UNCLASSIFIED

TID-2507(DEL.)

NUCLEAR SCIENCE AND TECHNOLOGY

Volume 1A, Issue 2

August 1955

Photostat Price $ 18.30

Microfilm Price $ 6.00

Available from the
Office of Technical Services
Department of Commerce
Washington 25, D. C.

ATOMIC ENERGY COMMISSION
UNITED STATES OF AMERICA

~~RESTRICTED DATA~~

Technical Information Service, Oak Ridge, Tennessee

UNCLASSIFIED

DECLASSIFIED

NUCLEAR SCIENCE AND TECHNOLOGY

A Journal Devoted to the Science and Technology of Nuclear Reactors and to Related Subjects

Contents

Special Announcement

The Editors have been informed that the following film has been made available:

"Motion Picture Progress Report No. 1 on the SGR Program," Feb. 1, 1955, 18-min. 16-mm color sound film. Contract AT(04-3)-49.
North American Aviation, Inc.
NAA-SR-1136 (Confidential-Restricted Data)

The design of the Sodium Reactor Experiment and its associated facilities is described in animated sequences. The fabrication, assembly, and operation of reactor components, such as the sodium pump, safety system, fuel rods, and moderator cans, are shown. Various aspects of the neutron physics and fuel-element development program are covered.

The film may be obtained from the following: U. M. Staebler, Civilian Power Reactors Branch, Division of Reactor Development, Atomic Energy Commission, Washington; Howard C. Baldwin, Director, Information Control Division, Chicago Operations Office; Edwin E. Stokely, Assistant to the Manager for Public Education, Oak Ridge Operations Office; Elton P. Lord, Information Service and Control Officer, Atomic Energy Commission, Washington; Miss Grace Wells, Information Specialist, New York Operations Office; Arthur R. Lee, Director of Information Division, Idaho Operations Office, and Douglas M. Frame, Assistant to the Manager, San Francisco Operations Office.

ARMY PACKAGE POWER REACTOR

Introduction

A. L. BOCH and R. S. LIVINGSTON
Oak Ridge National Laboratory

March 24, 1955

Ever since the concept of the "package" reactor was first advanced in 1952 by A. M. Weinberg of the Oak Ridge National Laboratory (ORNL) and L. R. Hafstad of the Atomic Energy Commission (AEC), a steadily increasing interest in this development has been evidenced by private industry. The package concept connotes a small compact reactor power plant designed to supply power reliably in relatively inaccessible or remote areas.

Early in 1953 a small group was established at ORNL to review the various reactor types and to select the one that appeared to lend itself best to the design of a thoroughly reliable and yet suitably inexpensive system. It appeared at an early date that such a package reactor plant as that being studied at ORNL would meet rather successfully the requirements of the Army for powering aircraft control and warning stations in the arctic regions. Subsequently, the project was so oriented.

During the spring of 1954 the AEC asked publicly for expressions of interest by private industry with respect to the possibility of constructing a package reactor plant on the basis of a lump-sum contract. Opinions were also solicited on what private industry would consider a practical guarantee period for the plant. The response was most gratifying in that approximately 100 firms replied with varying degrees of interest. A contractor-selection board was appointed by the AEC to evaluate the replies and to review the qualifications of the many interested firms.

On Aug. 19, 1954, invitations for proposals on the Army Package Power Reactor (APPR) were sent to 33 qualified contractors. A 90-day period was allowed for the preparation of proposals, Nov. 19, 1954, being the date established for the opening of the bids.

Several pioneering features unique to the nuclear power field were stipulated in the invitations; they were as follows:

1. The contractor is to assume full responsibility for the over-all design and construction of the facility.

2. The operation of the plant is to be guaranteed to the extent that a 700-hr performance test is required. The contractor is required to operate the plant in a safe manner for a continuous 700-hr period at an electric generation rate specified by the AEC but not to exceed the rating of the plant. During the 700-hr test a maximum of 40 hr is allowed for outage, i.e., time during which the plant cannot satisfy the requirements of the designated load.

3. A 6-month operating period under less

[Editors' Note: In processing the APPR section manuscript, the Editors were compelled to eliminate a number of figures that appeared in the original complete proposals so that problems of reproduction would not unduly delay publication.]

stringent conditions with payments geared to performance is also included.

4. A 2-year period is allowed for the design and construction of the plant. An additional year is allowed for testing and completing the 700-hr performance tests, and an additional 7 months is allowed for the 6-month test.

5. All the above conditions are to be fulfilled on the basis of a lump-sum contract.

On Nov. 19, 1954, 18 companies submitted impressively detailed, well thought out proposals for the APPR. After a thorough review of the proposals by the contractor-selection board and extensive comparisons of the technical features of the various proposals, the contract was awarded on Dec. 10, 1954, to the American Locomotive Co. for the sum of $2,096,753. Their proposal was judged to be the most favorable to the government from the standpoint of price, excellence of design, and responsiveness to other terms of the bid invitation. The price submitted by the interested companies ranged from $2,096,753 to $6,900,000. Twelve of the proposals were under $4,000,000, the second and third lowest bids being $2,462,355 and $3,037,586, respectively.

Because of the general excellence, and in many cases originality, of the proposals, it was deemed advisable to contact all 18 firms and request submission of their proposals for publication in *Nuclear Science and Technology.*

With the awarding of the contract for the APPR, the AEC agreed that ORNL should conduct certain development programs in support of the project. These programs and their current status are outlined briefly below:

1. Critical experiments will be performed to substantiate the nuclear calculations on the critical mass, control-rod effectiveness, and flux distribution. This program is currently under way.

2. The program for the development of the fuel plates, consisting of a matrix of UO_2, B_4C, and stainless steel and clad on all sides with stainless steel, is continuing. A full-size fuel assembly for use in a Submarine Thermal Reactor (STR) irradiation test has been successfully fabricated.

3. The irradiation tests of APPR type fuel elements designed to evaluate reliability and structural soundness at high burn-up is well under way. The assembly for the STR is now operating in the Mark I core at the National Reactor Testing Station, and results are expected during the summer of 1955. The assemblies for irradiations in the Materials Testing Reactor core are being fabricated.

4. Reactor-simulator tests to reevaluate the stability of the APPR with all the final parameters are in progress.

ARMY PACKAGE POWER REACTOR

American Locomotive Co. Proposal

KENNETH KASSCHAU

March 11, 1955

ABSTRACT

The American Locomotive Co. proposal for the Army Package Power Reactor is described. The reactor will be operated at the specified 10-megawatt power level and will have a net electrical output of 1825 kw. Conservative design has been employed throughout, and the end objective of remote installation has been borne in mind. The steam generator, which requires a 33° temperature difference to produce the superheated steam, is vertically mounted so that thermal convection will provide emergency cooling. An iron-water shield is provided which affords considerable weight saving if the aggregate for a concrete shield is lacking at a remote site. The main support member of the reactor vessel is the inner wall of the shield tank. This support, in combination with the vertical type steam generator, allows for very short pipe connections and leaves essentially no thermal expansion stresses in the pipe. A double-walled steel shell, 32 ft in diameter and 60 ft high with 2 ft of concrete between the linings, provides containment against nuclear accidents. When the reactor is refueled, spent elements may be removed under water to a deep water-filled storage pit outside the container.

1. INTRODUCTION

In a recent article in this journal,[1] a conceptual design for the Army Package Power Reactor (APPR) was discussed. The early history and background of the project is covered in that article, which summarizes the contents of Report ORNL-1613.[2] On Aug. 19, 1954, the Army Reactors Branch of the Atomic Energy Commission issued invitations for proposals for the design and construction of a prototype of the reactor to be located at Fort Belvoir, Va. The American Locomotive Company (ALCO) submitted a proposal, and on Dec. 10, 1954, it was awarded the contract. This article is a condensation of that proposal.[3] The design, construction, and initial operation of the plant are to be completed before Dec. 10, 1957.

The plant described in references 1 and 2 was designed to meet the needs and site conditions of a remote military base. Since the prototype reactor is to be constructed at a site in the United States, some of the design requirements were changed to meet these needs. In particular, containment of the maximum credible nuclear incident must be provided. There is no load requirement for heating, but all the useful output of the plant is to be converted into electricity.

Furthermore, the reactor is to be used as a training facility for troops and specialists who might eventually be required to operate and service remote plants. The original requirement that all components be transportable by air is still valid even though the site is not remote.

2. GENERAL CONSIDERATIONS

2.1 High Lights of the ALCO Design

The ALCO design for the APPR is an 1825-kw generating plant embodying modern large-plant features that provide good efficiency even at moderate temperatures. Conservative design has been employed throughout, and reserve capacity has been furnished so that the plant components themselves will not materially reduce the net power output.

The reactor will be operated at the specified 10-megawatt power level, and the coolant temperatures will be limited to the values employed in the design described in Report ORNL-1613. If experience at the APPR and elsewhere indicates the possibility of increasing the operating level and temperature of the primary system, the plant output can be increased up to 20 per cent without overload.

The end objective of remote installation has been borne in mind in the design of all components, which meet the required size and weight limitations specified in the inquiry. In this connection special mention is made of the steam generator, which requires only a 33° temperature difference between the primary and secondary fluids to produce the superheated steam. The vertical mounting of the steam generator achieves satisfactory thermal-convection emergency cooling with a minimum elevation of the entire unit.

Shielding of the reactor with iron and water provides a pilot-model demonstration of a remote-location shield design. The aggregate for a concrete shield is lacking in some of the remote areas, thus necessitating transportation of a weight far in excess of the cement weight. Iron, fabricated conveniently in well-equipped shops, is easier to transport, faster to assemble, and is appreciably lighter than the concrete aggregate it replaces. The water region of the shield provides a logical and convenient storage place for spent fuel elements, although this system is not employed in the current proposal for reasons noted below.

In the event of a major svstem rupture, containment of all contaminated vapor is provided by a container 32 ft in diameter and 60 ft high. The enclosed volume will contain all the steam generated by flashing of the superheated primary and secondary system water volumes, supplemented by fission-product-induced evaporation for 2 hr after the accident. An outer steel shell $^3/_4$ in. thick constitutes the vaportight container, which is reinforced and protected from missiles by a concrete lining 2 ft thick. A thin inner steel lining makes decontamination possible. The shielding effect of the container wall has been accounted for in the design of the shield.

The handling of spent fuel departs from the prototype concept in order to improve the flexibility of the installation. When the reactor is refueled, spent elements are removed, under water, to a deep water-filled storage pit outside the vapor container. Not only are fuel elements at all times accessible from the outside for removal to a reprocessing plant, but it is also possible to return to the reactor a partially spent fuel element previously discharged.

2.2 Research and Development Program

ALCO proposes a broad, balanced research and development program in support of the reactor design. The pursuit of this program will increase the contribution that this reactor will make to over-all reactor technology. As a part of this program ALCO will review the entire flow sheet from the logistic point of view, assessing the value of each component by evaluating its contribution to cycle performance against its cost of transportation and assembly time.

2.3 Fabrication of Fuel Elements

This proposal is predicated on the use of fuel elements and control rods fabricated by the Oak Ridge National Laboratory, ORNL, as described in the invitation. Should developments indicate the desirability of reconsidering this decision at a later date, ALCO will willingly review the matter.

2.4 Anticipated Performance

With the reactor operating at its designed 10-megawatt level, the plant performance is as shown in Fig. 1, where performance characteristics corresponding to cooling-water temperatures of 70 and 80°F are shown. All the major qualities are given in Fig. 1 except the details of the auxiliary power requirements. For this purpose the auxiliary power, exclusive of heating and air conditioning, has been determined from manufacturers' data on the proposed equipment, giving a total auxiliary power load of 180 kw.

On this basis the output of the proposed power station will be as follows:

Back pressure, in. Hg	Gross generation, kw	Net output, kw	Net station heat rate, Btu/kw-hr
1.5	2105	1925	17,800
2.5	2005	1825	18,800

3. REACTOR COMPONENTS

3.1 Reactor Core and Vessel

The reactor is essentially unchanged from the design proposed in references 1 and 2. The core is contained within a reactor vessel 4 ft in diameter and 8 ft 5 in. high, similar to that described in Report ORNL-1613. Cooling water is introduced immediately below a diaphragm, which supports the reactor and serves to direct the water to the entrance plenum chamber at the bottom of the reactor vessel. From the bottom the water flows upward through the fuel elements to the exit plenum chamber, from which the exit nozzle leads the water to the steam generator.

3.2 Support for Reactor Vessel

A novel feature of the ALCO design for the APPR is the support of the reactor vessel. The main support member is the inner wall of the shield tank (see Fig. 2), which consists of a heavy-wall cylinder, strong and stable against overturning moments and not materially affected by temperature changes. The reactor is supported at a plane slightly below that of the outlet pipe connection. This arrangement essentially eliminates the relative vertical expansion of the reactor vessel, permitting the steam generator and accessories to be mounted on rigid supports at the same level as the reactor. This mounting location also results in short piping connections, thus minimizing the horizontal expansion due to temperature change. The horizontal expansion will be accommodated by laterally flexible columns underneath the steam generator, circulating pumps, and pressurizer. Such a support, in combination with the vertical type steam generator, permits a compact plant with the shortest possible piping connections and one that has essentially no thermal-expansion stresses in the pipe.

3.3 Control-rod Drive Mechanism

The control-rod drive mechanism in the ALCO proposal was based on the design described in references 1 and 2. However, a bottom-mounted control-rod drive system offers the possibility of great improvement in over-all reactor design. The chief advantage stems from the improved safety because it will render the blind-operated latch mechanism unnecessary. Under no circumstances can the control rods be removed with the reactor-vessel cover. A second important advantage is the more compact reactor structure, following elimination of the superstructure. Design and development of a bottom drive unit are being undertaken.

4. PRIMARY COOLANT SYSTEM

4.1 Primary Flow Circuit

The primary and secondary flow circuits are shown in Fig. 3. In the primary cooling cycle 4000 gal/min of pressurized water is circulated through a closed loop consisting of the reactor, the steam generator, and two circulating pumps. The primary circuit is maintained at 1200 psia to preclude boiling (saturation temperature 568°F). Water leaves the reactor at 450°F (being automatically held constant at this temperature for all loads) and enters the steam generator, passing through U tubes that are surrounded by water in the secondary circuit. The primary water is cooled to about 431°F at full load and, on leaving the steam generator, is returned to the reactor by means of a motor-driven pump.

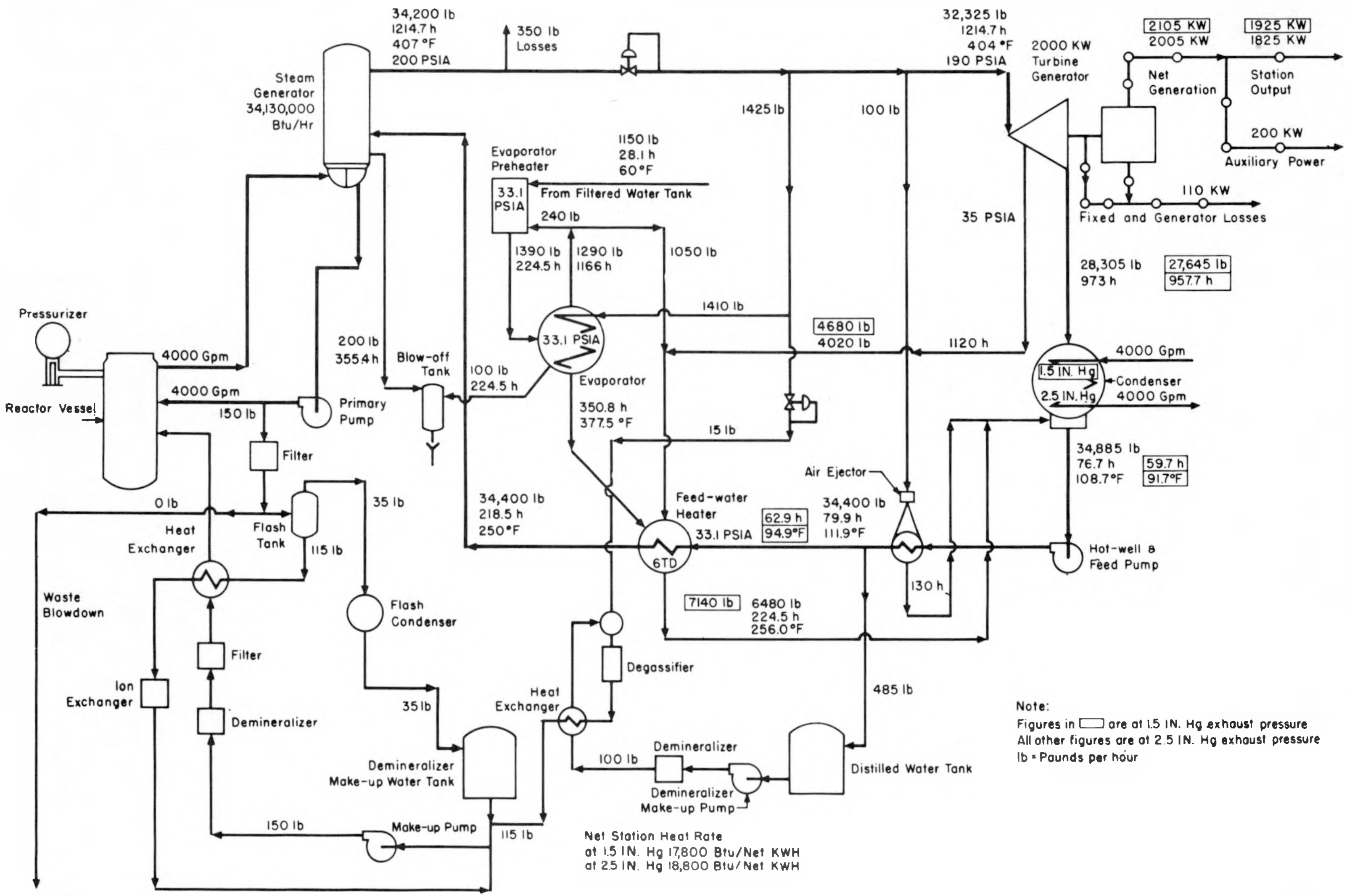


Fig. 1—Heat balance diagram.

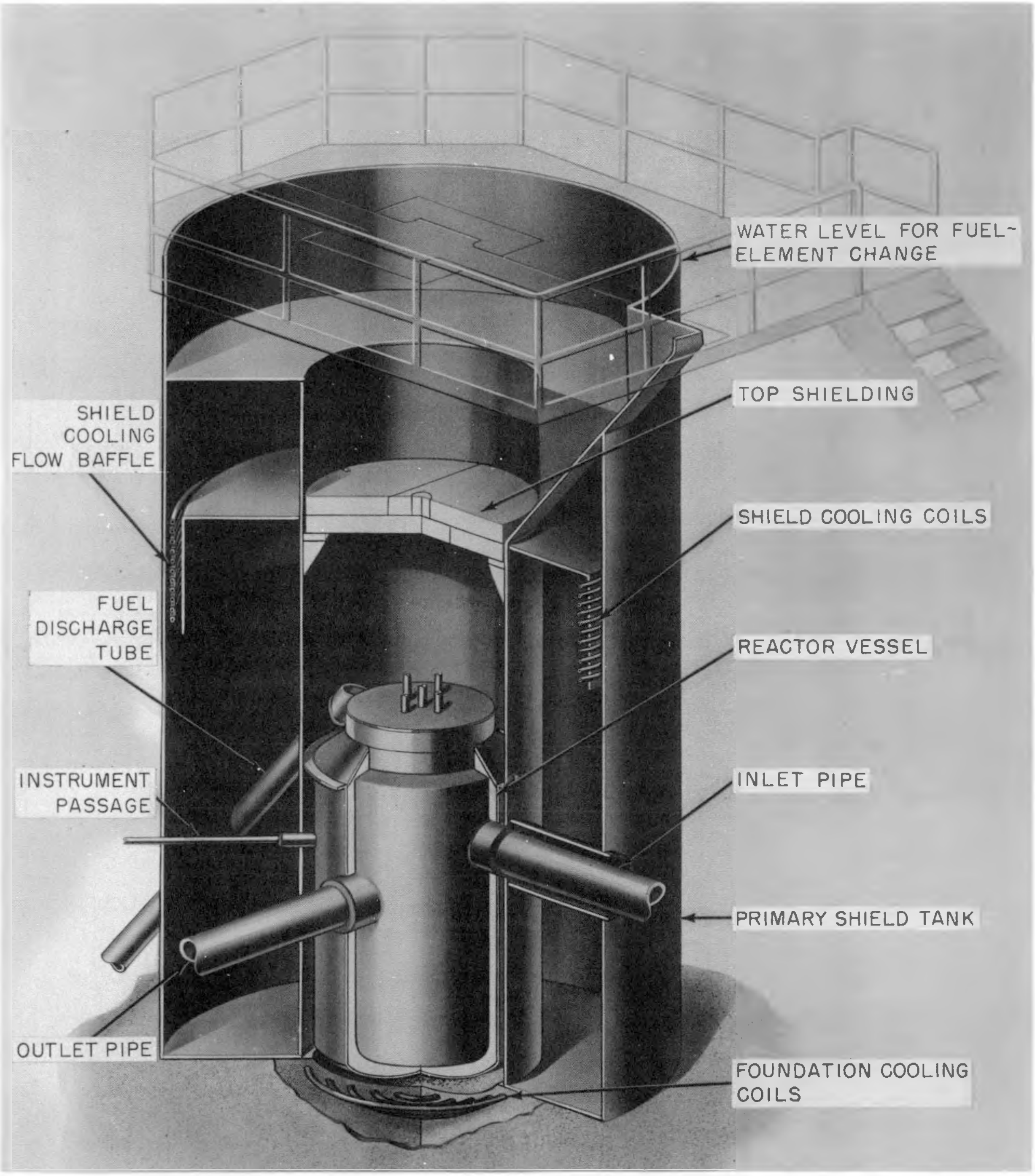
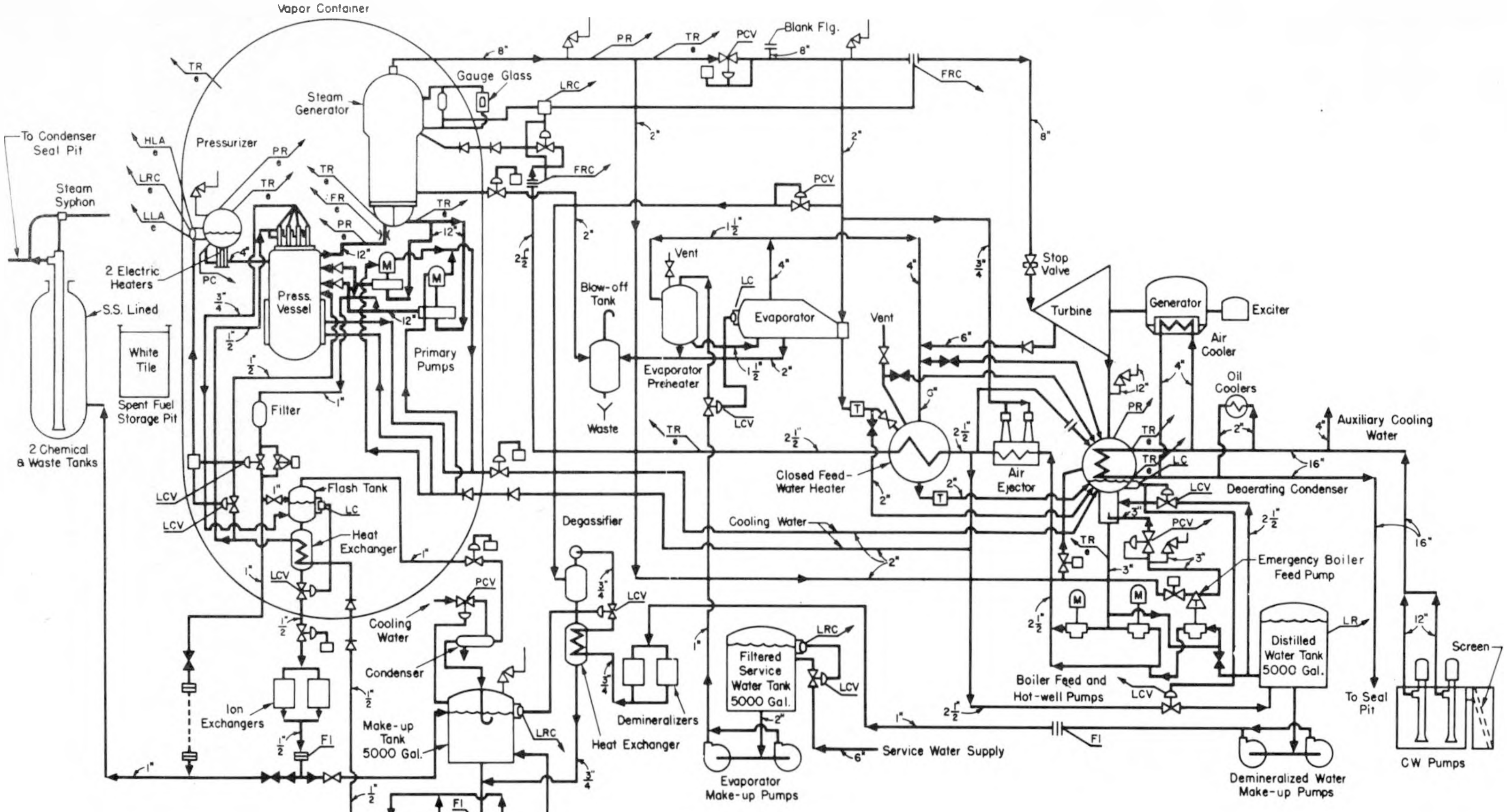


Fig. 2—Cutaway view of the primary shielding tank and reactor.



Indicating Pressure Gauges and Thermometers not shown

BPV Back Press Valve
PCV Pressure Control Valve
LCV Level Control Valve
HLA High Level Alarm
LLA Low Level Alarm
FR Flow Recorder
FI Flow Indicator
PR Pressure Recorder
TR Temperature Recorder
LC Level Controller
PC Pressure Controller
Motor-Operated Valve

Normally Closed
Safety Valve
Trap
Orifice
Temporary Hose Connection Between Normal Blind Flanges.
Valve in Outgoing Line from Vapor Container Automatically Closes on Pressure Rise

Trip Valve
Control Board Mounted Instrument
Locally Mounted Instrument
Board Mounted Instrument, Electric Transmission

Fig. 3 — Fundamental flow diagram.

4.2 Pumping Provisions and Coolant Circuit

The primary coolant circuit includes two "canned-rotor" type motor-driven coolant pumps, only one of which is required for full-load operation. One hundred per cent stand-by is provided for by automatic start-up of the stand-by pump in the event of failure of the operating unit.

Each pump is provided with a balanced swing check valve that remains in the closed position on the idle pump to prevent backflow caused by the pressure head generated by the operating pump. The motor and pump impeller can be removed for maintenance or replacement by cutting open the seal weld.

4.3 Pressurizer and Heaters

Pressure is maintained in the primary coolant circuit by a pressurizer vessel containing two 50-kw heaters. A relief valve on this pressurizer protects the primary circuit from overpressure. In addition, a control valve of restricted-flow capacity, remotely operated from the central control room, permits venting noncondensable gases from the primary circuit.

Although not shown in the figures, a high-pressure tank of hydrogen will be connected to the pressurizer through an automatically operated valve. In the event of power failure causing loss of heat in the pressurizer, the hydrogen pressure can be used to maintain system pressure and to prevent boiling in the reactor core even during the early part of the cooling-off period.

4.4 Primary System Water Supply

Water for the primary system is drawn from the distilled water tank, and, in passage to the make-up storage tank, it goes through a demineralizer and degassifier. Since storage of the water may permit the pickup of corrosion products, a second demineralizer is installed after the storage tank and before delivery to the primary system. In actual practice the demineralizers may well be used as alternates, since the corrosion products picked up in the primary loop are separately removed by the ion exchangers in the purification system in order to confine the radioactivity. The hydrogen for corrosion control is introduced to the water after withdrawal from the storage tank.

5. STEAM SYSTEM

5.1 Secondary Flow Circuit

Steam produced in the steam generator is led to the turbine (see Fig. 3), where it is expanded from the nominal 200-psia level to the $1\frac{1}{2}$-in. Hg back pressure provided by the condenser. Condensate, collected in the condenser hot well, is returned by the condensate pumps, first through the condensing side of the air ejectors and then through the feed-water heater to the steam generator.

5.2 Steam Generator

At full load steam is formed in the steam-generator shell at 200 psia (382°F saturation temperature) at the normal heat-transfer rate in the steam generator. The steam is superheated 25° to reduce condensation. At decreasing loads and with a constant primary coolant-inlet temperature to the steam generator of 450°F, the saturation temperature on the secondary steam side will approach 450°F; thus at no load steam will be generated at nearly 425 psia.

A three-element level-regulating valve is provided in the feed-water connection to the steam generator. Measurements of both steam and feed-water flow provide anticipatory action on the basic control from the steam-generator water level, thus ensuring closer level regulation.

5.3 Steam Turbine

The steam generator and steam turbine are connected with a single 8-in. carbon-steel pipeline, which will permit a line pressure drop of only 10 psi so that the turbine inlet pressure will be approximately 190 psia. The turbine will be designed for 250-psig inlet pressure to provide an operating margin for increased inlet pressure at partial load. A pressure-reducing valve is provided which will throttle the steam to maintain a 250-psig line pressure when decreasing load causes the steam pressure to rise above 250 psig. A safety valve ahead of the pressure-reducing valve protects the steam generator, and a safety valve following the pressure-reducing valve protects the turbine against overpressure.

Steam from the turbine exhausts to a surface condenser of the deaerating type. Combined vertical hot well and feed pumps remove the condensate from the condenser hot well and deliver it to the steam generator, thus avoiding the use of two sets of pumps. Normally only one of two 30-hp motor-driven condensate pumps will be used. A steam-driven pump is provided for feeding the steam generator during loss of electric power.

5.4 Air Ejector and Feed-water Heater

The condensate being returned to the steam generator is led first to the condensing side of the main condenser air ejector. Here the condensing action assists the ejector to maintain the desired main condenser vacuum, and the heat picked up by the feed water serves to preheat it. From the air ejector the feed water is led to the feed-water heater, which at full load raises the temperature to 250°F. The main source of heat is steam extracted from the turbine at 35 psia, although some economy is achieved by using the steam from the make-up evaporator to supplement the extracted steam.

5.5 Main Condenser Circulating Water

Circulating water for the condenser and cooling water for auxiliary services are taken from Gunston Cove on the Potomac River. A water pump well with dredged inlet will be built in the shallow cove to ensure a supply of water at all tide levels.

Water is circulated by motor-driven 2500-gal/min, 60-ft-head vertical submerged pumps, two of which are in operation at full load, although a single pump can carry appreciably more than one half load. The circulating water is discharged from the condenser to a seal pit in order to maintain as high a vacuum as practical at the condenser water box, thereby reducing pumping power.

6. REACTOR PHYSICS AND SHIELDING

6.1 Reactor Physics

When ALCO submitted the proposal with a preliminary design, the status of the reactor-physics calculations were believed to be adequately in balance with the remainder of the considerations.

The requirements for critical mass, control-rod worth, and temperature coefficient of reactivity were obtained from Report ORNL-1613, Supplement 1.

6.2 Description of Shielding

To provide a compact biological shield that can be readily erected, the major part of the shielding is accomplished by an iron-water shield immediately around the top and sides of the reactor. A cross section of the shield configuration is indicated in Fig. 2. Although intended specifically for installation at the Fort Belvoir site, the iron-water shield is well adapted for use at remote locations since it will require less field erection time, take up less space, and may represent less transportable weight than a concrete shield to a remote location. The iron-water shield is appreciably lighter than the equivalent concrete structure if coarse aggregate must be transported to the remote site. If aggregate is available at the site, the concrete shield offers a saving in transported weight. For shielding of areas outside the vapor container, the vapor-container wall provides additional radiation attenuation.

The basic element of the shield is an annular welded-steel watertight tank 30 in. thick to provide the neutron attenuation necessary to prevent activation of equipment in the vapor container. Around the shield tank is a ring of cast iron 11 in. thick which will provide the chief gamma shield with increased thickness in the sectors toward the control and turbine rooms.

6.3 Cooling of Shield

To remove radiation and conduction heat from the shield, cooling water is circulated through 11 coils of $1\frac{1}{4}$-in. pipe inside the tank. To aid natural convection a baffle near the top of the shield tank directs the hot water from the inside past the cooling coils. Heat generated in, or transferred to, the cast-iron ring and the top shielding disks is removed by conduction to the surrounding shield tank and by convection to the air. Heat is removed from the concrete below the reactor vessel by embedded cooling coils (Fig. 2).

6.4 Shielding for Fuel-element Handling

For fuel-element changing the space above the reactor is flooded to a depth of 12 ft above the top edge of the opened vessel. Elements may then be continuously submerged while being transferred to the discharge tube. The short portion of the fuel-element discharge tube between the shield-tank wall and the vapor container is surrounded by concrete to prevent excessive radiation during passage of the fuel element.

7. CONTAINMENT

7.1 Maximum Credible Accident

The maximum credible accident for which the vapor container (described in Sec. 7.2) is designed is a failure, resulting from unforeseeable causes, in the primary-system piping between the reactor vessel and the steam generator. Under such failure conditions it is expected that a sudden release of pressure will allow flashing to occur throughout the primary system, discharging all the primary system water into the vapor container. It is also possible that the rupture may produce missiles up to 50 lb in weight with a velocity as great as 700 ft/sec. Such missiles could break the steam-generator shell, discharging the steam and water of the steam system.

7.2 Vapor Container

The vapor-container structure has been given the most careful consideration because of its extreme importance to the safe operation of the APPR at the Fort Bevoir site. The design offered by ALCO consists of a cylindrical steel container with spherical ends constructed to withstand an internal pressure of approximately 50 psig (see Fig. 4). Inside this shell is 2 ft of reinforced concrete, which gives stiffness to the shell and protects it against high temperatures in the event of a primary-system rupture. The outer steel shell is not subject to large variations in temperature and therefore can be buried in the ground and supported on a concrete base without expansion difficulties. The concrete lining provides a ruptureproof container that will withstand any reasonable size of missile that might result from a rupture of the primary or high-pressure system. This lining also contributes to the shielding for personnel. The interior surface of the concrete is lined with light steel to facilitate cleaning.

Locating this vapor container with one end buried in the ground places the reactor vessel below ground level for increased safety and at the same time results in a more compact power plant, thus minimizing the cost of piping and electrical wiring and equipment.

A 6-in. emergency drain pipe will be provided, extending from above ground to the floor of the container. Normally sealed the pipe can be opened from the outside and connected to an external pump to remove contaminated cooling and washdown water after the maximum credible accident. Openings for an access door and an equipment hatch will be circular heavy-wall tubes with manhole covers inside the container held in place by bolts. The joint of the equipment-hatch opening at the top of the structure will be seal welded.

For the access door at the side of the container, an additional manhole cover will be provided which may be seal welded from the outside. Since there will be no concrete in this access hole, it will be filled with water whenever the access door is sealed. Draining this water will permit ready access to the interior of the vapor container for reloading the reactor.

7.3 Interior Cooling

It will be necessary to provide cooling for the interior of the vapor container at all times when the reactor is in operation. It has been estimated that the heat loss from the equipment will amount to approximately 50,000 Btu/hr when the APPR is running at full output. It is planned to provide cooling by circulating river water through steel-pipe coils, constructed of 2-in. pipe, placed approximately 1 ft from the interior wall of the container and spaced approximately 2 ft apart (see Fig. 4). This arrangement will permit cleaning of the interior surface when decontamination is required. In addition to the normal cooling, a second coil of pipe equipped with sprinkler heads will be provided. Normally this second coil of piping will not be in use, but in case of a rupture it will be connected to the fire-protection system and will cool the container as it washes down the walls.

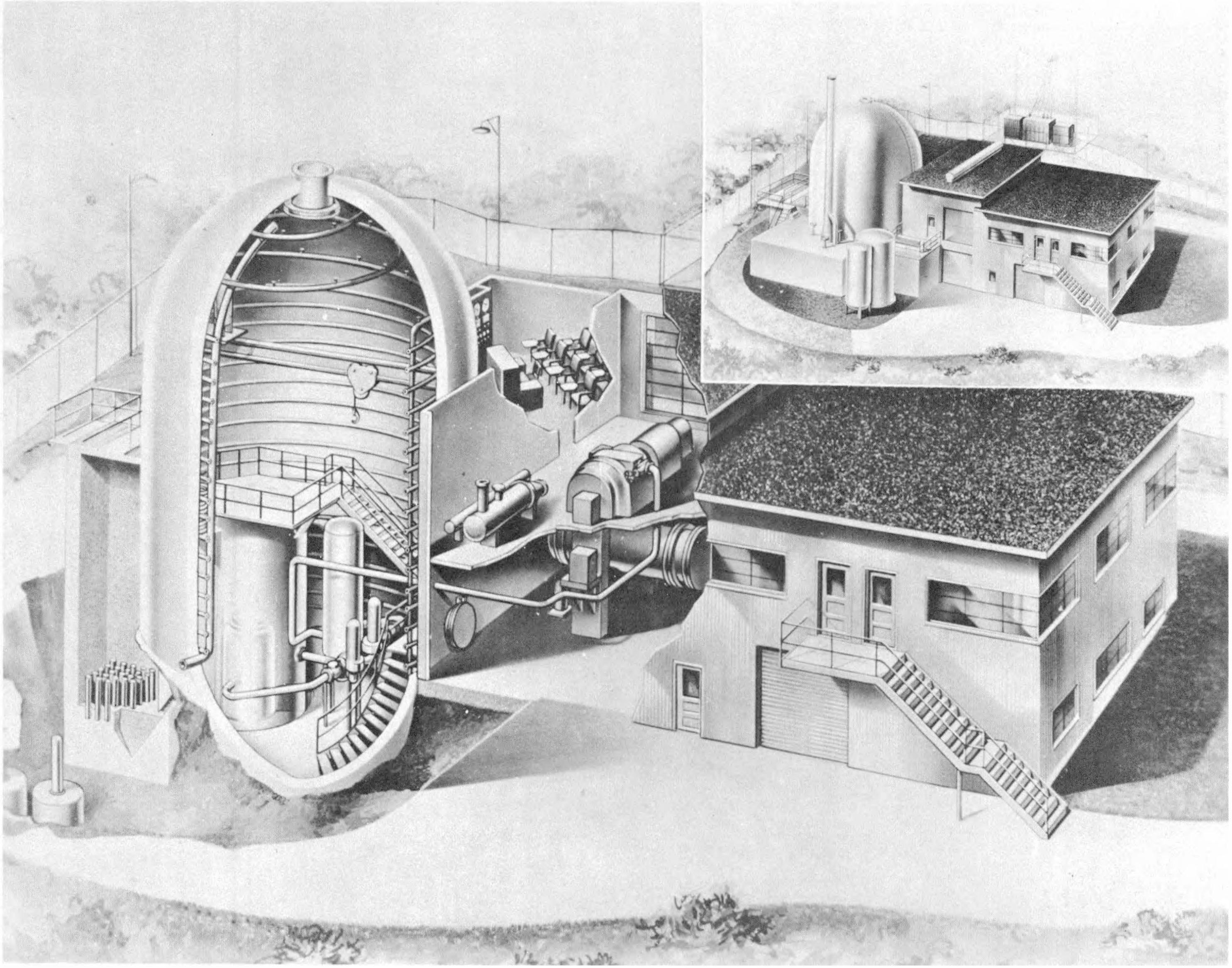


Fig. 4 — Cutaway view of the APPR.

7.4 Ventilation and Decontamination

ALCO has interpreted the ventilation requirements of the vapor container to mean that, during any period of occupancy by personnel, the space will be ventilated at the required rate of six fresh-air changes per hour. During reactor operation the requirement of a leaktight container governs so that the container is sealed without ventilation. This arrangement obviates the need for normally open ventilation ducts, which must be closed leaktight in the event of an accident.

8. ELECTRICAL SYSTEM

The electrical power is generated by a 2000-kw 4160-volt 3-phase 600-cycle turbine generator unit. Since the electrical tie-in at the Fort Belvoir substation will be converted to 4160 volts, only circuit breakers are required in the transmission system. An outdoor substation is provided, however, to supply power for station service at 460, 208, and 110 volts. For the station services two 300-kva transformers are provided, one being used as a reserve in the event of failure of the normal transformer. An emergency 125-volt d-c battery system is provided in the event of complete power failure.

9. BUILDING AND AUXILIARY EQUIPMENT

9.1 Building

Steel-frame buildings with insulated metal walls and roofs (see Fig. 4) were selected after considering economy, portability, and ease of erection during severe weather conditions. The walls consist of a flat and a fluted metal plate, enclosing at least $1\frac{1}{2}$ in. of Fiberglas board or equal. The exterior surface is protected by Galbestos (H. H. Robertson Co., Pittsburgh) or equal. The roof comprises an overhanging metal deck covered with insulation and a tar-and-gravel roofing.

The floors are to be made of concrete, supplemented by a surface hardener, except in locations having tile flooring. Asphalt tile is used in the offices, classroom, laboratory, locker room, instrument-repair room, and control room. The bottom slab will be supported on the ground, but the upper floor will be supported from the steel framing.

