PRESSURIZED WATER REACTOR OBJECTIVES

The pressurized water reactor concept by definition is one in which light water acts as both reactor coolant and moderator and is maintained at a pressure sufficiently high to keep the saturation temperature during normal operation above the bulk temperature of the coolant leaving the reactor. The primary coolant passes through a steam generator where it transfers heat to secondary water forming steam which drives the main turbine. Primary and secondary water are both in closed loops completely isolated from each other. The concept is in the category of thermal converters, whose primary objective is to achieve economic nuclear power in high energy cost areas by 1968 in central stations ranging in size from 25,000 to 200,000 KWe or larger.

Details of the concept and its objectives are discussed in Chapters II and III of the report.

C. GENERAL RESEARCH AND DEVELOPMENT PROGRAM

General research and development programs, both completed and underway, for the most significant problems in the pressurized water reactor concept are described in chapter IV. Because of the advanced status of PWR's, the development areas are fairly well defined. The many megawatt hours of successful operation of naval reactor plants, of the Army Package Power Reactor, and of the Shippingport Atomic Power Station are the best evidence of the established technology of the concept.

Some of the research and development programs discussed in Chapter IV are:

1. The measurement of nuclear constants.
2. Development of computational methods.
3. The measurement of basic reactor parameters and reactor critical and exponential experiments.
4. Analyses of multiregion multicycle loading.
5. Studies of reactor kinetics and safety analyses.
6. Development work on fuel materials, particularly UO_2, and on fuel fabrication techniques.
7. Development work on fuel cladding and core structural materials and on control rod materials.
8. Development work on materials for the primary system and on the effects of irradiation on these materials.
9. Analytical and experimental investigations of thermal and hydraulic problems.
10. Studies on primary coolant chemistry.
11. Development of improved designs for primary system components such as reactor vessel, pressurizers, steam generators, main coolant pumps and other pumps, valves, piping, etc.
12. Development of improved designs for auxiliary systems such as waste disposal, coolant purification, emergency cooling, fuel handling, ventilation, etc.

D. REACTOR DATA

Reactor data and information are given in Chapters V, VI, and VII in a Table of Reactor Data, in cross section views of various reactors, and in System Flow Diagrams. Reactors so described are: the Army Package Power Reactor, SM–1 (APPR–1), and its followers, SM–1A (APPR–1A) and SM–2 (APPR–1B); the NS Savannah (NMSR); the Belgian Thermal Reactor (BR–3); the Saxton Experimental Reactor; Shippingport (PWR–1) and the advanced Shippingport (PWR–2); Indian Point (CETR); Yankee (first core); the Westinghouse Electric Corp. 165 MWe Project; the AEC Design Study (APWR); and PM–1.

E. HISTORY AND OPERATING EXPERIENCE

1. Background

The first reactor to demonstrate operationally the technical feasibility of the pressurized water concept was the land-based prototype of the powerplant of the *Nautilus*. Construction of this prototype, the STR Mark I, began in 1951 at the National Reactor Testing Station in Idaho, and substantial amounts of power were first generated in May 1953. This plant was the forerunner of all the naval reactor plants discussed in Chapter VIII, Section A, and indeed of Shippingport, APPR, and of all of the pressurized water reactor plants now under construction or being designed in the United States.

2. Naval Reactor Plants

Table VIII–1 lists 36 naval vessels, constructed, under construction, or planned, that utilize pressurized water reactor plants.

3. The Army Package Power Reactor

Operating experience of the Army Package Power Reactor, SM–1, is described in Chapter VIII, section B. This was the first plant constructed under the Army nuclear power program. Full power operation of 20 MWt was attained in April 1957, and by April 1959, the plant had generated over 91,000 thermal megawatts.

The SM–1 has proved to be an extremely stable power plant, during both planned and unplanned load changes. There were some difficulties encountered with the moisture separator, with pumps (except for the primary system canned motor pump), and with lack of instrument reliability and high instrument noise levels. These problems have been rectified or alleviated. In general, measured radiation levels have been found to be reasonable and the shielding design to be conservative. Fission product activities in the primary coolant seem to indicate that some fuel cladding defects exist and that the magnitude of the defects is increasing with time. The feasibility of maintenance of primary system units after irradiation has been demonstrated operationally. The plant operation has met all design requirements.

4. Shippingport Atomic Power Station

The world's first full scale nuclear central power station, Shippingport, went into operation in December 1957. Its operating history is discussed in Chapter VIII, Section C. The Shippingport Atomic Power Station, in addition to being a successful part of the electric utility system of the Duquesne Light Co., is furnishing and has furnished valuable information for the design of the Army Package Power Reactors, the Belgian Thermal Reactor (BR–3), the Indian Point-Consolidated Edison plant, Yankee, the Saxton Experimental Reactor, and others of this type. As of June 1959, Shippingport had generated over 334,000 megawatt hours of electricity. Much of this power was generated while operating for long periods at the demands of the utility system, both as a base load or a peak load plant. At other times the plant was operated, independently of utility requirements, according to an established schedule for reactor testing. The seed fuel has operated for over 5,000 equivalent full power hours and the life, originally designed for 3,000 hours, is now estimated to be 6,000 equivalent full power hours.

Shippingport has been found to be extremely stable under transient load conditions, even under inadvertent transients that resulted in load changes exceeding design limits. A program of load varying tests has explored the reactor performance for design range transients. Xenon oscillations have been deliberately established to determine the stability of the core with respect to these oscillations.

There have been several mechanical component problems at Shippingport, including damage to the stator in one of the four main coolant pumps. Valves have leaked and leaks developed in several tubes of one of the four steam generators. These difficulties have all been rectified.

Radiation intensities within the reactor plant are lower than predicted. They approached steady-state levels after a few months of operation. Fission products in the primary coolant, although below design level, indicate the presence of one or more defected rods in the UO_2 blanket.

Shippingport as a whole is extremely stable and highly responsive to load changes. It is an excellent load peaking station. Operating experience to date indicates that it is simpler to operate than an equivalent coal-fired station.

5. Construction and Operation Schedules

Chapter IX summarizes the construction and operation schedules for the reactors described in the Reactor Data Table of Chapter V and also summarizes the operating histories of SM–1 and Shippingport.

F. PRESENT LIMITATIONS AND PROBLEMS

A discussion of present day limitations of pressurized water reactors and a description of some of the problem areas is given in Chapter XI. These are not inherent limitations but represent, rather, inadequacies of our present technology and point the direction for potential improvements. Some of the more significant of these problem areas are:

1. Reduction of hot channel factors.
2. The thermal design limitation of no bulk boiling in the core.
3. Limits on maximum fuel center temperature and on maximum heat flux.
4. Limits on size of reactor vessel and on the total number of heat transfer loops that may be added to a single reactor vessel.
5. Effects of irradiation damage to pressure vessel and other primary system materials.

G. BIBLIOGRAPHY

The final chapter of the report, Chapter XI, gives a fairly complete bibliography for Shippingport, SM–1, NS *Savannah*, Indian Point, Yankee, BR–3, and pressurized water reactors in general.

II. CONCEPT DESCRIPTION

The pressurized water reactor concept by definition is one in which the reactor coolant and moderator, light water, is maintained at sufficiently high pressure so that the bulk temperature of the coolant leaving the reactor is below the saturation temperature during normal operating conditions.

As the primary system operating pressure increases, the steam pressure in the secondary system may also be increased. The plant efficiency, however, reaches a value where only small incremental gains result from further increases in primary system pressure. The value of primary system pressure where this "plateau" occurs depends on the characteristics of the particular plant. For the plant conditions listed in Figure II–1, increasing the primary system pressure above 2,500 p.s.i.a. results in only small incremental improvement in plant efficiency.

The optimum primary system pressure is a function not only of plant efficiency, but also of equipment and component costs. The optimum pressure will vary from plant to plant. Historically, pressurized water reactors have been designed for a range of primary system pressures from about 1,200 to over 2,000 p.s.i.a. The maximum pressure is set by practical fabrication considerations as well as by optimum economics.

Present status does not permit bulk boiling in the hottest regions of the core but development of the pressurized water reactor is in this direction. To obtain high power density

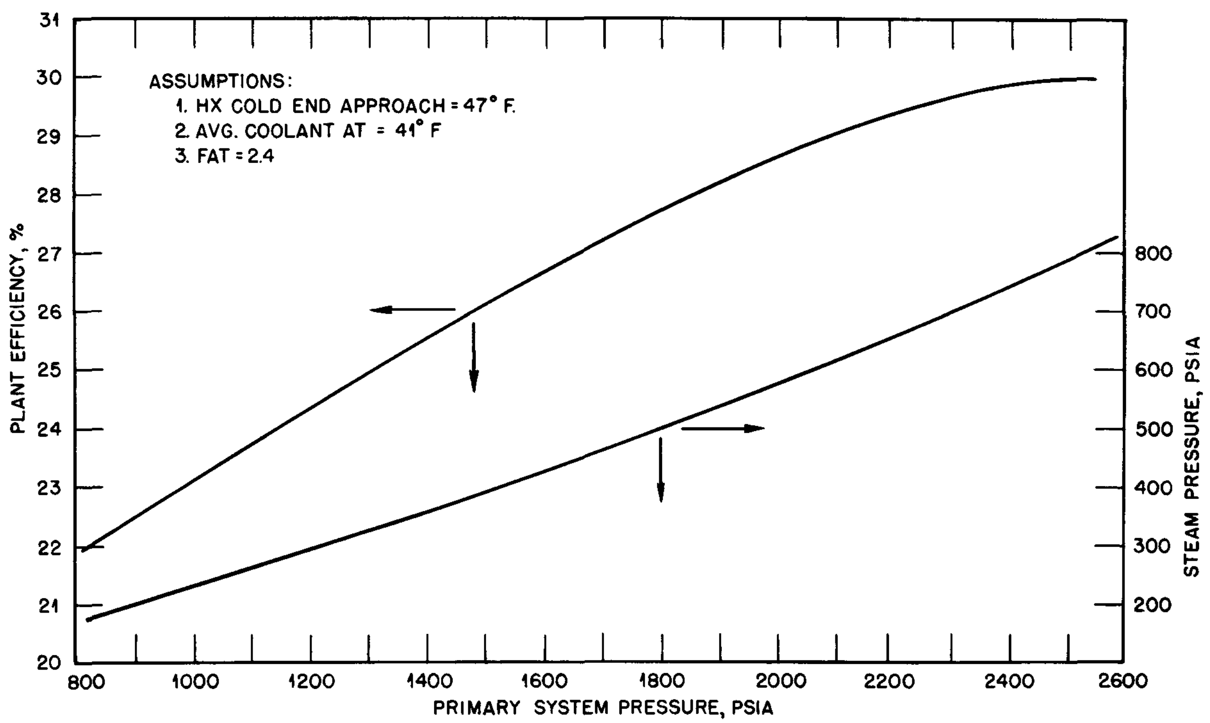

FIGURE II–1.—Effect of primary system pressure on plant efficiency and steam pressure.

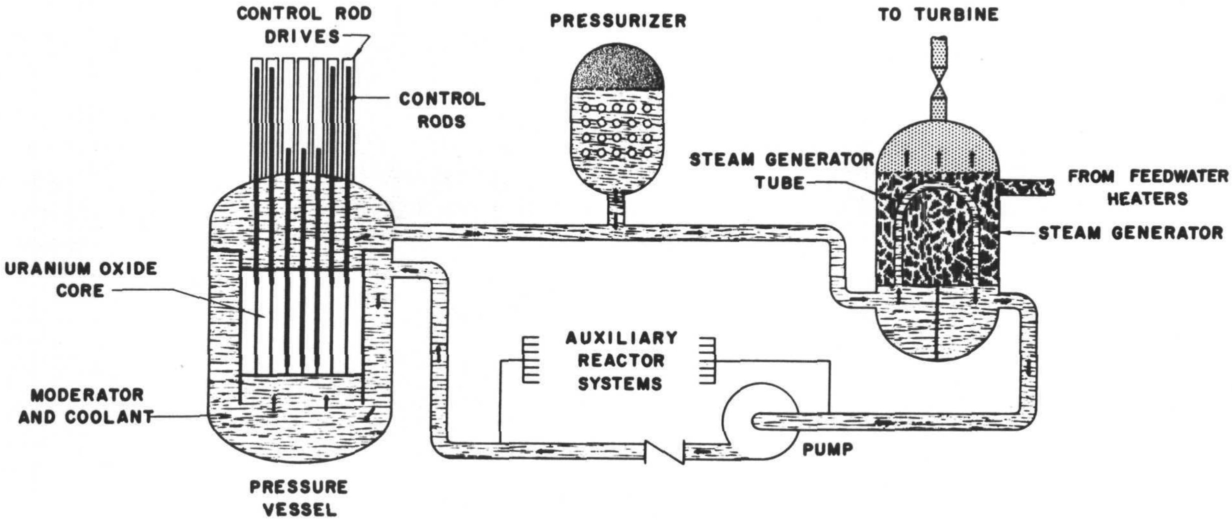

FIGURE II-2.—Pressurized water reactor. Basic flow diagram.

in the boiling water reactor, on the other hand, the trend is to greater subcooling so that these two reactor types seem to be approaching the same condition although from opposite ends. For this reason it has been suggested to classify these two concepts as water reactor types but for the purpose of this study the current status report will be limited to that of the pressurized water reactor.

Net steam is not generated in the core of the pressurized water reactor, the primary coolant being passed through a steam generator where it transfers its heat to a secondary working fluid forming steam which drives the main turbine. The primary and secondary coolant water are both in closed loops completely isolated from each other. A basic flow diagram of the pressurized water reactor system is shown in Figure II-2.

Since it is not feasible to operate a light water moderated and cooled reactor as a breeder or on natural uranium feed, this concept qualifies only in the thermal converter category.

Light water cooling and moderation classifies the pressurized water concept as a thermal reactor operating on a slow neutron energy spectrum.

Fuel is assembled in rods, plates, or pins and may be in the form of metal, cermets, carbides, etc., but the oxide, UO_2, seems favored at present because of its radiation resistance, high melting point, ease of fabrication, compatability with many cladding materials, and chemical stability in water. Plutonium and thorium can also be utilized as fuel and fertile material respectively.

As explained above, the cycle is indirect but both natural or forced primary coolant circulation can be used depending upon the power output and the physical size of the plant.

III. PRESSURIZED WATER REACTOR OBJECTIVES

The pressurized water reactor is in the category of thermal converters. The primary objective of thermal converters is to achieve economic nuclear power in high energy cost areas by 1968. The reactors in this category should be suitable for central station application in a size range of 25,000 to 200,000 KWe or larger.

IV. GENERAL RESEARCH AND DEVELOPMENT PROGRAM

A. SCOPE OF RESEARCH AND DEVELOPMENT PROGRAM

The established technology of the pressurized water reactor concept is best attested to by the successful operation of naval reactor plants, the Army Package Power Reactor, and Shippingport which have totaled many megawatt hours of operation. Yet there remain many areas where further research and development is required to develop the full potential of the pressurized water reactor concept. Because of the advanced status of PWR, the potential development areas are fairly well defined. The full realization of the PWR potential, though, requires the availability of a pressurized water reactor experimental facility which may be used as a testing tool. Presently, such a facility does not exist in the civilian power reactor program. The Saxton Experimental Reactor, now under design, will be the first facility of this type.

The general research and development program for pressurized water reactors as described in this chapter is not meant to be all-inclusive but to describe the most significant problems in the areas of reactor physics, materials, heat transfer and fluid flow, primary system chemistry, instrumentation and control, components, and auxiliary systems.

B. REACTOR PHYSICS

In surveying the present state of the reactor physics aspects of pressurized water reactor design, it is convenient to divide the field into three areas. These are: Reactor Statics, Nuclear Design, and Reactor Dynamics and Safety Analysis. The following paragraphs review prior work in the field and the present status.

1. Reactor Statics

The overall objective of this area of reactor physics is to generate the information and theory required for efficient evaluation and design of pressurized water reactors. Reactor statics may arbitrarily be divided into three sections. The first of these is nuclear physics and basic nuclear data, in which studies of nuclear cross sections, nuclear resonance phenomena, and basic physical models for neutron reactions and transport are considered. The second section encompasses computational methods. This area is concerned with making accurate and sufficient calculations of reactor parameters such as flux distributions, control requirements, kinetic parameters, etc. A final section is reactor experiments in which basic information is obtained and methods of analysis are verified.

a. Nuclear Physics and Basic Nuclear Data

This category includes both theory and experimental data on nuclear physics.

Work Completed

The basic theory of neutron slowing down and transport has already been developed. Absorption cross sections in the thermal neutron energy range have been measured and reported for all materials of interest in BNL-325 (2d edition). Measurements have been made over the entire energy range of interest for fissionable and major moderator and structural materials. Accuracy and resolution, however, leave much to be desired.

Work Underway

Some effort is directed at improving the accuracy and precision of already measured values. Values of γ and λ for U-233 are receiving extensive attention. In theoretical work, calculation of cross sections and understanding of theory needs improvement. Additional work on basic cross section information is desirable and measurement of inelastic scattering cross sections of reactor materials should be extended. More measurements of neutron absorption resonances in reactor materials should be made and better resolution obtained for the uranium and plutonium isotopes as well as for structural materials such as zirconium, aluminum, steel, nickel, cobalt, etc. Better information is needed on the higher isotopes of plutonium to determine more precisely the effects of their buildup on reactivity.

b. Computational Methods

The basic objective of this category is to obtain greater accuracy, rigor, and detail in calculations and at the same time perform calculations with more speed and efficiency.

Work Completed

The major contribution of the past in performing reactor calculations is the development of high speed digital computing machines. The development of these machines has allowed calculations to be performed which would be unheard of if done by hand or with desk calculators. A large number of one and two dimensional multigroup diffusion theory computer programs have been developed at many sites within this country. Several transport theory computer programs have also been developed. Simplified fuel depletion studies have been developed and are in use. A partial list of computer codes and their present status is given in Table IV–1.

Work Underway

The major effort in the field of computation methods at the present time is to obtain more detail and greater rigor. Monte Carlo codes are being developed at many AEC facilities. Core kinetic codes are also being developed at many laboratories. A three-dimensional fuel burnup code has been developed at Bettis. Codes which determine thermal neutron spectrum changes as a function of position are also presently under development.

The effectiveness of computational methods can be improved by more accurate input data. The development of simplified codes to study three dimensional burnup and more detailed consideration of transport theory in fuel depletion codes are desirable. Coordination of nuclear design calculations with thermal and hydraulic core considerations should be established so that single design codes which will perform all the necessary analytical work in core design are developed.

c. Reactor Experiments

This category includes measurement of basic reactor parameter data such as Doppler coefficients and ages; and reactor critical and exponential experiments.

Work Completed

The major amount of work applicable to pressurized water power reactors has been done on single region, water moderated, rod lattices at Bettis, BNL, Westinghouse APD, and Babcock & Wilcox. These experiments have been done with both U and UO_2 fuel rods of various diameters.

TABLE IV-1.—DIGITAL COMPUTER PROGRAM STATUS

Title	*Computer*	*Source*	*Description*	*Status*
ATBAC	IBM-704	Bettis Plant	Analysis of thermal transient conditions in a plate type, pressurized water core with channeled flow	Under study
BINTO	IBM-704	Bettis—Internuclear	Calculates steady state temperature distribution in two dimensional r-z geometry Accommodates one or two-pass coolant flow Requires power distributions as input	Being checked out
BURNOUT	IBM-704	General Motors	Part of the MAGNUM system for reactor analysis Determines fuel composition as a function of time using flux distribution calculated by the GNU II code	On hand.
CANDLE	IBM-704	Bettis Plant	One dimensional (slab, cylinder, sphere) four group, fuel depletion program using WANDA to compute neutron flux distribution and solving differential equations for fuel concentration at each spatial point and time interval	Operational.
CAP-1	IBM-704	WAPD	A modified one-group, uniform burnup lifetime calculation for uranium fueled reactors Calculates material number densities, macroscopic cross sections, poison concentrations and multiplication factor as a function of time Has enrichment search feature for determining initial enrichment for a prespecified lifetime	Operational
CAT	IBM-704	WAPD	Calculates the thermal conditions and flow distribution in the hot channel and the average channel coolant of a heterogeneous reactor core as a function of position and time and determines the location and time of the development of a burnout condition Input required is the heat flux as a function of position and time in the average channel and the hot channel and the mass flow rate as a function of time in the average channel	Operational
COFIT	IBM-704	Bettis Plant	Uses a least-squares technique to fit a cosine function to experimental data	Operational.
CURE	IBM-704	KAPL—Computer Usage Company	A two dimensional (r-o, r-z, X-Y) multigroup or few-group diffusion theory program for neutron flux and power distribution in a reactor core	On hand
DARED	IBM-704	WAPD	A data reduction program for assembling and correcting data from foil measurements at the WREC and fitting by the method of least squares to a function of the form A cos [B (x-c)] or J_0 [B (x-c)] This program incorporates COFIT and JOFIT as subroutines	Operational
DONATE	IBM-650	Bettis Plant	A calculation of thermal neutron cross sections and macroscopic reactor parameters averaged over a Wigner-Wilkins thermal neutron spectrum Similar to the SOFOCATE program on the IBM-704	Operational
EXFIT	IBM-704	Bettis Plant	Uses a method of least squares to fit an exponential function to experimental data	Operational
FOO 31	IBM-704	Bettis Plant	Uses a least squares technique to fit a function of the form A e^{Bx}+C to a given set of points	Operational.
BNU II	IBM-704	General Motors	One dimensional (slab, cylinder, sphere), multigroup (32), multiregion neutron diffusion code used in the MAGNUM system Computes spatial distribution of neutron flux and reactor power density and multiplication factor Optional output includes many group and region dependent macroscopic nuclear parameters	On hand
JOFIT	IBM-704	WAPD	Uses a least squares technique to fit a function of the form A J_0[B(x—c)] to experimental data	Operational
Lil Abner	IBM-650	Bettis Plant	One dimensional (slab, cylinder, sphere), multiregion, few group diffusion theory calculation of reactor multiplication factor and neutron flux distributions Similar to the WANDA program on the IBM-704	Operational
MAGNUM	IBM-704	General Motors	A nuclear code system for the IBM-704 consisting of the GNU II neutron diffusion code and other programs such as BURNOUT for carrying out related computations	On hand.
MUFT III	IBM-650	Bettis Plant	A multigroup computation of the few group constants (macroscopic cross section and diffusion coefficients) needed for input to flux distribution and reactivity programs.	Operational.
MUFT IV	IBM-704	Bettis Plant	An IBM-704 version of the MUFT III code described above.	Operational.

TABLE IV-1.—DIGITAL COMPUTER PROGRAM STATUS—Continued

Title	*Computer*	*Source*	*Description*	*Status*
MERLIN	IBM-704	WAPD	A synthesis of the CAP-1 and WANDA programs developed in Reactor Engineering for studying non-uniform burnup of uniformly loaded reactors and burnup of two and three region reactor cores with cycled fuel. Provides beginning and end of life fuel characteristics and flux distributions. Has search features for determining fuel enrichment necessary for a prespecified reactor lifetime or cycle time.	Operational.
PDQ	IBM-704	Bettis Plant	A two dimensional (r-z, X-Y) 2-4 group, diffusion theory program. Computes neutron flux distribution and multiplication factors for a multiregion reactor core.	Operational.
POLYPHEMUS	IBM-704	Bettis Plant	Monte Carlo study of neutron penetration through water slabs. Calculates neutron dose rates and dose buildup factors for several energies.	On hand.
PRESTO	IBM-704	WAPD	This is a program for the analysis of pressurizer transient operation. It is designed to provide information required in the development of a reactor control system simulation on the analogue computer.	Operational.
QED	IBM-704	Bettis Plant	A two dimensional, two group, diffusion theory calculation in rectangular coordinates. Furnishes flux distributions and region averaged fluxes as well as eigenvalue of η for the critical reactor.	Operational.
REP	IBM-704	AEC Computation Center. New York University.	Utilizes the Monte Carlo technique to calculate resonance escape probability in a heterogeneous lattice.	Operational.
SLOB 6	IBM-704	WAPD	A computation of material concentrations as a function of position and time in a slab exposed to an incident neutron flux. Designed to study burnup of control rods.	Operational.
SNG	IBM-704	Los Alamos	A few group or multigroup one dimensional (slab, cylinder, sphere) transport theory program utilizing the S_n approach to transport theory. Computes flux distributions and the eigenvalue of η for an arbitrarily high order of S_n procedure.	Operational.
SOCC	IBM-704	WAPD	This program is a combination of the CAP-1 program for fuel burnup with the SOFOCATE program for calculation of thermal neutron cross sections. It will provide macroscopic reactor cross sections, diffusion coefficients, material number densities and multiplication factor as a function of time for a low enriched, uranium fueled pressurized water reactor. An enrichment search feature will permit determination of the initial fuel enrichment required for a specified average power and core lifetime.	Being debugged.
SOFOCATE	IBM-704	Bettis Plant	A calculation of thermal neutron cross sections and macroscopic reactor parameters averaged over a Wigner-Wilkins thermal neutron spectrum. Similar to the DONATE program for the IBM-650.	Operational.
STIRRUP	IBM-704	WAPD	Calculates transient reactor power, delayed neutron concentration, reactivity, and average temperatures of fuel, clad and moderator in a pressurized water reactor with cylindrical fuel rods subjected to step or ramp reactivity perturbation.	Operational.
SWAP MU AND NU	IBM-704	Bettis Plant	Computes the uncollided particle flux as a function of the distance from a homogeneous cylinder containing a uniform, isotropic source distribution, assuming exponential attenuation of the particles both within the cylinder and within the attenuating slab or shells.	On hand.
TABLEX	IBM-704	WAPD	Produces a table of the functions $e^{\lambda t}$, $\lambda e^{\lambda t}$ and $1-e^{-\lambda t}$ for specified values of λ, initial time, final time and time increment.	Operational.
TURBO	IBM-704	Bettis Plant	A two-dimensional (r-Z, X-Y) 2-4 group diffusion theory program for calculating neutron flux distribution and fuel depletion as a function of time.	On hand.

