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DESIGN OF THE ARGONNE LOW POWER REACTOR (ALPR)

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DISCLAIMER

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ABSTRACT

This report describes the detailed design of a prototype "packaged" nuclear power plant constructed at NRTS, Idaho, during the period July, 1957 to July, 1958. The facility was transferred to a plant-operating contractor on February 5, 1959. The purpose of the plant is to alleviate fuel oil logistics and storage problems posed by remote auxiliary DEW Line radar stations north of the Arctic Circle. The ALPR (redesignated SL-1) is a 3-Mwt, heterogeneous, highly enriched uranium-fueled, natural-circulation boiling water reactor, cooled and moderated with light water. Steam at 300 psig, dry and saturated (421°F) is passed directly from the reactor to a conventional turbine-generator to produce electric power (300 kw nominal) and space-heating (400 kw) requirements consistent with rigid mechanical and structural specifications prescribed by the military, and dictated by the extreme geophysics prevailing at the ultimate site. The over-all design criteria emphasize: simplicity and reliability of operation and maintenance, with minimum supervision; minimum on-site construction; maximum use of standard components; limited water supply; utilization of local gravel for biological shielding; transportability by air lift; and nominal 3-year fuel operating lifetime per core loading. The "packaged" concept is incorporated for the initial erection. The plant is not designed for relocation.

The design criteria for the prototype necessitate special features. The fuel plates are clad with a newly developed aluminum-nickel alloy (X8001). Burnable-poison (B^{10}) strips are mechanically attached to the fuel assemblies to compensate the excess reactivity required for a nominal 3-year core operating lifetime. The control rods are actuated by rack-and-pinion drive extensions which incorporate rotary seals. Fuel exchange is accomplished without the removal of the pressure vessel head. The electrical power generated is used to operate plant auxiliaries; the "net electric power" is dissipated by resistors. The hot water for space heating is heated in a heat exchanger by 20-psig steam, use being made of the latent heat of vaporization, and all the heat is dissipated by a finned-tube, air-cooled heat exchanger.

Devoid of features peculiar to the immediate application, ALPR is adaptable for use in remote, limited access areas where natural fuels are available only at a very high cost.

DESIGN OF THE ARGONNE LOW POWER REACTOR (ALPR)

by

Eberhard E. Hamer *et al*.

I. INTRODUCTION*

In 1954, the Army Reactors Branch, Division of Reactor Development, AEC, was apprised of a requirement for nuclear power plants to replace diesel generators and boilers as the primary source of electrical power and space heat at remote auxiliary DEW Line radar stations north of the Arctic Circle. Designs for low-power nuclear plants were solicited by the Army Reactors Branch, to which the Laboratory responded.[1,2]

In late 1955, the Laboratory was authorized to design, construct, and test the prototype Argonne Low Power Reactor[3,6,7] as an ultimate member of a family of nuclear power plants to produce electric power and heat at remote military bases. The prototype design was to incorporate data and experience gained from the series of BORAX experiments at NRTS, Idaho, and to be consistent with associated problems of arctic construction, operations, maintenance, economics, and logistics. The issued specifications, with amendments, are reproduced as Appendix V. In brief, the major design objectives were:

(1) Transport by airlift. The reactor components were limited to "packages" measuring 7.5 x 9 x 20 ft, and weighing 20,000 lb.

(2) Maximum use of standard components.

(3) Minimum on-site construction.

(4) Construction on permafrost (tundra) terrain.

(5) Noninterference with radar signals.

(6) Use of local gravel for biological shielding.

(7) Minimum quantity of plant makeup water.

(8) Simplicity and reliability of plant operation and maintenance, with minimum supervision.

(9) Nominal 3-year fuel operating lifetime per core loading.

The prototype design evolved (Fig. 1) is a 3-Mwt, highly enriched uranium-fueled, direct cycle, natural-circulation boiling water reactor, cooled and moderated with light water. Steam at 300 psig, dry and saturated (421°F) is passed from the reactor to a conventional turbine-generator to produce electric power (300 kw nominal) and space heating (400 kw) required at the smaller auxiliary DEW line radar stations. The pertinent design characteristics are summarized in Tables 1 (see p. 39) and 2 (see p. 43).

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*M. Treshow

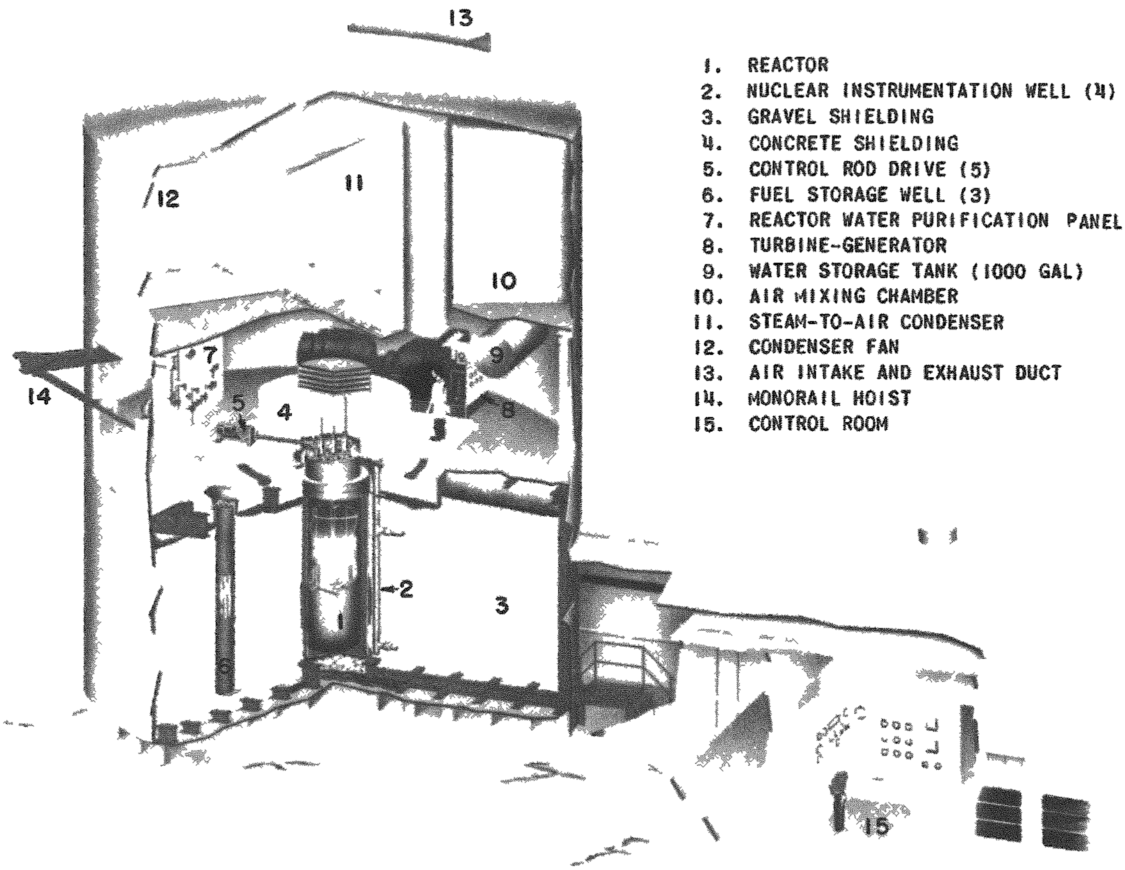

FIG. 1

SCHEMATIC OF ARGONNE LOW POWER REACTOR (ALPR) PLANT

Figure 2 is a photograph of a main DEW line radar station. It is the largest station in the DEW line network and requires more power output than is generated by an ALPR-type power plant. The inset shows the relative position of a proposed ALPR-type plant installation at a typical auxiliary radar station.

The electric power requirements of an auxiliary station are currently supplied by diesel-generator units. The heat requirements are supplied by hot water heated by the diesel engine exhaust gases and by boilers. The nuclear plant will not replace the existing electrical power and space-heating plants, since some standby capacity must be maintained; however, it will alleviate the fuel oil logistic problem.

In order not to limit future plant installations to one geographic location, the specifications for the prototype plant encompass the most stringent conditions imposed on the radar stations for which nuclear power is intended.

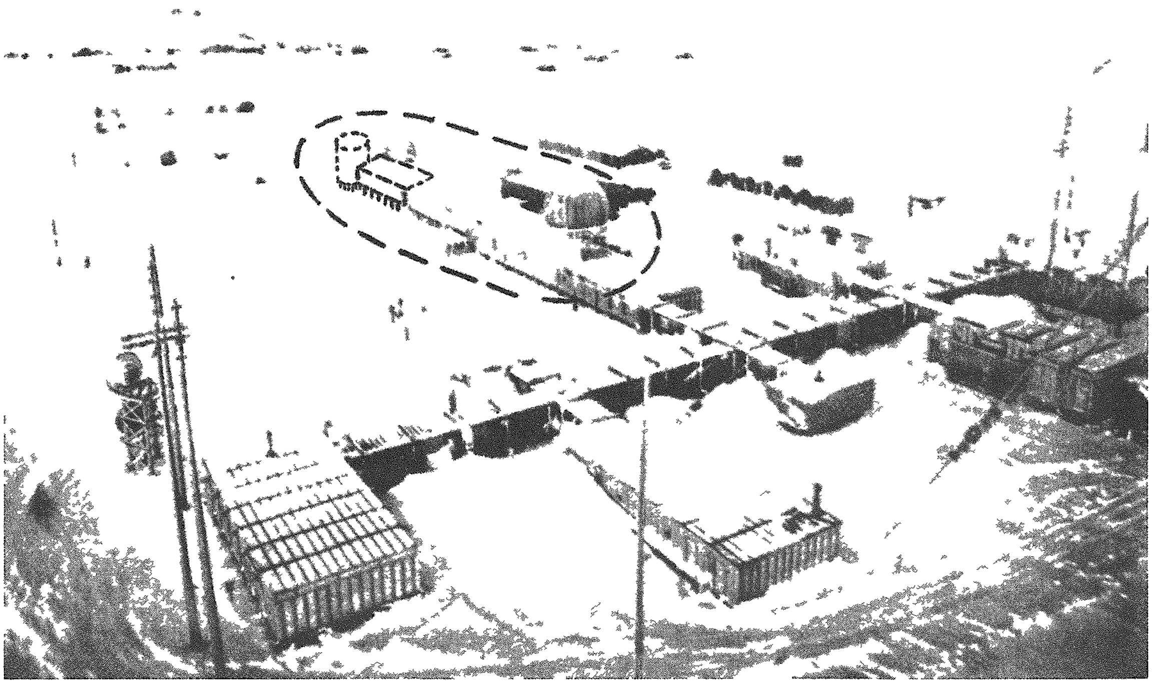

FIG. 2
MAIN DEW LINE RADAR STATION. INSET SHOWS RELATIVE POSITION OF PROPOSED ALPR-TYPE PLANT INSTALLATION AT AN AUXILIARY STATION.

1. Arctic Construction

The most adverse terrain in the arctic is permafrost (tundra), which precludes excavation, but on which pile construction has been proved acceptable. The prototype is constructed on concrete piers to simulate piles. In other locations, the plant can be set on rock or in an excavated area.

Owing to the wind velocities (up to 125 mph), the power plant building shell is circular and separate from the internal structure. This type of construction eliminates shell bracing and allows the building to "float" on the foundation piers. The condenser air-distribution inlet and exhaust ducts are located on the same side of the building and perpendicular to the prevailing winds. This orientation avoids any differential pressures in the air-distribution system due to outside disturbances.

The range of ambient air temperature for design in the arctic is -60°F to +60°F. The temperature range causes large linear expansion of the building shell. The "floating" characteristic of the shell allows for vertical expansion, and the anchoring of the shell to the center foundation piers allows for radial expansion. The severe cold also presents a problem with respect to plant component and fluid freezing. Therefore, all cold air drawn into the building by the condenser air-distribution and building ventilation systems is mixed with recirculated warm air to maintain operating air temperatures at 40°F, or higher.

2. Isolation of Reactor Plant from Radar Operations

The specified noninterference with radar operations resulted in the design of an all-welded steel building with a height limited to 50 ft above grade (corresponding to the bottom of the radar antenna). Grid screens are installed in the inlet and exhaust air ducts to complete the isolation of the plant from the radar beam.

The radar equipment requires a power supply with high-quality frequency and voltage control. Accordingly, radar electrical power is supplied by a separate system isolated from any transients induced by motor-starting currents or equivalent. In addition, the turbine speed is controlled by a sensitive, fast-load response, governor unit of the type used with calendar drives in paper mills.

3. Reactor Shielding

The ALPR features the use of locally available gravel for the bulk biological shielding. (Gravel of the type used in the prototype is found near the radar stations.) This feature serves to reduce the need for transporting weighty shielding constituents (e.g., cement) and is consistent with the prescribed airlift concept of plant mobility.

The gravel is conveniently contained within the circular building shell and serves to shield other auxiliary components, i.e., spent fuel storage wells, resin columns, and storage tanks.

4. Steam Plant Condenser System

Cooling by river or sea water was assumed to be unavailable. Therefore, variations of two alternative condenser systems were studied: an air-glycol-steam system, and an air-steam system (hereafter referred to as the air-cooled condenser system). The selection of the air-cooled condenser system was based on engineering considerations, since cost estimates of each system did not differ significantly. The disadvantages associated with the air-glycol system were: potential leakage of glycol into the reactor system, use of special low-temperature materials, construction of a larger plant, and higher steam condenser back pressure during the summer months.

It was recognized that the operating characteristics of an air-cooled condenser were unknown; however, the potentiality of a more compact and efficient plant could not be disregarded.

5. Reactor Plant Control System

A reliable automatic control system was devised, consistent with the prescribed minimum plant supervision and maintenance. A change in the plant power demand affects the steam pressure. Pressure deviations from a set point and outside of a dead band are sensed by an instrument which then transmits a signal to initiate insertion or withdrawal of reactor control rods. Steam bypass control is not used for normal plant operation.

6. Special Features

The prototype plant includes certain features designed to increase its utility:

a. The fuel elements are clad with an aluminum-nickel alloy (X8001) originally developed by the Laboratory.

b. Burnable-poison (boron-10) strips are mechanically attached to the fuel assemblies to compensate the excess reactivity required to ensure a core operating lifetime of ~3 yr.

The National Reactor Testing Station, near Idaho Falls, Idaho, was selected as the site for the prototype plant. Construction of the site and plant started in July, 1957. Certain structural deviations were allowed in view of the climatic differences between the Idaho site and the ultimate arctic site. For example, the quality of the structural steel was downgraded, and the thickness of the insulation was reduced. Also, the arctic building train "module" was simulated, in part: the Site Support Facilities Building is positioned relative to the power plant and the control room position is confined to the space equivalent in a train "module."

The Laboratory acknowledged custody of the facility on July 3, 1958. A program involving reactor-component installation, initial testing, operation, and training of military cadre was initiated. On August 11, the reactor achieved criticality with a minimum unpoisoned fuel loading. Electrical power and space heat was generated on October 24. The program was completed and the facility transferred to a plant operating contractor on February 5, 1959.

The final construction cost was $1,716,919.24, of which $1,242,125.43 is chargeable to the nuclear power plant, and $464,793.81 for the supporting facilities.

The broad scope of the ALPR Project makes it difficult to acknowledge the names and specific contributions rendered by all personnel engaged during the various stages of design, development, construction, and initial operation of the ALPR. The Reactor Engineering Division of the Laboratory was assigned primary responsibility for the project. Support was rendered by other Laboratory Divisions. The following Laboratory Staff personnel participated in the denoted areas of design, reactor-component development, and/or fabrication or procurement phases, construction, and operation. Also specific responsibilities in the preparation of this report are indicated by the authorship footnoted at the beginning of the various sections and appendices.

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C. R. Braun[c]	Assistant Division Director; Construction and Operations Project Engineer
C. R. Breden	Reactor water chemistry and corrosion testing
C. F. Bullinger[d]	Reactor control rods and drives
A. P. Gavin	Reactor water chemistry
N. R. Grant	Corrosion testing
E. E. Hamer	Engineer-in-Charge of mechanical design, construction, and operation; Resident Field Engineer; Chief Loader; Alternate Chief Operator.
J. M. Harrer	Nuclear instrumentation and reactor control systems
A. Hirsch	Reactor control systems; Operations Engineering
H. H. Hooker	Engineer-in-charge of instrumentation and electrical design, construction, and operation; Resident Field Engineer; Alternate Operations Coordinator; Alternate Chief Operator.
G. L. Jorgensen	Plant structural engineer
W. J. Kann	Reactor control rods and drives; Operations Engineering Staff

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[b]Currently with Combustion Engineering, Inc.
[c]Currently with Allis-Chalmers Mfg. Co.
[d]Currently with Advance Technology Laboratories, Division of American Standard

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A. L. London[a]	Reactor heat transfer
P. A. Lottes	Reactor heat transfer
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G. C. Milak	Reactor core; Operations Engineering Staff
H. Pearlman[b]	Fuel element design and corrosion studies
R. L. Ramp	Nuclear instrumentation
A. D. Rossin	Reactor shielding; Alternate Chief Physicist
D. H. Shaftman	Chief Physicist
A. Smaardyk	Special plant components; Operations Engineering Staff
M. Treshow[c]	Design Project Engineer
F. Verber	Power generation and distribution
R. J. Weatherhead	Reactor heat transfer

In addition, Design Draftsman F. J. Simanonis, Chief Draftsman J. J. Shimkus, and other members of the division Drafting Department contributed significantly to the project. Personnel in the division Scheduling Section and the Secretarial Section were particularly helpful in expediting the work.

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P. Stoddart	Operations Health Physics
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[a]On loan from Stanford University

[b]On loan from Atomics International, Division of North American Aviation, Inc.

[c]Currently with General Atomics Division, General Dynamics Corp.

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J. F. Schumar[a]	Associate Division Director
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W. R. Burt	Alloy core and fuel plate rolling
J. E. Draley	Aluminum-nickel alloy development
W. J. McGonnagle	Non-destructive testing of fuel plate
R. A. Noland	Fuel plate design and fabrication
W. E. Ruther	Aluminum-nickel alloy development
R. L. Salley	Alloy core and fuel plate rolling
D. E. Walker	Fuel plate design and fabrication

Central Shops

H. V. Ross[b]	Superintendent
G. M. Lobell	Planning Engineer
L. E. Wright[c]	Planning Engineer Supervisor
C. T. Zymko	Welding Specialist

Contractors

Under the direction of the Chicago Operations Office, AEC; the Idaho Operations Office, AEC; or the Argonne National Laboratory, several private companies contributed substantially to the design and construction of the ALPR facility. Some of these are as follows:

Pioneer Service & Engineering Company Chicago, Illinois	Architect-Engineer
Fegles Construction Company, Inc. Minneapolis, Minnesota	General Construction Contractor

[a]Currently with General Atomics Division, General Dynamics Corp.
[b]Currently with Combustion Engineering, Inc.
[c]Currently with General Atomics Division, General Dynamics Corp.

Alco Products, Inc. Schenectady, New York	Control Rod Drives
Bogue Electric Manufacturing Company Patterson, New Jersey	Emergency Power Supply
Chicago Bridge & Iron Company Chicago, Illinois	Reactor Building Shell and Structural Steel
Illinois Water Treatment Company Rockford, Illinois	Reactor Water Purification System
Ingersoll-Rand Company Chicago, Illinois	Feed-water Pumps
Lasker Boiler & Engineering Corporation Chicago, Illinois	Pressure Vessel and Support Cylinder
Leeds and Northrup Company Philadelphia, Pennsylvania	Process and Neutron Instrumentation
Modine Manufacturing Company Racine, Wisconsin	Air-cooled Condensers
Nelson Electric Company Tulsa, Oklahoma	Metal-Enclosed Low-Voltage Switchgear
Worthington Corporation Wellsville, New York	Turbine-Generator Unit

Many other organizations contributed to the project as suppliers of equipment, or as subcontractors, to one or more of the aforementioned companies.

Atomic Energy Commission

The close cooperation of the Army Reactors Branch, Division of Reactor Development, for whom the plant was designed and constructed; the Chicago Operations Office, who administered the design phase of the project; and the Idaho Operations Office, who administered the construction contracts, was a major factor in attainment of the program objectives.

II. SITE AND BUILDINGS*

A. Site

The prototype ALPR facility is located at the National Reactor Testing Station, Idaho (see Fig. 3). The geological structure at the reactor site consists of lava rock covered with a few feet of sedimentary rock and clay overburden. No large cracks or voids were found in the lava rock during preliminary soil tests (core drillings) or subsequent excavation. No major drainage or clearing problems were encountered; however, the lava rock did cause extra effort during both excavation and trenching of pipe and conduit runs.

B. Buildings

The prototype ALPR facility occupies an area approximately 350 ft square surrounded by site security fencing (see Fig. 4). The facility is self-supporting and is limited to plant operation, experimentation, and routine maintenance.

The structures include a pump house, a 50,000-gal. water storage tank, a water and sewage chlorination house, a guard house, a support facilities building (containing the reactor control room), and the power plant building which houses the reactor and power-generating plant.

1. Power Plant Building

a. Foundation

The Power Plant Building is the only structure designed to simulate construction in an Arctic "permafrost region."[16] As shown in Fig. 5(a), the building is supported 2 ft above grade level on short concrete piers embedded in a reinforced concrete slab. The air space between the building bottom and grade level serves to: (1) reduce heat transfer to the ground, which would cause deep thawing of the permafrost; and (2) alleviate the problem of snow drifts.

The central nine piers are capped and welded to the bottom of the building shell. The outer piers are capped with graphited-bronze bearing plates to permit radial movement of the building induced by temperature changes.

In a permafrost area the concrete piers would be replaced by wood piles set in the permafrost [see Fig. 5(b)]. The bearing plates would be replaced with caps welded to the building shell, allowing the piles to compensate for radial movement of the building.

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*G. L. Jorgensen

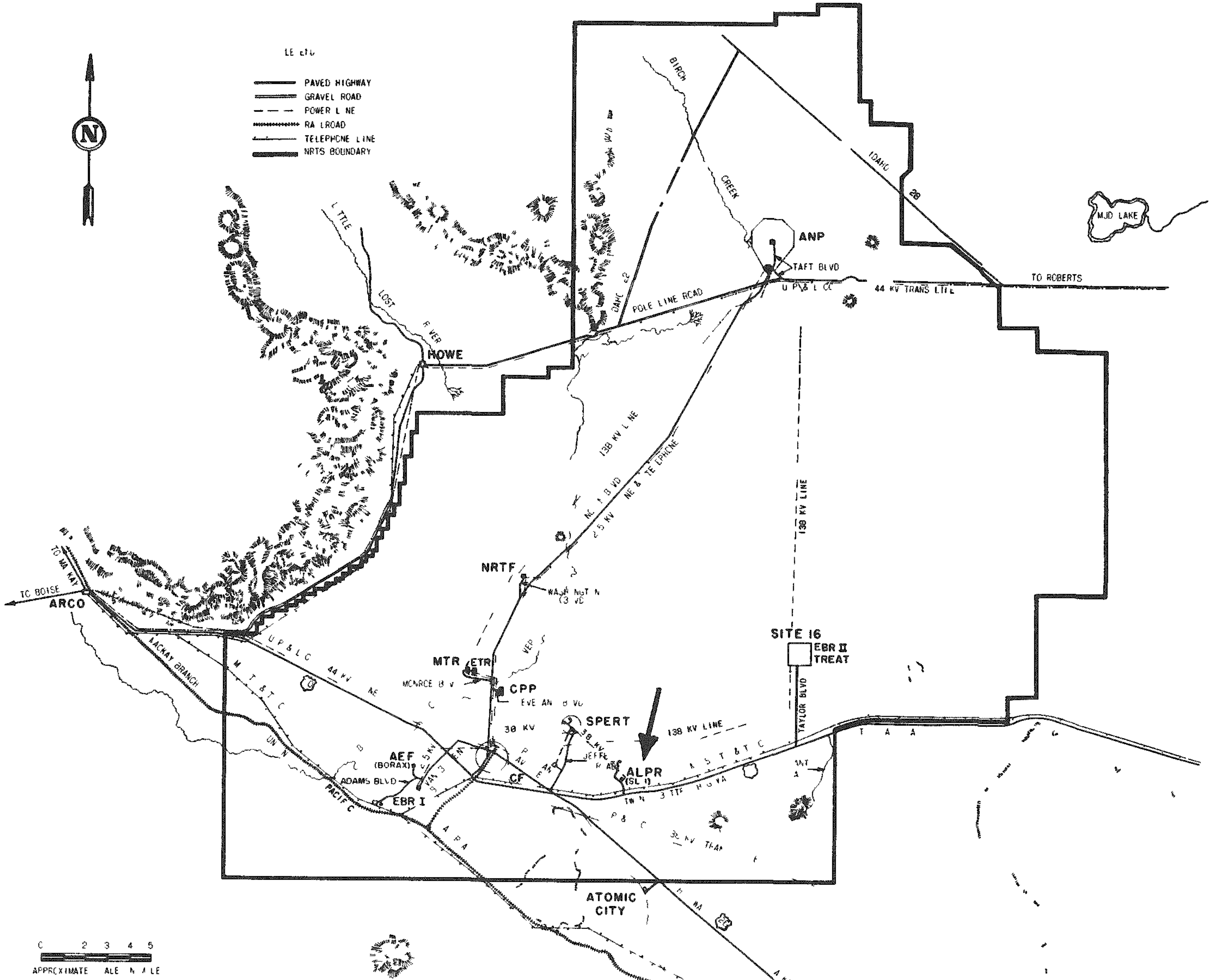
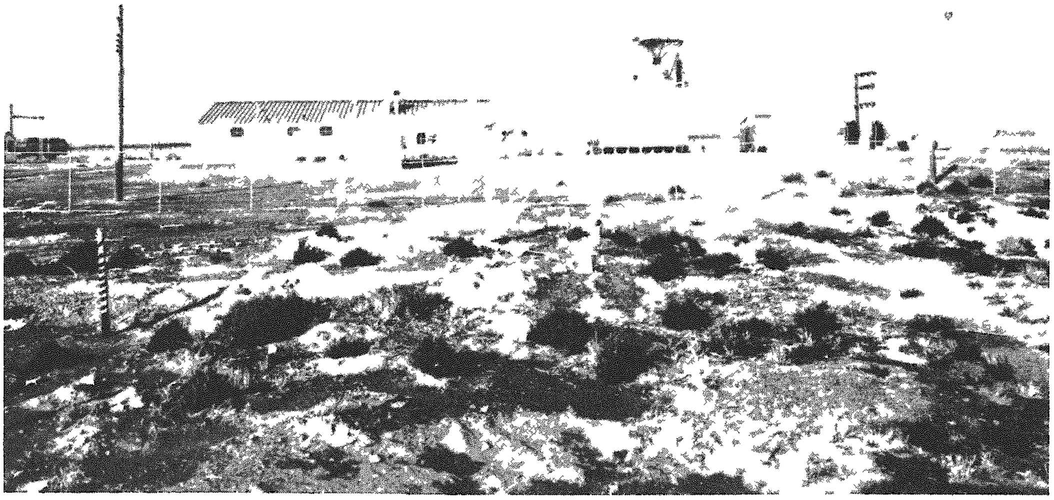


FIG. 3
NATIONAL REACTOR TESTING STATION



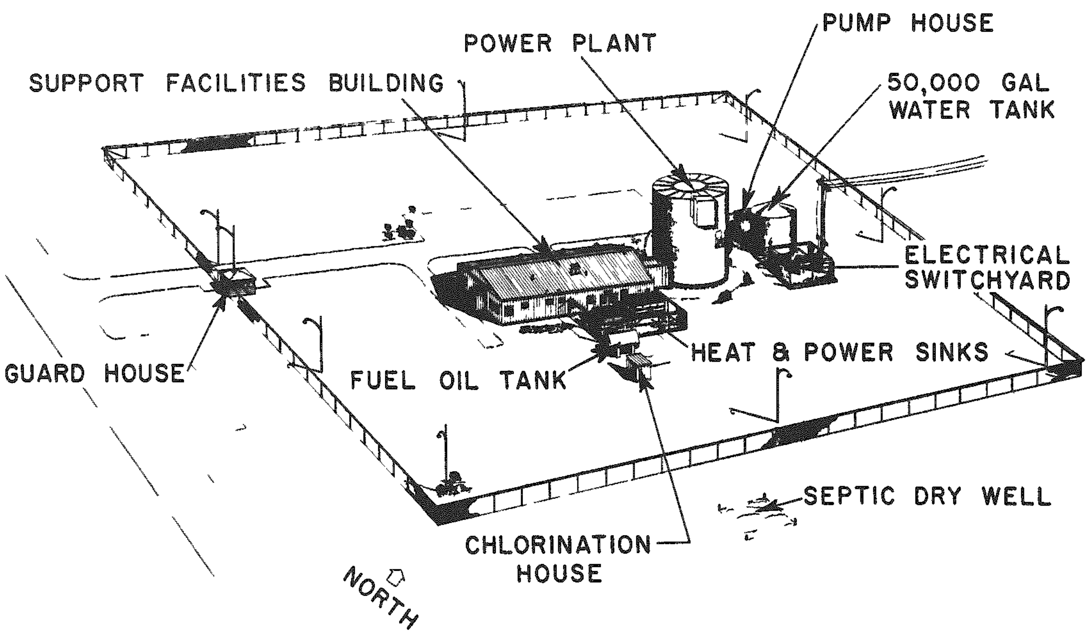
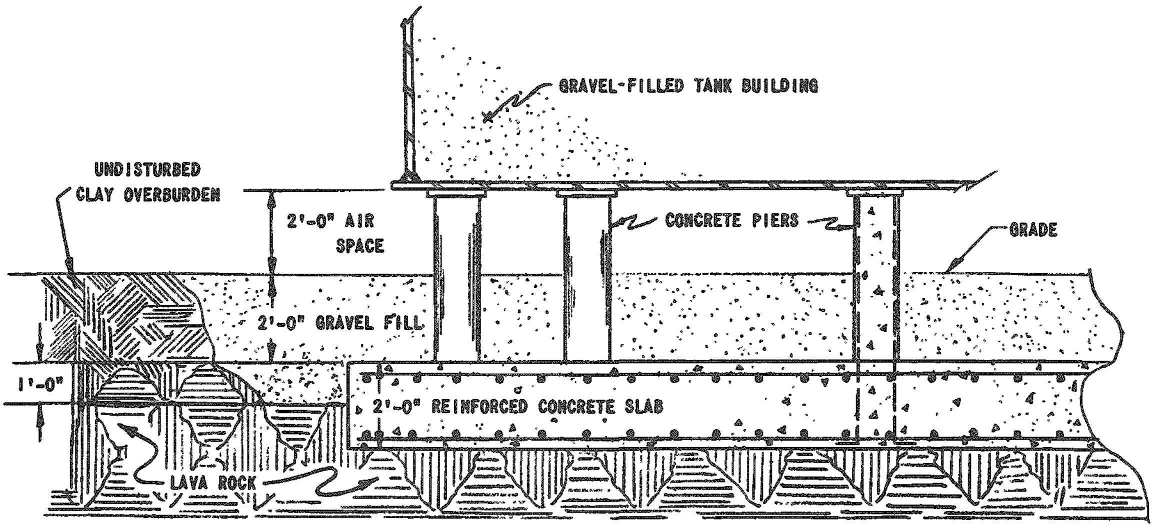
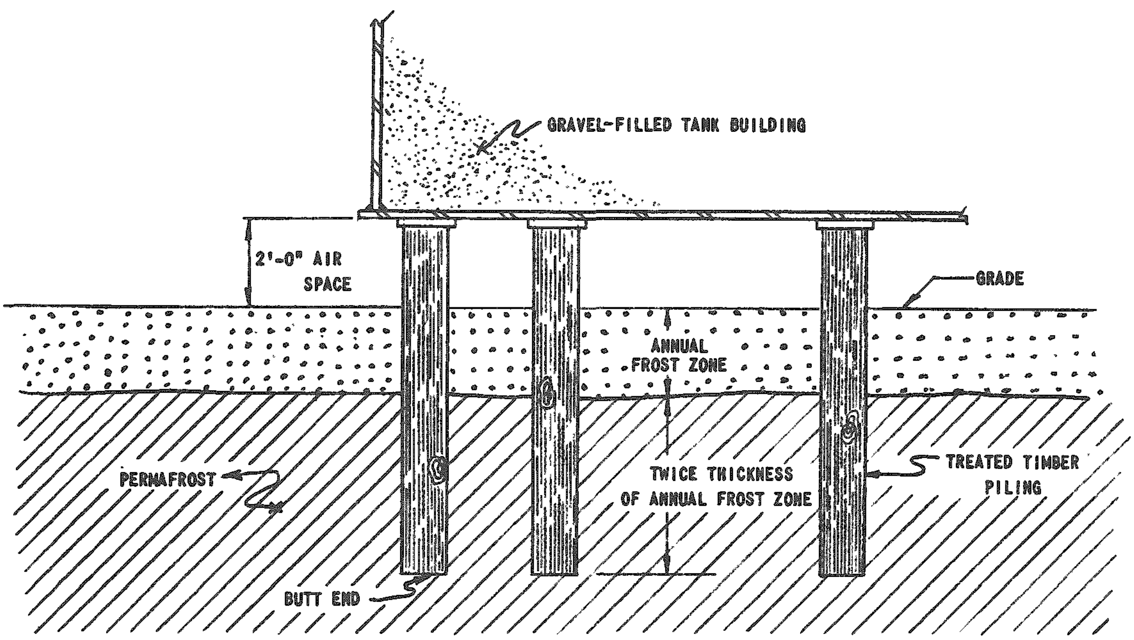


FIG. 4
ARGONNE LOW POWER REACTOR SITE



(a) FOUNDATION AT IDAHO SITE



(b) TYPICAL FOUNDATION IN PERMAFROST AREA

FIG. 5

COMPARISON BETWEEN FOUNDATION FOR PROTOTYPE POWER PLANT BUILDING AND PROPOSED FOUNDATION IN PERMAFROST AREA.

b. Building Shell

The building shell is a Structural Grade A-283-D steel tank (38 ft 7 in. in dia and 48 ft high), with a conical roof and a flat bottom. The diameter of the building represents the minimum size compatible with cost, reactor shielding, and operating floor-equipment space to reduce thawing of the permafrost and drifting of the snow.

The top of the tank is 50 ft above grade, on a plane corresponding to the bottom of a radar station antenna floor. The roof is fabricated of $\frac{3}{16}$-in. steel plate and is reinforced with structural steel angles. The tank walls are $\frac{1}{4}$-in. plates, and are increased to a thickness of $\frac{5}{16}$ in. at a point 8 ft above the tank bottom. The bottom plate is $\frac{3}{4}$ in. thick.

All joints between plates are continuous welded to prevent radar signal interference that might be incurred by gaps in the tank walls, or by internal electrical equipment exposed to the radar beam at the parent site.

Site conditions and economy were the criteria for selection of Structural Grade A-283-D steel plate for the prototype Power Plant Building. However, the conditions prevailing at ultimate Arctic sites (temperature: +60°F to -60°F; wind speed: 125 mph) may necessitate the use of a high-stress steel, i.e., Carilloy T-1,[a] for certain portions of the steel tank. One portion, in particular, is the lower tank wall around the gravel shield. The continuous settling of the gravel during periods of maximum expansion of the tank, and the resistance of the gravel to subsequent contraction of the tank may promote stresses in excess of specified values for Grade A-238-D steel at -60°F. One solution designed to reduce the stress buildup would be to insulate the exterior of the tank wall in this area.