Two 5-ton nominal capacity air-to-air heat pumps provide summer air conditioning and winter heating for the areas as required. These will be supplemented by five electric unit heaters in the areas not air conditioned.

9.2 Spent-fuel Pit

A pit for the storage of spent fuel is provided immediately outside the vapor container (see Fig. 4). It is approximately 28 ft deep and is lined with white tile. Illumination is provided to facilitate storage of the fuel elements in stainless-steel holders at the base of the pit, spaced far enough apart to prohibit reaching a critical mass and geometry even with fuel elements at the peak of their reactivity. Fuel is transferred to the storage pit from the reactor vessel through a submerged tube (see Fig. 2). Fuel may thus be moved from the reactor vessel to the discharge chute with a minimum of 12 ft of water above the element, which will provide shielding for the operator during the transfer process.

Since the pit remains full of water at all times, whereas the reactor region is dry during operation, a simple but effective plug valve has been designed for the fuel-discharge chute. The plug is designed to withstand the pressure of water in the spent-fuel pit with no water in the tank above the reactor vessel; yet the plug is light enough to permit easy handling by the chain hoist in the vapor container.

In the event of a rupture and a consequent build-up of pressure inside the vapor container, the end thrust on the plug will be taken by a metal seat. The plug is so located in the fuel chute that it is fully protected by concrete against a rupture of the high-pressure system, and hence it is capable of maintaining the integrity of the vapor container even in case of a rupture of the reactor vessel itself.

9.3 Waste Holdup Tanks

The 1000-gal waste holdup tanks will be buried underground and will receive the waste water from the blowdown system, together with the wastes from the sample room and the chemical laboratory. These quantities will all be small, amounting to a maximum of approximately 20 gal of water per hour. Each tank will be equipped with a steam siphon, permitting a controlled discharge of this water to the condenser seal water tank at a lower elevation.

9.4 Combined Make-up and Filtered Water Tanks

The purge water from the primary system will contain moderate amounts of radioactive contaminants. The expected activity is such that, for complete freedom of movement of personnel, a small amount of shielding is required around the tank. Since the flow rate is low and since the impurities have relatively short half lives, an appreciable amount of self-shielding is accomplished by baffling the interior of the tank. Water entering the tank enters first a 1000-gal inner reservoir, centrally disposed within the main tank. The flow rate is such that a two-day holdup is achieved before the incoming water reaches the top of the inner baffle, where it is permitted to mix with the remaining 4000 gal in storage. By this device the most active material is confined to the center, and the already cooled bulk of water surrounds it and acts as a shield.

Since the self-shielding is insufficient to permit complete personnel freedom, the remaining shielding is achieved by enveloping the tank within the raw filtered water tank. Therefore the latter tank is designed to give approximately a 1-ft-thick shield of water around the sides of the demineralized make-up water tank.

9.5 Air Ejector

The removal of noncondensables from the condenser is accomplished through the use of a steam-jet air ejector, using steam at full pressure in ejector nozzles. Condensation of the jet steam is accomplished by feed water returning to the steam generator. Since the light-load feed-water rate may not permit adequate condenser action, an orifice return of feed water to the main condenser is provided. This maintains a continuous low-volume recirculation that is adequate to permit attaining the desired condenser vacuum at low flows, but it is still a small enough item at reasonable loads so that it does not noticeably affect plant efficiency.

9.6 Emergency Cooling of Reactor

In the event of a complete electric power failure or the failure of both coolant pumps, heat must be removed from the reactor by thermal circulation alone. It can be expected that the control and shim rods will operate under these conditions to reduce the reactor output to a minimum. However, there will be decay heat, which has to be removed if damage to the reactor is to be prevented. The piping and equipment in the high-pressure system have been designed to facilitate thermal circulation under these conditions.

The fission-product heating 1 sec after shutdown will amount to approximately 2,000,000 Btu/hr but reduces rapidly to approximately 200,000 Btu at the end of 24 hr. No thermal circulation in the high-pressure system can occur unless there is a temperature difference between the inlet and outlet sides of the steam generator. It is therefore planned to use a steam-driven boiler feed pump that is automatically started in the event of a power failure to the two coolant pumps. This steam-driven pump not only ensures a supply of feed water to the steam generator but also provides a means of removing heat from the secondary side of the steam generator, and thus from the high-pressure coolant system, at a rate approximately equal to that of the fission-product heating in the reactor.

To provide a steam load in case the feed water to the steam generator is throttled owing to changes of level in the steam generator, an automatic dump valve will be provided which will discharge steam from the high-pressure steam line direct to the condenser or to the atmosphere. The main condenser has been provided with an atmospheric relief valve to permit such operation.

10. FUEL-HANDLING PROCEDURES

10.1 Facilities for Fuel Handling

The fuel elements are handled under water and are discharged to a chute through the vapor-container structure into the bottom of a deep storage pit for spent fuel. The fuel chute and deep storage pit, which have already been described, can be seen in Figs. 2 and 4.

10.2 Removal of Spent Fuel

After the vapor container has been cooled and decontaminated, the access doors can be opened and preparations can be made for removing the spent fuel. First, approximately 2 ft of demineralized water will be pumped into the tank directly above the top flange of the reactor vessel. The shielding disk will be removed, and the pit will be filled up to the control-rod mechanism. The bolting that holds the cap on the pressure vessel will be loosened, and then the entire compartment above the reactor vessel will be flooded as the top is lifted out, keeping the control equipment dry. The water level reaches a height equal to that in the outside fuel storage pit.

After the pressure between the tank and the deep pit has been equalized, the special plug valve (see Sec. 9.2) will be removed from the fuel chute.

Each fuel element will be removed by manual tongs and discharged through the transfer tube to the base of the deep pit, where manipulators, operated from the platform at the top of the pit, will be used to place individual fuel elements in the stainless-steel holders at the base of the pit. It is anticipated that the entire fuel charge can be removed in a period of approximately 3 hr.

10.3 Reloading of Reactor

The special plug valve will then be placed and properly seated. New fuel elements will be brought in and installed with the reactor vessel flooded with water. Suitable criticality tests will be made during the loading operation. Water from the fuel handling tank will then be pumped out by a portable pump to the fuel storage pit. When sufficient water is removed from the fuel handling tank, the pressure-vessel cap will be installed and bolted, and the remainder of the water will be pumped out.

With the reactor vessel closed the make-up pumps can be started to circulate the primary water through the demineralizers to remove contamination that entered when the top well was flooded. The shielding blocks above the reactor can be replaced, and the reactor can be started up.

ACKNOWLEDGMENTS

This article is a condensation of the ALCO proposal for the APPR which was written by a team of engineers, scientists, and consultants for ALCO. Mention should be made of the contribution of H. L. Weinberg and F. Fairbanks of the American Locomotive Co. staff and to Dr. A. H. Fox of Union College, consultant to the Atomic Energy Department. Particular mention should be made of the personnel of Stone & Webster Engineering Corp. who contributed substantially to the designs that were incorporated in the proposal and who have been retained by ALCO as the designers and constructors of the project.

REFERENCES

1. R. S. Livingston, A Conceptual Design for the Army Package Power Reactor, Nuclear Sci. Technol., 1(1): 31 (February 1955).
2. A. L. Boch, W. R. Gall, G. F. Leichsenring, and R. S. Livingston, A Conceptual Design of a Pressurized-water Package Power Reactor, Report ORNL-1613, July 8, 1954.
3. American Locomotive Co., Proposal for Design and Construction of Army Package Power Reactor, Nov. 17, 1954.

ABOUT THE AUTHOR

Kenneth Kasschau received the B. Mech. E. degree from Stevens Institute of Technology in 1936 and the M. Mech. E. from Harvard University in 1938. He worked at the Wright Aeronautical Corp. from 1938 until 1946 on aircraft engine design and development. In 1946 he was loaned to the Clinton National Laboratory as one of the members of the Daniels Power Pile Division. When that program turned toward the submarine program and the group was moved to Argonne National Laboratory, he was transferred to the NEPA Project, where he remained until 1950. He then joined the Oak Ridge Operations Office of the Atomic Energy Commission, first as Chief of the Reactor Division and later as Director of the Research and Medicine Division. He resigned from the Commission in July 1954 and joined the American Locomotive Co., where he is Manager of Engineering—Atomic Energy.

ARMY PACKAGE POWER REACTOR

American Machine & Foundry Co. Proposal

C. J. BROUS and F. S. HOLZER

March 11, 1955

ABSTRACT

The Army Package Power Reactor as conceived by the American Machine & Foundry Co. (AMF) is described. This small package reactor power plant, capable of generating a net electrical output of about 2000 kw, is based on a conceptual design prepared by the Oak Ridge National Laboratory and described in the references for this article. A proposal was submitted by AMF to the Atomic Energy Commission for the design, construction, and testing of this plant. The parts of the proposal which describe the plant, as envisaged by AMF, are condensed in this article.

1. INTRODUCTION

This is a greatly condensed version of a 589-page proposal submitted by the American Machine & Foundry Co. (AMF) to the U. S. Atomic Energy Commission (AEC) for the design, construction, and test operation of the Army Package Power Reactor (APPR) to be located at Fort Belvoir, Va. The plant proposed by AMF was to be a true prototype leading to standardization of a package type power plant for general deployment by the Army. The AMF concept for the Fort Belvoir prototype plant would bring the ultimate goal of the standard unitized-design APPR plant so close to reality that, after completion of the operating tests at Fort Belvoir, only minor redesign and modification would be required to place such a plant in operation at virtually any location designated by the Army.

2. GENERAL CONSIDERATIONS

An artist's concept of the APPR plant proposed by AMF is shown in Fig. 1. The AMF design for the APPR is the result of a careful study of the objectives of the government and is based on three primary design criteria, each of which is of major importance:

1. Containment of the maximum credible nuclear accident.
2. Simplicity as an aid to training, maintenance, and safety.
3. Ready adaptability to a standard unitized APPR design that is applicable, with minimum change, for locations practically anywhere in the world whether or not accident containment is required.

The determination of what constitutes a maximum credible nuclear accident can take little advantage of precedence for a satisfactory solution. The proper measures to contain such an accident are straightforward once the accident is defined. Careful study of the nuclear reaction and its control, an investigation of existing limited experience with reactor containment,

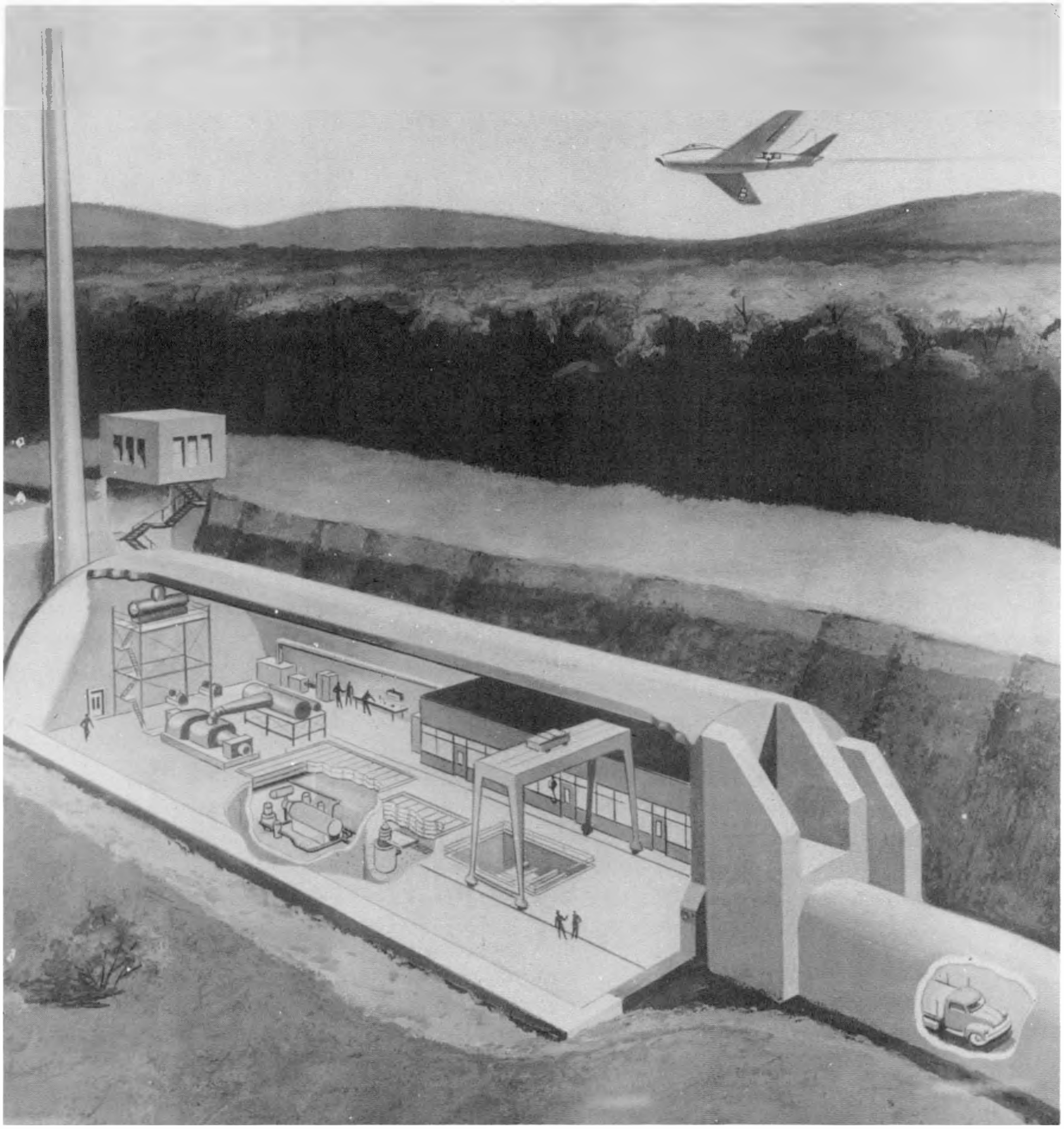


Fig. 1—Artist's concept of the APPR plant proposed by AMF.

together with an analysis of the rulings of the Advisory Committee on Reactor Safeguards convinced AMF that both operating and unloading accidents should be contained. Therefore the common containment method of placing the primary loop in a sealed tank was discarded. Since the damage that might result from such possible accidents at the selected Fort Belvoir site could be extremely severe, AMF concluded that the only realistic solution to the containment problem is a sealed underground building. The proposed arrangement for such a building permits a simple layout which can be applied also to plants where containment is not required, since the entire plant can be located above ground without redesign.

Since the proposed plant is to be used as a training facility, the design is geared to provide maximum accessibility of plant equipment, maximum ease and safety of operation, as well as convenient and straightforward handling and maintenance of components within the shielded area. The control system is foolproof and fail-safe. It will forestall potentially dangerous situations in a way that will reduce emergency shutdowns to an absolute minimum. The controls are so designed that their worst management could not result in destructive power surges. Routine maintenance and most repairs on the controls can be performed without shutting down the plant. Special handling tools are to be provided to eliminate any requirement for riggers or temporary construction during core handling. The entire unshielded portion of the plant is laid out on one level so that all electrical, steam, and water lines can be readily traced.

In accordance with the contract requirements the plant would have a guaranteed net electrical output of 1931 kw and a core life of 15 Mw-years total energy release and would be completed in 36 months. The AMF schedule called for development, design, and construction to be completed after 2 years, allowing one whole year for an intensive testing program, which would include thorough component testing, functional testing of every portion of the plant, as well as a 700-hr performance test.

3. REACTOR COMPONENTS

3.1 Reactor-core Assembly

AMF proposed to use fuel assemblies that are very similar to those developed by ORNL in the APPR feasibility study. These are flat plate type fuel assemblies similar to those employed in the Materials Testing Reactor (MTR) and the Submarine Thermal Reactor (STR). The fuel plates consist of 93.5 per cent enriched uranium dioxide retained in a matrix of sintered stainless-steel powder and clad with 304L stainless steel. Eighteen of these plates are assembled into a fuel assembly whose over-all length is about 35 in. and whose cross section is approximately a 3-in. square. A satisfactory method for fabricating such fuel assemblies has been developed by ORNL.

The exact composition of the core of the fuel plate remains to be determined by nuclear calculations. The total initial U^{235} loading is determined by the requirement to have an adequate criticality margin after an energy release of 15 Mw-years. This amount of U^{235}, estimated at 18.1 kg, would have to be apportioned among the total number of fuel plates in the core. Further, in order to keep the initial excess reactivity of the core low, small quantities of a neutron absorbing material would be mixed with the fuel powder. This material would burn out during the core life and thus compensate for the uranium burn-up. The final selection of the absorbing material had to be deferred until a stable material is found. ORNL is planning tests with zirconium diboride and boron nitrate for this purpose. Boron carbide, considered initially, was found to be unstable. The total number of fuel plates in the core is to be allowed to vary by a small margin from a reference value of 800, so as to permit an adjustment for calculation errors in the loading, even after the fuel elements have been fabricated.

The control rods are also based on the ORNL design for the APPR feasibility study. The lower section is the same as the fuel assembly except that 16 fuel plates are used. The upper section, which is the absorbing portion of the control rod, is a rectangular shell whose inside surface is covered with stainless-steel clad boron. The transition area between the two sections contains a circular piece of hafnium or clad cadmium to avoid a thermal-flux peak directly above the fuel section. The core is to contain five such control assemblies.

AMF proposed to supply both the fuel and the control assemblies for the first loading of the APPR. This decision was reached after completing an extensive technical evaluation of the fuel-fabrication problem. Included in this evaluation was the consideration that it would be advantageous to the AEC and the Army to have an industrial supplier for the APPR fuel elements for future plants as well as for the installation at Fort Belvoir.

3.2 Control-rod Drive Mechanisms

One major problem in a pressurized reactor is a control-rod drive that is highly reliable and yet prevents leakage of primary coolant from the reactor vessel. A solution to this problem was developed by AMF for the APPR feasibility study of ORNL. This solution employs a rotary seal through which a rack and pinion drive, located inside the pressure vessel, is located.

The seal is basically a labyrinth pressure breakdown type and is designed for a leakage rate of 10 lb/hr at a water pressure of 1200 psi and a water temperature of 600°F. The over-all seal length is 6 in. and the shaft clearance is 1 mil. The seal consists of a number of floating rings made of stellite 3 and a series of K monel bushings. These are assembled in a 304 stainless-steel housing. Clean cold water, under a pressure higher than operating pressure, is to be introduced into the seal near the pressure end. Leakage from the seal will therefore be clean water. The high-pressure seal water is to be supplied by the make-up water pumps.

The motive power is transmitted through the seal to the pinion by a motor package designed to the following specifications, which were determined from reactor kinetic calculations: The control-rod speed will be limited to 2 in./min, up or down. The rod can be positioned with an accuracy of 0.01 in., and the position will be indicated at all times, including during and after scram, with an accuracy of 0.5 per cent. During scram the rod will drop the full distance (22 in.) in 0.7 sec. This time includes a 50 msec breakaway time and rod deceleration due to snubbing.

The motor package consists essentially of a Diehl low-inertia motor that drives the pinion shaft through a worm-and-gear reduction unit and a magnetic clutch. The low-inertia rotor permits the motor to come up to speed quickly and to stop with very little coasting after the power is removed. Moreover, it is adaptable to control by switches and by servo amplifiers so that the same motor package can be used for shim rods and for regulating rods. The magnetic clutch is mounted on the spindle between the driving gear and the seal. The rod is scrammed by deenergizing the clutch. Pinned to the outside of the spindle before it enters the spindle seal is a clock type torsion spring which constantly loads the output spindle and pinion in a direction to drive the rod to the full-in position. The spring is a scram device whose sole function is to overcome the friction in the instrumentation gearing, seal, and water so that the rod will drop with the required velocity. Rod position indication is accomplished by means of a small synchro coupled to a take-off shaft geared to the output spindle.

A latch unit is attached to the lower end of the rack inside the pressure vessel. The latch will transmit linear motion to the control rods. It will automatically release the control rod when the rod is in its lowest position, thus allowing removal of the pressure-vessel cover while the rod remains in the reactor; also the latch will automatically grip the rod when the pressure-vessel cover is replaced. During extensive testing the latch jaws have released the rod on every occasion.

3.3 Reactor Structure

The reactor pressure vessel will be designed according to ASME standards for unfired pressure vessels, the design presure being 1250 psi and design temperature, 650°F. The vessel will be 108 in. high and will have a maximum inside diameter of 4 ft. The cylindrical wall is to be 2.25 in. thick, including a 125-mil stainless-steel cladding. The cylindrical section will be approximately 6.5 ft long. The top end of this section will be welded to an elliptical head which, in turn, will be welded to a rectangular cylinder with a 6-in. wall thickness to provide sufficient area for mounting the studs for attaching the cover plate. A 2-in. thermal shield will be included. The cylindrical shield will be welded to the upper support plate and extend downward 49.5 in.

A ring structure will be attached to the thermal-shield assembly. This structure will be used to mount the grid and support structure for the reactor core. The structure consists of the skirt support plate, the upper assembly grid, the skirt, the lower assembly grid, and shock absorbers. Except for the upper grid these components are assembled as a unit and lowered into the pressure vessel. The fuel assemblies rest on the lower assembly grid. The upper assembly grid compresses the fuel assembly springs. Compression is maintained by cam latches mounted on the pressure-vessel support frame. The grid and support structure also provides bearings and shock absorbers for the control rods. Bearings are located in both the upper and lower assembly grid. The shock absorbers are open-top cylinders with a 0.125-

in. orifice at the bottom. Pistonlike ends of the control rods enter the cylinder, and the kinetic energy of the control rods is dissipated by forcing the water in the cylinder through the orifice. The absorbers are attached to the lower assembly grid.

3.4 Instrumentation and Control

The reactor is controlled primarily by means of five control rods, of which one is used as a safety rod, one as a regulating rod, and the remaining three as shim rods. The total worth of the five rods is approximately 0.25 δK. In order to minimize nonuniformities in burn-up, it is desirable to rotate the functions of the rods during the life of the core. The control system is arranged, therefore, so that any one of the rods could be used either as a safety rod or as a regulating rod. The selection of rod functions is effected by means of rotary selector switches on the control panel. Interlocks are arranged so that the safety rod is the first rod to be withdrawn in start-up and the last one to be inserted in shutdown.

Three important features of the control-system design provide a wide margin of safety against a start-up accident in which all rods start withdrawing at their maximum speed at a time at which the reactor is at a spontaneous level and continue withdrawing at this speed until they are caused to scram by a level trip. (This assumes that the period safeties fail to work.) One feature is that when all five rods are at their most effective point and move up at their maximum speed they will increase the reactivity by less than 0.07 per cent per second. A second feature is that during start-up the safety circuit will initiate a scram at a flux level corresponding to 10 kw. This low-level scram can be bypassed only after the reactor has reached safely a level of 1 kw. Calculations have shown that under the worst conditions the low-level trip would limit the power surge following a start-up accident to less than 750 kw. The third feature is that reliable period indication would be provided whenever the reactor is supercritical. This is attained with a single period channel with a range of 5×10^6 by assuring a high spontaneous level with a 10,000-curie antimony-beryllium source. This source would assure that the reactor power level is at least 2 watts when criticality is reached and would cost less than 0.65 per cent in terms of a multiplication factor or about 180 g of U^{235} in terms of critical mass.

A careful analysis of plant kinetics have convinced AMF that, for automatic control during power operation, only a slow-acting servo system responsive to persistent deviations in the coolant outlet temperature is required. All changes in power demand can be accommodated by the negative temperature coefficient, and the worst resulting temperature fluctuations would not result in film boiling. The reactivity control by the servo system would be limited to a maximum average rate of 5×10^{-5} per second, which is adequate for following the most rapid possible xenon transients. The proposed servo system moves the regulating rod in increments proportional to the temperature deviation. After each movement there is a 30-sec waiting period to allow the temperature to settle down at a new value before any further corrective action is taken. This scheme promises excellent stability.

The proposed control system incorporates permissive and overriding circuits that will prevent the operator from initiating hazardous actions, will override the actions of the operator if they tend to produce an unsafe condition, and will automatically take action when the operator fails to take a necessary action. These circuits have been designed so that, whenever possible, they should forestall an automatic emergency shutdown and thus keep the reactor plant "on the line" under as many conditions of faulty operation as possible. The permissive and overriding circuits will, of course, limit the freedom of action of the servo system, as well as that of the operator. A description of these circuits is precluded by the space limitations of this greatly condensed version of the proposal.

The control system operates the motors of the five rod drives, and the safety system controls the excitation of the magnetic clutches. The safety system and the associated nuclear instrumentation is based to a large extent on the one developed by ORNL for the MTR, which is adequately described elsewhere in the literature. Departures from this tried and proved system were proposed only where subsequent technological improvements made greater reliability possible or where greater operating continuity could be obtained without sacrificing safety. Moreover, some departures were proposed where they are dictated by special requirements of the APPR.

As an example of one of the special safety features that was proposed, an interlock that combines flux level and outlet temperature will be described. In order to prevent film boiling in the hottest coolant channel, the fuel temperature in this channel must not be allowed to rise more than 15°F in the proposed design. When the power demand is suddenly decreased from maximum to zero and no rods are operated, the average fuel temperature rises no more than 5°F. The rise in fuel temperatures is due to a rise in coolant inlet temperature, but this rise in coolant temperature causes the neutron flux to drop at the same time as it causes the decrease in the rate of heat removal from the fuel elements. The two effects on the fuel temperature thus tend to cancel out. It may thus be concluded that as long as rods are not withdrawn after the power demand drops, no harmful temperature transients will occur. If, however, an ill-advised operator tries to counteract the effect of the temperature coefficient by withdrawing rods, as he notices that the flux level drops (this effect takes place over a period of about 90 sec), a dangerous situation may develop. This situation is safeguarded against by an interlock circuit that responds to a weighted sum of outlet temperature and flux level. If the coolant outlet temperature rises, the flux level should drop. For every coolant outlet temperature above the design value (450°F), there will be a maximum safe flux level, which will be less than the flux at full power. The interlock will operate whenever the maximum safe flux level corresponding to the measured outlet temperature is exceeded. This circuit would have to be calibrated periodically against actual flux at full power so as to take into account the increase in neutron flux with fuel burn-up. This is just one example of many special safety features.

The proposed reactor is designed with only one type of rod drive mechanism. Although tried and proved safety devices are used, it must be granted that, since all rods are alike and operated by the same system, a malfunction that might cause one to fail could cause the other four to fail at the same time. Thus as an emergency backup it is proposed to have a tank of boric acid pressurized with helium and connected to the primary loop through a safety valve that can be opened by a special emergency handle on the control console. A glass must be broken to gain access to this handle, and the handle must then be positively pulled in order to open the valve and force the boric acid into the primary loop.

4. PRIMARY COOLANT SYSTEM

A closed-loop circulating system is used to carry reactor heat to the heat exchanger. Primary coolant water is forced by the circulating pumps through the core where it picks up the heat of the nuclear reaction. The heater water in the core then flows through primary piping to the main-stream generator where it loses its heat in the generation of steam. From there primary coolant water flows back to the pumps, completing the loop. The primary coolant water is maintained at a pressure of 1200 psi by the pressurizer. A water purification system is used to obtain a condition of high purity in the primary loop at all times. Under design conditions the circulating system will provide an output to the main steam generator of 4000 gal/min of coolant water at 450°F. The system is designed to handle the full reactor output of 10 megawatts plus a reasonable overload heat output.

Calculations were performed to determine the heat-transfer conditions that occur when the design output is exceeded and the coolant flow is decreased. The results of the calculations are as follows: If the reactor output rises to 31 megawatts, bulk boiling will occur in the hottest channel. If the reactor output rises to 11.5 megawatts, surface boiling will occur at the hottest fuel-plate surface. If the coolant flow is reduced to 43 per cent of normal, bulk boiling will occur at 10 megawatts. If the coolant flow is reduced to 84 per cent of normal, surface boiling will occur at 10 megawatts.

The primary circulating pumps are to be of the canned rotor type. Two pumps, each capable of circulating 4000 gal/min, will be supplied. They will be in parallel with only one pump operating. Back flow through the pump that is not being used will be prevented by check valves. The pumps will be vertical single-stage units with bottom suction and horizontal discharge.

In case of circulating pump failure or intentional shutdown of the reactor, means of cooling the core to take away fission product decay energy must be provided. It is intended that this emergency or normal shutdown cooling will be accomplished by natural convection through the primary circulating loop.

The pressurizing system consists of a surge tank located at the high point in the system, electrical heaters in the surge tank, and controls for operating the heaters and the primary system blowdown valve. The 25-kw immersion heaters maintain the water in the pressurizing tank at 567°F and supply steam to maintain a pressure of 1200 psi. The heaters are sized to restore equilibrium pressure conditions after any reasonable transients within about 2 min.

The proposed water purification system uses a make-up and blowdown principle. It consists of a distilled water supply, make-up pumps, high-pressure supply piping, control-rod drive seal leak-off piping, blowdown equipment, filtering equipment, and hydrogen-injection equipment for maintaining satisfactory limits on solid particles in the system and on dissolved oxygen.

Instrumentation will be provided to measure primary-coolant flow, reactor-coolant outlet temperature, temperature rise in reactor, pressurizer pressure, make-up pump pressure, pressurizer liquid level, waste-tank liquid levels (two tanks), shield-tank liquid levels (four tanks), miscellaneous temperatures within shield, water conductivity, temperatures of pump-coolant outlet, motor packages, and stack.

5. PHYSICS

The hand and machine calculations performed by ORNL, which are summarized by Livingston in the February 1955 issue of this journal,[1] were reviewed carefully by AMF. The assumptions which were made in these calculations were considered. It was decided that very few, if any, additional machine calculations would be required. However, a second critical experiment using actual fuel and control-element subassemblies was proposed. This experiment might be conducted in a swimming-pool facility as soon as the subassemblies are available. It would furnish the final check on the criticality, the control-rod worth, and the control-mechanism requirements and would also check the control-rod fabrication at the earliest possible date.

In order to make the maximum possible use of the work performed in the ORNL feasibility study, it is proposed that the final core design should depart as little as possible from the ORNL reference design. In order to evaluate the effects upon critical mass, core life, control requirements, and control-rod worth of any departures from the reference design values dictated by detailed design considerations, it is proposed to use perturbation methods based on the modified two-group method. For this purpose the method would be refined by revising certain sensitive assumptions to obtain the widest possible agreement with experimental results and also the results of the more detailed machine calculations.

A study of the effect of nonuniform burn-up in the axial direction is proposed, and a method for this investigation, using the modified two-group diffusion theory, is suggested.

6. SHIELDING

Since the reactor vessel and the primary-coolant equipment are to be underground, the cost of the biological shield contributes very little to the over-all cost of the plant, and a very conservative shielding design can be employed. It is to be a design objective that radiation levels anywhere above the shield will be below 10 per cent of tolerance. In the steam generating compartment the radiation level will be well below tolerance 1 hr after the reactor is shut down. On the basis of preliminary estimates it should be possible to meet these objectives by a 9-ft concrete plug above the reactor well, a 5-ft concrete plug above the steam generating compartment, and a 9-ft wall of clean soil lined with concrete between the reactor and steam compartments. (See Fig. 2.)

7. CONTAINMENT

Once the site of the APPR became known, the hazard-containment problem became the greatest single factor affecting over-all building design. After careful consideration AMF has

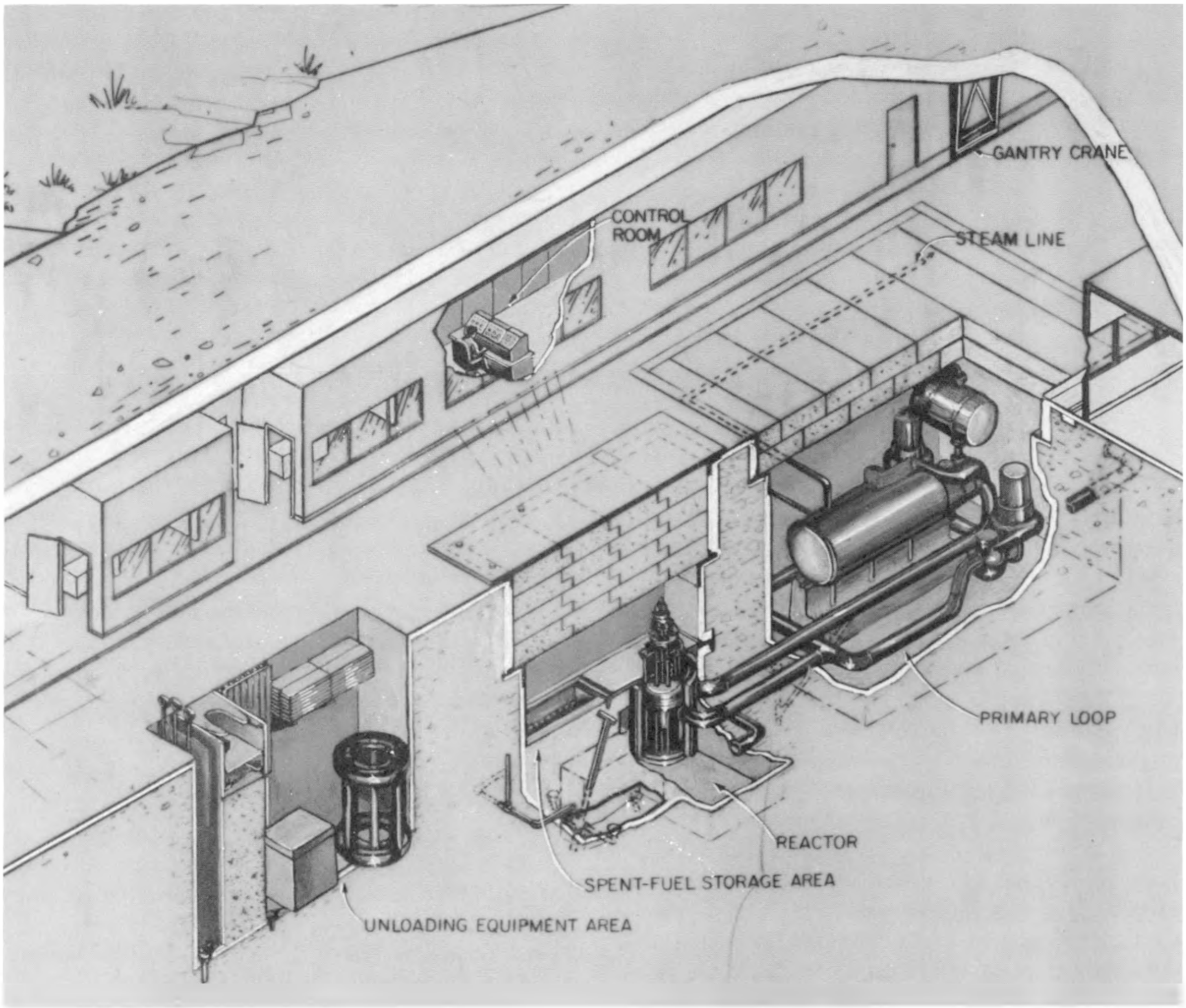


Fig. 2—Plant layout isometric, showing reactor and primary-loop components and handling facilities of the APPR plant proposed by AMF.

reached the conclusion that there are two credible accidents that must determine the containment scheme:

1. During full-load operation all control rods are inadvertently withdrawn and held in the out position owing to a simultaneous mechanical failure. The final emergency safety system (boric acid) is not actuated. The heat then generated by the reactor core vaporizes the water in the entire primary loop. The fuel elements melt, releasing fission products to the vaporized water. The increase in pressure resulting from the water vaporization causes a leak somewhere in the primary system. This leak allows the contaminated vapor to escape into the containment vessel.

2. During the unloading procedure, when the shield plugs have been removed from the reactor pit and the existing water level in the reactor pit is 2 ft above the top of the reactor pressure vessel, the five control rods are inadvertently withdrawn while the entire fuel core remains intact, causing the reactor to go critical. This will cause boiling of the water and then melting and dispersal of the fuel elements by the steam generated.

These two possible accidents were analyzed, and two containment schemes were proposed. In one scheme the operating accident is contained in a high-pressure vessel enclosing the primary system, and the unloading accident is contained in a gastight building. The vessel would have to withstand 21 psig and the building, 4 psig. In the second scheme the containment of both accidents is achieved by placing the entire plant underground within a gastight structure designed to withstand a pressure of 21 psig. Access to the interior of the plant is gained from either end of the building through two gastight doors, one of which must be closed before the other can be opened.

After a very careful weighing of both schemes, AMF decided that the second scheme was preferable. The major factor influencing this decision was that the entire plant design with the exception of the concrete shell enclosure could be used for a ground-level installation with minor modifications, so that the plant would be a true prototype for a remotely located installation where hazard containment would not be required. In the first scheme the plant design would have to be changed radically to the extent that the analogy with a remotely located plant would be lost.

8. STEAM AND ELECTRIC SYSTEM

The proposed steam and electric system consists of a turbine generator, a steam condenser, hot-well condensate pumps, condenser–cooling water pumps, feed-water heaters, boiler–feed water pumps, and evaporator, together with all associated pipes, valves, and auxiliary equipment. The components of the system are conventional items that may be purchased commercially and will therefore not be described in this condensed version.

A pressure-regulating valve at the steam outlet of the steam generator is to be provided as a means for keeping constant steam pressure at the turbine input for improved plant efficiency. The same valve is used also to shut off the steam load under conditions in which continued steam removal from the steam generator might cause the reactor temperature to drop at a dangerous rate or to an undesirably low value. This valve is actuated automatically immediately after a scram or when the coolant outlet temperature drops below 420°F.

9. BUILDING AND AUXILIARY EQUIPMENT

The proposed Fort Belvoir APPR plant is an underground facility. The general building arrangement may be seen from Figs. 1 and 2. Access and exit for personnel and equipment is provided by the access tunnel and the emergency-exit stairway. This underground arrangement provides complete protection to the thickly populated area in the vicinity of the plant. Within this underground facility the primary and secondary loops of the plant are arranged exactly as they would be located in a remotely located plant without containment. Two shielded compartments protect plant personnel from radiation; these spaces are called the "reactor compartment" and the "primary-loop compartment."

A shield tank, which will be filled with water for reactor fueling operations, is provided over the reactor as a part of the reactor compartment. Two spent-fuel element storage tanks, the reactor vessel compartment, and the shield tank, comprise the over-all reactor compartment. The primary-loop compartment provides gamma-ray shielding of the steam generator, loop pressurizer, pumps, primary piping, and auxiliaries. The secondary loop requires no shielding and is laid out on one level, exactly as contemplated for a remote plant.

10. HANDLING EQUIPMENT

The core-handling problem was studied carefully, and special tools have been designed that make quick loading and unloading of the core possible without requiring highly skilled and experienced personnel. A frame assembly is to be used to lower and raise the pressure vessel cover. This assembly provides a mounting and locating platform for two stand-by control-rod drives, which must be used during initial loading. A nut-removal tool is used to remove the nuts which hold down the reactor cover. A grid-assembly unlatching tool and a grid-assembly removal tool are used in conjunction with the frame assembly to raise or lower the upper assembly grid. A fuel-element tool is used to lift fuel assemblies from the core and transport them to the storage area. A control-rod unlatching and removal tool is needed to disconnect the upper section of the control rod and transport it to the storage racks. The fuel section of the control rod is gripped and transported by a long control-rod removal tool. A gasket tool is needed for removing and replacing the reactor cover gasket. In addition to these special tools, a coffin, a building gantry crane, as well as temporary flooring and walkways are required for the loading and unloading operations.

11. SUMMARY

The selection of the proposed design was governed primarily by the considerations of eliminating all possible hazard to the population in the surrounding area and to produce a plant that could be built in a remote area with little modification, so as to have a true prototype for the intended application. To a great extent AMF has used as a guide the conceptual design prepared by ORNL. Some refinements have been made to the primary cooling system. The handling problem has been studied extensively and improved handling tools have been designed. Much emphasis has been placed on the safety and control of the entire plant, and the kinetics of the plant have been considered carefully. AMF is convinced that the plant as outlined in this proposal is reliable and safe and fully meets the requirements of the Army.

REFERENCES

1. Robert S. Livingston, A Conceptual Design for the Army Package Power Reactor, Nuclear Sci. Technol. 1(1): 31 (February 1955).
2. A. L. Boch et al., A Conceptual Design of a Pressurized-water Package Power Reactor, Report ORNL-1613.

ABOUT THE AUTHORS

Chris J. Brous is a graduate of New York University (B.S.M.E.). He is presently Manager of the Nuclear Projects Section at American Machine & Foundry Co. He has had twelve years of experience in development, mechanical design, administration, construction, reactor engineering, and materials. He presently directs the design, development, and fabrication of various AEC projects. While at Argonne National Laboratory as Associate Engineer, he conducted in-pile experiments on materials and test flow systems. He also developed special auxiliaries for water reactor plant systems. He was Project Engineer in charge of material test and selection for the water-moderated submarine reactor program.

Franz S. Holzer is a graduate of the Polytechnic Institute of Brooklyn (B.E.E.). He has been connected with the American Machine & Foundry Co. since 1952 as a reactor control and reactor system engineer. During 1954, he was in charge of a project to develop standardized mechanisms for the control of nuclear power reactors and worked on the system analysis on several nuclear reactor plants. Before that time, he worked on the development, design, and installation of an entire control system for high-level production reactors at Savannah River.