TABLE IV-1.—DIGITAL COMPUTER PROGRAM STATUS—Continued

Title	Computer	Source	Description	Status
WANDA	IBM-704	Bettis Plant	A few (2-4) group, one-dimensional (slab, cylinder, sphere) diffusion theory program for calculating neutron flux and power distribution and multiplication factor. It provides region volumes and region averaged values of the neutron fluxes and sources. It has "search" features for determining critical reactor configurations or compositions.	Operational.
WB TSG 1	IBM-704	Bettis Plant	This program calculates thermal stress in cylinders with internal heat generation. It calculates the stresses from the temperature distribution determined from specified boundary conditions and internal heat source strength.	On hand.

Some of the pertinent critical experiments on which data exist are:

(1) Aluminum clad UO_2 of 1.3 percent enrichment.

(2) Stainless steel clad UO_2 of 2.7, 4.0, and 4.43 percent enrichment with water/uranium volume ratios of 2.19/1, 2.93/1 and 3.87/1.

(3) Two region critical experiments with stainless steel clad and UO_2 fuel of 4.43 percent and 2.7 percent enrichment with a water/uranium volume ratio of 2.93/1.

(4) $U^{235}O_2$-ThO_2 lattice with U-233 substituted for U-235 in selected elements.

Work Underway

Three region critical experiments with stainless steel clad UO_2 of 1.6 percent, 2.7 percent and 4.43 percent will be performed at Westinghouse APD. These experiments on multiregion lattices will be of great assistance in studying flat power distribution reactors. "Hot" critical and exponential experiments are being done or being considered at other facilities. A continuing program of low enrichment pressurized water critical experiments is being carried out at Bettis with emphasis on more basic measurements.

Among the additional experiments which should be undertaken are better measurements of Doppler coefficients in fissionable and fertile materials. More crtical experiment work should be done with plutonium fuel reactors. (A few measurements on plutonium bearing fuel rods in an exponential experiment at BNL have been completed.) More information on hot critical experiments should also be done—perhaps with irradiated fuel elements. More work in multiregion critical experiments is also needed with bigger cores and different structural materials. The effects of xenon oscillations require much additional study in large reactors. Present indications are that these oscillations will not be a source of concern in practical sizes of low enrichment cores because of the presence of a large Doppler coefficient.

2. Nuclear Design

The ultimate objective of the nuclear designer is to design a reactor which has a uniform power distribution, a long life, a simple and economical system of control, and great flexibility in response to load demands.

Work Completed

To date, concepts such as chemical control and burnable poisons have been confined to small, high enrichment reactors. Specifically Boiling Reactor Experiments II, III (Borax II, III), and Experimental Boiling Water Reactor (EBWR) have operated at power with boric acid reactivity control. SM-1 has operated with B_4C dispersed in the fuel as a burnable poison. In addition boric acid has been used for startup and shutdown control in Borax IV and will be employed in BR-3 and Yankee. Hence, some information is available and applicable to the design of low enrichment pressurized water power reactors. Methods of analysis for fuel cycling studies have been developed, but in small detail.

Work Underway

Studies on fuel cycling in the radial direction of the reactor have been made at many facilities. The value of multiregion, multicycle loading in reducing hot channel factors, peak-to-average burnup, and reactivity requirement has been firmly established. Preliminary studies of power shaping with control rods and fuel bearing control rod extensions are underway. Current experiments for Yankee show that 24—7.865-inch cruciform Ag-Cd-In control rods have a total worth of about 15 percent $\Delta K/K$. Work is underway on chemical control for power operation to reduce control rod requirements.

3. Reactor Dynamics and Safety Analysis

The basic objective of this field is to obtain reactors which are simple and easy to control, have maximum safety, and perform satisfactorily when integrated into a complete reactor power system.

Work Completed

Simplified methods of analyzing operational transients and reactor accidents have been developed at most reactor facilities. Experimental work has been performed by Phillips Petroleum in the Special Power Excursion Reactor Test (SPERT) experiments and by Argonne with the BORAX experiments. Both these programs are being carried out at the National Reactor Testing Station in Idaho. Detailed explanations of the various phenomena from the theoretical standpoint are not yet available.

Work Underway

Work is continuing on reactor safety experiments at Idaho, and model representations are being improved at various facilities. Methods of measuring reactor kinetic parameters are also being developed. In the immediate future better methods of analyzing reactor transients need to be developed. Improved analytical techniques are needed to represent more accurately the behavior of all of the components in the reactor system. Of particular importance is the treatment of the spatial dependence of reactor kinetics in the reactor core. As advanced digital computers become available, overall system studies will supplement work with analog computers. Meanwhile, analytical methods involving combinations of analog and digital computational systems should be developed.

Improved methods of measuring the kinetic characteristics of existing reactors require development. One such method would be to use autocorrelation techniques in observing the "noise" in nuclear instrumentation. This might allow determination of neutron lifetime, reactivity coefficients, transfer functions, and other parameters without interrupting the operation of a power reactor.

C. MATERIALS

1. Fuel

Work Completed

Extensive development work has been completed on UO_2 fuels. This work has demonstrated UO_2 to be dimensionally stable, easily fabricated, chemically compatible for water systems, and to possess high retentivity for fission gases. These properties have been investigated under severe radiation conditions both as reactor fuels and in in-pile test loops. Tests at burnups of $>$2,500 MWD/T have been performed on UO_2 rods at various heat fluxes. Recent loop experiments (CRVM) at Chalk River on the U irradiation of UO_2 platelets have demonstrated a burnup of 68,600 MWD/Metric Ton. Fission gas retention has been shown to be a function of UO_2 density. For high density material in the range of 93–95 percent of theoretical the experimental data on fission gas release shows considerable scatter but generally falls within the range of 0.1 percent to 32 percent released for exposures up to 12,000 MWD/T with center temperatures exceeding melting. For the design of the Yankee reactor a value of 14 percent fission gas release is assumed. Fuel rod sizes are limited by either center melting temperature or by burnout heat flux. For PWR reactors practical rod sizes based on current limitations are generally between 0.3 to 0.5-inch diameter.

The oxide form of fuel has been almost universally adopted for use in PWR. However, a relatively large number of metal and metal alloy fuels have been investigated for this service, among which are: U–6.8 w/o Zr; U–2 w/o Zr; U–7 w/o Zr—1½ w/o Nb; U–Mo; U_3Si.

The physical properties of UO_2 have been reasonably well established for types and grades currently manufactured. However, a good correlation of measured physical characteristics with fabricability has not yet been established. The preparation of UO_2 powders and the technology of sintered compacts up to 97 percent theoretical density have been demonstrated. Economic justification of compacts of >95 percent density for power reactor use is questionable with current technology.

Work Underway

Work is underway to extend the limits of firm data necessary for the design of reactor cores having greater burnup, improved thermal characteristics, and lower fabrication costs.

Work is continuing in the correlation of UO_2 physical properties with production methods and fuel performance in order to permit more reasonable and adequate specifications and product uniformity.

The major undesirable property of UO_2 for fuel use is its low thermal conductivity. Work is underway to investigate other compounds, uranium silicides for example, that show some promise of considerably better thermal conductivities and perhaps adequate other properties.

Work on improved pelleting techniques to permit increase in L/D ratios and to reduce "hour glassing" and hence eliminate grinding for dimensional control is underway. Alternate fabrication methods that show promise of reducing fuel element production costs are being investigated. In this area, development of both swaging and extrusion of UO_2 elements are being studied as promising alternatives to pelletization techniques. Irradiation programs to evaluate swaged elements have only recently gotten underway.

The development of fabrication techniques for alternative fuel elements shapes, such as cored pellets, annular elements, and flat plates, are being studied in an attempt to improve the thermal performance of fuel elements.

The current conservative design criteria limits fuel center temperature to less than the fuel melting point to assure that fuel element integrity is maintained. Criteria limits heat generation to ~15 KW/ft. However, recent experimental work indicates that operation at heat rates sufficiently high to cause center temperatures to exceed the melting point cause the formation of a hollow axial core in the fuel element by either a melting or sublimation mechanism, and that after the hollow core has developed no further deformation or change in fuel geometry occurs. Under these conditions experimental evidence indicates heat rates in excess of 15 KW/ft may be achieved provided that adequate provision is made in design for fission gas pressure buildup. Continuing investigations at Hanford and Chalk River are underway to test the feasibility of this concept.

The desire to reduce the amount of parasitic material within the core suggests the use of thinner cladding material that collapses on the fuel at operating temperature and pressure. Investigations are underway to determine the feasiblity of using collapsed cladding in lieu of the current free standing cladding designs. The feasibility is related to the controversial problem of thermal racheting and growth of fuel column under thermal cycling. Current data on whether or not this phenomenon occurs and on its magnitude are contradictory. Investigations of dished level pellets to alleviate the problem are also under study.

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2. Cladding Materials

Work Completed

The development of cladding materials for PWR types has in the past been primarily limited to the evaluation and development of either stainless steels or zirconium alloys. The feasibility of stainless steels from the standpoint of fabrication, corrosion and chemical compatibility, and resistance to radiation damage has been well established. The major disadvantage of stainless steel is its relatively high neutron absorption cross section (about 2.7 barns).

The desire to develop a lower cross section material for reactor use led to the extensive development program for zirconium and the development of the Zircaloys with a cross section of about 0.19 barns. Zirconium technology has been adequately developed to the point of practical utilization as a fuel cladding material. At the temperatures of the PWR its further utilization is hindered primarily by economic considerations and to a lesser degree by the progressive nature of the corrosion that can occur in case of a cladding defect in a water environment.

Work Underway

Development work currently in progress on zirconium materials is directed towards more economical production and fabrication methods, improved product uniformity, and better yields. In addition, some work is underway to develop new zirconium alloys, such as Zr–Nb or Zr–Nb–Sn, that promise better strength characteristics as elevated temperatures. Zr–5 percent Nb–2 percent Sn has an ultimate strength of 137,000 p.s.i. at 650° F. compared to 29,000 p.s.i for annealed Zircaloy 2 at the same temperature.

Aluminum alloys have found some acceptance in boiling water systems. Development work is underway to establish aluminum alloys that may be satisfactory for the more extreme environmental conditions of the PWR systems. Alloys of Al with some Ni such as ANL X–8001, show promise of improved corrosion properties as do alloys containing trace additives of Zr, Be, or Si. Suitable strength and corrosion characteristics at system temperatures have not been attained for this type of material and its application is restricted to low temperature systems (less than 450° F.). Work on sintered aluminum powder alloys currently in progress shows promise of meeting strength requirements, but corrosion resistance in high temperature water environments is poor relative to Zircaloy or stainless steels. Aluminum base materials in general require an acid water chemistry to reduce the corrosion rate. A pH of 3.5 is typical and may be achieved by phosphoric acid additions.

The potential of Fe-Al alloys as a substitute for stainless steel is under study. Strength characteristics at elevated temperatures are good, corrosion rates in water systems may be acceptable, and compatibility with UO_2 is probably satisfactory below 1,500° F. The cross section (1.8 to 2.2 barns) is lower than for stainless steel, but weldability is poorer. Unfortunately the higher the aluminum content, the poorer is the ductility of the alloy. Continuing work is necessary to establish radiation damage characteristics and to evaluate economics.

One way to extend core life and to operate at higher burnups is to introduce burnable poisons into the reactor core. A number of development programs are underway to investigate the feasibility of incorporating the burnable poisons in the cladding material. A typical system being considered is a dispersion of B_4C in zirconium alloys.

3. Control Rods

Work Completed

Hafnium has been developed as a control rod material. The desirability of hafnium for PWR control rods is well established and it will probably remain the preferred material from a technical standpoint. The continued use of hafnium, however, is limited by availability and relatively high cost. These items have led to investigations of alternate materials.

Work Underway

Alternate control rod materials being studied include boron dispersions, solid boron-carbide in sealed containers, rare earth dispersions, and Ag-In-Cd alloys. In general, the dispersion of boron in corrosion resistant materials has led to embrittlement. As a result the boron content is maintained below 2 percent by weight in stainless steel. (Boron steel containing 1.2 percent enriched boron costs about $70/lb in strip form). Solid boron-carbide in sealed or compartmented containers shows promise and is actively being investigated. Rare earth oxides have been used in SM–1 and are planned for use in the SM–2 reactor. These dispersions, however, are relatively expensive (Eu_2O_3 costs ~$1,000/lb) and require cladding.

Studies of silver base alloys have progressed to the stage where they have been specified as control rod materials for reactors going into operation in 1960. This material is only slightly more expensive than boron stainless steel but eliminates possible distortion as a result of helium formation upon irradiation. Development work is continuing on Ag-In-Cd and Ag-In-Cd-Sn to improve their relatively low creep strength and corrosion resistance. Limited work on the quaternary compound indicate improved mechanical and corrosion properties compared with the ternary while decreasing the control worth by only about 2–3 percent.

4. Primary System Materials

Work Completed

The primary system of PWR reactors consists of the reactor vessel, steam generator, pressurizer, main loop piping, and valves. Past work has demonstrated the feasibility of constructing these components of corrosion resistant stainless steels. Difficulties in maintaining components in a radioactive environment and the desire to maintain a low level of corrosion products in the circulating system suggested the development of these stainless steel systems. Initially systems were constructed of the stabilized types (347 or 348); subsequent development has shown that non-stabilized grades (304) having better weldability are adequate for use in the pure water environment of PWR's.

Work Underway

Work underway is directed toward proving the feasibility of substituting more economical materials in the primary system. Consideration is being given to the use of low alloy steels for such applications as heat exchanger tubing and main piping. Metallurgical and chemical studies are directed toward the reduction of stress corrosion for stainless steels, reduction of crack susceptibility on welding, and the development of high strength alloys that will permit the reduction of pipe wall thickness.

The effect of irradiation damage to pressure vessel materials is a subject of considerable concern. Irradiations of pressure vessel materials increases the yield strength, which may be beneficial in that it increases the factor of safety based on stress alone, but when plastic flow is needed to equalize stresses it may be undesirable. Experimental evidence indicates serious reduction in impact strength in low alloy steel for integrated fast fluxes as low as 5×10^{18} nvt at a temperature of 500° F. The reduction in impact strength is accompanied by an increase in transition temperature from brittle to ductile fracture. Continuing studies are being made on this behavior.

D. HEAT TRANSFER AND FLUID FLOW

Work Completed

Extensive analytical and experimental investigations have been performed on the thermal and hydraulic problems of PWRs. The following limits have been established based on a conservative interpretation of the work completed:

(a) Maximum ratio of heat flux to burnout flux of 1/1.5.

(b) Center fuel temperature less than fuel melting point.

(c) No bulk boiling in the core during steady state operation.

(d) Fuel rod size limit such that the fuel rod will remain intact during a loss of flow accident.

The most important summary of burnout data applicable to PWR's is presented in WAPD–188. The effects of geometry, fluid enthalpy and mass flow rate are correlated principally for 2,000 p.s.i.a. systems. Other investigators have examined the Bettis data and suggested alternate correlations to obtain greater accuracy on the effects of varying mass flow rates. The revised correlations are reported in KAPL–M–D1G–TD–1, KAPL–M–D1G–TD–2, and Transactions of the American Nuclear Society, 1959 Annual Meeting, paper 12–6, p. 95. Typical PWR operating conditions are illustrated by the Yankee plant full power burnout flux of 980,000 Btu/hr ft^2 as determined by the Bettis design correlation. The mass flow rate is 2,454,000 lbs/hr ft^2 and the bulk fluid enthalpy is 583 Btu/lb at the point of minimum burnout safety factor (2.19/1).

There is a lack of agreement on the effect of surface roughness and L/D ratio on the burnout heat flux. Experimental data from Argonne National Laboratory (ANL–4627) indicate that the surface roughness can influence the burnout flux but the L/D ratio has little effect. The reverse interpretation is obtained from Bettis experiments. Additional work is needed to resolve the problem.

Work Underway

Work is underway to extend the knowledge of thermal and hydraulic behavior sufficiently to justify relaxation of the current conservative design limitations. This work includes studies of transition boiling phenomena, the influence of flow instability and flow redistribution during bulk boiling on burnout, and improved effective thermal conductivity determinations for UO_2 under service conditions. Studies on transition boiling being performed by MSA Research Corp. are particularly important for evaluating the loss of coolant accident. Other aspects of thermal transients are a continuing subject of investigation. For example, studies of the effect of pellet size (fuel geometry) and core power density (KW/liter of coolant) on stored heat and hot channel exit enthalpy are being conducted to reduce the magnitude of thermal transients during loss of flow accidents.

E. PRIMARY COOLANT CHEMISTRY

Work Completed

A high degree of water purity is required in order to minimize corrosion, net radiolytic dissociation and the deposition of radioactive "crud" in the system. The deposition of "crud" on fuel elements or steam generator surfaces fouls the heat transfer and can lead to failures, and the settling of "crud" in stagnant areas of components can lead to severe maintenance difficulties.

Past work has done much to define the nature of the chemical problems and has developed ion exchange methods utilizing synthetic resins capable of maintaining water purity of one million ohm-cm in primary systems. Additional work has demonstrated the feasibility of the lithium-hydroxyl system for continuously maintaining pH control as well as for removing radioactive isotopes. Filtration and evaporation methods for removal of insoluble impurities have been investigated. Experience in pressurized water reactors, containing stainless steel or zirconium, has shown that the required coolant conditions for plant operation are low oxygen and neutral or alkaline water. Oxygen in the coolant is scavenged initially by the addition of hydrazine after which hydrogen in the coolant of approximately 25 ml (STP)/kg of coolant is maintained during operation. Concentration of dissolved oxygen is kept below 0.14 ppm. Additional experimental work is required.

The factors influencing net radiolysis of water have been established. The technology of external catalytic recombiners has been developed to the point that suitable and highly efficient (but not necessarily optimum) designs for the recovery of nominal flows of radiolytic gas are available.

Extensive research and development has been carried out on corrosion inhibition. Alkaline corrosion inhibitors have been developed that show considerable beneficial effect but not without accruing some disadvantages, especially in systems utilizing lithium. These disadvantages include:

(1) Lowered efficiency of performance or decontamination factor of ion exchange resins.

(2) Necessity to use lithium–7 both as the corrosion inhibitor and in the resin bed if tritium formation is to be avoided.

The problem of decontamination and removal of corrosion products from metal surfaces has received considerable study. These investigations have led to the multisolution (alkaline permanganate-ammonium citrate) process for decontamination of stainless steel.

Research and development efforts have shown that boric acid and possibly certain borate salts show promise for use as chemical poisons. Out-of-pile tests have shown no serious corrosion problem for stainless steel and Zircaloy under conditions of low dissolved oxygen. With high levels of boric acid and oxygen, as during refueling operations, the possibility of galvanic corrosion exists. Dissolved galvanic corrosion inhibitors such as potassium acid phosphates and potassium chromates have been found to be effective.

Work Underway

Consideration is being given to the development of thermal and radiation stable inorganic ion exchange resins that will permit essentially full flow clean-up of circulating streams without the thermal losses inherent in the temperature limitations (about 250° F.) of synthetic organic resins. Zirconium phosphate and oxide are examples of stable synthetic inorganic ion exchange materials. Zirconium phosphate is reported to exhibit properties typical of a polyfunctional weakly acid cation resin. The oxide is weakly basic and acts as an anion exchanger. Both materials exchange ions rapidly and reversibly and have substantial capacities.

Continuing work on gas recombination systems is directed toward the development of internal recombination catalysts. The internal recombination system has potential advantages, especially for systems which evolve very large quantities of gas.

Work continues on the effect of high temperature and radiation on corrosion inhibitors and on the possibility of a combined corrosion inhibitor and soluble poison.

Work on decontaminating agents is continuing in an effort to develop a single solution procedure that will permit reduction in solution waste volumes.

F. INSTRUMENTATION AND CONTROL

Work Completed

The inherent regulating characteristics of pressurized water reactor power plants did not require extensive work on the development of systems for the control of nuclear reactors. Circuits which use magnetic amplifiers were developed merely as a means to increase the overall reliability of the instrumentation and control systems.

At Shippingport, the FEDAL System was developed for determining the failure of any particular fuel element.

Control Rod Position Indicating Systems were developed which use variable inductance coupling between primary and secondary windings of a series of transformers. Motion of the control rod extension through these windings provides variable permeability.

Work Underway

Efforts are now being conducted to develop nuclear instrumentation circuitry for detection of boiling in the reactor core. This essentially consists of demodulating noise signals at the reading unit during steady state operation. The appearance of oscillatory signals during operation is an indication of boiling. As PWR cores become larger, the probability of xenon oscillation increases. Instrumentation, physics, and control rod programming studies are underway to provide solutions to this problem should it materialize.

G. COMPONENTS

1. Reactor Vessels

Work Completed

The development of reactor vessels for PWR's has been an evolutionary process since the origin of the concept in 1948. Developments of a manufacturing nature have permitted the construction of progressively larger and thicker walled vessels. Processes for cladding carbon steel and low alloy vessels with 300 series stainless steel have been improved. Both weld bead deposit and spot welded plate cladding is in use in addition to metallurgical bonded plate. Some developments in nozzle penetrations, vessel closures, welding, and heat treating have been made.

Work Underway

Work in progress is of the same nature as previous development. Studies are being carried out on the following subjects:

(*a*) Better gasketed closures to avoid seal welds.
(*b*) Nozzle penetration design.
(*c*) Methods of installing and removing vessel stud-bolting.
(*d*) Use of high strength materials.
(*e*) The use of "multilayer" construction.

2. Pressurizers

Work Completed

Since the completion of the basic development on the STR project, there has been little development on pressurizers. The technique of design and fabrication is similar to that of the reactor vessel with the possible exception of the electrical immersion heaters employed for pressure control. A minor change was the location of the heaters in wells so that they could be replaced readily.

3. Steam Generators

Work Completed

The steam generator unit consists of the tube sheet, tubing, shell, and steam separating unit. A major development task was concerned with the selection of tube material. Extensive work was done; many materials were tested for general corrosion, stress-corrosion, and erosion-corrosion behavior in high temperature water. Although various materials proved adequate from the standpoint of their behavior in the corroding environment, stainless steel was ultimately selected because of its availability and ease of fabrication. Several different design configurations were developed by various suppliers. These include the straight-through tube design, the horizontal U-tube U-shell design, and the vertical U-tube configuration. All of these designs have performed successfully. Major development effort has been devoted to the tube to tube-sheet welding and the cladding of carbon steel water boxes and tube sheets with stainless steel.

Work Underway

Present efforts are being devoted to increasing the size of the unit and developing refined manufacturing processes. Adequate recirculation rates and moisture separation on the secondary steam side in large size units are areas of concern and investigation. Examples of manufacturing development are semiautomatic tube to tube sheet welding and stainless steel clad application. Investigation of other materials of construction is being pursued in experimental steam generators (notably in the A1W reactor). Materials under study include carbon steel, low alloy steels, and high alloys.

4. Pumps

Work Completed

The major development work on hermetically sealed canned motor pumps (which provide zero leakage) was performed on the STR reactor project. Work completed on the canned pump includes the development of bearings capable of operating in high pressure, high temperature water, fabrication of thin stainless steel or Inconel liners for rotor and stator, cooling of motor windings, and high pressure electrical terminal seals. The size and efficiency of the canned pump has been steadily increased, culminating in the PWR Shippingport pumps.

Work Underway

Studies are now underway to increase the size and performance of the canned motor pump further. Pumps of 35,000 GPM capacity are now capable of being constructed. It is expected that a pump rated in excess of 40,000 GPM can be produced with increased manufacturing facilities. Improvements in bearings, electrical insulation, and motor cooling are under study. For example, alumina is a possible substitute for the present Graphitar bearings. Potting compounds injected into the stator windings will improve motor cooling and permit the use of higher voltages. Alternatives to the canned motor pump are being studied by many suppliers in an effort to reduce costs. A shaft sealed pump of either break-down seal type or mechanical face seal type is economically attractive. The problem is to produce a seal of high reliability and with low seal water and leak-off flow rates. In the past shaft seal boiler circulating pumps have exhibited poor reliability unless high seal water flow rates were maintained. It will require a major effort with no assurance of success to develop a low leakage seal of 4- to 5-inch diameter size for 2,000 p.s.i.a. service.

5. Valves

Work Completed

The development of a complete line of valves suitable for PWR operation was performed during the STR project. The main achievement was the construction of hermetically sealed, hydraulically operated stainless steel valves for all primary system services. Recent efforts have been devoted to the design of controlled leakage valves which permit conventional construction and conventional motor operators. This simplification eliminates the need for 4-way pilot valves which in the past have lacked reliability. A major construction development has been the acceptance of cast material for valve bodies and other pressure parts, thereby effecting appreciable cost reductions.

Work Underway

Virtually no development work is in progress. A normal engineering refinement and size increase is taking place. Design revisions are being made as required by operating experience.