Exclusive of the area surrounding the gravel shield, the structural design meets the prescribed building specifications.

c. Interior Structure

The building shell encircles, but is independent of, a bolted, structural steel framework. This framework supports the main operating floor, the fan floor, and the power plant equipment. The bottom of the framework comprises a network of steel beams welded to the tank bottom plate to distribute the loads over the foundation piers.

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[a]U.S. Steel Corporation

The operating floor is at an elevation 21 ft 4 in. above grade. The floor is formed primarily by the concrete biological shielding above the reactor. The balance of the flooring is constructed of $\frac{1}{4}$-in. steel, checkered plates.

The fan floor is at an elevation 34 ft above grade and is constructed of metal deck panels.

All field connections are made with high-tensile bolts.

d. Personnel Access and Service Openings

Two stairways, one exposed and the other enclosed, provide personnel access to the operating floor. Both stairways are installed outside the building shell. The exposed stairway serves as an emergency exit. The enclosed stairway interconnects the Power Plant and the Support Facilities Building containing the control room.

Personnel access from the operating floor to the fan floor is by means of a ladder through a hatch and a personnel air lock.

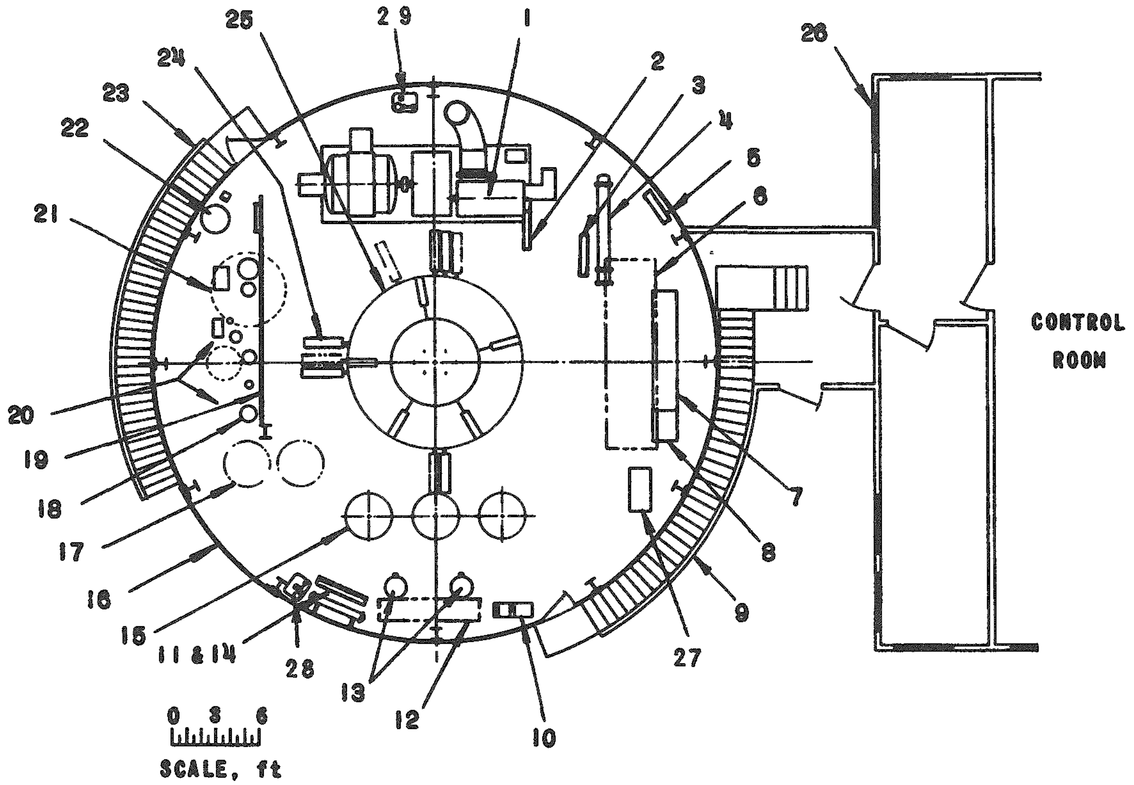
Additional openings include: (1) a cargo door that opens onto the operating floor; (2) a cargo hatch in the fan floor to facilitate transfer of equipment from the operating floor; and (3) a large opening in the sidewall of the fan floor to supply and exhaust air for the main condenser system.

Electrical troughs and conduit between the Power Plant and the Support Facilities Buildings pass through the wall of the building shell and the interconnecting passageway.

e. Equipment

The Power Plant Building (Fig. 1) is filled with gravel to a point 17 ft above the bottom plate. Embedded in the gravel are the reactor complex, three fuel-storage wells, two waste-storage tanks, a reactor water-purification system vault (containing the resin columns and heat exchanger), a contaminated-water retention tank, four reactor-instrument wells, miscellaneous piping, and ventilation ducts. The gravel is covered with a nonflammable pitch emulsion to contain radioactive gas or dust.

The equipment installed on the operating floor is shown in Fig. 6. The turbine-generator set is mounted on an isolated concrete pad which is supported by steel columns that are welded to the tank bottom plate. The balance of the equipment is supported by steel floor beams that transfer most of the loading to six steel columns centrally located around the reactor.

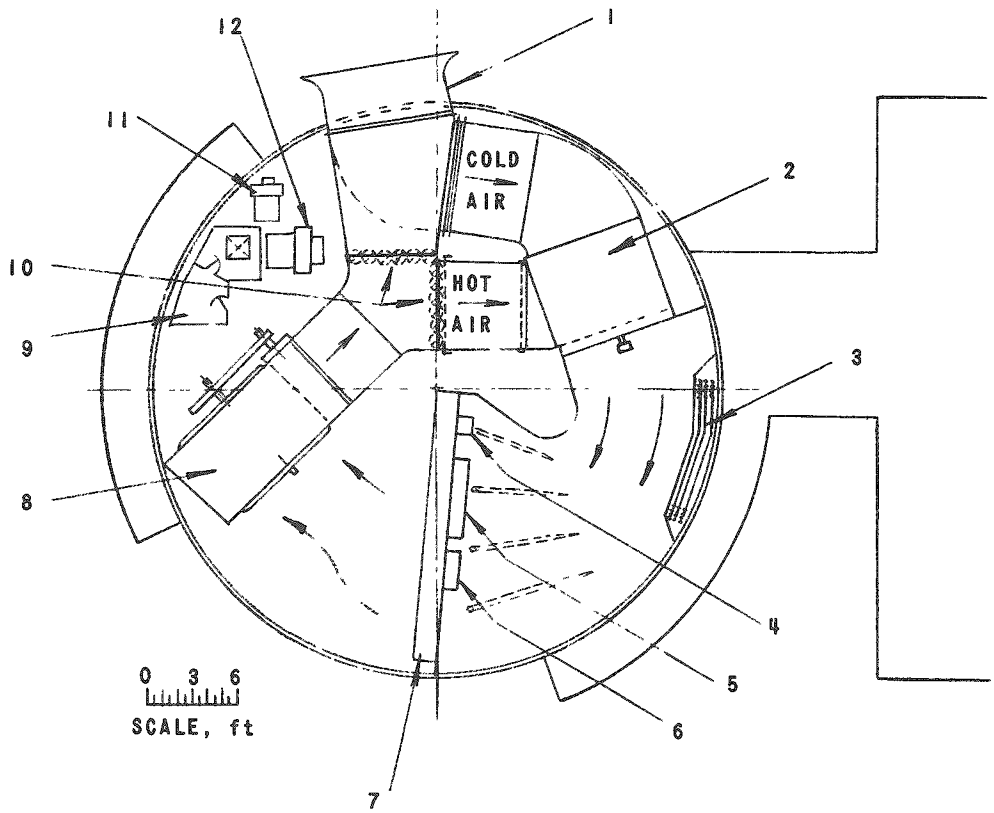


1. TURBINE - GENERATOR
2. TURBINE GENERATOR PANEL
3. LOCAL INSTRUMENT PANEL NO. 3
4. HEAT EXCHANGER - SPACE HEATING
5. TURBINE AUXILIARY OIL COOLER
6. WATER STORAGE TANK (OVERHEAD)
7. MOTOR CONTROL BOARD
8. NUCLEAR INSTRUMENT HIGH VOLTAGE SUPPLY PANEL
9. COVERED STAIRWAY
10. CONDENSATE CIRCULATING PUMP
11. LOCAL INSTRUMENT PANEL NO. 2
12. HOT WELL (OVERHEAD)
13. FEED WATER PUMPS
14. LOCAL INSTRUMENT PANEL NO. 1
15. FUEL STORAGE WELLS
16. EQUIPMENT DOORS
17. WASTE STORAGE TANKS
18. FEED WATER LINE FILTER
19. PURIFICATION SYSTEM PANEL
20. PURIFICATION SYSTEM AREA
21. CONTAMINATED WATER RETENTION TANK
22. BORON STORAGE TANK
23. OPEN EMERGENCY STAIRWAY
24. CONTROL ROD DRIVE MOTORS
25. CONCRETE SHIELD
26. SUPPORT FACILITIES BUILDING
27. PLANT MAKE-UP WATER DEMINERALIZER
28. HIGH PRESSURE CONDENSATE RETURN TANK
29. LOW PRESSURE CONDENSATE RETURN TANK

FIG. 6
OPERATING FLOOR PLAN

The operating floor is serviced by three cranes: a 10-ton bridge crane over the turbine-reactor-storage well area, a 10-ton monorail hoist over the cargo door, and a 1-ton jib crane in the reactor water-purification system work area.

The fan floor (Fig. 7) houses the main air-cooled condenser system, several small air-cooled condenser systems, building ventilation equipment, and the personnel air lock (Fig. 8). During operation the fan floor is at a negative pressure of ~ 2 in. H_2O. The main condenser system comprises an air-mixing chamber, a finned-tube condenser with an integral precooler, a fan, and the necessary dampers and ductwork.



1. AIR OUTLET-UPPER DUCT
   AIR INLET-LOWER DUCT
2. AIR MIXING CHAMBER
3. FUEL STORAGE VENT CONDENSER
4. AIR EJECTOR AFTER CONDENSER
5. HOT WELL CONDENSER
6. TURBINE OIL COOLER
7. AIR-COOLED CONDENSER
8. CONDENSER FAN
9. PERSONNEL AIR LOCK
10. DAMPERS
11. SHIELD AND INSTRUMENT EXHAUST FAN
12. BUILDING VENTILATION FAN

FIG. 7
FAN FLOOR PLAN

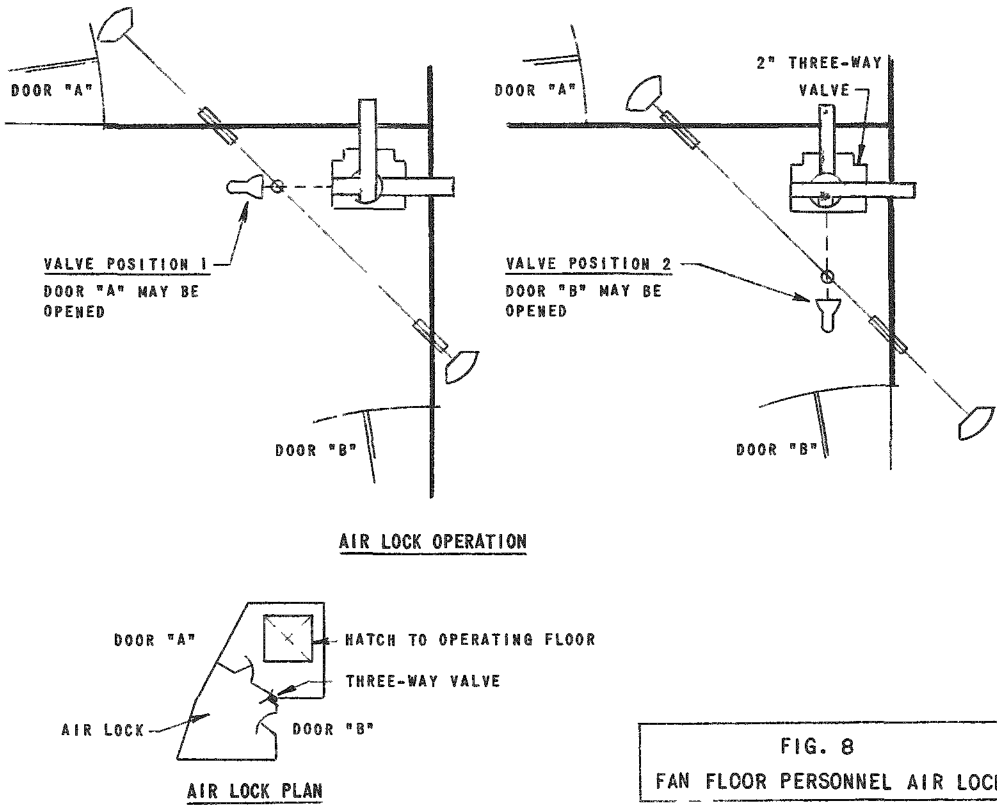


FIG. 8
FAN FLOOR PERSONNEL AIR LOCK

f. Lighting, Heating, Ventilating, and Insulation

The inner surface of tank walls above the gravel fill, and the under surface of the roof are insulated with a 2-in. layer of spray-type asbestos. The insulation is coated with a nonflammable pitch emulsion to provide a surface rigid enough to withstand a scrub-type decontamination. This type of insulation is not amenable to installation in the field. Extensive reworking and patching was required before an acceptable application was achieved.

Although adequate for the prototype site, additional insulation would be required at ultimate sites with a specified temperature of -60°F. During plant operation, enough heat is generated in the building to maintain above-freezing temperatures; however, during periods of plant shutdown the condensers and other water-containing components must be protected from freezing. Space limitations within the building necessitate an outside application, e.g., a rigid-type block insulation properly secured and covered. The application of insulation around the gravel-containing portion of the building shell would also serve to reduce stresses in the steel walls, as mentioned previously.

The building ventilation (Fig. 9) comprises five systems, four of which are interconnected. The four interconnected systems function as follows:

(1) The reactor shield enclosure ventilation system draws air from the operating floor area through the joints in the top shielding blocks, circulates the air around the control rod drives, past the pressure vessel head and top flange, and into a circular plenum around the pressure vessel, at an elevation just below the pressure vessel flange. From this plenum the air is circulated through narrow slots between the two horizontal, steel shielding plates, into a second circular plenum, and then to an exhaust fan that is common to the other three systems.

(2) The reactor instrument well ventilation system draws air from the operating floor area through a register beneath the nuclear instrument preamplifier panel, and distributes it into the bottom of each instrument well. The air is then circulated up through each well (past the instrument) into an exhaust duct. The duct discharges to the exhaust fan. The air flow is regulated by a damper and balanced with the air flow from the reactor shield closure ventilation system.

(3) The gravel exhaust system draws air from the gravel shield to maintain a negative pressure that prevents radioactive contamination of the areas above. The air is exhausted through a perforated pipe embedded in the gravel, and is filtered before entering the exhaust fan.

(4) The gravel air space exhaust system draws air from between the operating floor and the mastic seal on top of the gravel. This air is exhausted into the gravel shield exhaust duct through a barometric damper. Although no change has been made, it appears that this system is unnecessary.

The four systems utilize the same exhaust fan, located on the fan floor, which discharges to the atmosphere.

The fifth building ventilation system provides an independent source of fresh air to the operating floor area. During plant operation, air is drawn through the main condenser ducts, mixed to the proper temperature by a thermostatically controlled mixing chamber, and supplied by a fan to the operating floor area through two registers. The air is discharged to the atmosphere by the exhaust fan. During plant shutdown, the air is exhausted through a floor register leading to the main condenser exhaust duct.

Auxiliary heating of the building is provided by four unit heaters, two on each floor. The heaters are supplied with hot water from a boiler in the Support Facilities Building. Auxiliary heating at a future location would depend on facilities available at the particular site.

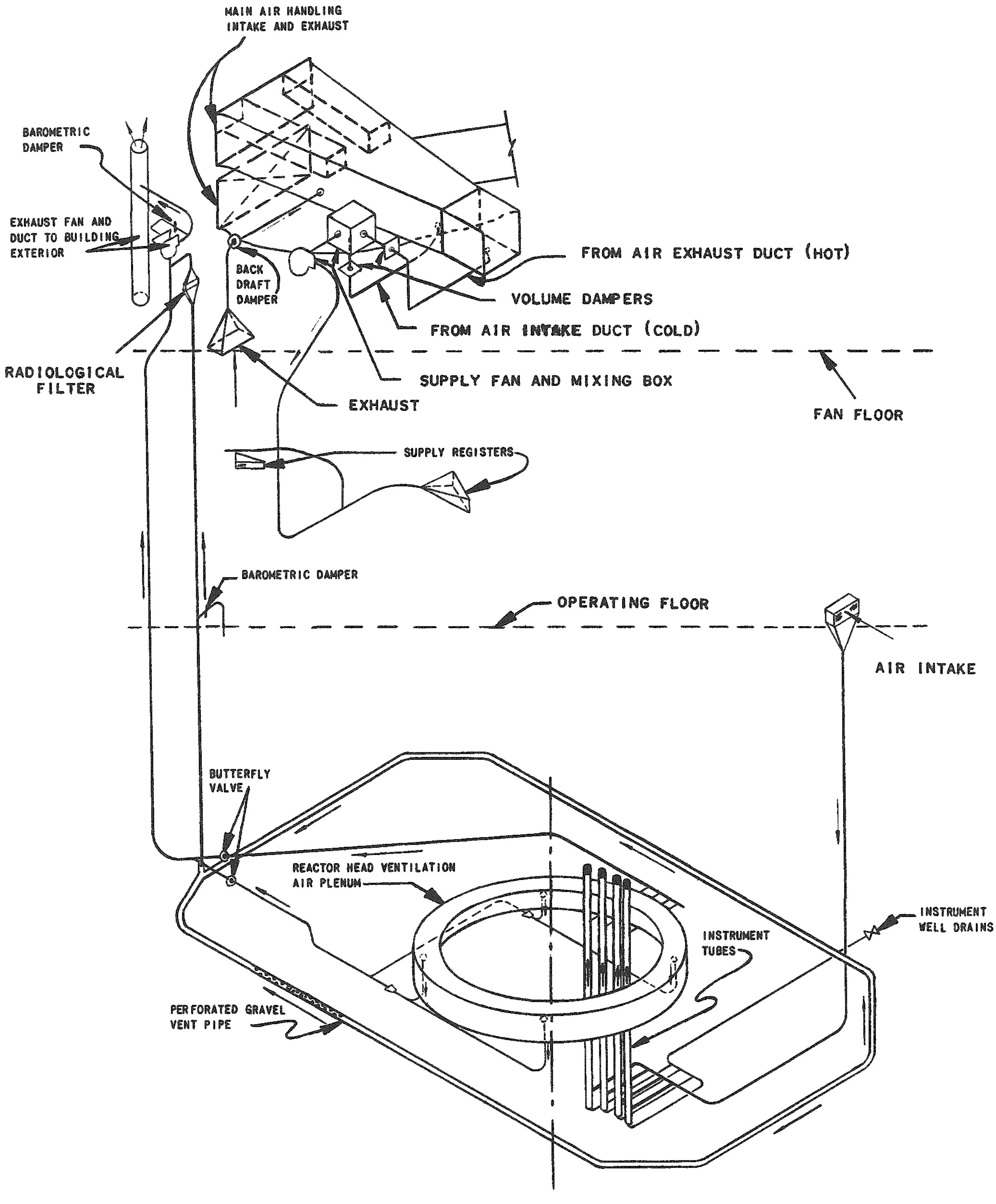


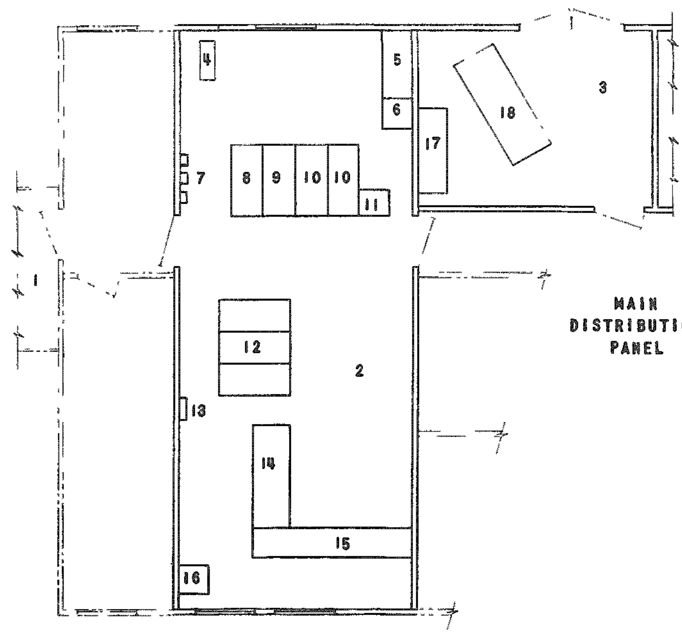
FIG. 9
PLANT VENTILATION FLOW DIAGRAM

Incandescent lighting is used throughout the building to lessen instrument interference. Power receptacles of 208 volts, 3-phase, and receptacles of 120 volts, single-phase, are located conveniently on each floor. Individual emergency battery-operated lights are located at critical points in the building.

2. Support Facilities Building

The Support Facilities Building is located adjacent, with access, to the Power Plant Building. The building is a prefabricated metal structure (40 ft x 100 ft) with ample window area, and six doors for personnel and equipment access. The interior is insulated and features panelled walls and ceiling. Space heating is supplied by a forced hot-air system. The building is also lighted with incandescent fixtures, and has 208-volt, 3-phase power receptacles, and 120-volt, single-phase receptacles.

The Support Facilities Building houses the control room for reactor operation (Fig. 10). An equivalent area would be required at any future site. The balance of the structure and the facilities contained therein is designed to satisfy the prototype site requirements rather than to simulate prototype installation. In addition to the control room, the Support Facilities Building contains laboratories for chemical analyses and low-level activity decontamination, simulated heat and power load equipment, diesel generator, furnace room, maintenance shops, offices, and personnel facilities.



1. PASSAGEWAY TO REACTOR BUILDING
2. CONTROL ROOM
3. ENGINE ROOM
4. SIMULATED HEAT LOAD PUMP
5. EMERGENCY POWER SUPPLY BATTERIES
6. EMERGENCY POWER SUPPLY PANEL
7. SIMULATED HEAT LOAD WATER FLOW AND TEMPERATURE INDICATORS

MAIN DISTRIBUTION PANEL:

8. LIGHTING AND POWER DISTRIBUTION PANEL
9. IDAHO POWER SWITCHING PANEL
10. GENERATOR SIMULATED LOAD & DIESEL GENERATOR SWITCHING PANELS
11. MOTOR CONTROL PANEL

12. GENERATOR CONTROL PANEL
13. CONDENSER INLET AIR TEMPERATURE INDICATOR
14. PROCESS CONTROL PANEL
15. NUCLEAR CONTROL PANEL
16. ROD DRIVE MOTOR CONTROL CABINET
17. EMERGENCY POWER SUPPLY MOTOR-GENERATOR SET
18. DIESEL GENERATOR

FIG. 10
PLAN VIEW OF CONTROL ROOM

C. Utilities

The site is supplied with 13.8-kv electric power from the Idaho Power Company. Emergency power is supplied by a diesel-generator unit and batteries on the site.

Telephone service is interconnected with the local NRTS system.

Water for domestic use, plant service, and fire protection is obtained from a deep well on the site, and is stored in a 50,000-gal tank at grade level. The stored water can be circulated to prevent freezing during cold weather.

The required water is distributed from the pump house. Plant water (unchlorinated) is pumped through a $1\frac{1}{2}$-in. line (at a rate of 15 gpm and at a pump discharge head of 60 ft) to the overhead storage tank (1000 gal) in the power plant building. The water is demineralized prior to entry into the reactor system.

Water for fire protection and domestic use in pumped independently through 6-in. lines leading to two fire hydrants and the chlorination house. The treated water is supplied through a $2\frac{1}{2}$-in. line to the support facilities laboratory sinks, lavatories, fire hose cabinet, and boiler.

Two waste-disposal systems are provided. Noncontaminated wastes drain to a septic tank, through a chlorination tank, and into a dry well. Contaminated wastes drain to a separate retention tank. The contents are pumped out periodically for ultimate disposal off the site.

III. DESIGN SUMMARY*

The design criteria and characteristics of the ALPR are summarized in Table 1. The approximate weights and costs of major plant components are listed in Table 2.

Table 1

SUMMARY OF ALPR DESIGN CRITERIA AND OPERATING CHARACTERISTICS

General	
Reactor Core Power Rating	3 Mwt (nominal)
Steam Production Rating	9,020 lb/hr (nominal)
Operating Pressure	300 psig
Operating Temperature	421°F
Feedwater Temperature	
Reactor	175°F
Condenser Hot Well	134°F
Core Life (design)	3 yr (nominal)
Average Power Density in Core Coolant	17.5 kw/liter
Average Heat Flux	21,500 Btu/(hr)(ft^2)
Turbine-Generator	
Power Rating	300 kw
Frequency[a]	60 cps
Voltage[a]	120/208 volts
Phase (4-wire)	3
Power Factor	0.80
Plant Factor	0.70
Standby Power Equipment Capacity (diesel-electric)	60 kwe
Domestic Heat Load Capacity (hot water)	400 kwt
Elevation at Prototype Site (above sea level)	5,000 ft
Ambient Temperature (design[b])	-60°F to + 60°F
Maximum Wind Speed (design)	125 mph
Maximum Building Height (above grade)	50 ft
Site Surface Condition (assumed basis for construction)	Permafrost
Site Materials for Construction	Local Gravel
Transportability of Components	Airlift

* H. H. Hooker

[a] See Appendix V; Operational Requirements - section 5.

[b] Not incorporated in building shell or insulation design.

Table 1 (Cont'd.)

Core

Maximum Horizontal Cross Section	~35 in. x 35 in. (corners removed)
Fuel Loading, Uranium	
40-assembly, U^{235}(minimum, poisoned)	14.00 kg
59-assembly, U^{235}(maximum, poisoned)	20.65 kg
Overall Metal-to-Water Ratio	0.5
Loading of Burnable Poison (40-fuel-assembly core)	
Number of 0.021-in.-thick strips	16 half-length
Number of 0.026-in.-thick strips	40 full-length

Fuel Assembly

Overall Dimensions	$3\frac{7}{8}$ in. x $3\frac{7}{8}$ in. x $34\frac{1}{2}$ in.
Weight of U^{235}	350 gm
Number of Plates	9
Average Water Channel Gap	0.310 in.

Fuel Plate

Overall Dimensions	0.120 in. x $3\frac{25}{32}$ in. x $27\frac{13}{16}$ in.
"Meat" Dimensions	0.050 in. x 3.500 in. x 25.800 in.
Cladding Thickness	0.035 in.
Cladding and "Meat" Diluent	Aluminum-nickel alloy
"Meat" Volume	74 cm^3
Weight of Al-Ni Alloy "Meat" Diluent	~195 gm
Weight of U^{235}	38.9 gm
Weight of Uranium (~93 w/o U^{235})	41.7 gm
Uranium in the "Meat"	~17.60 w/o
Uranium in the "Meat"	~2.21 a/o

Burnable Poison Strips

Overall Dimensions (full length)	$3\frac{7}{8}$ in. x $25\frac{13}{16}$ in. in thicknesses of 0.021 in. or 0.026 in.
Weight B^{10} per Strip	
0.021-in. thickness	0.4 gm
0.026-in. thickness	0.5 gm
B^{10} in strip	~0.43 w/o

Nuclear Data for 40-fuel-assembly Reference Core at 3 Mwt

Average Thermal Flux in Fuel (fresh reactor)	~ 6.2×10^{12} (n)(cm)/(cm^3)(sec)

Table 1 (Cont'd.)

Max/Avg Flux Ratio in Hot Fresh Reactor (no steam voids)	Calculated	Measured
Radial (gross)	1.6	-
Control Cell	1.3	-
Axial (control rods out)	1.3	-
Axial (control rods halfway in)	1.7	-
Reactor (control rods out)	2.7	-
Reactor (control rods halfway in)	3.5	-
Reactivity Changes (approximate), dollars (1 dollar = 0.7%)		
Temperature (cold→operating), fresh reactor	-2.2 to -2.9	-3.3 (± 0.5)(rods banked at ~21 in.)
Xenon + Samarium	-3.7 (fresh reactor)	3.5[a]
	-4.3 (depleted reactor)	-
Steam Voids (dependent on rod positions)	-1.9 to -2.9	-3.3(± 0.6)(estimated)
Xenon Override	-0.4 (fresh reactor)	~ -0.3
	-1.4 to -2.1 (depleted reactor)	-
Neutron Lifetime	4×10^{-5} sec to 8×10^{-5} sec (over range of core conditions)	

Heat Transfer and Fluid Flow for 40-Fuel Assembly Reference Core at 3 Mwt

Average Power Density in Core Coolant	17.5 kw/liter
Steam Flow	9,020 lb/hr
Average Steam Void in Fuel Assembly Channel	~9%
Average Steam Void in Moderator of Core	~7%
Exit Steam Quality	~0.7%
Water Recirculation Ratio (lb water per lb steam)	130
Feedwater Inlet Temperature	175°F
Subcooling of Water at Core Inlet	1.5°F to 2°F

[a] Equilibrium xenon measured worth ~2.3 dollars (1.6% at ~2.6 Mwt; estimate is ~2.5 dollars (1.7%) at 3 Mwt. Equilibrium samarium computed worth ~1 dollar (~0.7%).

Table 1 (Cont'd.)

Average Boiling Length	20 in.
Total Heat Transfer Area	475 ft^2
Average Heat Flux	21,500 Btu/(hr)(ft^2)
Maximum Heat Flux	75,000 Btu/(hr)(ft^2)
Average Temperature at Center Line of Fuel Plate	450°F
Maximum Temperature at Center Line of Fuel Plate	470°F
Average Temperature at Surface of Fuel Plate Cladding	440°F
Maximum Temperature at Surface of Fuel Plate Cladding	460°F
Control Rods	
Number of Cross-type Rods	5
Additional Spaces for Tee-type Rods	4 (not used on reference core)
Spacing	$8\frac{13}{16}$ in. (square lattice)
Length of Cadmium Section	34 in.
Thickness	
Cadmium ("meat")	0.060 in.
Aluminum-Nickel Alloy (clad)	0.080 in.
Total Metered Travel	30 in.
Scram Time (30 in. of rod travel, from rest)	2 sec
Withdrawal Rate[a]	Final: Rod No. 9:: 1.8 in./min; Rods No. 1, 3, 5 and 7:: 2.85 in./min
Weight of Control Rod	
Cross-type Rod No. 9	48 lb
Cross-type Rod No. 1, 3, 5 or 7	42 lb
Tee-type Rod	37 lb
Pressure Vessel	
Diameter, Outside	4.5 ft
Wall Thickness	
Base Material	$\frac{3}{4}$ in.
Stainless Steel Clad	$\frac{3}{16}$ in.

[a] Adjustable to meet criterion of maximum rate of reactivity addition (~0.01%/sec); set initially at 3 in./min. At 1.8 in./min., the maximum rate of reactivity addition by the center rod in the cold fresh reactor is ~0.02 dollar/sec (~0.01(5)%/sec).

Table 1 (Cont'd.)

Height (less head)	$14\frac{1}{2}$ ft
Design Pressure	400 psig
Thermal Shield	$\frac{3}{4}$ -in. stainles steel
Pressure Relief Settings	
First Stage (to Condenser)	350 psig
Second Stage (to Atmosphere)	385 psig
Level of Water above Core	4 ft 4 in.
Total Weight (empty)	~26,000 lb
Average Volume of Steam Dome	~80 cu ft
Total Weight of Contained Water (Operating)	~8000 lb
Maximum Thermal Stresses due to Gamma Heat in Vessel	750 psi
Maximum Thermal Stresses due to Gamma Heat in Cover	600 psi
Type of Head Closure	Double gasketed with leakoff

Table 2

APPROXIMATE WEIGHT AND COST OF MAJOR PLANT COMPONENTS

	Weight lb	Cost $
Reactor Pressure Vessel (empty)	26,000	39,000
Water Purification System Complex	24,000	51,000
Turbine-Generator	21,000	47,000
Pressure Vessel Support Cylinder and Thermal Shield	15,000	20,000
Permanent Concrete Shielding	80,000	-
Removable Concrete Shielding		
Heaviest Segment	15,000	-
Total	75,000	-
Permanent Steel Shielding	42,000	-
Condenser Fan	13,000	16,000
Electrical Switchgear	11,500	33,000
Condenser and Vapor Pre-cooler	10,500	24,500
Process Instrumentation and Panel	5,000	32,500
Nuclear Instrumentation and Panel	5,000	38,500
Control Rod Drive Assemblies (10)	4,000	91,000
Feed-water Pumps (2)	3,000	8,500
Emergency Power Supply	2,000	6,000
Permanent Gravel Shielding	2,000,000	17,000
Power Plant Building		
Shell	108,000	76,000
Structural Steel	110,000	94,000

A. Plant Operation

During normal plant operation, saturated steam at 300 psig and 421°F is generated within the core. As shown in the flow diagram (Fig. 11), the major portion of the steam flows from the reactor pressure vessel directly to the turbine and then to the air-cooled condenser. A small amount of steam is used to operate the plant air ejectors and the turbine gland seal system. The balance of the steam flows through the bypass regulating valve and primary steam-bypass line to the space-heating heat exchanger, and to a secondary bypass line around this heat exchanger. The steam flow to the heat exchanger is regulated by a thermostatic valve having a temperature-sensing element in the secondary water circuit. Condensate formed in the heat exchanger is flashed to the main steam condenser and the hot well by way of the flash tank. Steam in excess of the space-heating requirements is bypassed directly to the condenser through a back pressure-regulating valve in the secondary bypass line.

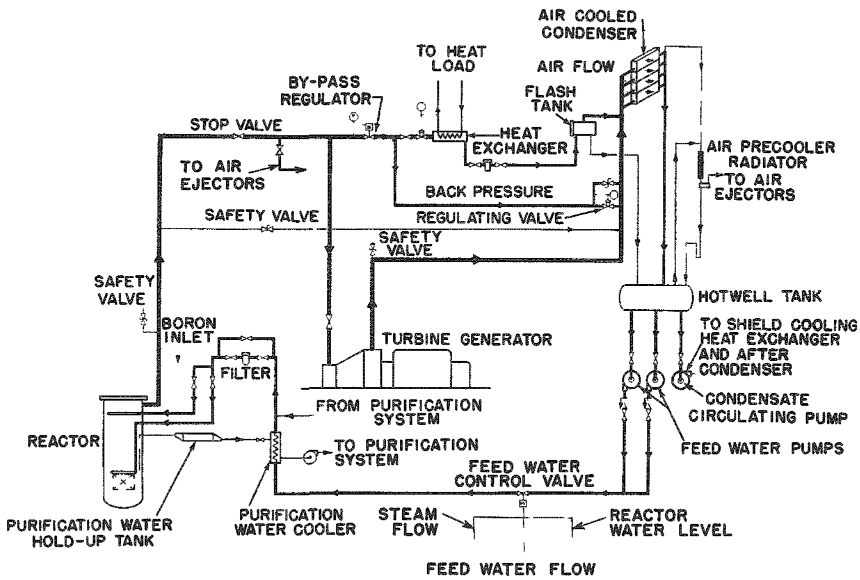


FIG. 11
FLOW DIAGRAM

The power level of the reactor may be controlled either manually or automatically. Manual control of reactor power is used in conjunction with automatic adjustment of the bypass regulating valve. The valve operates to maintain a relatively constant steam pressure by compensating for minor changes in turbine steam flow.