ARMY PACKAGE POWER REACTOR

Ford Instrument Co. Proposal

CHARLES M. RICE

March 25, 1955

ABSTRACT

The Ford Instrument Co., with the Catalytic Construction Company as principal subcontractor, submitted a proposal to build and test the Army Package Power Reactor for $3,037,586.

Unique features of the design include: (1) driving control rods from beneath the reactor with the actuating system located exterior to the reactor compartment; (2) use of concrete vessels for containment in the event of a rupture in the primary system; and (3) use of hydraulically actuated control valves in the steam plant.

A windowless noncombustible building would be provided. Maximum efficiency, minimum expense, simplicity, and safe operation were the primary design requirements for both the plant and the building.

1. INTRODUCTION

The Ford approach to the preparation of a proposal for construction of the Army Package Power Reactor (APPR) was basically as follows:

1. Select that portion of the over-all job that Ford was capable of performing.

2. Determine those areas in which there was either insufficient time or need to verify data developed by the Oak Ridge National Laboratory (ORNL).

3. Request from qualified firms fixed-price subcontract bids with subsequent selection of the most favorable for handling the conventional portions of the plant and the building.

The above approach resolved itself with Ford handling the primary system, including preliminary design and selection of all components containing pressurized water, control and instrumentation of the reactor, and incident determination; Catalytic Construction Company was selected as the principal subcontractor by virtue of having presented the best of several excellent proposals for design and installation of the secondary (steam) system, installation of the primary system, and construction of the building. It was also determined that, prior to being awarded the contract, no attempt would be made to verify or amplify ORNL calculations of critical mass, flux distribution, control-rod effectiveness, and shielding requirements.

2. GENERAL CONSIDERATIONS AND DESIGN BASIS

The three stated objectives to be accomplished by the construction of the APPR at Fort Belvoir were: (1) to solve technical construction and operational problems associated with a reliable nuclear power plant; (2) to provide firm cost information, operating parameters, and engineering test data necessary to adapt the system to a remote arctic location; and (3) to provide a training facility for troops and specialists who might eventually be required to operate and service remote plants. Our design, therefore, is directly applicable only to a plant located at Fort Belvoir. Modifications would be necessary to adapt this plant for use at different locations and under different basic operating conditions.

The building was designed to be of noncombustible permanent type construction, windowless, relatively airtight, and adequately ventilated, with one door for personnel and one for trucks. The building was designed to prevent emission to the environment of quantities of radioactive material under any conditions other than actual rupture of the building. Designs for containment within this building allow access to the reactor for loading, inspection, maintenance, and replacement of mechanical components and would prevent release of contaminants to the building under the maximum credible incident that could be conceived. The major consideration in both building and reactor-system design is to provide maximum efficiency at minimum expense, consistent with safe operation.

The power cycle for this plant would consist of two main systems: The primary pressurized-water system and the secondary steam system. Heat developed in the reactor core would be transferred to the pressurized water that circulates through a steam-generating type heat exchanger. The steam developed in this exchanger would then be used to operate a conventional steam power plant.

In the primary system water at 450°F would leave the reactor core at the rate of 4000 gal/min, pass through the tube side of the main heat exchanger, where heat would be transferred to the steam cycle, and would be returned to the reactor by canned rotor circulating pumps. An electrically heated pressurizer would maintain a system pressure of 1200 psi to prevent boiling in the primary loop and would serve as a surge for volume fluctuations in the system. An intermittent bleed would be taken from the primary system to maintain a low concentration of corrosion products. Steam-cycle condensate would be further filtered and demineralized to be used for make-up water.

In the secondary system most of the steam generated on the shell side of the main heat exchanger would pass through the turbine and thence to the water-cooled surface condenser. The condensate would be recycled to the deaerating feed-water heater by the hot-well pumps. The remainder of the system would be used in the steam jet ejector and to heat the evaporator.

Contaminated water from the primary system would be passed into two submerged holdup tanks. These tanks would be programmed on alternate cycles of fill, hold, and drain. The contaminated water would be discharged below the surface of the river at a sufficiently controlled rate to maintain established radioisotope concentration tolerance levels.

3. PRIMARY SYSTEM

The major components in the primary system are the reactor pressure vessel, the reactor core, including control rods, control-rod drive and actuating system, the water-treatment facilities, a pressurizer, the heat exchanger, and connecting piping. Collectively these components were selected to comply with the basic criteria of safety, reliability, ease of maintenance, and low cost, in order of decreasing importance. Sections 3.1 to 3.7 give a pertinent description of each of these components.

3.1 Reactor Pressure Vessel

The pressure vessel would be a vertically mounted steel tank with a flanged opening in the top head. Five pipes would be welded to the bottom head and would project vertically downward. An outside diameter of 52 in., a height of 9 ft, a top head opening of 30 in., and 2 inlet openings of 12 in. would be pertinent construction features. The material of construction would be carbon steel with type 304 stainless-steel cladding.

3.2 Fuel Elements and Control Rods

It was anticipated that fuel elements and control rods would be fabricated by the Sylvania Electric Products, Inc. Their fabrication techniques are believed to have advantages over other methods, such as high degree of core to cladding bond integrity, elimination of fusion welding, and simplified assembly methods. The control rods were designed to eliminate the use of expensive materials and to incorporate other features necessitated by driving control rods from the bottom.

3.3 Control-rod Drive and Actuating System

The development of a hydraulic drive system exterior to the reactor compartment to actuate the control rods from beneath the reactor has numerous advantages. The most readily apparent are: (1) an increase in inherent safety of the reactor system; (2) the accessibility of actuators during reactor operation; and (3) simplification of reactor unloading procedure.

A bottom drive system is believed to be safer than a top drive both in case of an incident and during the normal reactor unloading procedure. In the event of a pressure surge sufficient to blow the lid off the pressure vessel, the control rods, not being attached to the top, would remain in the reactor and would not therefore be ejected rapidly from the core. A top drive necessitates latches that must be released by remote means. With no available positive indication that the latches have released the rods, it is foreseeable that one or more latches could fail, the lid be raised, and the reactor, being cool, put on a fast period. Even if the stuck latches were discovered in time, a rather touchy problem of releasing them appears certain.

The actuators would be located in a basement room adjacent to the chamber immediately below the reactor. Sufficient shielding would be provided between these rooms to allow maintenance and routine inspection of the actuators during reactor operation. The subreactor room would contain the five pipes from the pressure vessel in which the control-rod racks would be located and to which the seals and backup bearings would bolt. The rods would be driven by gears and shafts operating through the rotary seals. The subreactor room would be included in the containment volume, and shafts from the actuators to the rod drive gears would be sealed by stuffing boxes.

3.4 Pressurizer

The pressurizer would be a horizontally mounted stainless-clad pressure vessel, approximately 60 in. long by 44 in. in diameter. Flanged connections would be provided at one end for two removable 50-kw immersion heaters. The contents of the vessel would be water and saturated steam at 1200 psia.

3.5 Heat Exchanger and Primary Piping

The steam generator would be a horizontally mounted single-pass type reboiler. On the tube side would be 1200 psia water at 450°F, and on the shell side saturated steam would be generated for full load at the rate of 34,000 lb/hr at a temperature of 382°F and a pressure of 203 psia. The steam generator would have a diameter of 5 ft, a head diameter of $3\frac{1}{2}$ ft, and a length of 12 ft and would weigh 18,500 lb empty. Type 304 stainless steel would be used for the tube and header, and the shell would be carbon steel. All piping in the primary system would be welded or extruded stainless steel.

3.6 Water Purification System

Filters and ion-exchange type demineralizers, with facilities for regeneration, would be used to treat condensate from the steam condenser prior to injection as make-up water in the primary loop. Two separate systems, one for standby operation, would be provided. The units would be oversized to eliminate the necessity for a storage tank and would reduce the water impurity content from approximately 5 ppm to less than 1 ppm.

3.7 Reactor Instrumentation

Standard electronic reactor-control instrumentation would be provided with sufficient duplication so that failure of any single instrument would not require plant shutdown for repair.

Two independent start-up channels operating log-count rate meters and recorders from fission chambers would be provided. Interlocks on the recorders would prevent rod withdrawal with the count rate below a prescribed level.

Two logarithmic channels operating from compensated ionization chambers would be provided. Readings would be supplied from these channels to log and period recorders. Each channel would also include period and sigma amplifiers for period scram.

For level scram three safety channels operating from parallel-plate chambers would be incorporated. A spare magnet amplifier is supplied for use with either the period or level scram circuits. The magnet amplifiers supply power to the scram solenoid valves of the hydraulic actuators.

A channel with a removable BF_3 chamber would be supplied for low-level experiments.

The control room would record radiation levels from monitrons located throughout the plant, on the waste tanks, and on the main steam lines. Portable survey instruments and personnel monitoring equipment would also be supplied.

4. PRIMARY-LOOP CONTROL

The reactor-coolant outlet temperature would be measured and recorded and would automatically control the position of a selected control rod. The temperature in the reactor outlet leg would be maintained at 450°F. Heat flow into the steam system would be regulated for partial load by allowing the pressure and temperature in the steam side to increase. A control valve downstream of the heat exchanger would maintain a constant downstream pressure at 200 psia. An instrument measuring the coolant temperature rise across the reactor would interlock with the alarm system to give both visual and audible indications of excessive temperature rise. A separate two-point recorder on the inlet and outlet temperature would also interlock with the scram system. These independent channels ensure safety in the event of an instrument failure.

The negative temperature coefficient permits simplification of the automatic rod control servo loop. Transient temperature variations would be regulated by the temperature coefficient and the long-range variations caused by fuel burn-up, and other factors would be regulated by the automatic rod control system.

Coolant flow rate would be measured and recorded. Alarms would indicate a rate fall to some predetermined level, and automatic scram would be initiated at some lower set rate.

An indicating recording controller would measure the pressure in the primary-coolant pipeline and provide on-off control for the heaters in the pressurizer. The scram system would be actuated if the pressure departed by predetermined amounts from 1200 psia. Since the pressurizer acts as a volume surge, the liquid level therein would be continuously monitored. A level-indicating recording controller would actuate appropriate valves to maintain the desired level and would also interlock with the alarm and scram systems.

The coolant conductivity would be recorded. Excessive conductivity would be reduced by increasing the make-up rate.

Other instrumentation in the primary loop provides suitable control and/or indication for the following: coolant-pump operation, make-up water-pump operation, control-rod seal leakage, demineralizer, flow, make-up water conductivity, and sump-tank level.

5. CONTAINMENT

The problem of containment in the event of a nuclear incident with this reactor is divided into three major parts:

1. Determination of the maximum credible incident.
2. Establishment of design philosophy for containing this incident.
3. Detailed design to effect containment.

The maximum rate of change of reactivity could be controlled by limiting the rate at which control rods could be driven into the core and remembering that, with the rods actuated from the bottom, no possibility is seen for blowing the rods into the core. This should preclude the possibility of a so-called "nuclear explosion." It is hoped that an additional safeguard could be incorporated by an investigation of the use of burn-out poisons, other than boron, which might limit the available reactivity in the core at any time to a tolerable quantity.

It was therefore determined that a mechanical failure in the primary system concomitant with a rupture or melting of fuel elements would be the maximum credible incident. Such a primary-system rupture would lead to a rapid release of primary coolant as steam containing fission products with a subsequent build-up of pressure.

As a basic philosophy for containment it therefore becomes necessary to restrain completely the steam and fission products discharged from the primary loop. A calculation was made, using a most conservative assumption that the flash steam would not condense and thus lose no energy, of the pressure build-up in the containment volume for a conservative estimate of the maximum quantity of water present in the primary loop. This calculation indicates that a reasonable containment volume can be designed with a corresponding design pressure of 100 psia.

The cylindrical compartment containing the reactor vessel and the compartment containing the pressurizer, heat exchanger, and pumps were designed as concrete pressure vessels.

Although it would be possible to furnish the top head in concrete, for economy and ease in handling our design specifies a conventional steel ellipticallly-dished head. Additional concrete required to meet shielding needs would be added around the pressure vessels and reinforced for the calculated shrinkage and temperature stresses. Shield cooling as required would be provided.

In the event of a radioactive spill, contamination of the concrete would be precluded by the installation of $^3/_{16}$-in. carbon-steel plate on the side walls and bottom head. This plate would be sealed by welding to form a container within the cylindrical compartments and would be used in construction as the internal concrete form. The joints between the steel dished heads and the concrete would be sealed against pressure leaks by means of an asbestos gasket.

6. STEAM SYSTEM

A unique feature about the steam-plant control-system design is that all control valves would be hydraulically actuated. It is felt that air-actuated valves would freeze in the arctic where the plant would ultimately be used. An antifreeze compound would be used in the proposed hydraulic system for arctic application.

The steam system consists of the following standard units:

1. One 2000-kw 4160-volt 3-phase 60-cycle straight condensing turbine-generator unit with all the necessary accessories to make a complete installation.
2. A horizontal two-pass nondivided water box surface condenser complete with all auxiliaries such as air exchanger, spring supports, condensate pumps, and airmeter.
3. One single effect horizontal evaporator to supply distilled water from boiler feed-water make-up. The vapor and condensate from this unit will be used for heating the deaerating feed-water heater.
4. A one-tray type deaerating heater with a design outlet capacity of 35,000 lb/hr and a deaerated storage capacity of not less than 15 min. Approximately 30,000 lb/hr of the water supply to the heater is condensate from the surface condenser at a temperature of approximately 100°F. The remaining quantity is make-up supplied from the evaporator at 240°F.
5. A steam ejector type vacuum pump, two stages with combined heat inter-after steam throttle valve, steam strainer, and duplex drain control for automatically returning the condensate from the ejector to the main condenser.

7. ELECTRICAL SYSTEMS

Temporary service for construction power, lighting, and heating would be carried by an aerial feeder at 2300 volts.

Substation H327 would have added to it a 4160-volt 3-phase 60-cycle outdoor type switchgear unit to be used as a tie circuit breaker controlling Virginia Electric Power Company power from the secondary of the existing 2000-kva transformer to the Engineer Research and Development Laboratory bus as well as two similar units for the control of a normal and an emergency feeder to the APPR building.

An indoor metal-clad switchgear and power-center assembly would be located inside the APPR building and would consist of the following:

1. Two switchgear units to control a normal and an emergency circuit from Substation H327.
2. One switchgear unit for control of a 2500-kva turbine generator.
3. One auxiliary section for exciter control equipment and for the regulator for the 2500-kva generator.
4. One switchgear unit for the control of a feeder to a power center.
5. A power-center transformer.
6. An enclosed switchgear section containing two circuit-breaker units for the control of two circuits at 480 volts.

All transformers would be of standard manufacture with the power-center transformer being a dry type with the high-voltage winding for 4160 volts, 4 wire, Y connected, and the low-voltage winding for 480 volts, 3 wire, Δ connected. The lighting transformers would be dry type, 3 phase, 480-208/120 volt.

Motors would be of the squirrel-cage induction type for across-the-line starting and would be capable of meeting or bettering the starting torque requirements. Starting equipment for motors at 440 volts and higher would be of the circuit-breaker combination type and grouped in a motor control center in the operating area. Manual equipment with overload protection and located within sight of the motors controlled by them would be provided for motors 208 volts and less.

Control switches and indicating lights for the tie circuit breaker at H327 and all circuit breakers at APPR, synchronizing equipment for the generator, and start-stop push buttons with red and green indicating lights for all auxiliary equipment would be mounted on a control panelboard in the central control room.

Provisions would also be made for internal and perimeter lighting, thermocouple and instrument operation, visual and audible alarm systems, telephones, an American District Telegraph (ADT) fire alarm connecting to the Fort Belvoir ADT system in Building 366, and a complete grounding system.

8. BUILDING AND EXTERNAL EQUIPMENT

The reactor building would be divided into a power-plant section and an administration section. The plant section would measure 37 ft 6 in. by 85 ft with a height of 30 ft over the first floor. The reactor and associated shielding, as well as the steam-generating and condensing equipment, would be located in this section. A vault for fuel-element storage and an equipment loading area of 750 sq ft would be included.

The two-story administration section would contain offices, lavatories, locker and shower rooms, a storage and tool crib, the electrical equipment area, and some work area on the first floor. The second story would contain a laboratory, classroom, instrument repair and calibration room, and the control room.

The building would be a steel-frame structure with exterior walls consisting of prefabricated galvanized steel sheets. The roof would be constructed of precast lightweight reinforced-concrete slabs supported on steel framing.

An overhead chain-operated crane with a 15-ton capacity capable of operating the full length of the building would be provided in the plant section. The control room would contain an operator's console and 18 relay racks containing control equipment for both the reactor and the steam systems.

9. UNLOADING PROCEDURE

The compartment over the reactor would be flooded after cooling the primary system. The concrete shield top blocks would be removed and stacked to one side. The chamber over the containment vessel would then be flooded and the dish head would be removed and would be placed behind the stacked blocks. Using a specially designed high-torque wrench, the bolts on the reactor vessel lid would be unbolted and the lid would be removed and would be placed behind the stacked blocks. The upper grid assembly plate would next be unfastened and would be lifted out of the vessel using an unlatching and removal tool. The fuel elements could then be removed from the core and could be placed in storage racks provided in the skirt plate around the reactor within the concrete container.

The control rods would be prepared for removal by running the rods to their uppermost position. At that time the lower assembly grid is cleared by the fuel section. The absorber and fuel sections, which combined comprise the removable portion of the control rods, would be rotated through 45 deg to disengage them from the lower drive-rack section. The control-rod racks could be removed, if necessary, by disengaging the backup bearing assemblies from the rod drive pipes at the bottom of the pressure vessel. These assemblies would have a stud that would normally prevent driving of the rack beyond a set position.

Locating the control-rod drives beneath the reactor would allow much greater accessibility to the top for removing the pressure-vessel lid and much greater freedom in handling the lid after removal. The necessary wrenches and tools for loading and unloading have been primarily designed and appear readily fabricable.

10. SUMMARY

The Ford bid for construction, testing, and operation of this plant for six months was \$3,037,586. Roughly one-third of this price was for testing, operation, and incidentals, such as reports, liaison, approvals, developmental experiments, and bond fees.

It is felt that the major contribution of the APPR project to nuclear energy development is its demonstration that a significant number of concerns feel that the reactor art has reached a stage where reasonable fixed-price bids can be made for construction of small nuclear power plants.

ACKNOWLEDGMENTS

A major share of the credit for the proposal on which this article is based is due to K. P. Johnson and M. Silverberg of Ford Instrument Co. and W. E. Kelley and R. Sunderland of Catalytic Construction Company.

In addition, we are extremely grateful to S. Roboff and L. Smiley of Sylvania Electric Products, Inc., and numerous members of the Atomic Energy Commission, the National Laboratories, and other bidders for their guidance, consultation, and assistance.

ABOUT THE AUTHOR

Charles M. Rice is a project supervisor with the Ford Instrument Co. Prior to joining Ford early in 1954, Mr. Rice was for three years a physicist with the Reactor Branch, Oak Ridge Operations Office of the Atomic Energy Commission. He went to the AEC from Oglethorpe University where he had been associate professor of physics. Mr. Rice is a graduate of the Oak Ridge School of Reactor Technology and has the M.S. degree in physics.

ARMY PACKAGE POWER REACTOR

Foster Wheeler Corp. Proposal

THEODORE STERN

March 9, 1955

ABSTRACT

The Army Reactors Branch of the AEC, in cooperation with the U. S. Army Corps of Engineers, will build a nuclear package power plant at Fort Belvoir, Va., which will be the prototype for a package plant suitable for a remote location.

Fixed-price proposals were requested by the AEC for the design, construction, and test of this nuclear plant as based on the concept developed by Oak Ridge National Laboratory (ORNL) and presented in Report ORNL-1613.

This article is a condensed version of the Foster Wheeler Corp. proposal. The considerations that the AEC applied in choosing a reactor concept and the criteria that were to be used in selecting the prime contractor guided Foster Wheeler in the design proposal. Standard equipment was chosen wherever possible, and necessary special equipment was specified on the basis that its design would entail a minimum amount of development work. In addition to supplying the buildings and equipment for the plant, the proposal included containment features that were necessitated by the densely populated area in which the prototype plant will be located.

The cost of all the equipment and facilities, including research and development, and test and performance operations is $3,018,200. The primary loop represents $1,825,000 of this amount, and the steam and electrical system represents $808,400.

1. INTRODUCTION

Comparison studies of small conventional power plants and nuclear power plants have shown that military and economic advantages may accrue from the use of transportable nuclear power systems at certain remote arctic bases. As a result of these studies the Army Reactors Branch of the AEC, in cooperation with the U. S. Army Corps of Engineers, prepared specifications for the design and construction of a prototype nuclear power plant at Fort Belvoir, Va. The prototype plant will be used to determine the economic characteristics of the plant and the feasibility of the construction and operation of this type of unit in an arctic environment. In addition, the facility at Fort Belvoir will be used as a training center for military engineers, operating crews, and maintenance personnel.

On Aug. 19, 1954, invitations for proposals for the Army Package Power Reactor (APPR) project were sent to 33 companies or jointventure groups. The scope of the project was the complete design, construction, and testing of a prototype nuclear power plant. The technical specifications for the plant were based on the reactor concept developed in a study prepared by the Oak Ridge National Laboratory (ORNL) and presented in Report ORNL-1613, A Conceptual Design of a Pressurized-water Package Power Reactor. (A summary of this report appeared in the February 1955 issue of this journal.)

Fixed-price bids for this contract were due on Nov. 19, 1954, and the proposal of the American Locomotive Company was selected from the 18 proposals received by the AEC.

A condensed version of the Foster Wheeler Corp. proposal is given in this article. It was submitted because it is thought that a comparison of the various proposals may be of interest to reactor engineers in that the proposals indicate the various possible approaches to a problem that was presented to all on the same basis.

2. GENERAL CONSIDERATIONS

Selection of the prime contractor was based on three criteria: responsiveness, lowest price, and the degree of contribution of the proposed design toward the most practical and economical unit for use by the Army in the field, consistent with the AEC specifications.

The reactor concept presented in Report ORNL-1613 is a heterogeneous light-water-cooled and -moderated system using highly enriched solid uranium oxide fuel in a stainless steel, boron carbide matrix clad in stainless steel.

In proposing a design for the APPR project, Foster Wheeler was guided by the criteria that were to be used in selecting the prime contractor and by the considerations that the AEC applied in choosing a reactor concept. Accordingly, developed and proved standard components were used wherever possible, and necessary special equipment was chosen on the basis that its design would entail a minimum amount of development work. Since the facility was to be the prototype for a plant to be constructed in the arctic, it was decided that this plant would be most practical if the reactor components and their arrangement were such that the complete unit was reliable and safe, requiring minimum maintenance under continuous operation, and capable of conversion for use in the arctic with the least number of modifications. Equipment arrangement, shielding, and containment were also influenced by the training-facility objective.

Since lowest price was one of the criteria that was to be used in selecting a contractor, it was realized that the above objectives had to be compromised with the cost of the equipment. Therefore the over-all objective of the plant design was as uncomplicated a system as possible, capable of arctic service, without sacrificing reliability or safety.

Special mention of the considerations given to containment is warranted. The proposed contract between the contractor and the AEC stated that if the Advisory Committee on Reactor Safeguards agreed with the maximum credible accident described by the contractor but termed inadequate the measures described for containment of this accident and if the Commission required the contractor to make changes in his containment to comply with the recommendations of the Committee, then the Commission would determine (1) whether the containment proposed by the contractor is inadequate to contain the maximum credible accident or (2) that the required changes based on the Committee recommendations were due to opinion differences and were not the result of actual inadequacies in the containment measures. If the determination of the Commission were the former, the contractor would make the required changes without additional compensation.

If the Commission required changes in the containment measures because the Committee disagreed with the contractor's statement of the maximum credible accident, an equitable adjustment would be made for any increase or decrease in the cost of containment.

The technical specifications for this plant did not differ except in one respect from the specifications that may be set for any other 10-megawatt (heat) nuclear plant that must supply continuous power. No single component of the plant, with the exception of the turbogenerator, main condenser, containment vessels, waste tanks, and shield, could exceed 10 tons in weight nor 7 by 7 by 18 ft in over-all dimensions.

3. PLANT DESCRIPTION

3.1 Over-all Plant Layout

The plant location is Fort Belvoir, Va., at a site along Gunston Cove which connects with the Potomac River.

The entire primary loop is contained within a steel envelope consisting of two container vessels and connecting piping. The reactor container vessel has two spent-fuel-element storage trays for storing fuel elements in a subcritical lattice and a support for holding a coffin during refueling. The biological shield is concrete, the exposed surfaces of which are dustproofed and hardened.

Filtered air, drawn through the space between the containers and the shield, provides cooling for both the shield and the container vessels. The air is exhausted through screens to the atmosphere through a 125-ft stack.

3.2 Primary Coolant System

The power cycle consists of a primary coolant system and a steam system. The reactor is a heterogeneous water-cooled and -moderated stainless-steel unit using highly enriched solid uranium fuel elements arranged in convenient subassemblies.

The water in the primary loop is maintained at 1200 psia and is circulated through the reactor at a sufficient rate such that nonboiling conditions prevail. The reactor is regulated by five control rods, only one of which is used as a regulating rod; this rod is set to maintain the coolant outlet temperature from the reactor at 450°F. The cooling water is circulated through 12-in. piping to an integral U-tube boiling type steam generator and then to one of two canned-rotor circulating pumps, each of which can supply sufficient head to overcome friction in the loop for a flow rate of 4000 gal/min. The system is so designed that, in the event of failure of both pumps, the reactor will be shut down and the decay heat will be removed by natural circulation. The primary-loop flow diagram and heat balance are shown in Fig. 1.

A small portion of the primary coolant water is continually purged from the system to maintain a solids concentration of about 2 ppm and a conductivity of 4 micromhos. The purge rate is controlled by the conductivity of the primary loop and by the pressurizer liquid level. If the level becomes too high, the purge rate is automatically increased, thus returning the level to its normal position.

Make-up to the primary loop is taken from the plant demineralizer and is introduced into the system by one of two positive displacement pumps, which are actuated by a signal from the pressurizer liquid-level recorder controller. One of the make-up pumps is held in standby, and, if one of the pumps fails to start from a signal of low liquid level, the second pump will automatically be cut in. The make-up water contains hydrogen at a concentration that is sufficient to ensure a maximum oxygen content of the primary coolant water of no more than 0.1 ppm.

All piping and metal surfaces in contact with the primary coolant water are made of type 304L stainless steel, and the primary loop is so designed that expansion stresses will be absorbed without requiring movement of either the reactor pressure vessel or heat exchanger.

3.3 Steam System

Saturated steam at 203 psia is produced in the steam generator when the reactor is operating at full load. This steam passes to a turbine generating unit consisting of a 2000-kw steam turbine designed for steam conditions of 200 psia and 382°F at the turbine throttle, driving a three-phase 2500-kva 4160-volt generator with a direct connected exciter.

The turbine exhausts to a surface condenser, and the condensate is pumped from the hot well by either of two 50 gal/min horizontal centrifugal pumps through an air ejector and into a deaerating heater. Two 75 gal/min horizontal centrifugal pumps take their suction from the deaerating heater and deliver the feed water to the steam generator in the primary loop. Make-up is supplied through a mixed-bed demineralizer with adequate provision for storage of treated make-up water.

A screen house at the river bank is equipped with two traveling screens and three vertical motor-driven circulating water pumps. Two of these pumps are required for full-load operation; the third is a spare.

Three-element control is provided for this system, with one element on feed water to the steam generator, one element on the steam-generator water level, and the third element on the flow to the turbine. The feed pumps will be provided with automatic recirculation control to protect them during no feed water flow periods.

The gross electrical output of the turbo-generator is 2055 kw. Auxiliary power requirements are 125 kw for the steam system and 115 kw for the primary coolant system; therefore the net available output is 1815 kw.

Power is generated at 4160 volts and, through suitable switching equipment and connections, is delivered to a 4160-volt bus and then over an outdoor line to an existing substation. Necessary relaying equipment is provided to permit the local power company circuit to remain the preferred source of supply to the substation, with the APPR circuit operating in parallel on the secondary bus, without permitting any back surge of power into the power company lines. A station auxiliary transformer, obtaining its supply from the 4160-volt bus, delivers power through a system of switching equipment to various APPR plant auxiliaries.

3.4 Buildings

The building that houses the power plant and service facilities has a reinforced concrete foundation, a structural steel framework with insulated metal siding, and a built-up roof laid on a steel roof deck. A 15-ton crane with a 1-ton auxiliary hook operates over the power-plant section.

The screen house is of the same general type of construction as the main building. In addition, a guardhouse is provided, which is composed of insulated metal siding and roof supported by a light structural steel frame.

Improvements include landscaping, roadways, fences, and adequate parking facilities.

4. REACTOR COMPONENTS

4.1 Reactor-core Assembly

The reactor, shown in Fig. 2, is very similar to that shown in Report ORNL-1613. The Sylvania Electric Products, Inc., fuel-element assemblies are duplicates of those in Report ORNL-1613, except that zirconium diboride is added as a burn-out poison instead of boron carbide, which would have caused difficulties in the fabrication of the fuel plates. Before the fuel assemblies are installed in the reactor core, they are checked and tested for integrity of the bond, material distribution, and resistance to thermal shock and thermal-cycling failure.

The fuel plates are designed for use in both the fuel assemblies and the fuel section of the control-rod assembly. The control rods differ from those discussed in Report ORNL-1613 in that the hafnium specified for the connecting ends of the two-piece control rod may be replaced by a cheaper material. Both boron steel and a stainless-steel-clad silver-cadmium alloy are considered good possibilities. The latter alloy was corrosion tested by Sylvania, and the results were encouraging; however, no decision can be made until further tests are conducted.

4.2 Control-rod Drive Mechanism

The control rods are operated by a canned motor-driven mechanism, which is the latest improved version by the Westinghouse Electric Corporation of a mechanism developed for the Submarine Thermal Reactor (STR) (see Fig. 2). Essentially, the mechanism consists of a reluctance type motor rotor directly coupled to a roller nut. This nut and the rotor are canned and operate submerged in the primary coolant water. When the mechanism is in operation, the roller nut is held firmly (by magnetic force) in contact with a lead screw attached through a latch mechanism to the control rod. The latch mechanism is similar to that employed in Report ORNL-1613. Surrounding the canned rotor of the drive mechanism is a polyphase-wound stator. Electrical rotation of this stator field turns the rotor and roller nut, which, in turn, raises or lowers the lead screw.

The drive mechanism contains a fail-safe magnetic release that permits the screw and control rod to fall free on signal or on loss of magnet power. Each rod is provided with its own power supply so that singular rod motion or various ganged motions are possible, permitting maximum flexibility.

The design of this mechanism assures zero leakage of the reactor fluid. The outer casing

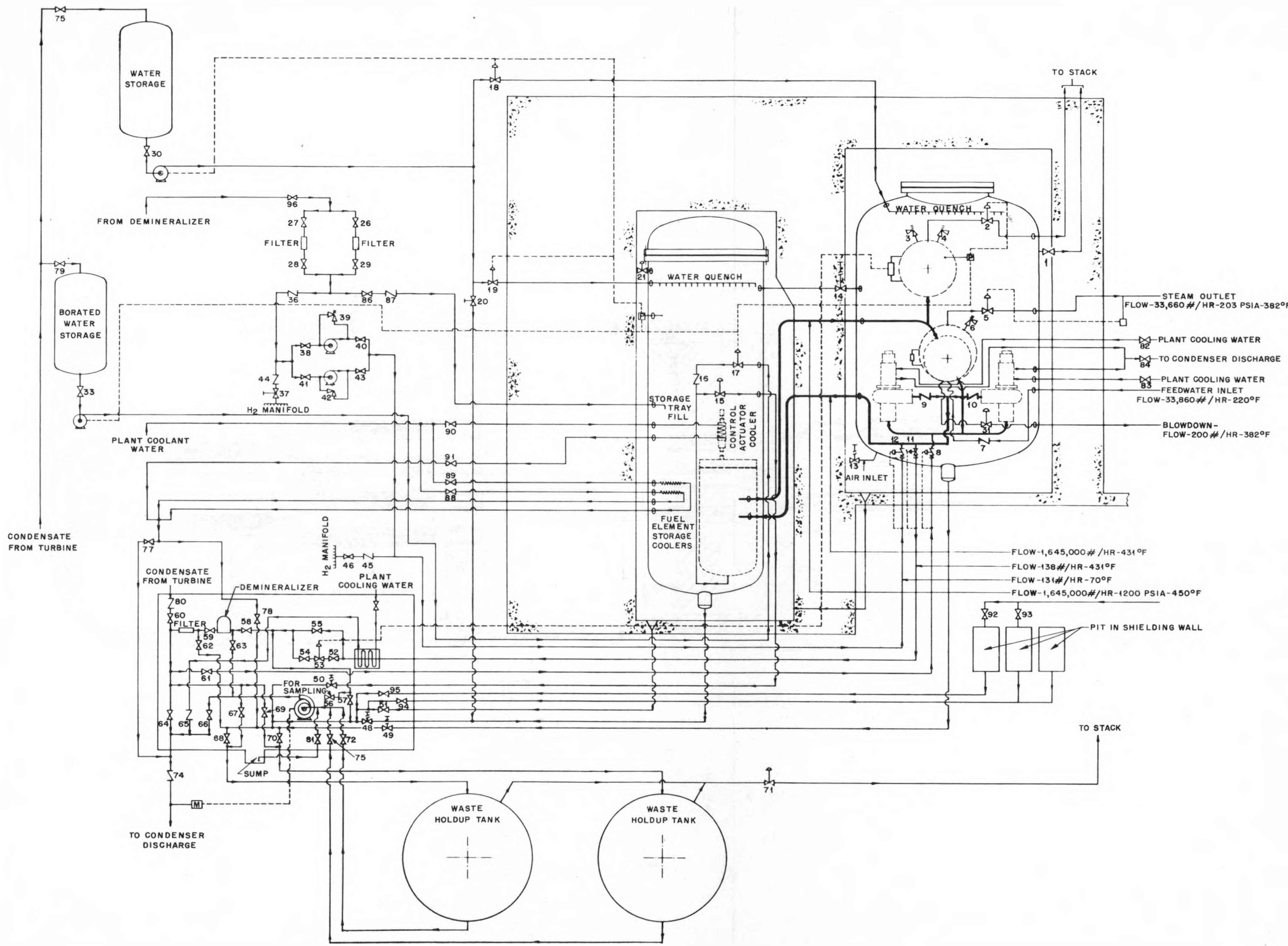


Fig. 1 — Primary-loop flow diagram and heat balance.

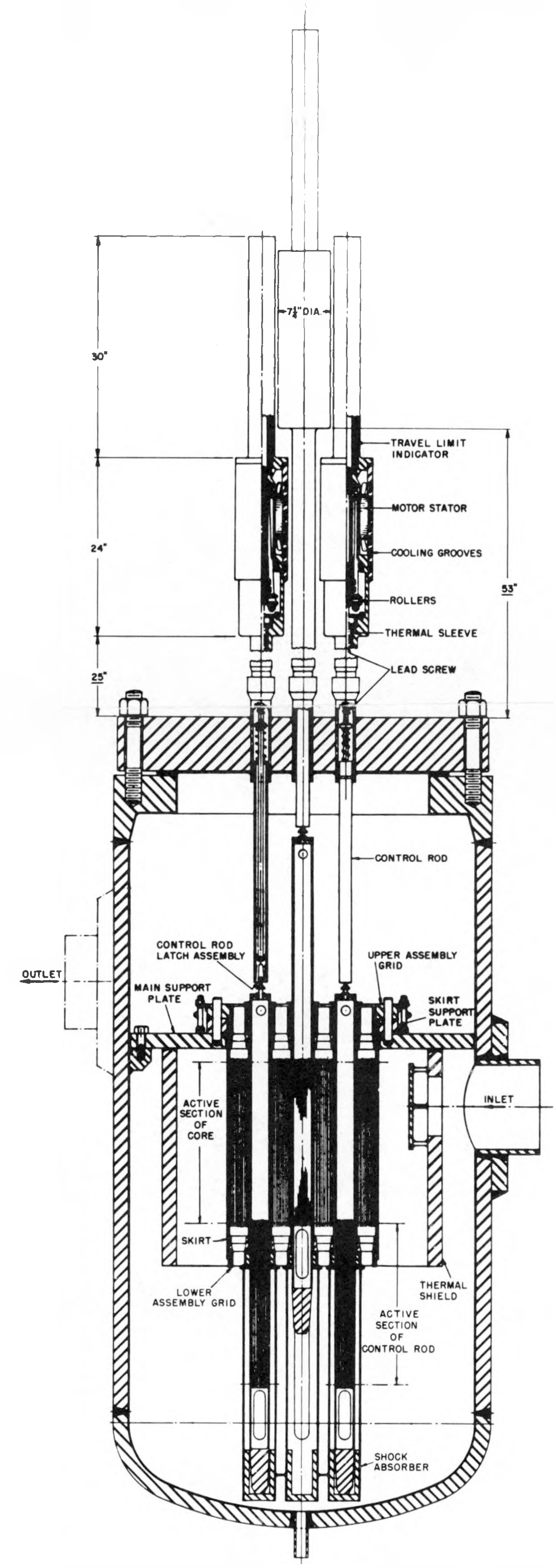


Fig. 2—Reactor assembly and control-rod drive mechanism.

is seal welded, and proper protection is supplied for the electrical and cooling-water leads so that the mechanism can operate under water.

A thermal sleeve surrounding the lead-screw extension restricts the flow of high-temperature fluid through the rotor cavity; therefore the fresh-water cooling system on the motor stator is able to maintain reduced temperatures around the mechanism parts. As a result the ambient temperature outside the mechanism-enclosing can is not restricted to a 150°F range, thus reducing venting requirements.

4.3 Reactor Pressure Vessel

All pressure vessels are designed to the ASME code. The design conditions of the reactor vessel are 1250 psia and 650°F. The base materials for the pressure parts are SA-212 grade B carbon steel for plate and SA-105 grade II carbon steel for forgings; all internal surfaces of the vessel and the cover are lined with type 304L stainless steel. The vessel has an over-all height of 9 ft $2^1/_4$ in. and a 4-ft I.D. Welding on the vessel wall is back-chipped and rewelded with stainless-steel filler rod to provide continuity of the inside stainless-steel surface. All longitudinal and circumferential butt welds on the pressure parts are radiographed. In addition, root passes and final inside passes on the pressure-vessel welds are examined by Zyglo. Weldments in carbon steel are examined by Magnaflux. After assembly the vessel without the head cover is thermally stress relieved.

4.4 Pressurizer

Connected to the primary loop, at a point just before the coolant enters the steam generator, is a pressurizer. Three 36-kw immersion type heaters are used to maintain the pressure of 1200 psia on the system. As the pressure drops, a pneumatic relay turns on one of the 36-kw units. A further drop in pressure actuates relays that turn on the remaining heaters in 18-kw steps. Safety valves located on the pressurizer relieve steam at overpressures. The pressure-vessel shell and heads are made of SA-212 grade B carbon steel clad with type 304L stainless steel. The welds are tested as before, and the pressurizer is stress relieved after welding.

4.5 Steam Generator

The steam generator is of integral design with the heat-transfer surface and the steam separating equipment included within a single vessel. A U-tube bundle is utilized, which eliminates the differential thermal-expansion problems encountered in straight-through heat exchangers. The tube sheet is welded to the heat-exchanger vessel, eliminating leakage of primary water into the steam system.

Chevron driers, furnished in the steam space, permit only high-purity steam to flow to the turbine. The over-all length of the generator is 11 ft 2 in., and the outside diameter of the head is 54 in. This represents a considerable reduction in size over a straight-tube steam generator.

4.6 Primary Coolant Circulating Pumps

The primary coolant is circulated through the system by one of two canned-rotor pumps placed in parallel in the loop. The pumps provide for zero leakage, and, since the pump and motor form an integral unit, only two bearings are required, thus eliminating alignment problems. The pump and motor construction of this design permits the removal of the motor and impeller from the casing as a single unit. Material of construction is type 304 stainless steel.

4.7 Liquid-waste Disposal System

The liquid-waste disposal system is comprised of two 14,500-gal carbon-steel holdup tanks, a pump, cooler, demineralizer, and filter. All this equipment, with the exception of the holdup tanks, is located in an underground room approximately 10 ft outside the biological shield. Each of the two tanks, which are buried in the ground, has a capacity capable of holding all the primary-loop water as well as the water used to flood the reactor container vessel during refueling.

All lines to the waste tanks are controlled by valves located in the underground room. Although it is possible to obtain access to this room while the reactor is on, the manual valves can be operated by extensions ending in the building itself.

The cooler, located in the primary purge line, is used to prevent flashing of the hot water to steam. The pump is a self-priming centrifugal type providing a 20-ft lift. Valving is so arranged that the waste in the tanks can be discharged directly to the river, where it is diluted with the condenser circulating water discharge, or the wastes can be passed through the demineralizer and filter and then to the river (or it can then be returned to either one of the holdup tanks). The demineralizer is of the replaceable cartridge type, and the filter can be backwashed to either holdup tank.