6. Piping

Work Completed

Piping for PWR systems has been of the seamless (extruded or drawn) or hollow forged and bored type. The welding of heavy wall stainless steel piping has been extensively developed. Weldability and weld cracking were found to be very sensitive to material composition. A recent development is centrifugally cast and hydroforged piping for large sizes. This manufacturing process has promise of reducing cost and increasing the pipe size (diameter) that can be fabricated.

Work Underway

Present efforts are a continuation of the previously mentioned developments.

H. AUXILIARY SYSTEMS

Work Completed

Auxiliary systems for liquid and gaseous waste disposal, coolant purification, emergency cooling, fuel handling, ventilation, etc., are necessary to all reactor systems. Safe design criteria and feasible engineering techniques to meet these requirements have evolved from past work on both military and commercial reactors.

Work Underway

The relatively small amount of work currently in progress on auxiliary systems is confined to the development of specific adaptations of known technology to individual reactor system requirements and to methods of simplifying these systems. Analysis of operating experience is providing valuable information to justify the simplification or elimination of auxiliary systems in future reactors. For example, present designs do not include steam bypass systems for dumping steam, as is the case with Shippingport. In addition, automatic control has been found unnecessary because of the large negative temperature coefficient of reactivity.

V. REACTOR DATA TABLE

Pertinent data on pressurized water reactors that have been built, are under construction, or under design are presented in the following table. Actual operating data are specified for SM-1 (APPR-1) and PWR-1 (Shippingport).

REACTOR

Plant designation	SM-1 (APPR-1) Army package power reactor	SM-2 (APPR-1A)	SM-2 (APPR-1B)	NMSR (NS *Savannah*)	BR-3 Belgian thermal reactor	Saxton Experimental reactor
Sponsored by	U.S. Dept. of Defense and U.S. A.E.C.	U.S. Dept. of Defense and U.S. A.E.C.	U.S. Dept. of Defense and U.S. A.E.C.	U.S. Maritime Adm. and U.S. A.E.C.	CEEN (Centre d'Etude de l'Energie Nucleaire).	General Public Utilities and Westinghouse Electric Corp.
Reactor designed by	Alco Products, Inc.	Alco Products, Inc.	Alco Products, Inc.	The Babcock & Wilcox Co.	Westinghouse Electric Corp.	Westinghouse Electric Corp.
Plant location	Fort Belvoir, Va.	Fort Greely, Alaska.		NS *Savannah*	Mol, Belgium	Saxton, Pa.
Purpose	Remote base power supply.	Remote base power and space heat.	Remote base power supply.	Ship propulsion	Power and training.	Experimental
Present status	Operating	Under construction.	Preliminary design.	Under construction.	Under construction.	Preliminary design.
Start of construction	Oct. 1955	June 4, 1958		April 1957	Nov. 1957	Sept. 1959
Initial criticality	April 8, 1957	1960		Feb. 1960	July 1960	Nov. 1961
Full power operation	April 20, 1957	1960		Mar. 1960	Aug. 1960	Dec. 1961
Thermal power rating of total plant, KWt.	10,000	20,000	27,600	69,000 max. continuous.	40,900	20,000
Thermal power rating of reactor, equilibrium core, KWt.	10,000	20,200	27,600	74,000	40,900	20,000
Thermal power rating of fossil fired superheater, KWt.	None	None	None	None	None	None
Gross electrical power rating, KWe.	2,000		7,760	Not applicable	11,500	~5,500
Net electrical power rating, KWe.	1,855	1640+38.3 x 10^6 Btu/hr space heat.	7,060	22,000 SHP		~5,000
Net KWe/gross KWe, %	92.7					91
Net plant thermal efficiency, %.	18.6		26	24		25
Total steam generated, lb/hr.	34,000	70,000	103,358	261,500	154,400	73,000
Steam flow to turbine, lb/hr.	32,000		102,708	206,550	153,800	73,000
Fuel element:						
Shape	Plate		Plate	Rod	Rod	Rod
Overall dimensions, plate or rod, inches.	23.0 x 2.76 x 0.03		22.0 x 2.938 x 0.040	length 69	56 x 0.343	40 x 0.4
Fuel material	UO_2 in SS		UO_2 in SS	UO_2	Sintered UO_2	Sintered UO_2
Cladding material	304 SS		347 SS	304 SS	348 SS	SS
Cladding thickness, inches.	0.005		0.005	0.035	0.0205	0.018
Fuel isotope weight % of total fuel.	93.2% of U-235	Highly enriched U-235.	93.4% of U-235	inner, 4.2; outer, 4.6.	inner, 3.7; outer, 4.4	~3.9
Average conversion ratio (equilibrium condition).				0.40	~0.40	~0.40
Core inventory	22.5 Kg U-235	22.37 Kg U-235	36.2 Kg U-235	7,100 Kg	2,280 Kg	1,270 Kg
Average burnup, equilibrium condition, MWD/metric ton of U and Th metal.						
Guaranteed	None		None	None	None	None
Design	5,475		9,855	7,352	5,940	7,100

DATA TABLE

PWR-1 Shippingport (at 4,500 EFPH)	PWR-2 Shippingport (modified)	Indian Point (CETR)	Yankee (first core)	165 MW project	AEC design study (APWR)	PM-1
U S A E C and Duquesne Light Co	U S A E C and Duquesne Light Co	Consolidated Edison Co of New York, Inc	Yankee Atomic Electric Co		U S A E C	U S Dept of Defense and U S A E C
Westinghouse Electric Corp	Westinghouse Electric Corp	The Babcock & Wilcox Co	Westinghouse Electric Corp	Westinghouse Electric Corp	Combustion Engineering, Stone & Webster	The Martin Co
Shippingport, Pa	Shippingport Pa	Buchanan N Y	Rowe, Mass			Sundance, Wyo
Power	Power	Power	Power	Power		Power and 7×10^6 Btu/hr heat
Operating	Preliminary design	Under construction	Under construction	Design	Design	Preliminary design
April 1955		Dec 1956	Nov 1957	Aug 1959		June 1960
Dec 2 1957		April 1961	Oct 1960	Mar 1963		July 1961
Dec 23, 1957			Dec 1960	June 1963		Aug 1961
231,000	530 000	795 000	392,000	615,000	685 000	9,000–10,000
231,000	530 000	585 000	392,000	615,000	685,000	9,000–10 000
None	None	210 000	None	None	None	None
67,000	150,000	275,000	116,000	180,000	248,120	1,000
60,000		255 000	107,000	165,000	236,420	1,000
89 5		92 8	92 2	91 6	95 3	
26		32 5	27	27	34 5	12 5 to 14 3
860,000		2,200,000	1,840,000	2,350,000	2,865,300	30,000 to 37,000
850,000		2,200,000	1,840,000	2 350,000	2,564,910	21,000 to 27,000
Blanket—rod Seed—plate		Rod	Rod	Rod	Rod	Tubular
Rod—64 7 L x 0 411 D Plate—70 75 x 2 05 x 0 069		138 x 5 7 sq 14 x 14 array	90 x 0 340	102 x 0 45	114 x 0 45	0 25 to 0 5
Blanket—UO_2 Seed—U-Zr	Sintered UO_2	UO_2-ThO_2 pellets	UO_2	Sintered UO_2	Sintered UO_2	UO_2 in SS
Zircaloy-2		304 SS+Boron	348 SS	SS	347 SS	348 SS
Rod, 0 027 Plate, 0 015		0 0205	0 021	~0 028	0 022	0 006
Rod, Nat'l U Plate, 92 3		6 4 'as metal" fully enriched	3 4	inner, 2 6 middle 2 7 outer, 2 8	inner, 2 6 outer, 3 4	Highly enriched
Blanket, 1 13 Overall, 0 56		0 46	0 546	0 7	0 68	
Blanket 12800 Kg UO_2 Seed 75 Kg U-235		1,100 Kg U-235 and 16,100 Kg Th as metal	20,400 Kg	39,300 Kg	53,000 Kg	28 Kg U-235
						None
None		None	None	14,700		33-38% of U-235 atoms
Blanket 7300		20,400	7,830	14,700	13,000	

REACTOR DATA

Plant designation	SM-1 (APPR-1) Army package power reactor	SM-2 (APPR-1A)	SM-2 (APPR-1B)	NMSR (NS *Savannah*)	BR-3 Belgian thermal reactor	Saxton Experimental reactor
Lifetime per cycle, full power hours.	13,000		8,500	8,250	7,000	11,000
Max central fuel temperature, °F.	742		655	3310, nominal channel, inner pass.	(11.65 kwt per ft of fuel rod).	(12.13 kwt per ft of fuel rod).
Max heat flux, 10^3 Btu/hr-ft^2.	224		630	275	442	410
Overall hot channel factors (nuclear and mechanical):						
$F_Q=\frac{\text{Max heat flux in core}}{\text{Ave heat flux in core}}$	4		3.68	4.42	4.69	4.69
$F_{\Delta T}=\frac{\text{Max change in enthalpy of coolant through core.}}{\text{Ave change in enthalpy of coolant through core.}}$	3.04			1.83	3.48	3.48
Core active heat transfer Flow percent of total included in FΔT.	100			90–Yes	90–No	90–No
Dimensions of core, ft	1.83 x 1.70 x 1.70		Height, 1.83 Diam., 1.87	Height, 5.5 Diam., 5.19	Height, 2.79 Diam., 4.67	Height, 3.33 Diam., 2.87
Reactor power density, kwt/ft^3 of core.	2,080			617	1,430	930
kwt/liter of core	73.4			21.8	50.5	32.8
Reactor specific power, kwt/kg of U and Th.	445	904	763	9.72	17.95	15.7
$\frac{\text{Vol } H_2O}{\text{Vol U+Th metal}}$	4.4			1.32 (+ steel)	4.48	5.5
Coolant inlet temp., °F	431.6	423	500	495.6	491	494
Coolant outlet temp., °F	450	446	525.7	520.4	515	516
Operating pressure, p.s.i.a.	1,200	1,200	2,000	1,750	2,000	2,000
Coolant flow rate, lb/hr	1.66 x 10^6		3.17 x 10^6	8 x 10^6	5 x 10^6	2.8 x 10^6
Ave coolant velocity through core, ft/sec.	4.1			Outer, 9.29 Inner, 8.40	7.7	3.6
Reactor vessel material	A-212 C & B			SA 212-GrB	302 GrB	302 GrB
Reactor vessel I.D., inches.	48		55	98	58	58
Reactor vessel thickness, including cladding, in.	2.75			6.61	4.484	4.5
clad thickness, in	0.25			0.109	0.11	0.25
Primary coolant piping nominal diam.	12 in		14 in. Sch 140 16 in. Sch 140	12.56 in. I.D	12 x 1.375 in. 16 x 1.750 in.	
Number of loops	1		1	2	2	1
Number of operating main coolant pumps per loop.	1 (1 standby)		1	2	1	1
Main cooland pump type	Canned motor		Canned motor constant speed.	Canned motor	Canned motor single stage.	Canned motor
Pump head at rated flow, p.s.i.	11		(116 ft)	66.5	50	30
Flow rate per pump, gpm.	4,000		8,000	5,500	6,250	7,000
Number of steam generators.	1		1	2	1	1

TABLE—Continued

PWR-1 Shippingport (at 4,500 EFPH)	PWR-2 Shippingport (modified)	Indian Point (CETR)	Yankee (first core)	165 MW project	AEC design study (APWR)	PM-1
Blanket—19,200 Seed—6,000.	10,000	14,400	10,000	7,500	24,100	17,500.
636 at metal surface.		4,800	(11.65 kwt per ft of fuel rod.)	(13.3 kwt per ft of fuel rod.)	(10.1 kwt per ft of fuel rod.)	
Blanket—425 Seed—520		470	446	384	296	100 to 250.
Blanket, 7.5		3.57	5.17	3.7	3.61	
Blanket, 2.6		2.38	3.36	2.4	2.46	
		87-No	90-No	90-No		
Height, 6 Diam., 6.8		Height, 8.5 Diam., 6.5	Height, 7.69 Diam., 6.31	Height, 8.5 Diam., 8	Height, 9.5 Diam., 8.2	Height, ≈2.5. Diam., 2.
1,060		2,150	1,560	1,440	1,370	1,100 to 1,300.
37.4		75.8	55	50.8	48.4	38.8 to 45.8.
Blanket, 11.7		34	19.2	15.65	12.9	321 to 357.
Blanket 3 Seed 1.1		1.83	3.00	2.8	2.3	
486	504	485	495	498	582	420 to 490.
514	556	517	532	544	618	460 to 540.
1,800	2,000	1,500	2,000	2,150	2,000	1,200 to 2,000.
22.6 x 10^6	29 x 10^6	53.9 x 10^6	37.8 x 10^6	39.1 x 10^6	45 x 10^6	
Seed, 20 Blanket, 9.5		19	14	8.5	12	
302 GrB		SA 212-GrB	302 GrB	302 GrB	302 GrB	
109		117	108	126	126	30 to 36.
8.375		7.06	8.0	9.25	9.0	
0.25		0.109	0.11	0.25	0.25	
15 in. I.D.		24 in. O.D. Sch 160.	20 in. O.D. 24 in. O.D.	24, 20 in.	22 I.D. x 2.85 in.	6 in.
4 (3 operating)	4	4	4	4	4	1.
1		2	1	1	1	1.
Canned motor	Canned motor	Canned motor vertical.	Canned motor single section.	Canned motor single stage vertical.	Controlled leakage centrifugal.	
	123	125	79.1		45	(25 to 40 ft.)
19,000	18,000	16,000	23,700	24,500	32,000	1,200 to 2,200.
4		4	4	4	4	1.

REACTOR DATA

Plant designation	SM-1 (APPR-1) Army package power reactor	SM-2 (APPR-1A)	SM-2 (APPR-1B)	NMSR (NS *Savannah*)	BR-3 Belgian thermal reactor	Saxton Experimental reactor
Steam generator (secondary) outlet conditions						
Full load pressure, p.s i a	200	200	600	474	525	500
Full load temp , ° F	407	382	486	462	472	476
Steam flow per generator, lb/hr	34,000		103,500	130,750	154,400	73,000 ..
Assumed effective heat transfer surface, % of total				100%	99 3%.	
LMTD, ° F	57		78		29 4	27.5.
Steam turbine						
Full load throttle pressure, p s i a	190		590	450	510	300
Full load throttle temp , ° F	404		484	456	469	
Turbine back pressure, inches Hg A	1 5		1 5	1 5	1 5	
Description	8 stage with reduction gear and single casing			High and low pressure stages	2 section	
Means of reactor control	7 rods, burnable poisons, fuel followers	7 rods, europium oxide burnable boron poison	Rods, EU_2O_3 in SS, burnable poison, B^{10} in fuel	Rods	Rods, boron solution for shut down	
Primary water chemistry						
pH (room temp)	6 4 to 9 0			neutral	10	neutral
Oxygen control initial				hydrazine	hydrazine	hydrazine
Oxygen control operation				hydrogen	hydrogen	hydrogen

TABLE—Continued

PWR-1 Shippingport (at 4,500 EFPH)	PWR-2 Shippingport (modified)	Indian Point (CETR)	Yankee (first core)	165 MW project	AEC design study (APWR)	PM-1
500	600	420	500	500	1,050	200 to 500
467	486	449	467	467	550 6	381 to 467
215,000		550,000	460,000	587,500	716,325	30,000 to 37,000
		13,773 ft ^{2} 100%	94%			~90%
31		48 5	44 4	50	48	30 to 100
485		340	465	465	1,000	200 to 500
463		1,000	460	460	545	381 to 467
1 5		1 5	1 5	1 5	1 5	6 to 11.5
			Tandem compound, double flow	Tandem compound, double flow, two cylinder, non-reheat.	3 cylinder single shaft	9,000 RPM, condensing type
Rods		21 Hf rods, boron in fuel clad, 18 (max) fixed shim blades with 80% B^{10} boric acid for start up	Ag-In-Cd rods, chemical poison for shutdown	Ag-In-Cd rods, chemical poison for shutdown	Rods and burnable poison	Rods and burnable poison
10 hydrazine	neutral hydrazine		neutral hydrazine	neutral hydrazine	neutral hydrazine	
hydrogen	hydrogen		hydrogen	hydrogen	hydrogen	

VI. CROSS SECTION VIEWS OF REACTORS

This section contains the following reactor cross section views:

Figure VI–1.—SM–1 (APPR–1) Reactor Core, Pressure Vessel and Shield (from report APAE–23).

Figure VI–2.—SM–2 (APPR–1B) Vertical Section, Reactor Vessel (from report APAE Memo–197).

Figure VI–3.—SM–2 (APPR–1B) Plan Section, Reactor Vessel (from report APAE Memo–197).

Figure VI–4.—Reactor Vessel Internals, NS *Savannah*.

Figure VI–5.—BR–3 Reactor Vessel and Internals.

Figure VI–6.—Saxton Reactor Vessel; Longitudinal Cross Section.

Figure VI–7.—Saxton Experimental Reactor Core Cross Section.

Figure VI–8.—Shippingport PWR–1 Pressure Vessel and Core.

Figure VI–9.—Shippingport PWR–1 Pressure Vessel Cross Section Above Core.

Figure VI–10.—Indian Point, Reactor Vessel & Internals.

Figure VI–11.—Indian Point, Layout of Core for Consolidated Reactor Showing Placement of Fuel Elements, Thermal Shields, and Reactor Vessel Wall.

Figure VI–12.—Yankee Reactor Vessel and Internals.

Figure VI–13.—165 MW Project Reactor Vessel.

Figure VI–14.—APWR—Perspective of Reactor (from TID–8502 (Part 3)).

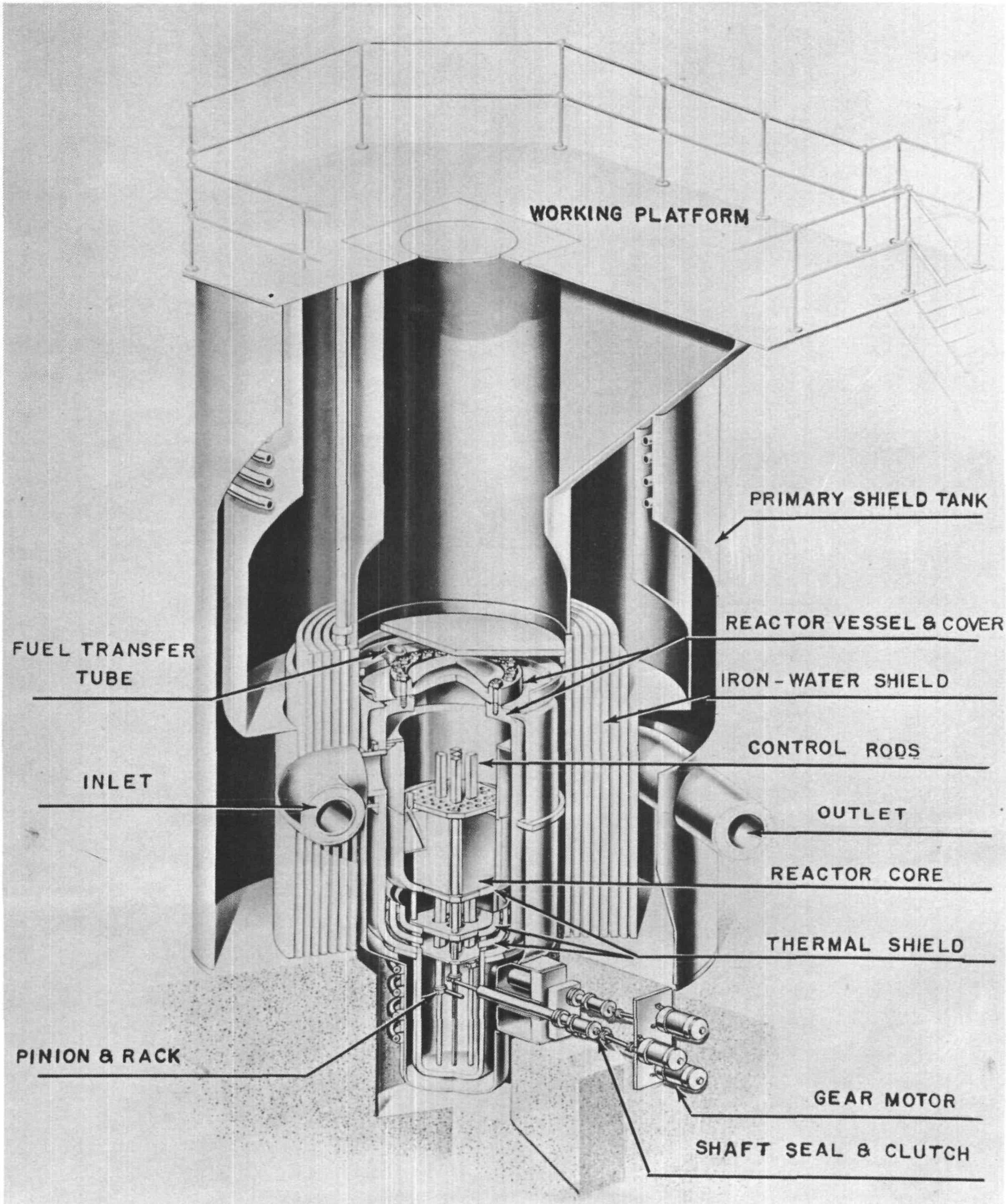


FIGURE VI-1.—SM-1 (APPR-1) reactor core, pressure vessel, and shield (from report APAE-23).

540997 O—60——3

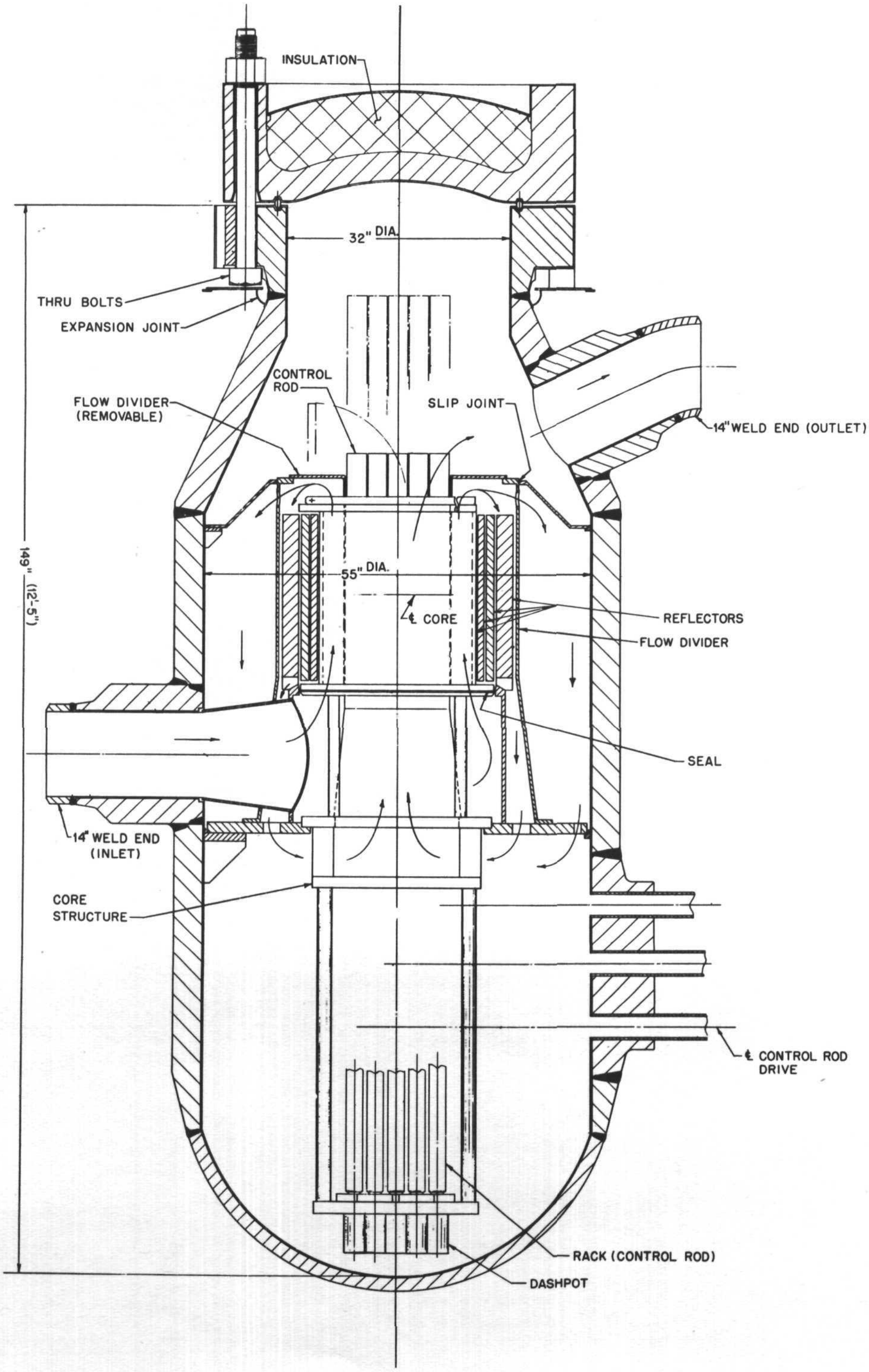


Figure VI-2.—SM-2 (APPR-1B) vertical section, reactor vessel (from report APAE Memo-197).