When automatic control of reactor power is employed, the steam pressure-regulating function of the bypass regulating valve is not required. The valve is adjusted manually to supply only the steam required for the simulated space-heating system.

In the prototype plant, the heat transferred to the secondary (water) circuit of the simulated space-heating system is dissipated to the atmosphere by means of an air-cooled heat exchanger.

All exhaust steam is passed to the main air-cooled condenser on the fan floor. Cooling air is drawn through the condenser by a motor-driven centrifugal fan. During freezing weather, the temperature of the condenser inlet air is raised by mixing the incoming stream of outside air with an automatically controlled fraction of the condenser exhaust air. The speed of the condenser fan and, hence, the flow of cooling air, is reduced automatically during periods when maximum condenser capacity is not required. An air-ejector system continuously removes the noncondensible gases from the system.

Condensate flows by gravity from the condenser to the hot well. From the hot well it is pumped by one of two feed pumps through the reactor water-purification-system cooler and filter, to the reactor vessel where it is discharged through the feed-water spray ring. In the event of failure of the operating feed-water pump, the standby pump is started automatically.

A reactor water-purification system is provided to maintain water purity. Water is pumped at approximately 3 gpm, and up to 5 gpm, from the reactor vessel through a heat exchanger (where it is cooled by the incoming feed water) to ion exchange beds. It then flows to the feed-water line where it is mixed with the feed water and returned to the reactor.

A portion of the condensate from the hot well is pumped through an auxiliary system where it is utilized as a coolant for the shield-cooling system, gland air-ejector precooler, and air-ejector aftercondenser, and as feed pump gland sealing water.

Excessive reactor steam pressure is relieved through two pressure-actuated valves. The first exhausts to the condenser. The second, which is set at a higher pressure, exhausts to the atmosphere outside the building. Excessive condenser pressure is relieved through a pressure-actuated valve which also exhausts to the outside.

B. Reactor Core

In effect, the shroud structure for the reactor core divides the core into sixteen compartments, twelve designed to accommodate a 2 x 2 array of assemblies, and four corner compartments contoured to contain three assemblies each (Fig. 12). Thus, the core capacity is fifty-nine fuel assemblies, plus one source assembly. The sides of the compartments form shrouds for $\frac{1}{2}$-in.-thick water channels in which the reference system of five cross-shaped control rods and the system of four tee-shaped rods move.

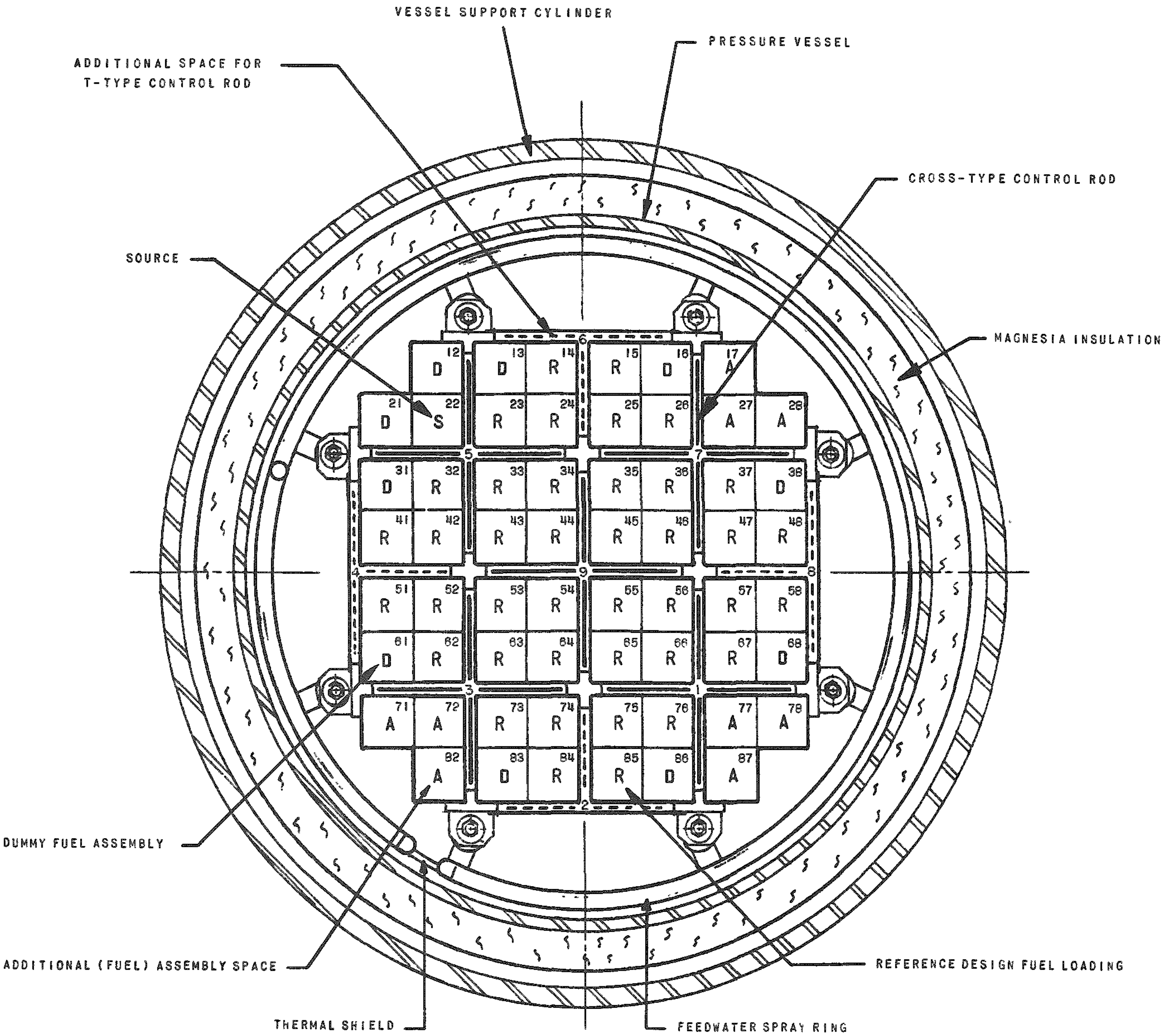


FIG. 12
REFERENCE 40-FUEL-ASSEMBLY CORE LOADING FOR 3 Mwt OPERATION

The core structure is supported on eight brackets welded to a $\frac{3}{4}$-in.-thick stainless steel cylindrical annulus acting as the (radial) thermal shield. In turn, the thermal shield is supported on lugs welded to the pressure vessel. The entire core structure is fabricated of an alloy of aluminum with ~1 w/o nickel (type X8001).

The reference core loading for 3-Mwt operation, shown in Fig. 12, includes forty fuel assemblies and one source assembly. Ten dummy fuel assemblies are used to fill out compartments and thus to provide lateral support for the other assemblies. Each assembly is $3\frac{7}{8}$ in. square and $34\frac{1}{2}$ in. long. The fuel assembly contains nine flanged fuel plates which are welded to two side plates. The fuel plate consists of a 25.8-in.-long, 0.050-in.-thick core of uranium, enriched to 93 w/o U^{235}, in an alloy with aluminum plus 2 w/o nickel, clad with 0.035 in. of X8001. Each fuel plate contains ~39 gm U^{235}. Fuel plates are separated by 0.031-in.-thick water coolant channels.

To assist in control of the excess reactivity required for the nominal core life of three years, thin burnable-poison strips, extruded from a mixture of aluminum-nickel powder and a powder highly enriched in B^{10}, are tack-welded to one or both side plates of fuel assemblies. In the reference 40-assembly core, one full-length (25.8 in.) strip containing ~0.5 gm B^{10} is welded to a side plate of each fuel assembly, within the active core region. In addition, one half-length strip containing ~0.2 gm B^{10} is welded to the other side plate of each of the central sixteen fuel assemblies, in the bottom half of the active core.

The control rods consist of a cadmium absorber section, 0.060 in. thick and 34 in. long, confined between two 0.080-in.-thick sheets of X8001 alloy. The cross-shaped rods have an absorbing span of 14 in.; the portion of the tee-shaped rod pointing toward the center of the core has an active length of 7 in. and the other span has a total active length of $11\frac{1}{2}$ in. (The tee rods are not used in the 40-assembly reference core.) Control rods are driven by rack-and-pinion mechanisms through combination magnetic and over-running mechanical clutches (see Fig. 13). When the clutches are de-energized, the rods fall by gravity through the full 30 in. of metered travel within two seconds. If the rod should stick during its free fall, the rod drive over-running clutch will engage and the rod will be driven past the region of resistance. The central cross control rod (No. 9) is moved by its rod drive at the rate of 1.8 in./min, corresponding to a maximum rate of change of reactivity of ~0.02 dollar/sec [~0.01 (5)%/sec)] in the cold fresh reactor. The remaining four cross control rods are driven at a rate of 2.85 in./min, since each is considerably less effective than rod No. 9.

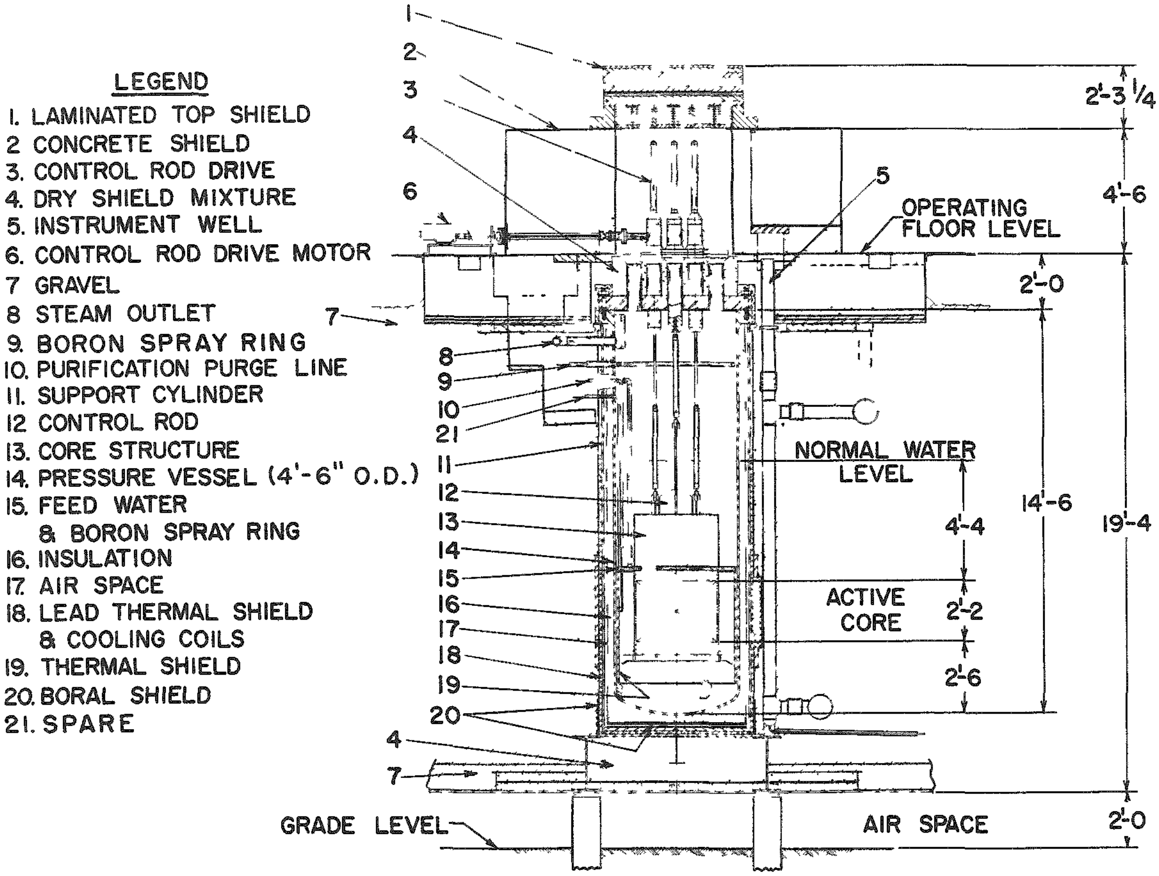


FIG. 13
REACTOR INSTALLATION, VERTICAL SECTION

C. Reactor Pressure Vessel

The reactor vessel (see Fig 13) is fabricated of SA-212, Grade B Firebox Quality steel and measures 4 ft 6 in. in outside diameter by 14 ft 6 in. in length. The upper end is flanged and machined to accommodate a flat cover plate which is bolted against two concentric gaskets to effect a pressure-tight closure. Any outleakage of steam across the inner gasket is collected in an annulus between the gaskets and returned to the system. The inside surface of the vessel and the cover plate is clad with Type 304 stainless steel.

All openings are limited to the cover plate and the upper sidewalls of the vessel proper. The openings in the cover plate include nine 6-in. flanged nozzles for the control rod drives, and one 4-in. nozzle and one 2.5-in. nozzle for liquid-level instrumentation. The openings in the vessel wall include nozzles for the steam outlet (4-in.), feed-water inlet ($1\frac{1}{4}$-in.), shutdown cooling water inlet ($1\frac{1}{4}$-in.), purification system water outlet (1-in.), and a spare (1-in.).

The outer surface of the vessel is covered with a layer of magnesia insulation (3 in. thick) which is retained by a steel jacket ($\frac{1}{4}$ in. thick).

D. Pressure Vessel Support Cylinder

The reactor vessel is suspended from its upper flange, which rests on the support cylinder (see Figs. 12 and 13). This cylinder is fabricated of carbon steel ($\frac{7}{8}$ in. thick) and consists of two flanged half-cylinders that are bolted together. The assembly rests on a steel plate (1 in. thick) which, in turn, is supported by building structural steel.

The support cylinder also constitutes a substantial part of the thermal shielding. The outer surface is covered with a layer of lead ($1\frac{1}{4}$ in. thick) which is retained by a steel jacket ($\frac{1}{4}$ in. thick). At core level, the thickness of the steel jacket is increased to $1\frac{1}{4}$ in. In the area adjacent to three nuclear instrument wells, the $1\frac{1}{4}$-in. steel band is replaced with an additional 3 in. belt of lead. A $1\frac{1}{4}$-in. layer of lead is also installed above the 1-in. steel plate that supports the cylinder.

Heat generated within, or conducted to, the support cylinder, outer thermal shield, and bottom thermal shield is removed by cooling water that is circulated through copper cooling coils embedded in the lead.

E. Shielding

The bulk of the biological shielding is local gravel (see Fig. 13), which extends upward from the bottom steel plates of the building shell to a level approximately 18 in. below the operating floor. In addition to shielding the reactor, the gravel serves as a convenient shielding medium for radioactive system components, i.e., ion exchange beds, spent fuel storage tanks, and contaminated water (retention) tank.

Beneath the reactor vessel, the lead thermal shield is augmented by a Boral sheet for thermal neutron capture, and a bed of mixed steel punchings, boric oxide, and sand.

The space surrounding the reactor vessel cover nozzles is also filled with a mixture of steel punchings, boric oxide, and sand. The mixture extends upward from the vessel cover to slightly below the nozzle flanges, and is retained by a steel jacket.

The top biological shield consists primarily of steel plates and concrete. A ring of concrete, flush with the operating floor level, surrounds the top of the vessel. Above this ring, and resting on it are five pie-shaped blocks of concrete poured in structural steel frames. The contiguous surfaces of the blocks are stepped to reduce radiation streaming.

The shielding directly above the control rod drives is comprised of alternate layers of Masonite hardboard and steel. The steel affords gamma shielding, and the hydrogenous Masonite provides neutron attenuation.

IV. OPERATIONAL CHARACTERISTICS*

A Plant Performance Test(8) consisting of 500 continuous operating hours for the complete plant at maximum power was the most comprehensive operational test performed by the Laboratory. Figure 14 represents a typical plant heat balance for the test.

A. Power Plant

The master flow diagram of the power plant and the valve designations and descriptions, are illustrated in Figs. 15 and 16, respectively. In Table 3 are tabulated the design versus operating characteristics at rated plant power. Deviations from the design calculations are noted and discussed. The power plant was accessible at all times to personnel. Time limitations imposed in some areas were due to radiation levels above tolerance (AEC) and are discussed in section VII.

1. Steam Systems

The main steam system (see Fig. 17) consists of the piping, valves, and instrumentation necessary to supply 300-psig, dry and saturated (421°F), reactor steam directly to the turbine-generator unit, turbine gland seals, air ejectors, and indirectly to the simulated heat-load heat exchanger (space-heating system).

The main steam line was originally designed to incorporate a steam separator and a pressure-regulating valve at the exit of the reactor pressure vessel. Experiments with BORAX indicated that the value of the steam separator was doubtful and that a reactor pressure-regulating valve was unnecessary. Subsequent ALPR plant operation verified the results of the BORAX experiments. Operating pressures are higher than the design pressure because the pressure drops induced by the separator and valve were eliminated.

The bypass steam system (see Fig. 17) supplies steam to the simulated heat-load heat exchanger at a pressure of 40 psig. Steam generated in excess of that required by the turbine-generator unit, turbine gland seals, air ejectors, and space-heating system is routed to the main air-cooled steam condenser by discharging into the turbine exhaust piping. The system capacity is such that upon turbine throttle trip the full steam load can be rejected to the steam condenser.

Originally, the automatic-demand control feature of the plant necessitated the incorporation of an automatic bypass steam flow regulating valve (0-3000 lb/hr) for changes in plant load. The valve is positioned in response to the deviation of reactor steam pressure from a set point.

*E. E. Hamer

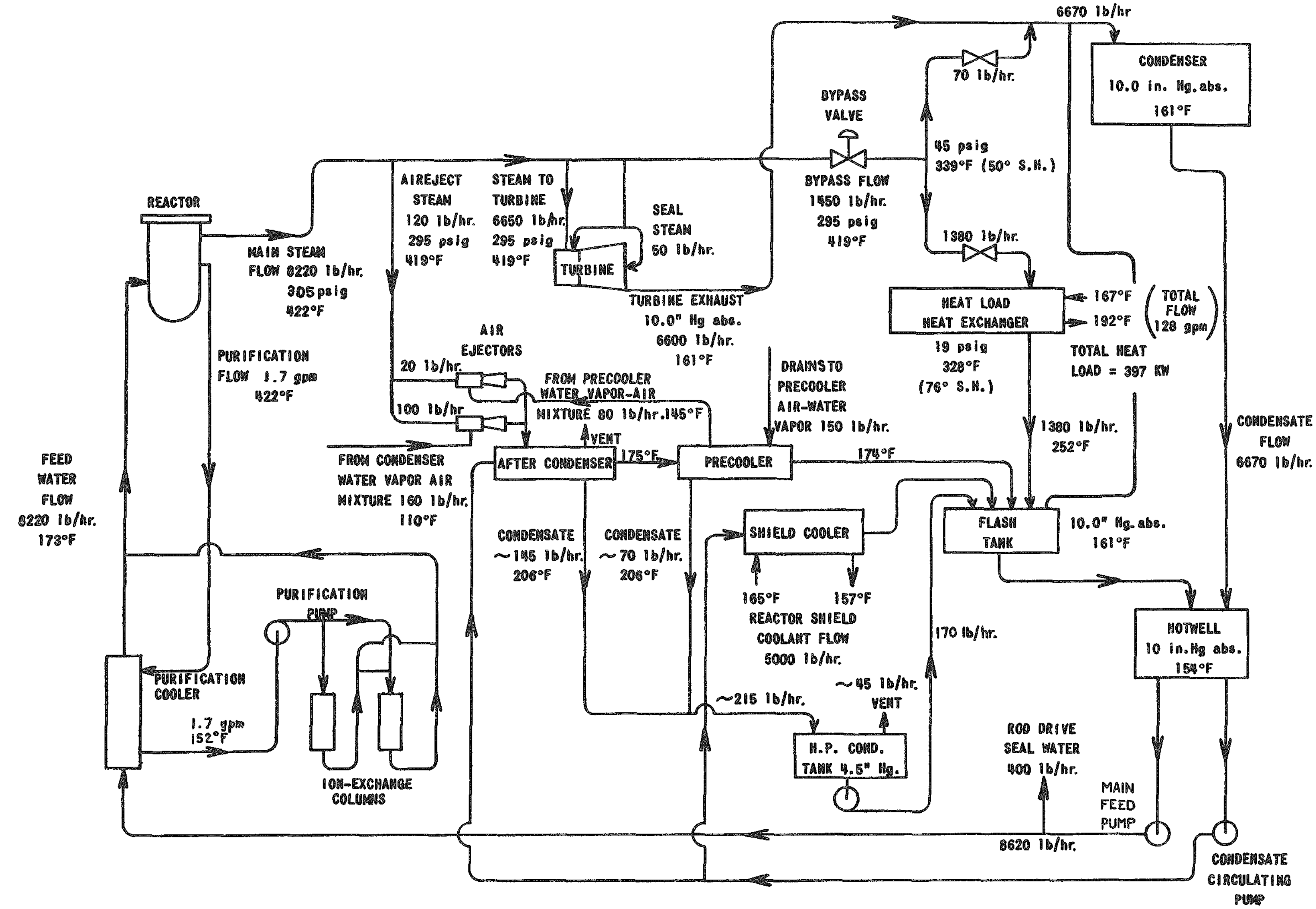


FIG. 14
TYPICAL HEAT BALANCE FOR PLANT PERFORMANCE TEST AT 300 kwe
NOVEMBER 30, 1958

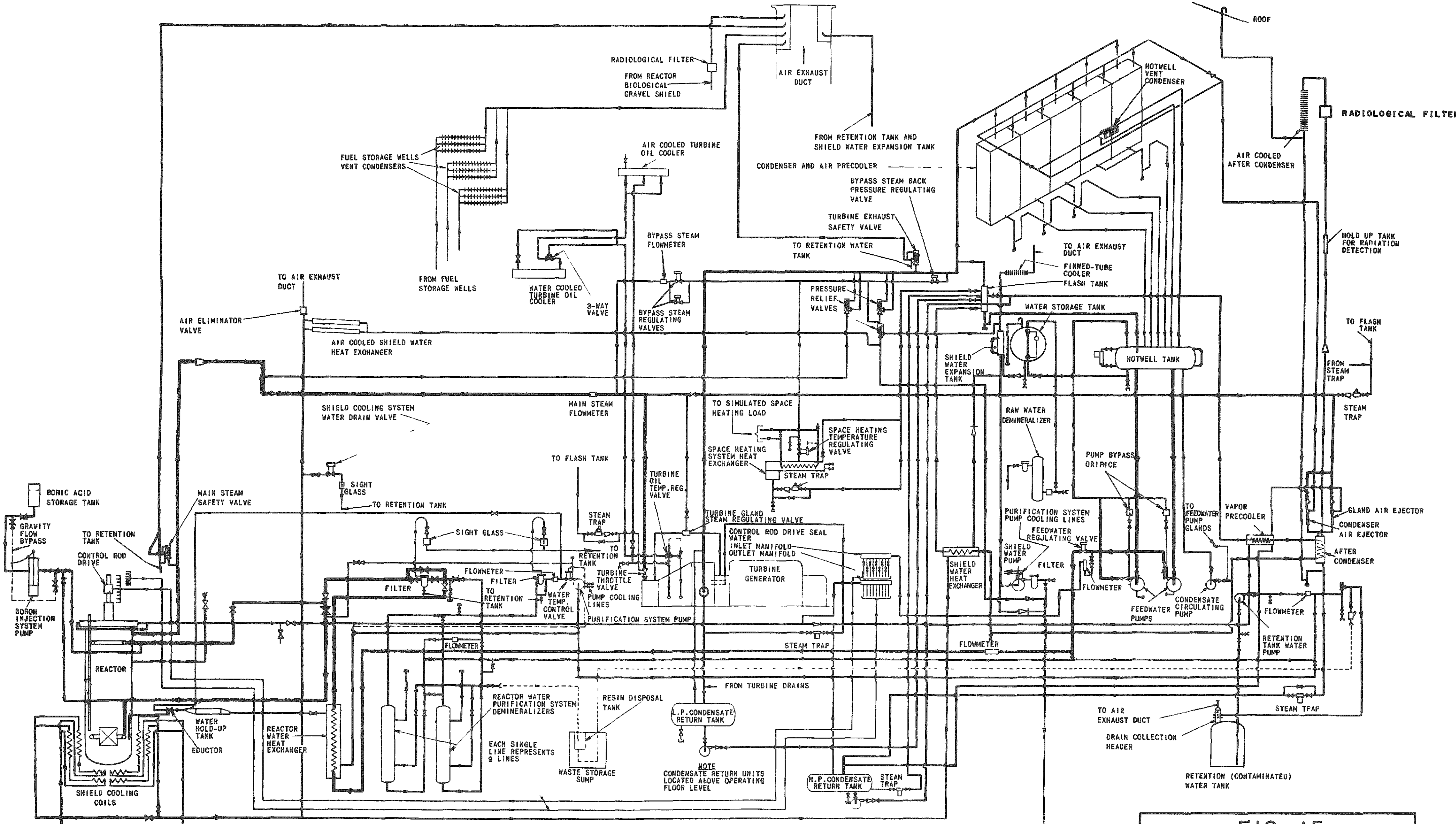


FIG. 15
MASTER FLOW DIAGRAM

MAIN STEAM SYSTEM

SYMBOLS	DESCRIPTION
MS-1	REACTOR SHUTOFF
MS-2	TURBINE TRIP-THROTTLE
MS-3	STEAM BYPASS REGULATING 0-3000 LB/HR (1" VALVE)
MS-4	STEAM BYPASS REGULATING 0-10,000 LB/HR (1½" VALVE)
MS-5	STEAM BYPASS BACK PRESSURE REGULATING (40 PSIG)
MS-6	STEAM BYPASS PRESSURE RELIEF (50 PSIG)
MS-7	REACTOR PRESSURE RELIEF TO CONDENSER (350 PSIG)
MS-8	REACTOR PRESSURE SAFETY TO ATMOSPHERE (385 PSIG)
MS-9	TURBINE EXHAUST SAFETY TO ATMOSPHERE (5 PSIG)

AUXILIARY STEAM SYSTEM

AS-1	AIR EJECTORS STEAM LINE SHUTOFF
AS-2	CONDENSER AIR EJECTOR STEAM SHUTOFF
AS-3	GLAND AIR EJECTOR STEAM SHUTOFF
AS-4	TURBINE GLAND STEAM SHUTOFF
AS-5	SPACE HEATING STEAM SHUTOFF
AS-6	SPACE HEATING STEAM REGULATING
AS-7	SPACE HEATING STEAM SUPPLY LINE DRAIN
AS-8	SPACE HEATING HEAT EXCHANGER VENT
AS-9	SPACE HEATING HEAT EXCHANGER VENT AND CONDENSATE DRAIN AT FLASH TANK SHUTOFF
AS-10	**CONDENSATE FROM L.P. CONDENSATE TANK**

FEEDWATER SYSTEM

FW-1	FEEDWATER PUMP #1 INLET SHUTOFF
FW-2	FEEDWATER PUMP #1 OUTLET CHECK
FW-3	FEEDWATER PUMP #1 OUTLET SHUTOFF
FW-4	FEEDWATER PUMP #2 INLET SHUTOFF
FW-5	FEEDWATER PUMP #2 OUTLET CHECK
FW-6	FEEDWATER PUMP #2 OUTLET SHUTOFF
FW-7	FEEDWATER REGULATING (MOTOR OPERATED)
FW-8	FEEDWATER LINE DRAIN
FW-9	CONTROL ROD DRIVE COOLING WATER REGULATING

COOLING WATER SYSTEMS (CONDENSATE)

CW-1	CONDENSATE CIRCULATING PUMP INLET SHUTOFF
CW-2	CONDENSATE CIRCULATING PUMP DISCHARGE SHUTOFF
CW-3	CONDENSATE CIRCULATING LINE TO SHIELD COOLER DRAIN
CW-4	CONDENSATE CIRCULATING LINE FROM SHIELD COOLER TO FLASH TANK SHUTOFF
CW-5	CONDENSATE CIRCULATING LINE FROM AIR EJECTOR PRECOOLER TO FLASH TANK SHUTOFF

FEEDWATER PUMPS GLAND SEAL VALVES CARRY NO SEPARATE DESIGNATION

SHIELD COOLING SYSTEM

SYMBOLS	DESCRIPTION
SC-1	SHIELD COOLER PUMP INLET SHUTOFF
SC-2	SHIELD COOLER PUMP AND TANK DRAIN
SC-3	SHIELD COOLER PUMP DISCHARGE SHUTOFF
SC-4	SHIELD COOLER PUMP CHECK TO AUXILIARY COOLER
SC-5	SHIELD COOLING COIL #1 THROTTLING AND SHUTOFF
SC-6	SHIELD COOLING COIL #2 THROTTLING AND SHUTOFF
SC-7	SYSTEM DRAIN (MANUAL)
SC-8	SYSTEM BYPASS AND DRAIN (SOLENOID)
SC-9	SYSTEM PRESSURE RELIEF
SC-10	SYSTEM EXPANSION TANK CHECK TO SHIELD COOLER
SC-11	SHIELD COOLER PUMP AND SYSTEM FILTER BYPASS CHECK

MAKE-UP WATER SYSTEM

MW-1	RAW WATER DEMINERALIZER SOLENOID VALVE BYPASS
MW-2	RAW WATER DEMINERALIZER TANK INLET SHUTOFF
MW-3	RAW WATER DEMINERALIZER TANK BYPASS
MW-4	RAW WATER DEMINERALIZER OUTLET SHUTOFF
MW-5	RAW WATER DEMINERALIZER DRAIN
MW-6	RAW WATER DEMINERALIZER TANK VENT
MW-7	RAW WATER DEMINERALIZER INLET SHUTOFF (SOLENOID)
MW-8	RAW WATER DEMINERALIZER INLET SHUTOFF (MANUAL)
MW-9	WATER STORAGE TANK TO CONDENSER HOTWELL SHUTOFF
MW-10	WATER STORAGE TANK TO SHIELD COOLING EXPANSION TANK SHUTOFF

REACTOR WATER AND CONTAMINATED WATER (RETENTION) SYSTEM

RP-1	FEEDWATER SYSTEM SHUTOFF
RP-2	FEEDWATER FILTER INLET SHUTOFF
RP-3	FEEDWATER FILTER BYPASS SHUTOFF
RP-4	PURIFICATION SYSTEM INLET SHUTOFF
RP-5	MIXED BED DEMINERALIZER OUTLET REGULATING
RP-6	FEEDWATER FILTER OUTLET SHUTOFF
RP-7	PURIFICATION SYSTEM OUTLET SHUTOFF
RP-8	MIXED BED DEMINERALIZER INLET SHUTOFF
RP-9	CATION BED DEMINERALIZER INLET SHUTOFF
RP-10	CONTAMINATED WATER (RETENTION) TANK-CATION BED DEMINERALIZER INLET SHUTOFF
RP-11	CATION BED DEMINERALIZER OUTLET SHUTOFF
RP-12	CONTAMINATED WATER (RETENTION) TANK-MIXED BED DEMINERALIZER INLET SHUTOFF
RP-13	PURIFICATION SYSTEM PUMP BYPASS SHUTOFF
RP-14	CATION BED DEMINERALIZER OUTLET TO CONTAMINATED WATER (RETENTION) TANK SHUTOFF
RP-15	CATION BED DEMINERALIZER FLOWMETER INLET REGULATING
RP-16	MIXED BED DEMINERALIZER BYPASS SHUTOFF

REACTOR WATER AND CONTAMINATED WATER (RETENTION) SYSTEM (CONTD.)

SYMBOLS	DESCRIPTION
RP-17	MIXED BED DEMINERALIZER OUTLET TO CONTAMINATED WATER (RETENTION) TANK SHUTOFF
RP-18	DEMINERALIZERS OUTLET TO CONTAMINATED WATER (RETENTION) TANK HOSE CONNECTION SHUTOFF
RP-19	CONTAMINATED WATER (RETENTION) TANK PUMP DISCHARGE HOSE CONNECTION SHUTOFF
RP-20	FEEDWATER FILTER VENT SHUTOFF
RP-21	FEEDWATER FILTER DRAIN SHUTOFF
RP-22	PURIFICATION SYSTEM FILTER VENT SHUTOFF
RP-23	PURIFICATION SYSTEM FILTER INLET DRAIN SHUTOFF
RP-24	PURIFICATION SYSTEM FILTER DRAIN SHUTOFF
RP-25	CONTAMINATED WATER (RETENTION) TANK PUMP INLET HOSE CONNECTION SHUTOFF
RP-26	CONTAMINATED WATER (RETENTION) TANK PUMP INLET SHUTOFF
RP-27	CONTAMINATED WATER (RETENTION) TANK PUMP DISCHARGE FOR PURIFICATION SYSTEM PUMP PRIMING SHUTOFF
RP-28	CONTAMINATED WATER (RETENTION) TANK PUMP DISCHARGE SHUTOFF
RP-29	CONTAMINATED WATER (RETENTION) TANK PUMP DISCHARGE HOSE CONNECTION SHUTOFF
RP-30	FEEDWATER LOWER SPRAY RING SHUTOFF
RP-31	FEEDWATER UPPER SPRAY RING SHUTOFF
RP-32	EDUCTOR VALVE

BORON PUMP SYSTEM

BP-1	BORON TANK DRAIN SHUTOFF
BP-2	BORON TANK DRAIN HOSE CONNECTION SHUTOFF
BP-3	BORON PUMP BYPASS HOSE CONNECTION SHUTOFF
BP-4	BORON PUMP DISCHARGE SHUTOFF
BP-5	BORON PUMP SYSTEM CHECK
BP-6	BORON PUMP SYSTEM LOWER SPRAY RING SHUTOFF
BP-7	BORON PUMP SYSTEM UPPER SPRAY RING SHUTOFF

FIG. 16

VALVE DESIGNATIONS AND DESCRIPTIONS

Table 3

DESIGN VERSUS OPERATING CHARACTERISTICS

(300 kwe and 400 kwt)

Item	Description	Design Data	Operation Data
1.	Reactor Main Steam		
	a. Flow rate, lb/hr[a]	9020	8220
	b. Pressure, psig	300	305
	c. Temperature, °F	421	422
2.	Turbine Throttle Steam		
	a. Flow rate, lb/hr[a]	7500	6600
	b. Pressure, psig[b]	280	295
	c. Temperature, °F[b]	415	419
3.	Turbine Exhaust Steam		
	a. Flow rate, lb/hr[a]	7500	6600
	b. Pressure, in. Hg abs.[c]	5	10
	c. Temperature, °F	135	161
4.	Turbine Gland Seal Steam		
	a. Flow rate, lb/hr[d]	30	50
	b. Pressure, psig[b]	280	295
	c. Temperature, °F[b]	415	419
5.	Bypass Steam		
	a. Flow rate, lb/hr	1400	1450
	b. Pressure		
	1. Regulating valve inlet, psig[b]	280	295
	2. Regulating valve outlet, psig	40	45
	c. Temperature		
	1. Regulating valve inlet, °F[b]	415	419
	2. Regulating valve outlet, °F	338 (50°S.H.)	339 (50°S.H.)
6.	Space Heat Steam		
	a. Flow rate, lb/hr	1400	1380
	b. Heat exchanger pressure, psig[e]	15	19
	c. Heat exchanger temperature, °F	328 (76°S.H.)	328 (76°S.H.)
	d. Condensate		
	1. Flow rate, lb/hr	1400	1380
	2. Temperature, °F[c]	240	252
7.	Bypass Steam Excess		
	a. Flow rate, lb/hr[f]	zero	70
	b. Pressure, psig[g]	40	45
	c. Temperature, °F	338 (50°S.H.)	339 (50°S.H.)
8.	Main Steam Condenser		
	a. Condensing rate, lb/hr[a,f]	7500	6670
	b. Pressure, in. Hg abs.[c]	5	10
	c. Temperature, °F[c]	135	161
9.	Flash Tank Condensate		
	a. Flow rate, lb/hr[h]	1550	1550
	b. Pressure, in. Hg abs.[c]	5	10
	c. Temperature, °F[c]	135	161

Table 3 (Cont'd.)