4.8 Reactor Control System

The control-circuit design is in keeping with the plant design objective — as uncomplicated a system as possible, capable of arctic service, without sacrificing reliability or safety. This was accomplished by utilizing the experience Westinghouse gained in the operation of the STR control and safety system.

A schematic diagram of the control and safety circuit is shown in Fig. 3. The source and intermediate range comprise a pulse and d-c channel, respectively, with associated power supplies, level and period indicators, and level and period recorders. Start-up interlocks are provided.

A bistable magamp in the intermediate-range period circuit provides a shutdown signal to the bistable-magamp magnet supply, which in turn causes the magnetic clutches to release their hold on the control rods. A self-contained test panel provides pulse, level direct current, and period signals for checking and alignment of all channels. Three power-range channels operating in parallel provide level indication through a selection meter and three-point recorder. Each channel at a flux level of 1.5 that of full load will trip a bistable magamp whose output into the sigma bus will in turn trip a power bistable magamp, thus removing holding current from the magnetic clutches. The outputs of the channel bistable units may be set by prior switching so that a trip of any two of the three available will be required for shutdown. This coincidence feature reduces the possibility of reactor power interruption if a malfunction or transient should occur in one channel. The alarm circuit, provided for all trips and interlocks, will signal the operator that a second flux-level trip will cause shutdown.

A resistance thermometer keeps the outlet temperature of the primary coolant constant. The power level, represented by the flux, is fed into a magamp comparator as a check on the temperature controller. If the power level exceeds a certain maximum setting, the rod-controller motor will act to insert the regulating rod to lower the power level. Therefore the temperature is the main controlling factor up to this maximum power setting, at which time the flux level overrides the temperature signal and in turn becomes the controlling factor.

No vacuum tubes are used in the power-range indicator, control, and safety circuits. All components except the vacuum tubes (in the start-up circuit) are designed for extra long life, and all high-impedance circuits are sealed and desiccated for protection from atmospheric conditions.

All readings (including radiation and leakage checks from the boiler leak detector, air-borne particle detector, and air and waste monitors) are indicated in the control room on a console or on the control panels. The console contains all those functions that the operator requires to maintain control of the reactor in the function of start-up, power-range operation, and safety, with supervisory control of the steam system.

5. CONTAINMENT

The maximum credible accident that is assumed to occur in the APPR is that of a rupture in the primary loop while operating at rated power.

The primary loop is contained within a steel envelope consisting of two large cylindrical vessels appropriately interconnected. The 12-in. piping between the reactor vessel and the steam generator is enclosed in a 24-in. pipe envelope connecting the container vessels. When the reactor is in operation, the containers are completely sealed.

All lines passing through the containers, such as the primary-loop make-up line, are fitted with valves located inside the containers. In the event of a rupture in any of these lines, the valves will automatically close, preventing primary coolant from escaping to the outside of the containers.

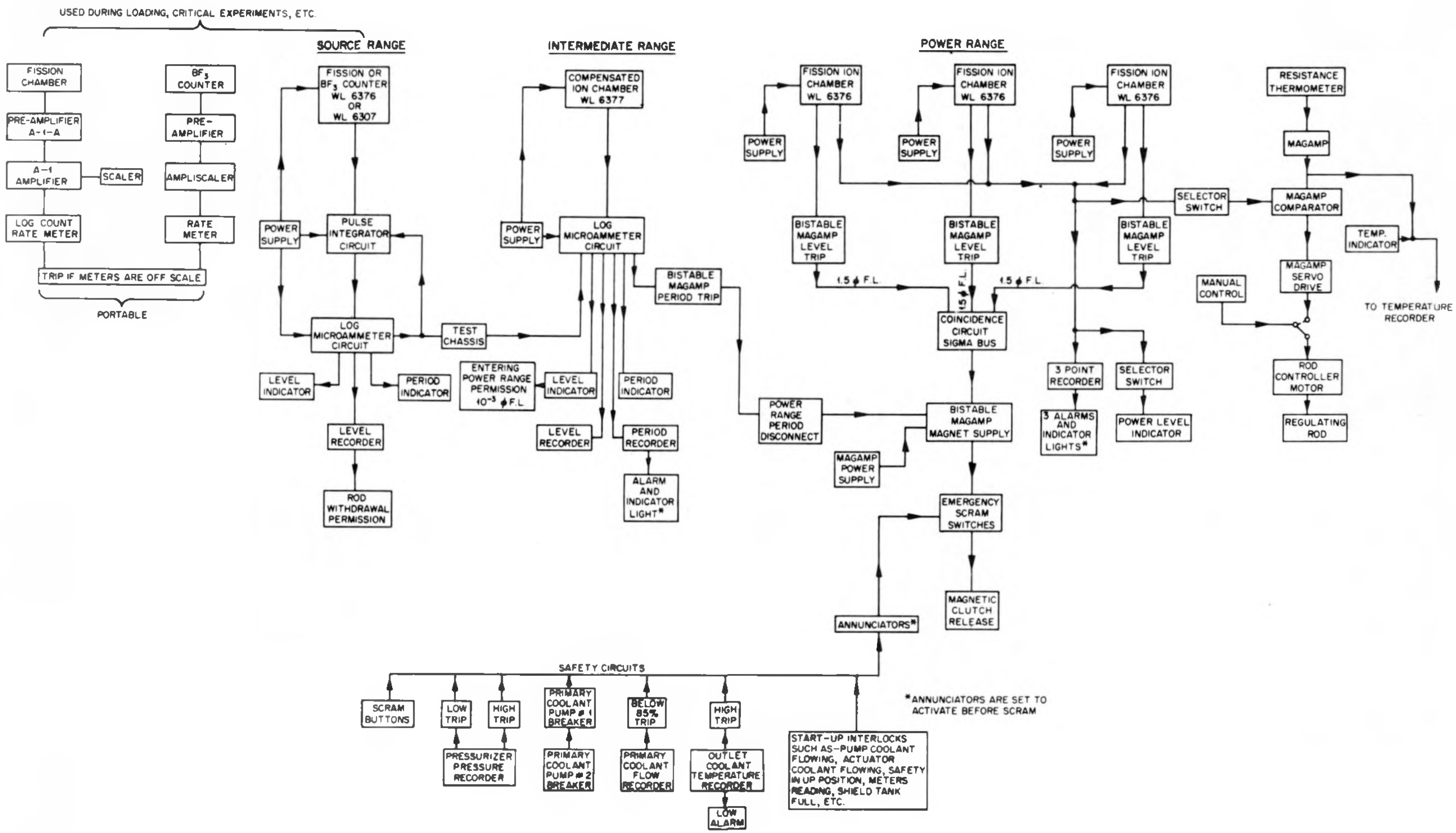


Fig. 3 — Diagram of control and safety system.

A rupture in the primary loop permits the primary coolant water to flash (through the break) to steam and liquid water at a lower pressure and temperature. If the break is such that all the primary coolant flashes from 450°F (equivalent to 10 sec additional operation at 10 megawatts), the pressure inside the containers will be slightly under 150 psia.

The reactor should scram from any one of several signals actuated by the rupture. Should the rods not drop (the case where the break may distort the rods), then the reactor may not be subcritical until approximately 70 per cent of the primary-loop water has flashed to steam. However, it is undesirable to have the water flash to steam because the fuel elements will melt. Therefore additional water must be added to cool the fuel elements; but at the same time the reactor must be subcritical, and the pressure build-up in the containment vessels must be curtailed. This is accomplished by pumping borated water into the reactor (to cool the elements and add poison to the core) and by introducing spray water at the top of the containers (to prevent excessive build-up of pressure by condensing steam).

The spray system is started when the container pressure exceeds 50 psia. Borated-water pumping occurs automatically when the primary-loop pressure drops to 500 psia and the container pressure rises to 50 psia. At a pumping rate of 200 gal/min, it takes 1.25 min for water to reach the bottom of the fuel elements. The maximum temperature the fuel elements reach before they are completely covered by borated water is 1460°F. In the event of a rupture occurring at the bottom of the reactor vessel, the fuel elements will reach a temperature of 2175°F before they are covered by borated water.

Failure of both the borated water and spray system will cause melting of the fuel elements, but the heat-removal rate of the air outside the containers is such that it can remove the decay heat generated. Even if air cooling is not available, there is no danger of melting the containers, although a 10-psi pressure rise may be expected.

For missile protection it is possible to install a wire netting, coated with Gunite, several inches away from the inside of the container walls. The containers themselves are designed according to the ASME pressure code for the containment of a lethal substance. The material of construction is carbon steel.

Decontamination is easily accomplished. The inside of the containers can be flooded and washed using the existing spray quench system. It is also possible to acid wash through the quench system. All waste can be passed from the containers directly to the existing waste-disposal system.

6. REFUELING OPERATION

After the reactor is shut down, all control rods are run down until unlatching lights indicate that they are in the unlatched position. Water from the storage tank is pumped into the reactor container vessel, forcing the air in the container into the steam-generator container, where it is slowly bled into the stack. When the container is almost completely filled, a valve in the top is opened to permit fresh air to replace the old air that remained in the container. In the meantime, the top shielding plugs are removed, and, because the blower is still in operation, air will be drawn from the building into the shield through the top plug opening. The top closure of the reactor container is unbuttoned and placed in the storage pit in the shield, and then four of the shield plugs are replaced to reduce radiation leakage.

Next the reactor pressure-vessel closure is unbolted and is raised 2 in. Each control drive rod is then driven to its top position while nuclear instrumentation is watched to make sure that the control rods are not being lifted. (If possible, positive unlatching indication should be provided to make this step unnecessary.) After the drive rods are raised and coolant and electrical lines are disconnected, the closure plate with the drive mechanisms is raised and placed in its storage pit.

The upper assembly grid is then placed in storage, and the fuel elements are moved from the core to the storage trays located next to the reactor in the container. New fuel elements are loaded into the core under conditions simulating a critical experiment.

The closures are replaced in exactly the reverse procedure, and the water in the reactor container vessel is drained to the holdup tanks in the waste system. The water used for refueling is not demineralized and is in contact with the carbon container during refueling. Some mixing of this water with the water in the primary loop has probably occurred; therefore the loop water is circulated through the system with the bypass demineralizer line open. While this operation is in progress, the heaters in the pressurizer are turned on to build up pressure, and at 200 psia the reactor vessel blowdown valves are opened to blow out any solid material that might have fallen into the vessel during the refueling operation. After the conductivity has been reduced to the proper level, the plant is ready for power operation.

7. COST ANALYSIS

The cost of the equipment and facilities described above, including fuel elements and control rods, research and development, various test operations, reports to the Advisory Committee on Reactor Safeguards, and a 700-hr performance test, is $3,018,200. Research and development includes sample fuel-element irradiation tests, development of a substitute for hafnium for the connecting ends of the control rods, and development of a more suitable control-rod snubber and unlatching indicator. Approximately 200 hr of calculations on the UNIVAC or the equivalent is required to complete the reactor-operation analysis. The fuel, in a form suitable for this application, is received from the AEC at no cost.

The primary-loop cost (including containment) represents $1,825,000 of the total, and the steam and electrical system (including shielding) represents $808,400. The cost of the containment feature was estimated to have added $150,530 to the cost of the plant. This includes the actual cost of the equipment in the safety system plus estimated costs for the increase in the total cubic feet of concrete required, as well as the extra expense of installing the equipment.

Plant cost without research and development, reports to the Safeguards Committee, the 700-hr performance test, and the containment feature is approximately $2,400,000.

With this figure as the basis for the capital investment for a plant of this type, the cost per kilowatt of electricity (net) is $1320. Again, on the same basis, the cost per kilowatt of heat is $240.

ACKNOWLEDGMENTS

The proposal submitted to the AEC, of which this is a condensed version, represents the efforts of many people both in the Foster Wheeler Corp. and in other companies.

Credit is due the Pioneer Service & Engineering Co. for providing the building and turbine plant design. Many other companies helped us arrive at costs.

ABOUT THE AUTHOR

Theodore Stern is a project engineer in the Nuclear Energy Department of the Foster Wheeler Corp. He received the B.Mech.E. degree in 1951 from Pratt Institute and has done graduate work in mathematics and physics at New York University. He attended the 1951 to 1952 session of the Oak Ridge School of Reactor Technology and later joined the Foster Wheeler Corp. AEC Industrial Participation Program. His present activities are concerned with reactor design.

ARMY PACKAGE POWER REACTOR

Kidde Companies Proposal*

JAMES J. BARKER, JOHN FAAS, and
WILLIAM L. WEBB†

March 3, 1955

ABSTRACT

The design proposed herein is similar to that described in Report ORNL-1613[1] with modifications to meet the Army Package Power Reactor requirements and to adapt the plant to the Fort Belvoir site.

The primary flow is 6000 gal/min at 450°F and 1250 psia. A purge demineralizer and a boron-solution emergency scram are included. Saturated steam at 240 psia is generated using a simple regenerative cycle with three stages of feed-water heating. The net generation is 1800 kw for a 10-megawatt reactor input with 85°F cooling water in the two-pass condenser. The largest heat loss is in the purge circuit.

Ordinary concrete, supplemented by water and earth, is used for shielding. Concrete thicknesses range from 5 to 8 ft around the reactor compartment to 2 ft around the remainder of the primary circuit.

The process auxiliaries consist mostly of water storage tanks, exchangers, pumps, and demineralizers.

Total containment is provided for all radioactive emanations, gaseous or liquid. A partly buried spherical steel enclosure is provided to contain the maximum credible accident, which occurs as a result of a relatively slow accumulation of energy in the water of the primary circuit and leads to rupture of the system and flashing of the water to steam.

1. INTRODUCTION

The design and specifications that were proposed for the Army Package Power Reactor (APPR) are based on Report ORNL-1613[1] modified to meet the design requirements accompanying the Invitation for Proposals. The design was developed primarily to determine feasibility and to estimate cost.

The requirements (Appendix B to the invitation) limited the design fairly closely to that outlined in Report ORNL-1613. However, we did not feel that this limitation was unduly restrictive since the reactor proposed in this report is eminently suitable for the purpose. The only significant modifications in the primary loop were an increase in flow rate from 4000 to 6000 gal/min and the addition of a demineralizer. The design of the secondary loop

*Original proposal signed by: John F. Kidde, President, Walter Kidde & Company, Inc., William Collins, President, Walter Kidde Constructors, Inc., Karl Cohen, Vice President, Walter Kidde Nuclear Laboratories, Inc.

†Affiliated with the American Gas & Electric Service Corporation as head of the Nuclear Power Section. This corporation acted as consultants on power-generation equipment.

was modified to take into account the increased power demand, the removal of the space-heating requirement, and the elimination of arctic conditions. Other modifications, including a boron-solution scram system, are described later on.

A principal addition to the 1613 design is the containment provision. The site is near a populated area and is near one of the most sensitive spots in the country, Washington, D. C. There are possibilities of winds in the direction of Washington, and the hydrology shows that radioactive materials entering ground waters could enter public water supplies. We therefore adopted a philosophy of total containment of all radioactive emanations, gaseous or liquid, and have provided a sealed reactor enclosure for this purpose. The size of this enclosure was determined by the flashing of the primary coolant under conditions of the maximum credible accident. Our evaluation of this accident showed that the total energy released in the accident would be only 40 per cent larger than that released by flashing of the water in the primary loop in the event of failure during normal operation.

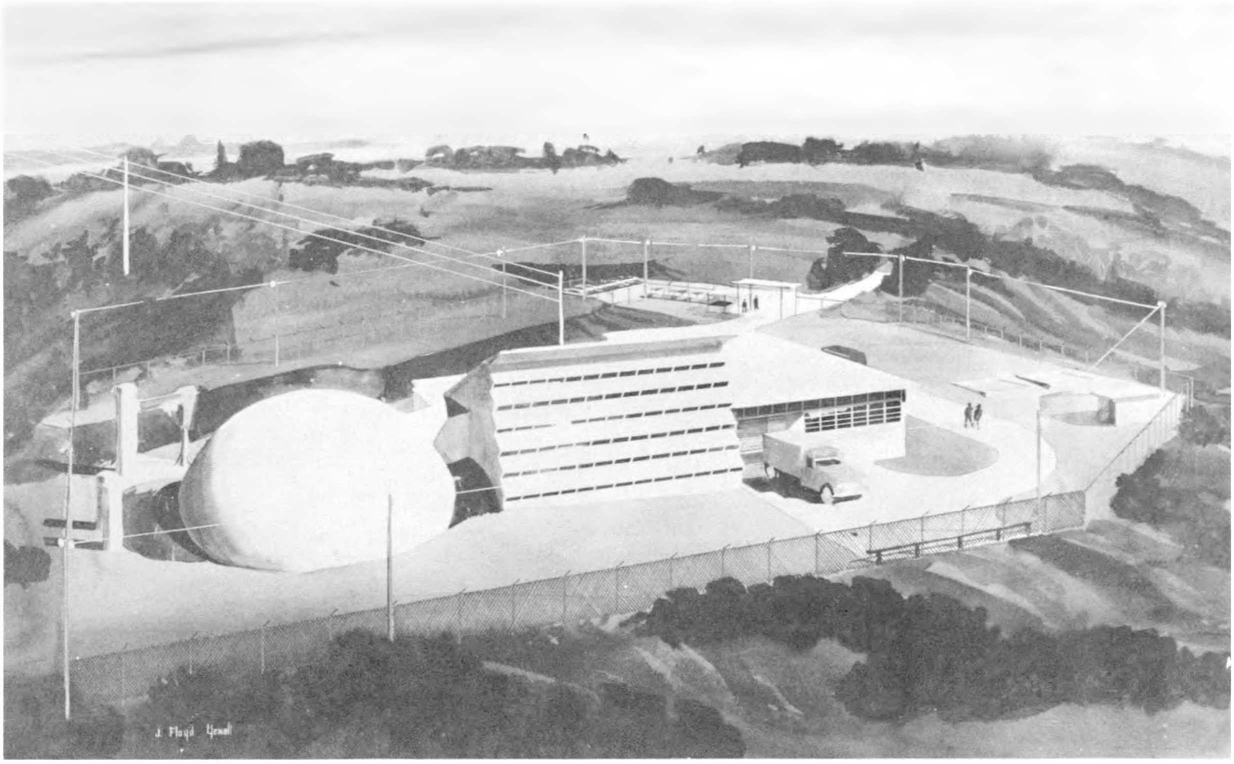


Fig. 1—Army Package Power Reactor.

The layout of buildings and auxiliaries in this particular design is closely determined by local characteristics and requirements. Adaptation to other locations could be made in a straightforward way. All essential parts of the plant can be transported by air to an isolated site.

Figure 1 is an architectural rendering of the plant. The reactor and primary circuit are inside the spherical enclosure (part of which is below ground) and the steam plant, control room, laboratories, shop area, offices, etc., are housed in the auxiliary building. Figure 2 gives the heat and power balance for the plant. The reactor supplies 6000 gal/min of water at 450°F and 1250 psia to the boiler, which delivers saturated steam at 235 psia to the turbine. With 4300 gal/min of cooling water at 85°F from

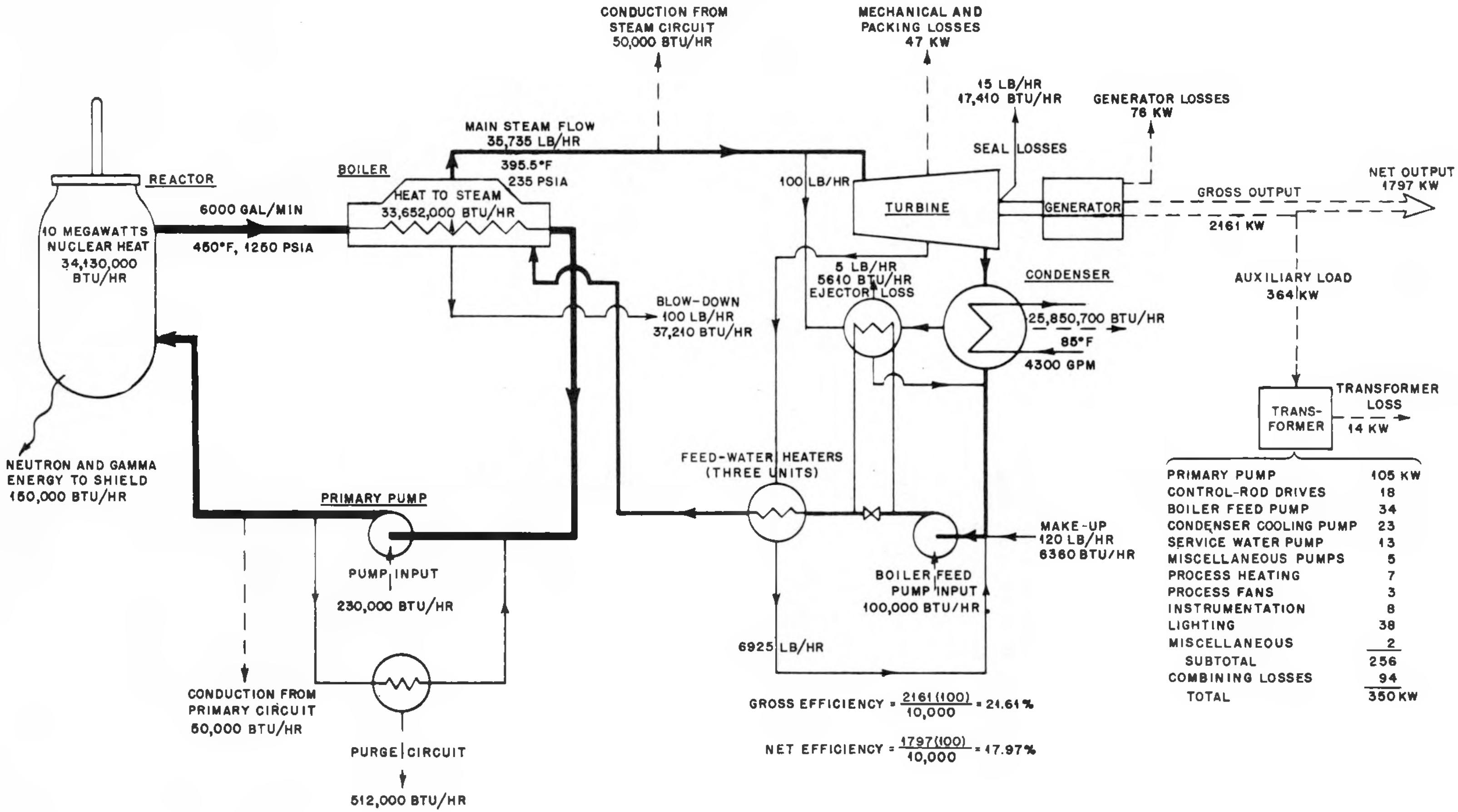


Fig. 2—Heat and power balance.

Gunston Cove pumped through the condenser, the net output of the plant is 1800 kw.

2. REACTOR COMPONENTS

Our studies on reactor components concentrated on core heat transfer, nuclear control and instrumentation, the control-rod drive mechanism, and reactor-shield configuration and heat dissipation.

Table 1 — Summary of Key Characteristics of Reactors*

Fuel elements:
- Type: Rectangular, flat, UO_2 – stainless steel – B_4C core, clad with 5 mils of type 304L stainless steel
- Dimensions: Core width, 2.5 in.; length, 22 in.; thickness, 0.02 in.; over-all, 2.76 by 23 by 0.03 in.
- Spacing between plates: 0.134 in.
- Fuel plates per fuel assembly: 18; number of fuel assemblies: 40
- Fuel plates per control-rod assembly: 16; number of control-rod assemblies: 5
- Composition of core: UO_2, 17.94 wt. %; stainless steel, 81.88 wt. %; B_4C, 0.18 wt. % w/O

Core:
- Average diameter: 22.2 in.
- Height: 22 in.
- Inventory, 17.7 kg of U^{235} new; 10.2 kg of U^{235} after 15 Mw-years
- Heat-flux area: 550 sq ft on fuel assemblies, 61.1 sq ft on control-rod assemblies
- Water-flow cross section: 2.14 sq ft
- Stainless-steel content: 98 kg in matrix, 208 kg total

Reflector: Water, thickness, 7 in.

Thermal shield: Steel and water

Vessel: Over-all dimensions approximately 4.5 ft diameter by 9 ft high

*Abstracted from reference 1.

We tentatively accepted the reactor core dimensions, the number, fuel loading, and dimensions of the fuel rods, and the number and dimensions of the control rods, all as described in ORNL-1613. Table 1 gives the basic information on the reactor.

2.1 Required Coolant Flow Rate

During periods of maximum reactivity the heat-flux area in the core is 550 sq ft, since the 61.1 sq ft of fuel-plate area on the bottom sections of the control rods is then out of the reactor core. This results in an average heat flux for the design case of 62,000 Btu/hr/sq ft.

Preliminary calculations (Table 2) show that a total flow rate of 4000 gal/min is insufficient to suppress incipient boiling in the core when allowances are made for unfavorable heat-flux distributions. We based the design of the plant on a total flow rate of 6000 gal/min to provide some safety on reactor-core heat transfer.

2.2 Nuclear Control and Instrumentation

The nuclear control system is designed to meet the following principal requirements:

1. Control of reactivity over life of reactor and over extreme ranges of conditions from cold clean to hot, maximum xenon poisoned.
2. Rod motion to be sufficiently rapid to permit start-up after scram within one-half hour.
3. Maximum rod withdrawal rate and scram delay time are coordinated to avoid damage to fuel elements in a start-up accident.
4. Interaction of rod motion with the tendency of the reactor to keep its average temperature constant will avoid self-exciting oscillations over a power range from 5 to 100 per cent of design. Power fluctuations under steady load conditions will be held to 2 per cent.

Other requirements that the control system must meet are that it fails safely, that continuous indications of rod positions are given, that rod positioning be accurate to a distance corresponding to a small variation of the reactor temperature, that rods be adequately cooled, and that they be spaced in the reactor core and their motions be programmed in such a way as to create the most uniform flux pattern at all times.

A design requirement is that the reactor can be shut down with only 80 per cent of the rods functioning. If the central rod does not function, the calculations do not show a comfortable margin of safety between the effectiveness of the four eccentric rods and the maximum reactivity of the core. Critical experiments on control-rod effectiveness will eliminate most of this uncertainty. The balance of the discussion will be valid whether the final number of rods is five or six.

The nuclear control system is based on the use of Westinghouse equipment, with the following special characteristics:

1. The circuits leading to the sigma bus do not use vacuum tubes and therefore may be expected to be relatively easy to maintain.

2. There is no need for a mechanism to move the fission chamber.

3. The use of a continuous stream of nitrogen gas through the ion chambers is avoided.

The block diagram is basically that shown in ORNL-1613 with a few additional features:

1. Two of the power range ion-chamber circuits must call for scram before the rods will be released. When only one calls for scram, a warning light on the console lights up.

2. A direct power-measuring system is added. In the event that the flow decreases below a critical value, the reactor is scrammed by the "flow-level trip."

3. A hot-leg temperature trip is added, set 10°F above the normal operating range, to counter possible operating error under manual control.

4. An entirely independent boron-solution safety system is added in case a number of rods do not function (e.g., tilting of the pressure vessel head in a boiler explosion). This has one automatic trip (hot-leg temperature at 490°F) and manual trips.

5. A pressure interlock is provided to prevent reactor operation at over- or underpressures.

Experience with the Submarine Thermal Reactor and Materials Testing Reactor fast servo systems showed them to be somewhat temperamental, and the present tendency is toward simple servos or manual controls. We provided for either. More extensive studies of the dynamics of the reactor-control system and the power circuit were planned before a final choice of the control parameters (degree of proportional and derivative control, etc.) was to be made.

Table 2—Hot-channel Temperatures in Reactor Core as a Function of Total Flow Rate, Heat-flux Distribution, and Heat-transfer Coefficient*

Flow rate, gal/min	4000				6000			
Max./av. heat flux	4		2.5		4		2.5	
Safety factor on h†	1	0.8	1	0.8	1	0.8	1	0.8
Hotspot temperature, °F	576.3	601.1	522.0	537.3	539.8	558.0	501.5	512.8
°F below boiling point at 1200 psia	−9.1	−33.9	45.2	29.9	27.4	9.2	65.7	54.4
°F above inlet temperature	144.8	169.6	90.5	105.8	102.2	120.4	63.9	75.2
Power increase to initiate boiling, % design power	−6.29	−20	50	28.2	26.8	7.64	102.9	72.3

*Design power level of reactor, 10 megawatts.
Average heat flux, 62,000 Btu/hr/sq ft with all control rods inserted.
Coolant outlet temperature, 450°F, average.
Coolant flow area in core, 2.14 sq ft.
Coolant flow distribution in core, uniform.
Axial heat-flux distribution, cosine, 1.31 max./av.
†Heat-transfer coefficient calculated from the equation

$$\frac{hD_e}{k} = 0.023\ (Re)^{0.8}\ (Pr)^{1/3}\ \text{(safety factor)}$$

Boiling point at 1200 psia, 567.2°F.

2.3 Control-rod Drive Mechanism

Either of two alternate control-rod drive mechanisms, in association with proper design of the control and safety circuits, would be satisfactory. One is the Westinghouse Mark II type canned drive, and the other is the rack-and-pinion design described in ORNL-1613. Our preliminary specification would have to be the Westinghouse drive because it is presently developed, but we would change to the Oak Ridge type if developments during the next few months show that it could be perfected in time. A main advantage of the Oak Ridge type of drive is the low power required to operate it.

We planned to develop and fabricate the ORNL control-rod drive ourselves in our Belleville plant. We would probably add, to the specifications in ORNL-1613, a slow speed, approximately 2 in./min, on one of the outer rods for smoothness of regulation.

2.4 Reactor Vessel and Shielding

The reactor vessel is essentially that described in ORNL-1613. Three feet of water shielding has been added around the vessel to reduce the heat generated in the ordinary concrete shielding to tolerable values. The insulation on the vessel is canned.

Detailed calculations of the stresses in the reactor vessel and of the proper distribution of iron and water in the thermal shield have not been made. Our preliminary investigations indicate that some revisions may be needed in the final design.

During operation the water shield surface is level with the top flange of the reactor vessel. Almost all the radiation that leaks from the vessel is captured in the water shield. Only the 850 Btu/hr that leaks through the air cavity above the vessel, where the control-rod drives are located, is to be absorbed in concrete.

The concrete reactor pit is lined with a type 304 stainless-steel membrane to prevent the demineralized shield water from being contaminated. In an emergency post water may be supplied to the reactor pit.

3. PRIMARY CIRCULATING SYSTEM

The primary system is indicated by the heavy lines on Fig. 3. The main components are the Reactor (V-1), the boiler (E-1), the canned rotor circulating pump (P-1) and spare pump (P-2), the pressurizer and surge drum (V-3), and the closed-loop purge circuit consisting of a regenerator (E-11), subcooler (E-12), and demineralizer (V-10). The primary circuit is located within a gastight enclosure (heavy dashed line).

Table 3 — Primary System

Reactor outlet temperature, °F	450
Reactor inlet temperature, °F	437.6
Operating pressure, psia	1250
Cooling-water flow rate, gal/min	6000
Purge-loop flow rate, gal/min	20
Water volume in system, gal	2000
Hydrogen concentration in water, cm^3 at S.T.P. per kg of H_2O	100
Pressure losses in circuit, ft of primary water	
Reactor	15
Piping	35
Boiler	12
Total	62
Heat losses, Btu/hr	
Reactor vessel	
Neutron energy	130,000
Gamma energy	15,000
Conduction	10,000
Piping	15,000
Pressurizer	4,000
Boiler	20,000
Subtotal	194,000
Purge circuit	512,000
Total heat loss	706,000
Average corrosion rate	
$Mg/cm^2/month$	0.1
G/day	8.4
Concentration of solids in water, ppm	0.078
Rate of increase in solids concentration with no purge, ppm/day	1.3

The circulating rate is 6000 gal/min. Water enters the reactor at 437.6°F, flows down past the thermal shield, turns and flows up through the core, leaving the reactor at an average temperature of 450°F. The operating pressure is 1250 psia. Water flows from the reactor to the boiler and is then pumped back to the reactor. A small side stream is taken off the pump discharge, flows through the purge circuit, and is returned to the pump inlet. A bleed line between the pressurizer and the pump suction keeps the water in the pressurizer leg at loop temperature.

Table 3 summarizes the main features of the primary system. The water inventory is larger

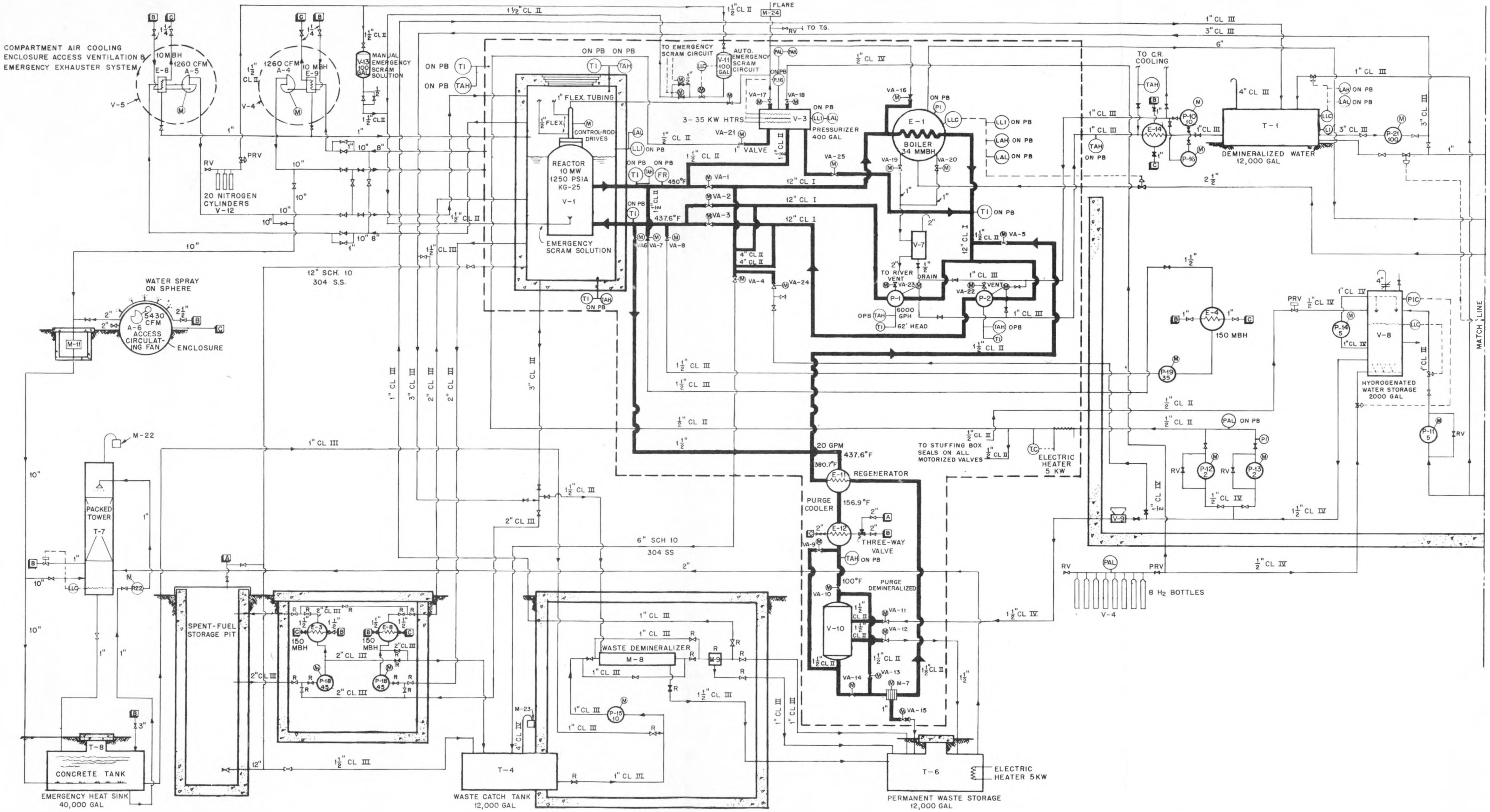


Fig. 3 — Flow sheet, primary system.

than that quoted in ORNL-1613 because of three factors: (1) it includes the volume in the pressurizer, (2) the flow rate is higher, which tends toward larger equipment, and (3) the steam pressure is higher, which results in a larger boiler. Hydrogen is added to the primary-system water to minimize corrosion. The purge circuit maintains the concentration of solids in the water at about 0.08 ppm. With the purge circuit shut off the solids concentration would build up at a rate of 1.3 ppm/day. The pressure loss is 62 ft of primary water. Heat losses total about 700,000 Btu/hr, most of which is sustained in the purge circuit.

The main piping contains three remotely operated block valves (VA-1, VA-2, and VA-3) to isolate the reactor from the rest of the system. Either VA-2 or VA-3 would normally be closed since one pump is a spare. There are no check valves in the system.

3.1 Primary-system Auxiliaries

(a) Afterheat. The system is designed so that fission afterheat may be removed by natural circulation of primary water. However, an afterheat cooler (E-4) and pump (P-19) loop are provided so that maintenance may be performed on the boiler side of the main block valves without unloading the reactor.

(b) Pump and Control-rod Drive Cooling. The main circulating pumps and the control-rod drives have canned motors, which must be cooled to protect the windings and ensure long life of the equipment. Demineralized water is circulated through cooling passages in these equipments by a pump (P-10), and the heat removed is rejected to the service water through a cooler (E-14). The pump and cooler operate at low pressure and are located outside the enclosure.

(c) Seals on Motorized Valves. All valves on the primary system are motorized through stuffing-box seals. A small inflow of heated hydrogenated water is maintained through the seal to prevent leakage of primary water.

(d) Hydrogenated Water. The primary system is designed for no out-leakage. The average make-up requirements are low since in-leakage through the seals on the motorized valves is normally the major load. On start-up, or during maintenance, however, a large supply of make-up water is needed. One full-system volume is provided in the hydrogenated-water storage tank (V-8), which also serves as the hydrogenator. The primary system is filled by the pressure in the hydrogenated-water storage tank, and hydrogenated water is used to flush resin to and from the purge demineralizer. Thus resin may be changed without shutting down the reactor and without access to the enclosure.

(e) Demineralized Water. Post water is demineralized in the main feed-water demineralizer, which supplies make-up water to the plant. This 7 gal/min unit supplies water with 0.5 ppm total solids and a resistivity of 8×10^6 ohm-cm. A storage capacity of 12,000 gal is provided (T-1 on Fig. 3).

(f) Shield Cooling. Heat is removed from the shield water through a cooler (E-8) in a closed loop. The cooler and a pump are located in a shielded pit outside the enclosure.

The heat loss to the air cavity above the reactor vessel is removed by circulating the air through a cooler (E-9) with a fan (A-4).

The heat generated in the concrete shielding on the boiler side of the block valves is negligible.

(g) Emergency Scram Solution. Concentrated boron solution is provided for emergency scram in case of failure of the control rods. An automatic system is located within the enclosure close to the reactor but outside the biological shield, and a manual system is located outside the enclosure. Each system stores 100 gal of concentrated boron solution under 2400-psi nitrogen cylinder pressure. Each system has a high-flow valve to be used when the primary system pump is operating and a low-flow valve to be used when the pump is stopped.

The manual system is provided as a last-ditch measure. It could prove very useful if rupture of the primary circuit resulted in damage to the control rods. Since it is located outside the enclosure, it is easily maintained and may be refilled with poison in a short time.

(h) Radioactive Waste. The waste treatment system consists of a catch tank (T-4), a demineralizer (M-8) and filter (M-9), and a permanent storage tank (T-6). All systems that contain, or may contain, radioactive waste are connected to the waste treatment system.

The catch-tank capacity is sufficient for six complete flushings of the primary system. The waste demineralizer is capable of treating a full catch tank in 20 hr. The permanent storage tank is equipped with an electric heater, a vent condenser, and a filter.

(i) Vents. The vents on the primary system are manifolded to the permanent waste storage tank during operation. While the system is being filled, the hydrogen displaced from the system is vented through a filter and a flare.

(j) Shakedown Heating. Reactor heat will bring the system up to operating temperature during normal operation of the primary system. However, during the initial shakedown, it will be necessary to heat the system to operating temperature for a check run. Since the canned motor pumps are subject to limited service at low temperatures, temporary electric heaters will be installed on the primary system.

(k) Decontamination. It is expected that there will be little need for an integrated decontamination system for the plant. Nevertheless, the equipment and enclosures will be designed with due consideration of ease of decontamination. A portable rig with an adequate supply of detergents and cleaning acids will be kept on hand for this service.

(l) Instrumentation. In addition to the nuclear-control instrumentation described in Sec. 2.2 and the process instrumentation indicated on the flow sheets, Figs. 3 and 4, certain other nuclear instrumentation, which is considered as standard equipment, will be provided. The main items are as follows: (1) fission-product monitor, (2) boiler leak detector, (3) seal water monitors, (4) through-enclosure line monitors, (5) vent monitors, (6) airborne-activity detectors, and (7) portable survey instruments.

4. SHIELDING

The main biological shield is made of ordinary concrete. Supplemental shielding is provided by water around the reactor vessel and by earth around the partly buried enclosure.