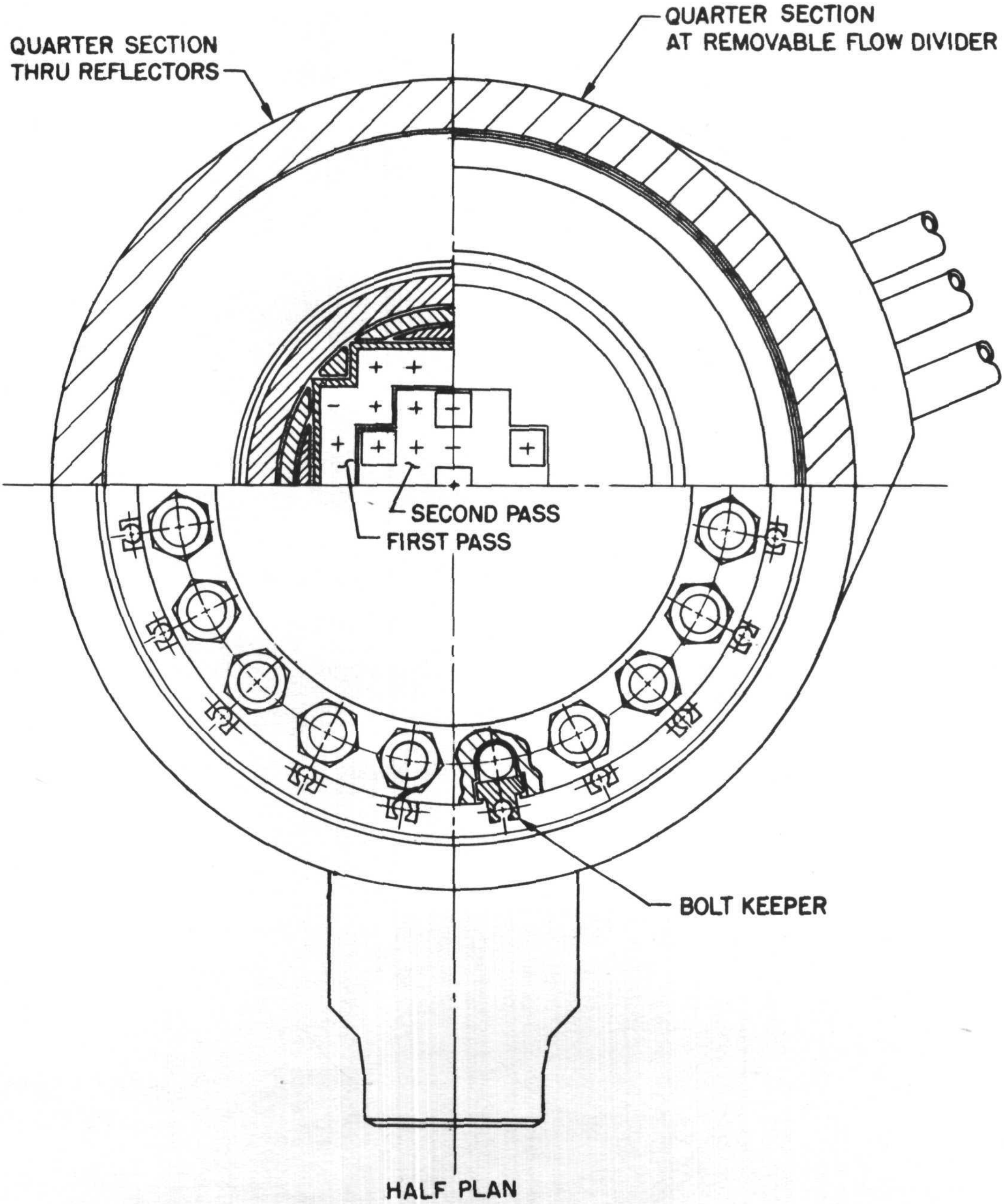


FIGURE VI-3.—SM-2 (APPR-1B) plan section, reactor vessel (from APAE Memo-197).

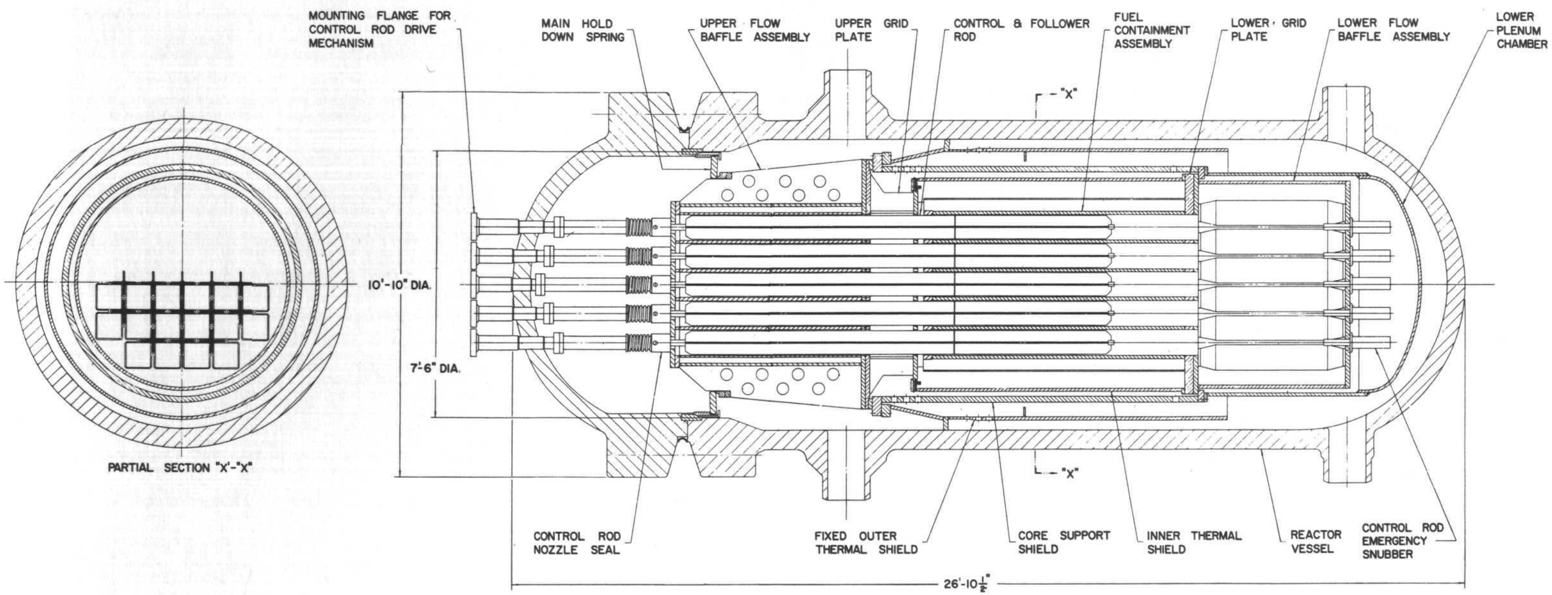


FIGURE VI–4.—Reactor vessel internals, NS *Savannah*.

540997 O—60 (Face p. 32)

FIGURE VI–5.—BR–3 reactor vessel and internals.

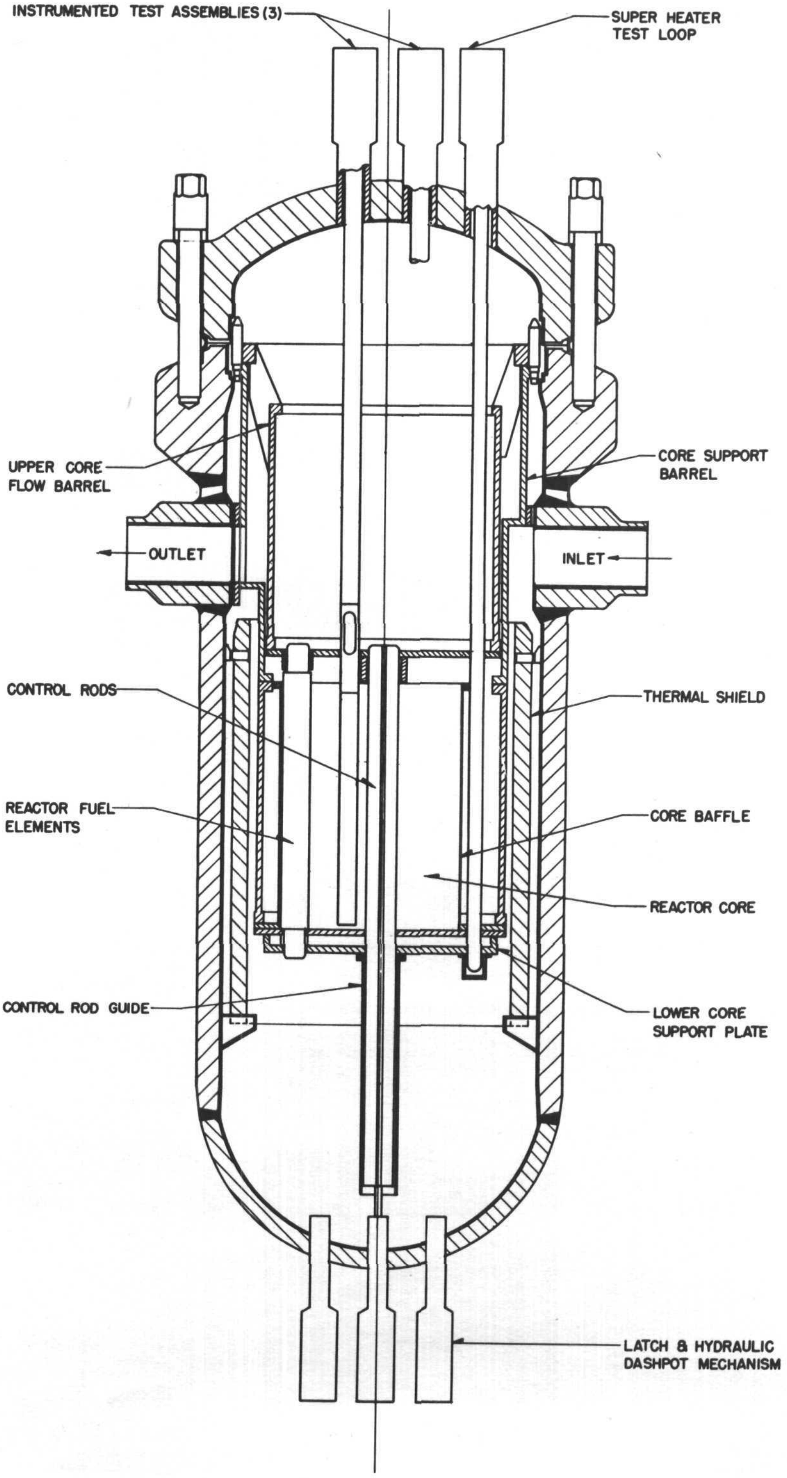


FIGURE VI-6.—Saxton reactor vessel; longitudinal cross section.

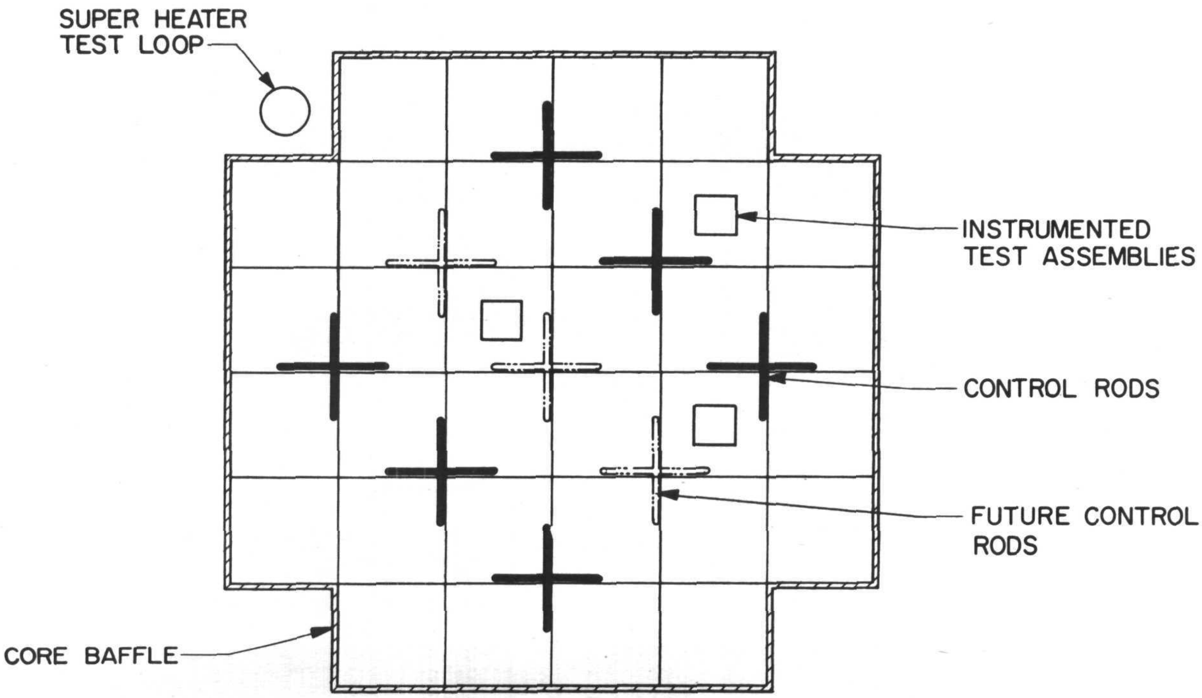


FIGURE VI–7.—Saxton experimental reactor core cross section.

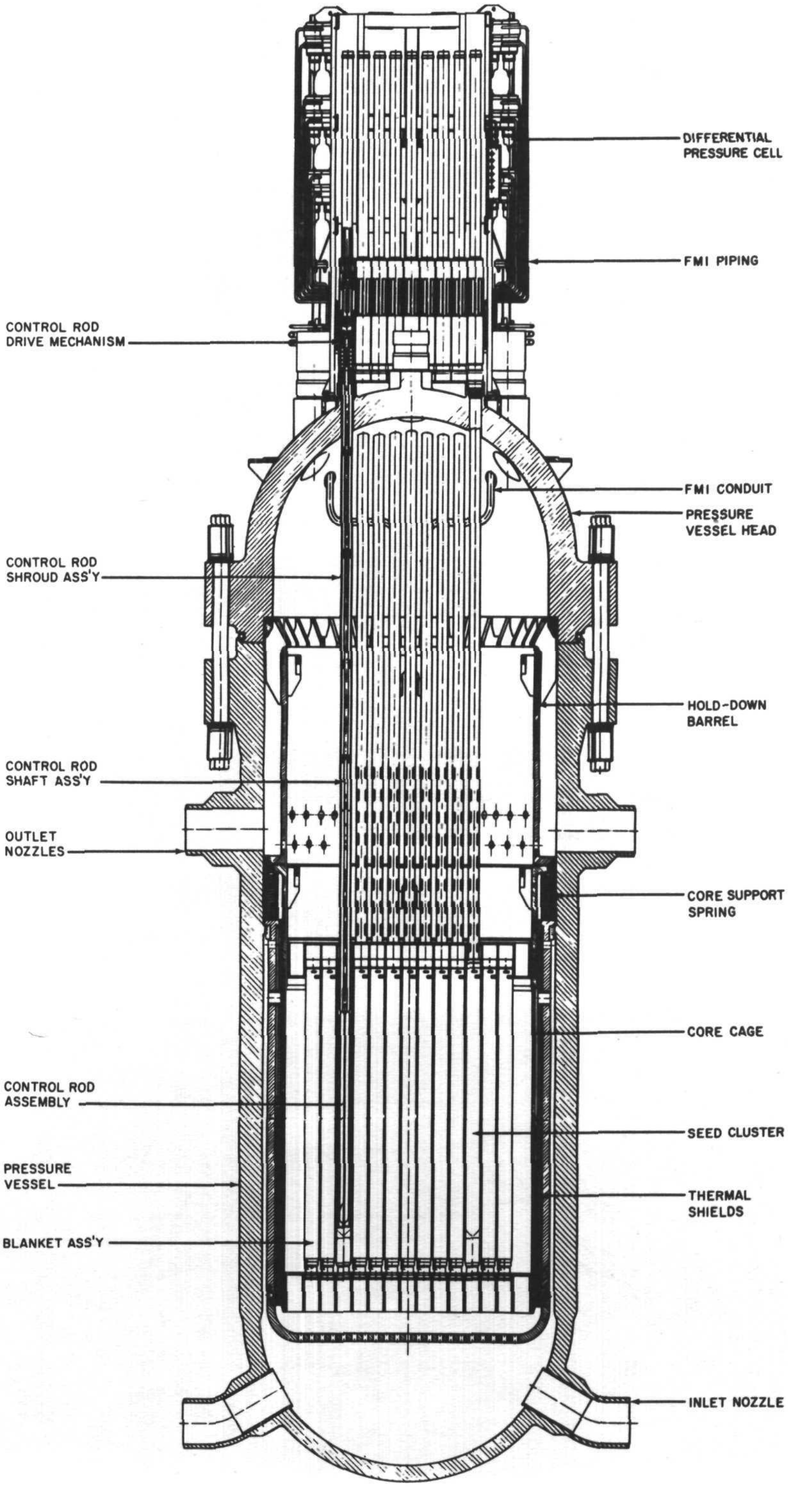


FIGURE VI-8.—Shippingport PWR-1 pressure vessel and core.

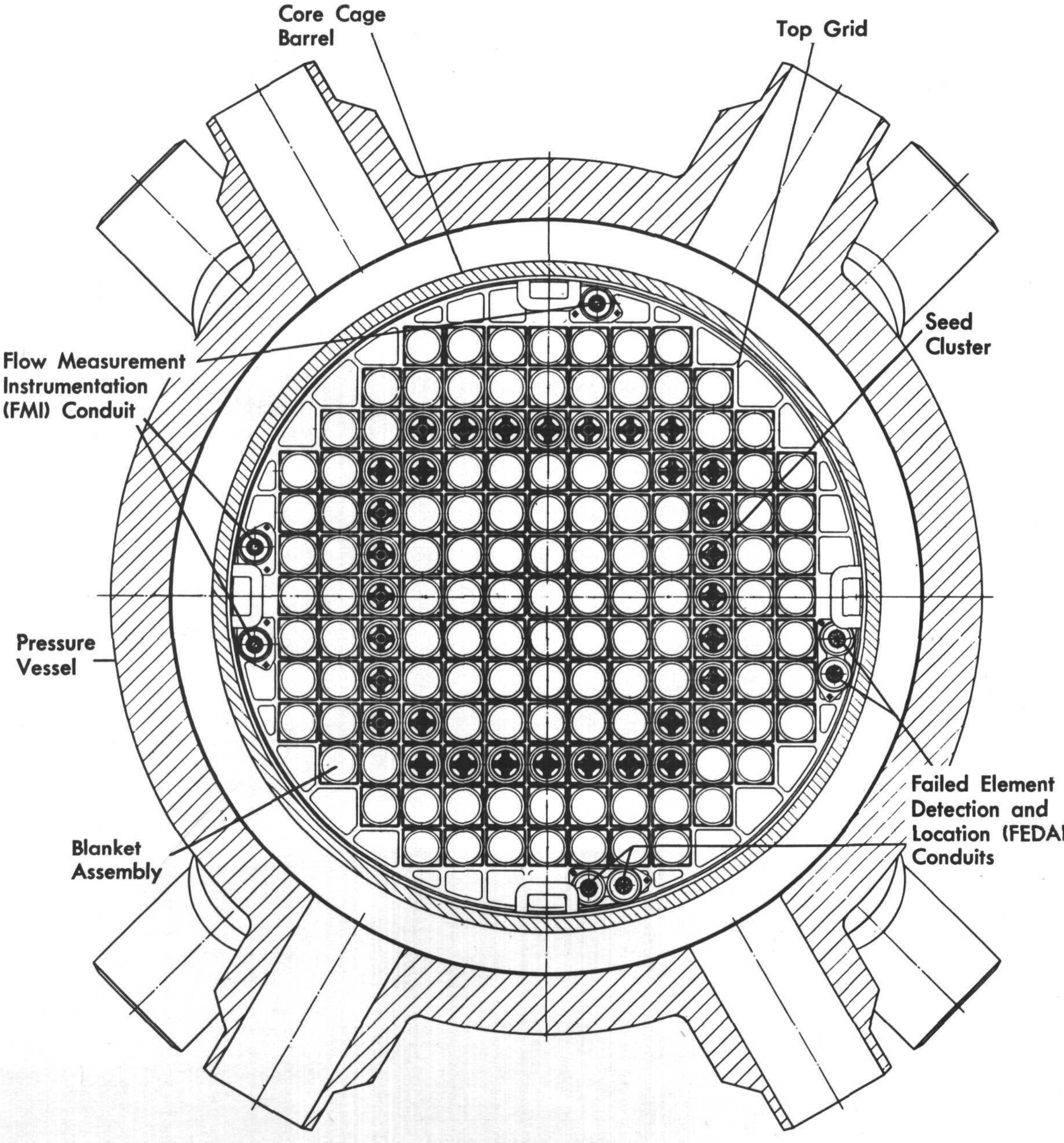


FIGURE VI-9.—Shippingport PWR-1 pressure vessel cross section above core.

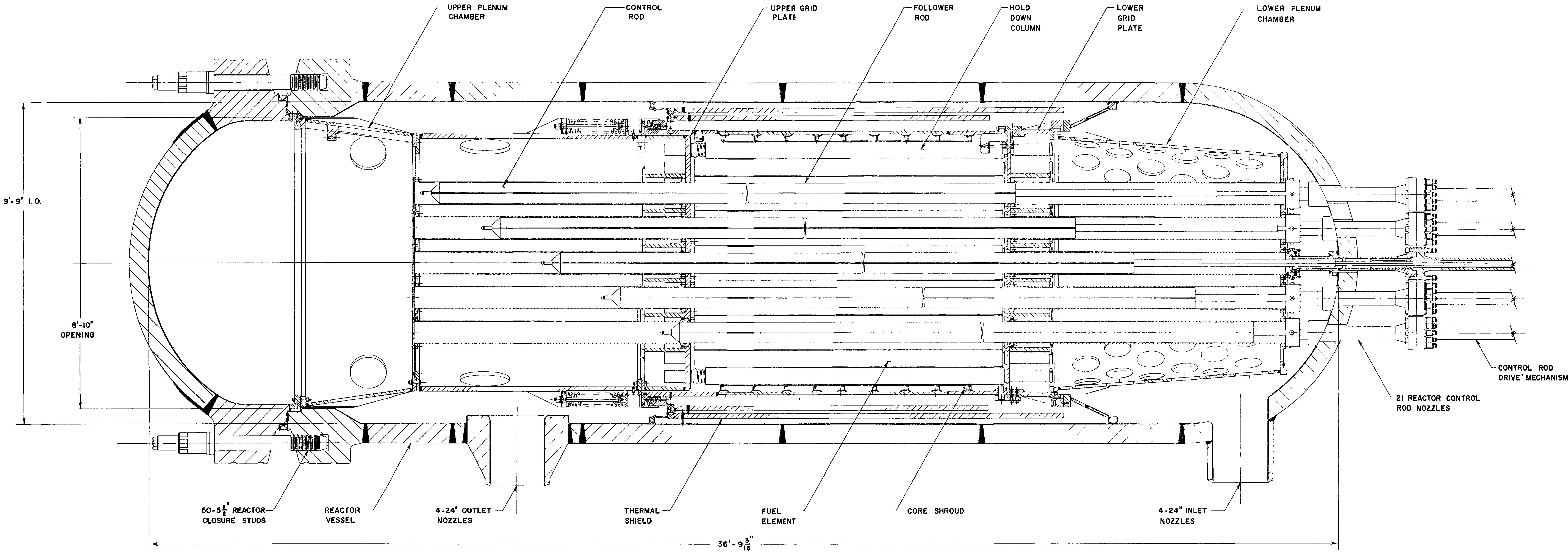


FIGURE VI-10.—Indian Point reactor vessel and internals.

540997 O—60 (Face p. 36)

Figure VI-12.—Yankee reactor vessel and internals.