Item	Description	Design Data	Operation Data
10.	Hot Well		
	a. Flow rate, lb/hr[i]	9020	8220
	b. Pressure, in. Hg abs.[c]	5	10
	c. Temperature, °F[c]	130 (5° sub-cooled)	154 (7° sub-cooled)
11.	Rod Drive Seal Water Flow Rate (5 drives), lb/hr	400	400
12.	Feedwater		
	a. Flow rate, lb/hr[a]	9020	8220
	b. Temperature		
	1. From hot well, °F[c]	130	154
	2. To reactor (after regenerative heating), °F[j]	175	173
13.	H. P. Condensate System		
	a. Flow rate		
	1. Vent, lb/hr[k]	zero	45
	2. Liquid, lb/hr[k]	215	170
	b. Pressure, in. Hg bar.[l]	-3	-4.5
	c. Temperature, °F	206	206
14.	Air Ejector System		
	a. Steam flow rate		
	1. H.P. condensate system, lb/hr	20	20
	2. Condenser system, lb/hr	100	100
	b. Steam pressure, psig[b]	280	295
	c. Steam temperature, °F[b]	415	419
	d. Vapor flow rate		
	1. H.P. condensate system, lb/hr	80	80
	2. Condenser system, lb/hr	160	160
	e. Vapor pressure		
	1. H.P. condensate system, in. Hg bar.[e]	-3	-4.5
	2. Condenser system, in. Hg abs.[c]	5	10
	f. Vapor temperature		
	1. H.P. condensate system, °F	145	145
	2. Condenser system, °F	110	110
	g. Aftercondenser water temperature		
	1. Inlet, °F[c]	130	154
	2. Outlet, °F	156	175
	h. Precooler water temperature		
	1. Inlet, °F	156	175
	2. Outlet, °F[m]	162	174
	i. Drains to precooler, lb/hr	150	150
	j. Condensate flow rate		
	1. Aftercondenser, lb/hr	145	145
	2. Precooler, lb/hr	70	70
	k. Condensate temperature		
	1. Aftercondenser, °F	206	206
	2. Precooler, °F	206	206
15.	Reactor Water Purification System		
	a. Reactor water flow rate, lb/hr[n]	5	1.7
	b. Reactor water temperature		
	1. Cooler influent, °F	421	422
	2. Cooler effluent, °F[n]	170	152

Table 3 (Cont'd.)

Item	Description	Design Data	Operation Data
16.	Reactor Shield Cooling System		
	a. Coolant flow rate, lb/hr	5000	5000
	b. Coolant temperatures		
	1. Cooler inlet, °F(o)	190	165
	2. Cooler outlet, °F	150	157
17.	Space Heat Load Water System(p)		
	a. Flow rate, gpm	135	128
	b. Temperatures		
	1. Heat exchanger inlet, °F	150	167
	2. Heat exchanger outlet, °F	170	192
	c. Heat load, kw	400	397

NOTES: (a) The rate of steam flow in the operational reactor is less than that designed because the actual overall rate of flow of steam through the turbine is 22 lb/(kw)(hr) as compared with the predicted value of 27.33 lb/(kw)(hr). The 20 psig increase in steam pressure over the rated pressure of 275 psig accounts for approximately 0.8% decrease in the steam rate. Also, the temperature increase of 8°F over the rated temperature accounts for approximately 0.6% decrease in the steam rate. The decrease in the main steam condenser vacuum from the design of 5 in. Hg abs to 10 in. Hg abs accounts for approximately a 10% increase in the turbine steam rate. This agrees closely with the turbine performance guarantee of 20.65 lb/(kw)(hr) and an overdesign of approximately 7 lb/(kw)(hr).

(b) The operational turbine throttle pressure is higher than the design pressure because of the deletion of a steam separator and a reactor pressure-regulating valve. The temperature change corresponds to a change in the saturation temperature.

(c) The operational turbine exhaust vacuum is higher than the design because the main steam condenser efficiency is lower than predicted. The saturation temperature also changes to correspond to the operating vacuum.

(d) The operational turbine gland seal steam flow rate is greater than the design value because the vacuum to which the turbine gland seals discharge is operating at -4.5 in. Hg bar instead of the predicted -3 in. Hg bar. This is due to the higher efficiency of the air ejector.

(e) The steam pressure of the operational space-heating heat exchanger is higher than design because the 400-kw heat load could not be maintained at a lesser pressure.

(f) The operational bypass steam excess is for control convenience.

(g) The operational bypass steam pressure is 45 psig instead of 40 psig because of pressure fluctuations and the possibility of decreasing the steam pressure to the space-heating system during a turbine load increase.

(h) The flash tank condensate flow rate is defined here as the difference between the reactor main steam flow and the condensing rate in the main steam condenser. Systems such as the condensate circulating system to the air-ejector aftercondenser, precooler, and reactor shield-cooling system are construed as closed loops with the flash tank as an intermediary and are disregarded.

Table 3 (Cont'd.)

(i) The hot well flow rate is defined as the reactor main steam flow. The condensate circulating system is construed as a closed loop and is disregarded. The rod drive seal water system is also a closed loop for one-half of the flow rate. The other half of the flow rate enters the reactor at the control rod drives. Plant makeup water is added to the hot well.

(j) The operational feedwater flow regenerative temperature varies directly with the rate of flow of water in the reactor purification system.

(k) The H.P. condensate system flow rate through the vent was construed as a closed loop for all intent and purpose. The effluent from the rod drive cooling system is disregarded.

(l) The operational H.P. condensate system vacuum is lower than the design value because of a more efficient air ejector.

(m) The operational precooler temperature outlet is cooler than the inlet due to the entering low temperature drain water.

(n) The operational reactor water purification system flow rate was set at the minimum rate consistent with efficient water cleanup. The cooler outlet water is dependent upon flow rates up to the shutoff setting of the temperature regulating valve.

(o) The operational shield cooling system cooler inlet temperature is lower than the design value due to the inclusion of conservative shield radiation heating calculations.

(p) The operational space-heating load water temperatures are dependent upon the flow rate which was selected as a matter of convenience.

Experimentation resulting in a system which does not utilize the regulating valve during load changes is described in section XI-B-1-c. The valve is presently used to regulate the supply of steam to the simulated heat-load heat exchanger. A second valve (0-10,000 lb/hr) functions as a manually operated pressure-relief valve during a turbine throttle trip incident or when full-power reactor operation is desired without the use of the turbine-generator unit. With the 0-10,000-lb/hr valve closed, the 0-3000-lb/hr valve functions to compensate for changes in plant load when the reactor power is being controlled manually.

The turbine exhaust steam is condensed in the main air-cooled steam condenser. The disagreement between design and operating conditions is believed to be attributable to the condenser and is discussed in section IX-B. The piping (see Fig. 17) also provides a means to discharge steam to the condenser from the bypass steam system, flash tank, and pressure-relief valves.

The turbine gland seal system, air ejectors, and space-heating system (see Fig. 17) are discussed in section IX.

2. Condensate Systems

The condensate resulting from the condensing of the steam and vapors from various systems is collected in the plant hot well (see Fig. 18).

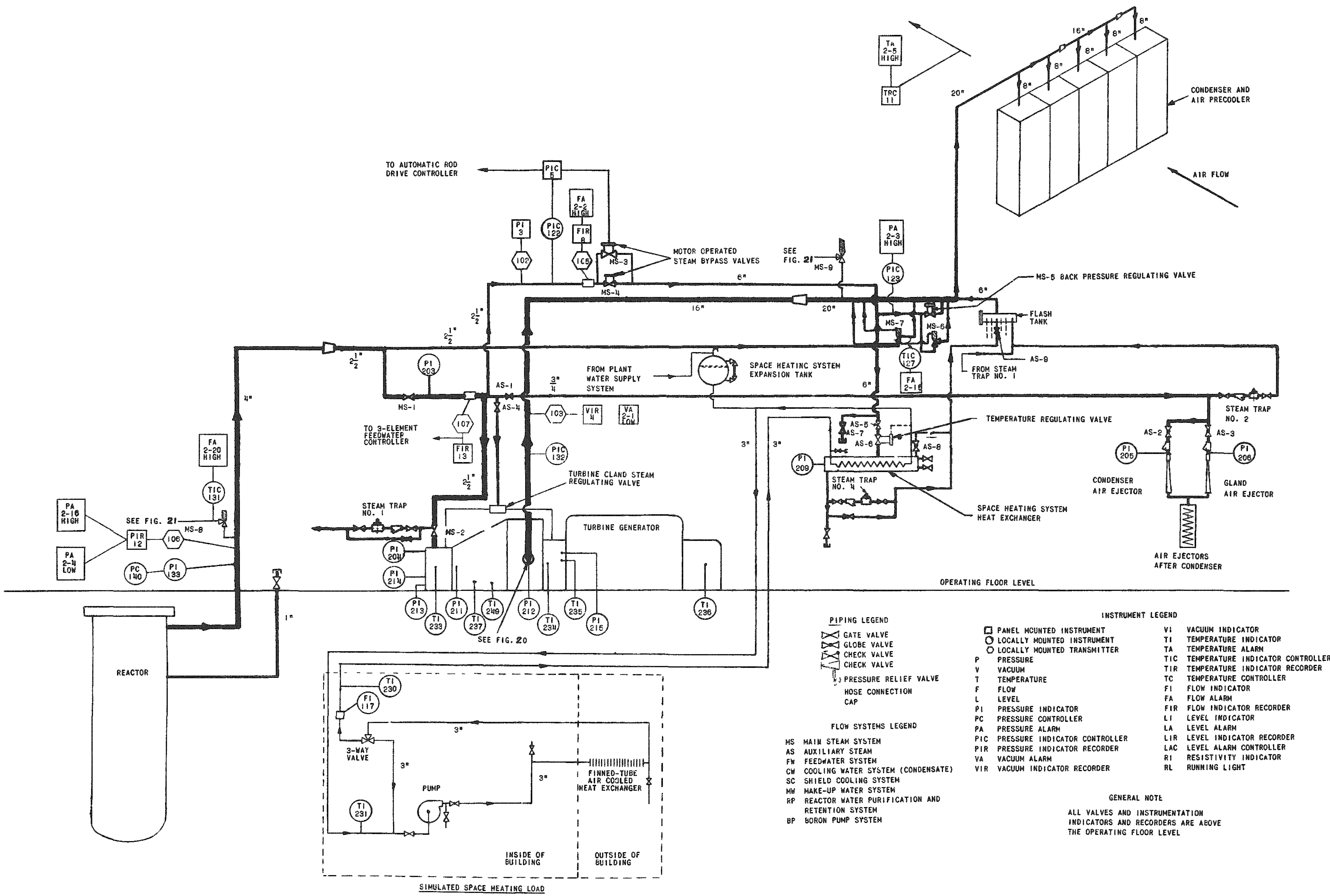


FIG. 17
STEAM SYSTEMS AND SPACE HEATING FLOW DIAGRAM

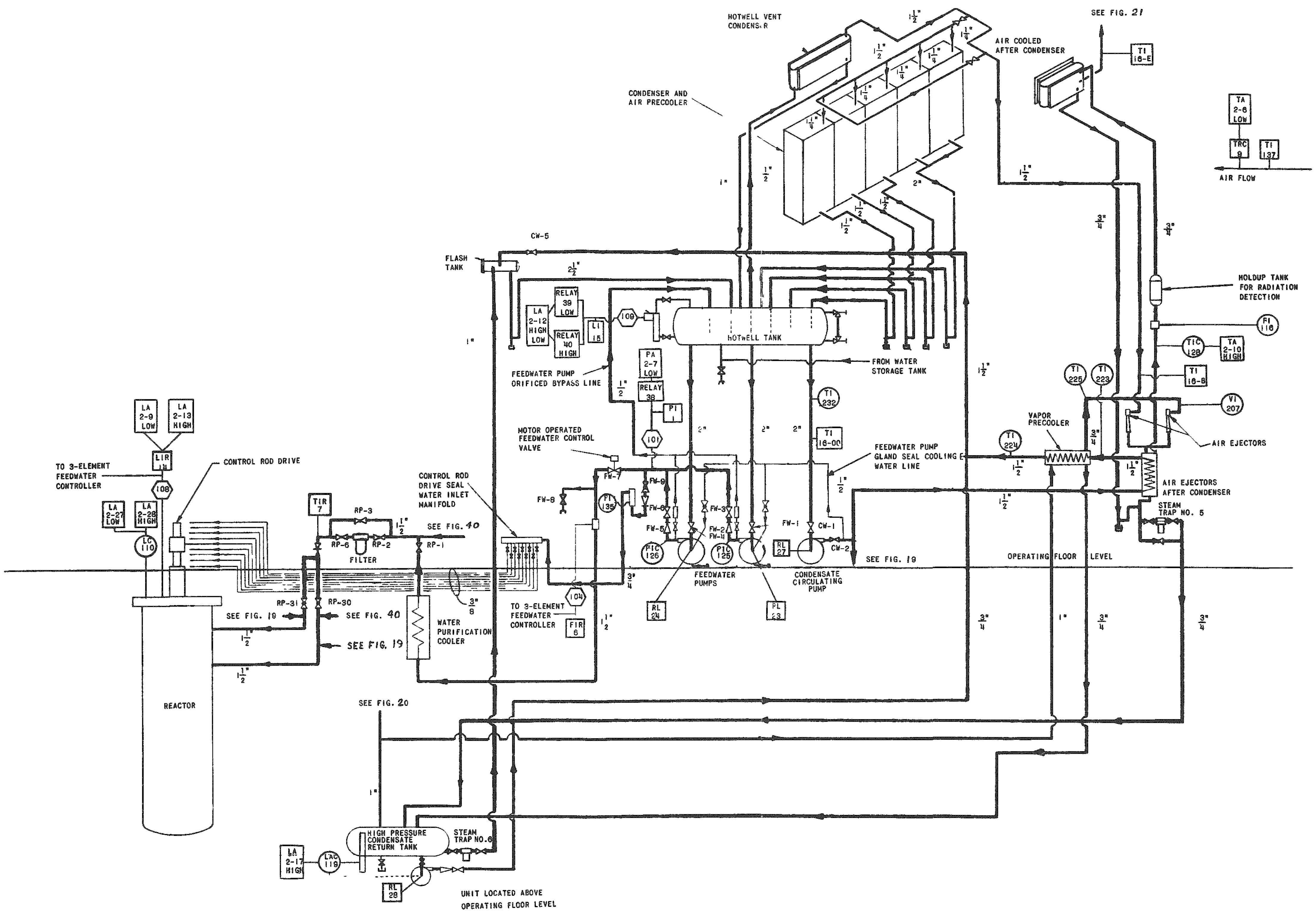


FIG. 18
FEED WATER, COOLING WATER (CONDENSATE), AIR EJECTORS, CONDENSER, ROD DRIVE COOLING AND DRAINS SYSTEMS FLOW DIAGRAM

The temperature of the condensate in the hot well is dependent upon the main steam condenser vacuum. One of the two feedwater pumps returns the condensate to the reactor as feedwater (see section IX-C). Before entering the reactor, the feedwater is filtered and preheated to approximately 175°F, which, when mixed with the reactor downcomer water at 421°F, results in a subcooling of approximately 2°F at the entrance of the fuel channels. The preheating is accomplished by heat exchange with the reactor purification water from the reactor (see section VI-A). Another function of the feedwater and pump is to provide a sidestream flow of water coolant to cool and flush the rotating break-down seal of the control rod drives.

A secondary system (see Fig. 18) using condensate provides the coolant for the air-ejector aftercondenser, precooler, and the reactor shield-cooling system (see Fig. 19). The air-ejector aftercondenser and precooler are discussed in section IX-B. The operation of the shield-cooling system to date has indicated that forced-coolant circulation is not necessary and that natural circulation cools the external thermal shield sufficiently.(9) A canned-rotor pump is used to circulate the condensate. The coolant is returned to the hot well by way of the flash tank. Another function of the system is to provide a water seal and coolant to the feedwater pump gland seals (see Fig. 18). This water is not recovered.

The condensate-return systems (see Figs. 18 and 20) are the collection systems for the plant. The LP (low-pressure) system services the turbine drains and exhaust piping only. The receiver tank is open to the turbine exhaust line to which it can flash steam. A canned-rotor pump pumps the liquid to the flash tank.

The HP (high-pressure) system for condensate return receives the drainage from the pressure vessel vapor leak-off gasket groove, turbine gland seals, control rod drive seals, air-ejector after-condenser, and vapor precooler. The receiver tank is subject to evacuation by the air ejector with the precooler as intermediary. The liquid condensate is discharged to the lower-pressure flash tank by means of a steam trap. Condensate in excess of the steam trap capacity is pumped to the flash tank by a canned-rotor pump.

3. Auxiliary Systems

The following systems in Fig. 19 are discussed elsewhere as noted: Boric Acid Injection System - section VI-B; turbine lubricating oil system - section IX-A (Turbine-Generator Unit); Primary Water Makeup System - section IX-D; and Space-heating System - section IX-E.

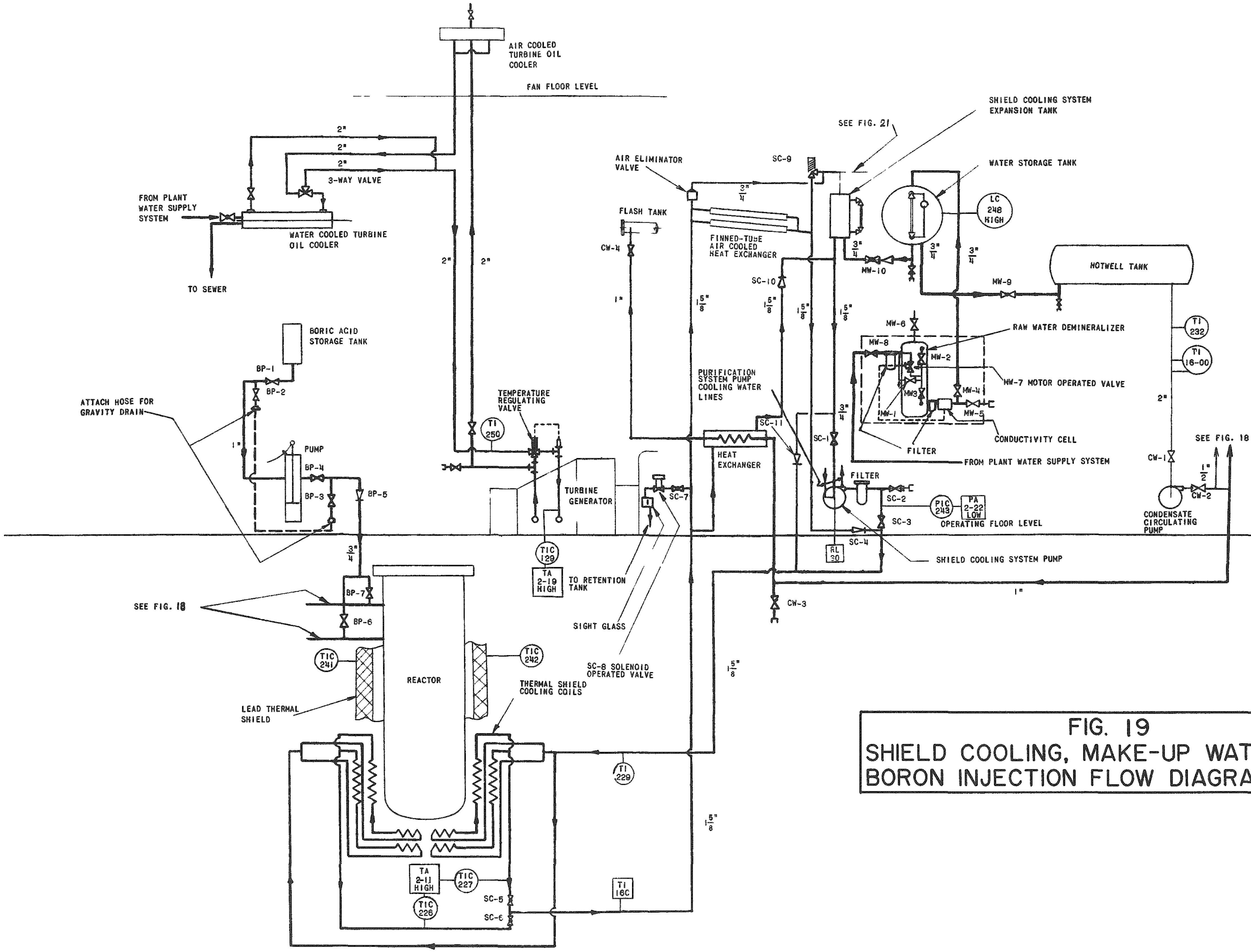


FIG. 19
SHIELD COOLING, MAKE-UP WATER AND BORON INJECTION FLOW DIAGRAM

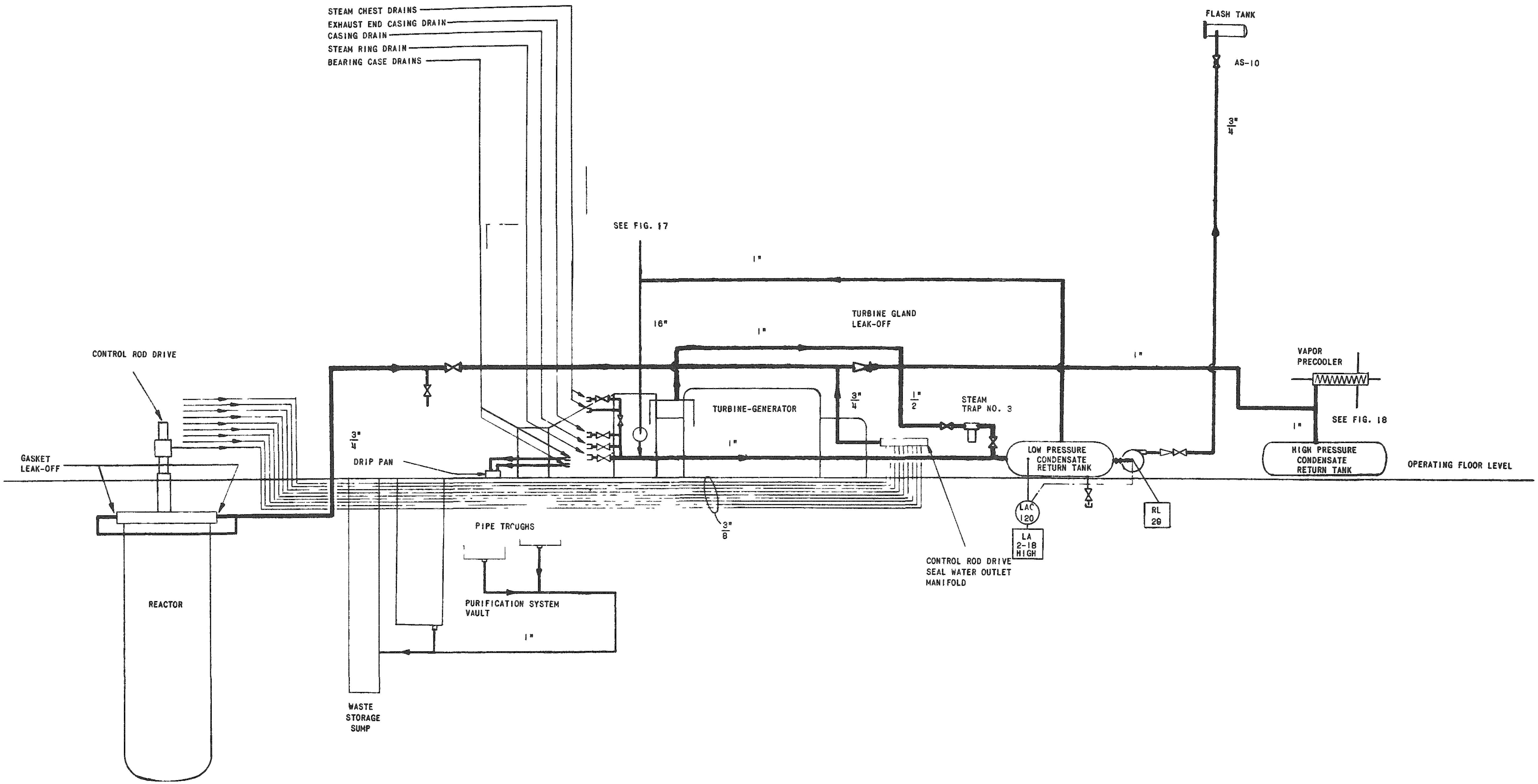


FIG. 20
MISCELLANEOUS DRAINS AND VAPOR LEAK-OFF FLOW DIAGRAM

4. Pressure Relief and Safety Valves

The main steam line (see Fig. 21) incorporates a pressure-relief valve (throttling) and a safety valve (popping). The former is set to open at 350 psig and discharges to the turbine exhaust line. The safety valve is set to open at 385 psig and discharges to atmosphere. The turbine exhaust line incorporates a condenser pressure-relief valve which is set to open at 5 psig and also discharges to the atmosphere.

B. Plant Startup and Shutdown[17,18]

No attempt is made to preheat the reactor for startup purposes. Heating of the reactor is accomplished with reactor power while limiting the period to not less than 30 sec and a heating rate of not more than 3°F/min. When at a steady heating power (~10% full power), the period meter is bypassed. Auxiliary equipment and systems are started as needed.

After the operating pressure (300 psig) is reached, reactor power is increased to yield a steady-state steam production of 1500 to 2000 lb/hr through the bypass steam system, with the 0-3000-lb/hr regulating valve on "automatic" operation and the 0-10,000-lb/hr valve closed. The turbine is then started and brought to speed. During this procedure, the reactor power is increased by manual withdrawal of the control rods in order to maintain 1500 lb/hr of bypass steam. The turbine governor acts to control the turbine speed.

When at steady-state power, the rate of feedwater is regulated to correspond with the rate of main steam flow. The feedwater-regulating valve is then placed on "automatic." With the entire plant operating satisfactorily, the bypass steam-regulating valve is placed on "manual" and the reactor is put on "automatic demand control."

Following this sequence, the electrical and space-heat loads may be added.

Routine plant shutdown is accomplished by first removing the electrical and space-heat loads. During this procedure, the reactor power is reduced automatically to maintain the preset pressure of 300 psig. When at steady-state power and at a condition of no-load, the bypass steam valve control is switched to "automatic" and the reactor control to "manual" operation. Reactor power is then reduced to a power level corresponding to a bypass steam flow rate of 500 lb/hr. Before placing the feedwater-regulating valve in "manual" operation, the rate of feedwater flow should correspond to the rate of main steam flow at a reactor water level of zero in. (the datum line). (This water level is maintained during plant shutdown and the level is gradually increased to +12 in. during the time the reactor is being depressurized.)

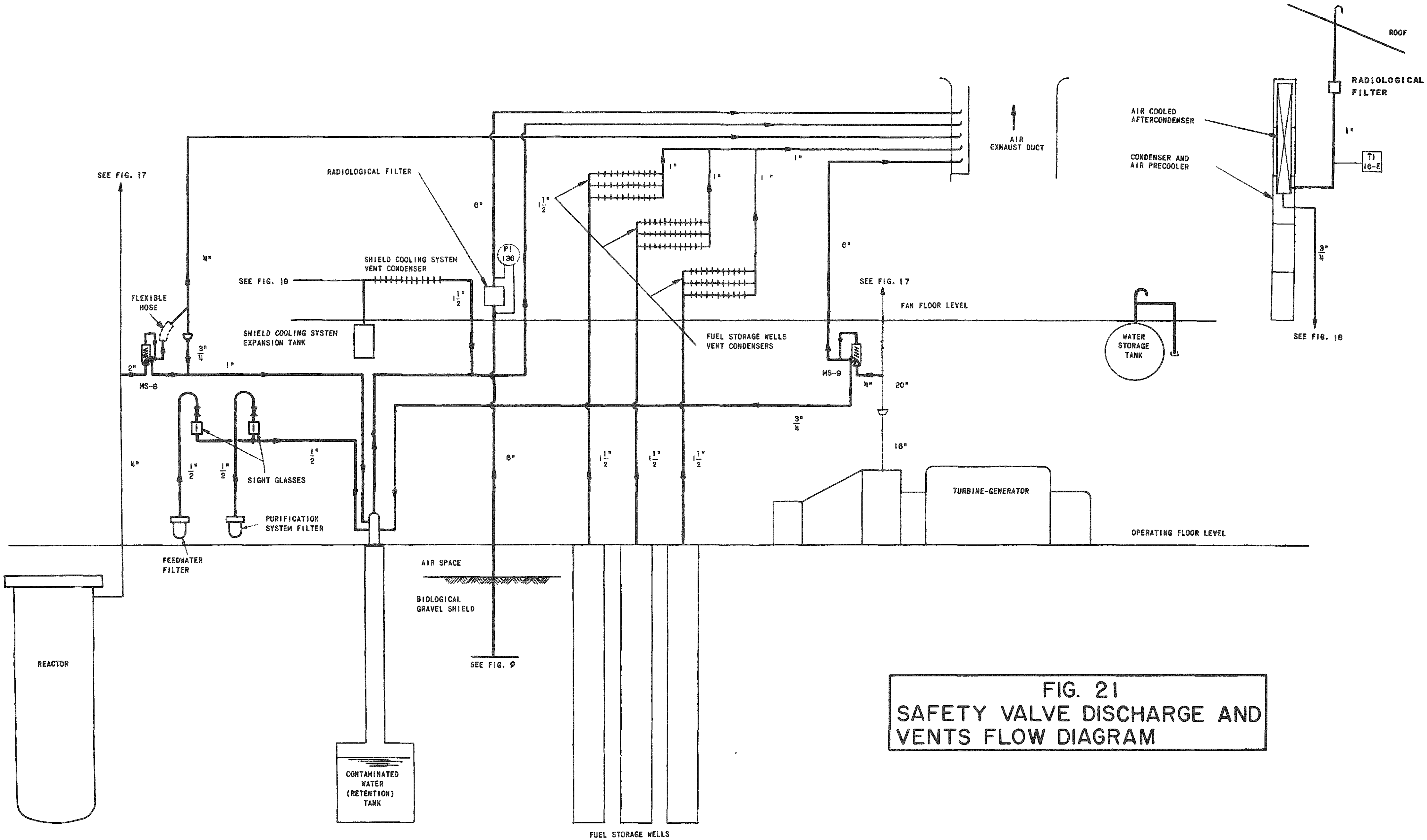


FIG. 21
SAFETY VALVE DISCHARGE AND VENTS FLOW DIAGRAM

With the bypass steam flow rate at 500 lb/hr, the turbine is secured. Shutdown of the reactor is effected by dropping all rods with the reactor "shutdown button." The reactor is then depressurized by blowing steam to the condenser at a rate which produces a water temperature drop of 3°F/min. Other items of equipment are then secured at times dictated by their operation functions.

In the event of a reactor scram, plant power from the turbine-generator unit is lost and the emergency power system automatically switches to battery power. The steam bypass and feedwater valves close automatically to prevent rapid loss of reactor pressure. The plant power is restored by switching to the diesel-electric or the NRTS distribution system. The plant is restarted as previously discussed except for the pressurizing of the reactor, since only the mass steam flow was lost in the incident.

V. REACTOR COMPONENTS

A. Core*

In the light of favorable experimental work with BORAX I, II, and III, it was decided to use aluminum plate-type fuel elements, assembled into approximately 4 x 4-in. compartments, for the ALPR. The "meat" of the fuel plate is similar to that of the BORAX plate, that is, an alloy of aluminum with highly enriched uranium (~93 w/o U^{235}). One very significant difference is that 2 w/o nickel has been added to the aluminum to increase the resistance of the alloy to corrosion by the boiling water.

Early experimental results indicated that the addition of 1 w/o nickel to aluminum increased the corrosion resistance (see Appendix IV). The choice of this aluminum-nickel alloy (type X8001) for the fuel plate cladding was substantiated by additional corrosion research,[21] much of it with water at higher temperatures and pressures than the ALPR conditions. Since a nominal core life of three years is expected, a cladding thickness of 0.035 in. was selected. The alloy was also used for the core components because it is inexpensive, adequately strong, easy to fabricate, and appears to have excellent resistance to corrosion. In addition, the nuclear properties compare favorably with other feasible structural materials.

The X8001 aluminum-nickel (formerly designated M-388) alloy composition was developed by the Laboratory and is given in Table 4.

Table 4

X8001 ALUMINUM-NICKEL ALLOY COMPOSITION

Element	Weight Per cent	Element	Weight Per cent
Ni	0.9-1.1	Ti	0.01-0.02
Cu	0.00-0.15	Pb	0.01 max.
Si	0.00-0.15	Sn	0.01 max.
Fe	0.4-0.6	B	0.001 max.
Mn	0.02 max.	Li	0.001 max.
Mg	0.02 max.	Cd	0.001 max.
Zn	0.02 max.	Co	0.001 max.
Ce	0.01 max.		

*G. C. Milak

The entire core is fabricated of aluminum-nickel alloy with the exception of certain minor items, such as the fuel assembly gripper pin, core hold-down bolts, and spacer springs, which are made of stainless steel. Figure 22 is a plan view of the core and Fig. 23 represents the core vertical section. Design data for the core loading, fuel assembly, and fuel plate are noted in Table 5.