The concrete shield forms two compartments within the enclosure: a heavily shielded compartment for the reactor and a lightly shielded compartment for the remainder of the primary circuit. The partition between compartments is approximately 5.9 ft thick. The side walls of the reactor compartment taper from 5.9 to 4.9 ft, the thickness of the back wall. The slabs over the reactor compartment are 7.9 ft thick. The other compartment is shielded by walls and removable slabs which are 2 ft thick.

Air vents interconnect the shielded compartments and the expansion volume inside the enclosure. The air cooling system maintains the air temperature below 150°F.

The radiation level outside the spherical enclosure during reactor operation will be less than 300 mrep. Since access to the enclosure is barred while the reactor is in operation, the radiation level directly over the slabs can and will be higher. When the reactor is down, the level over the slabs will be considerably below tolerance.

5. CONTAINMENT

There are two main types of accident which may endanger the reactor structure: (1) a power excursion on a short reactor period and (2) mechanical or human failure, not necessarily accompanied by a power excursion.

The maximum accumulation of energy in the system will not occur in a short burst of power (the fuel elements will burn out first) but by a gradual increase in temperature of the primary circuit, not interrupted by boiling. This type of accumulation is favored, not by power excursions and hot channels, but by uniform low heat fluxes. The maximum credible accident is the extreme of this type.

We conclude that the APPR is a safe reactor, but the possibility of fuel-element burn-out and pressure-vessel ruptures cannot be excluded. The reactor safety systems are devised to make these possibilities remote except for sabotage. The containment system is devised to protect the public in any event.

The entire primary system, including shielding, is enclosed in a steel sphere, 54 ft in diameter, capable of withstanding 36 psig internal pressure and providing 70,000 cu ft expansion volume. Effluent from this sphere, in case of an accident, will go to a blow-down drum, scrubbing tower, and filter.

The maximum credible accident that this sphere is designed to contain consists in the following:

1. Primary circulating system is operating at design pressure and temperature.
2. Operator has placed system on manual control (possibly to correct for xenon burn-out after temporary partial-load operation).
3. Temperature in primary system rises (slowly) but is not corrected by operator until saturation temperature is reached.
4. Automatic temperature trips fail.
5. After saturation temperature is reached, pressure rises until frangible disks are shattered and primary system flashes to ambient pressure.
6. Reaction stops; manual boron scram prevents new start-up if control rods are damaged.

An alternate route to this accident could be systematic false indications from the temperature sensing instruments with the reactor on either servo or manual control. The operator would have to ignore pressure-rise indications from the secondary circuit.

The result of this accident will be to liberate mildly radioactive steam throughout the containing sphere. Fission products, from a possibly simultaneous fuel-element failure, would remain in the heavily shielded region of the reactor and would present no radiation hazard outside. Table 4 gives the equilibrium flash conditions within the enclosure as a function of the enclosure volume. After such an accident the pressure build-up rate is slow enough (less than 12 psi/hr) to allow ample time for bleeding the enclosure.

Table 4—Equilibrium Flash Conditions in APPR Enclosure as a Function of Enclosure Volume*

Volume for expansion, cu ft	Pressure, psia	Temp., °F	Water vaporized, lb-moles	Mole fraction of water vapor in gas	Fraction of water flashed
22,250	119.5	328	264.9	0.841	0.343
44,500	70.4	284	293.1	0.745	0.3795
66,750	52.9	260.5	309	0.674	0.400
89,000	43.9	244.2	316.7	0.613	0.410

*Primary system initially contains 13,900 lb of liquid water saturated at 1500 psia (2000 gal at 52 lb/cu ft).

Enclosure initially filled with air at 14.7 psia and 150°F.

No heat is lost from system during expansion.

Constant volume specific heats are: air, 51.7 Btu/(lb-mole)(°F); steam 6.5 Btu/(lb-mole)(°F).

The absorption system consists of a filter (M-11), a large reservoir of water (T-8), a packed tower (T-7), and another filter (M-22). The reservoir stores enough water to absorb all the heat in the gases within the enclosure with less than a 50°F rise in temperature. The system is bled at a rate of about 1000 cu ft/min (atmospheric). When the pressure in the enclosure gets too low to drive the gases through the absorption system, the reactor compartment air circulating fan (A-4) is pressed into service as an emergency exhauster.

The enclosure is uninsulated and is coated on the outside with aluminum paint. If access is required on hot summer days, the sphere may be cooled by spraying service water over the outside surface at a rate of 70 gal/min. A 5400 cu ft/min circulating fan (A-6) located within the enclosure provides sufficient air movement for comfort, and the reactor-compartment air circulating fan (A-4) is used to pull fresh air through the enclosure.

6. STEAM SYSTEM

6.1 The Cycle

A simple low-pressure regenerative feedwater heating cycle with steam at 240 psia and no more than 0.3 per cent moisture is used. A pressure-regulating valve maintains turbine-throttle pressure at 220 to 235 psia. Power is delivered at a generation voltage of 4160 volts to a bus, which is connected through overhead lines, breakers, disconnects, and transformer to the 22-kv substation. Turbine exhaust is condensed in a two-pass condenser using cooling water from Gunston Cove. Condensate is heated to 290°F in three extraction feed-water heaters before returning to the heat exchanger. Figure 4 is a flowsheet for the steam plant.

6.2 Heat and Power Balance

Figure 2 shows the heat and power balance for the plant. The net electrical output is 1797 kw with 85°F cooling water. The auxiliary load is 364 kw, including a contingency of 98 kw. Table 5 shows the effect of cooling-water temperature on gross generation.

Table 5—Gross Generation as a Function of Cooling-water Temperature

Condenser cooling-water inlet temperature, °F	Gross generation, kw
85	2161
65	2290
40	2433

6.3 Heat Exchanger

The primary side of the heat exchanger is designed for 1500 psi and 475°F, and the shell side is designed for 450 psi. Pressure drop in the tube side is 3 psi at 6000 gal/min. At full load 35,735 lb/hr of saturated steam at 240 psia will be generated. The heat exchanger, which is 17 ft long by 5 ft diameter, has approximately three hundred and eighty 1- by 0.055-in.-wall type 304L stainless-steel tubes 11 ft long. All other surfaces in contact with the primary coolant are type 304 stainless steel. All secondary side surfaces are carbon steel.

6.4 Pressure-regulating Valve

An automatic pressure-regulating valve located in the main steam line maintains steam pressure at the turbine throttle. This pressure would otherwise tend to rise at part load because of the reduced temperature difference in the heat exchanger. A regulating valve appears to be the best solution, although it represents a full load pressure loss of about 5 psi. Other schemes considered were: (1) to use a heat-exchanger bypass in the primary-coolant circuit and (2) to design the front end of the turbine for the highest pressure expected, about 425 psia.

6.5 Turbine Generator

The turbine generator is designed for steam at 200 psia (good for 230 psia) to exhaust at 2.0 in. Hg absolute. Standard utility type equipment is provided on the turbine, including the following: stop throttle valve, speed governor (4 per cent regulation overload range), multiple control valves, separate overspeed governor, remotely operated speed changer, oil pump and oil reservoir and cooler, steam-turbine-driven auxiliary oil pump, pressure gauges, above-seat drain, extraction openings, and steam seal exhauster and condenser.

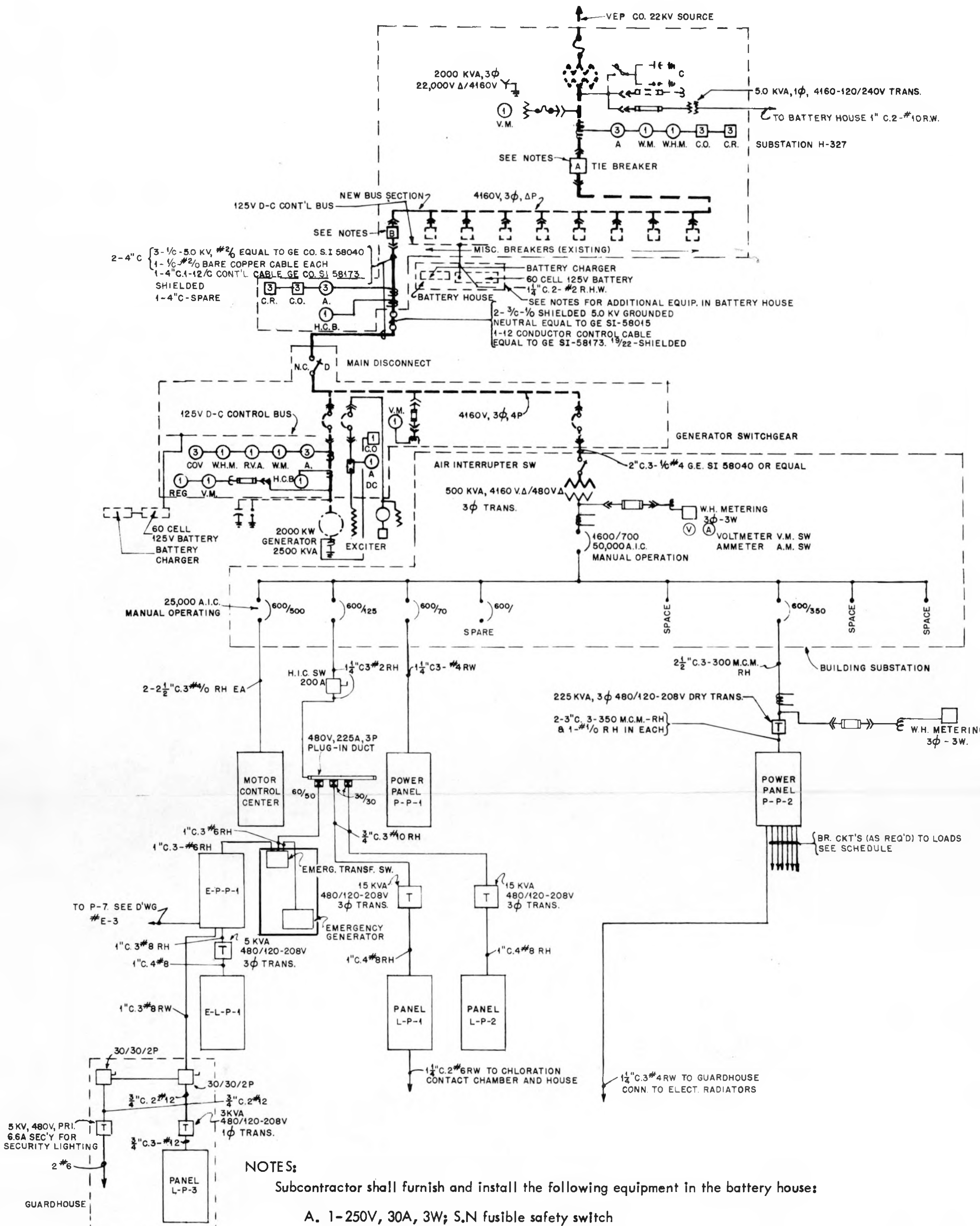
The turbine is of rugged design of suitable materials with special attention being paid to the problem of the relatively high moisture content of the steam in the latter stages of the turbine.

The generator is rated 2000 kw, 2500 kva, 3600 rpm, and 4160 volts totally enclosed and self-ventilated, with a 25-kva 125-volt direct-connected self-ventilated self-excited exciter. Accessories include: ventilating fans and closed ventilating system, air coolers for operation 85 per cent clean with maximum cooling-water temperature of 95°F, field-discharge resistor, stator winding temperature detectors, and hand-operated field rheostat.

The turbine generator is mounted so as to isolate the building from vibration. The turbine weighs 12 tons, and the generator and exciter weigh 11 tons. The over-all length is 22 ft. An overhead crane is provided to erect and maintain the machine.

6.6 Main Condenser

The main condenser is of two-pass design with 3250 sq ft of surface on the ¾-in. by 14.0-ft Admiralty metal tubes. The shell is 54 in. in diameter and 17 ft long. Water boxes are cast iron, and the shell is welded steel. An atmospheric relief valve is provided. Cooling-water velocity in the tubes is 7 ft/sec. With



NOTES:

Subcontractor shall furnish and install the following equipment in the battery house:

A. 1-250V, 30A, 3W; S.N fusible safety switch
B. 1-4 CKT. 3-2W; S.N: Panel with 1-30/20 2P. 250V breaker (for elect. heater) and 3-20/20, 1P 125V breakers (1 for light and recept; 1 for rectifier and 1- spare)
C. 1-3KW 240V, 1φ, 2W space heater with controls (see specs.)
D. 1-150W type "A" fixture and 2- receptacles
E. 1-20A, 125V SW (for light)
F. 1-60A, 2P, 230V, d-c fu. disk: SW (for outgoing d-c battery cable)

Fig. 5—Wiring diagram.

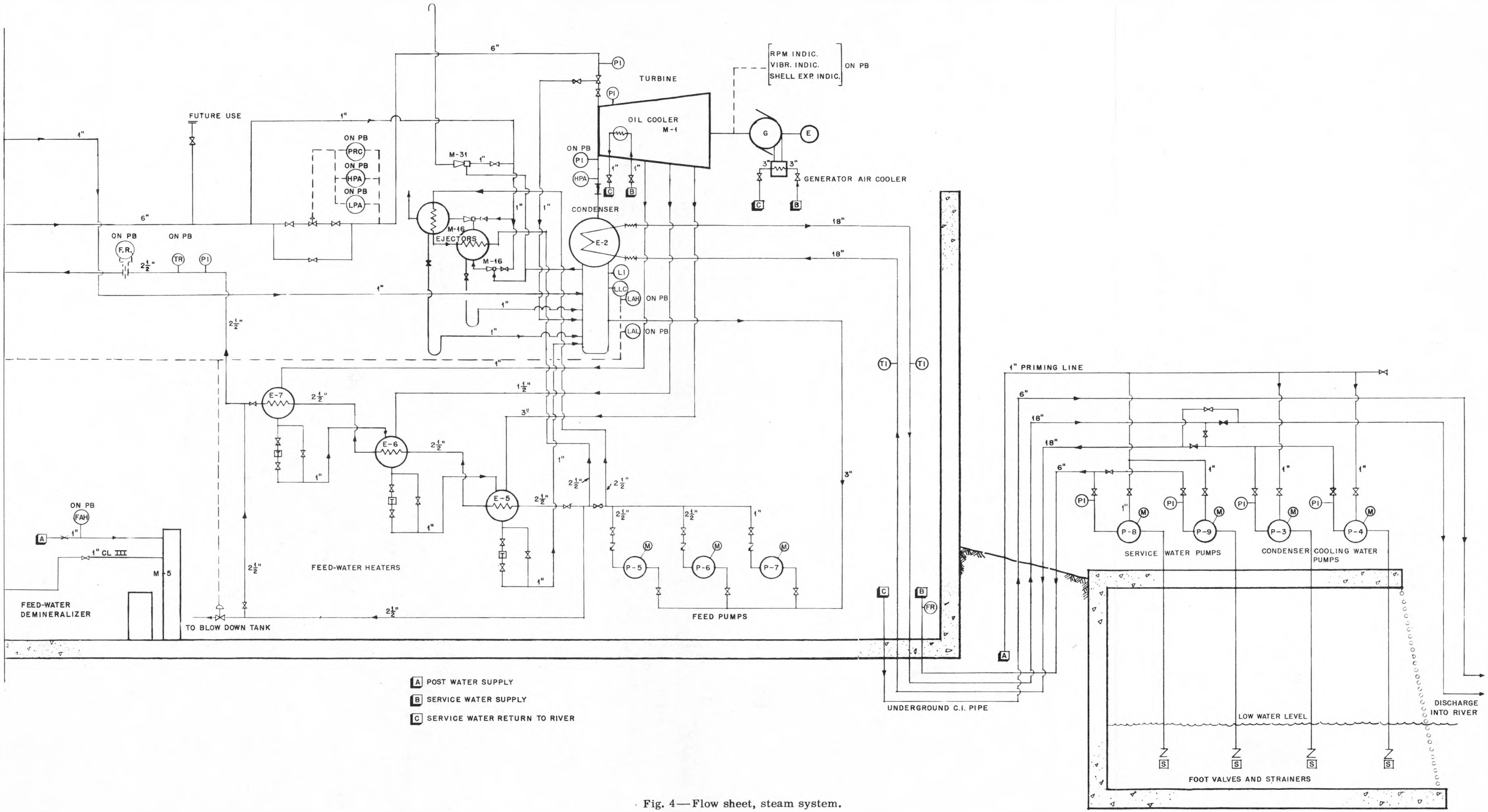


Fig. 4—Flow sheet, steam system.

85°F cooling water and 85 per cent cleanliness factor, a vacuum of 27.7 in. Hg is expected. A 500-gal deaerating hot well removes oxygen from the condensate so that the concentration is no higher than 0.01 cm^3/liter and provides surge capacity for the secondary system.

Cooling water is taken from Gunston Cove through a channel which is dredged to assure a supply of water under all conditions. Two 12- by 12-in. 4300 gal/min centrifugal pumps, one of which is a spare, driven by totally enclosed motors, draw water from this channel through strainers and foot valves and deliver it into a cast-iron pipe leading to the condenser. These pumps and motors are outdoors at the edge of the river. They must be primed before starting and are arranged so that the suction pipes can be removed. Cooling water is discharged at a location removed from the intake and below the lowest free level of Gunston Cove. A water-box ejector is supplied to create the syphon during start up.

A steam-jet air pump and hogging ejector is provided to remove noncondensible gases from the condenser. The former is a twin-element (either of which can carry the load) two-stage ejector with inter- and after-condensers. Motive steam is provided directly from the main steam header at 240 psia.

6.7 Combination Hot-well Boiler-feed Pump

The usual hot-well and boiler-feed pumps are combined into one pump in this design to simplify operation and control. Two of these pumps are provided, rated at 75 gal/min at 1000-ft head. At all loads the pumps deliver water at about 450 psia. At full load this pressure has to be reduced to the 240 psia in the heat exchanger, which is done by the feed-water regulating valve. At very light loads the pressure in the heat exchanger rises, but a 25-psi pressure drop is still available to regulate water flow, which in turn controls the level in the heat exchanger.

A separate battery-powered d-c driven pump with a 2 gal/min capacity at 60-ft head is provided to supply water to the heat exchanger in the event of loss of auxiliary power.

6.8 Feed-water Heaters

Studies indicated that if a single regenerative feed-water heater cycle were used as a base, the following increases in generation at constant heat input would result by the addition of heaters:

No. of heaters	Gain in generation, %
1	Base
2	2.2
3	3.0
4	3.4

It was judged that the use of three closed low-pressure heaters with drain traps would be justified economically and would not reduce the reliability or simplicity of the cycle.

6.9 Controls

The control of the secondary cycle is extremely simple and is accomplished by four main elements: (1) load on the unit is controlled by the operator manually positioning the turbine speed changer to give the load output desired, (2) the throttle pressure is automatically controlled by the pressure-regulating valve, (3) water level in the heat exchanger is automatically regulated by a two-element control, which measures feed-water flow and drum level and acts to position the feed-water regulating valve, and (4) the hot-well level is regulated by make-up and dump valves.

Level is maintained in the feed-water heaters by steam traps. If these fail, bypasses are provided for manual operation. Service water and other secondary functions are attended by the operator.

Suitable automatic devices and interlocks are provided to protect the equipment during transients. In so far as possible annunciators are provided to give operators an indication of impending difficulties, and necessary corrections can be made without disturbing the operation of the unit.

7. ELECTRICAL SYSTEMS

Figure 5 is a wiring diagram for the electrical systems. An aerial tie line connects the APPR generator to the H-327 Substation 4160-volt bus. Circuit breakers are installed between the generator and the APPR bus and at the point where the tie line connects to the H-327 bus. A disconnect switch is provided where the tie line connects to the generator bus. A third circuit breaker is installed between the 2000-kva transformer and the 4160-volt bus in Substation H-327. The neutral of the APPR system is permanently grounded. Pilot wires, suspended on the same poles as the aerial tie line, protect the system and permit the operator to remotely control and synchronize the system independent of, or in parallel with, the Virginia Electric Power Company.

Energy for the APPR plant is provided by a 4160/480-volt 500-kva unit substation with low-voltage secondary circuit breakers. The lighting, heating, ventilating, and air-conditioning systems are served by 480/120/208-volt 3-ϕ dry type transformers. A series type security lighting system, energized by a constant-current transformer and controlled from the guard house, is provided. American District Telegraph (ADT) signal and telephone services are included. An emergency generator rated at 25 kw, 480 volts, and 3 ϕ is provided. Batteries supply power to the electrical controls.

8. BUILDINGS AND AUXILIARY EQUIPMENT

In determining the disposition of the various components, it was felt that the plant could best be served, from the standpoint of economy and efficient operation, by placing all services in one structure. This objective has been accomplished except for minor items such as the spent-fuel storage pit and guardhouse.

The main structure, designated as "auxiliary building," contains the control room, turbine area, maintenance shops, and, in a separate wing, the offices and classroom. The office wing has continuous fixed sash above a masonry wall for maximum light penetration. A continuous overhang has been provided to minimize glare from the sun. Above the sash a fiberboard panel conceals roof construction and ventilation ducts. For the turbine area, or high-bay portion, insulated corrugated siding was felt to be more suitable. To obtain outside light on the sloping west wall of the turbine area and eliminate glare, a sawtooth wall was designed to deflect the light into the building.

Existing contours were taken into consideration in the orientation of the auxiliary building and reactor sphere. Necessary roads and outside services were placed as close to the main structure as was deemed practical for the purpose of security control.

The following is an outline of the building services to be provided with this power plant:

1. Potable water.
2. Fire protection.
3. Hot and cold domestic water.
4. Sanitary waste treatment.
5. Building plumbing, including fixtures and piping.
6. Electric heating system.
7. Exhaust and supply ventilation systems where required.
8. Air-conditioning systems where required.
9. Lighting and emergency lighting.
10. Telephones.
11. ADT system coordinated with the post alarm system.
12. Fences.
13. Roads and parking area.
14. Grading and seeding.
15. Guardhouse.
16. A river-water intake and pump platform.

9. REACTOR LOADING PROCEDURE AND EQUIPMENT

When the reactor is to be unloaded and recharged, the reactor is shut down, and the primary system is gradually cooled, first through the main exchanger and then through the afterheat cooler. If necessary the primary system may be drained and flushed. When the system is cool, the water level in the reactor pit is raised 12 ft, the enclosure is checked for activity and is opened, and its circulating and ventilating fans are turned on. The loading crew moves in, lifts the slabs off the reactor pit with the 360-deg rotating jib crane, and swings them out of the way. The control-rod drives are disengaged, and the vessel cover, on which the drives are mounted, is unbolted with impact wrenches, is cautiously lifted clear of its guide dowels, and is placed aside completely submerged in the temporary water shield (the control-rod service connections are flexible and waterproof). Spent fuel is removed from the reactor with tongs and is transferred underwater to the spent-fuel storage pit, which is located outside the enclosure, through a lock that consists of a 12-in. pipe and two 12-in. valves. The vessel, core-support structure, and control rods are inspected with underwater telescopes, and then fresh fuel assemblies are placed in the core. The vessel cover is replaced and the control-rod drives checked for proper engagement. The slabs are replaced and, the enclosure is cleared, closed, and tested for tightness. The water level in the reactor compartment is lowered to the top flange of the vessel and the system is ready for start-up.

10. SUMMARY

The proposed design is similar to that described in ORNL-1613. Modifications that were made include a spherical enclosure around the primary system for containment in case of failure, an increase in the primary-system flow rate and the addition of a purge demineralizer, the addition of a boron-solution scram system, water shielding around the reactor, and other changes to adapt the design to the Fort Belvoir site, such as the use of cooling water and a waste treatment system.

The primary-system flow rate was increased to 6000 gal/min to provide some safety on core heat transfer. Canned rotor-rod drives are specified instead of the rack-and-pinion type, but we planned to develop and fabricate the ORNL drive ourselves to take advantage of its lower power consumption. Water was added around the reactor vessel to protect the concrete shield and facilitate cooling. This requires canned insulation on the vessel. A stainless-steel liner was added to the reactor pit to prevent contamination of the water. The thickness of the concrete shield varies from between 5 to 8 ft around the reactor to 2 ft around the remainder of the primary system. Supplemental shielding is provided by earth around the partly buried enclosure.

The reactor operates at 10 megawatts and delivers water to the boiler at 450°F and 1250 psia. The steam pressure is 240 psia. A net generation efficiency of 18 per cent is obtained with the simple regenerative cycle using three stages of feed-water heating with cooling water at 85°F. The major heat loss in the plant is sustained in the purge circuit of the primary system. The process auxiliaries consist mostly of storage tanks and pumps for demineralized water, contaminated waste water, hydrogenated water, and cooling water.

The maximum credible accident, which occurs as a result of a relatively slow accumulation of energy in the water of the primary system, is not much worse than rupturing the primary circuit under normal conditions. Containment is provided by a spherical steel vessel with a net volume of 70,000 cu ft which is designed according to code to hold a pressure of 36 psig. In case of accident the enclosure may be bled to an absorption system.

The building was designed to take advantage of existing contours and to be an efficient, pleasant place to work. Reactor loading procedures were conceived to avoid the use of complicated remote-handling devices and to minimize exposure to radiation.

REFERENCE

1. A. L. Boch et al., A Conceptual Design of a Pressurized-water Package Power Reactor, Report ORNL-1613, July 8, 1954.

ABOUT THE AUTHORS

James J. Barker graduated from New York University (NYU) in 1943 with the B.Ch.E. degree and worked in the Process Department of The Kellex Corporation on the design of K-25. He spent some time at Oak Ridge during the start-up of the gaseous-diffusion plant and then was drafted for military duty. In 1949 he joined the Atomic Energy Division of the H. K. Ferguson Co. He obtained the M.Ch.E. degree from NYU in 1950 and is currently working on his doctorate. Barker has been with Walter Kidde Nuclear Laboratories, Inc., since it was formed in 1952.

John Faas graduated from Purdue University in 1940 with the B.S.M.E. degree and joined the H. K. Ferguson Co., where he became a specialist in the design and construction of chemical-processing plants. Mr. Faas joined Walter Kidde Constructors, Inc., in 1954. In his present position he has the complete responsibility for all detailed mechanical design carried out by that company.

William L. Webb received the B.S.E.E. degree from Purdue University in 1921 and started his professional career with the General Electric Company at Schenectady. From 1923 to 1926 he worked as a relay engineer for the Appalachian Power Co. He has been with American Gas and Electric Co. since 1926. In 1943 he was appointed head of the Chemical Section of the Mechanical Engineering Division with responsibility for the supervision of all chemical services to American Gas and Electric properties, including design, erection, and operating problems. He was appointed to his present position in 1951 and attended the Oak Ridge School of Reactor Technology during 1952 and 1953. Webb is a member of ASME, ASTM, and EEI.

ARMY PACKAGE POWER REACTOR

The Kuljian Corporation Proposal

HERBERT G. JOHNSON*

February 15, 1955

1. INTRODUCTION

The concept of calling for competitive bids on a lump-sum-basis fixed price is a decided innovation, as regards Atomic Energy Commission (AEC) contracts, and marks a definite milestone in the application of atomic energy to peacetime use.

In light of the many criticisms leveled at industry for its (excessive) caution and reluctance to pioneer, except on a cost-plus basis, the response to the APPR invitation should carry conviction. At least a substantial segment was willing, ready, and anxious to participate on the basis of a fixed price, with penalties, for a completely integrated atomic power plant placed into operation on a close time schedule, even in the face of less than complete engineering, data, and specifications. Such a response would be unusual, even for a conventional thermal plant of the same size; therefore the conclusion is inescapable—private industry is ready to assume full responsibility, provided the risk or gain may be properly evaluated.

It is hoped that before long other groups in the AEC will follow this excellent and effective example (common in industry) of developing detailed and carefully worked out job specifications of their requirements, which will then be submitted to qualified competitive bidding, rather than calling in one of a favored few to negotiate on a nebular outline. Obviously such a method calls for considerably more work by the contracting agency in advance of the bidding, but the payoff in better prices, better designs, and all-round better performance is axiomatic and is, in fact, the foundation for our competitive capitalistic and democratic system.

2. GENERAL CONSIDERATIONS

In the invitation for proposals of Aug. 19, 1954, it was stated that selection would be made based upon (1) responsiveness, (2) lowest adjusted price, and (3) degree of contributions.

Ordinarily in such circumstances we would have assumed that lowest price would be the first consideration. However, such a conclusion did not seem warranted in light of the subsequent selection of Fort Belvoir as the site. We felt that such close proximity to Washington must change the emphasis from low price to maximum security.

Therefore, although our practical experience indicated that low price would govern, our common sense dictated that we include safety features to provide against all conceivable hazards or contingencies, more or less regardless of dollar cost.

*Now with the Johnson Engineering Company, Havertown, Pa.

Therefore we decided to put our primary emphasis in the design of the plant on safety and reliability. Next in importance we considered the practicability for army field use and the contributions to the art. Then we evaluated the economics of power generated and available at the switchboard per dollar of investment. We also considered important the flexibility of the facility for training purposes and the adaptability of the reactor to another type of operation, i.e., the production of boiling water instead of hot pressurized water. In the latter case the turbine could easily be renozzled for the higher steam pressure (e.g., 400 to 450 psig) with a consequential increase of at least 10 per cent in utilization of fuel.

3. REACTOR DESIGN

In several important respects our reactor design differed from that outlined in Report ORNL-1613; however, it had the same capacity under similar operating conditions.

We felt that in the interest of safety it would be advantageous to reduce the amount of primary coolant in the system under pressure; also in the event of the failure of both pumps, the thermosyphon circulation should be able to carry a substantially full load. Also the minimum energy should be expended to induce circulation through the loop.

The design we proposed features a built-in heat-exchanger boiler and practically eliminates the primary loop. The design of the fuel elements was unchanged, but their primary water passage was made straight through with equal distribution to all elements. The heated pressurized water flows out from the upper cylindrical chamber of the reactor and spirals downward through approximately 1200 sq ft of 16-gauge stainless-steel tubes ($^3/_4$ in. O.D.), each with an effective length of 12 ft, in a somewhat spiral vertical arrangement in the annulus chamber between the high-pressure stainless-steel inner reactor shell and outer carbon-steel shell of the heat-exchanger section. Steam is generated in the annulus chamber and flows upward to a steam-moisture separating drum before going to the turbine throttle. The cooled pressurized water drops directly from the lower annular tube header into the suctions of the circulating pumps. In case of the failure of one pump, the other automatically picks up the load. If both pumps fail, the pump circuit acts as a return bend with practically no resistance to thermosyphon flow. Check valves prevent backflow.

The reactor design is almost equally suited to operation as a boiling-water type of reactor, in which case the steam connection is made to the top of the reactor and the return condensate is brought into the lower annular tube chamber. The existing annular steam chamber is then abandoned, and the tubes serve as downcomers of a forced-recirculation steam boiler of which the fuel elements are the boiler tubes. With the high base velocity of water flow through the fuel elements (approximately 4 ft/sec, exclusive of steaming), excellent heat transfer and element cooling are assured, and local hot spots are eliminated by excellent distribution of the tremendous mass of forced-circulation cooling medium (4000 gal/min, cooling water).

Thus by relatively simple valving and piping changes, our pressurized-water reactor could be demonstrated as a boiling-water reactor (with its higher thermal-energy conversion), and the plant need not become obsolete on the development of satisfactory (interchangeable) boiling-water fuel elements for the reactor.

Even when operated under boiling-water conditions at forced recirculation, the design provides the same thermosyphon circulation safeguard against pump failure, causing loss of circulation provided for the pressurized-water operation.

Another feature of the reactor offered included special attention to the reliability of control-rod mechanisms, using the Westinghouse canned drive or, as an alternate, the Ford hydraulic drive, not depending on gravity as in the type described in Report ORNL-1613.

4. PRIMARY COOLANT SYSTEM

As is pointed out in Sec. 3, the primary coolant system is much abbreviated in our design. We still employ 2000 to 4000 gal/min circulating pressurized-water pumps with canned motors (one as 100 per cent spare), each equipped with a 12-in. free-swinging check valve to prevent backflow when a pump is not operating. However, owing to the elimination of all loop piping, except a short 12-in.-diameter fabricated T between the pump's discharge and riser connection to the bottom of fuel elements, the friction loss and the horsepower is reduced to one-half that required in the arrangement given in Report ORNL-1613.

The heat exchanger is not only simplified but is a part of the reactor housing, with a considerable reduction in size and weight.

All parts in contact with pressurized water would be stainless steel, and make-up water would be evaporated, microfiltered, deoxygenated, hydrogen-injected, and demineralized. The pressurizing system would be similar in principle to the arrangement in Report ORNL-1613 and is illustrated with the rest of the system in Fig. 1.

5. PHYSICS

We have made no contribution to the nuclear physics aspects of the ORNL-1613 design, which on preliminary analysis appeared to be entirely sound and conservative from the theoretical viewpoint (according to our consultant physicists).

It was not practical for us to conduct any simulated prototype tests on actual highly enriched fuel elements, or dummy elements fabricated to this design, and, short of actual high burn-up rate experiments over an appreciable time period under service conditions, in our opinion any tests we might run would be indicative rather than conclusive.

The stainless-steel-cladding fabrication technique appears adequate to solve the hot-spot problem, with suitable poisons available for override. The likelihood of vaporizing or rupturing the fuel-element jacket seems rather remote with the great mass of cooling-medium flow and thin thermal barrier, especially with the improved coolant distribution envisioned by us, provided the design is structurally stable. This again will be determined conclusively only by actual experience, although a fair indication would be obtained by the proposed dummy-element tests, which we understand are now being conducted at Oak Ridge.

6. SHIELDING

The proposed shielding appeared entirely adequate, but, as a further safeguard, we proposed burying the entire "hot" portion of the plant. This would not ordinarily be necessary, but, since it only affected the price slightly, we considered the additional safeguard warranted. Furthermore, in the event of an atomic incident, the cleanup and ultimate disposal, if necessary, would be greatly expedited.

The nature of our reactor design increases the effectiveness of its shielding, as well as concentrates the radioactivity to a small volume, which is much easier to shield. This makes it more practical for army field use where burying may not be practical because of permafrost (we encountered water at Fort Belvoir, which increased the cost of excavation and installation).

7. CONTAINMENT

We devoted a great deal of thought to the problem of containment, and allocated a considerable portion of the cost, directly and indirectly, to provide against all contingencies that we could conceive as feasible (Fig. 1).

Here again our reactor design combining the heat exchanger made it practical to enclose the whole "hot" system in a stainless-steel pressure skin, which would have been costly for the ORNL-1613 arrangement.

Furthermore, since the design was so concentrated and contained, we were able to work out condensing, cooling, and decontamination as a regular controlled cycle. This was not with the idea that such incidents would become routine, but so that, by the use of slightly radioactive tracers, the cleanup procedure could be taught as a regular training routine. This would avoid panic catastrophies occurring in case of damage by enemy action or an accident precipitating an atomic incident.

Even in case (highly improbable under any normal operation) that the design pressure of the containment vessel was exceeded, or in case the pressure skin was ruptured while under containment pressure, decontamination was provided.

It is possible that we may have "overcontained" our design, but we still feel it was justified under the circumstances. After years of experience we shall know better what we can safely dispense with, but, in the meantime, "better be too safe" seems to be a wise policy.

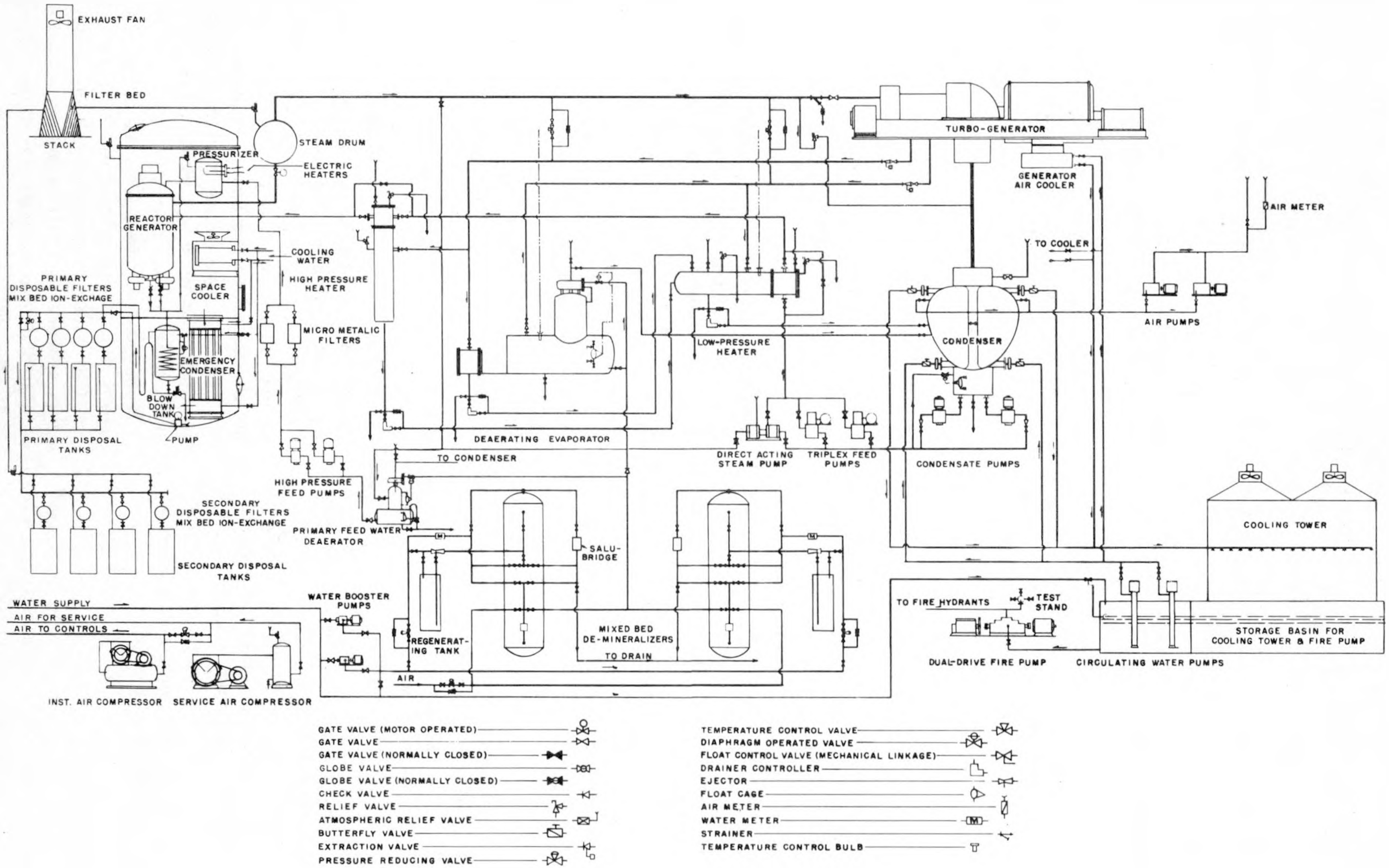


Fig. 1—Single-line flow diagram for proposed APPR power plant.

8. STEAM SYSTEM

At first glance the steam system appears to be entirely conventional (Fig. 2). There is not a great deal of scope for originality in this type of equipment; as a result, in view of the large number of similar existing installations, complete reliability and predictability can be expected.

However, on start-up of a new thermal power plant of the most common variety, troubles do occur and all too frequently. It is actually a bit unusual to process the boiler, turbine, and auxiliaries, test them individually, and then throw the plant on the line without a false start or two.

In an ordinary thermal power plant these troubles with bearings, lubrication systems, vacuum seals, pumps, and the hundred and one things to go wrong are always irritating and sometimes costly, but only rarely are they catastrophic. It is generally true that the worst things that are going to happen to a station ordinarily occur upon start-up, but as a rule nothing in the connected electrical system is damaged or seriously endangered because of the reliability of protective devices.

Unfortunately in an individual atomic-reactor power plant of this type, there is no way to test the steam system as a whole unit under full load (when most serious troubles occur) until the reactor is activated at full power. It is possible to simulate "no load" by feeding back from the connected public utility enough electrical energy to the pressurized-water system to generate a nominal amount of steam under pressure for a "spinning" test, but that is all.

Therefore in the light of the many serious accidents that have occurred even with experienced operators and well-run steam plants (such as the recent Philadelphia Electric and Detroit Edison blowups, which caused millions of dollars in damage and months of outage), we decided that reasonable safety demanded that the steam plant be run for a minimum of several months at full load before activating the reactor and that heavy-load swings be induced to test regulation and stability during the period.

This decision "loaded" our direct costs (for equipment, personnel, and fuel) by almost 15 per cent, but Kuljian top management could not assume the heavy responsibility for the operation (under bond) without this safeguard.

We firmly believe that such a rigorous procedure is indicated because of the location of Fort Belvoir and the world-wide attention likely to be focused upon the results to be obtained. Any serious trouble could set atomic-power generation back by years, especially if it could have been avoided by such a precaution (regardless of cost), and might cause an unfortunate "loss of face" in world politics.

9. ELECTRICAL SYSTEM

The electrical system, as illustrated in Fig. 3, is complete with all normal safeguards and instrumentation for fully automatic control of the station from a central control point.