540997 O—60 (Face p. 37)

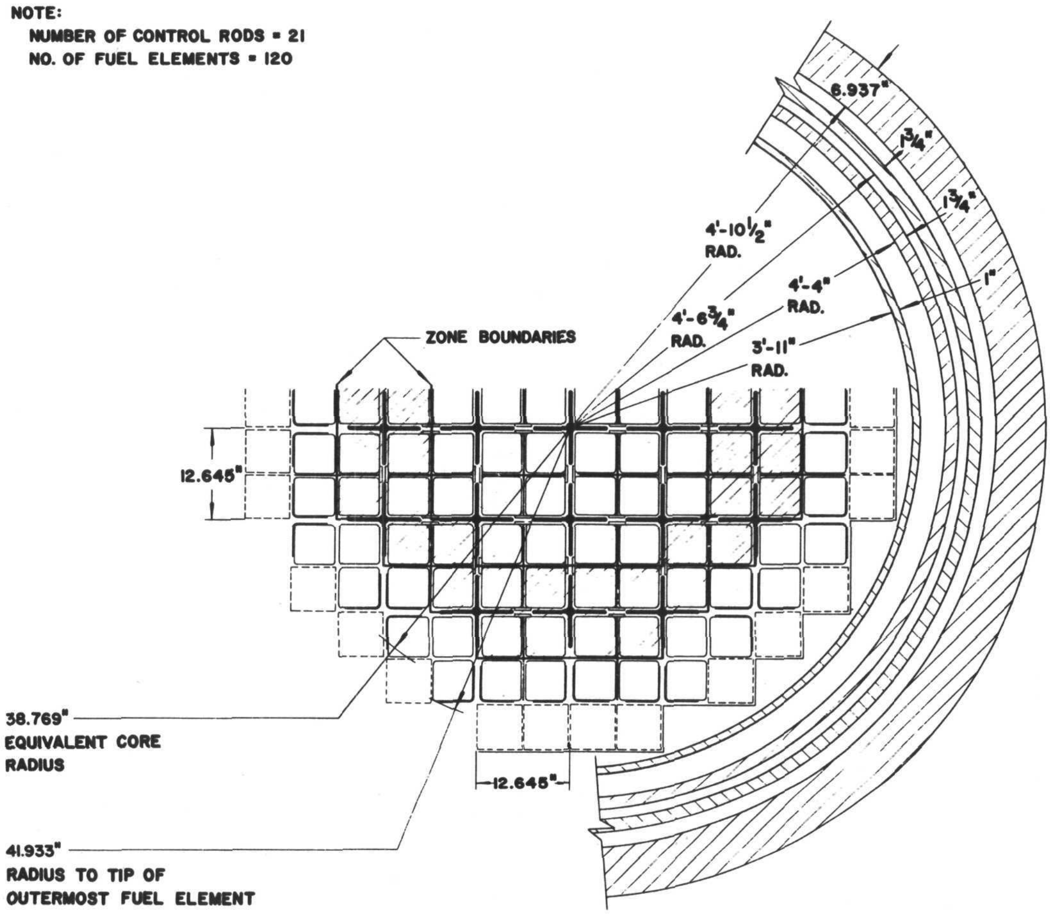


FIGURE VI–11.—Indian Point layout of core for Consolidated reactor, showing placement of fuel elements, thermal shields, and reactor vessel wall.

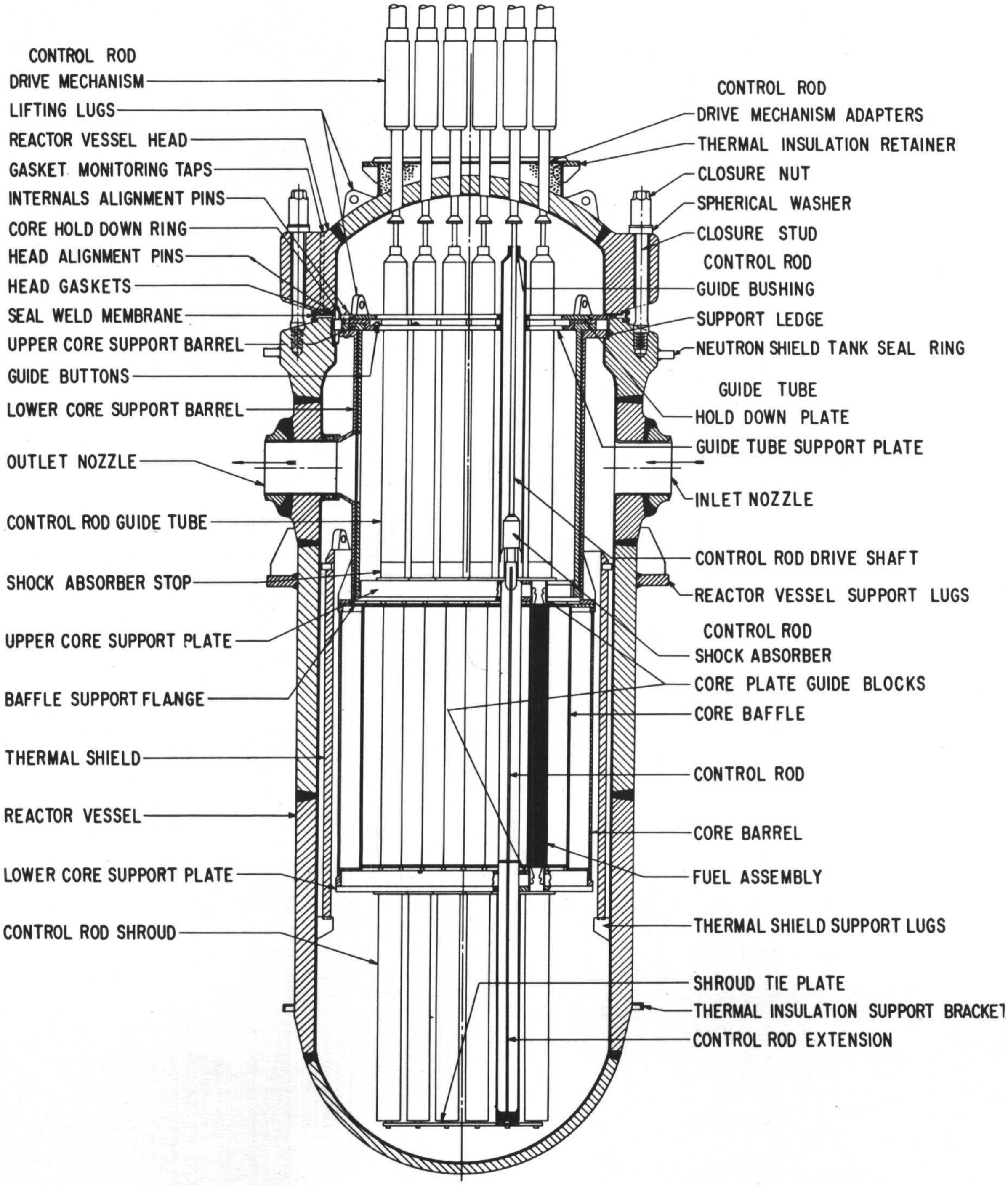


FIGURE VI-13.—165 Mw project reactor vessel.

540997 O—60 (Face p. 38)

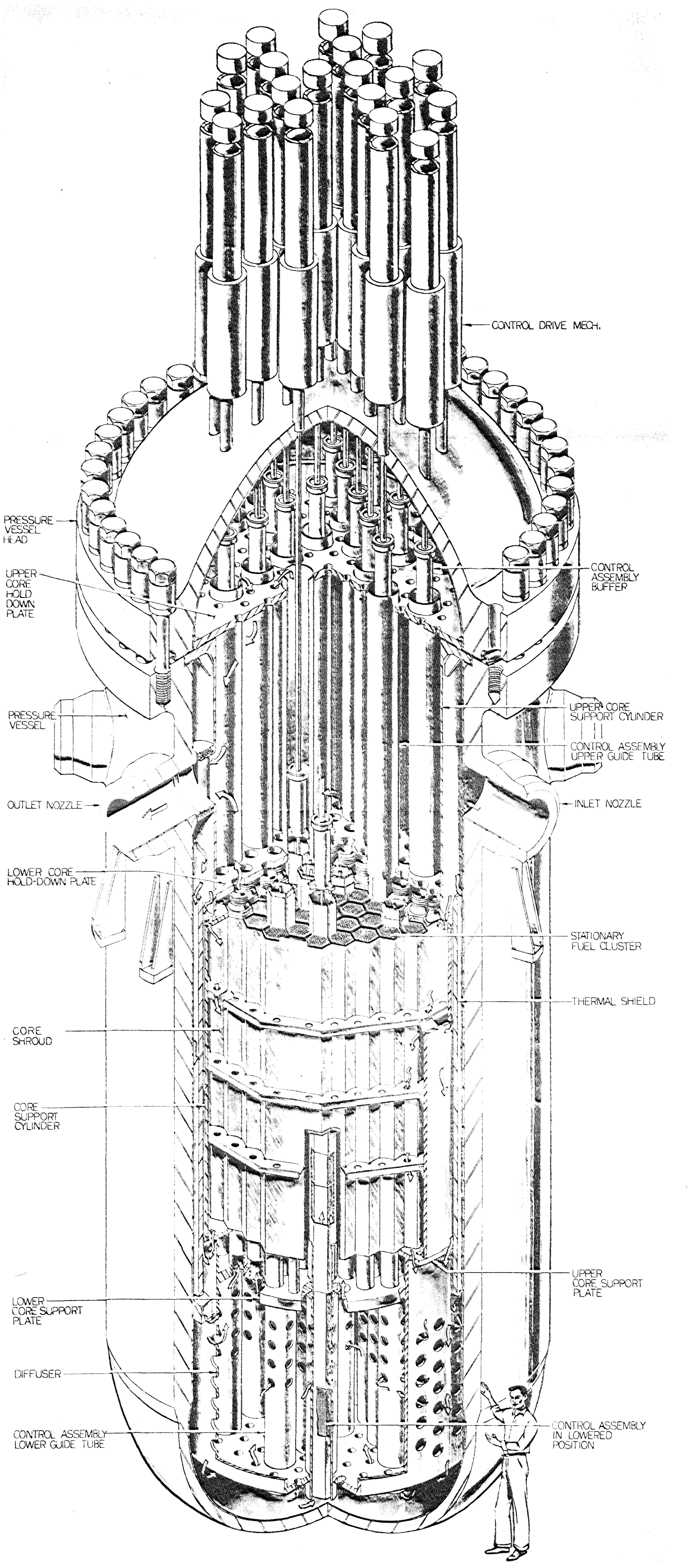


FIGURE VI–14.—Perspective of advanced pressurized water reactor (from TID–8502 (Part 3)).

VII. SYSTEM FLOW DIAGRAMS

This section contains the following reactor system flow diagrams:

Figure VII-1.—APPR-1 Heat Balance Diagram (from report APAE-10, Vol. II).

Figure VII-2.—SM-2 Basic Flow Diagram.

Figure VII-3.—Basic Flow Diagram for NS *Savannah* (adapted from Bull. AER-54, Babcock & Wilcox Co.).

Figure VII-4.—Basic Flow Diagram Belgian Thermal Reactor.

Figure VII-5.—Basic Flow Diagram Saxton Hook-on Reactor Plant.

Figure VII-6.—Basic Flow Diagram, PWR-1.

Figure VII-7.—Basic Flow Diagram, Indian Point.

Figure VII-8.—Basic Flow Diagram, Yankee Atomic Electric Plant.

Figure VII-9.—Basic Flow Diagram, 165 MW Project.

Figure VII-10.—Heat Balance Diagram, APWR (from TID-8502 (Part 3)).

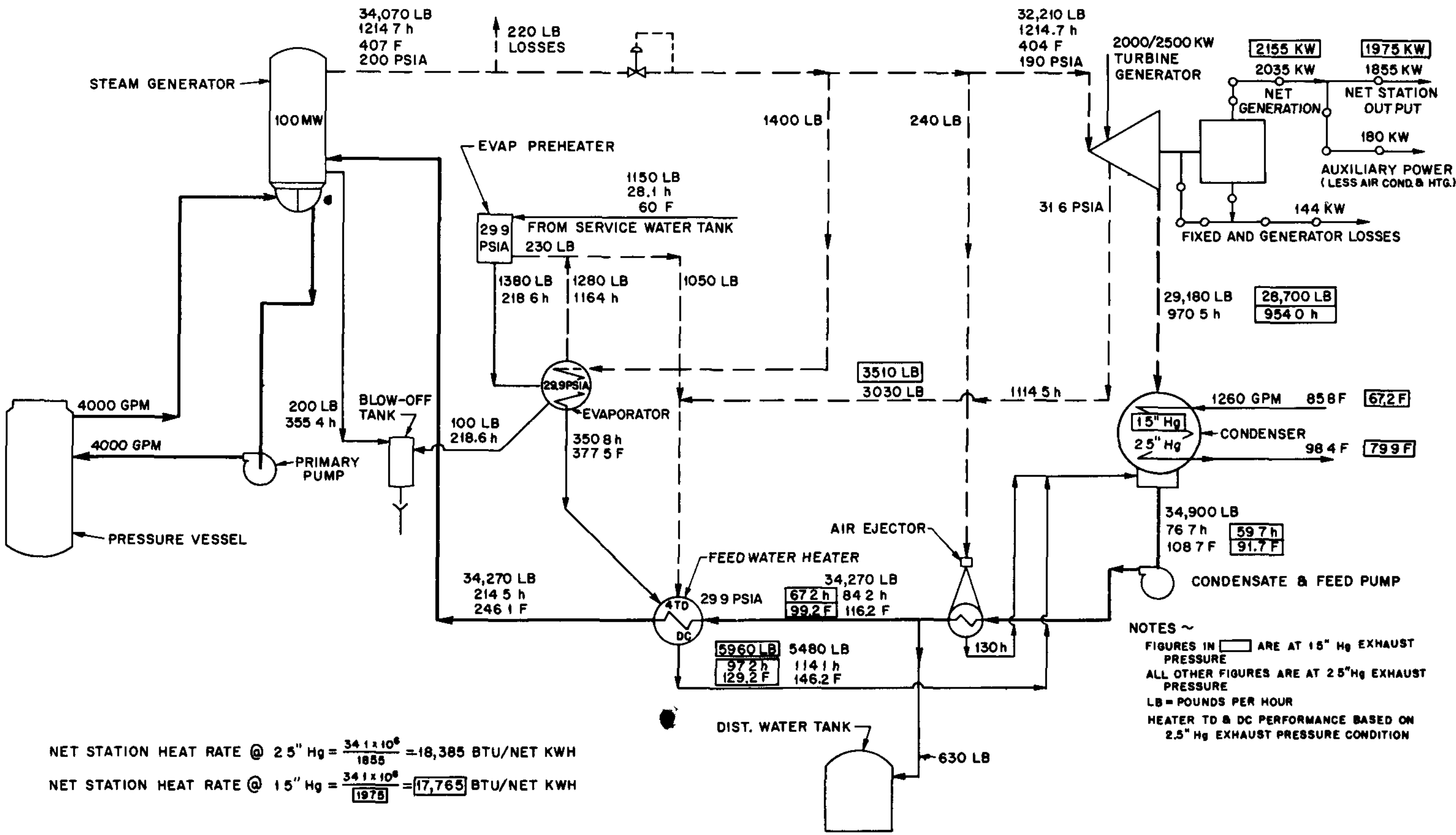


FIGURE VII-1.—APPR-1 heat balance diagram (from report APAE-10, Vol. II).

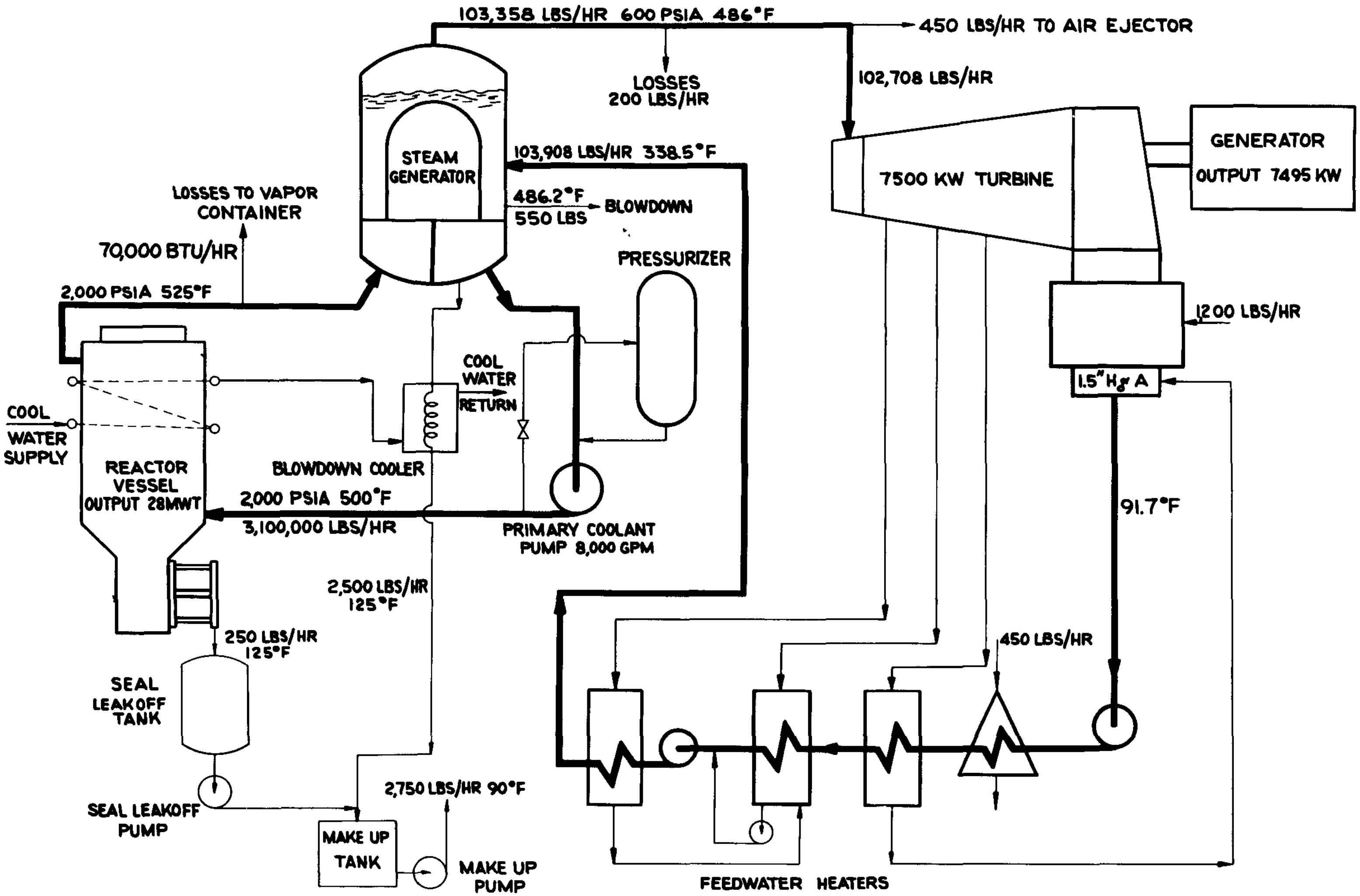


FIGURE VII-2.—SM-2 basic flow diagram.

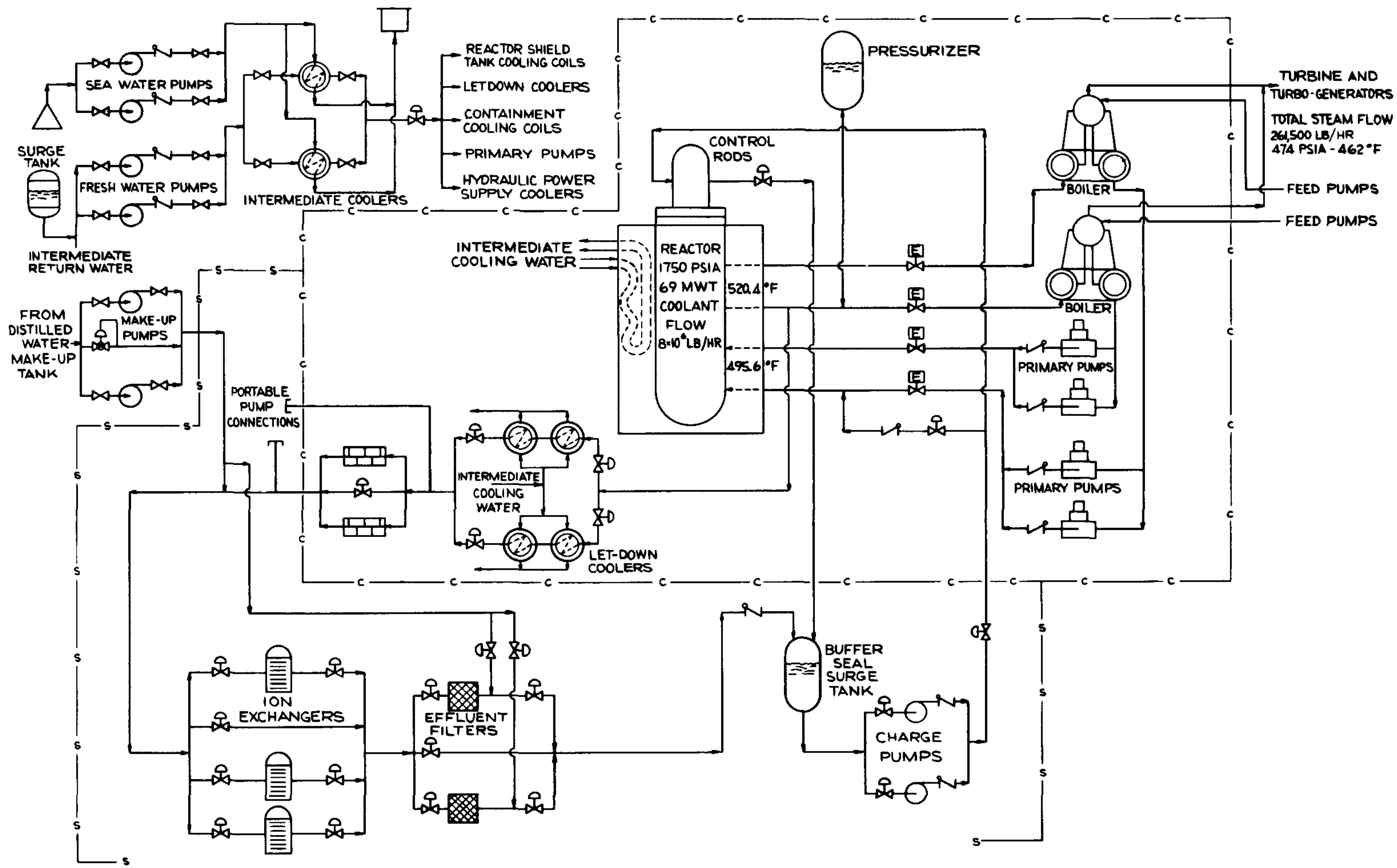


FIGURE VII-3.—Basic flow diagram for NS *Savannah* (adapted from Bull. AER-54, Babcock & Wilcox Co.).

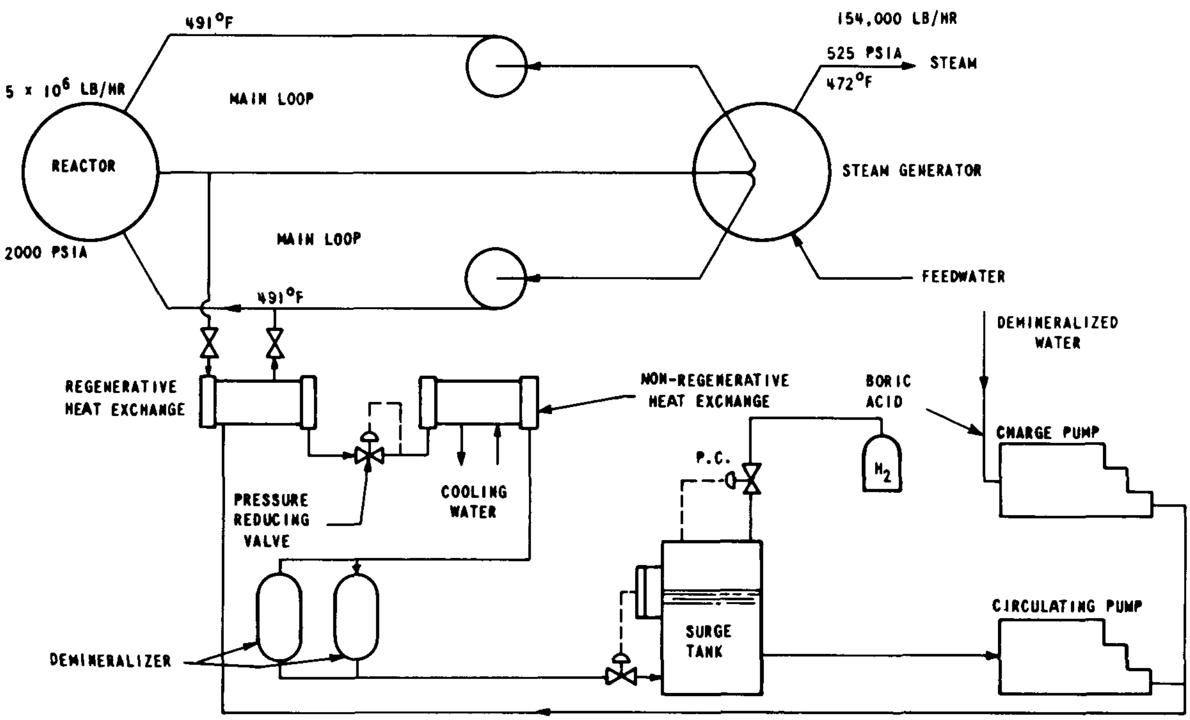


FIGURE VII-4.—Basic flow diagram, Belgian thermal reactor.