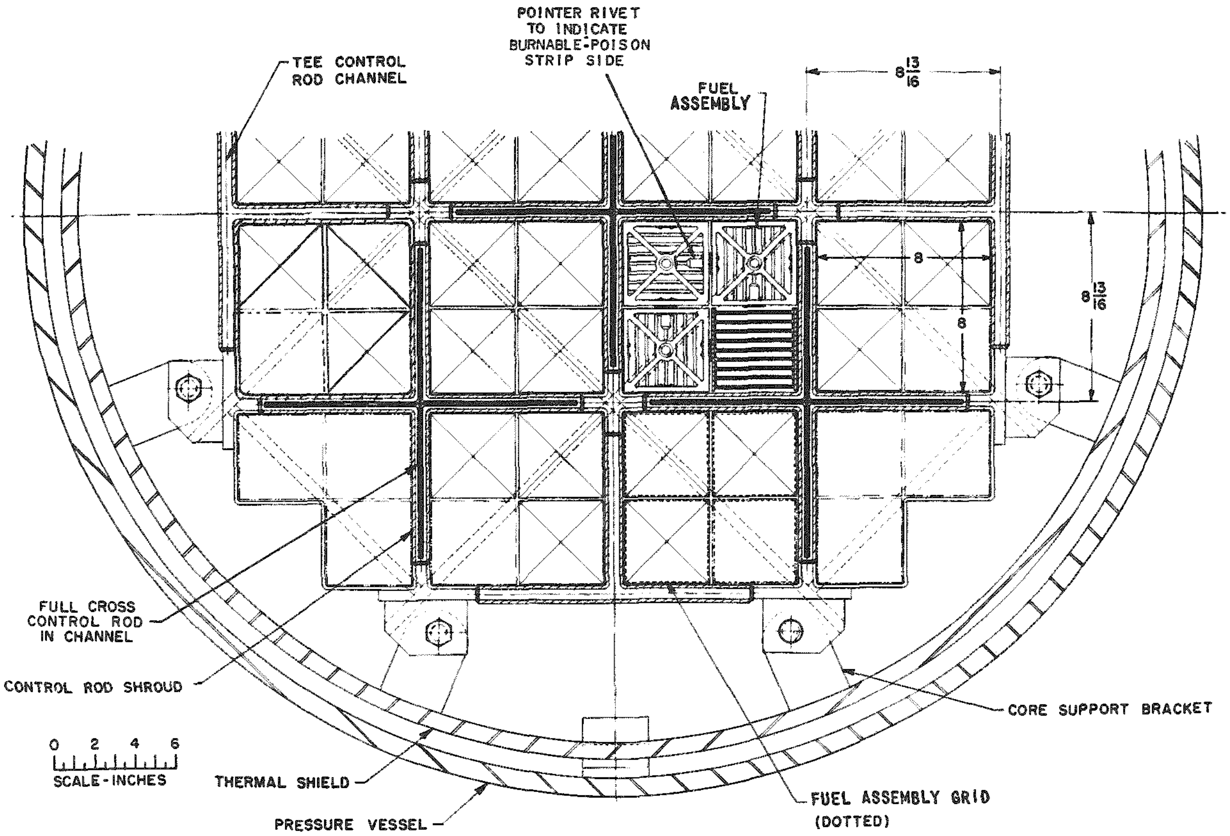


FIG. 22
CORE PLAN, SECTION

1. Fuel Assembly*

The nuclear and mechanical design of the fuel assembly(22) for the nominal three-year operating core lifetime resulted in a core structure consisting of sixteen compartments (fuel assembly cells) containing a maximum loading of fifty-nine fuel assemblies plus one source assembly.

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*G. C. Milak

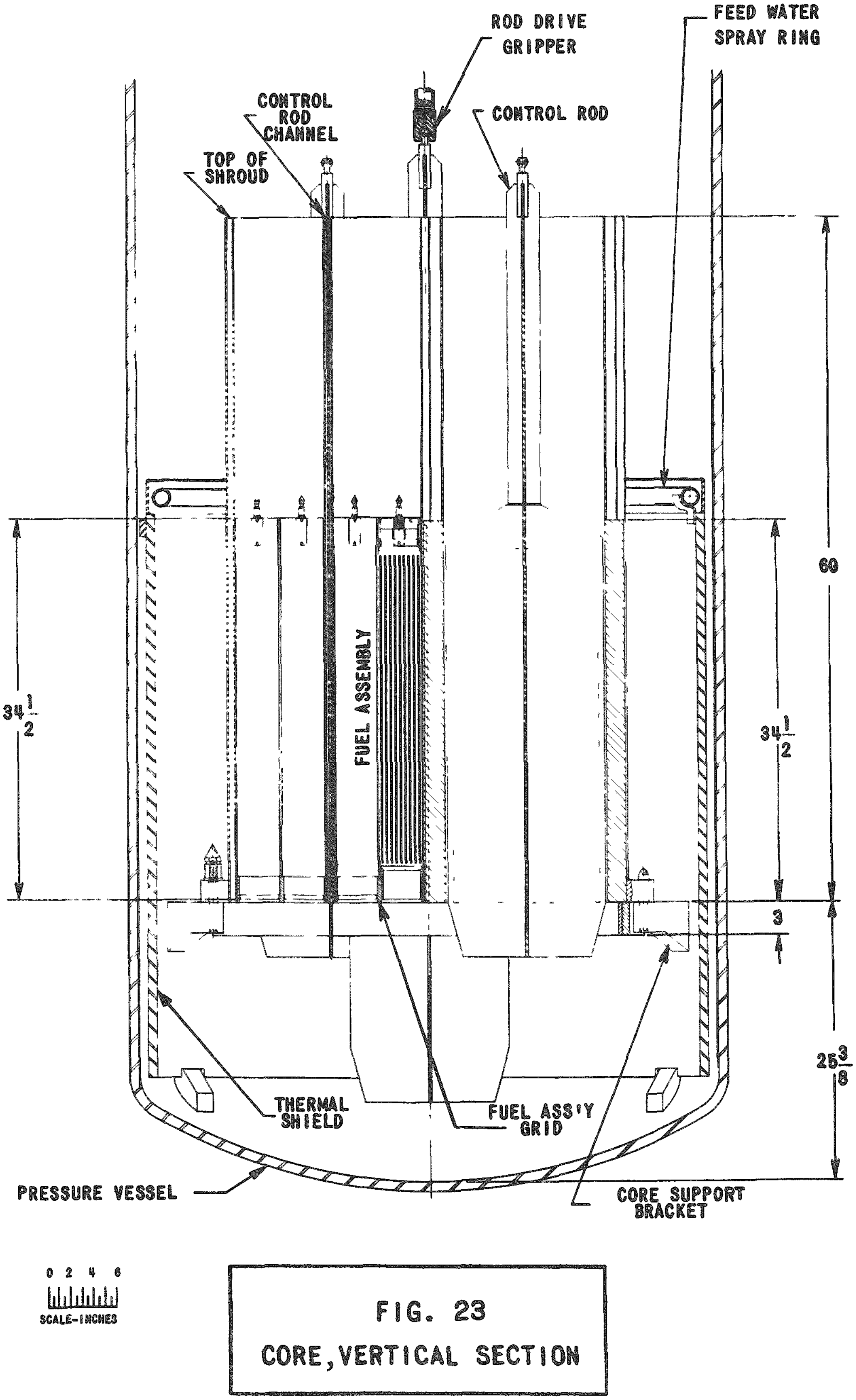

FIG. 23

CORE, VERTICAL SECTION

Table 5

CORE DESIGN DATA

Core	
Length of Active Core	25.800 in.
Maximum Horizontal Cross Section	~35 in. x 35 in. (corners removed)
Fuel Loading of Uranium	
40-assembly, U^{235} (minimum, poisoned)	14.00 kg
59-assembly, U^{235} (maximum, poisoned)	20.65 kg
Overall Metal-to-water Ratio	0.5
Core Heat Transfer Area (40-fuel assemblies)	~475 ft^2
Loading of Burnable-poison (40-fuel-assembly core)	
Number of 0.021-in.-thick strips	16 half-length
Number of 0.026-in.-thick strips	40 full-length
Fuel Assembly	
Overall Dimensions	$3\frac{7}{8}$ in. x $3\frac{7}{8}$ in. x $34\frac{1}{2}$ in.
Weight of U^{235}	350 gm
Number of Plates	9
Average Water Channel Gap	0.310 in.
Fuel Plate	
Overall Dimensions	0.120 in. x $3\frac{25}{32}$ in. x $27\frac{13}{16}$ in.
"Meat" Dimensions	0.050 in. x 3.500 in. x 25.800 in.
Cladding Thickness	0.035 in.
Cladding and "Meat" Diluent	Aluminum-nickel alloy
"Meat" Volume	74 cm^3
Weight of Al-Ni Alloy "Meat" Diluent	~195 gm
Weight of U^{235}	38.9 gm
Weight of Uranium (93 w/o U^{235})	41.7 gm
Weight-per cent Uranium in the "Meat"	~17.60
Atom-per cent Uranium in the "Meat"	~2.21
Burnable-poison Strips	
Overall Dimensions (full length)	$3\frac{7}{8}$ in. x $25\frac{13}{16}$ in. with thicknesses of 0.021 in. or 0.026 in.
Weight B^{10} per strip	
0.021-in. thickness	0.4 gm
0.026-in. thickness	0.5 gm
Weight-per cent B^{10} in strip	~0.43

Each fuel assembly (see Fig. 24), $34\frac{1}{2}$ in. long and $3\frac{7}{8}$ in. square, contains ~350 gm U^{235}. An assembly is composed of a lower end (locating) fitting, nine flanged fuel plates, two side plates, and a top end fitting equipped with a stainless steel gripper pin. The flanged edges secure correct spacing of the plates. The fuel plate includes a 0.050-in.-thick center portion ("meat") of an alloy of uranium, enriched to ~93 w/o U^{235}, with aluminum containing 2 w/o nickel; the "meat" is clad with 0.035-in.-thick aluminum containing 1 w/o nickel (type X8001). The "meat" is 3.500 in. wide and 25.800 in. long. The finished clad plate width is 3.710 in. wide (exclusive of the flanges) and $27\frac{13}{16}$ in. long. The top end fitting has four rectangular slots through each of which is fitted a flat-type stainless steel spring insert. These springs provide proper fuel assembly spacing and vertical alignment in the core.

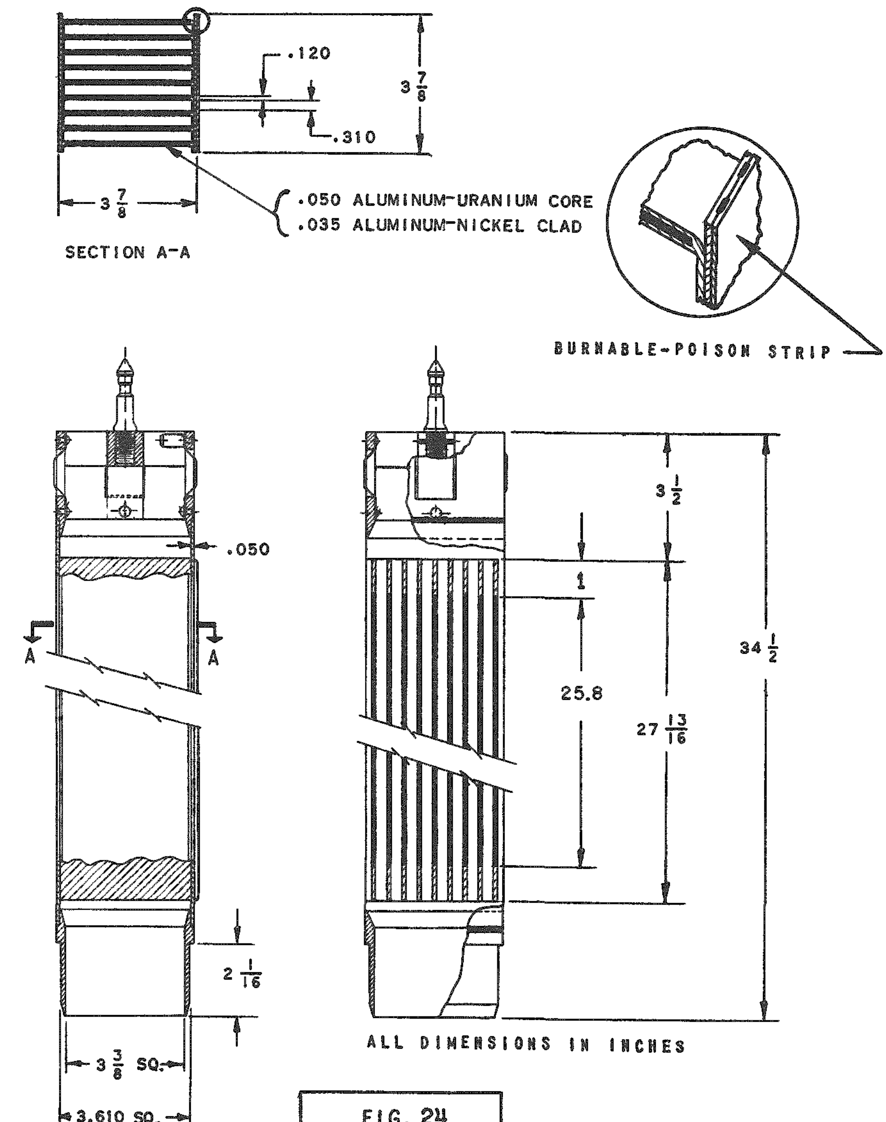

FIG. 24
FUEL ASSEMBLY

Thin strips of an extruded aggregate of powders of aluminum, type X8001, and of boron, highly enriched in B^{10}, are fusion welded to the edges of one or both side plates of the fuel assembly. Each strip is $25\frac{13}{16}$ in. long, the approximate length of the active core ("meat"), and $3\frac{7}{8}$ in. wide, which is the width of the fuel assembly side plate. One full-length strip, containing nominally 0.5 gm B^{10}, is welded to a side plate of each fuel assembly of the reference core; one half-length strip (containing ~0.2 gm B^{10}) is welded to the bottom half of the other side plate of each of the central sixteen fuel assemblies.

Dummy fuel assemblies are made from a one-piece, hollow, rectangular extruded section of X8001. Both ends are machined to the geometry of the fuel assembly and are equipped with a stainless steel gripper pin. The dummy assemblies are placed in the vacant fuel assembly positions of partially filled assembly cells to provide lateral support for the assemblies. Figure 12 shows the 40-fuel-assembly core loading for the reference design, dummy assembly positions, and the extra fuel assembly positions.

2. Core Support Lattice*

Since the reactor was designed to operate with natural convection of the water coolant, the core support lattice and shroud were constructed so as to offer a minimum of restriction to the recirculation of the coolant. The structure of the core support is of an interlocking and welded bar construction, consisting of four 42-in. lengths of aluminum-nickel extruded bar stock alloy, 3 in. wide and $\frac{1}{2}$ in. thick (see Fig. 25). The bars are slot machined to interlock on a 90-degree crisscross arrangement, after which all junctions are welded. An additional bar is welded across the four pairs of open ends in the lattice to supply additional rigidity and load-bearing strength.

A system of four cross-shaped and eight tee-shaped stanchions provide the shroud support. The cross-shaped stanchions are slot-machined to fit directly over the interlocking junction of the lattice. The eight tee-shaped stanchions are welded to the lattice. This construction provides the main structural skeleton for the core support and shroud. The eight corners of the lattice are then welded to eight blocks, or support pads, in which holes have been drilled to accommodate six hold-down bolts and two alignment pins. The lattice pads serve the purpose of attaching the entire core shroud to the support brackets of the pressure vessel thermal shield (see Fig. 23).

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*G. C. Milak

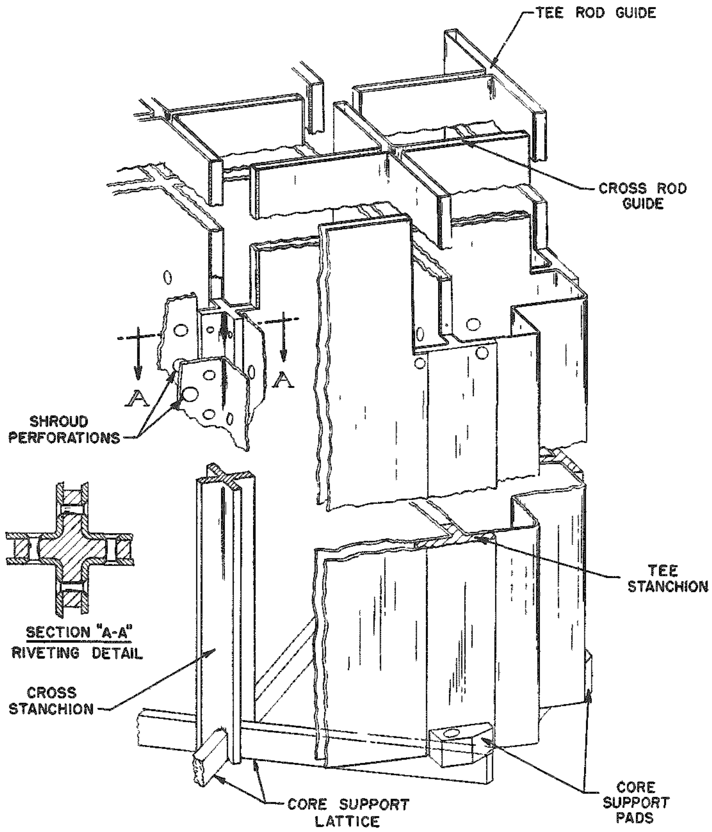


FIG. 25
QUADRANT SECTION CORE SUPPORT AND SHROUD

3. Core Shroud*

The core shroud consists primarily of $\frac{5}{32}$-in.-thick aluminum-nickel alloy (type X8001) sheet stock, 62 in. in length and 16 in. in width. The sheet stock is press-brake formed in length-wise angles. Equally spaced holes of 2 in. diameter are punched in each section of the inner shroud to provide a more efficient water-circulation system within the shroud. Each hole is slightly chamfered to eliminate sharp edges which might cause "hangup" of the control rods or fuel assemblies. The (four) outer shell plates of the enclosure are not perforated, and thereby produce a "chimney" effect beneficial to natural circulation. The use of angle shapes provides for an easily assembled structure which is divided into cells for fuel assemblies. The individual cells are formed by flush-riveting the angles to the vertical cross and tee stanchions. Riveting was used in lieu of welding to eliminate post-assembly cleanup other than degreasing and to eliminate the thermal stresses and warping associated with welding. The upper section, above the active core region, required

*G. C. Milak

the welding of thin strips to define this portion of the control rod shrouds. Each of the twelve cells thus formed is of square cross section to accommodate four fuel assemblies each, and the four corner cells are contoured to contain three fuel assemblies each. At the same time, the sides of these cells serve as shroud for the lower section of the control rod channel. There are five cross-type control rod channels and four tee-type control rod channels.

Welded at the bottom and on the inside of each fuel assembly cell is an egg-crate-shaped fuel assembly support consisting of an interlock and welded cross around which a rectangular-formed band is welded. These units have been designed to contain and act as a support for the bottom end fitting of a fuel assembly.

The core support and shroud is inserted into the vessel with a simple four-point sling. The component rests on and is fastened to, an arrangement of eight symmetrically located stainless steel support brackets welded to the thermal shield. Each of six of these support brackets has been drilled and tapped to take a 17-4 PH stainless steel, $\frac{5}{8}$-in. diameter hold-down bolt. These bolts are inserted through the core support pads and are fastened to the support brackets. The remaining two brackets are drilled and tapped for two special tapered alignment pins which fit through the support pads. The pins provide proper alignment and orientation of the core support and shroud.

Space limitations between the thermal shield and the shroud structure necessitated the use of a special hold-down bolt and bolt tool (see Fig. 26). The tool (a tee-handle socket wrench) consists of a tube, with a tee-shaped handle, through which a rod (bolt retainer), threaded on one end, is inserted. The other end of the bolt retainer has a winged nut. The bolt head is inserted into the socket of the wrench and is engaged by the bolt retainer. The wrench with the bolt is lowered into the vessel and a few threads of the bolt are engaged with the thermal shield support bracket. The bolt retainer is then detached from the bolt head and the bolt is then tightened with the socket wrench. The wrench is then removed. Another bolt is then attached within the wrench socket and the procedure is repeated until all six hold-down bolts have been inserted and tightened.

The core support lattice and shroud, and the fuel-handling equipment (coffin, gripper mechanism, viewers, etc.) were constructed in the Central Shops Facility of the Laboratory.

The use of extruded cross and tee-shaped vertical stanchions represented a very significant savings in delivery time and overall fabrication costs. Since the alloy X8001 was not available in "finished" form (sheets, bar stock, etc.), billets were purchased and the necessary dies procured. The machining and welding that would have been required if bar stock had been used in the fabrication of the crosses and tees was almost completely eliminated because the extrusion tolerances were held closely. The use of these extrusions also resulted in a negligible amount of overall twist in the vertical shroud sections. As a result, the core shroud and support was kept straight within $\frac{1}{32}$ in. in a length of 60 in.

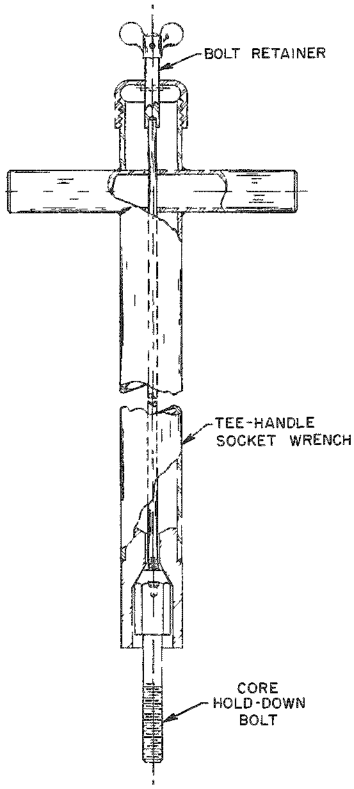


FIG. 26
CORE HOLD-DOWN BOLT AND BOLT TOOL

4. Control Rods*

Two different control rod shapes are included in the core configuration (see Fig. 12). During part of the critical experimentation, both the five cross and four tee-shaped control rods were used.

Figure 27 illustrates a typical cross-shaped rod. The five cross rods are $14\frac{1}{4}$ in. wide across the blade edges, with approximately 14 in. of cadmium absorber. The four tee-shaped rods are 12 in. wide with a perpendicular leg that is $7\frac{3}{16}$ in; the cadmium absorber is approximately $11\frac{1}{2}$ in. by 7 in. The center cross rod, an off-center cross rod, and a tee rod weigh approximately 48, 42 and 37 lb, respectively.

*W. J. Kann

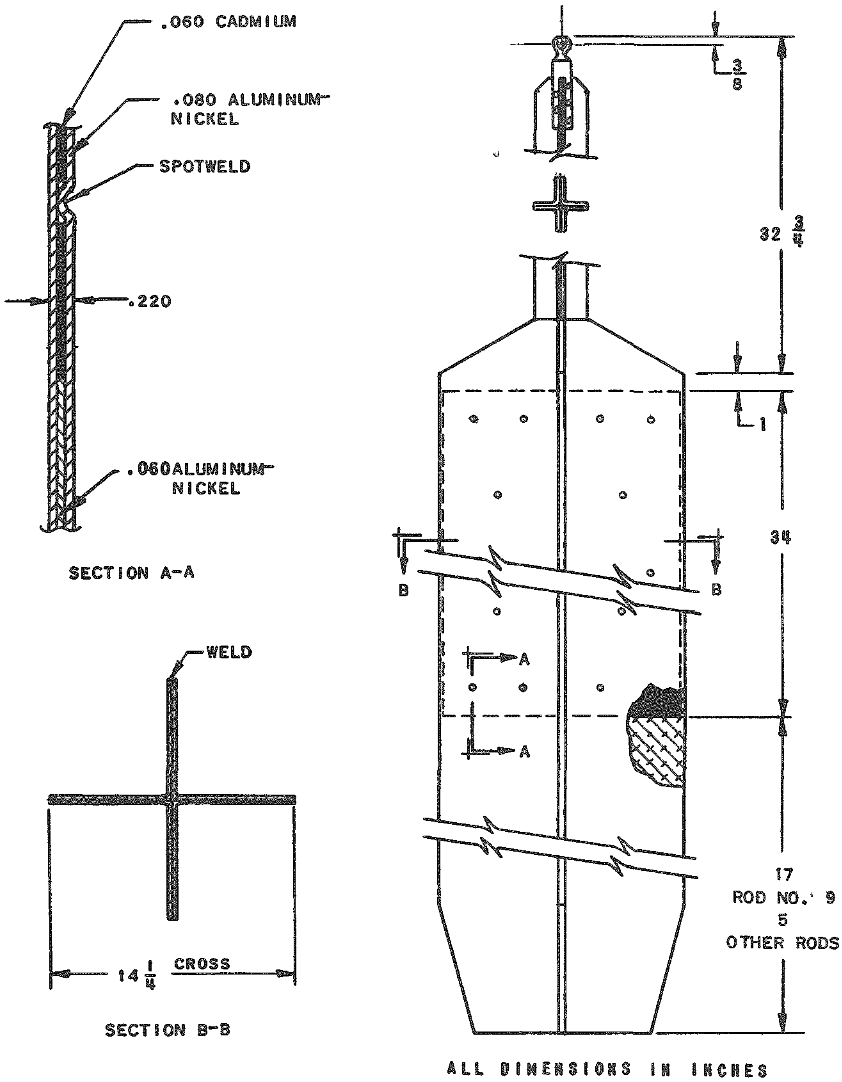


FIG. 27
CROSS-TYPE CONTROL ROD

A sandwich-type construction was used to fabricate all rods. No special jigs, fixtures, or processes were necessary. Only standard commercial machine shop techniques of forming, welding, etc., were used during manufacture.

For each cross rod, four sheets of 0.080-in.-thick type X8001 aluminum-nickel alloy were formed into angles. The active portion of each rod, a 0.060-in.-thick by 34-in. long cadmium sheet, was inserted between the aluminum side plates. Filler pieces of 0.060-in.-thick aluminum were used at the upper and lower ends of the rod. Alternate side plates were dimpled (in the cadmium area) and holes for mating were drilled in the cadmium. This assembly was then edge-welded and spot-welded together. A 3-in. cross-shaped extension was attached to the upper end of the rod. The extension was fabricated from four $\frac{1}{8}$-in. bars formed into angles and spot-welded together. A ball joint end fitting of Type 304 stainless steel was riveted to the top of the cross extension to provide a means of simple disconnection from the drive mechanism. With the drive mechanism disengaged from the rod, the bottom end of the fitting acts as a positive stop for the control rod as it allows the rod to rest against the top of the core shroud.

Fabrication of the tee rods was, for the most part, the same as for the cross rods.

Installation of the rods did not require any special handling tools. The rods were inserted into the shroud guides by hand prior to the installation of the pressure vessel head.

Critical experiments were conducted using all nine rods as previously mentioned. However, the five cross rods were adequate for control of the 40-assembly reference design core and the four tee rods were eventually removed.

The critical position of the five-rod bank (in the cold, fresh, 40-assembly core) was approximately 12.6 in. As the core cycle progresses, the reactor hold-down position of the rod bank will exceed 10 in. Less than half of the effective cadmium (Cd^{113}) at that position is burned up in six years of operation at average power. Therefore, it is conservative to conclude that the burnup-life of the control rods is at least two core cycles with respect to the effects of neutron absorption. This is equivalent to at least eleven megawatt years of reactor operation. Control rod design data are tabulated in Table 6.

Table 6

CONTROL ROD DATA

Number of Cross Rods	5
Additional Spaces for Tee Rods	4 (not used on reference core)
Spacing	8-13/16 in. (square lattice)
Length of Cadmium Section	34 in.
Thickness	
Cadmium ("meat")	0.060 in.
Aluminum-Nickel Alloy (clad)	0.030 in.
Total Metered Travel	30 in.
Scram Time (30 in. of rod travel from rest)	<2 sec
Withdrawal Rate[a]	
Rod No. 9	1.8 in./min
Rods No. 1, 3, 5 and 7	2.85 in./min
Weight of Control Rod	
Cross Rod (off-center)	42 lb
Cross Rod (center)	48 lb
Tee Rod	37 lb

[a]Adjustable to meet the criterion of maximum rate of reactivity addition (~0.01%/sec); set initially at 3 in./min. At 1.8 in./min, the maximum rate of reactivity addition by the center rod in the cold fresh reactor is ~0.02 dollar/sec [~0.01 (5)%/sec].

5. Neutron Source*

The startup neutron source must be of sufficient intensity that the neutron instrumentation can detect the power level of the multiplied source neutrons in the subcritical reactor. Due to the uncertainty in the calculation of the neutron spectrum in the instrument region, a generous safety factor was allowed in the design of the source.

In the prototype plant, neutron counters are used to give indication of power level at startup. The sensitivity of the Westinghouse BF_3 proportional counters used is given as 4.5 counts/sec per unit thermal neutron flux. However, it was desired that the source strength be high enough to be able to use the compensated ion chamber to check the approach to criticality. The minimum flux observable by the chamber is limited by the gamma-ray background from fission products.

At a time $2\frac{1}{2}$ hours after shutdown, the fission product gamma rays yield 0.7×10^{-9} amp from the uncompensated ion chamber. A neutron flux of 2×10^3 n/(cm^2)(sec) is required to yield a current of 0.1×10^{-9} amp. With the compensated chamber, an additional two decades of sensitivity are available; thus a flux of 20 n/(cm^2)(sec) was required.

The attenuation of neutrons from the core to the instruments was estimated by using removal theory, and the results of the rough calculations were compared with the results of an experiment(23) performed in the Laboratory's shielding tank facility.(24) It was concluded that the fast flux at the instrument would be 1.2×10^{-3} times the fast flux at the reactor core surface, and that the effective thermal neutron flux level would be about one-tenth as high. The fast flux at the reactor surface, with the reactor 10% subcritical, was calculated to be 4×10^{-4} n/(cm^2)(sec) per neutron born at the core center. Then a factor of one-sixth was applied because the source is at one corner of the core. Thus it was calculated that a source of 10^9 n/sec would yield a count rate of about 20 counts/sec and a current approaching 10^{-9} amp from the instruments.

The design of the source is shown in Fig. 28. The antimony rod, 12 in. long and $\frac{3}{4}$ in. in diameter, is canned in aluminum-nickel alloy (type X8001). An aluminum extension rod is threaded to a stub on the top of the can. At the top of the extension rod is a standard tip to accommodate the fuel-handling gripper, and the bottom has a small grooved tip which presses against a spring clip to prevent rod movement. The beryllium block is $3\frac{5}{8}$ in. square by 14 in. long and has a 1.100-in.-diameter hole to accommodate an aluminum-nickel-alloy liner tube and the antimony rod. The beryllium block is canned and mounted in a dummy fuel assembly.

*A. D. Rossin

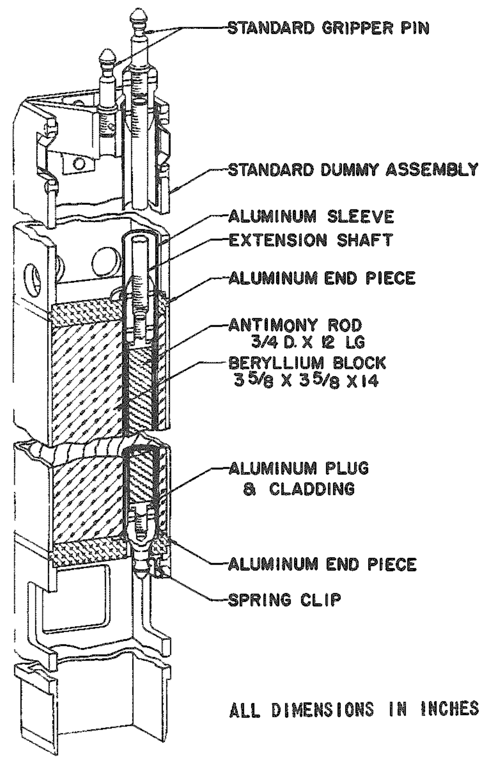


FIG. 28
SOURCE ASSEMBLY

However, the antimony rod gripper pin extends above the assembly gripper pin, making it impossible to attach the gripper to the block assembly with the rod in it. This eliminates the possibility of withdrawing the neutron-radiating source as a whole from the core. There is no residual activity in the beryllium and it can be handled once the 2.3-min aluminum activity decays. However, there will be some long-lived Co^{60} and Co^{58} activities. The activated antimony rod can be handled with the fuel-handling coffin provided that it is separated from the beryllium block.

Gamma rays emitted by the activated antimony rod generate photoneutrons by (γ,n) reaction with the beryllium atoms of the block. The antimony rod was irradiated initially in the MTR; its activity decays with a 60-day half life. During ALPR operation, however, the antimony will be under irradiation and its activity will be regenerated. Calculations indicated that, in the core, the antimony rod would receive an average thermal flux of about 10^{12} n/(cm^2)(sec). The yield was calculated directly using tabulated cross sections, and again by extrapolation from the Wattenberg Source.[25] The two methods indicated that a yield of about 10^8 n/sec could be expected. In order to have an ample source of neutrons, in view of decay losses after irradiation and uncertainties in the analysis, use was made of an irradiation facility in MTR which delivered an unperturbed thermal flux of 6-7 x 10^{13} n/(cm^2)(sec). The first antimony rod (which was damaged and therefore not used) was irradiated for six weeks and the activity (about 1700 curies) on withdrawal was calculated to be twenty times that expected after long-term irradiation in the ALPR. The permanent source rod was estimated to be about 1000 curies on withdrawal from the MTR.

The source proved ample for approaches to critical. To determine a true critical, the antimony rod was withdrawn approximately 4 ft above the beryllium block. When the instruments were placed in their permanent locations, it was evident that the source was strong enough to provide an adequate counting rate. The ion chambers, however, will not detect source power over background.

It is apparent that the source as designed delivered at least its anticipated yield. It does not appear economical to increase the source yield to the level where it can be detected by the ion chambers before startup; hence, it should be possible to reduce the cost of the source. Two recommendations, therefore, are as follows:

(1) The beryllium block represents the biggest item of fabrication cost. It could be designed as a rectangular block lying entirely between the antimony rod and the core, thereby eliminating at least half the quantity of beryllium required. Also, the machining of the antimony rod hole is eliminated.

(2) A reactor with a lower flux and irradiation price rate than the MTR would be sufficient for activation of the antimony rod.

B. Control Rod Drive Mechanism*

The control rod drive is a rack-and-pinion type mechanism. It is designed to meet the conditions outlined in Table 7.

Table 7

CONTROL ROD MECHANISM DATA

Reactor Operating Pressure	300 psig
Reactor Operating Temperature	421°F
Design Pressure	400 psig
Design Temperature	444°F
Number of Control Rods	
Cross-type	5
Tee-type (not used in reference core)	4
Control Rod Spacing	8-13/16 in. (square lattice)
Control Rod and Extension Rod Weight	~100 lb
Control Rod Travel	30 in.
Withdrawal and Insertion Rate (Driven)	
Center Rod (No. 9)	1.8 in./min
Off-center Rod (Nos. 1, 3, 5, and 7)	2.85 in./min
Scram Time	<2 sec
Scram Velocity (maximum)	~4 ft/sec
Dashpot Travel	3 in.
Seal Water Coolant Temperature	135°F

*W. J. Kann

Figure 29 illustrates the arrangement of the control rod drive components that are part of, and directly in contact with, the reactor. The rack and pinion, the pinion support bearings, and the rack back-up roller operate in an atmosphere of saturated steam and water above the reactor vessel.

The rack and its extension shaft are connected to the control rod by means of a ball and socket-type joint. The control rod is provided with a spherical end fitting. The cylindrical extension shaft socket is split in half to produce the gripper jaws. These jaws are pinned to the extension shaft. The jaws are held in place by a sleeve extending over the outside diameter. For disassembly, a built-in threaded shaft and nut-puller lifts the sleeve to permit the jaws to open.

The stellite bushing in the bottom of the shield plug and the rack back-up roller are the guide points for the rack and the extension shaft.

The rack and the extension shaft are fabricated from Armco-type 17-4 PH stainless steel and are fully hardened. The rack teeth are chrome plated for additional wear resistance. The extension shaft is also chrome plated over the length in contact with the stellite bushing.

A 20-degree stub tooth involute pinion (12 diametral pitch and 22 teeth) is mated with the rack. It is fabricated from the same material as the rack, and the teeth are also chrome plated.

The pinion is supported by two ball bearings with stellite balls and races and with stainless steel retainers. An Oldham-type coupling is used between the pinion and the pinion drive shaft.