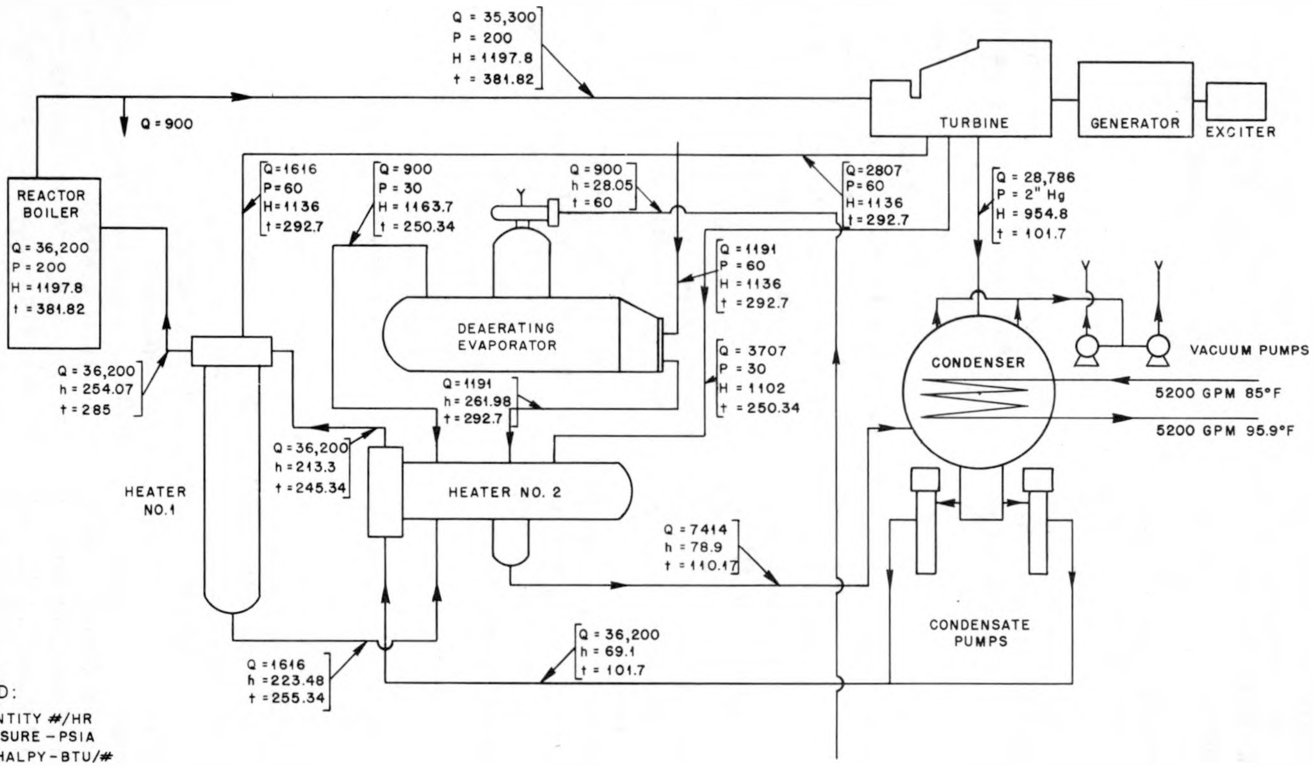
Duplicate 480-volt buses, arranged for energization independently, feed duplicate equipment and drives. Since the entire station is equipped with electrically driven auxiliaries (with but few exceptions, such as, pressurizing feedwater pump), the dependability of electric energy must be assured; however, an electrical generator driven by a small internal-combustion engine is available to drive the auxiliaries in extreme emergencies.

Using the main 2500-kva turbogenerator as our primary electrical source, we provide a backup secondary electric service from the public utility and assure its reliability by the use of two independently routed incoming services. One of these is solely to service one of the auxiliary load-center substations; the other is for the main outgoing service as well.

10. BUILDING AND AUXILIARIES

In general design and specification the building (Fig. 4) follows the plan described in Report ORNL-1613, except it is modified to suit our equipment layout and containment. The building would be unusually airtight in order to maintain a controlled negative pressure in all areas except personnel areas, which would be positive pressured above atmosphere for safety.

In case of escape of radioactive gases or dusts into the building, they would be picked up by the unidirectional ventilating system and deposited in the filters at the base of the fume stack. Owing to the negative pressure in the building, all leakages would be into the building, thus eliminating any possibilities of bypassing



LEGEND:
Q = QUANTITY #/HR
P = PRESSURE – PSIA
H = ENTHALPY – BTU/#
t = TEMPERATURE °F
h = SENSIBLE HEAT – BTU/#

TURBINE HEAT RATE = $\dfrac{35{,}300 \times 1197.8 + 900 \times 28.05 - 36{,}200 \times 254.07}{2190.8} = 15{,}105$ BTU/KWH

Fig. 2 — Heat-balance diagram for proposed APPR power plant.

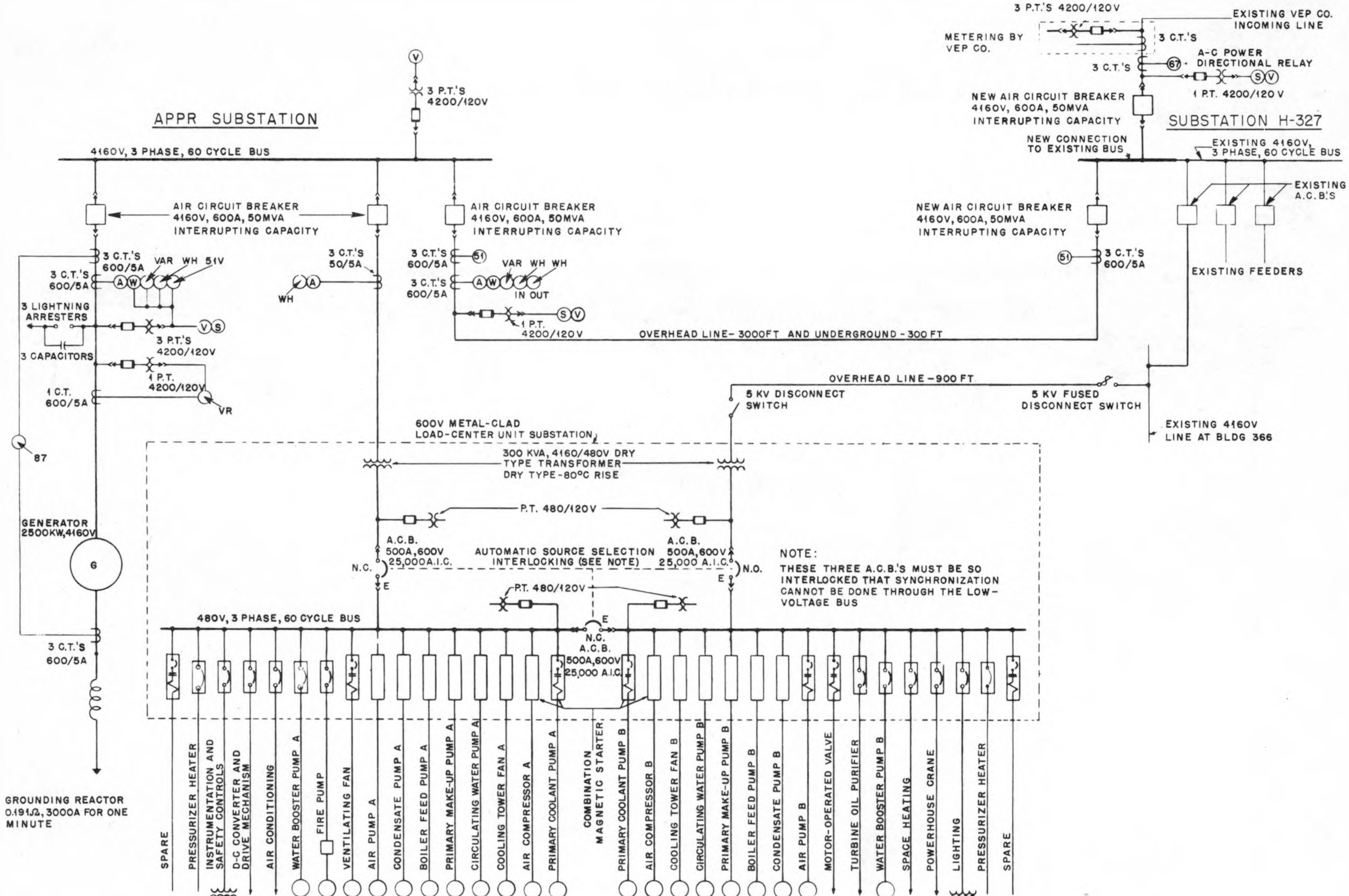


Fig. 3 — Electrical single-line diagram.

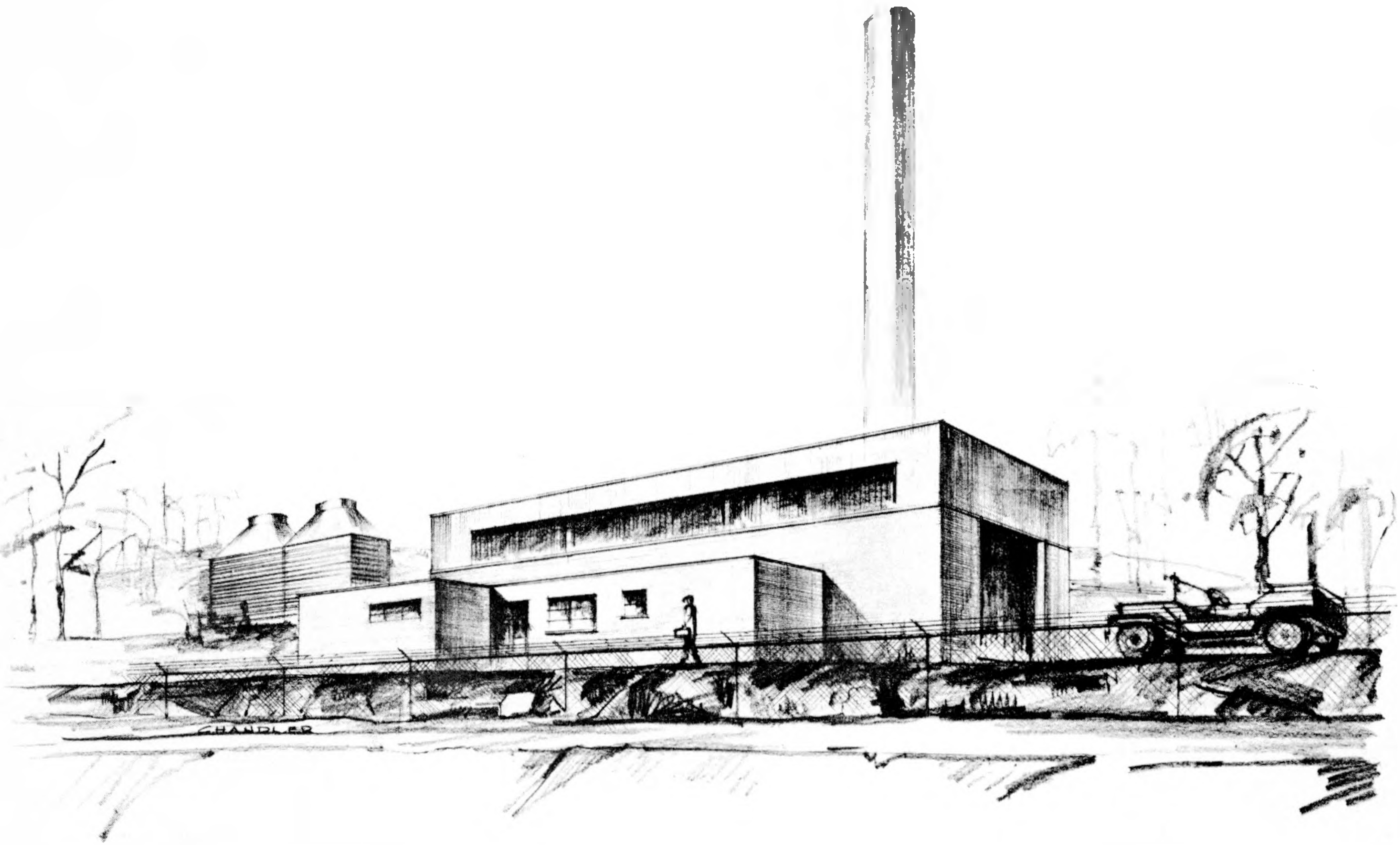


Fig. 4 — Perspective sketch of proposed APPR power plant.

Table 1 — Construction Cost Comparison (APPR Vs. Conventional)

Project item	Estimated installed price APPR (ORNL-1613)	2500-kw conventional (steam-electric)
Site improvements: fencing, clearing, roads, sewage, and drainage	$ 96,550.00	$ 80,000.00
Building: concrete, foundations, excavations, structures, and crane	277,700.00	175,000.00
Primary system (including fuel elements): reactor, containment, disposal, piping, wiring, steam boiler, and coal-handling facilities	817,280.00	150,000.00
Nuclear control and monitoring	197,230.00	
Secondary system (steam-electric): turbogenerator and auxiliaries, piping, wiring, and cooling water	540,650.00	500,000.00
Tests: hydrostatic, thermal, full-load steam, and 700-hr	576,790.00	50,000.00
Construction: erectors, equipment and rentals, and field personnel	247,750.00	120,000.00
Engineering: design, consultants, drawings, and instruction books	144,000.00	100,000.00
Contingencies, overhead, and profit	650,986.00	100,000.00
Performance bond	35,000.00	15,000.00
Total bid price	$3,583,936.00	$1,290,000.00
Price per kilowatt of maximum capability	$1,630.00	$515.00

the filters. However, the personnel areas, being under positive pressure, would be doubly protected against radioactive particle infiltration.

Usual auxiliary equipment such as overhead traveling crane and fire-protection service would all be in accordance with the latest and the best power-plant practice.

In addition, owing to poor cooling-water conditions at Gunniston Cove, we provided a multicell induced-draft cooling tower, the basin of which is designed with extraordinary storage capacity for emergency use. The make-up water would be from wells; city water would be used for general service and emergency use.

11. REACTOR LOADING, PROCEDURE AND EQUIPMENT

We provided the most complete reactor-tool handling facilities which we and our subcontractors, Sperry and Westinghouse, could devise. Here again performance and reliability were placed far above initial investment.

Our basic approach was again that normally the station was to be centrally and automatically operated, although capable of manual operation for training or in emergency. Since this is in effect "blind flying," we felt that the handling of reactor tools could well follow the same philosophy.

We planned to have the operation of the reactor tools suspended from the crane monitored by a closed TV circuitry and operated entirely from behind the protective glass of the control room. The reactor was capable of being removed bodily and lowered onto the floor or into a pit so that a replacement could be installed. Other items of equipment were planned for replacement for rapid servicing, to be followed by field repairs at convenience as and when practiced. A high loading bay under the craneway provided for loading of shielded equipment for transport.

12. COST ANALYSIS

A comparison of the APPR (ORNL-1613) and a 2500-kw conventional reactor (steam-electric) as to construction costs is given in Table 1.

Table 2 gives a comparison of the APPR and the conventional type reactor as to predicted operational cost.

13. OVER-ALL PERFORMANCE

Although the net generation of our station from 35,300 lb of steam per hour, after allowing 175 kw for auxiliaries, is only 2014.8 kw effective at the outgoing bus bar, this may be considered close to the practical limit for this cycle and size of unit. The net station heat rate of 16,300 Btu/kw-hr at the switchboard is creditable for this size of unit.

However, by operating our reactor to produce boiling water at 400 to 450 psig instead of pressurized 450°F water, a net increase in excess of 10 per cent in net station output would result from the same fuel consumption, or a net station heat rate of 14,670 Btu/kw-hr, which is better thermal efficiency than some conventional steam-electric 50,000-kw stations still in service.

Of course, operation at back pressure instead of vacuum would result in a still higher over-all thermal recovery, although the actual electrical generation would be reduced. This back-pressure cycle would benefit even more than the condensing cycle through the production of 400-psig steam from water boiled directly in the reactor.

Compared with the conventional coal-fired steam-electric power plant having essentially the same electric generating plant, it is apparent that the APPR net station heat rate would be much lower since the conventional plant suffers a fuel conversion to steam loss of 15 to 20 per cent of the heat content in the fuel burned, whereas the atomic steam generator operates at almost 100 per cent thermal transfer.

Table 2 — Predicted Operational Cost Comparison* (APPR Vs. Conventional)

Item	Operation expense, cents/kw-hr: APPR (ORNL-1613)	Operation expense, cents/kw-hr: 2500-kw conventional (steam-electric)
Labor (12-man staff, average)	0.48	0.48
Fuel charges	0.38	0.80
Fixed charges (10 per cent/year)	3.58	1.29
Other expenses (including repair and maintenance)	0.30	0.20
Taxes	0	0
Total	4.74	2.77

*The above costs are predicted on the following basis: Load factor on 8000 hr per year, 60 per cent; life of plant and equipment, 20 years; APPR fuel life (30 months) at 60 per cent load, $40,000 per year; average net output at 16,300 Btu/kw-hr (APPR), 1250 kw; cost of coal ($0.40 per million Btu), $80,000 per year at 20,000 Btu/kw-hr; fixed charges, 5 per cent on investment plus 4 per cent interest plus 1 per cent insurance.

14. SUMMARY

Although responsiveness, price, and contribution were the stated bases for selection, we were forced to conclude that safety and reliability should be emphasized in view of the plant being located in close proximity to Washington. Our reactor design deviates from the ORNL-1613 design in combining the steam generator with the reactor, which reduces the volume of pressurized water, the circulating water friction loss, and space requirements. Also the design provides for the possibilities of operation as a boiling-water reactor, with an increase in fuel economy. The primary coolant system is reduced to two pumps, check valves, and a 12-in. welded T.

The fuel elements and physics of the reactor are identical with those described in Report ORNL-1613, and the shielding is similar but simplified owing to the compact design. The containment is most complete, to preclude any possibility of contamination or radiation damage, and is designed to be used for decontamination training as well.

The secondary steam turbogenerating system is quite conventional, but, as a special safety feature, we provide elaborate test facilities for full load steam to ensue the reliability of the steam-electric load to hold down the reactor. The electrical system is featured by a dual incoming-outgoing service and double auxiliary buses.

The building provides full crane service and features protective pressurized construction for personnel areas. The reactor tooling is handled by closed TV visual control from the main control room, and the health and radiation monitoring are fully automatic and cover all essential areas.

The net output from 35,300 lb/hr of steam generated is 2014.8 kw after allowing 175 kw for auxiliaries. The unit price, fully installed, tested, and in operation, is approximately $1630 per kw of maximum installed capability compared to about $515 per kw for an equivalent coal-fired steam-electric plant. However, direct comparison of prices is not justified in view of the many special charges against the APPR.

ABOUT THE AUTHOR

Herbert G. Johnson is a graduate of the Massachusetts Institute of Technology, 1927, after which he spent one year as apprentice engineer with the Terry Steam Turbine Co. on turbine design and test.

He later was General Manager of the C. H. Wheeler Mfg. Co., producers of heat-transfer equipment (condensers, cooling towers, pumps, and heat exchangers). Subsequently, he was Vice President of the Madison Iron Works, producers of marine equipment, and District Manager of the Kewanee-Ross Corporation, producers of heat exchangers.

At the time this article was written, he was Vice President (Consulting Engineering) of The Kuljian Corporation, Philadelphia. He is presently with the Johnson Engineering Company, Havertown, Pa.

ARMY PACKAGE POWER REACTOR

Merritt-Chapman & Scott Corp. Proposal

I. B. PURDY

March 7, 1955

1. INTRODUCTION

The use of atomic energy presents a tremendous challenge and opportunity to industry. Merritt-Chapman & Scott Corp. is in an excellent position to undertake the construction of atomic-energy installations because of its cumulative experience gained through years of varied construction activity, equipment acquired and developed to meet the demands of specific projects, and broad range of diverse facilities available through its subsidiaries.

The Merritt-Chapman & Scott offer for the design, construction, and test operation of an Army Package Power Reactor (APPR) at Fort Belvoir, Va., contemplated the cooperation of Ford Instrument Co. of Long Island City, N. Y., and Smith Hinchman & Grylls, Inc., of Detroit, Mich., as major subcontractors.

2. GENERAL CONSIDERATIONS

Three general considerations were kept constantly in mind in the development of the plan for a package power reactor as outlined in the offer submitted. They are as follows:

1. Use of the Atomic Energy Commission's concept of design for a pressurized-water reactor.
2. Adaptation of this design to permit development of a portable reactor that could be used safely for training personnel at the Fort Belvoir site.
3. Design of a building to contain the reactor and its components, with maximum safety a basic factor.

3. REACTOR COMPONENTS

3.1 Fuel Elements and Control Rods

It was anticipated that fuel elements and control rods would be fabricated by Sylvania Electric Products, Inc. Their fabrication techniques are believed to have advantages over other methods, such as high degree of core-to-cladding bond integrity, elimination of fusion welding, and simplified assembly methods. The control rods were designed to eliminate the use of expensive materials and to incorporate other features necessitated by driving control rods from the bottom.

3.2 Control-rod Drive and Actuating System

The development of a hydraulic drive system exterior to the reactor compartment to actuate the control rods from beneath the reactor has numerous advantages. Those most readily apparent are (1) an increase in the inherent safety of the reactor system, (2) the accessibility of actuators during reactor operation, and (3) the simplification of reactor-unloading procedure.

A bottom drive system is believed to be safer than a top drive system, both in case of an incident and during normal reactor-unloading procedure. In the event of a pressure surge, sufficient to blow the lid off the pressure vessel, the control rods, not being attached to the top, would remain in the reactor and would not therefore be ejected rapidly from the core. A top drive necessitates latches that must be released by remote means. With no available positive indication that the latches have released the rods, it is foreseeable that one or more latches could fail; the lid could be raised; and the reactor, being cool, could be put on a fast period. Even if this were discovered in time, a rather touchy problem of releasing stuck latches appears certain.

The actuators would be located in a basement room adjacent to the chamber immediately below the reactor. Sufficient shielding would be provided between these rooms to allow for maintenance and for routine inspection of the actuators during reactor operation. The actuator room would contain the five pipes from the pressure vessel in which the control-rod racks would be located and to which the seals and backup bearings would bolt. The rods would be driven by gears and shafts operating through the rotary seals. The subreactor room would be included in the containment volume, and shafts from the actuators to the rod-drive gears would be sealed by stuffing boxes.

4. PRIMARY COOLANT SYSTEM

The primary loop equipment consists essentially of a reactor pressure vessel (housing the reactor itself), a pressurizer, a steam generator, and the contiguous piping and circulating pumps.

The heat source of the steam plant, which is the reactor core, is contained within the 9-ft by 4-ft 4-in.-diameter steel pressure vessel. The vessel, designed to withstand 1200 psia, would be mounted vertically with a flanged access opening at the top and drive-rod assemblies projecting downward.

The pressurizer is a 5-ft by 3-ft 8-in.-diameter vessel mounted on the system to maintain the required pressures and to avoid flashing of the primary-system coolant water. Immersion heaters within this vessel provide the energy to develop the required pressure of 1200 psia and approximately 400°F of coolant water.

The steam generator is a horizontal U-tube single-pass exchanger. The heat exchange will be between pressurized high-temperature water on the tube side and steam on the shell side.

Two centrifugal canned pumps, one of which would be connected for 100 per cent stand-by operation, are provided to circulate 4000 gal/min of purified coolant water.

Coolant piping would be schedule 80 12-in. steel pipe. Vessels are also of steel and are pressure tested. The interior of each vessel burner would be stainless-steel clad. The entire primary system is planned to be of welded construction, eliminating leakage to practically nothing.

Inasmuch as the primary loop contains the radioactive elements, the reactor, and the radioactive primary coolant water, most of the problems peculiar to an atomic power generating station pertain to this area.

Measures to protect against the hazards involved in the operation of such a power source, however remote, offer the greatest challenge to the ingenuity of the designers.

Shielding for operating personnel is, of course, a fundamental problem. As discussed in more detail in Sec. 6, sufficient masses of ordinary concrete were employed.

Another problem peculiar to an atomic nuclear power plant is the disposal of contaminated waste.

Holdup of contaminated water from the primary system is provided for 1 gal/min of blowdown. Provision must also be made for holdup of the entire 1800 gal of water in the primary system to permit draining and maintenance.

Preliminarily, it was estimated that two 5000-gal underground tanks with the necessary pumps, drains, and control interlocks would be required. Vents, connected into the radioactive exhaust stack, would open only when draining or filling the tanks.

Reactor Instrumentation

Standard electronic reactor-control instrumentation would be provided with sufficient duplication so that failure of any single instrument would not require plant shutdown for repair.

Two independent start-up channels operating log count-rate meters and recorders from fission chambers would be provided. Interlocks on the recorders would prevent rod withdrawal with the count rate below a prescribed level.

Two logarithmic channels operating from compensated ionization chambers would be provided. Readings would be supplied from these channels to log and period recorders, and each channel would also include period and sigma amplifiers for period scram.

Three safety channels operating from parallel-plate chambers for level scram would be incorporated. A spare magnet amplifier is supplied for use with either the period or level scram circuits, The magnetic amplifiers supply power to the scram solenoid valves of the hydraulic actuators.

A channel with a removable BF_3 chamber would be supplied for low-level experiments.

The control room would record radiation levels from monitrons located throughout the plant, on the waste tanks, and on the main steam lines. Portable survey instruments and personnel-monitoring equipment would also be supplied.

5. PRIMARY LOOP CONTROL

The reactor coolant-outlet temperature would be measured and recorded and would automatically control the position of a selected control rod. The temperature in the reactor-outlet leg would be maintained at 450°F. Heat flow into the steam system would be regulated for partial load by allowing the pressure and the temperature on the steam side to increase. A control valve downstream of the heat exchanger would maintain a constant downstream pressure at 200 psia. An instrument measuring the coolant-temperature rise across the reactor would interlock with the alarm system to give both visual and audible indications of excessive temperature rise. A separate two-point recorder on the inlet and outlet temperature would also interlock with the scram system. These independent channels ensure safety in the event of an instrument failure.

The negative temperature coefficient permits simplification of the automatic rod-control servo loop. Transient temperature variations would be regulated by the temperature coefficient and the long-range variations caused by fuel burn-up, and other factors would be regulated by the automatic rod-control system.

Coolant flow rate would be measured and recorded. Alarms would indicate a rate fall to some predetermined level, and automatic scram would be initiated at some lower set rate.

An indicating recording controller would measure the pressure in the primary-coolant pipeline and would provide on-off control for the heaters in the pressurizer. The scram system would be actuated if the pressure departed by predetermined amounts from 1200 psia. Since the pressurizer acts as a volume surge, the liquid level therein would be continuously monitored. A level indicating recording controller would actuate appropriate valves to maintain the desired level and would also interlock with the alarm and scram systems.

The coolant conductivity would be recorded. Excessive conductivity would be reduced by increasing the make-up rate.

Other instrumentation in the primary loop provides suitable control and/or indication for the following: (1) coolant-pump operation, (2) make-up water-pump operation, (3) control-rod seal leakage, (4) demineralizer, (5) flow, (6) make-up water conductivity, and (7) sump-tank level.

6. CONTAINMENT

It was first determined that a nuclear explosion should not be considered a credible incident. Because of the design of the control-rod mechanism, actuated from below the reactor, it is impossible for the rod to enter the core or to be driven into the core quickly enough to cause an explosion; nor could loss of the primary coolant through a rupture in vessels or piping cause an explosion. Should a major rupture occur, the loss of water from the primary loop could conceivably cause melting of the fuel elements with the subsequent release of fission products and steam.

To provide sufficient volume for the approximately 1800 gal of 1200-psia water in the loop to flash and to dissipate its energy to atmospheric, or near atmospheric, pressure was found to require a prohibitively large volume, i.e., the entire power-plant building, including the related office space. Moreover, use of the building as the containment measure would mean an increase in hazards to operating personnel, as well as increased costs in providing a thoroughly airtight building.

After plotting a graph revealing the pressure build-up vs. volume required to contain the flashed steam, it was decided to provide steel pressure tanks as the first line of defense and to allow the building, with minor modifications to normal construction, to provide a reasonably tight and secure secondary line of defense.

The entire primary system, therefore, was designed to be fitted inside two steel pressure vessels.

An economical volume vs. pressure relation was obtained by the provision of two additional underground expansion tanks. Pressures required to be contained by the expansion tank system were then kept below 100 psi, a pressure well within the normal limits of pressure-vessel design.

The reactor vessel and drive-rod mechanism was fitted into a tank 23 ft 6 in. by 7 ft in diameter. The pressurizer, heat exchanger, and pumps were placed within a second vessel 18 ft by 12 ft in diameter. Piping between the reactor and the heat exchanger was encased in steel pipe sleeves welded to, and connecting, the two tanks.

For rapid equalization of pressures, as well as for access, maintenance, and inspection of tanks and contained equipment, the four tanks are interconnected by 60-in.-diameter steel pipe. Heat-exchanger tubes can easily be inspected or pulled for maintenance. In fact, each piece of equipment may be inspected, maintained, or replaced with reasonable convenience.

In laying out the primary system to fit into a limited space, the problems of providing for thermal expansion are multiplied. Space could not be sacrificed for expansion loops or heavy expansion joints in piping. On the other hand, the possibility of rupture due to excess stresses imposed by high pressures and temperatures could not be ignored. The contained equipment therefore was designed to "float" within the containment vessels, the anchor point being the reactor vessel at its control-rod mechanism. The containment vessels and the connecting pipes, where cast in the concrete shield, have been given freedom of movement by a corrugated-metal liner sheet between the steel vessel and the concrete shield.

As a secondary line of defense the building itself is designed to provide containment. Although building walls are designed with ordinary insulated metal panels and the roof is designed with precast concrete slabs, all joints are sealed with mastic to be watertight on the outside and airtight on the inside. Windows have been eliminated, and the only exterior pedestrian door will be a double air-lock type. In the event of an incident, all openings to the outside which are not normally closed and sealed will "fail safe" to provide a building as airtight and leakproof as is practicable. The ventilation system, of course, will automatically "fail closed" during an emergency, but it can be reactivated from the guard house nearby for the controlled exhaust of contaminated air through radioactive filters and the steel stack, which is approximately 70 ft high.

7. STEAM SYSTEM

The steam system, or secondary system, is a conventional power-plant design. It includes such major items of equipment as turbogenerator with accessories, exhaust condenser with air ejector and condensate pump, deaerating feed-water heater, evaporator for make-up water, feed-water pumps, cooling towers, circulating pumps, and two deep-well make-up pumps.

The steam turbine will be a condensing type exhausting to a water-cooled condensing system. The generator is rated at 2090 kw at an 80 per cent power factor.

The condenser is a horizontal surface condenser with circulating water at inlet temperatures of 85°F and at a flow rate of nearly 5000 gal/min from induced-draft cooling towers.

Cooling water for the condenser is obtained by circulating water through a cooling tower. Make-up water for the cooling tower will be obtained from two drilled wells.

Condensate from the condenser will be pumped to the feed-water heater for heating, deaerating, and storage. Feed water will be heated by extracted steam from intermediate stages of the turbine.

The deaerating feed-water heater will be sized with a storage capacity of 1800 gal to provide an available source of distilled water for charging the pressurized primary loop. Feed-water pumps will pump feed water to the heat exchanger in the primary system.

All pumps will be sized for full-load operation with duplicate pumps for 100 per cent stand-by operation.

The evaporator for distilling water required for make-up purposes will be normally operated from high-pressure steam but will include electric heaters for operation during plant shutdown or when no steam is available.

All instrumentation and controls will be arranged for fully automatic operation of both the primary system and the secondary system and will include all shutdown and safety devices required for normal and emergency operation.

8. LOADING AND UNLOADING PROCEDURE

After the primary system is cooled, the compartment over the reactor would be flooded. The concrete-shield top blocks would be removed and stacked to one side. The chamber over the containment vessel would then be flooded, and the dish head would be removed and placed behind the stacked blocks. By the use of a specially designed high-torque wrench, the bolts on the reactor-vessel lid would be unbolted, and the lid would be removed and placed behind the stacked blocks. Next, the upper grid-assembly plate would be unfastened and lifted out of the vessel by an unlatching and removal tool. The fuel elements could then be removed from the core and could be placed in storage racks provided in the skirt plate around the reactor within the concrete contained.

The control rods would be prepared for removal by running the rods to their uppermost position. At that time the lower grid assembly is cleared by the fuel section. The absorber and fuel sections, which comprise the removable portion of the control rods, would be rotated through 45 deg to disengage them from the lower drive-rack section. The control-rod racks could be removed, if necessary, by disengaging the backup bearing assemblies from the rod-drive pipes at the bottom of the pressure vessel. These assemblies would have a stud that would normally prevent driving the rack beyond a set position.

It is believed that removal of the control-drive mechanisms from the top of the reactor simplifies the unloading problem, and preliminary designs of the necessary equipment, such as the high-torque wrench, indicate that the equipment to meet the needs could be fabricated.

9. COST

The Merritt-Chapman & Scott Corp. proposal for the design, construction, and test operation of the APPR was approximately $3,150,000. Of this total almost one-half was for engineering, development work, and test operation.

The contemplated schedule was one year for design, drawings, and approval; one year for construction; and eight months for completion of the 1000-hr test.

The cost of technical personnel to supervise and record the additional six-month operating test was about 80 per cent of the approximate bid of $440,000 for this work.

10. CONCLUSIONS

1. Design and construction of a reactor for the purpose desired by the Army is practical.
2. Design and construction of package reactors for industrial use is economically feasible and practical in certain localities.
3. Specialist companies that must be utilized as subcontractors or material suppliers by general contractors are expressing increasing interest in atomic energy project work.
4. Experience gained in preparing the estimate for the design, construction, and test operation of the APPR will be reflected in lower cost estimates for future similar projects.

ARMY PACKAGE POWER REACTOR

Raymond Concrete Pile Co. Proposal

J. C. SCHLECK* and THEODORE STERN†

March 10, 1955

ABSTRACT

This proposal was predicated on utilizing to the fullest possible extent the ability of established companies to meet the requirements of nuclear power electric generation in accordance with accepted commercial practice.

Gibbs & Hill, Inc., prepared basic sketches, heat balances, and flow diagrams for the complete plant and wrote specifications covering all major mechanical and electrical items of equipment. On the basis of these specifications firm prices were requested from several manufacturers for each item. For this purpose the primary system was considered as a unit.

Plant orientation, layout, and construction represent an equitable compromise between considerations of first cost, operating efficiency, and the experimental nature of the project.

Response by potential suppliers of the primary system was excellent and indicates that industry is prepared to furnish all components of a nuclear-powered electric generating plant on a competitive commercial basis.

In view of the rapidly increasing potential for the application of nuclear energy to the economic production of commercial power, Raymond Concrete Pile Co. and its engineers, Gibbs & Hill, Inc., welcomed the opportunity of developing preliminary engineering for a proposal to design, build, and operate a complete nuclear powered generating plant. This proposal was predicated on utilizing to the fullest possible extent the ability of established companies to meet the requirements, in their particular fields, of the new and unusual concepts of nuclear power electric generation in accordance with accepted commercial practice.

Following this basic concept, Gibbs & Hill prepared the required preliminary sketches, heat balances (Fig. 1), flow diagrams (Figs. 1 and 2), and engineering specifications for all major items of equipment and all basic systems. Specifications were submitted to several manufacturers in each specialized field, with a request that firm proposals be offered for furnishing the necessary equipment. In the preparation of the specifications, the primary system was considered as an integral unit, including reactor, steam generator, pressurizer, pumps, containment, controls, and instrumentation for the generation of steam. Proposals were requested on this basis to follow the presently accepted commercial procedure under which a conventional steam-generating boiler would be purchased.

*Gibbs & Hill, Inc.
†Foster Wheeler Corp.

The remaining items, comprising essentially the conventional steam electric generating system, circulating-water system, building, etc., presented no particularly unusual problems.

Response to requests for quotations on the primary system was most gratifying. Of six invitees, three submitted prices and designs for a complete system as requested. The remaining three deviated from the request because they did not, in general, desire to assume responsibility for the operation of an integrated installation. Of those offering a complete installation, it was determined that the proposal presented by the Foster Wheeler Corp. was the most suitable for meeting the requirements of this project. Nuclear Development Associates were most helpful in that they assisted in establishing basic specifications for the primary system.

1. GENERAL CONSIDERATIONS

Within the basic design criteria established by the Invitation for Proposals and associated documents, the fundamental concepts of the plant presented in the proposal are predicated on providing the utmost economy commensurate with sound engineering practice based on many years of steam power plant experience.

The terrain of the plant site presented some complications in the evaluation of the cost of site preparation against initial and operating penalties inherent in the high circulating-water head and in the provisions necessary to ensure proper dissipation of air-borne contamination which might become necessary through the primary-system exhaust stack.

Because neither station output nor required load factor was specified, it was difficult to evaluate first cost of heat-cycle improvement against any resultant increase in net power; therefore it was decided that in view of the experimental nature of this installation only a minimum of capital expenditure could be justified for heat recovery.

With these factors in mind the basic design of the installation was developed to provide maximum economy for the particular conditions of intended utilization and terrain; however, due consideration was given flexibility in the event that a similar installation should be desired under entirely different physical and operating conditions.

2. REACTOR COMPONENTS

2.1 Reactor-core Assembly

The reactor is very similar to that shown in Report ORNL-1613. The fuel-element assemblies are duplicates of those in Report ORNL-1613 except that zirconium diboride is added as a burn-out poison instead of boron carbide, which would have caused difficulties in the fabrication of the fuel plates.

The fuel plates are designed for use in both the fuel assemblies and the fuel section of the control-rod assembly. The control rods differ from those discussed in Report ORNL-1613 in that the hafnium specified for the connecting ends of the two-piece control rod may be replaced by a cheaper material. Both boron steel and a stainless-steel-clad silver-cadmium alloy are considered. The latter alloy was corrosion tested by Sylvania Electric Products Co., and the results were encouraging; however, no decision can be made until further tests are conducted.

2.2 Control-rod Drive Mechanism

The control rods are operated by a canned motor-driven mechanism, which is the latest improved version by the Westinghouse Electric Corporation of a mechanism developed for the Submarine Thermal Reactor (STR). Essentially the mechanism consists of a reluctance type rotor directly coupled to a roller nut. This nut and the rotor are canned and operate submerged in the primary coolant water. When the mechanism is in operation, the roller nut is held firmly (by magnetic force) in contact with a lead screw attached through a latch mechanism to the control rod. The latch mechanism is similar to that employed in Report ORNL-1613.

The drive mechanism contains a fail-safe magnetic release that permits the screw and the control rod to fall free on signal or on loss of magnet power. Each rod is provided with its own power supply so that singular rod motion or various ganged motions are possible, permitting maximum flexibility.

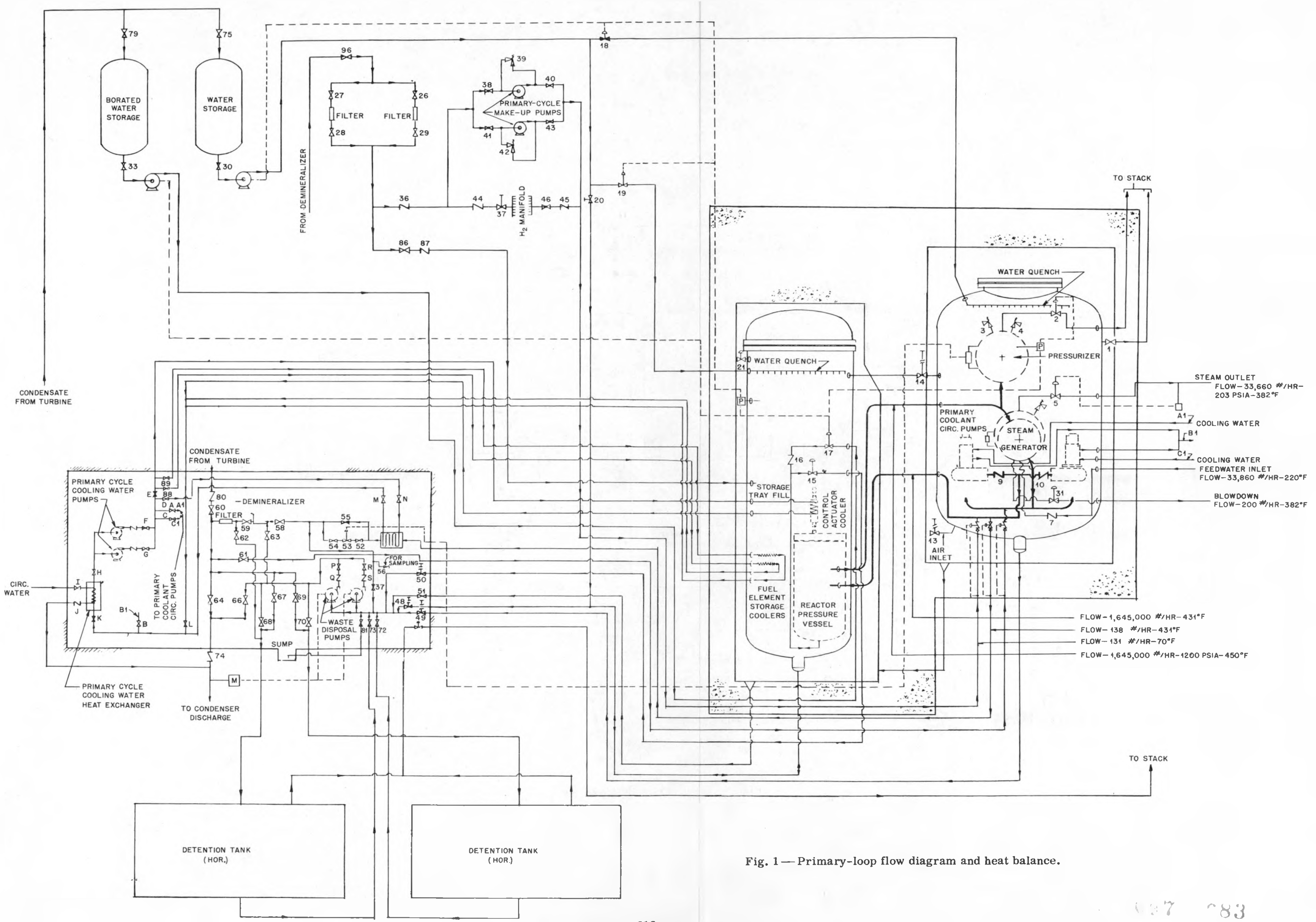


Fig. 1 — Primary-loop flow diagram and heat balance.