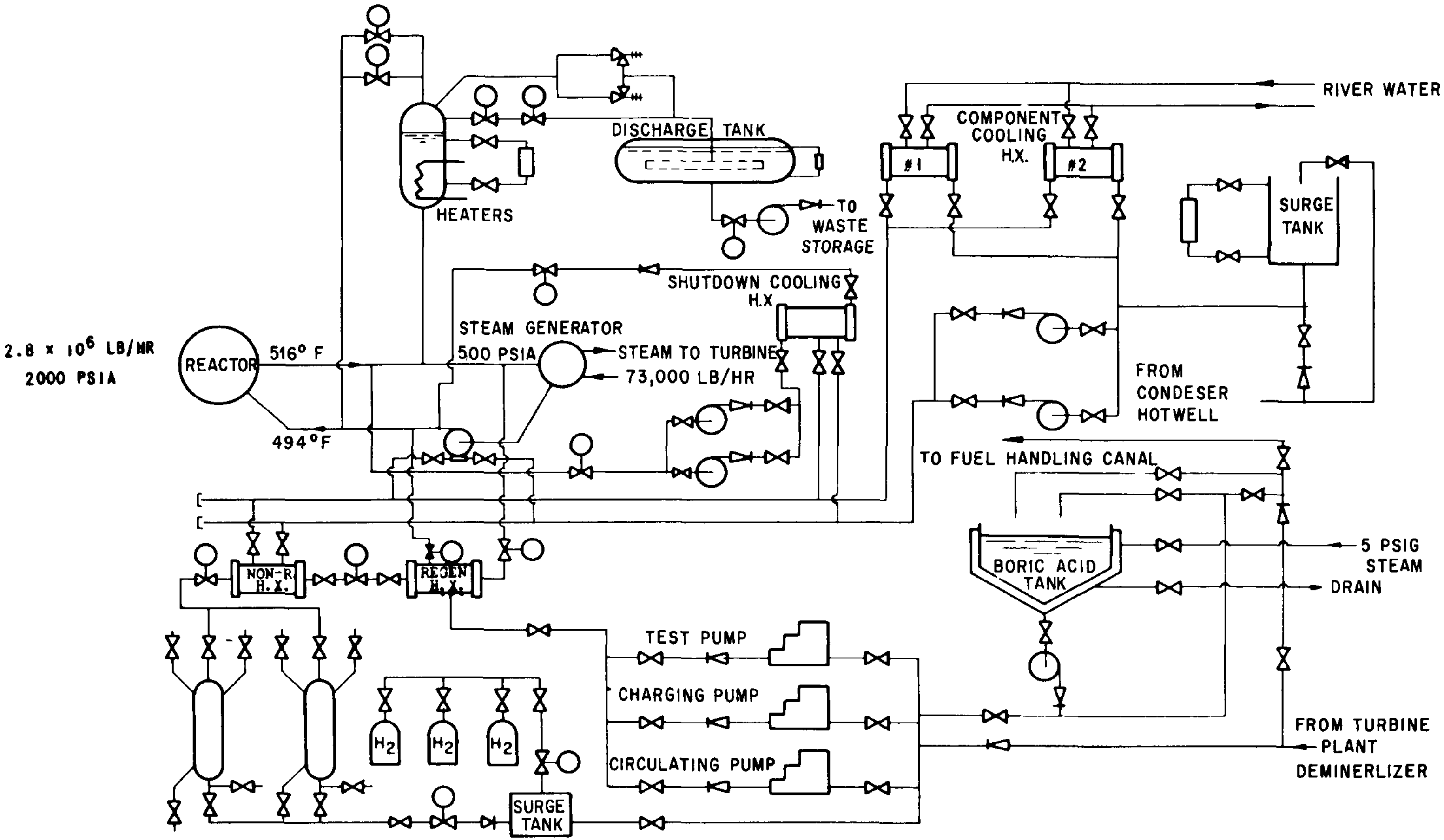
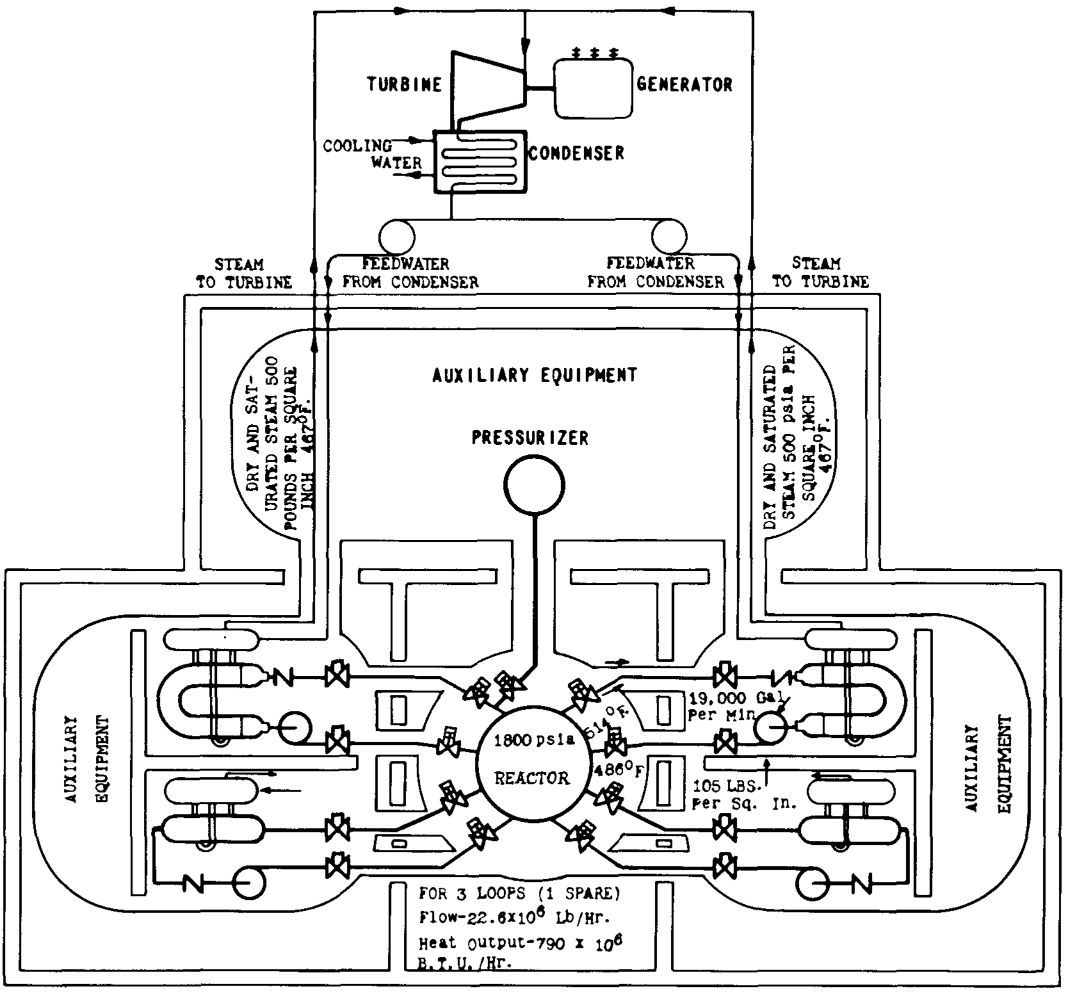


FIGURE VII-5.—Basic flow diagram, Saxton hook-on reactor plant.



MANUAL CAPPED VALVE

HYDRAULICALLY OPERATED VALVE

FIGURE VII–6.—Basic flow diagram, PWR–1.

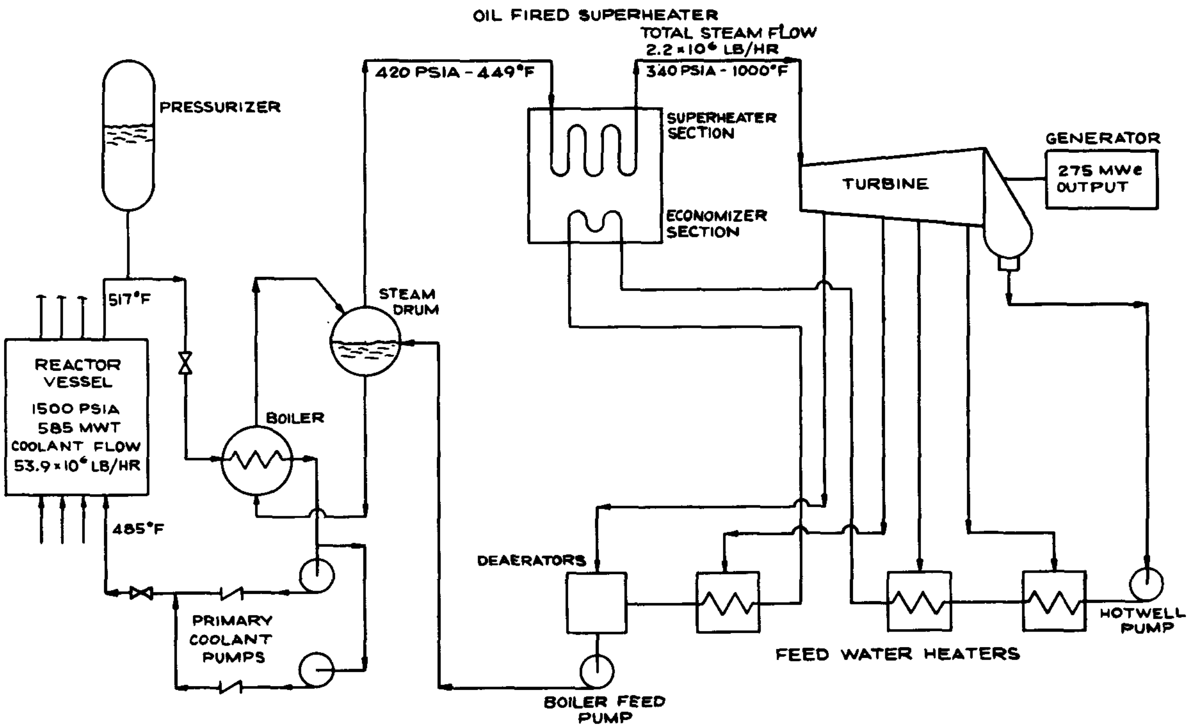


FIGURE VII-7.—Basic flow diagram, Indian Point.

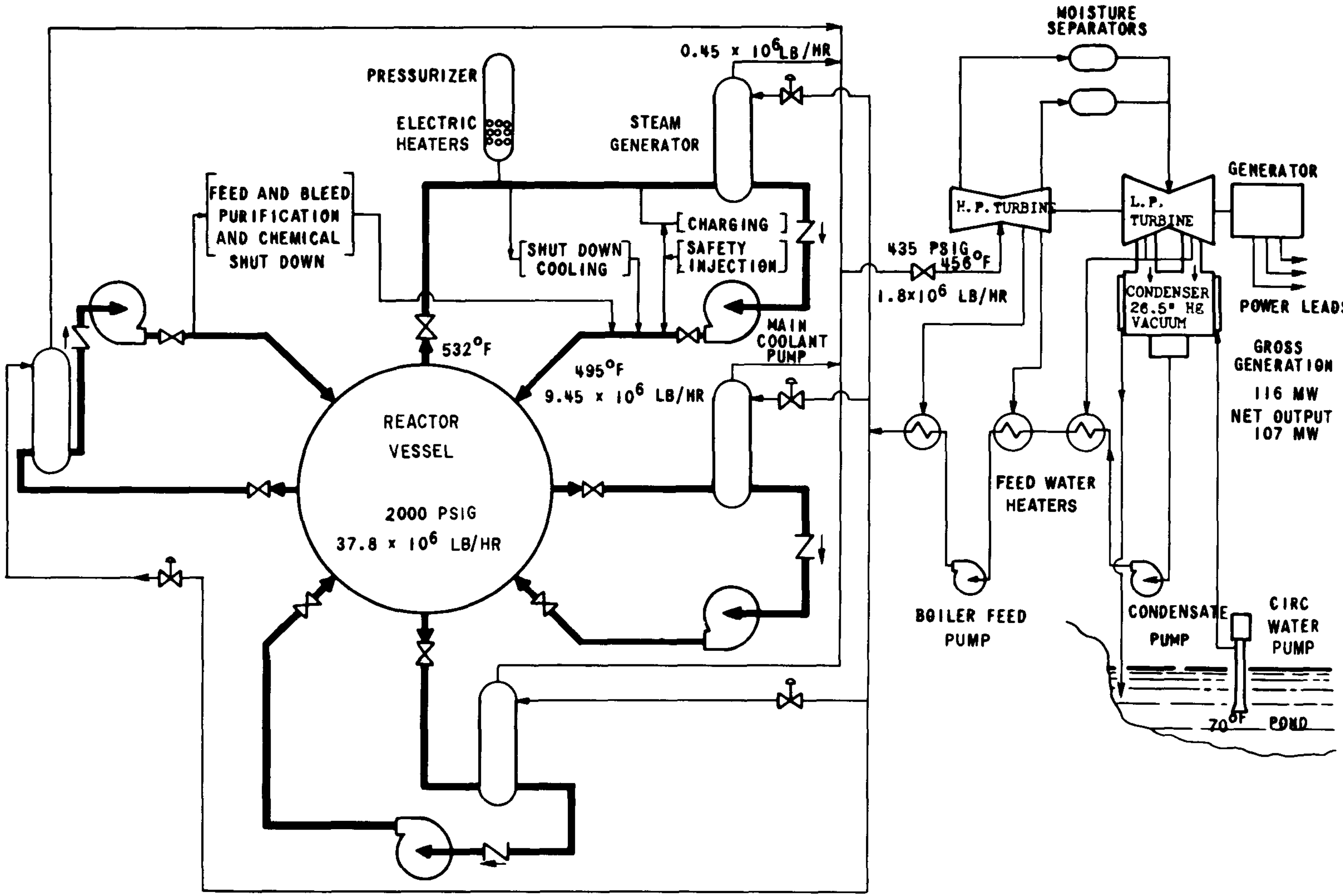


FIGURE VII–8.—Basic flow diagram, Yankee Atomic Electric Plant.

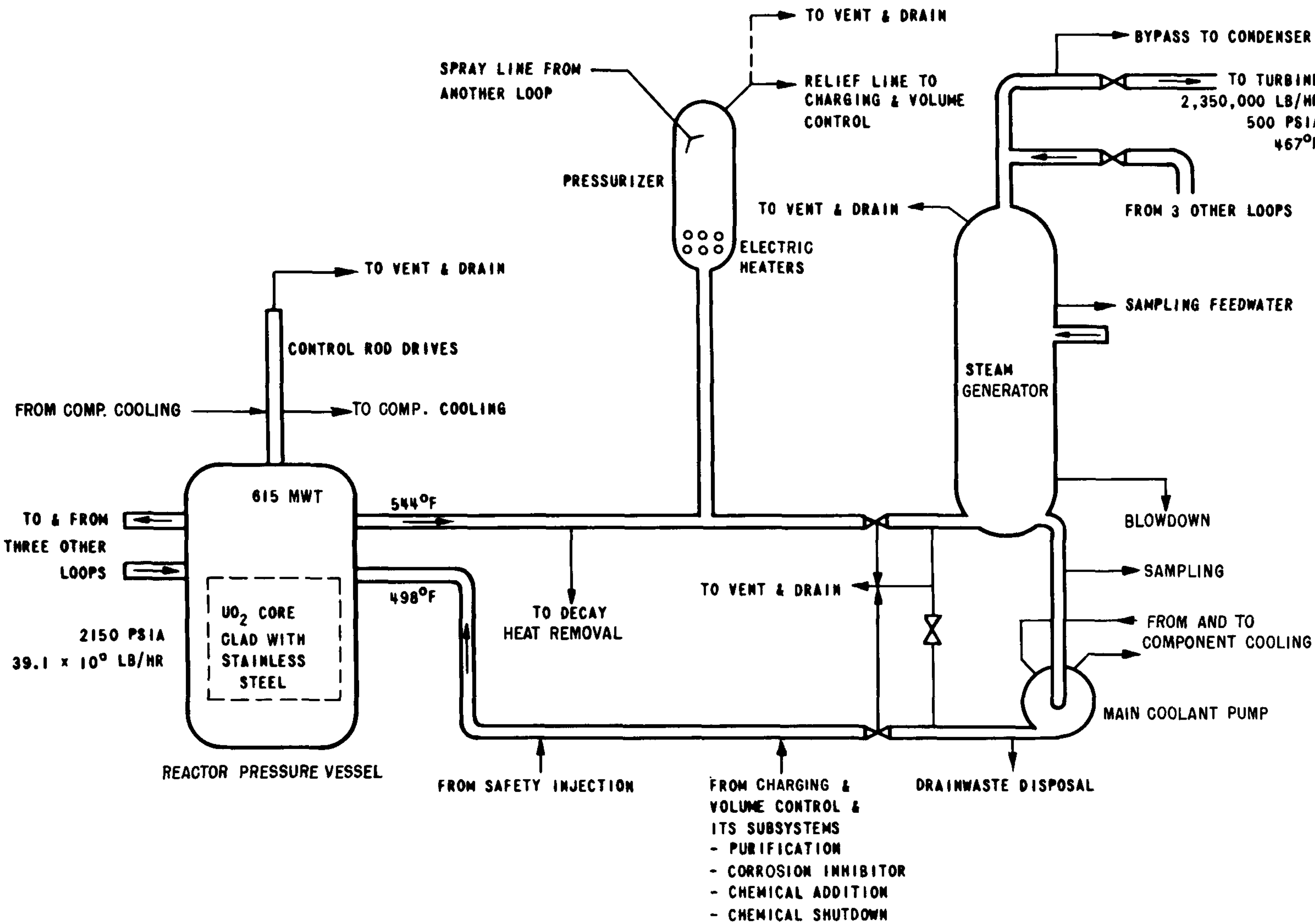
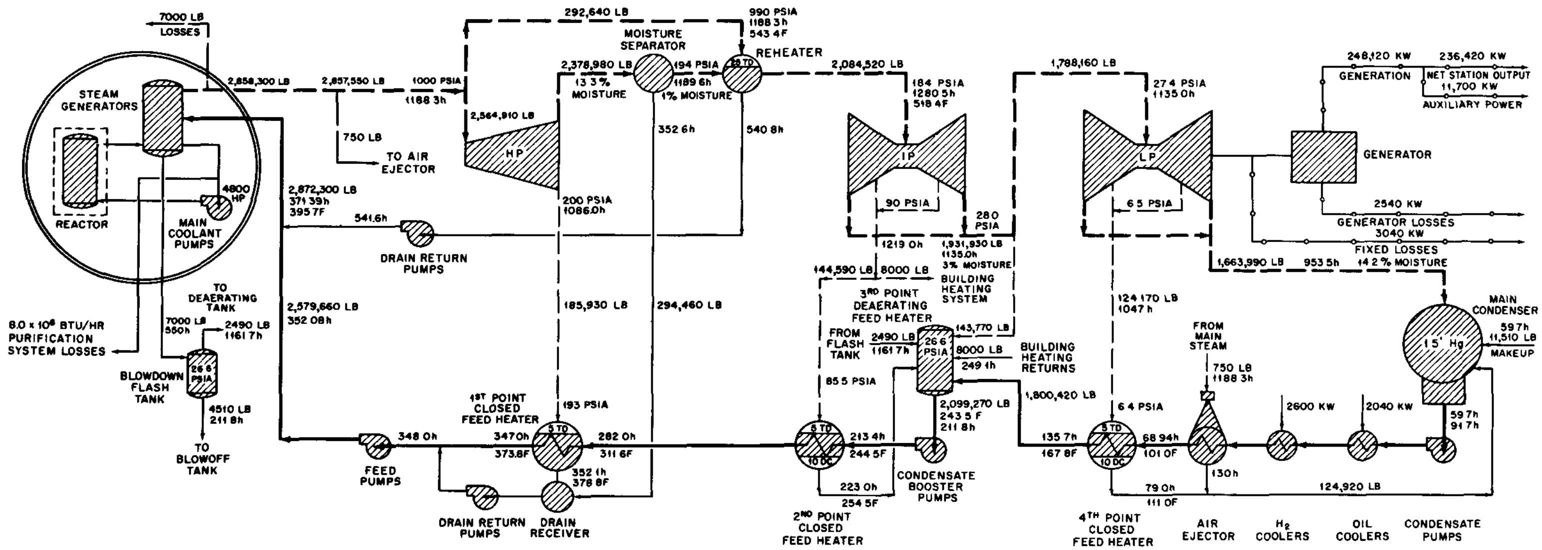


FIGURE VII–9.—Basic flow diagram, 165 Mw project.



TURBINE HEAT RATE = $\frac{(248,120+2540+3040-2040-2600)3412.75+1,663,990(953.5-59.7)+124,170(79.0-59.7)}{248,120}$ = 9430 BTU/KW-HR

STATION HEAT RATE = 9890 BTU/KW-HR

NOTES

Purification steam losses: 8×10^6 at all loads.
Steam losses: 7000 lb at all loads.
Steam-generator blowdown: 0.25% of steam generation.
Extraction line pressure drops are expressed as the following percentages of the absolute pressure at the turbine connection to the extraction line:

Line:	1st point	2nd point	3rd point	4th point
Per cent:	3.5	5.0	5.0	1.0

Extraction pressures shown at or near turbine are pressures at the turbine connection to the extraction line.
Heater entrance pressures are shown outside heater symbols.
The percentage pressure drop from the high-pressure turbine exhaust to the reheater inlet and from the reheater inlet to the reheater outlet is calculated at all loads.
Auxiliary power requirements have been calculated for main coolant pumps, steam-generator feed pumps, and condensate booster pumps, and have been estimated for all other equipment.

LEGEND

— —	STEAM
——	WATER
—o—	POWER
LB	FLOW, POUNDS PER HOUR
H	ENTHALPY, BTU PER POUND
F	TEMPERATURE, DEGREES FAHRENHEIT
TD	TERMINAL DIFFERENCE, DEGREES FAHRENHEIT
DC	TERMINAL DIFFERENCE (DRAIN COOLER), DEGREES FAHRENHEIT
KW	KILOWATTS
" Hg	PRESSURE, INCHES OF MERCURY, ABSOLUTE
PSIA	PRESSURE, POUNDS PER SQUARE INCH, ABSOLUTE
PSIG	PRESSURE, POUNDS PER SQUARE INCH, GAUGE

FIGURE VII-10.—Heat balance diagram, APWR (from TID-8502 (Part 3)).

VIII. OPERATING EXPERIENCE

A. NAVAL REACTOR PLANTS

The basic technology of the pressurized water reactor concept was developed for naval reactor plants. Similarly, most of the operational history of pressurized water reactors is intimately connected with the operational history of the nuclear plants constructed under the Navy program.

STR Mark I, the full scale land based prototype of the *Nautilus*, attained initial criticality on March 31, 1953, and operated at very low power until late May 1953 to obtain the initial necessary physics and radiation shielding information. On May 31, 1953, Mark I was placed in power operation and full design power was reached on June 25. Not one part of the nuclear plant indicated failure to meet specifications and it took less than one month after first power generation before Mark I was operating smoothly at full power.

The initial purpose of the Mark I plant was to prove the feasibility of nuclear power propulsion for naval vessels, to train operators, and specifically to eliminate design deficiencies in the *Nautilus*' plant. A few years ago Mark I with a new core operated 66 days and nights continuously at full power, a significant indication of the developed technology and reliability of the pressurized water reactor concept. The Mark I is now being used to study new developments in the technology, design and operation of improved pressurized water plants.

The Mark I plant has been the forerunner for a whole family of naval plants for submarines and surface ships. Table VIII–1 is a listing of the naval vessels that utilize pressurized water reactor plants. All of these plants, though, use highly enriched fuel to minimize core size; in a central station nuclear power plant, the economics would dictate the use of slightly enriched uranium.

B. SM–1 (APPR–1)

The SM–1 was the first plant to be constructed under the Army nuclear power program. Construction was started in October 1955, the reactor went critical on April 8, 1957, and full power operation, 10,000 KWt, was first achieved on April 20. An initial 700-hour endurance test was completed by July 2 with only 7 hours 28 minutes of down time.

1. Transient Response

The SM–1 has proven to be an extremely stable powerplant. This has been demonstrated by results of operation during both planned and unplanned load changes.

During an early full power run a circuit breaker failure resulted in dropping the reactor load from full load to station load. The transition was so smooth—without any operation control action—that it was nearly a minute before the operators realized that the breaker had failed.

Subsequently a planned load drop transient was induced by tripping the turbine off the line. During this test the reactor did not scram and the control rods were not moved. During the test the primary system pressure rose sharply from 1,210 to 1,300 p.s.i.g. and then dropped until the pressurizer heaters came on at 1,200 p.s.i.g. The pressure peak was well below the design pressure of 1,600 p.s.i.g. The reactor ΔT dropped to essentially zero on loss of demand and remained at a constant value without oscillation. Immediately on loss of load the steam pressure rose from 200 to 435 p.s.i.g. and the steam temperature rose from 420 to 460° F.; both then steadily decreased.