A seal is used where the pinion drive shaft penetrates the pressure vessel. The seal assembly consists of a stellite guide bushing and a 5-element labyrinth-type pressure breakdown seal. This seal has 5 stationary and 5 floating rings made of stellite. The guide bushing is fluted to allow the passage of water that is introduced between it and the seal elements. Water for the seal is supplied by the feedwater pump. The water provides cooling for the seal and drive components, and the use of feedwater assures that clean water is used to flush the seal and keep dirt particles to a minimum. Leakage through the seal is collected in a lantern ring and returned to the high-pressure condensate return tank. A garter-type shaft seal is used to reduce leakage to the atmosphere. During testing, leakage was approximately 1 gallon per hour.

The transmission and position-indicator assembly (see Fig. 30) is located outside the 3-ft-thick biological shield (see Figs. 31 and 32).

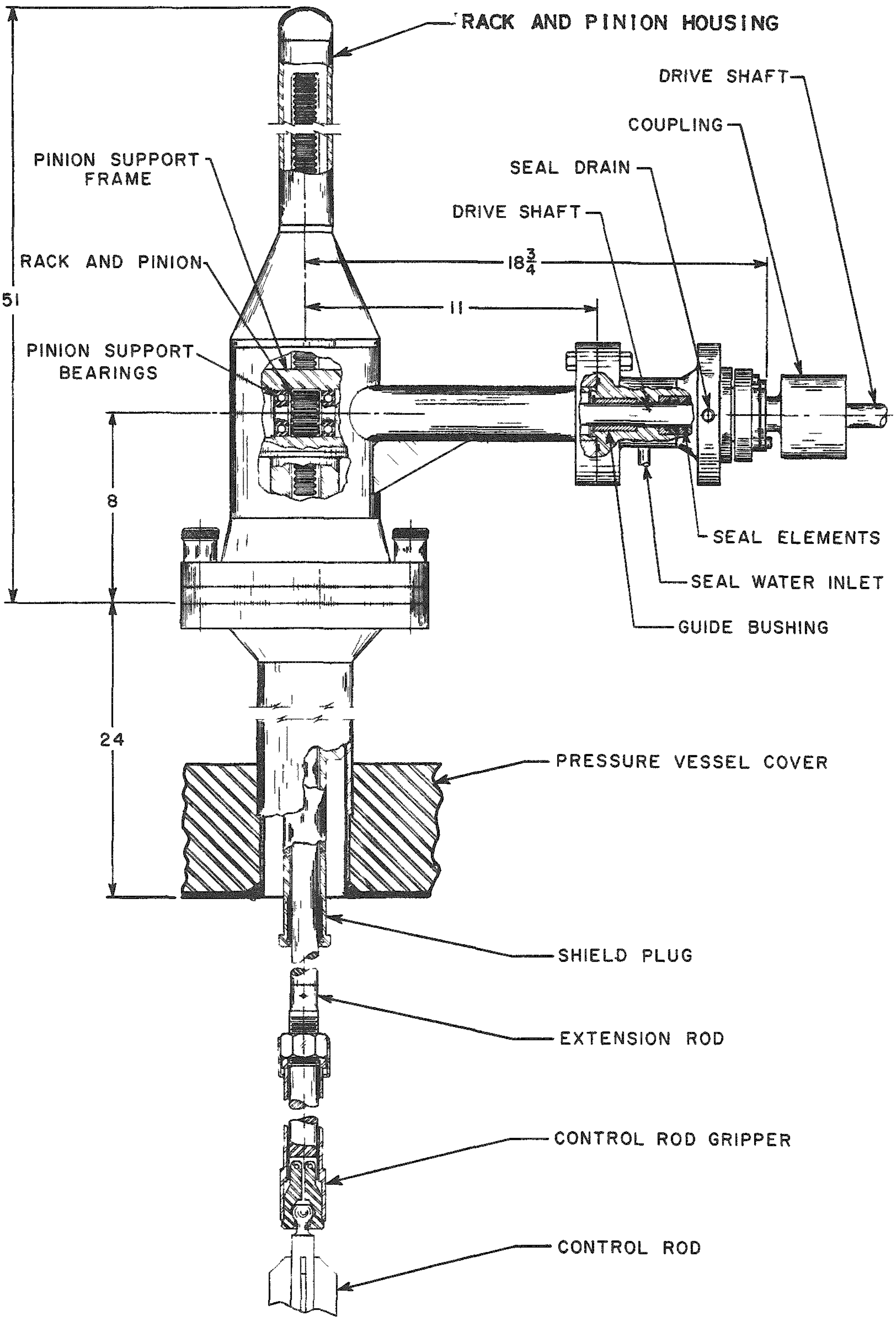


FIG. 29
CONTROL ROD DRIVE

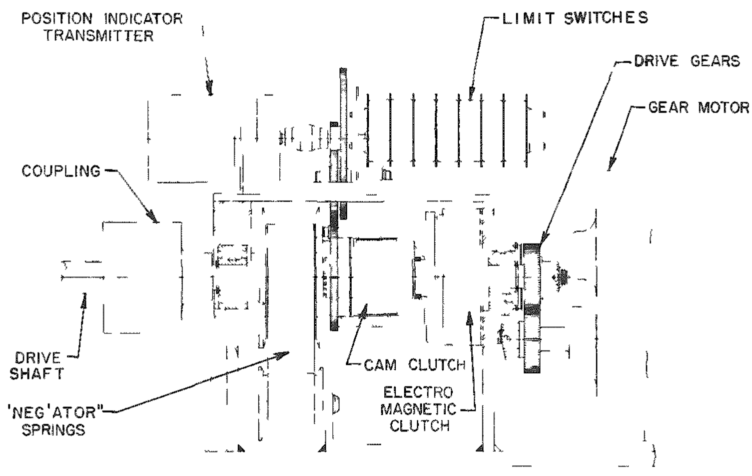


FIG 30
TRANSMISSION AND POSITION INDICATOR

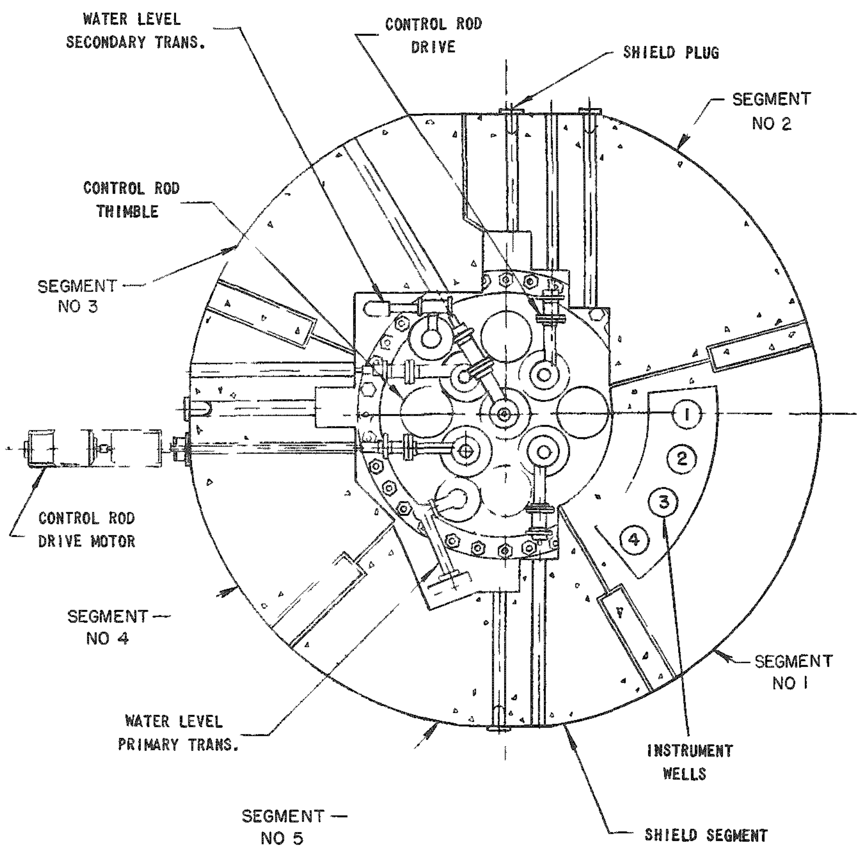


FIG 31
CONTROL ROD DRIVES (5)
AND TOP SHIELDING

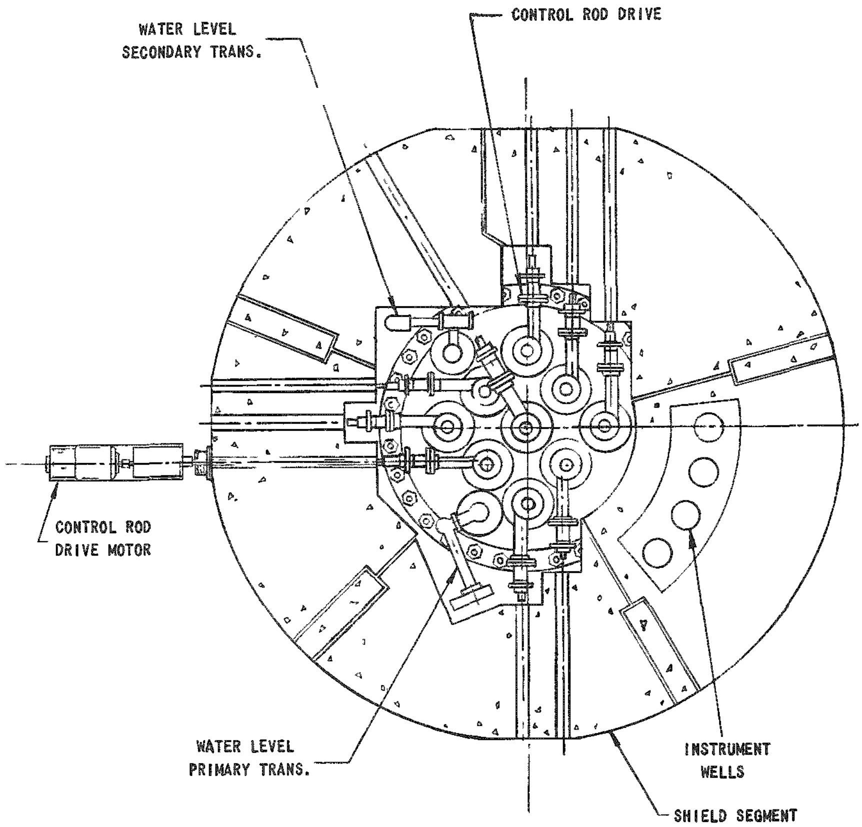


FIG. 32
CONTROL ROD DRIVES (9)
AND TOP SHIELDING

To connect the pinion drive shaft and the transmission, a universal coupling and shaft is used.

The transmission is primarily an assembly of 2 clutches and 2 springs. An electromagnetic clutch is used to drive the rod in either direction. However, should it fail, the cam clutch, which is unidirectional, will allow the rod to be driven down. The "Neg'ator" springs limit the velocity of rod drop during scram to approximately 4 ft/sec. Scramming of the rod is accomplished by de-energizing the electric clutch. A pair of Inconel springs, mounted above the pinion support bearings and concentric about the rack, serve as the dashpot. Dashpot travel is 3 in. Rod insertion time measured during start up was 1.2 to 1.5 sec.

A gear on the "Neg'ator" spring drum drives a gear train that is coupled directly to the position-indicator synchro-transmitter. This arrangement assures the reactor operator of positive rod-position indication at the receiver in the control room at all times during plant operation.

The limit switches, also coupled directly, are used for actuating upper and lower limit motor switches and control panel board indicating lights.

A $\frac{1}{8}$-hp, $\frac{1}{2}$-rpm, 208-volt, 3-phase, 60-cycle gearmotor supplies the motivation required for the drive. It is equipped with a magnetic disc brake that stops the motor armature in about 200 msec. This provides the operator with "no coast" control and permits accurate positioning of the control rod.

Fabrication of the drive components did not require any special tooling; standard machine shop practices were used. The rack-and-pinion housing and the pinion support members are of AISI-Type 304 stainless steel. The transmission, except for "off the shelf" items and all other parts external to the primary reactor system, are fabricated from conventional materials such as carbon steel and brass.

A test rig (see Fig. 33), designed to simulate reactor conditions, was used to test a prototype mechanism. The testing established that AISI-Type 440 C stainless steel was not acceptable for the rack or pinion. Micarta ball bearing retainers were found not acceptable. A pinion shaft guide bushing of nylon also proved unsatisfactory. Various cooling schemes were tried to eliminate the introduction of water into the seal, but none proved practical. For position indication, a multiturn potentiometer transmitter was used initially, but the voltmeter scale of the receiver was not accurate or 100% linear.

The chrome plating on the extension shaft, the rack-and-pinion teeth, and the pinion drive shaft has proved very satisfactory. Very little wear was evident after more than 10,000 cycles of test operation. Where the seal floating rings rode on the pinion drive shaft, definite burnishing was visible. The rack-and-pinion teeth showed a pitch line burnish pattern. The extension shaft showed some scratches which did not penetrate the plating. In no case has any spalling, pitting, or failure of the plating been experienced.

Installation on the reactor (see Fig. 34) did not require any unusual tools or techniques. A special grid structure for mounting the drives was used during the "cold" critical experiments. Final installation on the reactor head resulted in removing a "Neg'ator" spring from the transmission. This was necessary to fulfill scram requirements. It was assumed that certain inaccuracies of alignment between the core shroud and the vessel head nozzles tended to move rods to one side or the other in the guide channel. This resulted in a slight retarding force.

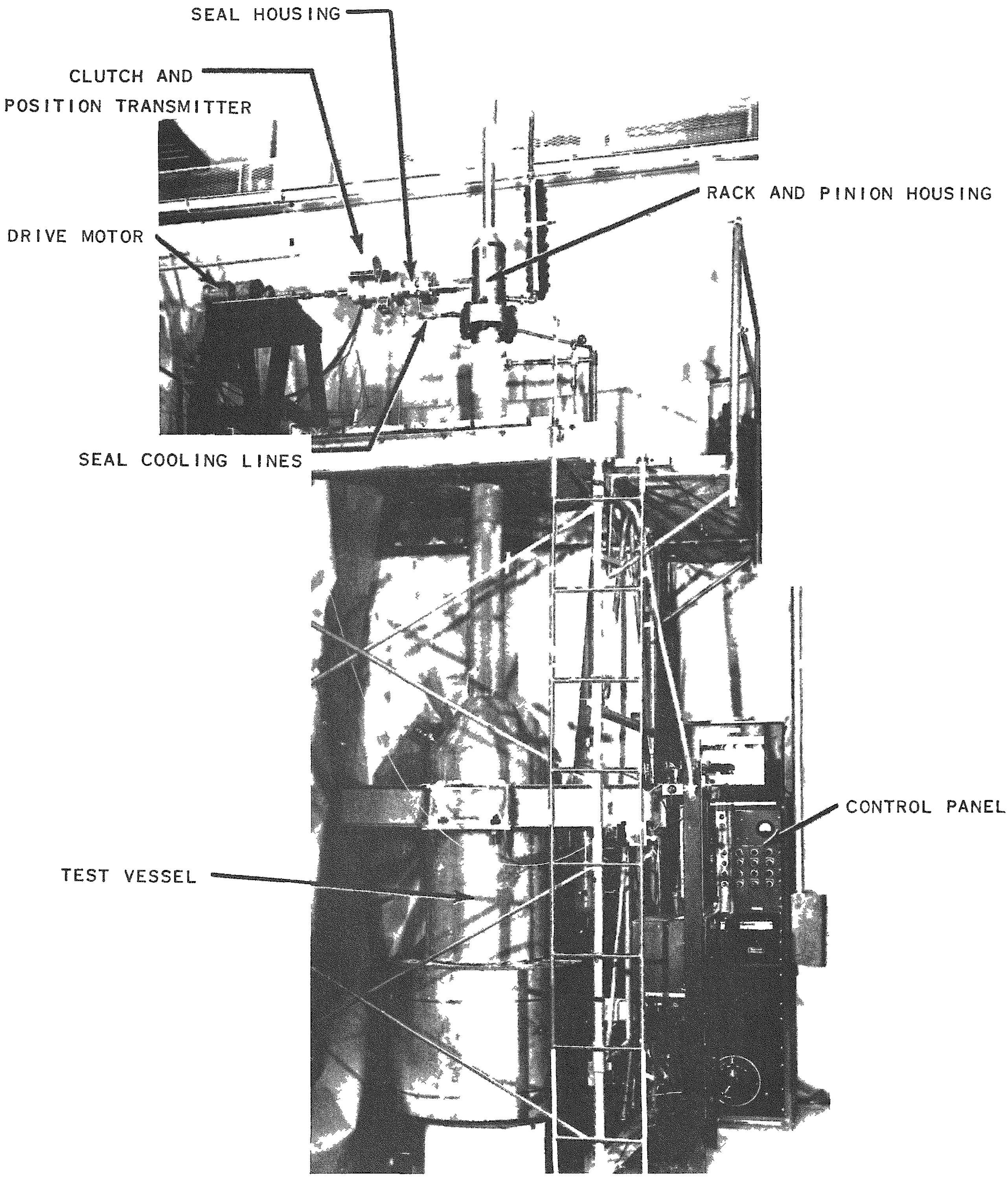


FIG. 33
CONTROL ROD DRIVE TEST FACILITY

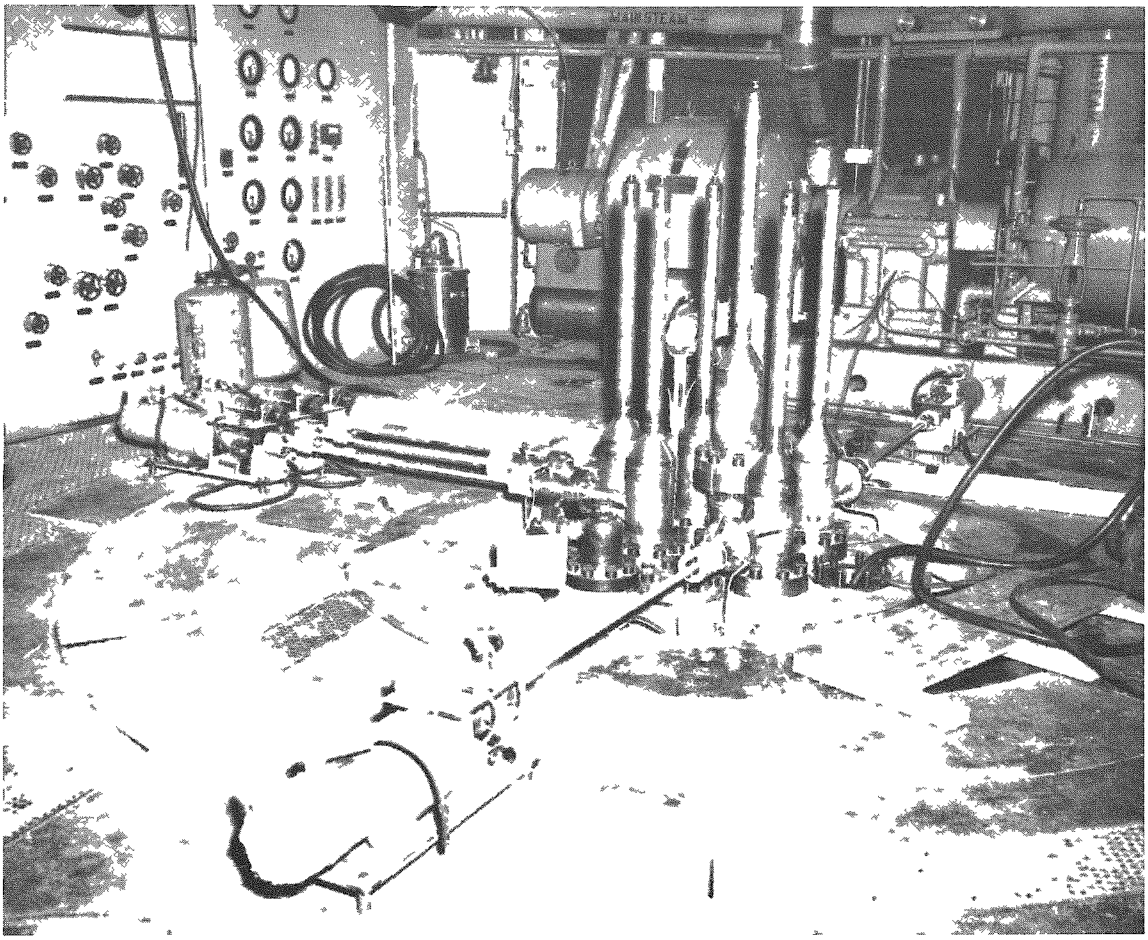


FIG. 34
CONTROL ROD DRIVE INSTALLATION FOR "HOT" CRITICALS

C. Pressure Vessel and Support Cylinder*

The pressure vessel with its appurtenances is supported by the support cylinder to form a reactor complex that incorporates radiation thermal shielding and facilities for the reactor core, control rods and drives, and instrumentation (see Figs. 35, 36, and 37). In this manner, preassembly before shipment is possible. Also, support is provided by a simple steel beam structure on the building bottom plate, and the additional biological shielding gravel is poured in contact with the support cylinder component (see Fig. 13).

*E. E. Hamer

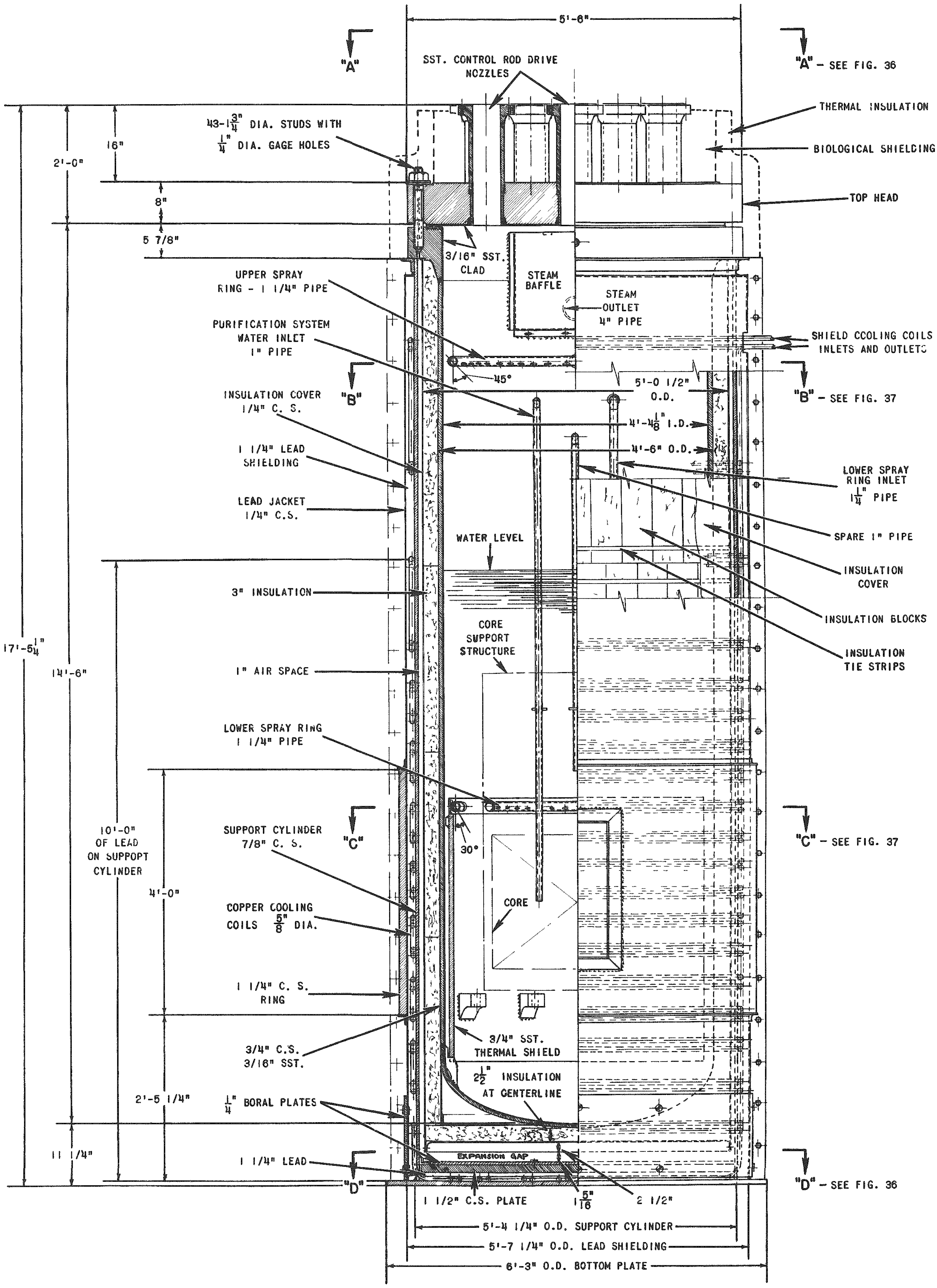
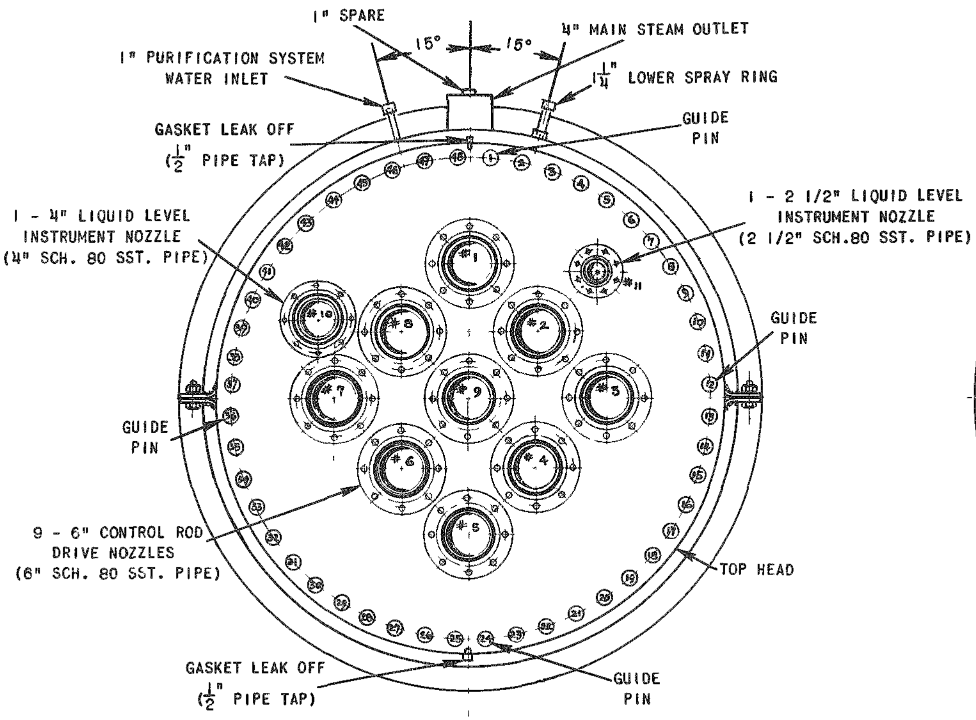
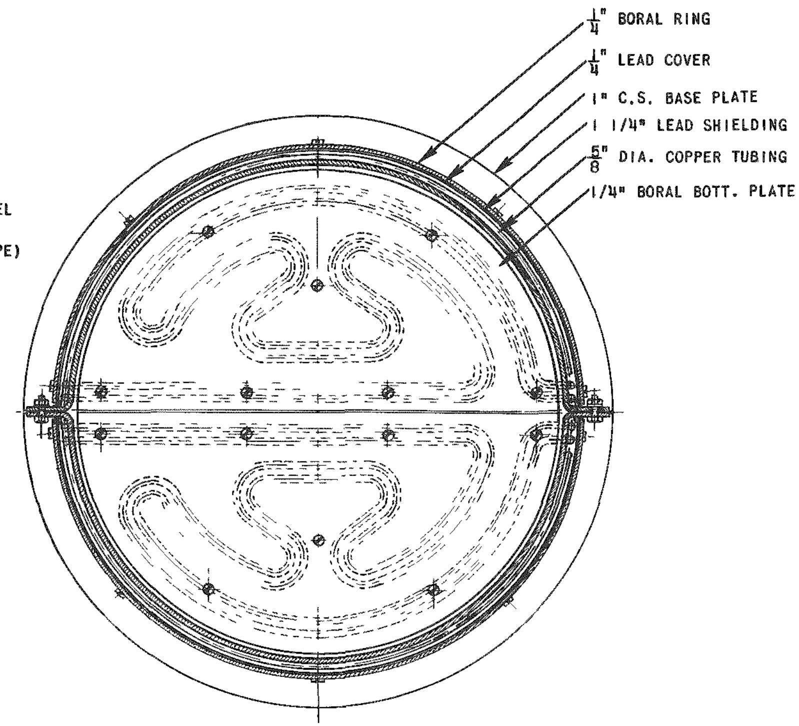


FIG. 35
PRESSURE VESSEL AND
SUPPORT CYLINDER ASSEMBLY

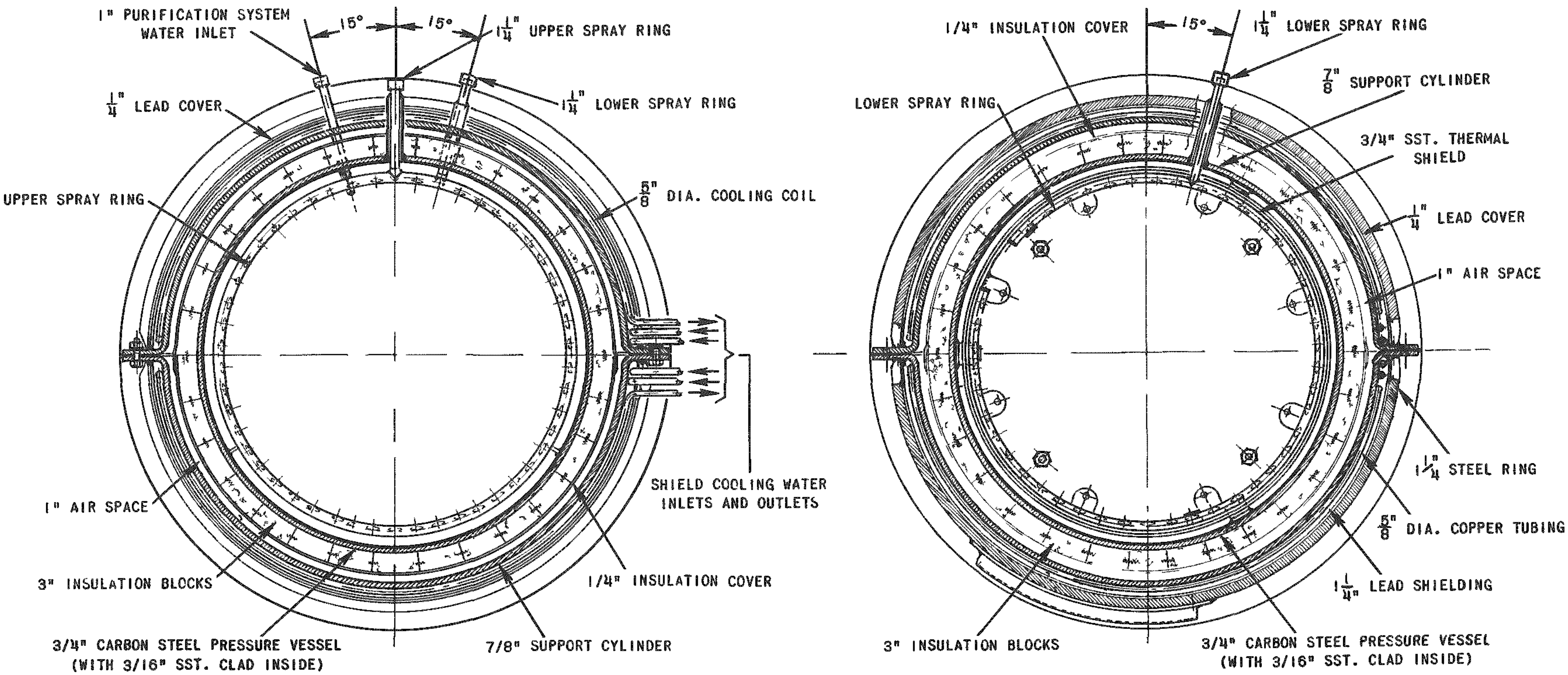


VIEW "A"-"A"



SECT. "D"-"D"

FIG. 36 (SEE FIG. 35)
PRESSURE VESSEL AND SUPPORT CYLINDER ASSEMBLY
TOP VIEW AND BOTTOM SECTION



SECT "B"-"B"

SECT. "C"-"C"

FIG 37 (SEE FIG 35)
PRESSURE VESSEL AND SUPPORT CYLINDER ASSEMBLY
SECTIONS

The control rod drives are top-mounted as the result of precluding excavation for the building, together with limiting the height of the building to 50 ft. However, the pressure vessel head need not be removed for fuel exchange. Fuel exchange is accomplished through the rod drive nozzles whose openings are flush with the operating floor (see Fig. 49 p. 126).

The complex also meets the design requirements on transportability by airlift (see Appendix V). The pressure vessel consists of two major components: the vessel, and the head. The support cylinder is composed of two similar halves. Each component, when crated for shipment, is less than $7\frac{1}{2}$ ft x 9 ft x 20 ft and weighs less than 20,000 lb.

The size of the pressure vessel, and therefore also the size of the support cylinder, was dictated, in part, by the size of the reactor core. The core was designed to accommodate as many as 59 fuel assemblies, providing added flexibility for power production. Also, a large steam dome volume was deemed necessary to reduce water carryover and to eliminate the necessity of the inclusion of expensive steam-drying equipment. The steam dome also provides a steam "buffer" volume, similar to an accumulator tank, thereby decreasing the potentially adverse effects normal to a "solid" system. The length of the vessel and the location of the core are such that filling the vessel with water provides adequate water shielding during reloading of the reactor core (see section VIII-A).

1. Pressure Vessel

A summary of the pressure vessel design requirements[26] is represented in Table 8. The operating position is vertical. The vessel is supported on the underside of the top flange by the support cylinder. The reactor core is supported by the vessel internal thermal shield which, in turn, is supported by lugs welded to the vessel. By this construction, only a few obstructions are in the coolant path to the fuel assembly inlet channels

The reactor control rods are mounted on the 6-in. vessel head nozzles. Nine locations are available corresponding to the core configuration (see Fig. 12). Two additional nozzles, a 4 in. and a $2\frac{1}{2}$ in., are provided for the liquid level-sensing instruments.

The vessel wall penetrations are near the vessel support location to reduce the effects of thermal expansion. The penetrations are five pipes that perform the following function:

(1) A 4-in., Sch. 160 steam outlet.

(2) A $1\frac{1}{4}$-in., Sch. 40 lower spray ring welded to a 2-in., Sch. 80 thermal sleeve, a construction which reduces the stresses resulting from cold feedwater injection. The line is also connected to the boric acid-injection system.

(3) A $1\frac{1}{4}$-in., Sch. 40 upper spray ring welded to a 2-in. Sch. 80 thermal sleeve, a construction which reduces the stresses resulting from cold boric acid injection. The line is also connected to the feedwater system.

(4) A 1-in., Sch. 40 pipe to supply reactor water to the water-purification system.

(5) A 1-in., Sch. 40 pipe that is a spare.