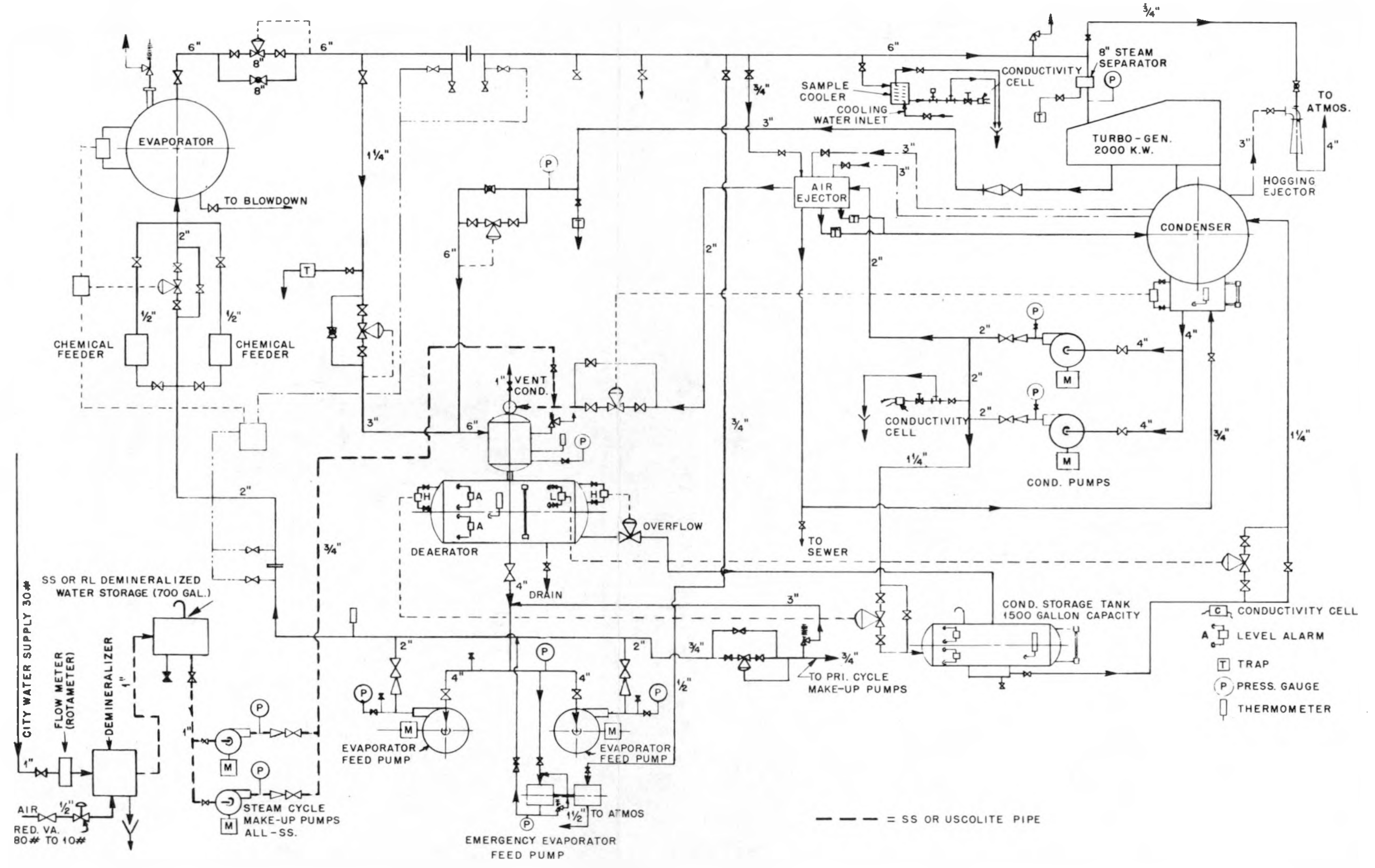


Fig. 2—Steam-cycle flow diagram.

The design of this mechanism assures zero leakage of the reactor fluid. The outer casing is seal welded, and proper protection is supplied for the electrical and cooling-water leads so that the mechanism can operate under water. A thermal sleeve surrounding the lead-screw extension restricts the flow of high-temperature fluid through the rotor cavity. Therefore the fresh-water cooling system on the motor stator is able to maintain reduced temperatures around mechanism parts. As a result the ambient temperature outside the mechanism-enclosing can is not restricted to a 150°F range, thus reducing venting requirements.

2.3 Reactor Pressure Vessel

All pressure vessels are designed to the appropriate ASME code. The reactor vessel is designed for a pressure of 1250 psia and a temperature of 650°F. The base materials for the pressure parts are carbon steel SA-212 grade B for plate and SA-105 grade II for forgings, and all internal surfaces of the vessel and the cover are lined with type 304L stainless steel. The vessel has an over-all height of 9 ft $2\frac{1}{4}$ in. and a 4-ft I.D. Welding on the vessel wall is back-chipped and rewelded with stainless-steel filler rod to provide continuity of the inside stainless surface. All longitudinal and circumferential butt welds on the pressure parts are radiographed. In addition, root passes and final inside passes on the pressure-vessel welds are examined by Zyglo. Weldments in carbon steel are examined by Magnaflux. After assembly is complete, the vessel without the head cover is thermally stress relieved.

2.4 Pressurizer

Connected to the primary loop at a point just before the coolant enters the steam generator, is a pressurizer. Three 36-kw immersion type heaters are used to maintain the pressure of 1200 psia on the system. As the pressure drops, a pneumatic relay turns on one of the 36-kw units. A further drop in pressure actuates relays that turn on the remaining heaters in 18 kw steps. Safety valves located on the pressurizer relieve steam at overpressures. The pressure-vessel shell and heads are of SA-212 grade-B carbon steel clad with type 304L stainless steel. The welds are tested as before, and the pressurizer is stress relieved after welding.

2.5 Steam Generator

The steam generator is of integral design with the heat-transfer surface and the steam separating equipment included within a single vessel. A U-tube bundle is utilized which eliminates the differential thermal-expansion problems encountered in straight-through heat exchangers. The tube sheet is welded to the heat-exchanger vessel, thus eliminating any leakage of primary water into the steam system.

Chevron driers, furnished in the steam space, permit only high-purity steam to flow to the turbine. The over-all length of the generator is 11 ft 2 in., and the outside diameter of the head is 54 in. This represents a considerable reduction in size over a straight-tube steam generator.

2.6 Primary Coolant Circulating Pumps

The primary coolant is circulated through the system by one of the two canned-rotor pumps placed in parallel in the loop. The pumps, which have been developed by Westinghouse, provide for zero leakage, and, since the pump and motor form an integral unit, only two bearings are required, eliminating alignment problems. The pump and motor construction of this design permits the removal of the motor and impeller from the casing as a single unit. Construction material is type 304 stainless steel, and the bearings are made of carbon graphite. The high-temperature-system fluid is kept from circulating through the motor by a thermal barrier and a labyrinth seal.

2.7 Reactor Control System

The control-circuit design is in keeping with the plant design objective—as uncomplicated a system as possible, capable of arctic service, without sacrificing reliability or safety. This was accomplished by utilizing the experience Westinghouse gained in the operation of the STR control and safety system. The source and intermediate range comprise a pulse and a d-c channel, respectively, with associated power supplies, level and period indicators, and level and period recorders. Start-up interlocks are provided.

A bistable magamp in the intermediate-range period circuit provides a shutdown signal to the bistable-magamp magnet supply, which in turn causes the magnetic clutches to release their hold on the control rods. A self-contained test panel provides pulse, level direct current, and period signals for the checking and the alignment of all channels. Three power-range channels operating in parallel provide level indication through a selection meter and three-point recorder. Each channel at a flux level of 1.5 that of full load will trip a bistable magamp whose output into the sigma bus will in turn trip a power bistable magamp, thus removing holding current from the magnetic clutches. The outputs of the channel bistable units may be set by prior switching so that a trip of any two of the three available will be required for shutdown. This coincidence feature reduces the possibility of reactor power interruption if a malfunction or transient should occur in one channel. The alarm circuit, provided for all trips and interlocks, will signal the operator that a second flux-level trip will cause shutdown.

A resistance thermometer keeps the outlet temperature of the primary coolant constant. The power level, represented by the flux, is fed into a magamp comparator as a check on the temperature controller. If the power level exceeds a certain maximum setting, the rod-controller motor will act to insert the regulating rod to lower the power level. Therefore the temperature is the main controlling factor up to this maximum power setting, at which time the flux level overrides the temperature signal and in turn becomes the controlling factor.

No vacuum tubes are used in the power-range indicator, control, and safety circuits. All components except the vacuum tubes in the start-up circuits are designed for long life, and all high-impedance circuits are sealed and desiccated for protection from atmospheric conditions.

All readings (including radiation and leakage checks from the boiler leak detector, air-borne particle detector, and air and waste monitors) are indicated in the control room on a console or control panel. The console contains all those control functions that the console operator requires to maintain control of the reactor in the function of start-up, power-range operation, and safety, with supervisory control of the steam system.

2.8 Primary Coolant System

The water in the primary loop is maintained at 1200 psia and is circulated through the reactor at a sufficient rate such that nonboiling conditions prevail. The reactor is regulated by five control rods, only one of which is used as a regulating rod; this rod is set to maintain the coolant outlet temperature from the reactor at 450°F. From the reactor the cooling water is circulated through 12-in. 304L stainless-steel piping to an integral U-tube boiling type steam generator. From here the water flows to one of two canned-rotor circulating pumps, each of which can supply sufficient head to overcome friction in the loop for a flow rate of 4000 gal/min. The system is so designed that, in the event of failure of both pumps, the reactor will be shut down and the decay heat will be removed by natural circulation.

A small portion of the primary coolant water is continually purged from the system to maintain a solids concentration of about 2 ppm and a conductivity of 4 micromhos. The purge rate is controlled by the conductivity of the primary loop and by the pressurizer liquid level. If the level becomes too high, then the purge rate is automatically increased, thus returning the level to its normal position.

Make-up to the primary loop is taken from the plant demineralized-water storage and is introduced into the system by one of two positive displacement pumps which are actuated by a signal from the pressurizer liquid-level recorder controller. One of the make-up pumps is held in stand-by, and, if one of the pumps fails to start from a signal of low liquid level, the second pump will automatically be cut in. The make-up water contains hydrogen at a concentration that is sufficient to ensure a maximum oxygen content of the primary coolant water of no more than 0.1 ppm.

All piping and metal surfaces in contact with the primary coolant water are of type 304L stainless steel, and the primary loop is so designed that expansion stresses will be absorbed without requiring movement of either the reactor pressure vessel or the heat exchanger.

2.9 Containment and Shielding

The maximum credible accident that is assumed to occur in the APPR is that of a rupture in the primary loop while operating at rated power.

The primary loop is contained within a steel envelope consisting of two large cylindrical vessels appropriately interconnected. The 12-in. piping between the reactor vessel and steam generator is enclosed in a 24-in. pipe envelope connecting the container vessels. When the reactor is in operation, the containers are completely sealed.

All lines passing through the containers are fitted with valves located inside the containers, and, in the event of a rupture in any of these lines, the valves will automatically close, preventing primary coolant from escaping to the outside of the containers.

A rupture in the primary loop permits the primary coolant water to flash to steam and liquid water at a lower pressure and temperature. If the break is such that all the primary coolant flashes from 450°F (equivalent to 10 sec additional operation at 10 megawatts), the pressure inside the containers will be slightly under 150 psia.

The reactor should scram from any one of several signals actuated by the rupture. Should the rods not drop (the case where the break may distort the rods), then the reactor may not be subcritical until approximately 70 per cent of the primary-loop water has flashed to steam. However, it is undesirable to have the water flash to steam because the fuel elements will melt. Therefore additional water must be added to cool the fuel elements; but at the same time the reactor must be subcritical, and the pressure build-up in the containment vessels must be curtailed. This is accomplished by pumping borated water into the reactor (to cool the elements and add poison to the core) and by introducing spray water at the top of the containers (to prevent excessive build-up of pressure by condensing steam).

The spray system is started when the container pressure exceeds 50 psia. Borated-water pumping occurs automatically when the primary loop pressure drops to 500 psia and the container pressure rises to 50 psia. At a pumping rate of 200 gal/min it takes 1.25 min for water to reach the bottom of the fuel element. The maximum temperature the fuel elements reach before they are completely covered by borated water is 1460°F. In the event of a rupture occurring at the bottom of the reactor vessel, the fuel elements will reach a temperature of 2175°F before they are covered by borated water.

Failure of both the borated water and spray system will cause melting of the fuel elements, but the heat-removal rate of the air outside the containers is such that it can remove the decay heat generated. Even if air cooling is not available, there is no danger of melting the containers, although a 10-psi pressure rise may be expected.

Decontamination is easily accomplished. The inside of the containers can be flooded and washed using the existing spray quench system. It is possible to acid wash through the quench system. All waste can be passed from the containers directly through the existing waste-disposal system.

A concrete biological shield is used, and all its exposed surfaces are dustproofed and hardened.

3. GENERATING SYSTEM

The proposed plant provides a gross electrical output of 2000 kw and a net station output of 1800 kw based on 85°F circulating water. The turbine generator is a standard 2000-kw nominal rated machine designed to operate under maximum steam conditions of 250 psig and 500°F total temperature and is complete with automatic governor-controlled valves, combination stop and throttle valve, complete lubricating system, one extraction point, and all other standard accessories. Because the additional initial cost of providing a turbine capable of utilizing the higher pressures imposed under low load conditions does not appear justified, a pressure-reducing station is provided.

Additional equipment includes a two-pass surface condenser with two-stage and priming ejectors, a deaerator heater supplied by turbine extraction steam, a demineralizing make-up water treatment for both primary and secondary systems, 8-hr storage facilities for treated water, chlorination equipment for circulating water, and other necessary auxiliary equipment.

Plant generation is at 4160 volts to provide direct integration with the projected distribution system for Fort Belvoir. The complete electrical system provides all facilities necessary for central control of plant electrical transmission as normally applied to public utility installations.

4. BUILDING AND AUXILIARY FACILITIES

Building specifications are based on the general usage for which the installation is intended. Because it is definitely established that training programs are to be maintained and it appears probable that the installation will receive much public attention, it is deemed desirable to depart from the most economical construction to provide a reasonable amount of interior decoration in the office, control room, and classroom section of the building.

The building design comprises light steel framing and concrete-block exterior walls except in the turbine and reactor room which are corrugated transite. The interior walls in the office and classrooms are generally of flush-joint concrete-block construction. Hung acoustical ceilings, asphalt-tile floors, and fluorescent lighting are provided in the control room, offices, and classrooms. In all other respects basic requirements control the design to achieve minimum cost.

5. REACTOR LOADING PROCEDURE AND EQUIPMENT

After the reactor is shut down, all control rods are run down until unlatching lights indicate that they are in the unlatched position. Water from the storage tank is pumped into the reactor container vessel, forcing the air in the container into the steam-generator container, where it is slowly bled into the stack. When the container is almost completely filled, a valve in the top of the container vessel is opened to permit fresh air to replace the old air that remained in the container. In the meantime, the top shielding plugs are removed, and, because the blower is still in operation, air will be drawn from the building into the shield through the top plug opening. The top closure of the reactor container is unbuttoned and placed in the storage pit in the shield with the crane hook. Four of the shield plugs are replaced to reduce radiation leakage.

Next, the reactor pressure-vessel closure is unbolted and is raised 2 in. Each control drive rod is then driven to its top position while nuclear instrumentation is watched to make sure that the control rods are not being lifted. (If possible, positive unlatching indication should be provided to make this step unnecessary.) After the drive rods are raised and the coolant and electrical lines are disconnected, the closure plate with the drive mechanisms is raised and placed in its storage pit.

The upper assembly grid is then placed in storage, and the fuel elements are moved from the core to the storage trays located next to the reactor in the container. New fuel elements are loaded into the core under conditions simulating a critical experiment.

The closures are replaced in exactly the reverse procedure, and the water in the reactor container vessel is drained to the holdup tanks in the waste system. The water used for refueling is not demineralized and is in contact with the carbon container during refueling. Some mixing of this water with the water in the primary loop has probably occurred; therefore the loop water is circulated through the system with the bypass demineralizer line open. While this operation is in progress, the heaters in the pressurizer are turned on to build up pressure. At 200 psia the reactor vessel blowdown valves are opened to blow out any solid material that might have fallen into the vessel during the refueling operation.

6. COST ANALYSIS

Cost estimates for the installation described herein are presented below in major categories.

Buildings and structures (including specific excavation, all general interior facilities, and equipment foundations)	$ 145,000
Outside facilities (including site preparation, landscaping, security, and circulating-water system)	110,000

Transmission and substation facilities	$ 55,000
Primary system (including engineering, development, and fabrication of fuel and control rods)	1,689,000
Shielding	131,000
Containment	156,000
Secondary system	400,000
Engineering and construction charges (including field testing, except six-month test)	692,500
Total cost	$3,378,500
Additional cost of six-month operating test	$ 247,000

7. SUMMARY

Raymond Concrete Pile Co. and Gibbs & Hill, Inc., took the position that a consulting engineer should perform the same general function in connection with the design of a nuclear powered electric generating station as has been practiced for many years by comparable organizations in the field of steam power generation. In this concept the manufacturer of equipment assumes responsibility for developing and manufacturing the various components that comprise the generating plant. It is the function of the engineer to interpret the requirements of the particular project and to apply to the best possible advantage the equipment available for fulfilling those requirements by creating designs for the complete installation.

The experience of Gibbs & Hill in preparing the proposal for this project has established beyond doubt that this procedure is feasible and that industry is now prepared to furnish the equipment required for such plants on a commercial competitive basis.

ABOUT THE AUTHORS

John C. Schleck, Mechanical Engineer, Gibbs & Hill, Inc., graduated from Brown University in 1946 with the B.S. degree in civil engineering. In 1953 he received the M.S. degree in civil engineering from Newark College of Engineering and is presently working toward his doctorate at Columbia University, taking courses in mathematics and nuclear physics. After completing a tour of duty in the Navy, he joined the Mechanical Engineering Staff of the Esso Standard Oil Company in 1947. In 1951 he joined Gibbs & Hill, Inc., as a hydraulic engineer on the Savannah River Project for the Atomic Energy Commission. Schleck spent 1 year at the Knolls Atomic Power Laboratory as a consultant during 1953 and 1954. His present activities with Gibbs & Hill are concerned primarily with the atomic energy program.

Theodore Stern is a project engineer in the Nuclear Energy Department of the Foster Wheeler Corp. He received the B.Mech.E. degree in 1951 from Pratt Institute and has done graduate work in mathematics and physics at New York University. He attended the Oak Ridge School of Reactor Technology in the 1951 to 1952 session and then joined the Foster Wheeler Atomic Energy Commission Industrial Participation Program. His present activities are concerned with reactor design.

ARMY PACKAGE POWER REACTOR

The Stearns-Roger Mfg. Co. Proposal

R. W. AKERLOW, F. THURLOW LACY,
and MACKEY M. PAYNE

March 14, 1955

1. INTRODUCTION

For at least the past 40 years, The Stearns-Roger Mfg. Co. has been engaged in designing and constructing steam power plants in the Rocky Mountain region. In the early days these plants used coal as a source of power. As the territory developed and gas and oil became available, Stearns-Roger engineered conversions to the new fuels. Now we have entered a new age in which the atom becomes a source of energy for developing power. Stearns-Roger is naturally interested in this latest advance and welcomed the opportunity to take part in bidding on the Army Package Power Reactor (APPR).

We found, in entering into this new kind of work, that the information developed and furnished to us by the Oak Ridge National Laboratory (ORNL) was invaluable in the preparation of our proposal. We received, and appreciated, numerous offers of assistance from other companies that had been previously engaged in the development and utilization of atomic energy. We felt, however, that we should attempt to utilize our established techniques and apply them to this new problem and, at the same time, learn as much as possible about the new source of power by doing as much as we could toward design of the primary loop.

2. PRIMARY LOOP

Starting with the ORNL 10-megawatt Conceptual Reactor Design as a heat source, we developed a heat balance, as shown in Fig. 1. We accepted the ORNL primary-loop design with minor modifications. Special attention was paid to furnishing a true no-leak system by eliminating all packing glands and stuffing boxes on the equipment selected. We realized that this decision would be more expensive in the original equipment, but we believed that it would reduce maintenance and make the operation of the plant more satisfactory. A brief description of the main components which were selected for the primary loop follows.

2.1 Reactor Pressure Vessel

The reactor pressure vessel reactor was to be fabricated of mild steel and was to have all internal surfaces clad with type 304L stainless steel. Control-rod drives as furnished by Westinghouse were selected because we believed their construction would eliminate all possibility of leakage and because they had been previously used and tested on a similar system. We decided to use core fuel assemblies as furnished by Sylvania Electric Products, Inc., primarily because of their better guarantee. The reactor pressure vessel, control-rod thimbles,

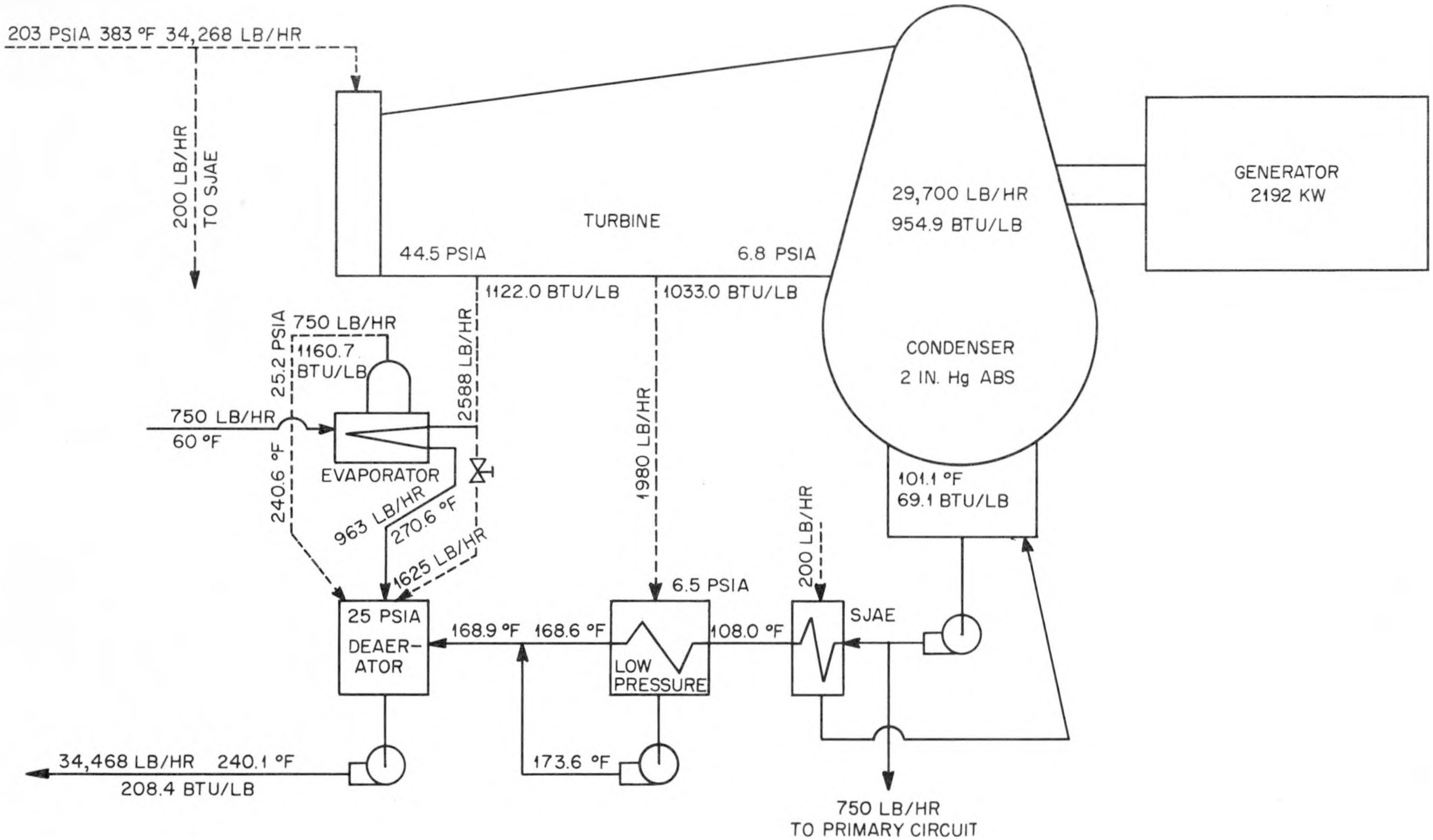


Fig. 1—Heat balance. Maximum gross generator output, 2192 kw; plant auxiliary power, 202 kw; and maximum net plant output, 1990 kw.

and internal fuel-plant support structures would all have been of Stearns-Roger design and fabrication.

2.2 Steam Generator

The steam generator was to be furnished by the Griscom-Russell Co. This generator would consist of a mild steel vessel having all internal surfaces clad with type 304L stainless steel. The tube bundle was to be of U type design and was to consist of type 304L stainless-steel tubes rolled and seal-welded into a tube sheet.

2.3 Pressurizer

The pressurizer was to be fabricated of mild steel, all internal surfaces being clad with type 304L stainless steel. Pressure was to be maintained by three 36-kw nine-step (each) Calrod immersion heaters. The purpose of the pressurizer was to maintain the primary system at a fixed pressure and to act as an expansion chamber. The pressurizer vessel was to be designed and fabricated by Stearns-Roger.

2.4 Two Primary Circulating Pumps

These pumps were to be of the canned-rotor type, as designed and fabricated by Westinghouse. Each pump was to have a capacity of 4000 gal/min at 26 ft head.

2.5 Two Primary-loop Check Valves

These valves were to be of Stearns-Roger design and fabrication, inasmuch as they were of a new type. Two check valves were to be used in the primary loop, one located on the discharge side of each canned-rotor pump. The proper valve would automatically close when pressure was applied under the check, and the other valve would remain open when the corresponding pump was in operation. The use of these valves enabled the primary loop to have a positive thermal circulation without the need of a valved by-pass when the pumps were not in operation. Thermal circulation was necessary to dispose of decay heat of the reactor core.

2.6 Primary-loop Make-up Demineralizer

This demineralizer was to be as furnished by the Cochrane Corporation and was to be sized to treat 30 gal/hr of secondary-loop condensate to the desired purity.

2.7 Primary-loop Make-up Pump

These pumps, positive displacement pumps each having a capacity of 2 gal/min at 1250 psi total head, were to be furnished by the Milton Roy Company. Make-up water for the primary loop would be taken from the secondary loop, demineralized, and delivered at a constant rate to the primary system. Blowdown of the primary system would be intermittent and would be governed by the liquid level in the pressurizer. In this manner the high-pressure make-up pump could be operated continuously with associated advantages. Reserve capacity for primary system make-up would be provided.

2.8 Contaminated Drain System

The contaminated drain system proposed for the primary-loop blowdown and building contaminate drains was to consist of the following equipment:

1. Drain pumps: One 5 gal/min centrifugal pump and one 50 gal/min centrifugal pump were to be furnished to pump the contaminated fluids.
2. The contaminated-drain make-up heat exchanger: A Griscom-Russell Co. twin G-fin section heat exchanger was to be provided to reduce the temperature of the primary-loop blowdown in order to protect the resins in the contaminated-drain demineralizer. The heat removed would be absorbed by the primary-loop make-up fluid.
3. Demineralizer: One Cochrane Corporation two-bed demineralizer of adequate size was to be provided to clean up all contaminated fluids.
4. The holding tanks: Two 1000-gal-capacity holding tanks were to be provided to receive the demineralizer effluent. These tanks were to be continuously monitored for radioactivity levels. When radioactivity was at an acceptable level, the fluids from the holding tanks were to be discharged to a leaching field.

3. PHYSICS

We felt that the physics of the reactor as calculated by ORNL would suffice for the purposes of our proposal. In the event a contract was awarded to us, we planned to have these nuclear calculations checked by an independent organization so that we might be in a position to guarantee plant performance.

4. SHIELDING

We used the recommendations of ORNL as a basis for the shielding requirements. The concrete enclosure for the entire primary loop was to be of reinforced concrete and would have served a dual purpose, namely shielding and containment. Stainless-steel tubing was to be cast in the walls of the compartment to provide a means of circulating cold water through the concrete and thus maintain safe operating temperatures for the concrete.

5. CONTAINMENT

The second purpose of the concrete enclosure was for containment, and we will now describe the provisions made to accomplish that end. We believed that there were two basic types of credible accidents to be considered in our design of containment: (1) a mechanical failure of some type, releasing large quantities of radioactive steam, and (2) a nuclear explosion. We felt that the possibility of a nuclear explosion was extremely improbable. For this reason a mechanical failure was established as the maximum credible accident upon which containment design would be based.

Our design for containment of radioactive material in case of accident to the system utilized the concrete shield. To reduce the pressure of released steam, we provided a means of condensing a portion of it rapidly and effectively. Suitable ducts from strategic points in the shield compartments led to an 8-ft-diameter by 20-ft-long tank of water. The steam flowing through these ducts would be distributed in the water by perforated pipes located near the tank bottom, thus condensing a portion of the steam and substantially reducing the final pressure. Access covers to the shielded compartments were to be bolted down and sealed so that no leakage of contaminated vapors could occur at this point.

6. SECONDARY LOOP

The steam system was to be of conventional design and would have utilized standard components throughout. We decided that a cooling tower would be a much more reliable and predictable means of obtaining condensing water than the use of Potomac River water. We also believed that it would be less expensive to build a cooling tower than to provide the structures necessary to use water from the river.

7. ELECTRICAL

The plant we offered was guaranteed to produce a maximum gross electrical output of 2192 kw and a guaranteed maximum net electrical output of 1990 kw when delivering electricity to a system having a 0.88 power factor. Auxiliary equipment power requirements are shown in Table 1.

The 4160-volt plant bus consisted of indoor metal clad switchgear with three electrically operated air circuit breakers. Breaker 1 was the main generator breaker and connected the generator to the 4160-volt plant bus. Breaker 2 was the main feeder breaker to substation H327 on the lines of the Virginia Electric Power Company system, and it could be closed after synchronizing the plant bus with substation H327. Breaker 3 was the plant station power breaker and fed the station power transformer. Suitable equipment was selected so that synchronization between the Virginia Electric Power Company system and the APPR plant generator could be accomplished from the plant control room.

The power line from the APPR plant to substation H327 consisted of 3C 250 MCM self-supporting aerial cable. A ground grid was designed beneath the plant which consisted of 4/0 cable and ground rods. A 100 amp/hr battery with rectifier type charger was supplied for the station battery. Other requirements of the plant, such as lighting, telephones, American District Telegraph system, and lightning protection, were to be provided as specified.

8. INSTRUMENTATION

A control room was to be located on the turbine operating floor level of the proposed build-

Table 1—Auxiliary Power Requirements
(Maximum gross output, 2192 kw; plant auxiliary load,* 202 kw; maximum net plant output, 1990 kw)

Description	No. req'd	No. in use	Head, ft	Capacity, gal/min	Motor, hp	Actual hp req'd	Motor efficiency	Load factor	Input kw
Hot-well pump	2	1	125	75	3	3	0.81	0.8	2.2
Circulating-water pump	2	1	61	5000	125	102	0.91	1.0	83.5
Boiler-feed pump	2	1	1155	75	50	41	0.90	1.0	34.0
Cooling-tower fan	1	1			50	47	0.90	1.0	39.0
Water-well pump	1	1	120	175	10	7.6	0.85	0.4	2.7
Low-pressure heater drain pump	2	1	90	5	1.5	1.5	0.78	0.8	1.2
Primary-loop circulating-water pump	2	1	26	4000	40 (kva)	26	0.71	1.0	27.4
Primary-loop make-up pump	2	1	2888	0.5	0.75	0.53	0.70	0.3	0.2
Contamination drain pump	1	1	60	5	0.25	0.12	0.70	0.8	0.1
Contamination drain pump	1	1	60	50	1.5	1.25	0.78	0.3	0.4
Shield coolant-water pump	2	1	100	25	1.5	1.47	0.78	1.0	1.4
Pressure heater	3	2			72	72		0.07	5.0
Battery charger and instruments	1	1			1	1		0.50	0.5
750-kw transformer losses	1	1							3.0
Miscellaneous for contingency									1.4
								Total plant auxiliary load*	202.0

*Not including heating, ventilating, air conditioning, and lighting load.

ing, and it would house all instruments and controls for the primary system, steam system, and the electrical system. The plant instrumentation and control would consist of three basic divisions. The first division would cover the primary loop and health monitoring systems; the second division would cover the steam system; and the third division would consist of instruments and controls for the electric generating and distribution system.

The proposed primary-loop system was designed to be as "fail safe" as possible and included three basic reactor scrams: (1) fast, from the safety channels or Log N period channel; (2) intermediate, from all trip circuits; and (3) slow, from all automatic rundown circuits. On the fast scram duplicate Log N period channels were included for additional safety. Contacts on the Log N recorder were provided to limit flux at start-up to a preset value and to provide a limit on high flux level at full power. Both these contacts were connected into the automatic rundown circuits. All fast, intermediate, and slow signals were to be annunciated. Duplicate log count-rate channels were proposed to enable the use of alternate fission chambers for calibration purposes. A servo recorder and a high-speed platinum resistance thermometer were used to control the temperature of the primary loop through the control-rod drive motors.

Two systems of health monitoring were included. The first, having approximately 12 channels, covered local and remote area monitoring. The second system, with 5 channels, was restricted to monitoring liquid wastes. All 17 channels were tied into an audible and visual annunciator.

The instrumentation and control for the steam system was to be of standard design familiar to all. Our instrumentation and control systems were of the electrical type rather than a pneumatic system frequently used. All motors in the plant could be operated from a panel board in the central control room. Control of the generator and electric distribution system was also to be performed from this point.

9. BUILDING AND AUXILIARY STRUCTURES

The building, which was to house the complete nuclear power plant, would also have included the specified offices, laboratories, classrooms, toilet rooms, etc. The building as designed was to be supported on concrete footings and walls below grade. It was to have concrete floors throughout, standard brick construction with a structural steel skeleton, interior partitions of hard burned brick, and metal doors and windows. The offices and classrooms were to be treated acoustically, and the entire building was to be air conditioned and electrically heated. We planned to drill a well to supply make-up water for all plant uses.

10. REACTOR-LOADING PROCEDURE

The proposed procedure for reactor loading and the equipment to be provided followed the suggestions outlined by ORNL, except that large impact wrenches would not be needed to remove the reactor cover. This need was alleviated by employing a cover sealing method utilizing keys. The access covers over the reactor compartment were designed in three horizontal sections so that when assembled the cover, would consist of three solid disks securely set in place. The overhead traveling crane was to have a capacity of 20 tons, which would easily lift one section of the access cover.

When access to the reactor was needed, provisions were made to flood the compartment as a first step. Following this, the concrete covers would be unbolted, removed, and placed on the floor beside the opening. Lead brick was to be provided for temporary shielding around the lower cover.

After the covers were removed, the flooded compartment would be completely visible by illumination provided within. The reactor cover would then be removed, after its bolts and keys had been unfastened, and it would be laid beside the reactor on a shelf provided in the compartment. Spent fuel elements when removed would be placed in subcritical storage racks adjacent to the reactor vessel. New fuel elements would be installed, and the covers would be replaced in reverse order of their removal. After the compartment covers were securely bolted in place, the water which acted as a shield would be drained off, and steps would be taken to put the primary system in operation from the control room.

11. COST ANALYSIS

We believe that a cost analysis of our bid would be of dubious value because of many indefinite conditions which surrounded this project.

12. SUMMARY

We feel that the experience we gained in the preparation of our proposal to the Atomic Energy Commission for the APPR was well worth while. We appreciated the cooperation which we obtained on every hand. It is our intention to continue our activity in the field of nuclear power, and we hope that some time in the future we shall be given the opportunity to build a nuclear power plant.

ABOUT THE AUTHORS

R. W. Akerlow graduated from the Westinghouse Trade School, South Philadelphia, in 1926. He was employed briefly by Consolidated Edison, and in 1929 became affiliated with the Stearns-Roger Co. His activities have centered in the steam power-plant field, with particular emphasis on steam-boiler engineering. His present activities include the development of nuclear engineering projects.

F. Thurlow Lacy received the B.Mech.E. degree from Columbia University. He was initially employed by the American Locomotive Co., Schenectady, and later by the Rio Grande Western Railroad, Denver. He joined the Stearns-Roger Co. in 1926 and has been engaged in shop projects, steam power-plant work, and gas and oil projects.

Mackey M. Payne received the B.Mech.E. degree from the University of Colorado. After leaving college he served with the University of Colorado Physics Department Upper Air Laboratory. He then worked at the General Electric Co., Hanford Atomic Products Operation, in the reactor design and construction section. He joined the Stearns-Roger Co. in 1954.

A High-flux Research Reactor for Large-volume Irradiations

R. G. MALLON and W. R. PEARCE

Bendix Aviation Corp.
Research Laboratories
Detroit, Mich.

June 3, 1955

ABSTRACT

This article describes the conceptual design of a fully enriched graphite-moderated research reactor intended for large-volume irradiations. The reactor is cooled and fueled by a solution of uranyl sulfate in light water which is circulated through the graphite in parallel tubes.

For a reactor power of 120,000 kw, an average thermal flux of 4.4×10^{14} *is obtained in the graphite. Fifteen through facilities are provided, one of which permits irradiation of specimens up to 16 by 16 in. in a thermal-neutron flux of* 10^{15}.

1. INTRODUCTION

Considerable interest has arisen in the so-called engineering test type of reactor, a facility intended for large-volume irradiations in high neutron fluxes. Such a facility would contain large experimental volumes within the core in which fuel elements or reactor components could be tested under the approximate conditions for which they are designed. The requirement of minimum critical mass, or maximum thermal-neutron flux per unit power, together with the large reactor size requirement, suggests the use of a moderator that presents a low cross section and a long migration length. Of the two such moderating materials, graphite and heavy water, graphite is most compatible with the high-temperature experimentation anticipated.

It is suggested that an ideal engineering-test facility would consist of a fully enriched graphite-moderated reactor employing aqueous fluid fuel.

The use of aqueous fluid fuel presents all the merits of the aqueous homogeneous power reactor yet is not subject to its inherent disadvantage of poor thermodynamic characteristics. Among these merits are:

1. Fuel fabrication and radiation damage to fuel are eliminated.
2. There is continuous operation without need of shutdown for reloading.
3. There is continuous removal of fission-product gases.

4. There is no fuel allotment for burn-up. This permits constant flux to be maintained at constant power.

5. There is inherent self-regulation. The production of steam displaces fuel as well as coolant-moderator; hence the reactor power is self-limiting.

For the high specific powers attainable with this type of reactor, continuous refueling is extremely desirable because of the rapid depletion of fuel.

The use of fluid fuel also permits reactor control to be performed largely, or perhaps entirely, by variation of fuel concentration. This allows reactivity compensation of very large quantities of experimental absorber. Furthermore, such control is a homogeneous effect; hence flux distributions need not be disturbed.

The development problems of the system are, in general, common to all fluid-fuel systems or to any high-flux reactor. Specifically these appear to be:

1. Remote handling of fluid fuels and off-gassing of decomposition products.
2. Proper design of inlet and exit plenums and heat exchangers with due regard to criticality hazards.
3. Maintaining a high temperature in graphite moderator to permit annealing of radiation damage.
4. Provision for handling internal fuel-solution leakage in the event of rupture of a process tube.

To determine the feasibility and nuclear characteristics of such a system, a conceptual design of a 120,000-kw reactor has been made.

2. REACTOR DESIGN CONSIDERATIONS

The nuclear characteristics of a graphite reactor employing a light water–uranyl sulfate fuel solution were examined with intent to optimize the thermal-neutron flux in the graphite per unit reactor power. The design constraints were that 120 megawatts of heat be removed without boiling the fuel solution and that the reactor be at least 8 ft in dimension to permit large-volume irradiations.