TABLE VIII-1.—PRESSURIZED WATER REACTOR POWER PLANTS FOR THE NAVY

Hull No.	Name	Shipyard	Type*	Keel laying	Launching	Sea trials	Commissioned
SUBMARINES							
SSN-571	NAUTILUS	Electric Boat	A	6/14/52	1/21/54	1/17/55	9/30/54
SSN-575	SEAWOLF	Electric Boat	A				
SSN-578	SKATE	Electric Boat	F	7/21/55	5/16/57	10/27/57	12/23/57
SSN-579	SWORDFISH	Portsmouth	F	1/25/56	8/27/57	8/19/58	9/17/58
SSN-583	SARGO	Mare Island	F	2/21/56	10/10/57	8/4/58	10/1/58
SSN-584	SEADRAGON	Portsmouth	F	6/20/56	8/16/58		
SSN-585	SKIPJACK	Electric Boat	A	5/29/56	5/26/58	3/8/59	4/15/59
SSR(N)-586	TRITON	Electric Boat	R	5/5/56	8/19/58		
SSG(N)-587	HALIBUT	Mare Island	GM	4/11/57	1/9/59		
SSN-588	SCAMP	Mare Island	A	1/23/59			
SSN-589	SCORPION	Electric Boat	A	8/20/58			
SSN-590	SCULPIN	Ingalls	A				
SSN-591	SHARK	Newport News	A	12/2/57			
SSN-592	SNOOK	Ingalls	A				
SSN-593	THRESHER	Portsmouth	A				
SSN-594	PERMIT	Mare Island	A				
SSN-595	POLLACK	Mare Island	A				
SSN-596	PLUNGER	Ingalls	A				
SSN-597	TULLIBEE	Electric Boat	S				
SSB(N)-598	GEORGE WASHINGTON.	Electric Boat	FBM	11/1/57	6/9/59		
SSB(N)-599	PATRICK HENRY	Electric Boat	FBM				
SSB(N)-600	THEODORE ROOSEVELT.	Mare Island	FBM				
SSB(N)-601	ROBERT E. LEE	Newport News	FBM	8/25/58			
SSB(N)-602	ABRAHAM LINCOLN.	Portsmouth	FBM				
SSN-603		New York	A				
SSN-604		New York	A				
SSN-605		Portsmouth	A				
SSN-606	TINOSA	Portsmouth	A				
SSN-607		Ingalls	A				
SSB(N)-608	ETHAN ALLEN	Electric Boat	FBM				
SSB(N)-609			FBM				
SSB(N)-610			FBM				
SSB(N)-611			FBM				
SURFACE SHIPS							
CG(N)-9	LONG BEACH	Bethlehem	Cruiser	12/2/57			
CVA(N)-65	ENTERPRISE	Newport News	Carrier	2/4/58			
	BAINBRIDGE	Bethlehem	Destroyer	5/15/59			

*A—Attack
F—Fleet-Type Attack
FBM—Fleet Ballistic Missile (Polaris)
GM—Guided Missile (Regulus)
R—Radar Picket
S—Small Attack

An increase in load transient was accomplished by increasing the electrical load from 225 to 2050 KW in 75 seconds. On increase in load, the reactor power rose very nearly as fast as the power demand and data indicate that negligible power overshoot developed. The shortest reactor period during the transient was 25 seconds, well above the scram setting of 3 seconds. Reactor ΔT rose sharply from an initial value of 4 to 20° and then gradually to 21° F., finally stabilizing between 20 and 21° F. Steam pressure dropped sharply from 322 to

175 p.s.i.g. and then rose to a new equilibrium pressure of 200 p.s.i.g. over a period of 45 minutes. Steam temperature dropped from 430 to 410° F. over a 10-minute period and stabilized at the latter value.

2. Mechanical Component Failures

During the initial tests it was determined that the steam generator failed to deliver the design superheat. The difficulty was caused by the moisture separator. A higher than anticipated pressure drop in the separator resulted in the drain drawing water up the dry side of the separator. Redesign and replacement of the separator corrected the difficulty.

Considerable trouble was experienced initially with pumps—except for the primary system canned motor pump. The most serious pump problem reported was high leakage rates through the packing of the make-up pumps. The trouble was traced to excessive wear of the stainless steel plungers. In one pump the S.S. plungers were replaced with Hastelloy and in the other with ceramic. Both replacement materials have proved serviceable.

Lack of instrument reliability has accounted for the most aggravating routine problems. The signal strengths from the ion chambers are so low as to require operation of the amplifier at excessively high gain levels to achieve adequate signal strength. Because of this condition electron tube serviceable life is very short before noise levels increase to the level where tube replacement is necessary. High noise levels have resulted in many false period scrams. The problem has been somewhat alleviated by installation of filters and by the correction of improper seals of mineral insulated wire terminations.

The original BF_3 counters failed during power operation, and have been replaced by those of another manufacturer.

3. Radiation Levels

In general the measured radiation levels have been found to be reasonable and the primary and secondary shielding design to be conservative. Table VIII–2 shows the measured levels at selected points at full power and at various times after shutdown. The high dose rate at the "pressurizer elbow" location is attributed to the accumulation of activity in this stagnant section.

4. Activity in the Primary Coolant

Short-lived activity in the primary coolant has been determined to be approximately equally divided between fission products and

TABLE VIII–2.—RADIATION LEVELS IN SM–1

Location	Full power operation		Hours after shutdown			
	Gamma	Thermal flux	7.9	13.9	24	80
	r/hr	10^3 n/cm² sec	mr/hr	mr/hr	mr/hr	mr/hr
Point of maximum radiation due to primary shield penetrations	15	13	54	28	25	16
Inner wall of vapor container	.3	10	2	1		
Adjacent superheat section of steam generator	4.0		39	27	21	14
Primary coolant inlet pipe outside primary shield tank	12			40		19
Elbow of 4-inch pressurizer line	(*)		300	300	320	240
Control rod drive pit	(*)		140	140		52

*Not measured during operation.

induced activity. The fission product activities are in excess of that attributable to surface contamination and are believed due to cladding defects. Data seem to indicate that the magnitude of the defects is increasing with time.

Radiochemical analyses of the primary water indicate long-lived species $Co^{60,58}$, Fe^{59}, Mn^{54} and Cr^{51}. A slow steady rise of activity has been observed. The measured dose rate in the steam generator is shown in Table VIII-3.

TABLE VIII-3.—SM-1 LONG-LIVED DOSE RATES (24 HOURS AFTER SHUTDOWN FROM FULL POWER)

Location	Dose rates (mr/hr)	
	11/7/58	3/7/59
Steam generator water box	5. 7	6. 2
Steam generator above No. 2 outlet	190	165
Steam generator next No. 1 outlet	10	14
Upper collar steam generator above steam line	94	400
Upper collar steam generator above inlet	60	100
Steam generator No. 2 outlet, bottom	49	65
Steam generator No. 2 outlet, top	49	70
Steam generator No. 1 outlet, bottom	47	63
Reactor inlet pipe, bottom	87	100
Reactor inlet pipe, top	117	260

The reactor was shut down in April 1959 and the steam generator head was lowered for inspection. At the start of this operation the dose rate on the outside of the steam generator head was 1 r/hr, primarily due to crud deposits in the drain line. After removal of the drain line and moderate flushing the level was reduced to 250 mr/hr. On lowering the head the dose rate on the inside of the head was 2 r/hr. In spite of this activity, the removal, inspection and replacement of the head was carried out with 140 manhours of mechanical work and a total of 673 manhours including nonproductive and waiting time. No overexposures of personnel were incurred in the operation. The operation demonstrated the feasibility of maintenance of primary system units under radiation conditions.

An extensive R & D program was carried out to develop decontamination procedures applicable to the SM-1 system. The recommended method of decontamination is a fill-flush procedure in which the solutions are introduced into the system, circulated at temperature and pressure for the requisite time, and then drained. The conditions under which the activated corrosion product scale is most effectively removed are as follows:

Caustic Permanganate Step (10% NaOH+5% $KMnO_4$).	30 minutes at 225° F. at 60–70 p.s.i.g. pressure (nitrogen).
Water Flush Step	5 minute water flush with demineralized water.
Citrate Combination Step (5% dibasic ammonium citrate +2% citric acid +½% Versene 9).	30 minutes at 220° F. at 60–70 p.s.i.g. pressure (nitrogen).
Successive Water Flushes	15 minute water flushes to restore system to operating levels of purity.

Based on loop experiments, a decontamination factor of at least 10 and probably 30 is anticipated in the actual plant.

5. Control Rods

The original control rods were made of amorphous boron in iron, clad in stainless steel. These were replaced by Eu_2O_3 in stainless steel, clad in stainless steel, because of a fear of swelling as a result of gas formation upon neutron absorption in boron. A test coupon of boron showed that this could happen, but no swelling was noticeable at the time the control rods were replaced.

6. Conclusions

The data presented has been extracted from numerous documents listed in the bibliography. The following extract from APAE-18 summarizes the experience with the reactor:

"The SM-1 has proven itself to be a reliable nuclear power plant—and has operated so as to meet all design requirements. Particularly noteworthy during the test was the response of the reactor to load demand, as the stability of the SM–1 exceeded all expectations."

By the end of April 4, 1959 SM–1 had generated a total of 91,980 MW of heat.

C. SHIPPINGPORT ATOMIC POWER STATION

The Shippingport reactor went critical on December 2, 1957, and supplied nuclear power for the first time to the Duquesne Light Co. system on December 18. Full power operation, 60,000 KW net electrical, was achieved on December 23.

Shippingport is the world's first full scale nuclear power station but as such it is also an important development facility. The method of operation, therefore, has been dictated not only by load considerations but also by test programs. Yet, the plant has operated at a load factor three times more than predicted. Testing has included 2 power runs of more than 1,000 hours each as part of a Reactivity Lifetime Test, and the seed fuel has operated for approximately 5,100 equivalent full power hours to date. The original design life was 3,000 equivalent full power hours and is now estimated to be 6,000 equivalent full power hours.

1. Transient Response

The stability of Shippingport on loss of load was adequately demonstrated at the end of the initial 100-hour full power run. While preparing for shutdown, and with the reactor still operating at full power, loss of condenser vacuum caused the turbine throttle valves to trip. No scram occurred, no relief valves popped, the reactor was maintained critical and the plant temperature and pressure stabilized without oscillation or control rod motion. (condenser vacuum had decreased to the turbine throttle's trip point while transferring auxiliary steam supply which caused the steam pressure to drop below that required for proper air ejector operation.) The reactor was not on automatic control and the operator did not move control rods manually until approximately 25 seconds after the throttle trip occurred. The peak disturbances in the reactor coolant conditions were as follows:

Coolant pressure rise	+180 p.s.i.
Coolant temperature rise	+12° F.
Coolant volume change	+31 cu. ft.

The rise in coolant pressure increased the pressurizer pressure to a valve just under the setting of the pilot-operated pressure relief valve.

A second loss of load from full power operation occurred on shutdown from the first 1,000-hour run when there were 1,690 equivalent full power hours on the core. For this second full-load loss, the reactor was in automatic control which was set to initiate rod insertion at 1° F. above the average value of 523° F. Rod insertion began in less than 3 seconds after the throttle was tripped, and continued until the reactor power was below 8 percent. The peak disturbances in the reactor coolant conditions were as follows:

Coolant pressure rise	+180 p.s.i.
Coolant temperature rise	+12.2° F.
Coolant volume increase	+33.3 cu. ft.

The temperature coefficient of reactivity had changed from -2.9×10^{-4} $\delta K/°$ F. at the time of the first incident to -2.3×10^{-4} $\delta K/°$ F. The fact that the second peak disturbances were not lower than those of the first incident is probably attributable to the lower value of temperature coefficient. In both cases the automatically controlled steam dump system was not in service.

On March 4, 1959, at approximately 3,600 equivalent full power hours the turbine was tripped off manually, with the plant at full power and normal temperature and pressure, to commence the Xenon Transient Test. A severe temperature and pressure transient resulted from a 10 to 20 second delay in inserting the reactor control rods and in operating the decay heat relief valve. The peak disturbances in this case were:

Coolant pressure rise	+300 p s i.
Coolant temperature rise	+26° F.
Coolant volume change	+38 2 cu ft.

The relief valve did not open since the plant was operating at 1,800 p.s.i. and the pressurizer relief valve set pressure was 2,300 p.s.i.

Most of the operating experience with the Shippingport plant has been accumulated in steady-state operation. There does not exist, therefore, a large amount of data on the operation of the reactor under transient conditions. The transient data that are available fall into two categories: (1) inadvertent transients and (2) specific load transient tests. The first type of transient has resulted in load changes outside of design limits, and the available data are limited to the full-load loss cases described above. The second type of transient has explored the reactor performance for design range transients only. The available data from these specific load transient tests are summarized in Table VIII–4.

The stability of the core with respect to xenon oscillations was determined at 480 equivalent full power hours by deliberately establishing a xenon oscillation and then following the reactor response over approximately two cycles. A slight divergent trend was detected. Analytic studies indicate that the oscillation amplitude increases or decreases in time depending upon the following parameters: power level, temperature coefficient, reactor lifetime history, and height of the controlling rod group.

Neutron detectors gave more immediate indication of the oscillation than thermocouples. Incidentally, a strong correlation was shown to exist between change in power in a given quadrant of the core and the resulting change in average boiler temperature for the loop corresponding to that section of the core. Variations in the hot leg temperatures of the loops indicated that cross flow among the core regions was not sufficient to provide a uniform hot leg temperature in all the loops.

2. Mechanical Component Failures

Shippingport has had several mechanical component difficulties. In June 1958, the stator in one of the four main coolant pumps was cut circumferentially by the lower end of the rotor stack. It is believed that operation of the pump at low system pressure caused the stator can to collapse. The removal of the

TABLE VIII–4 —PWR LOAD SWINGS—REACTOR PLANT RESPONSE DATA

Line No	Load swings in gross generated Mw		Rate of load swing	Peak avg temp change (F°)		Pressurizer volume surge (cu ft)		Pressurizer pressure surge (p s i)		Rod motion (number of times)		Pressurizer spray in use	
	3 loop	4-loop		3-loop	4 loop	3 loop	4 loop	3 loop	4 loop	3-loop	4 loop	3 loop	4 loop
1	5–20	5–20	3 Mw/sec		−4 8	−20	−16	−70	−60		1	No	No
2	25–40	25–40	3 Mw/sec		−3 2	−10 5	−12	−50	−42		0		No
3	50–62		3 Mw/sec			−6		−40					
4	61–46	60–45	3 Mw/sec		+4	+13	+16	+73	+70		1		Yes
5	40–25	40–25	3 Mw/sec		+3 7	+16	+16	+73	+85		0		Yes
6	20–0	21–5	3 Mw/sec		+5 2	+24	+20	+85	+90		0		Yes
7	62–46	60–40	3 Mw/sec		+3 1	+12		+100		1	0	No	No
8	40–24	40–25	3 Mw/sec		+3 7	+10 5	+12	+100	+90	0	0	No	No
9	23–8	22–5	3 Mw/sec		+3 1	+16 5		+110	+130				No
10		5–24	25 Mw/min		−6 2		−24 5						No
11		24–43	25 Mw/min		−6		−20				1		No
12		45–62	25 Mw/min		−2 8								
13		61–43	25 Mw/min		+3 6		+16		+100		0		Yes
14	43–23	42–22	25 Mw/min		+6 4	+13	+28	+120	+100		1	No	Yes
15		22–5	25 Mw/min		+4 3		+19 5		+105				Yes
16		6–23	25 Mw/min		−6 6		−32		−110		1		No
17		61–39	25 Mw/min		+4 2		+10 5		+95		1		No
18		43–21	25 Mw/min		+5 4		+16		+130		1		No
19		22–5	25 Mw/min		+4 7		+13		+110		0		No

pump motor, installation of the spare in the casing, and closing of the container required only two days. However, five additional days were required to replace piping, power cables, and catwalks, and to conduct hydrostatic and electrical checkout tests prior to operation of the spare unit.

During the early days of power operation, spurious motions of some of the fail-as-is type hydraulic valves occurred. A test was formulated to determine the causes and the conclusions were that two types of valve drifts were encountered:

(1) Valves drifted from the closed position when the water flash was vented. Air is normally directed into the top of the water flask to operate the valves.

(2) Valves bounced from both the open and closed positions during the operation of other hydraulic valves.

During initial plant operation it was found that both of the self-actuated pressurizer steam relief valves were leaking. Tests at several pressures indicated that the leakage may have been caused by thermal distortion or warpage. In September 1958, the primary plant operating pressure was reduced from 2,000 to 1,800 p.s.i. but the 2,300 p.s.i. set pressure for the relief valve was not altered. During subsequent plant operation, the primary plant leak rate has decreased from an average of 30 to 10 g.p.h. indicating that the valve leakage had been reduced substantially as a result of the spread between operating pressure and set pressure. The primary pressure was reduced to 1,800 p.s.i. and the average primary coolant temperature was decreased from 523 to 500° F. to bring to a safe limit the metallurgical stresses in the type 410 stainless steel housing of the control rod mechanisms.

On February 3, 1958, leaks were discovered in several tubes of one of the four steam generators between the secondary face of the inlet end tube sheet and the first tube baffle. Examination of the removed tube sections revealed that failure was caused by stress corrosion probably resulting from a combination of steam blanketing and boiler water chemistry out-of-specification with respect to free hydroxide. During initial plant operation, excessive blowdown was required to reduce silica concentration and at this time the boiler water chemistry could not be maintained within limits. To prevent steam blanketing in the future, two additional risers were installed between the inlet tube sheet and the first existing riser. In addition, sampling lines were installed in the defective area to determine the presence of a steam bubble and chemical concentrations. Other changes included modifications in the existing boiler sampling connections, the addition of thermocouples, and changes in the boiler water chemistry control to eliminate any possibility of free hydroxide. The defective tubes were successfully plugged with blind nipples and no difficulty has been experienced since the steam generator went back into service on May 14.

Inspection of the turbine moisture separator in late November 1958, indicated that the internals had been completely destroyed. Further inspection at a later date revealed that pieces of the turbine moisture separator were lodged in the turbine low pressure blades. The possible explanation for this failure is excessive mechanical vibration.

3. Radiation Levels

In general, measurements of the radiation intensities within the reactor plant container at various power levels showed these to be somewhat lower than predicted. A gradual increase in proportion to length of plant operation was observed for the first few months, after which steady-state levels were approached.

At Full Power

Readings on contact with the 18-inch primary coolant piping in each boiler chamber, after steady-state levels had been reached, showed values ranging from 6 r/hr to 14 r/hr. The radiation intensities dropped off rapidly away from the line. At the sight glass in each boiler chamber, radiation intensities varied from 150 to 250 mr/hr. Levels registered at various points in the containers are shown in Figure VIII–1 (each level shown is a maximum for that locality).

In the auxiliary chamber the maximum radiation level in the flash and blow-off tank compartment was 1 mr/hr. In the pressurizer compartment the levels ranged from 100 mr/hr to 400 mr/hr.

The reactor chamber was not entered during operation but readings made at the entrances showed a maximum detectable gamma flux of 6 r/hr. At the same point the thermal neutron flux was approximately 24,000 nv.

The auxiliary chamber and the purification valve access cubicle in each of the two boiler chambers are accessible at any time for maintenance or valving. The radiation intensities in these areas at full power vary between 0.03 mr/hr and 15 mr/hr, excluding the bottom level where in some areas a maximum of 150 mr/hr has been found.

50 Percent Power Level

The maximum radiation intensities on contact with the primary coolant lines ranged from 4 to 8 r/hr; whereas at the boiler sight glass the intensities were 40 to 90 mr/hr.

Isolated Loop—Full Power

Levels registered at various points in the boiler chamber after one loop has been shutdown with the reactor at full power, is shown in Figure VIII–2.

Shutdown Levels

The radiation levels recorded after reactor shutdown are shown in Figure VIII–3. The levels shown correspond to different lengths of time after shutdown. The values given for points in the 1A, 1B, and 1D boiler chambers were determined 15 hours after shutdown, while the values shown in the 1C chamber were determined 30 hours after shutdown. The radiation levels in the reactor chamber were determined approximately 96 hours after shutdown.

Contamination Levels

Contamination levels throughout the reactor plant have increased steadily as a result of increased maintenance, penetration of the primary coolant system, and the draining of primary lines and valves. Beta and gamma activities have been detected but no alpha contamination has occurred.

In June 1958, the primary side of one of the boilers was opened to inspect tube banks in the heat exchanger; the radiation associated with the contamination products was a maximum of 120 mr/hr located inside the primary coolant line. In the same month, a primary coolant pump motor failed and had to be replaced. The maximum contamination, measured on the impeller of the pump, was 500 mr/hr. The higher activity level in the pump is partially explained by the restriction to coolant flow which permits crud buildup.

4. Corrosion Products in Primary Coolant

The level of fission products in the primary coolant has been higher than was expected on the basis of observed levels of uranium contamination in structural materials, although the level is considerably below design levels. On the basis of experimental evidence, it is suspected that one or more defected UO_2 blanket rods exist.

Shippingport is the first reactor plant to be maintained at a pH of 9.5–10.5 by the use of LiOH addition. One disadvantage of using LiOH is the formation of tritium which is a potential hazard from the viewpoint of ingestion as a gas. Its maximum permissible concentration in water is relatively high, however, since the tritium is present as water. The reference specification for pH has been very easy to maintain and small increments of LiOH had to be added on only few occasions. The concentration of insoluble corrosion products (crud) has been approximately 3 ppb but

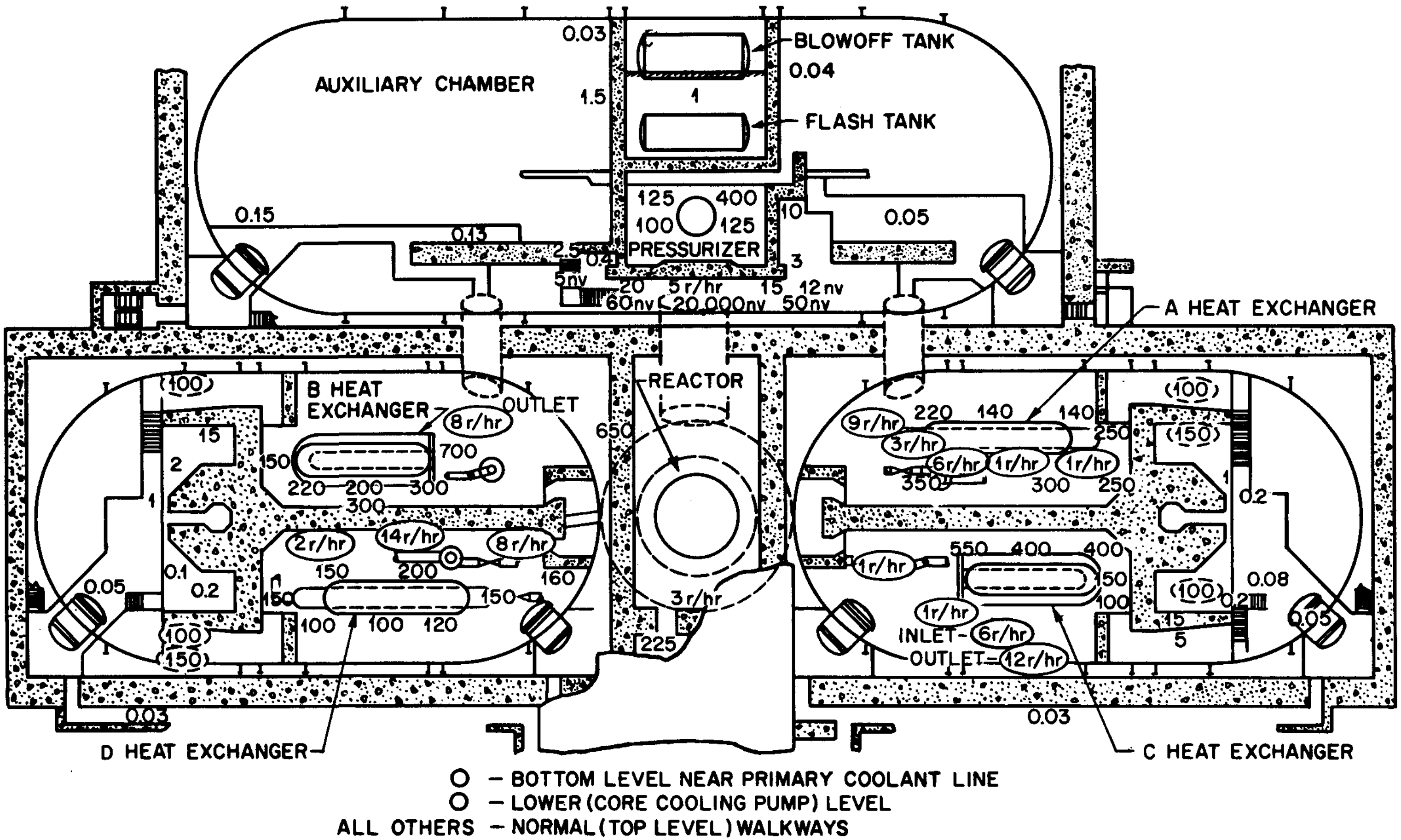


FIGURE VIII-1.—Maximum levels detected at 100 percent power in reactor plant containers (nv=thermal neutrons/cm²—sec; all other readings given in mr/hr unless designated as r/hr).