Table 8

PRESSURE VESSEL DATA

Design Pressure	400 psig
Design Temperature	
Water and Steam	450°F
Metal	500°F
Operating Pressure	300 psig
Operating Temperature	
Water and Steam	421°F
Metal	470°F
Outside Diameter	4 ft-6 in.
Length (face of flange to inside of bottom head)	14 ft-6 in.
Wall and Bottom Head Thickness	
Base	3/4 in.
Clad	3/16 in.
Top Head Thickness	
Base	8 in.
Clad	3/16 in.
Materials of Construction	
Base	Carbon steel type SA-212 Grade B Firebox, weldneck flange type SA-181 Grade II
Clad	Stainless steel Type 304
Bond	Roll-bond, plug welded sheet, and submerged arc weld
Internal components	Stainless steel Type 304
Code	ASME Unfired Pressure Vessel, Section VIII
Penetrations	
Vessel wall	(1) - 4 in., Sch. 160 (2) - 2 in., Sch. 80 thermal sleeve for the 1-1/4-in. Sch. 40 pipe (2) - 1 in., Sch. 40
Top head	(9) - 6 in., Sch. 80 (1) - 4 in., Sch. 80 (1) - 2-1/2 in., Sch. 80
Design Loads	Static load of 27,000 lb on any one 6-in. head nozzle when vessel is at atmospheric pressure
Weight of Vessel	~17,000 lb
Weight of Top Head and Bolting	~9,000 lb
Weight of Water (full)	~13,500 lb
Weight of Water (operating)	~8,000 lb
Operating Height of Water (cold)	9 ft
Volume of Steam Dome	~80 ft^3
Pressure-relief Valves	
To condenser	350 psig
To atmospheric	385 psig
Top Head Closure	Double gasket with leakoff
Bolting	(48) - 1-3/4-in. diameter
Material of Studs	ASTM, SA-193, B14
Material of Nuts	ASTM, SA-194 Class 2H
Radiograph	All welds
Hydrostatic Test	600 psig for 2 hr in a simulated vertical position
Helium Leak Test (vessel and head tested separately)	300 psig for 48 hr with maximum allowable leakage of 1.5 cc/day
Cleaning	Aluminum oxide blast Nitric acid passivation

The vessel is thermally insulated by 3 in of 85% magnesia block that has a conductivity of ~0 48 Btu/(hr)(ft^2)(°F) at operating temperature. The blocks are stapled together and banded circumferentially. The insulation is jacketed by $\frac{1}{4}$-in.-thick carbon steel which will restrain and confine any physical breakdown that may occur. The jacket also allows the vessel to be free standing.

a. Steam Baffle

The configuration for the steam baffle was selected as the simplest and most inexpensive, yet effective, design to reduce water carry-over. The normal steam-generation velocity over the area of the pressure vessel is approximately $\frac{1}{4}$ ft/sec in the first stage of steam-water separation. The second stage occurs at the underside of the vessel head before entering the steam baffle. Upon entering the baffle, the change of direction varies from 90 to 180 degrees, and the velocity increased by a factor of 100. The third stage is within the baffled volume, where a decrease in steam velocity by a factor of 2 occurs together with a 90° change in direction. The exit steam quality as determined by the throttling calorimeter method, has been measured above 99.5% with operation at the normal water level.

b. Spray Rings

Two spray rings are available for use by the feedwater system and the boric acid-injection system. Normally, the lower spray ring is used for the feedwater and the upper spray ring is for the introduction of the boric acid if reactor shutdown is necessary by this method.

The lower spray ring has forty eight $\frac{1}{4}$-in.-diameter holes through which feedwater is ejected at an angle of 30° with the vertical. This angle of entry into the downcomer provides for effective mixing and collapsing of the entrained voids carried over by the water recirculation. The elevation of the spray ring corresponds with the top of the fuel assemblies and the bottom of the core shroud cut-out through which a part of the water recirculation takes place (see Figs. 23 and 25).

The upper spray ring has twenty five $\frac{1}{4}$-in.-diameter holes through which boric acid is ejected at an angle of 45° with the vertical. This angle of entry places the reactor poison in the downcomer area for quick transport to the fuel assembly channel. Alternatively feedwater may be valved so as to be introduced through this spray ring, which produces a spray pattern over the entire core because of the higher pumping rate.

c. Thermal Shield

The pressure vessel internal thermal shield is in the region of the core (see Fig. 35) and serves to reduce the pressure vessel wall stresses due to the high thermal neutron fluxes and core gamma radiation. The shield is also effective in introducing a time lag in the heating of the external lead thermal shield should the shield-cooling system fail (see section VII-C). Other functions are to prevent thermal shock to the pressure vessel wall by the introduction of cold feedwater, and to provide a reactor core support.

The thermal shield conforms with the pressure vessel curvature and is fabricated from $\frac{3}{4}$-in.-thick stainless steel plate. A 1-in. gap exists between the shield and vessel to allow for cooling of the vessel wall by circulation of the water.

d. Vessel Gaskets

The vessel gasket design was based upon the continuing work performed on the EBWR pressure vessel. The ALPR and EBWR gasket configuration are alike except in size. Two spiral-wound gaskets, with a leak-off groove between them, are used (see Fig. 38). The gaskets are compressed in the gasket groove by the flat underface of the vessel head. The dimensions are as follows:

Inner - $53\frac{1}{4}$ in. x $54\frac{3}{4}$ in. x 0.175 in.

Outer - $55\frac{3}{4}$ in. x $57\frac{1}{4}$ in. x 0.175 in.

The gaskets are zinc-plated C.R.S. (cold rolled steel) with Teflon or asbestos filler, whichever performed more efficiently. To date, it appears that the asbestos filler-type gasket is more effective.

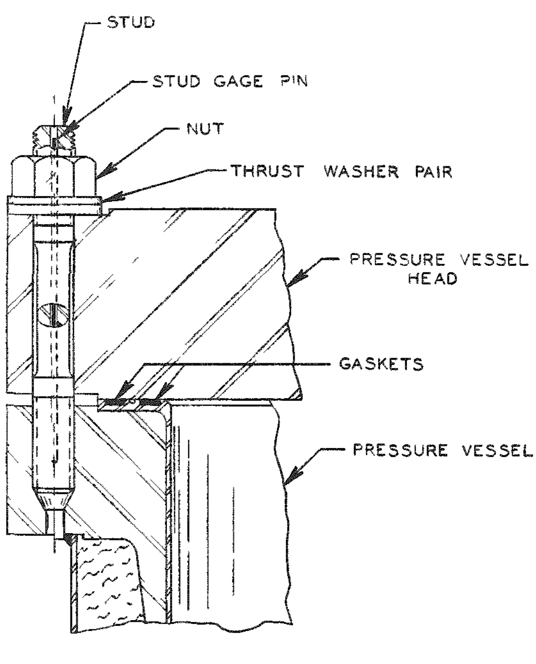


FIG. 38
PRESSURE VESSEL HEAD BOLTING

e. Vessel Head

The use of a flat head instead of a dome head, considering the number of nozzles (9 for control rod drives and 2 for instrumentation), reduced the overall cost materially. The gasket shear based upon reactor pressure and temperature has been compensated in the design of the gasket and groove (see section d above). Zero leakage as measured from the leak-off groove confirmed the selection of the head design.

The head thickness is determined primarily by material selection, pressure load, gasket load, and head perforations (ligament efficiency). Disregarding the stainless steel clad (by submerged-arc deposition) as a strength material, the calculated head thickness is 7.750 in. This is based upon the following for design conditions:

(1) Material - 17,500 psi allowable working stress.

(2) Pressure load - 96.0×10^4 lb

(3) Gasket load - 21.2×10^4 lb

(4) Ligament efficiency of perforations - 0.47.

The impact load (from the fuel assembly coffin) is non-existent during operation and is less than the pressure load when applied at atmospheric conditions. A stress of approximately 600 psi due to gamma heating is additive during reactor operation.

The nozzles for the control rod drives and water-level instruments are fabricated from Type 304 stainless steel pipe and include special flanges. Each of the nozzles is stepped on the outside diameter to absorb some of the impact shock when in a cold condition (see Fig. 35). The strength weld is primarily Type 308 stainless steel with the carbon steel head "buttered" with Type 310 stainless steel in the "J" groove. Assuming an overall joint efficiency of 50% and an allowable shear stress of 10,500 psi, and disregarding the effect of the press fit, the allowable load on the weld of the 6-in. nozzle is approximately 103,000 lb. Using design values, the pressure load is approximately 9,000 lb, and at atmospheric pressure the impact load is approximately 27,000 lb. The 4-in. and $2\frac{1}{2}$-in. nozzles have a lesser pressure load; therefore the impact load also governs The allowable nozzle loads on the welds is approximately 77,000 lb and 45,000 lb, respectively. It should be noted that the use of the $6\frac{1}{2}$-ton fuel assembly coffin is intended with the 6-in. nozzles only.

The stress in the nozzles due to the differences in the coefficients of expansion of the stainless steel nozzle and carbon steel head is difficult to estimate because the head is partially insulated by shielding above. The shielding is also a heat source because of its function. Further, the head is insulated by fiber glass on the ends, and each nozzle contains a shielding plug that is close fitting. The rod drives have an integral plug, and solid plugs are inserted in the vacant nozzles. Assuming a pessimistic temperature differential of 300°F, the stress in the nozzle is approximately 17,000 psi. Of the 17,000 psi, the initial pressure fit contributes 8100 psi, the temperature stress is 8400 psi, and the pressure stress is 500 psi.

Impact on a 6-in. nozzle by the fuel assembly coffin can occur at three critical areas. The downward force along the centerline has been discussed. The other two forces induce a bending stress in addition to the compression and shear stress previously mentioned. One force is applied downward and off center to a point on the outer diameter of the nozzle flange. This force induces a total stress of approximately 21,000 psi. The other force is applied horizontal and at the top edge of the nozzle flange. Assuming a pessimistic value for the force as equal to the weight of the coffin, the stress induced is approximately 27,000 psi. It should be noted that these values are based upon conditions of room temperature and atmospheric reactor pressure.

The impact load on the nozzles is more severe than on the head. The calculated maximum stress in the head due to a 27,000-lb load on any one nozzle at atmospheric pressure is 875 psi.

f. Bolting

The pressure vessel head bolting consists of forty-eight $1\frac{3}{4}$-in.-diameter studs, nuts, and double washers on a 5 ft 2 in. bolt circle (see Fig. 38). The stud material is ASTM SA-193 B14. The nut material is ASTM SA-194 Class 2H. The washers are SAE-1040, heat treated to R_C 35-37, and machined in pairs. The adjacent surfaces are ground and then lubricated with graphite at assembly.

The studs and nuts are a class 2 fit and 8 thds/in. Each stud shank is undercut to a diameter equal to the thread minor diameter. Also, each stud contains a gauge hole, $\frac{1}{4}$ in. in diameter and $13\frac{3}{4}$ in. deep, for the purpose of measuring the stud elongation with a depth gauge during the bolting procedure.[20]

The calculated bolting load is 117.2×10^4 lb. Of this, 96.0×10^4 lb is attributed to the pressure load and 21.2×10^4 lb is the gasket load. These correspond to bolt stresses of 10,200 psi and 2300 psi, respectively, for a total stress of 12,500 psi. The stud elongation representing this stress is approximately 0.006 in. Each stud has an area of 1.96 in.^{2} and thereby assumes a load of 24,500 lb.

The calculated bolt loading is construed as minimum and is below the allowable bolting stress of 20,000 psi for the material. The allowable stress corresponds to a stud elongation of approximately 0.010 in. and is presently used.

g. Vessel

The vessel (see Fig. 35) is fabricated from steel plate and includes an ellipsoidal head, ratio 2:1, at the bottom end and a weldneck flange at the top end. The vessel wall and ellipsoidal head material is carbon steel plate type SA-212 Grade B Firebox and clad with Type 304 stainless steel by the Lukens' roll-bond method. The weldneck flange material is carbon steel forging type SA-181 Grade II and clad with Type 304 stainless steel. The inside diameter surface is clad by stainless steel sheet plug-welded to the carbon steel, and the gasket area is clad by the submerged-arc method.

The vessel wall and bottom head thickness is $\frac{3}{4}$ in., carbon steel clad with $\frac{3}{16}$-in. stainless steel. The calculated design stress developed is 14,000 psi. At the operating pressure of 300 psig the calculated stress is 10,500 psi. In addition to this, a stress of 750 psi is induced at the core centerplane by gamma heating. Also, at the design conditions, no stresses are induced in the vessel wall by the difference in the coefficient of expansion between the carbon steel and stainless steel cladding materials.

The vessel flange configuration and calculations are illustrated in Fig. 39.

The vessel steam outlet nozzle is 4-in., Sch. 160 pipe requiring no reinforcing at the vessel wall due to the pipe wall thickness. This is also true for the two 2-in., Sch. 80 thermal sleeves through which the $1\frac{1}{4}$-in. cold water pipes pass (see Fig. 37). The 1-in., Sch. 40 pipes need no such provision.

h. Hydrostatic Test

The vessel and head were tested independently in the fabrication shop and integral in the field after installation. The shop tests were witnessed by the Hartford Insurance Co. inspector who also issued the code stamp indicating compliance with the ASME Unfired Pressure Vessel Code, section VIII. The field tests were performed as the result of the check on connecting piping.

The vessel head was hydrostatically tested at 600 psig for 2 hours with no loss in pressure.

The vessel was hydrostatically tested in a vertically suspended position. A blind head was provided and the required 600 psig was applied. During the time for the test, the inspector closely examined the vessel including the application of a hammer impact test. The vessel was code stamped thereafter.

i. Helium Leak Test

The helium leak test is not a requirement for ASME code acceptance. The conductance of this test followed the procedure for the EBWR vessel.

WELDING NECK FLANGE DESIGN

DESIGN CONDITIONS			
Operating Pressure, p		400	psi
Operating Temp.		500	°F.
Atmospheric Temp.		70	°F.
Flange Material		A-181-Gr-II	
Bolting Material		A-193-B 14	
Corrosion Allowance		0	
Allowable Bolt Stress	Oper. Temp. S_{op}	20,000	psi
	Atm. Temp. S_{atm}	20,000	psi
Allowable Flange Stress	Oper. Temp. S_{FO}	17,500	psi
	Atm. Temp. S_{FA}	17,500	psi

All Dimensions are Shown in the Corroded Condition

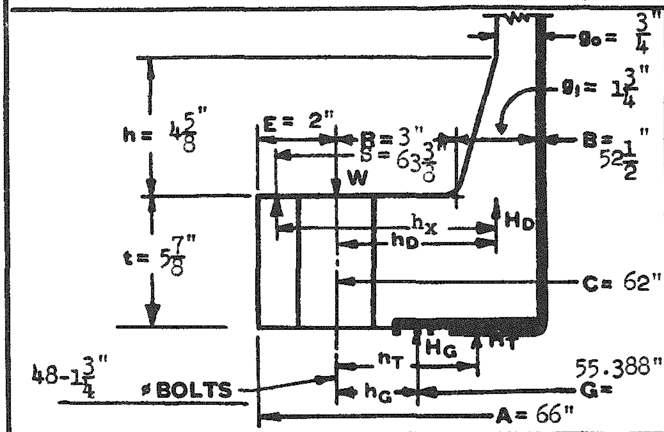
GASKET AND BOLTING CALCULATIONS			
Gasket Details carbon steel spiral wound asbestos filled	From Sheet B	N	(2) 3/4"=$1\frac{1}{2}$"
		b	(2) .306=.612
Facing Details special		y	2900
		m	2.5
$H_{Gy}=b\pi Gy=$			310,000
$H_{Gp}=2b\pi Gmp=$			212,000
$H=\frac{G^2\pi p}{4}=$ 960,000	$H_{Gp}+H=$		1,172,000
$A_{Bmin}=$The Greater of $\frac{H_{Gy}}{S_{atm}}$ or $\frac{H_{Gp}+H}{S_{op}}=$			58.6
$A_{Bact}=$			94
$W_{atm}=.5(A_{Bmin}+A_{Bact})S_{atm}=$			1,526,000
$W_{op}=H_{Gp}+H=$			1,172,000
Gasket Width Check	$N_{min}=\frac{A_{Bact}\times S_{atm}}{2y\pi G}=$		

FLANGE LOADS (OPER. CONDITION)		LEVER ARMS		FLANGE MOMENTS (OPER. CONDITION)	
$H_D=\frac{\pi}{4}B^2p=$	864,000	$h_D=R+.5g_1=$	3.875	$M_D=H_D\times h_D$ =	3,340,000
$H_G=W_{op}-H=H_{Gp}=$	212,000	$h_G=.5(C-G)=$	3.306	$M_G=H_G\times h_G$ =	700,000
$H_T=H-H_D=$	96,000	$h_T=.5(R+g_1+h_G)=$	4.028	$M_T=H_T\times h_T$ =	384,000
H_x = 50,000 Dead Weight		$h_x=.5(S-C)+h_D$	=4.56	M_x =	228,000

FLANGE LOAD (BOLTING UP COND.)		LEVER ARM		FLANGE MOMENT (BOLTING UP COND.)	
$H_G=W_{atm}=$	1,526,000	$h_G=.5(C-G)=$	3.306	$M_{atm}=H_G\times h_G=$	5,040,000

$M_o=$The Greater of $M_{atm}\times\frac{S_{FO}}{S_{FA}}$ or $M_{op}=$ M_{op} = 4,852,000; M_o = 5,268,000 (Equivalent to checking for M_{op} at allowable flange stress S_{FO} and separately for M_{atm} at allowable flange stress of S_{FA}) $M=\frac{M_o}{B}=$ 100,000

If Bolt Spacing Exceeds 2d+t Apply Correction Factor. $C.F.=\sqrt{\frac{\text{Bolt Spacing}}{2d+t}}=$ $M=\frac{M_o\times C.F.}{B}=$



SHAPE CONSTANTS			
K=A/B=	1.258	$h_o=\sqrt{Bg_o}=$	6.28
From Fig. I or Fig. Ia T=	1.8	h/h_o =	.737
Z=	4.55	From Fig. II F=	.782
Y=	8.6	V=	.182
U=	9.5	From Fig. IV f=	1
$g_1/g_o=$	2	$e=F/h_o$ =	.125
$d=\frac{U}{V}h_og_o^2=$	184		

t(assumed)	5.875		
$\alpha=te+1$	1.735		
$\beta=\frac{4}{3}te+1$	1.975		
$\gamma=\alpha/T$	.965		
$\delta=t^3/d$	1.08		
$\lambda=\gamma+\delta$	2.045		

Allowable	STRESS CALCULATION			
1.5 S_{FO}	Longitudinal Hub Stress, $S_H=fM/\lambda g_1^2$	15,900		
S_{FO}	Radial Flange Stress, $S_R=\beta M/\lambda t^2$	2,800		
S_{FO}	Tangential Flange Stress, $S_T=(MY/t^2)-ZS_R$	12,300		
S_{FO}	Greater of .5 (S_H+S_R) or .5 (S_H+S_T)	14,000		

"G" is moved .138" outward from center of double-gasket due to the special gasket configuration and "b" equivalent. $M_{op}=M_D+M_G+M_T+M_X$ $M_O=M_{atm}+M_X$

FIG. 39
PRESSURE VESSEL FLANGE
CONFIGURATION AND CALCULATIONS

The vessel was "bagged" in Pliofilm, all welds were isolated, and the vessel was divided into zones so that a detected leak could be isolated. These zones were as follows:

Zone 1 - The entire vessel "bagged" from the flange downward except for the nozzle area (steam outlet, feedwater inlet, purification system outlet, upper spray ring inlet, and spare).

Zone 2 - The nozzle area.

Zone 3 - The top horizontal weld area, the weld between the flange and upper vessel plate.

Zone 4 - The middle horizontal weld area, the weld between the upper and lower vessel plates.

Zone 5 - The bottom horizontal weld area, the weld between the bottom ellipsoidal head and the lower vessel plate.

Zone 6 - The upper vertical weld area, the weld seam of the upper vessel plate between the vessel flange and lower vessel plate.

Zone 7 - The lower vertical weld area, the weld seam of the lower vessel plate between the vessel bottom ellipsoidal head and upper vessel plate.

The preceding zoning resulted in a checker-board design whereby zone 1 is comprised of many isolated areas over the plate material.

The test was conducted at a helium test pressure of 300 psig with a maximum leakage allowable of 1.5 cc/day. The sensitivity of the mass spectrometer was 10^{-7} cc/sec or 0.518 cc/day. No leakage from the vessel was found.

It should be noted that, although the gaskets were not under test, there was a leakage rate of 16.8 ft^3/hr S.T.P. at the 300-psig test pressure. The leakage was through the inner gasket, and the helium was carried away in order to keep the test environment at a zero background level. Obviously, it was never intended that the gaskets confine helium under the test conditions.

2. Support Cylinder

The pressure vessel support cylinder locates and supports the pressure vessel and its appurtenances (see Figs. 13 and 35).

An external radiation thermal shield is made conveniently a part of the support cylinder. The unit is comprised of two similar halves to adhere to the transport weight requirement. Radially, the support cylinder is comprised of the following:

(1) A $\frac{7}{8}$-in.-thick carbon steel cylinder for the support of the pressure vessel at the top flange.

(2) A $1\frac{1}{4}$-in.-thick layer of lead to a height of 10 ft.

(3) Cooling coils, of $\frac{5}{8}$-in. OD seamless copper tubing, clamped to the carbon steel cylinder and embedded in the lead.

(4) A $\frac{1}{4}$-in.-thick carbon steel jacket to prevent the loss of the lead in the event of melting.

(5) A $1\frac{1}{4}$-in.-thick carbon steel ring in lieu of the $\frac{1}{4}$-in.-thick steel jacket at the reactor core centerplane. The ring is 4 ft high.

(6) A $5\frac{1}{4}$-in.-thick "bay window" of lead (thickness from the steel support cylinder) at the core centerplane, and 2 ft 6 in. high by $47\frac{1}{2}$ degrees wide. This lead "window" has a $\frac{1}{4}$-in.-thick steel jacket and protrudes through the steel ring in the area of three reactor instrument tubes, after installation.

(7) Boral, $\frac{1}{4}$ in. thick by 16 in. high, circumvents the base of the structure.

(8) Two thermocouple wells (each half) for temperature measurement of the lead by thermocouples.

The bottom of the support cylinder is comprised of the following:

(1) Boral, $\frac{1}{4}$ in. thick.

(2) A $1\frac{1}{2}$-in.-thick carbon steel plate.

(3) A $1\frac{1}{4}$-in.-thick layer of lead.

(4) Cooling coils, of $\frac{5}{8}$-in. OD seamless copper tubing, embedded in the lead.

(5) A 1-in.-thick steel base plate.

Each cylinder half was fabricated separately. The halves were then bolted together and the top machined. Construction requirements[26] provided that the support cylinder be concentric with the pressure vessel. The supporting steel plate of each half is fabricated from a single plate in such a manner that no strength welds are present. The $\frac{7}{8}$-in.-thick carbon steel support cylinder is overdesigned for the support of the 50,000-lb load, since the lead is free standing when in the solid state. In the event that the lead melts, the $\frac{1}{4}$-in. steel jacket and support cylinder (with its excess strength) will confine the lead. The steel cylinder has a pressure vessel support area of 173 in.^{2} and a slenderness ratio of 4. Treated as a short column, the pressure vessel component (with 100% contact) induces a compressive stress of approximately 300 psi. In addition to the lead and pressure vessel component, it is assumed that the gravel exerts a pressure of approximately 25, 000 lb. The fiber stress near the top of the support cylinder is 9,000 psi due to the cooling of the support shield by the cooling coilings, located ~16 in. from the top, and the heating by the pressure vessel flange in addition to the flange expansion at the top.

The lead and steel (thermal shield) attenuates the core gamma radiation which otherwise would be converted into heat in the biological gravel shield (see section VII). To dissipate the resultant heat in the thermal shield, cooling coils are imbedded in the lead portion of the shield. Water is used as coolant. Each half cylinder has two independent cooling-coil circuits. Although both circuits are normally in use, one circuit can carry the heat load if the other fails. The lead was placed between the tubing in sheets and contact was made by "wiping" the lead around the tubing at a temperature less than 850°F to prevent the amalgamation of the lead and copper. The coils and all the steel used as shielding were "tinned" to insure good bonding.

The $5\frac{1}{4}$-in.-thick lead "bay window" shadow shields three of the four reactor instrument tubes. This was done deliberately to provide one instrument location not shielded by a large thickness of lead for use with an uncompensated ion chamber. In this way the relative merits of a high and low-gamma background at the uncompensated ion chambers could be evaluated. The relative differences in instrument response time for similar instruments appears to be small, and the value of the added lead shield is questionable when considered in terms of operating experience.[10]

Materials of construction are as follows:

Carbon steel plate and sheet	ASTM SA-212 Grade B Firebox
Bolts	ASTM SA-193 Grade B7
Nuts	ASTM SA-194 Grade 4
Lead	ASTM B-29-55, Chemical Grade (Melting point ~618°F)
Cooling Coils	ASTM B-75-55, Seamless copper tubing Type DHP soft annealed and tinned on the outside surface, size $\frac{5}{8}$ in. OD x 0.049-in. wall
Boral	35% B_4C in cast aluminum clad on both sides by 0.041 in.

The cooling coils were hydrostatically tested at 500 psig and flow tested for 4 gpm per coil at a supply pressure of 30 psig. The lead closure was pneumatically tested at 5 psig with all welds soaped at the fabrication plant and after placement in the field.

VI. REACTOR PLANT AUXILIARIES*

A. Reactor Water Purification System

1. System Description

Water is pumped from the reactor through the purification system and returned into the feedwater line, ahead of the feedwater filter (see Fig. 40).

The intake to the purification system is located in the downcomer region near the center plane of the reactor core (see Fig. 35). Under normal operating conditions, the water in this vicinity is slightly subcooled by 1-2°F. A portion of this water enters the purge line and passes successively through the eductor, purification holdup tank, purification system cooler, purification system pump, temperature control valve, flow meter purification system filter, ion exchanger(s), feedwater line, and feedwater filter, and is then returned to the reactor.

A description of the major items in the system follows:

a. Eductor

The eductor is needed for the operation of the purification system when simultaneously the reactor is at atmospheric pressure, the temperature of the reactor water is above 170°F, and there is no feedwater flow to the purification cooler. The eductor is used to lift the reactor water to the reactor purification system pump, thus raising the pump inlet pressure and preventing flashing with subsequent loss of pump suction. The motive fluid for the operation of the eductor is supplied from the discharge of the purification pump through valve RP-32 (see Fig. 40). The eductor performance is shown in Fig. 41.

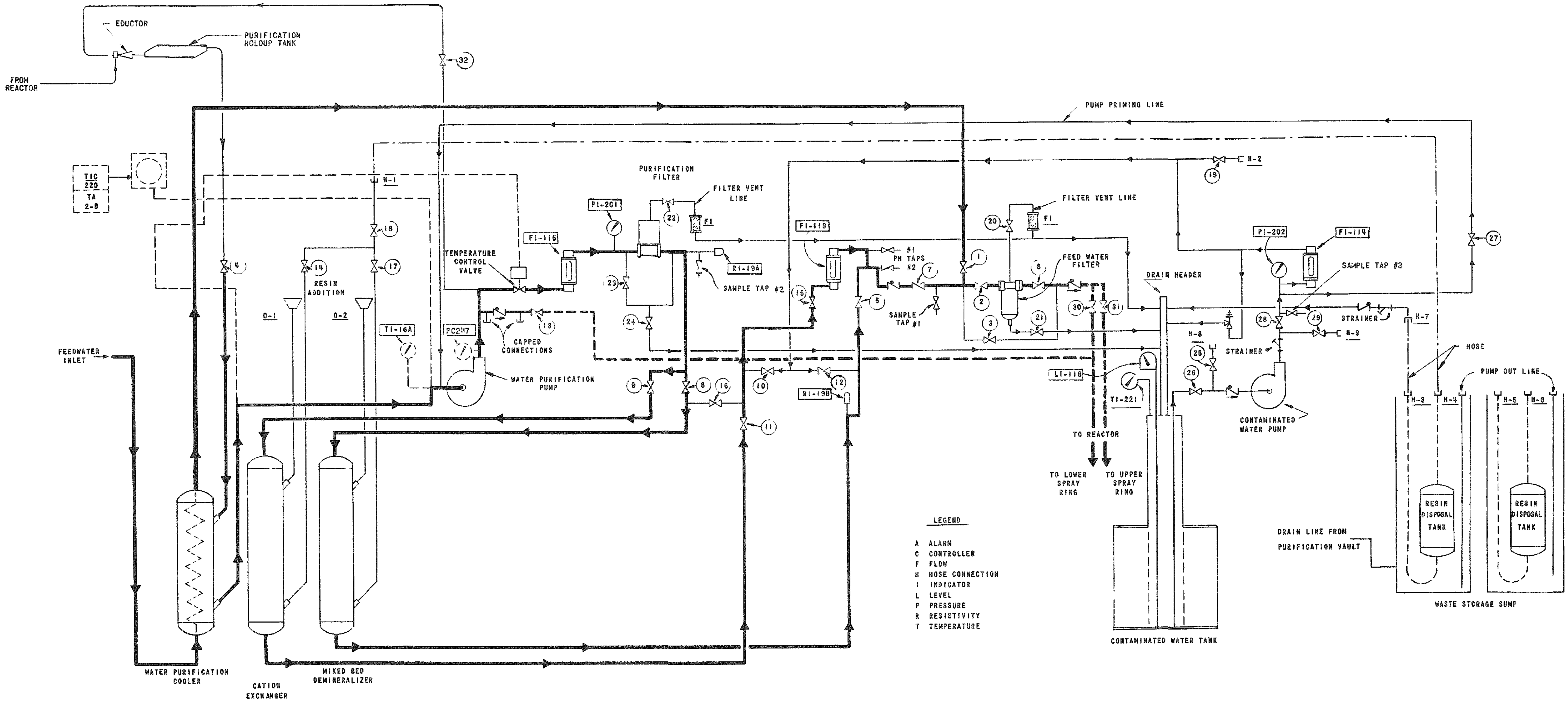
b. Purification Holdup Tank

The water coming from the reactor passes through a 5-gal purge water holdup tank which is located in a shielded region. Here the short-lived N^{16} activity is reduced to tolerable levels during the one to two minutes that it takes the water to pass through the tank.

c. Purification System Cooler

After leaving the purification holdup tank, the water flows through the shell side of the purification system cooler. Here the temperature is reduced by regenerative heat exchange to the feedwater passing through the tube side of the cooler on its way back to the reactor. Under normal conditions, operating the reactor at 3 Mw results in a feedwater flow of 18 gpm through the tube side of the cooler, sufficient to cool 3 gpm of purge water to 170°F.

*A. Smaardyk.



NOTE: ALL VALVE NO'S. ARE IDENTIFIED AS RP, SEE MASTER FLOW DIAGRAM

FIG. 40
REACTOR WATER PURIFICATION SYSTEM FLOW DIAGRAM

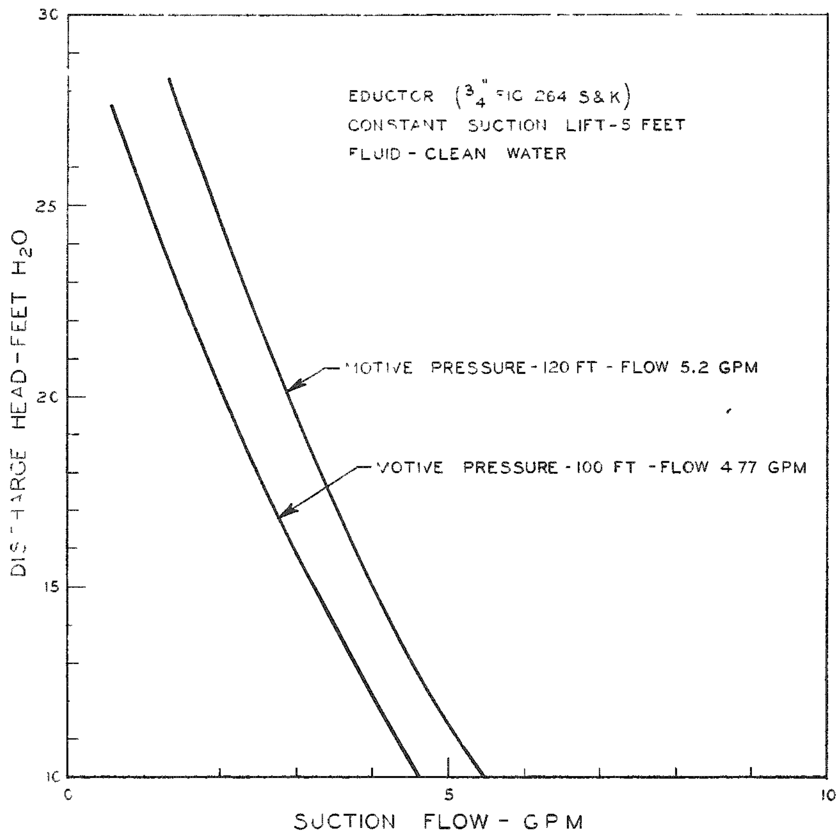


FIG 41
EDUCTOR PERFORMANCE

d. Purification System Pump

The water is circulated through the purification system at the rate of approximately 3 gpm, and up to 5 gpm, by the purification water pump. The pump is completely enclosed, with bearings and rotor submerged in the canned section of the pump, to prevent leakage of water. A small portion of purification water is bled from a point between the discharge of the pump and the temperature-control valve. This flow prevents overheating of the pump if run with the temperature-control valve closed and permits the valve to reopen after high purge water temperature causes complete stoppage of flow. The pump performance curve is shown in Fig. 42.

e. Temperature-control Valve

The temperature of the purge water is controlled automatically by a temperature-sensing Spence control valve located in the discharge line of the purification system pump. The valve is mercury actuated by a sensing bulb in the discharge line of the purification system cooler. It regulates the flow of purge water so that the temperature of the cooled water leaving the shell side of the cooler does not exceed 170°F. The water temperature can be adjusted by changing the valve set point.