2.1 Method of Calculation

Critical mass was determined by using the continuous-slowing-down theory and a reflector savings obtained by the one-group theory. Nuclear constants were chosen for each material on the assumption that the effective neutron temperature was determined by the temperature of the material. Thus the thermal cross sections of graphite are averaged over a Maxwell-Boltzmann distribution at 500°F, and those for water and for uranium are similarly determined at 150°F. Thermal utilization was adjusted by iteration in consideration of the greater thermal flux in the graphite region.

The ratio of average thermal flux in the graphite to average thermal flux in the fuel solution was determined by a two-group two-region diffusion calculation for a unit cell consisting of a tube of fuel solution surrounded by graphite. In addition to the requirement that fast- and thermal-neutron fluxes and current densities be continuous at the graphite-water interface, an additional condition is required at the outer boundary of the unit cell. Since k_∞ for the reactor is greater than unity, there must be neutron leakage from the unit cell. It was assumed that this leakage consists only of fast neutrons and that the thermal-neutron gradient is consequently zero at the cell boundary. This is equivalent to the assumption that the thermal-neutron gradient for the reactor is zero at the core-reflector interface.

2.2 Results

The average thermal flux in the graphite per unit reactor power is proportional to the ratio of average thermal fluxes (graphite to water) divided by the critical mass. This ratio is directly dependent upon three variables; the ratio of water to graphite, the reactor size, and the size of the unit cell. Over the range of variables investigated it appears that the degree of dependence is in the order mentioned. The optimum reactor is characterized by the least percentage of water, the smallest reactor size, and the largest cell radius, consistent with heat-removal requirements and the desired irradiation space.

3. A PROPOSED DESIGN

3.1 Reactor Data

A reference design of a 120-megawatt reactor is presented which seems to best utilize the

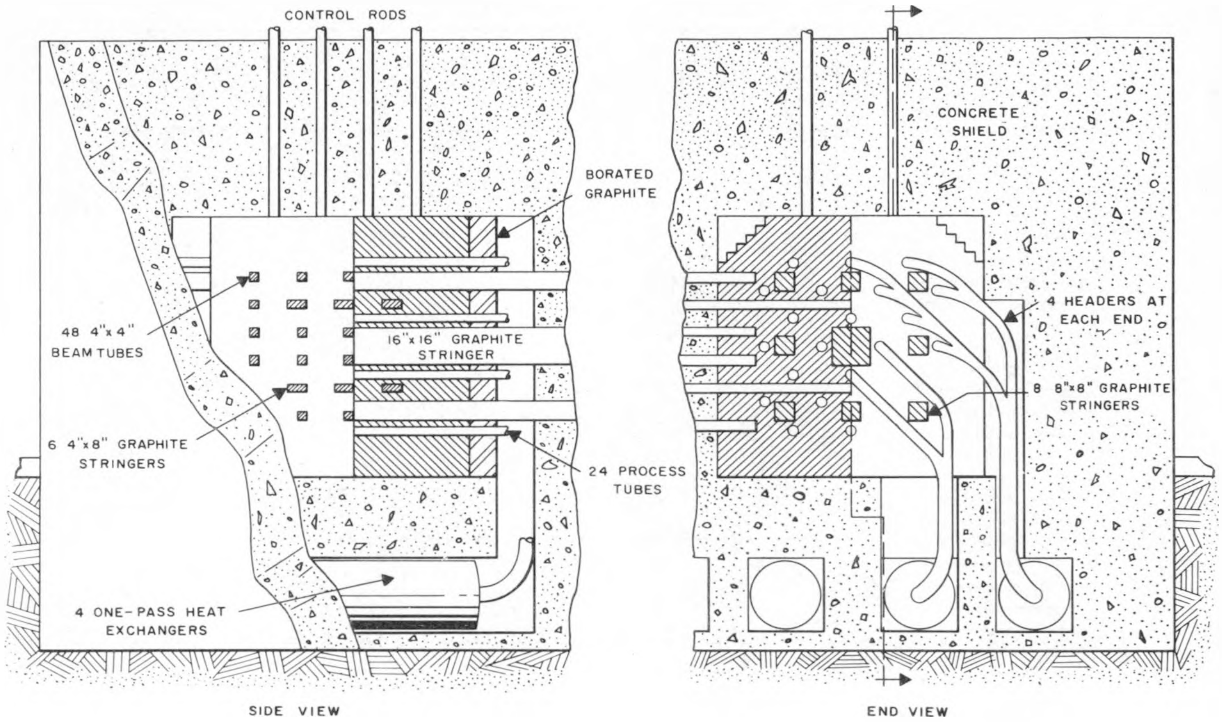


Fig. 1—Schematic layout of proposed reactor.

Table 1—Nuclear Design Data

Reactor power, 120,000 kw	
Hot critical mass, 10.2 kg of U^{235}	
Fuel concentration, 30 g/liter	
Fuel consumption, 160 g/day	
Core composition, vol. %	
Graphite (effective density = 1.65), 96.55	
Water, 3.0	
Zirconium, 0.45	
Neutron flux, cm^2/sec	
Av. ϕ_{th} in fuel solution, 3×10^{14}	
Av. ϕ_{th} in graphite, 4.4×10^{14}	
Av. ϕ_{th} in graphite at reactor center, 1×10^{15}	
Av. ϕ_F in fuel solution, 2.8×10^{14}	
Av. ϕ_F in graphite, 1.0×10^{14}	
Av. ϕ_F in graphite at reactor center, 2.3×10^{14}	
Core size, 8- by 8-ft right-circular cylinder	
Radial graphite reflector thickness, 20 in.	

Table 2—Power-removal Data

Process tubes:	
Number of tubes, 24	
Material, zirconium	
Dimensions, 3.375 in. I.D.	
3.625 in. O.D.	
Liquid velocity, 25 ft/sec	
Inlet temperature, 100°F	
Exit temperature (average tube), 145°F	
Exit temperature (hottest tube), 172°F	
Maximum wall temperature, 210°F	

trends indicated. A schematic layout of the facility is shown in Fig. 1. Design data are presented in Tables 1 and 2.

Flux distributions within a unit cell and for an average core location are plotted in Fig. 2. For the reference design, the average thermal flux in the graphite is 1.48 times that in the water. The maximum thermal flux of 10^{15} will be obtained at the midpoint of the 16- by 16-in. graphite stringer indicated in Fig. 1.

The graphite moderator is fabricated in 4- by 4-in. blocks, and the fuel tubes are mounted in staggered rows on 24-in. centers. Provisions for maintaining high graphite temperatures and for handling internal fuel leakage have been neglected in this study.

Tube wall temperatures have been calculated on the assumption that 5 per cent of the reactor

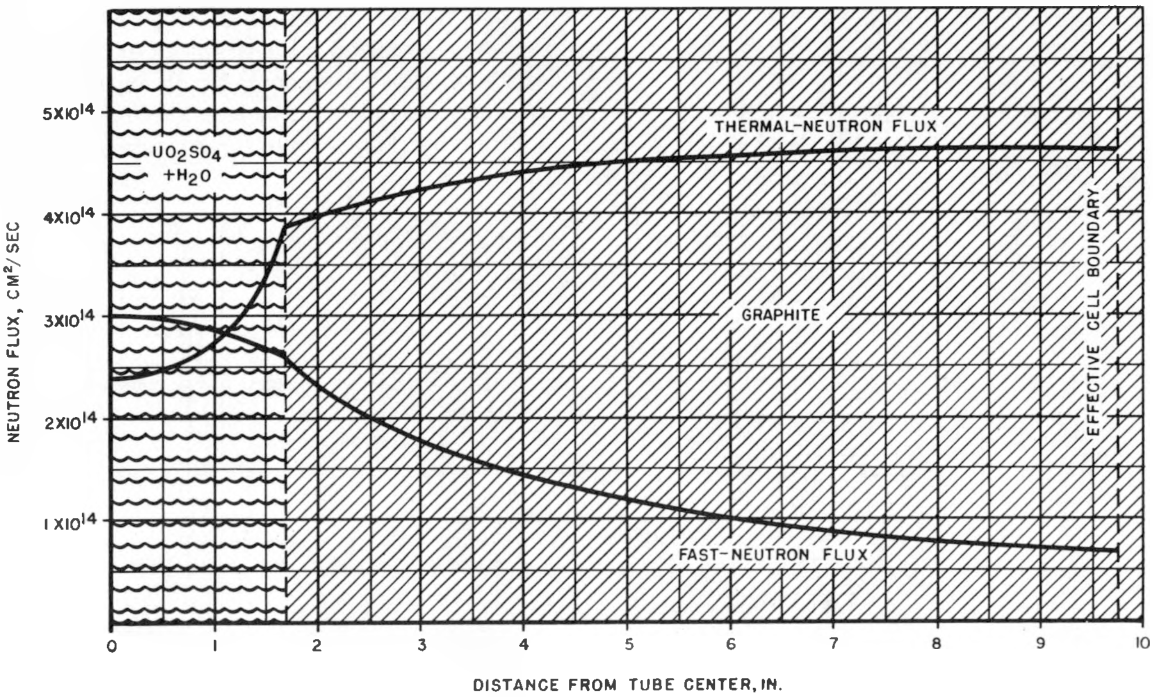


Fig. 2—Flux distribution for average core location.

heat is produced in the graphite. The closest approach of the wall temperature to the boiling point occurs at the hottest wall surface. Assuming that the system pressure at the suction side of the pump is atmospheric, the temperature at the hottest wall surface is 15°F below the boiling temperature. This margin could be extended to 48° by raising the system pressure by 1 atm.

3.2 Criticality Considerations External to the Reactor

Prevention of criticality of the fuel solution external to the reactor greatly influences the design of plenums and heat exchangers. It is suggested that borated graphite be employed in place of an axial reflector to better define the core boundary and to minimize the fission rate external to the core. Since a conventional tube header would, in itself, be critical, it is suggested that four manifolds be used at either end of the reactor together with four separate heat exchangers. For a manifold diameter of 8 in., a fluid velocity of 27 ft/sec is obtained. To be critical, an infinite cylinder of this diameter with infinite water reflection would require much greater fuel concentrations than might reasonably be anticipated.

To avoid the necessity of simultaneous concentration control in several parallel loops, the four-pass series arrangement of Fig. 3 is recommended. The fuel solution is alternatively heated and cooled four times before returning to the pump. The pressure drop through the reactor and heat exchangers is estimated to be 11 psi, and the total system head is about 41 psi. A 4200 gal/min pump of 100 hydraulic horsepower output would be required.

3.3 Experimental Facilities

The layout of Fig. 1 provides for the following experimental facilities: one 16- by 16-in. removable graphite stringer; eight 8- by 8-in. removable graphite stringers; six 4- by 8-in. removable graphite stringers; and up to forty-eight 4- by 4-in. beam tubes.

The graphite stringers would be in multiple sections and would be removable from either the ends or the sides of the reactor. The

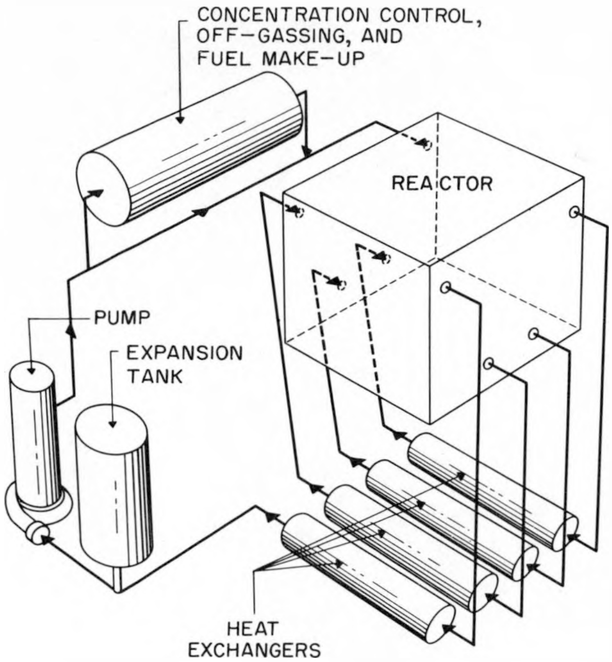


Fig. 3—Diagram of series flow arrangement.

through holes containing these stringers would normally be filled with graphite except for the length of slot which contains an experiment. A thermal column might be provided at the expense of removing some of the irradiation ports.

ABOUT THE AUTHORS

Richard G. Mallon received the B.S. degree in electrical engineering from Missouri School of Mines in 1951 and attended the 1951–1952 session of the Oak Ridge School of Reactor Technology. He has since worked as a nuclear engineer for Bendix Aviation Corp. Research Laboratories, and he is now with Advanced Scientific Techniques Research Associates, Milford, Conn.

William R. Pearce received the B.S. degree in mechanical engineering from Michigan College of Mining and Technology in 1951 and attended the 1951–1952 session of the Oak Ridge School of Reactor Technology. He has since worked as a nuclear engineer for Bendix Aviation Corp. Research Laboratories, and he is now with the Atomic Project of Monsanto Chemical Co., St. Louis.

REACTOR FUTURES

The Vapor-slurry Reactor

R. L. CROWTHER*

Institute for Atomic Research
and Department of Engineering
Iowa State College

ABSTRACT

This article describes a circulating-fuel reactor that utilizes a suspension of solid particles of fissionable material in a vapor as the coolant. The reactor core serves as the superheater in the vapor cycle. The coolant outlet temperatures from this reactor are not limited by pressure, and this allows utilization of the advantages resulting from the use of turbines designed for throttle inlet pressures and temperatures in the neighborhood of 1450 psig and 1000°F.

Several cycles that would be suitable for the vapor-slurry reactor design are discussed. A brief summary of a preliminary analysis conducted on a specific vapor-slurry reactor that utilized a water vapor—uranium dioxide fuel and a heterogeneous core is presented.

Two other possible reactor designs that utilize suspensions of fissionable solids in gases or vapors are described. These reactors were designated the "cyclone slurry reactor" and the "screw conveyer reactor." The cyclone slurry reactor operates similarly to the cyclone furnace. The screw conveyer reactor consists of a critical assembly of a heterogeneous array of screw conveyers.

1. INTRODUCTION

A study of the cycles that might be used to derive power from nuclear reactors indicated that possibly the concept that the reactor must replace the boiler in the steam cycle was invalid. A reactor that would act as the superheater in the cycle appeared to be feasible. Further analysis of this possibility led to a reactor design that was designated the "vapor-slurry reactor." The object of this report is to present this design as a possible future type of reactor.

2. DESCRIPTION

The vapor-slurry reactor utilizes a fuel consisting of solid particles of fissionable material suspended in a vapor. Steam appears to be the most suitable vapor for a thermal reactor, although the cycle is versatile and other vapor mediums might be used. A preliminary analysis was undertaken on this reactor, and a vapor slurry composed of H_2O vapor and UO_2 was

*Now with the Atomic Energy Division of The Babcock & Wilcox Co., Akron, Ohio.

chosen. A reactor-core design similar to that shown in Fig. 1 appeared to be desirable.

The flow diagram of the proposed reactor is shown in Fig. 2. The hot slurry particles and a required amount of recycled vapor are injected into the vaporizer. The hot particles and vapor vaporize the high-pressure liquid entering the vaporizer from the feed heaters and feed pumps. The slurry, which is a fissionable solid-saturated vapor mixture, passes from the vaporizer to the reactor and is heated by the resulting nuclear reaction. Thus, the reactor, which consists of a heterogeneous array of tubes surrounded by moderating material, serves to superheat the vapor and heat the slurry particles.

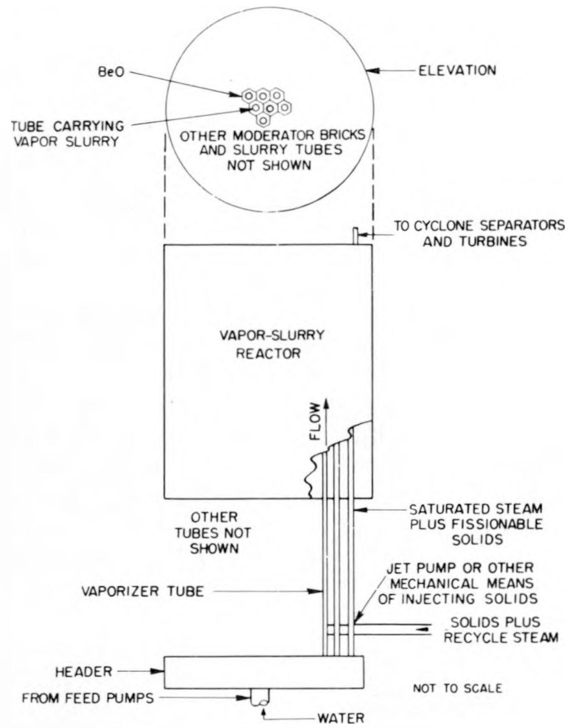


Fig. 1—Design for a possible reactor core and vaporizer.

At the reactor outlet, part of the slurry is recycled to the vaporizer, and the rest continues to the cyclone separators, which separate the solids from the vapor. The solids from the cyclone separators are then combined with the recycle slurry, and both return through the reheaters to the vaporizer where they act to vaporize water entering from the feed pumps.

The solids and recycle slurry are injected into the vaporizer by use of pressure differential, vibration, screw conveyance, or other similar means. Criticality is prevented in the circuits external to the reactor by the absence of moderator and by divided flow.

The separated vapor from the cyclones proceeds to the turbines and then to the condensers.

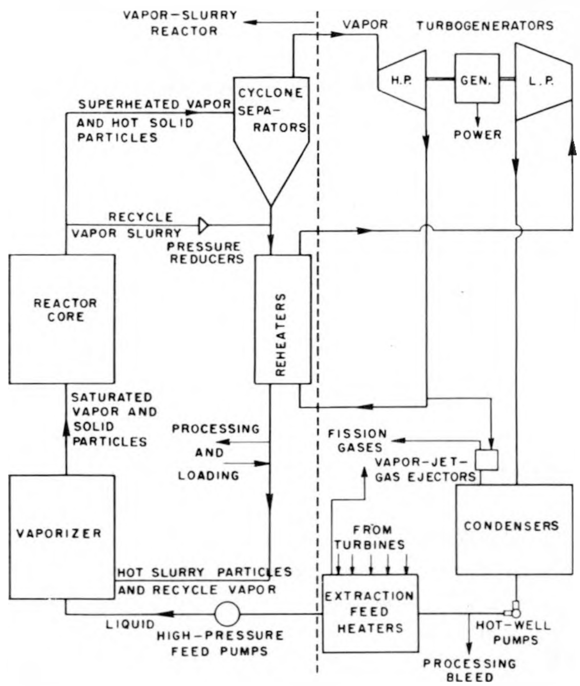


Fig. 2—The vapor-slurry reactor. The reactor serves as the superheater in the cycle.

Gases can be removed in the condensers by standard vapor-jet-gas ejectors. The condensate from the condensers is pumped through extraction feed heaters to the vaporizer. Additional gases that have been entrained in the condensate can be removed by use of one or more standard deaerating feed heaters.

Bleeds can be incorporated in the condensate and in the reheater sections of the cycle to allow for processing. The concentration of the slurry in the reactor can be controlled by varying the rate of injection of solids into the vaporizer.

It would be possible to operate the vaporizer as a heterogeneous boiling slurry reactor and thus eliminate the need for recycling part of the

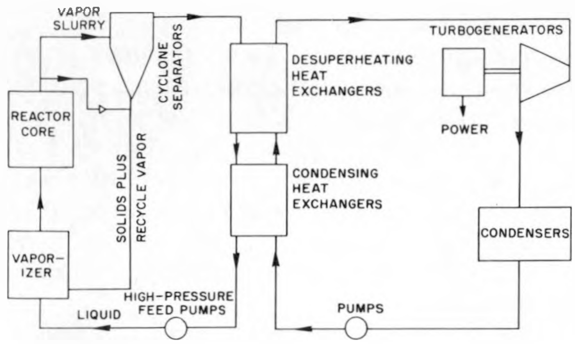


Fig. 3 — Alternate vapor-slurry cycle one.

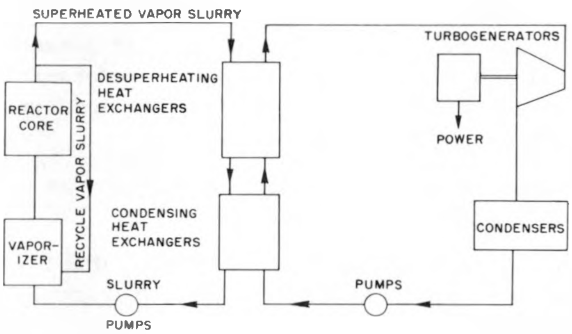


Fig. 4 — Alternate vapor-slurry cycle two.

vapor slurry. The large amount of structural material which will necessarily be present in the vaporizer and the possibility for nuclear instability in such a system would most probably rule out this proposal.

The preceding description assumes that radioactive vapor can be passed through the turbines and that high separation efficiencies can be achieved from the cyclone separators. Another possibility would be to pass the vapor or the slurry through desuperheating and condensing heat exchangers outside the reactor. Two versions of this alternative are illustrated in Figs. 3 and 4. It appears that the cycle shown in Fig. 3 would be more suitable because it eliminates the need for pumping a slurry in the heat-exchanger circuit. It also reduces problems of slurry caking in the heat exchangers and slurry erosion in the initial section of the vaporizer, which may be an array of high-velocity jet pumps fed by screw conveyers. On the other hand, with the system shown in Fig. 4 it is possible to use fluidized solid beds to achieve better heat transfer in the desuperheating heat exchangers. Both of these cycles could employ regenerative feed-water heating and reheat just as was proposed for the cycle shown in Fig. 2.

The cycle shown in Fig. 2 is preferable from the standpoint of thermal efficiency which could be achieved, but, if the cost is considered, the cycles of Figs. 3 and 4 appear to be more feasible.

3. FEASIBILITY OF USING SPECIFIC VAPOR SLURRY

A preliminary analysis was undertaken on the vapor-slurry reactor cycle shown in Fig. 2 to determine its feasibility. It was specified that a heterogeneous reactor-core design, similar to that shown in Fig. 1, would be used and that water vapor and uranium dioxide would be the vapor and solid components of the vapor slurry.

In the past, heterogeneous core arrangements for slurry reactors have been given little consideration. The heterogeneous type of slurry reactor appears to possess the following advantages over the homogeneous type:

1. There is less uncertainty in the flow and mechanical stability of the slurry. Turbulence induced by high-velocity flow through tubes may be used as a means of achieving mechanical stability. Matched densities, stabilizing agents, or complicated flow arrangements are not needed. Owing to high turbulence, no hot spots occur as a result of stagnant areas in the core, and there is lesser caking of slurry on structural materials.

2. There is versatility in the fuel-moderator-coolant arrangement, and the slurry need not be a good moderator.

3. Simplicity in design is possible, and there are less critical stress requirements owing to the possibility of the use of tubes to contain the slurry. Large pressure vessels with high stress concentrations due to entrance and exit holes and due to thermal stresses from absorption of radiation are not required.

4. There is less loss of neutrons by streaming owing to smaller holes entering and leaving the core.[1] This advantage would be more pronounced for a slurry that did not have moderating ability.

657 104

5. There is higher power density owing to greater velocities.

6. There is a possibility of providing for nonuniform fuel loading, thus achieving flat flux and minimum critical mass. It has been proved both experimentally and theoretically that flat flux is a criterion for minimum critical mass.[2,3]

7. There is a decrease in neutron resonance absorption owing to the geometry.

8. There is also a possibility of providing for a tube failure by simply plugging it and thus preventing lengthy shutdowns for repair.

The advantages of the homogeneous type of slurry reactor appear to be—fewer neutrons lost and thus lower enrichment required because of the use of less structural materials in the reactor core; less erosion of the structural materials and of the slurry particles because of lower velocities and less slurry contact with the structural materials; and less pumping power is required.

The vapor-slurry reactor cycle, shown in Fig. 2, which utilizes a steam-UO_2 slurry fuel, possesses several advantages over previous aqueous-fuel reactor designs.

1. The temperature and pressure at the reactor outlet are limited only by the maximum that slurry and tube materials can withstand. Previous reactor designs utilizing aqueous circulating fuels featured extremely high pressures and comparatively low temperatures in order to prevent the aqueous solution from flashing to steam. The maximum temperature of these designs was limited by the critical pressure of the water. The vapor-slurry cycle allows production of steam commensurate with modern steam power plant practice (1450 psig and 1000°F).

2. The absence of boiling in the reactor core eliminates instabilities caused by large and rapid bulk density changes and nonuniform fuel flow.

3. The slurry particles do not pass through the pumps. This alleviates many problems concerning pump erosion and bearings.

4. The intimate contact between the slurry particles and the carrying fluid allows for extremely rapid and efficient heat transfer both in the reactor and the vaporizer. Boiling-reactor designs with stationary fuel elements are prevented from attaining high superheat because of the poor heat transfer to superheated steam.

5. The heterogeneous flow arrangement in the reactor allows for high fuel velocities, resulting in high power density. It also allows for more accurate prediction of characteristics of fuel flow, resulting in elimination of instabilities that might occur if the reactor were one large tank.

6. The fuel distribution in the reactor may easily be made nonuniform to allow for minimum critical mass.

7. Operation at high temperatures, low pressures, and with reheat prevents turbine erosion due to "wet" vapor.

8. High temperatures and pressures result in high thermal efficiency and reduce the physical size of the turbines needed.

9. The cycle allows for the use of previously developed and standard turbines, condensers, and feed-water heaters with special provisions for sealing and maintenance.

Disadvantages and problems that may arise with this design are as follows:

1. Erosion of the tube materials, the cyclones, or the solid particles may cause serious difficulties.

2. Attrition of the slurry particles due to erosion or fission may prevent high separation efficiencies in the cyclone separators, resulting in fines getting to the turbines and condensers. A high pressure drop is required across the cyclones in order to achieve the necessary separation efficiency.

3. Highly radioactive steam entering the turbine cycle may present difficulties in sealing and maintaining the turbines, condensers, feedwater heaters, and pumps.

4. The entire cycle must be shielded, increasing the total plant weight and cost.

5. Large numbers of neutron-absorbing tubes in the core prevent the use of natural-uranium slurries.

6. Presence of fission gases and steam-soluble fission products may increase the corrosiveness of the steam in the turbine cycle.

7. Presence of too much noncondensable gas in the condenser may seriously reduce the heat-transfer coefficient.[4]

8. Caking of the particles on container materials might occur.

Solution of the above problems does not seem insurmountable. The hardness of UO_2 is reported to be from Moh 3.5 to 4, and therefore the erosion of container materials should not be serious if stainless steel is used throughout the system. Tube bends and other vulnerable spots can be lined with hard materials to prevent erosion.

The primary difficulty appears to be that of preventing particles from getting through the cyclone separators. Cyclone separators have been designed to operate at temperatures up to 1000°F, and collection efficiencies of 98 per cent have been achieved for separation of heavy concentrations of dust down to 2μ in diameter.[5–7] Extremely high separation efficiencies can be attained by arranging the separators in both parallel and series. Particle agglomeration has been achieved by the use of sound fields.[8] Stationary wave patterns set up by sound fields preceding the cyclones, in the cyclones, or between two cyclones in series could be used to agglomerate the fine fission or erosion particles and allow their separation. Subsequent turbulence in the reheaters or vaporizer would act to break up large agglomerates which might occur.

The turbines and their associated circuits will undoubtedly become radioactive because of the fission gases and soluble fission products. It appears that it might be feasible to submerge the turbines and condensers in water in floodable compartments. The water would act as a shield and would allow easy access for remote repair.

Continuous processing and continuous removal of gases will probably make the problems of additional corrosion and poor heat transfer in the condensers insignificant.

The cost of solving the preceding problems may make one of the alternative cycles illustrated in Figs. 3 and 4 more desirable than the cycle of Fig. 2. The difficulty of attaining high heat transfer in the desuperheating heat exchangers appears to be the chief problem. Finned tubes or other mechanical methods for achieving high heat transfer provide possible means of alleviating this difficulty.

In a preliminary design analysis of the cycle of Fig. 2, it was specified that the reactor would supply steam at 1450 psig and 1000°F to two 90,000- to 100,000-kw preferred standard turbine generators operating with 1000°F/1000°F reheat and five-stage extraction regenerative feed-water heating. The anslysis was based on

Table 1 — Summary of Preliminary Design

Heat power, 488,000 kw
Electrical power, 200,000 kw
Power density, 601 kw/liter
Per cent rated power, 110
Average flux, 5.83×10^{14} neutrons/cm^2/sec
Moderator, hexagonal BeO bricks
Enrichment, 16.67 per cent
Molecules of BeO per atom of U^{235}, 3860
Core shape, right-circular cylinder
Core diameter, 6.5 ft
Core height, 6 ft
Number of tubes, 877
Tube thickness, $\frac{1}{32}$ in.
Tube material, 347 stainless steel
Vapor slurry at core inlet, 1657 psia, 601°F
Vapor slurry at core outlet, 1600 psia, 1025°F
Core volume concentration, 2.5 per cent solids in slurry
Core flow rate, 24.45×10^6 lb of slurry per hour
Core outlet velocity, 100 ft/sec
Steam at throttle, 1450 psig, 1000°F
Throttle flow rate, 1.372×10^6 lb of steam per hour
Condenser pressure, 2.50 in. Hg absolute

a maximum condenser pressure of 2.50 in. Hg absolute with the turbogenerators operating at 10 per cent above rated load or 100,000 kw. It was assumed that a practical value for the volume concentration of the solids in the slurry flowing in the reactor core was 2.5 per cent. A velocity of 100 ft/sec at the core outlet was assumed. An approximate picture of reactor criticality was obtained by the use of modified Fermi age theory. The steam had an appreciable effect on the criticality of the reactor.

Reference 9 provides a detailed summary of the assumptions and the method of attack used in the analysis of the reactor. A summary of some of the factors of the preliminary design are presented in Table 1. This preliminary design indicates that the high power density and the high ratio of electrical power output of the turbogenerators to heat power which the reactor must produce are advantages of this type reactor.

4. REACTOR START-UP, SHUTDOWN, AND CONTROL

Several procedures are possible for starting the vapor-slurry reactor. One possibility would be to derive enough steam from an external source to operate the vaporizer. This is undesirable since it would result in appreciable extra plant cost and would be unsuited for mobile applications.

Possibly a more desirable method would be to operate the reactor as a heterogeneous boiling-slurry reactor until the desired power level was reached. The flow to the cyclone separators would be shut off until enough heat was generated to provide for saturated steam at the vaporizer outlet. The introduction of a water slurry instead of a steam slurry into the core would result in a smaller critical size. This could be compensated for by shutting off the flow of the slurry in certain tubes or by decreasing the concentration of the fissionable solids in the slurry. Experience would determine which of these was more favorable.

The reactor could be shut down by dumping the slurry particles into a series of tanks or into fluidized solid beds where the particles could be prevented from caking. The solids could be purged from the system by a high-pressure inert gas.

The vapor-slurry reactor could be controlled by varying the concentration of the solids in the slurry, by introducing neutron-absorbing gases such as xenon into the slurry, or by cutting off the flow of the slurry in certain reactor tubes. Various other possibilities, such as use of control rods, undoubtedly exist.

Calculations for the preliminary design pertaining to the cycle of Fig. 2 indicated that only about 15 per cent of the available delayed neutrons will aid in the control of the reactor. Investigation of the kinetics of circulating-fuel reactors has indicated that the negative temperature coefficient makes the reactor inherently stable even without the damping from delayed neutrons.[10] It has also been demonstrated that the circulation of the fuel causes damping of power oscillations in the reactor.[11] These analyses assumed that the inlet temperatures of the fuel remained fixed, whereas this may not be the case for the vapor-slurry reactor. Increases in temperature at the reactor outlet will be fed back to the reactor inlet. A control mechanism operating on the basis of the steam conditions at the vaporizer outlet and varying the concentration of the slurry would provide a method of controlling reactor power.

If the flow of steam to the turbines were decreased at the throttle, less steam would flow through the cyclone separators and more would be recycled to the vaporizer. The vaporizer would then produce superheated steam. If the negative temperature coefficient of the reactor were not great enough to decrease the reactor power sufficiently, the temperature at the reactor outlet would increase. Feedback of this temperature increase to the vaporizer would magnify the situation, and possible burn-up of the reactor could result. A dump valve preceding the turbine throttles could be used to alleviate this difficulty. When the throttle was closed, the dump valve would open, releasing steam to a tank or to the condensers and maintaining constant flow of steam through the cyclones. The dump valve could then be slowly closed, while the reactor control devices decreased reactor power.

A reactor emergency could be met by stopping the flow of solids and fluid into the vaporizer and purging the reactor core with a high-pressure inert gas.

A positive means of controlling the vapor-slurry reactor could be established, and control does not appear to be a limiting feature of the design.

5. OTHER POSSIBLE REACTOR DESIGNS

Consideration of two other reactor designs that utilize solids suspended in gases or vapors appears to be warranted. These designs were the cyclone slurry reactor and the screw conveyer reactor.

The cyclone slurry reactor, which would operate in a manner similar to that of the cyclone furnace, might prove feasible. A simplified diagram of one possible design of the core of this reactor is shown in Fig. 5. Solid particles of fissionable material are whirled at high velocity through the core by a tangential stream of secondary vapor. The solid particles are thrown to the outside of the core, where they form a dense, rapidly moving layer in which fission can take place. If the reactor is thermal, a moderator island could be introduced to the center of the core to aid in neutron moderation.

If a vapor were used to transport the solids, the cyclone slurry reactor core could be adapted to one of the cycles shown in Figs. 2 to 4. If an inert gas were used, one of the cycles shown in Figs. 6 or 7 could be utilized.

The cyclone slurry reactor has the advantages of simplicity and high power density. The cyclone furnace has been adapted[12] to steam power plants operating at supercritical conditions of 5000 psia and 1100°F, and it seems possible that this reactor might also be suitable for such use. The primary difficulties to be solved are erosion and possible nuclear instability due to rapid density fluctuations in the whirling cyclone. The slurry particles would be eroded very rapidly, and experiments would be needed to determine what limitations this would place on this design.

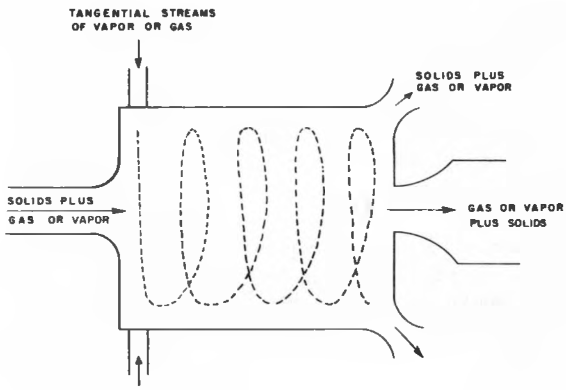


Fig. 5 — Cyclone slurry reactor.

Another possible reactor design consists of a heterogeneous array of screw conveyers moving solid particles of fissionable material through a critical assembly. The solids are carried external to the reactor by gas or vapor streams, and one of the cycles shown in Figs. 2 to 7 would be used to produce power. The gas or vapor would be used to deposit and remove the solids from the outlet and inlet of the reactor, and large quantities of vapor or gas would not flow through the reactor with the solids.

If the reactor were thermal, serious limitations would be placed on the materials that were used for the screws. Zirconium appears to be a possible screw material. However, zirconium is rapidly corroded by high-temperature steam, and this factor would have to be considered if water vapor were used to transport the solids. The screws can be cooled by making them hollow

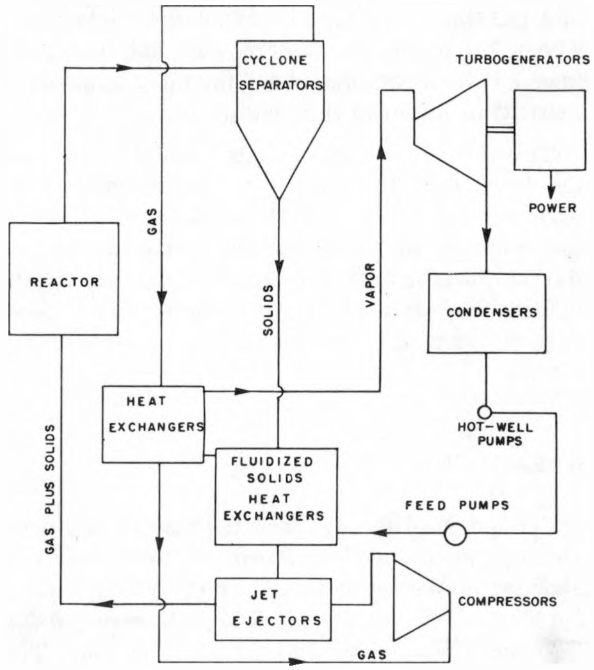


Fig. 6 — Scheme for the utilization of the Rankine cycle to derive power from a gaseous-suspension reactor.

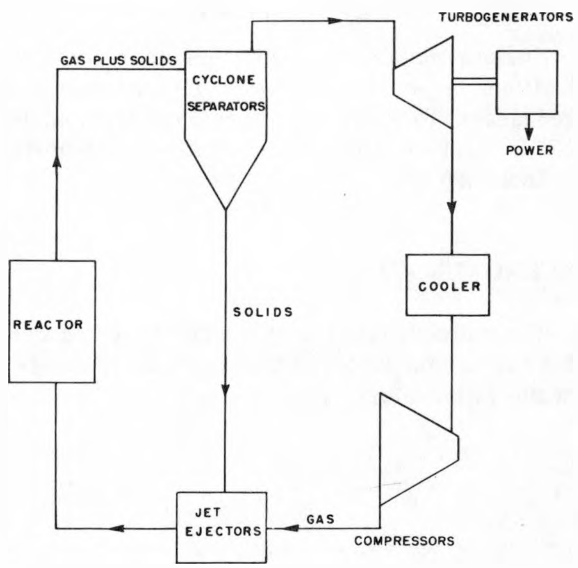


Fig. 7 — Scheme for the utilization of the Brayton cycle to derive power from a gaseous-suspension reactor.

and passing a coolant through their interiors. The solid particles must not be hot enough to fuse, and, therefore, heat transfer may be a critical problem in this design.

The screw conveyer reactor design possesses the important advantages of little erosion and high concentrations of fissionable material in the reactor core. The rate of conveyance of the solids through the core could be easily varied in each of the different screw conveyers, and the reactor design is inherently simple.

6. FAST REACTOR

The possibility of operating any of the previously described reactors as fast reactors should not be neglected. For the vapor-slurry cycle a slurry of fissionable solids suspended in mercury vapor would be a possible fuel. The fast reactor is plagued by the difficulty of removing heat from the core, and a slurry utilizing a nonmoderating vapor or gas provides an adequate solution to the problem. The previously mentioned screw conveyer reactor might prove to be especially adapted for operation as a fast reactor because of the heavy concentrations of fissionable material which can be conveyed through the core by the screws.

Nuclear instability due to fuel-geometry fluctuations is possibly a major factor which it is necessary to consider when investigating the use of fast reactors incorporating the features of these designs.

ACKNOWLEDGMENT

The author wishes to thank Dr. Glenn Murphy for his helpful advice on the preliminary analysis of the vapor-slurry reactor.

REFERENCES

1. D. J. Behrens, The Effect of Holes in a Reacting Material on the Passage of Neutrons, with Special Reference to the Critical Dimensions of a Reactor, Report AERE-T/R-103, May 24, 1949.
2. John W. Morfitt, Minimum Critical Mass and Uniform Thermal-neutron Core Flux in an Experimental Reactor, Reactor Sci. Technol., 4(4): 9 (December 1954).
3. Gerald Goertzel, Minimum Critical Mass and Flat Flux, Reactor Sci. Technol., 2(1): 19 (April 1952).
4. W. H. McAdams, "Heat Transmission," 3d ed., p. 342, McGraw-Hill Book Company, Inc., New York, 1954.
5. C. E. Lapple, Fluid and Particle Mechanics, University of Delaware, March 1951.
6. J. M. Dalla Valle, "Micrometrics," 2d ed., Pitman Publishing Corp., New York, 1948.
7. J. M. Dalla Valle, Dust Collector Costs, Chem. Eng., 60: 177 (November 1953).
8. F. J. Burger and K. Sollnar, The Action of Ultrasonic Waves in Suspensions, Trans. Faraday Soc., 32: 616, 1958 (1936).
9. R. L. Crowther and Glenn Murphy, Slurry Reactors, Master's thesis to be published as Iowa State College (ISC) report.
10. S. Tamor, Note on the Nonlinear Kinetics of Circulating-fuel Reactors, Report Y-F10-109, Aug. 15, 1952.
11. W. K. Ergen, The Kinetics of the Circulating-fuel Nuclear Reactor, Report AECD-3526, Mar. 30, 1953.
12. W. H. Rowand, Developing the First Commercial Supercritical Steam Generator, Power, 98: 73 (September 1954).

ABOUT THE AUTHOR

R. L. Crowther received the B.S. degree in mechanical engineering from Iowa State College in 1953 and the M.S. degree in nuclear engineering in 1955. During this time he was employed as a part-time research assistant in the Engineering Development Group at Ames Laboratory. He is now employed with the Atomic Energy Division of The Babcock & Wilcox Co.