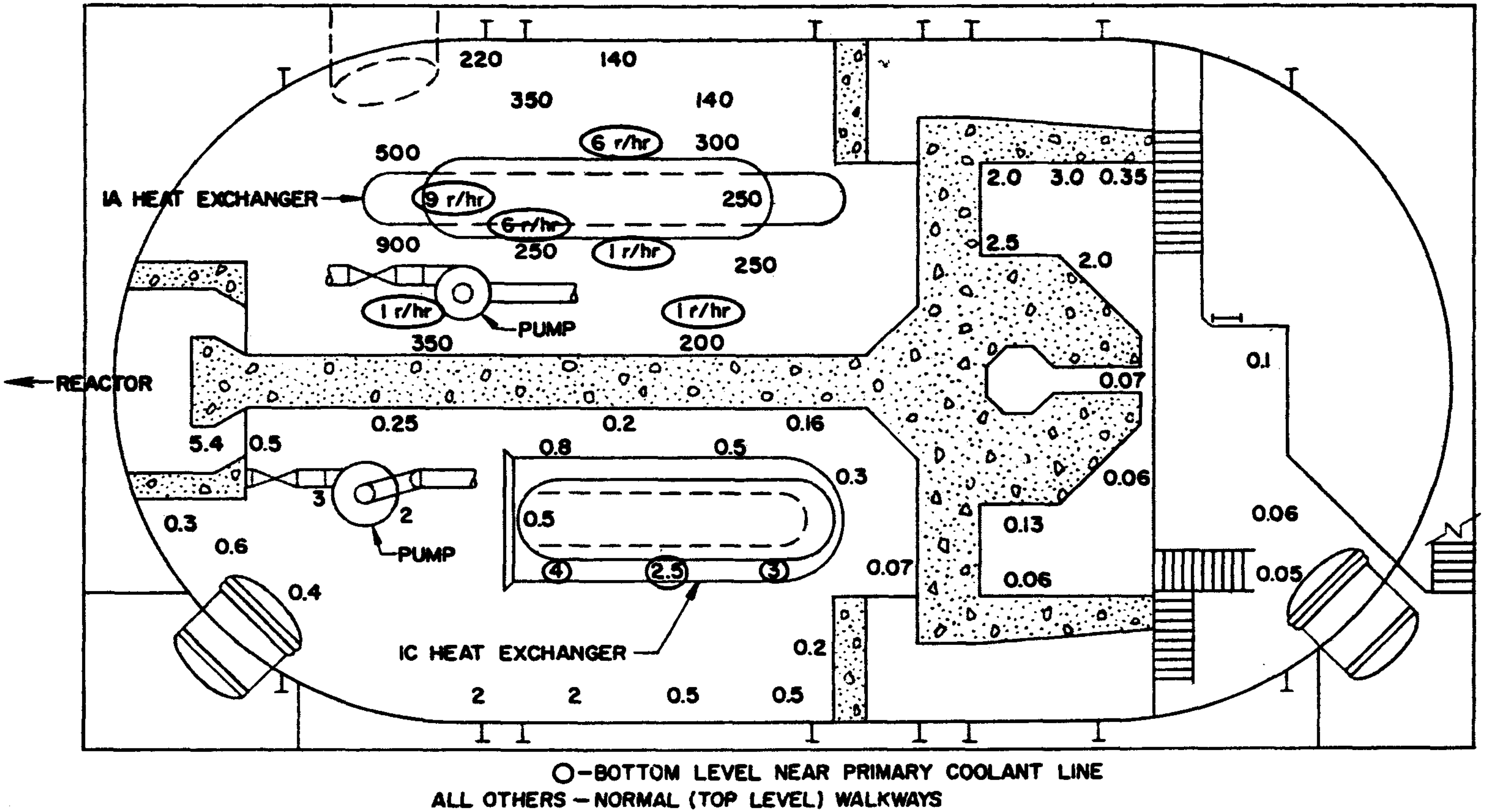


FIGURE VIII-2.—Maximum detectable levels in east boiler chamber 48 hr after isolation of 1C boiler; 1C boiler isolated; 1A loop in service (all readings given in mr/hr unless designated as r/hr).

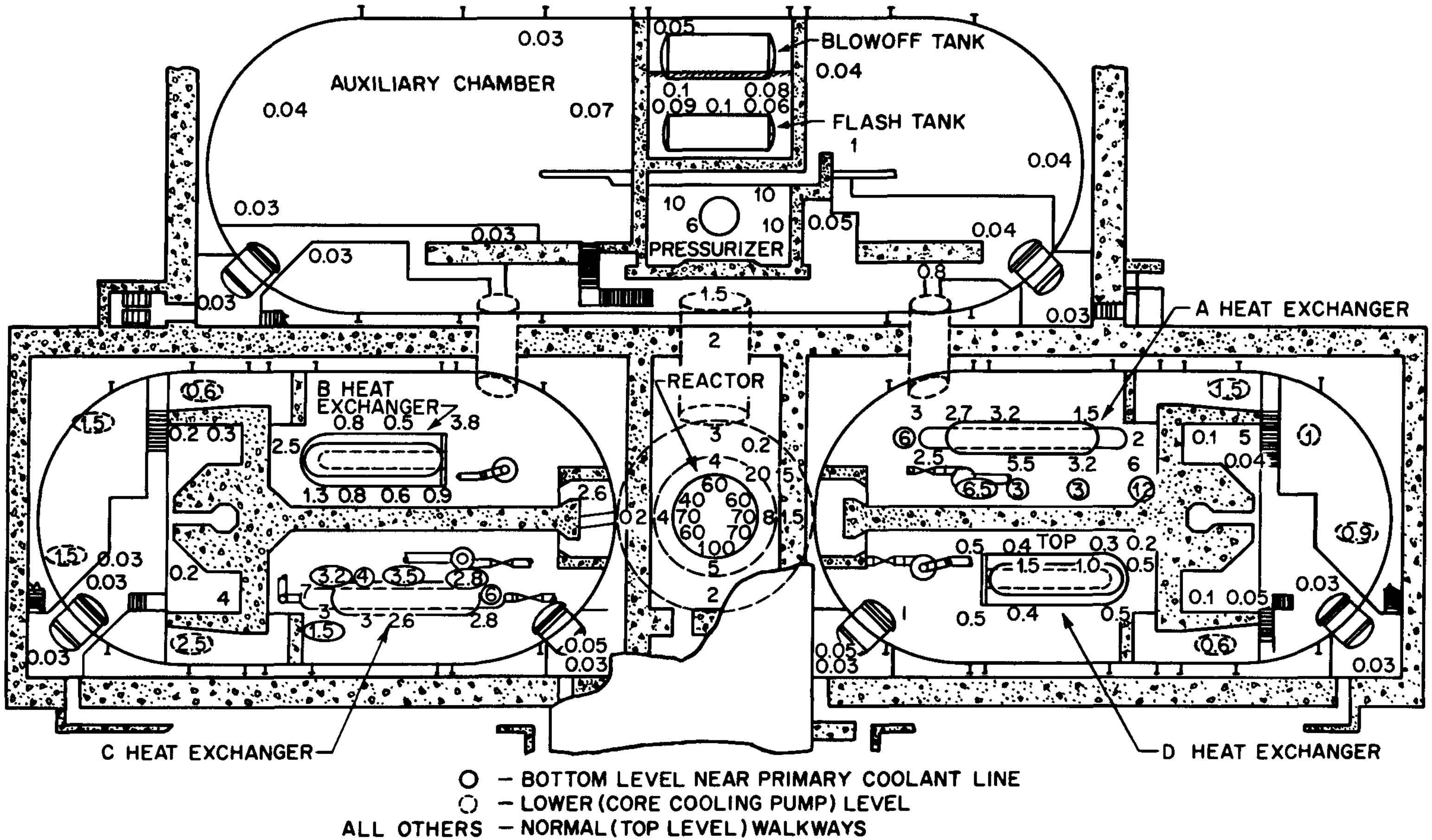


FIGURE VIII–3.—Radiation levels of reactor plant containers after shutdown at 1700 EFPH; A, B, and D loops surveyed 15 hr after shutdown; C loop surveyed 30 hr after shutdown (all readings given in mr/hr).

on one occasion did increase to 60 ppb after a cold startup. However, the level returned to 3 ppb in a few days.

The specification for dissolved hydrogen concentration requires 25 to 50 cc/kg of water. The instrumentation designed to monitor and record the hydrogen content continuously did not function properly and therefore a laboratory analytical procedure is used daily. During the first nine months of operation the hydrogen concentration was below specification approximately 15 percent of the time, the lowest determined value being 13cc/Kg, but nitric acid synthesis had not been detected during this period.

5. Conclusion

Most of the data presented on the operating experience of Shippingport has been extracted from WAPD-BT-12, Bettis Technical Review, April 1959, *One Year of Operating Experience at Shippingport.* The following is quoted from the conclusions stated in the report:

> The general operational performance of the Shippingport Atomic Power Station during the first year has compared quite favorably with that of conventional coal-fired stations. Initial startup and loading to full capacity proceeded on schedule without any major difficulties or equipment failures; those encountered since operation was begun have not been out of proportion with those of other conventional stations.
>
> The operating experience to date on the Shippingport station indicates that it is simpler to operate than an equivalent coal-fired station. The principal reason for this is the minimum of operator actions required for controlling the reactor. The station as a whole is extremely stable and highly responsive to load changes. It can be started up and shut down much more rapidly than a coal-fired station and can be operated in synchronism with the utility system at zero net output in readiness for instant load demands for extended periods with no adverse effects. All these factors make it an excellent load peaking station.

The 334,970 MWe-hours generated by June 1, 1959, through nuclear power supports the conclusion reached by the report.

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IX. CONSTRUCTION AND OPERATION SCHEDULES

This section contains the following curves for the pressurized water reactor systems that have been built, or are under construction or design:

Figure IX–1.—Schedule for Pressurized Water Reactors.

Figure IX–2.—Operating History of SM–1.

Figure IX–3.—Operating History of Shippingport Atomic Power Station.

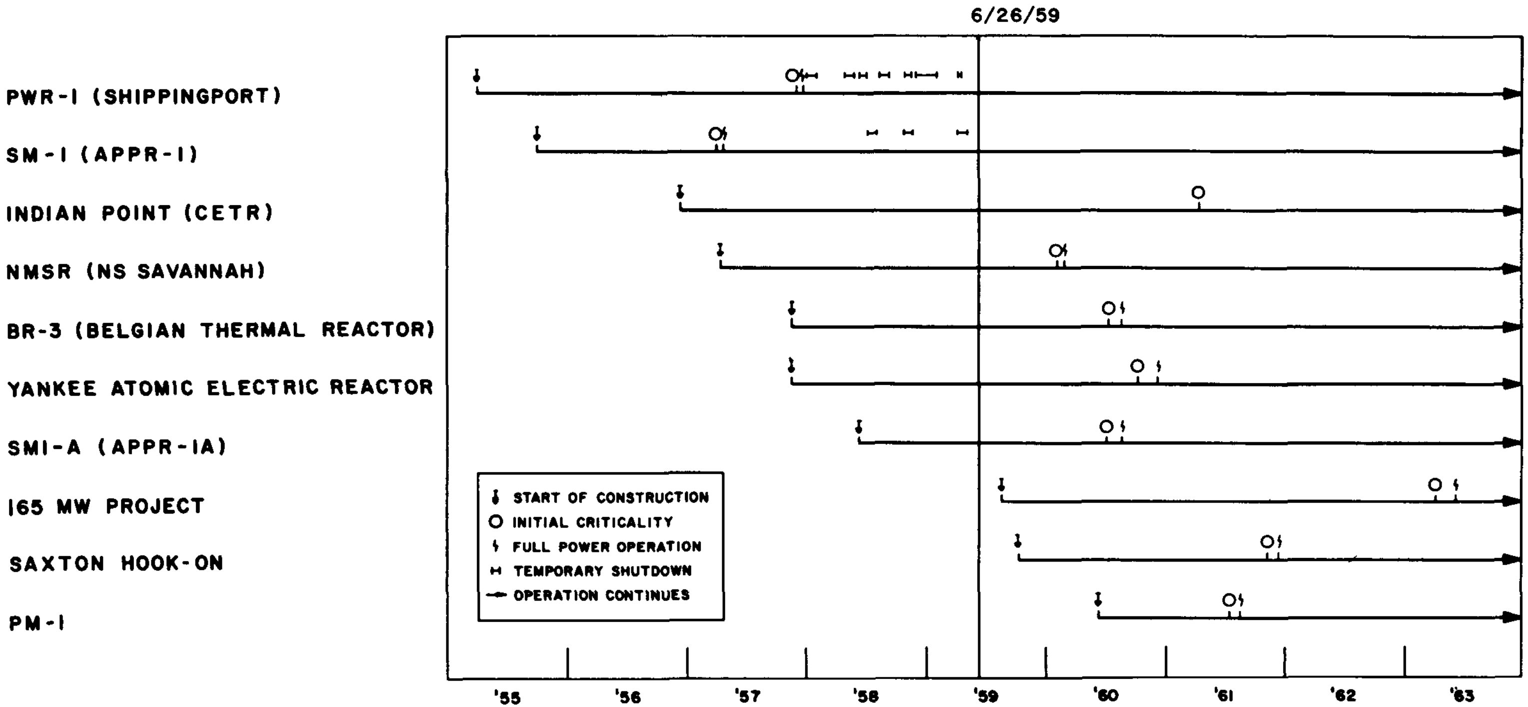

FIGURE IX–1.—Schedule for pressurized water reactors.

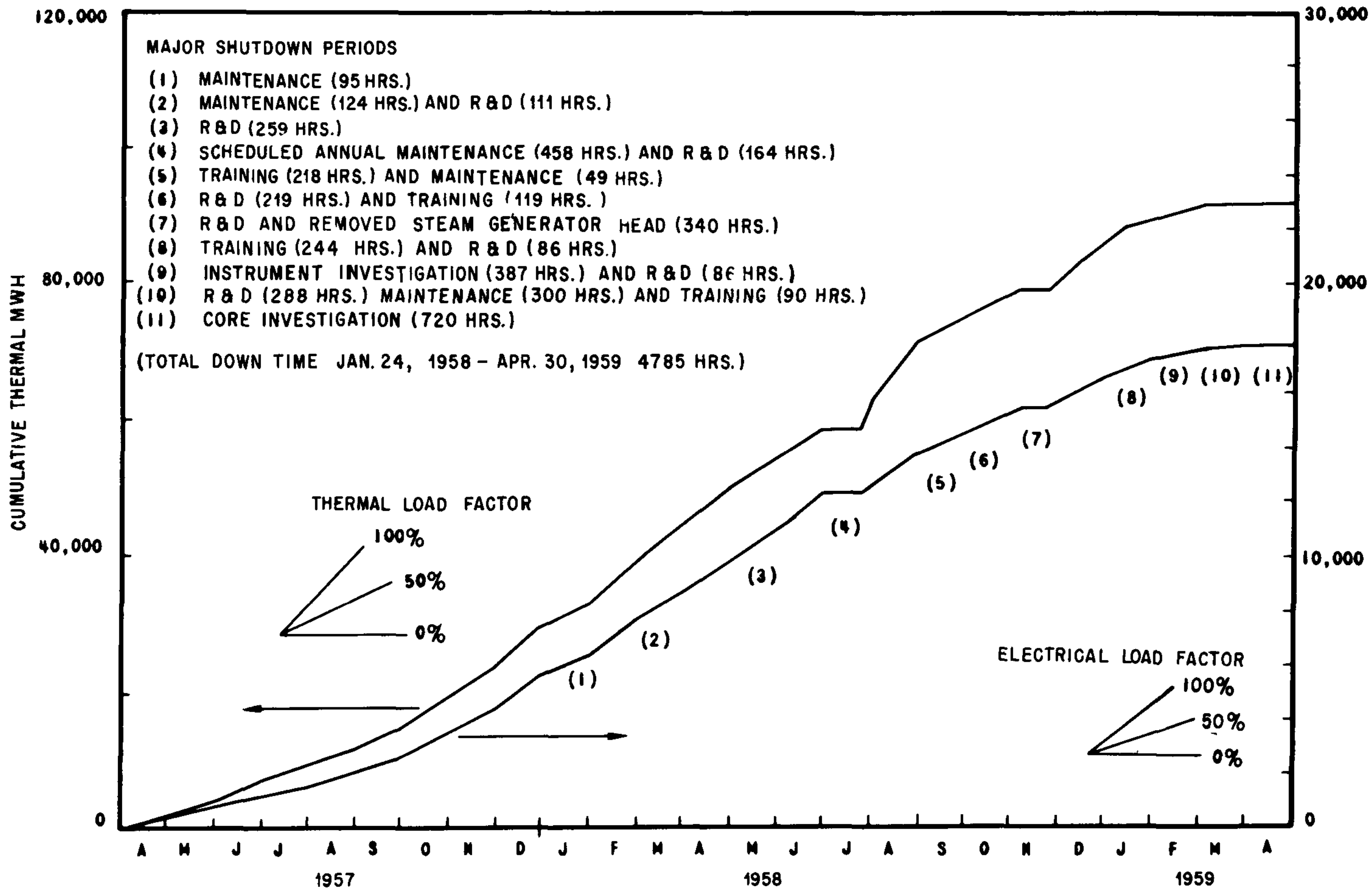

FIGURE IX-2.—Operating history of APPR-1.

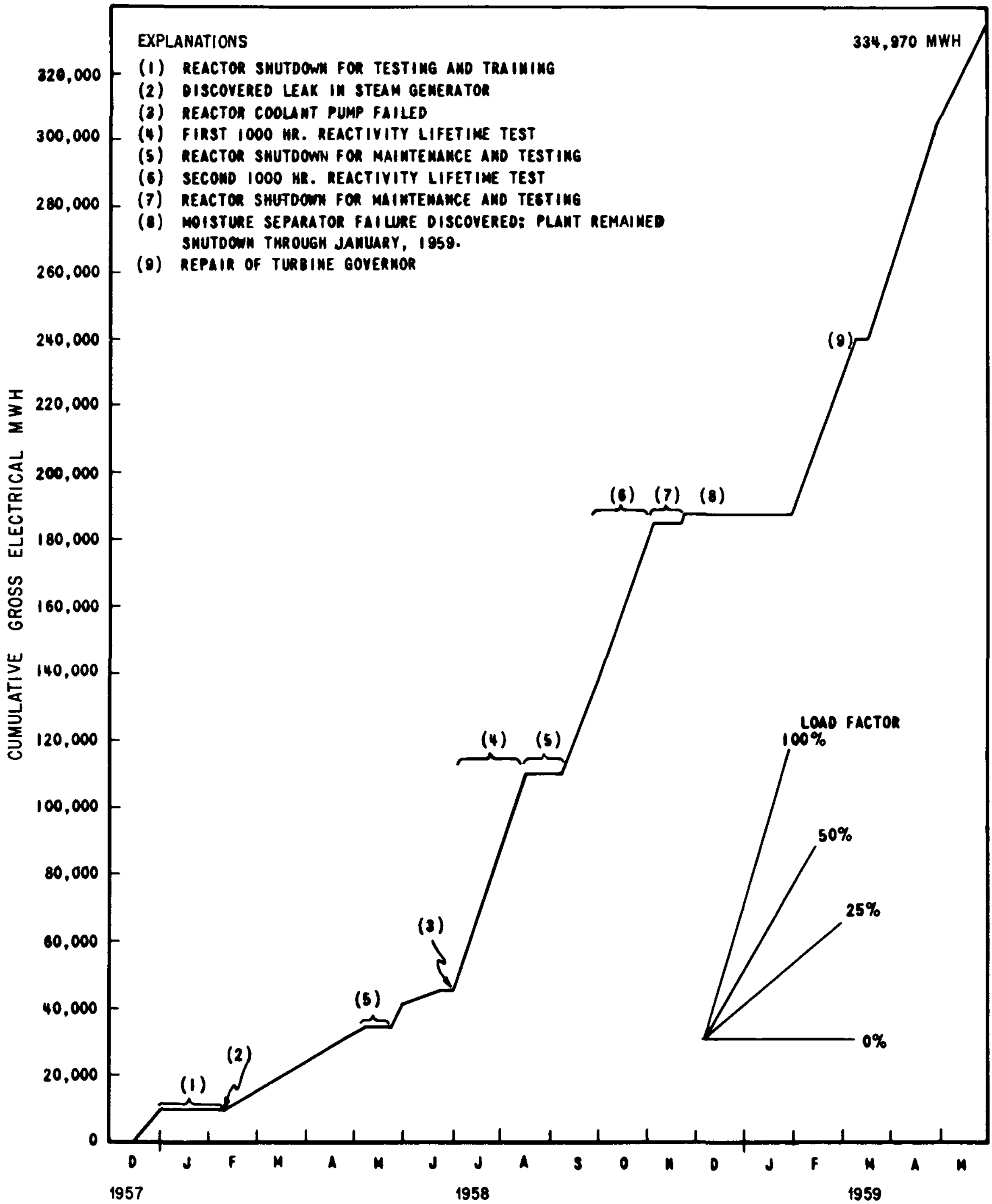

FIGURE IX–3.—Operating history of Shippingport Atomic Power Station.

X. PRESENT LIMITATIONS AND PROBLEMS

Some of the present day design limitations of the pressurized water reactor plant are listed and described in this section. These limitations are not necessarily inherent in the system but represent, rather, inadequacies of our present technology and point out areas for potential improvement.

It should be noted that all of the research and development work now underway, as described in Chapter IV, identifies present problem areas.

A. THE REACTOR CORE

1. Hot channel factors, when both nuclear and mechanical contributions are considered, are now no lower than a range of from 3.3 to 4.0 for F_Q and 2.3 to 2.7 for $F_{\Delta T}$. These factors are for multicycle, multiregion core loadings having a burn up of 5,000 to 10,000 MWD/T per cycle, utilizing control rod followers but no chemical shim. The reactors chosen for the present AEC-PWR study have an $F_Q=3.7$ and $F_{\Delta T}=2.4$, which are the same values specified for the 165 MW Project.

2. Present methods for calculating the lifetime of a core results in uncertainties in the range of 25 percent.

3. At present, the requirement for no bulk boiling in the core places a limitation on the thermal design of the core.

4. The limit for maximum center fuel temperature is set somewhere between 4,500 and 5,000° F. and is based on the design limitation that melting of the center of the fuel element shall not occur. Experiments at Hanford and Chalk River indicate that this limit corresponds to a KW/ft-of-rod value of 15.

5. Maximum heat flux is limited to about two-thirds of the burnout heat flux. In this study, the maximum heat flux used is less than one-half of burnout.

6. Other problem areas involving the reactor core are being studied under the research and development programs described in Chapter IV. These R & D programs include:

a. More accurate and precise measurements of nuclear constants such as cross sections, η and α for U–233, and neutron absorption resonances for nuclear materials.

b. Development of improved computational techniques, including simplified codes to study three dimensional burnup and more detailed consideration of transport theory.

c. Critical experiments for multiregion loadings, low enrichment cores, plutonium fuels, and for xenon oscillation.

d. Studies on reducing hot channel factors and control requirements by fuel cycling, power shaping with control rods and fuel bearing control rod extensions, and by chemical shim for power operation.

e. Development of improved analytical techniques for the treatment of the spatial dependence of reactor kinetics in the reactor core.

f. Study of autocorrelation techniques to observe and discount the effects of "noise" in nuclear instrumentation.

g. Further studies of the properties of UO_2.

h. Development of improved fuel pelleting techniques and of advanced fuel fabrication methods such as swaging and extrusion.

i. Development of collapsible clad fuel elements and of improved cladding materials such as improved zirconium alloys, iron-aluminum alloys, and sintered aluminum powders.

j. Studies of alternate control rod materials and control rod fabrication techniques.

k. Development of instrumentation and control systems to detect boiling in the reactor core and to detect and compensate for xenon oscillations.

B. THE PRIMARY REACTOR SYSTEM

1. The reactor vessel size is limited by the availability of forgings. About 10–11 inches is the maximum thickness now available. This thickness corresponds to an I. D. of about 142 inches for a vessel built to ASME code requirements with a design pressure of 2,500 p.s.i.

2. Transportability is also a limiting factor in reactor vessel size. A vessel with an O. D. of 168 inches is about the largest that can be shipped by rail. Larger vessels can be shipped by water.

3. As pointed out in Chapter IV, the effect of irradiation damage to pressure vessel materials is a subject of considerable concern. Experimental evidence indicates a reduction in impact strength in low alloy steel for integrated fast fluxes as low as 5×10^{18} nvt at a temperature of 500° F.

4. The total number of heat transfer loops that may be added to a single reactor vessel is limited by the space requirements for locating the required number of nozzles into the vessel and by loop layout.

5. Existing manufacturing and test facilities limit canned motor pumps to pumping capacities up to 35,000 g.p.m. and 85 p.s.i. However, present pumps cannot vent during operation or at any time that the temperature is below 200° F. and they must maintain a minimum pressure to avoid stator can collapse. Operating instructions for large canned motor primary pumps, for example, usually specify a minimum pressure of about 200 p.s.i. at the pump suction inlet.

6. The maximum diameter of a steam generator tube sheet which can presently be fabricated is limited by available forgings to about 96 inches. This places an upper limit on the number of steam generator tubes which may be used.

7. The maximum length of steam generator tubes is at present about 60 feet. This, combined with the preceding item 6, places an upper limit on the heat transfer surface area per steam generator.

8. The presently permissible maximum velocity of primary fluid through the steam generator tubes is about 20 fps. This, combined with item 6 above, places an upper limit on the mass flow rate per loop.

9. Other work mentioned in Chapter IV which is currently underway on primary reactor system problems includes:

a. Studies of the feasibility of using more economical materials in the primary system; for example, the use of low alloy steels for heat exchanger tubing and main piping.

b. Development of inorganic ion exchange resins that will be radiation stable and thermally stable at reactor temperatures.

c. Studies of advances in reactor vessel design such as gasketed closures instead of seal welds, improved nozzle penetration design, and the use of "multilayer" construction.

d. Development of canned motor pumps of increased size and performance and improvement in bearings, electrical insulation, and motor cooling.

e. Study of alternatives to the canned motor pump such as a shaft sealed pump.

C. OTHER COMPONENTS

1. The maximum permissible plate thickness for the vapor container is presently about 1.5 inches, if designed to ASME Code, because of stress relief requirements.

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U.S GOVERNMENT PRINTING OFFICE: 1960 O—540997