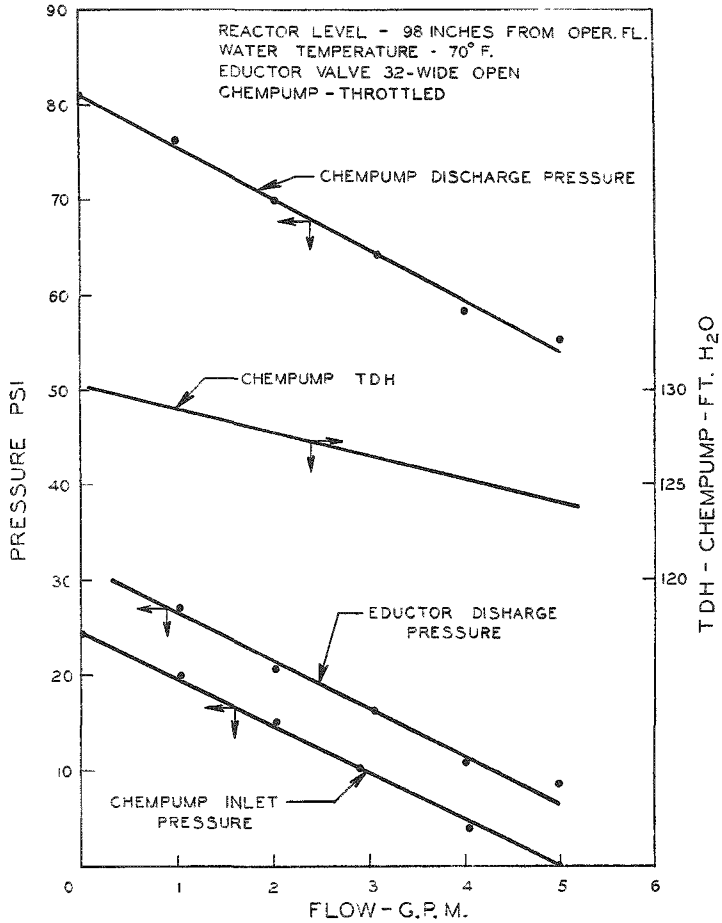


FIG. 42
ACTUAL FLOW PERFORMANCE

f. Flow Meter

Water flow through the purification system is indicated by a totally enclosed float-type flowmeter. An external indicator sleeve is operated by a magnet on the internally located float. The range of the flowmeter is 0 to 5 gpm.

g. Purification Filter

The purification filter is a single disposable cartridge of woven cotton material. It serves to remove particulate matter down to 5 microns in size. Since this particulate matter has been irradiated in the reactor vessel, the accumulation is radioactive. If a cladding failure should occur, the filter would serve to prevent highly activated particles from entering the resin columns. Because it is readily replaceable and rather inexpensive, the filter provides a simple means for extending the life of the costly resins. The filter cartridge is housed within a shielded container having a flanged and bolted removable head for easy replacement of the filter cartridge.

h. Ion Exchangers

Two columns are installed, each consisting of 6-in. diameter, schedule 40 pipe, 9 ft long, and containing ion exchange resin. At a maximum flow rate of 5 gpm through one column, the specific flow through the resin bed is approximately 25 gal/ft^2. The resin charge in each column is approximately $1\frac{1}{2}$ cu ft. Thus the resin depth is roughly $7\frac{1}{2}$ ft. The resin is supported on a 50-mesh screen, and another 50-mesh screen above the bed prevents resin carryover through the system, particularly during the resin changing cycle.

i. Shielding

The purification water cooler and ion exchange columns are located in a common vault below the surface of the gravel shielding in order to protect operating personnel against radiation from radioactive matter accumulated in these units. The vault is a vertical cylindrical tank with a removable shielded cover. This removable cover facilitates removal of components for repair or replacement should such be necessary.

The purification system filter, a 5-micron disposable cartridge type, is expected to accumulate radioactive matter and is shielded with a 2-in. thickness of lead on the sides and top. Because the unit is installed close to the floor, the bottom is left unshielded. All other purification equipment are unshielded above the floor and are readily accessible. A small amount of nickel, 2 w/o, has been added to the reactor fuel core alloy to increase corrosion resistance. For this reason and because the fuel core alloy contains only 17.7 w/o uranium, there is little likelihood that a localized clad failure would result in a rapid escape of large amounts of solid fission products to be picked up by the resin.

2. Operation of the Purification System

Normally, at power levels close to 3 Mw, flow through the system is throttled manually to a rate between 2 and 3 gpm. At reduced feedwater flow, the rate of flow of purge water is throttled further by the Spence temperature-control valve. The purge water must be lifted from the downcomer region of the reactor to the eductor and to the operating floor, and then down into the cooler and up to the purification system pump. The total suction head in the line from the downcomer area to the pump is 16 ft water.

Since 1°F of subcooling effectively increases the net positive suction head by 7 ft at operating pressure and temperature (see Fig. 43), roughly 2-3°F of subcooling should prevent flashing at the pump inlet. Normally, 1.5°F of subcooling is provided in the reactor downcomer by the incoming feedwater. This is sufficient to prevent flashing at the eductor inlet, located 2.75 ft above the normal reactor level, but is not sufficient to prevent flashing farther up in the line. Only a slight opening of the eductor valve should be necessary to further subcool the water a few degrees to assure satisfactory operation.

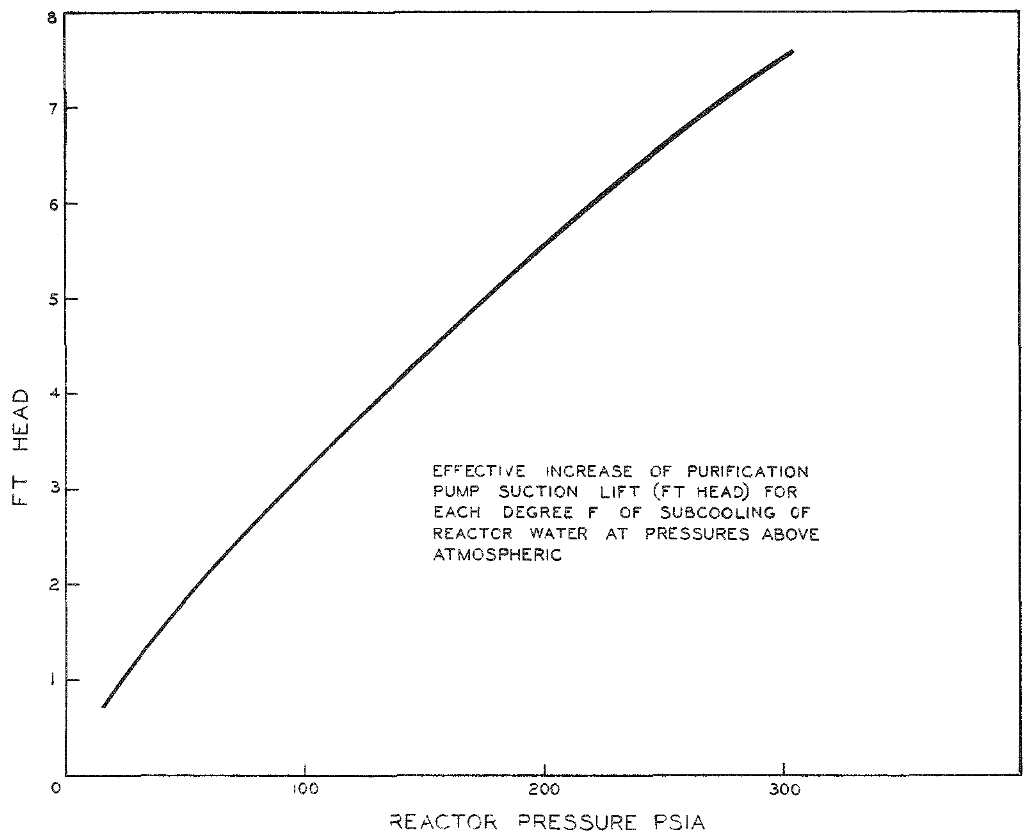


FIG. 43
EFFECT OF SUBCOOLING ON PUMP SUCTION LIFT

As shown in Fig. 43, a decrease in pressure necessitates more subcooling. Approximately 12°F of subcooling is needed at atmospheric pressure. When the reactor pressure is decreased, and before atmospheric pressure is reached, the eductor valve should be opened fully in order to provide a positive suction head on the inlet of the purification system pump. At temperatures between the boiling point at the existing atmospheric pressure and 170°F, the eductor should still be used. It should be emphasized that it is not possible to pump water through the purification system unless the purge water can be cooled by the feedwater as called for by the setting of the temperature regulator.

Water quality is measured both before and after ion exchange by means of conductivity cells and is continually indicated on the instrument panel in the control room. Sample taps are provided before and after the resin columns to check resistivity and pH by means of portable instruments.

By appropriate manipulation of valves, three different modes of flow through the resin beds can be accomplished. For control of pH, parallel flow through the two ion exchange columns was made possible. Alternatively, the purge water may be pumped in series through the two columns, or through one column only. In the latter case, the other column is on standby.

3. Replacement of Ion Exchange Resins

Valves are provided to permit isolation of the purification system from the high-pressure portion of the plant. After isolation, the resin can be removed hydraulically by means of the contaminated (retention) water pump. The resin is removed from the bottom of the demineralizer resin chamber by pumping water from the contaminated (retention) water-storage tank in a reverse direction. Portable hoses are connected temporarily during the process of resin replacement. In this way, the resin is forced out of the vessel through pipes and temporary hoses, and is transferred to a screened-off resin disposal container which is placed in one of two waste-storage sumps (see Fig. 6). The resin disposal container is removed later from the waste-storage sump and after the resin activity has reached a sufficiently low level. The equipment has been designed so that a 55-gal storage drum can be lowered into a storage sump. The 55-gal drum can be used to contain the resin disposal tank and provides an annular space for gravel shielding in the event that experience dictates this procedure. It has been estimated that for a dose rate of 100 mr/hr at the drum wall, the dose rate at the center of the waste can would be 1.1 r/hr. This would correspond to a source strength of 1.23×10^9 Mev/(sec)(ft^3) of resin, or approximately 1.6×10^{-7} times the total accumulated fission product activity of the reactor core.

Fresh resin is added through a top connection on each resin column.

4. Resin Characteristics

At normal purge rates of 2 to 3 gpm, the maximum feedwater flow of 18 gpm cannot cool the purge water below 170°F. The strong-cation type of resins are capable of operating at temperatures up to 250°F. The strong-anion type resins should be operated at temperatures below 140°F. The weak-anion type of resins, however, can withstand temperatures as high as 212°F. For these reasons, a mixed bed containing equal parts of weak-anion and strong-cation resins was initially specified for control of reactor water quality.

Actual operation indicated that this mixed bed did not remove weak acids of organic origin. It was found that satisfactory reactor water quality could be maintained only by the use of a single ion exchange column containing a mixture of strong-anion and strong-cation resins. The limitation of 140°F set for the strong-anion resin is based on the usage of this resin under circumstances of frequent regeneration. Because the ALPR resins are not regenerated, the reduction in life of the resin arising from operation at higher temperatures should not be serious. Regeneration of the ALPR resin is not advised because of the accumulated activity in the resin and because of short water supply at an arctic site. Accelerated laboratory tests[27] in which strong-anion resins of the ALPR type were refluxed at temperatures up to 200°F have indicated basicity loss (test for determining capacity of anion exchange resins to remove anion of weak acids) of approximately 16% at the end of 750 hr.

Wirth[28] similarly shows a curve derived from laboratory studies in which the basicity loss is roughly 13% after 6 months of operation at 200°F for the type of resin similar to the ALPR resin. Although the results of the accelerated laboratory tests are not necessarily directly applicable to ALPR operation, the tests do suggest the possibility of satisfactory operation at higher temperatures between 170 and 200°F for the strong-anion-type mixed bed.

For these reasons, the ion exchange system that was finally recommended consists of a single mixed bed made up of equal parts of strong-anion and strong-cation resins. The specifications for these nuclear grade resins are as follows:

(1) Strongly acidic, sulfonic cation exchange resin with a minimum exchange capacity of 4.7 milliequivalents per dry gram of resin in the hydrogen form.

(2) Strongly basic, quaternary ammonium anion exchange resin with a minimum exchange capacity of 3.5 milliequivalents per dry gram of resin in the hydroxide form.

B. Boric Acid-injection System

A boric acid-injection system (see Fig. 44) has been incorporated in the plant for the purpose of adding boric acid solution to the reactor, either at atmospheric or at operating pressure. It is considered a back-up shutdown system, which can be used at the discretion of the Chief Operator as a means of reducing reactivity, e.g., if control rods should be inoperable.

If the reactor is at atmospheric pressure, the boric acid can be introduced into the reactor by way of the upper spray ring by gravity feed through a portable bypass hose. At higher pressures, boric acid is introduced by means of a manually operated hydraulic plunger-type pump, through either the lower spray ring or upper spray ring.

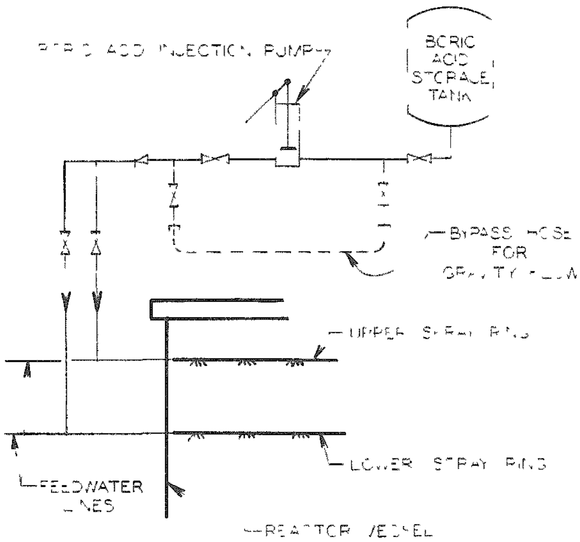


FIG. 44
BORIC ACID INJECTION SYSTEM

The system includes a boric acid storage tank containing 120 gal of boric acid solution. With a concentration of 100 grams H_3BO_3 per gallon of water, the storage tank contains enough poison to control ~20% reactivity (~29 dollars) in the cold fresh reactor, if there are ~900 gal water in the reactor vessel. The boric acid controls ~2.5 dollars/(gm H_3BO_3/gal reactor water) [1.7%/(gm H_3BO_3/gal reactor water)] in the cold system, and an estimated 2.3 dollars/(gm H_3BO_3/gal core fluid) [1.6%/(gm H_3BO_3/gal core fluid)] in the operating system.

Table 9 gives the pertinent tank data.

Table 9

BORIC ACID STORAGE TANK DATA

Diameter	30 in. ID
Height	41 in.
Total Volume	125.8 gal
Gallons per inch	3

The manually operated pump has a capacity depending on the number of hand strokes per unit time. Tests have been run up to 65 strokes per min. At only 25 strokes per min its capacity is approximately 50 gal/hr.

Table 10 gives actual system test data.

Table 10

BORIC ACID-INJECTION SYSTEM TEST DATA

Number of hand strokes applied per minute	65
Capacity per minute at 65 strokes	2.3 gal
Gravity flow through pump	1.5 gal/min
Gravity flow through bypass hose	4.6 gal/min
Time for boron fluid to reach upper spray ring with pump	15 sec
Time for gravity flow through pump	60 sec
Time for gravity flow through hose	12 sec

The boric acid-injection system has been used to add boric acid to the reactor at atmospheric pressure and pressures up to 300 psig during the initial criticality experiments when it was necessary to determine reactivity parameters for control. It was also used to pressurize the reactor system at pressures up to 600 psig to check the tightness of reactor vessel head gasket and control mechanism housings.

VII. SHIELDING*

A. Design

The primary consideration in the design of the prototype shielding was to minimize the weight of material to be transported to a remote site. Gravel was considered to be available at a proposed remote site, so the maximum use was to be made of the locally available material. Thus, the bulk of the reactor biological shield is gravel, which surrounds the reactor pressure vessel cylinder and fills the entire lower third of the plant building (see Fig. 1). The fuel storage wells, ion exchange beds, and 1000-gal contaminated water-(retention)-storage tank are imbedded in the gravel, and steam pipes are in troughs near the surface.

The external cylindrical thermal shield (see Fig. 35) is composed of a $1\frac{1}{4}$-in. layer of lead sandwiched between the $\frac{7}{8}$-in.-thick pressure vessel support cylinder and a $1\frac{1}{4}$-in.-thick cylindrical steel plate at the reactor core level. The bottom thermal shield is composed of a $1\frac{1}{4}$- in. layer of lead between the $1\frac{1}{2}$-in.-thick pressure vessel support cylinder bottom and a 1-in.-thick steel base plate. The shield cooling coils are embedded in this $1\frac{1}{4}$-in. lead layer.

Directly below the reactor (see Fig 13) is a $6\frac{1}{2}$ ft x $6\frac{1}{2}$ ft x 17 in. deep layer of steel punchings in addition to the water-cooled thermal shield, and outside of this is a 4-in. layer of steel punchings extending to a 6 ft 6 in. radius, all supported by 4-in.-thick wooden planks. Sand is mixed with the steel punchings to fill the voids. The $\frac{3}{4}$-in. steel plate building bottom underlies all the bottom shielding. A $\frac{1}{4}$-in. Boral sheet is used below the reactor vessel to capture thermal neutrons and thus to reduce the possibility of thermal neutron activation of air and airborne dust below the reactor building. During reactor operation, the air space below the reactor is inaccessible because of radiations; however, hazardous activation of argon or dust by neutrons is not a problem.

Local gravel is also used as the aggregate for all concrete employed in the reactor shield. The top shield is made from movable blocks which fit together to form a circular wall around the control rod drive area. The large blocks are keyed with small rectangular blocks in an effort to eliminate streaming of scattered radiation (see Fig. 31). The blocks are formed with angle iron at all edges to provide protection against chipping. Pipes provide access for the control rod drive motor shafts. Also, a 2-ft annulus of concrete forms the operating floor.

Directly above the reactor vessel head is a region, 15 in. high, through which the mounting flanges for the control rods pass. The remaining volume of this region is packed with a mixture of shielding material,

*A. D. Rossin

each cubic foot containing 100 lb of steel punchings and 30 lb of boric oxide, the balance being sand. The cavity formed by the concrete blocks and occupied by the rod drives is then covered with a stack of alternate Masonite and steel plates (see Fig. 45). The Masonite provides moderating material to slow down neutrons which leak through the region below, and the steel attenuates the remaining gamma radiation. By bolting a Masonite sheet to a steel plate, a unit is obtained which can be handled by the crane. Also, since the calculations involved are quite conservative, an opportunity was created to decrease the thickness and weight of this top shielding if operating experience so warrants (see section E-1).

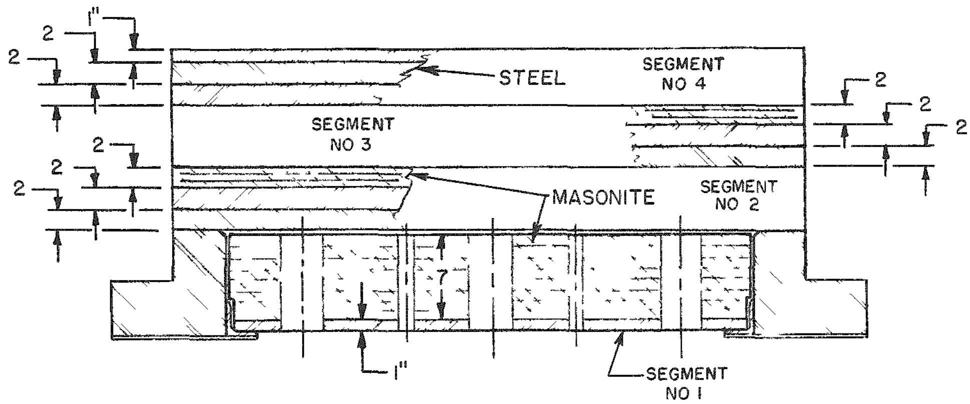


FIG. 45
REMOVABLE TOP SHIELD PLATE ARRANGEMENT

B. Calculations

The results of the shielding calculations are shown in Table 11. The original shielding calculations (see Appendix III) were made using conventional removal theory. The results were considered applicable in the regions of the reflector and thermal shield, and served as a starting point for considering flux levels in the gravel. Removal theory[29,30] determines a distribution of the effective fast neutron flux. Where sufficient moderating material is present, the distribution of removals from the fast group has been shown to determine an effective source distribution for thermal neutrons. The thermal flux is then calculated by diffusion theory, and the source distribution for capture gamma rays can be obtained.

In regions where no moderator is present or adjacent, the theory fails. In a medium like gravel, neutrons may be scattered away from the source many times without appreciable energy loss. In an effort to approximate the resultant transfer distance between the point of removal from the fast group and thermalization, a spatial shift of τ/λ (Fermi age divided by mean free path for fast neutrons) was applied to the thermal flux as calculated by removal and diffusion theory.

Table 11
CALCULATED RADIATION FLUX LEVELS AT 3 MEGAWATTS

Location	Nominal Distance from Centerline	Neutron Flux $[n/(cm^2)(sec)]$ Fast	Neutron Flux $[n/(cm^2)(sec)]$ Thermal	Gamma-ray Flux Mev/(cm^2)(sec)	Gamma-ray Flux r/hr
Radial (at level of midplane of core)					
Core outside surface	41.5 cm	8.28×10^{12}	1×10^{13}		
Inside surface					
Internal thermal shield	62.86 cm	1.1×10^{10}	3.5×10^{11}	5×10^{13}	
Pressure Vessel	66.35 cm	6.5×10^{9}	1.8×10^{11}	2.3×10^{13}	
External thermal shield (support cylinder)	79.4 cm	2.4×10^{9}	9×10^{10}	6.2×10^{12}	
Gravel	85.4 cm	1×10^{9}	3×10^{9}	2.7×10^{11}	1.6×10^{6}
One foot into gravel	115.9 cm	1×10^{8}	2×10^{9}	4×10^{10}	
Twelve feet into gravel	4.5 m (14.8 ft)	10	1×10^{3}		0.001
Outside of instrument shield	93 cm	10^{8}		1.4×10^{9}	4×10^{3}
Axial Up (along core vertical centerline)					
Underside of vessel lid	3.35 m (11 ft)	9×10^{6}	5×10^{7}	2×10^{12}	
Floor level	3.96 m (13 ft)	2×10^{3}	3×10^{5}		750
Above top shield plates	6.1 m (20 ft)	80	1×10^{3}		0.10
Axial Down (in air gap under reactor building)		4×10^{4}	8×10^{5}		100

Such a technique has some theoretical justification and was attempted as a method of predicting thermal fluxes in the gravel. The results are compared in Fig. 46 for the first few feet of gravel. The analysis was used to show that the thickness of gravel specified would be greater than that actually required. The thickness of gravel is determined by the size of the building needed to provide adequate space on the operating floor. Axially above the reactor, where steel was the major constituent, the uncertainty was covered by the use of large concrete blocks and the laminated steel-Masonite top shield plug.

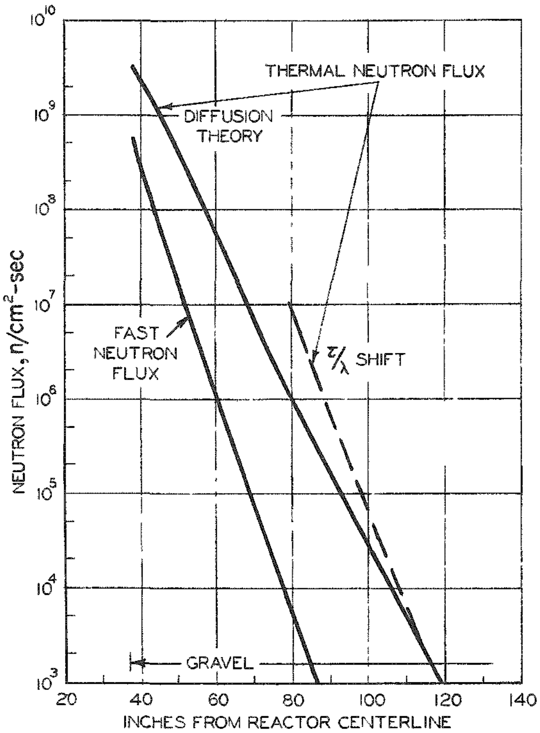


FIG. 46
CALCULATED NEUTRON FLUXES IN GRAVEL

C. Radiation Heating and Thermal Shielding

Calculations were made of the radiation heating in the pressure vessel wall and head and in the support cylinder. The results showed that excessive thermal stresses would not result. The lead and steel of the support cylinder and outer thermal shield attenuate core gamma radiation which would otherwise be converted to heat out in the gravel. Calculated rates of radiation heating in the biological gravel are relatively low. However, the thermal conductivity of the gravel is poor. Accordingly, the problem concerned with the temperature of the gravel was considered.

In calculating the temperature distribution, a thermal conductivity of 0.35 Btu/(hr)(ft^2)(°F) for gravel was used. This was estimated from handbook values and checked by a simple and rough experiment, in which a sample was heated and the resultant equilibrium temperatures were measured at various points. The gravel supported large temperature gradients, and a "k" of about 0.39 Btu/(hr)(ft^2)(°F) was obtained.

As a first approximation, the gravel was assumed to be an infinite cylindrical insulator with no heat dissipation from its outer wall and all heat removal occurring at the inner surface. The resulting asymptotic temperatures were above 800°F. When a conservative rate of heat transfer is assumed at the outer surface, the peak temperature is less and occurs one to two feet from the inner wall. Since these pessimistic calculations indicated that no components would be subjected to dangerous temperatures, further calculations in greater detail were not attempted. Heat is dissipated through the top and bottom of the biological gravel and through the various pieces of equipment imbedded in the shield, i.e., the purification vault containing the resin bed columns and heat exchanger, air-cooled instrument tubes, drain pipe, etc. Thus, the temperatures measured in the gravel during the Plant Performance Test [8] were far below the predicted maxima in Table 12.

Table 12

TEMPERATURES IN THE SHIELD

Thermocouple	Distance into Gravel	Temperature during Plant Performance Test (°F)
No. 1	9 in.	103 (max)
No. 2	1 ft 5 in.	87 (max)
No. 3	3 ft 1 in.	74 (max)
No. 4	6 ft	60 (max)
Storage Wells	~8 ft	90 (max) and 60 (nominal)
Concrete Floor	-	120 (nominal)
Instrument Wells	-	80 (nominal)

The shield-cooling system is designed to handle 40 thermal kilowatts, thereby providing sufficient capacity to carry away the heat equivalent of all of the radiant and thermal energy escaping the reactor pressure vessel. The system consists of a set of seamless copper tubes imbedded in a layer of lead on the outside of the pressure vessel support cylinder. The shield cooling water is circulated through the coils at the rate of approximately 15 gpm by an Aurora APCO single-stage turbine-type pump. The shield coolant dissipates its heat to the reactor condensate (condensate circulating system) in a Ross shell-tube-type heat exchanger. As designed, the system will continue to function adequately by natural circulation even if the pump should fail to operate.[9] However, an alarm is indicated in the control room if the temperature of the shield cooling water or lead thermal shield exceeds set limits. The upper limit is set at 185°F. During the Plant Performance Test, the effluent from the shield cooling coils was usually between 155 and 180°F.

The temperature of the lead never exceeded 195°F. The system was tested on natural circulation with the shield cooling pump bypassed. Coolant temperatures stabilized in the same temperature range observed during the Plant Performance Test.

Calculations were made of a hypothetical accident situation in which shield cooling is lost and no action taken. The results indicated that the lead around the cooling tubes could melt in a few minutes. Since this leaded region is completely inaccessible, a thermal shield was designed to fit inside the pressure vessel. This $\frac{3}{4}$-in. stainless steel cylinder attenuates much of the lower-energy core gamma radiation, and despite the fact that some high-energy secondary gammas are born in it, the net effect is an increase in time before possible melting of the lead to one-half hour after the accident. In addition, the thermal shield protects the vessel from high thermal neutron fluxes and also is used to support the core structure.

D. General Shield Performance

The design requirements (see Appendix V) limited the dose to be accumulated by an individual to 50 mr for a seven-day work week. Personnel do not normally spend long periods of time in the plant during operation; hence there is no need to design to 1-mr/hr levels. The shield was designed to keep the operating floor and regions outside the reactor building below 7.5 mr/hr (the standard set by the AEC to permit 300 mr/wk for a 40-hr continuous exposure work week). It was believed that a dose rate as high as 100 mr/hr could exist directly above the reactor top shield; this was no problem, since this region is accessible only with difficulty and there is no anticipated reason for personnel to be there during operation. An attenuation factor of ~10 was allowed for radiation scattered back to the operating floor from the ceiling and various pieces of equipment.

Although calculations were made at an early date of the minimum radial thickness of biological gravel required, space considerations on the operating floor soon fixed the diameter of the building and hence the radial shield thickness. Since the design guaranteed more thickness than the minimum required, further calculations were not pressed. However, due to the uncertainties in the preliminary calculation, several feet of gravel would have been specified as a safety factor. In actual construction, gravel of about 15% less density than that originally assumed was used. Thus the shield thickness, as constructed, was believed to be adequate, but not grossly overdesigned.

No shielding was believed to be required for any of the components or pipes which were not to be imbedded in the gravel. BORAX experience indicated that radiation levels at contact on the main steam pipe would be greater than $7\frac{1}{2}$ mr/hr but well within tolerance at a foot or more from the pipe. The turbine was not expected to be a radiation hazard. In general, it was believed that should an unexpected "hot spot" be found, steps could be taken to reduce the level, based on operating experience.

Table 13 gives activity levels at 21 different locations throughout the plant.

Table 13

ROUTINE HEALTH PHYSICS SURVEY

Location	mr/hr of Gamma Radiation	
	at Full Power (3 Mwt)	at 1 hr after Shutdown
Control room, rear of panel	0.11	0.005
Lower landing, by plate on stairway	0.09	0.004
Between lower and middle landing	0.05	0.004
Middle landing	0.03	0.004
Doorway, operating floor level	0.8	0.05
Doorway, chest height	0.8	0.05
Average around reactor top shield	2 to 8	0.10 to 1.0
Feedwater pump No. 1	1.0	0.13
Feedwater pump No. 2	1.0	0.13
Top, simulated heat load heat exchanger	300 (max)	1.2
NE end, simulated heat load heat exchanger	1.0	0.13
Bypass steam line	8	1.5
Main steam line	7	1.2
Middle, turbine control panel	5	0.37
Turbine casing	4	1.2
Main steam line, overhead by emergency door exit light	5	0.1
Purification system panel	8	1.8
Feedwater filter	13	1.7
Main steam line, behind purification system panel	13	1.1
Air ejector flow meter	4	0.25
Reactor, on top and center of shielding	1	0.05

These levels should be regarded as typical for similar conditions of power level and bypass steam flow rate after the reactor has been operating steadily. The particular data presented are based on a survey taken shortly before the end of the Plant Performance Test with the reactor at 8000-lb/hr main steam flow, and 1500-lb/hr bypass steam flow rates, and again one hour after reactor shutdown at the end of the Test. All readings are tabulated in milliroentgens per hour of gamma radiation. Those values above 1 mr/hr were taken with a JUNO survey meter, and those below 1 mr/hr were taken with a scintillation detector having a minimum sensitivity of 0.001 mr/hr. The general area background prior to reactor operation was 0.003 to 0.004 mr/hr. Activity levels on the fan floor averaged several mr/hr during operation.

The "hot spot" at the top of the simulated heat load heat exchanger is discussed in section E-4. Another location of high activity level to be considered is the floor region above the purification system vault, which indicated levels of 40 to 60 mr/hr after several days of operation. This radiation leakage contributes to the high activity levels at the purification panel. The purification system vault is not referred to in Table 13 but is discussed in section E-3.

E. Specific Design and Operating Considerations

1. Top Shield

The top shield consists of several layers of steel and Masonite. The former provides shielding for primary and secondary gamma rays above the vessel head and control rod drives, and the latter helps to make up for the lack of moderating material in the axial direction above the reactor. The lowest plate is Masonite with holes to accommodate the tops of the rod drive housings. The details of this and the remaining layers are shown in Fig. 45. The stack can be taken apart in four segments. Radiation measurements were taken with the reactor at a nominal full power of 8000-lb/hr steam flow rate. The results are given in Table 14.

Upon examination of the data, it becomes clear that it is safe to reduce the amount of top shielding. An array made up of the large Masonite segment and one of the thinner sets of steel-Masonite segments should be adequate. The anticipated dose rates at 3-Mwt reactor operation would be roughly 16 mr/hr gamma and 3-8 mrem/hr neutron above the top shield, and 3 mr/hr on the operating floor. The result is a saving of 9 in. of steel plates and 2 in. of Masonite plates, the total weight of which is 8000 lb. The cost of fabrication of these plates is also saved, and the lowering of the height of the stack permits handling of the remaining plates by a standard hook and chain attachment to the crane, rather than the steel beam arrangement which was actually employed.

Table 14

RADIATION LEVELS WITH VARIOUS TOP SHIELD ARRANGEMENTS

Configuration (Bottom to Top)	Dosage above Top Shield			On Operating Floor Gamma (mr/hr)	Segment[a] No.
	Gamma (JUNO) (mr/hr)	Neutrons/(cm²)(sec)			
		Fast	Thermal		
No top shields	200	5500	1700	10	-
7 in. Masonite + 1 in. steel	100	150	170	5	1
As above directly above 3-in. hole in 7-in. Masonite block	40	130	60	-	1
7-in. Masonite + 1-in. steel 4-in. Steel 2-in. Masonite	16	40	15	3	1 2 2
7-in. Masonite + 1-in. steel 4-in. Steel 2-in. Masonite 4-in. Steel 2-in. Masonite	3	15	8	3	1 2 2 3 3
Full Shield	1-2	10	2	3	1, 2, 3, 4[b]

[a]Segment numbers refer to Fig.45.

[b]Segment No. 4 is comprised of 5 in. of steel.

Although it is not feasible to do so on the prototype plant, it is recommended that in a subsequent design the outer radius of the concrete blocks could be reduced by 8 in. Valuable floor space could be saved. On the basis of shielding considerations only, it would be feasible to reduce the height of the outer part of the blocks by about a foot. However, this increases the complexity of the forms required for fabricating the blocks and does not save any floor space. Blocks with squared-off outer sides could also be used, saving some forming cost, but requiring some additional floor space.

2. Instrument Filter Shield

It is necessary that, at all times during reactor operation, the nuclear instrumentation give a positive indication of reactor power level. During low-power critical experiments, temporary instrument tubes were placed at conveneient locations in the vessel. The permanent instrument tubes are located in the gravel just outside the pressure vessel support cylinder and external thermal shield (see Fig. 13). Since there is no good moderating material present in this region, there was a question concerning an adequate flux of neutrons to provide sufficient current in an uncompensated ion chamber to activate a reactor scram signal at 150% of full power without amplification, and to measure multiplied neutrons from the source before starting to withdraw control rods. The latter condition is most difficult to attain during the period shortly after reactor shutdown, when gamma activity from fission products in the core creates a high background current in the ion chamber. The range is effectively extended downward another decade or two by the use of a compensated ion chamber; however, some response at the high end must be forfeited.

It was decided to add a lead filter to one portion of the outer thermal shield between the support shell and the instrument tubes (see Fig. 35). The $1\frac{1}{4}$-in. steel belt is replaced by 4 in. of lead, making a lead section $5\frac{1}{4}$-in. thick. The net effect of this change is to reduce the gamma-ray level at the instrument by a factor of 7 to 8 during operation, and by a factor of almost 40 at a time $2\frac{1}{2}$ hr after shutdown following long reactor operation. Replacement of steel by lead may raise the neutron flux level somewhat, but no quantitative calculation was made. The lead filter covers instrument tubes Nos. 1, 2 and 3 reading clockwise around the reactor top (see Fig. 31), leaving the fourth tube as covered by the original arrangement of materials. When comparing the instrument responses between the filtered and unfiltered tubes in actual operation, very little difference was observed with similar instruments. Thus, the value of the lead is questionable, since observed instrument responses easily exceed the minimum required for safe and dependable operation. It is not recommended that such a filter shield be specified in a future design.

3. Reactor Water Purification System Activities

Although consideration was given to anticipated activities in the purification system resin columns, no quantitative predictions were made. Many variables are involved, such as type and magnitude of corrosion, impurities, pH of the water, and entrainment. It was known from EBWR and BORAX experience that radioactive crud does build up in the resin and that noticeable activity levels result. Sodium activity (half-life of 15 hr) is inevitable due to the $Al^{27}(n,\alpha)Na^{24}$ fast neutron reaction. The design called for burying the resin beds below floor level and making use of gravel shielding where available.

After several days of continuous operation, the activity above the resin beds was observed to be 40 to 90 mr/hr. Most of this is due to Na^{24} and will decay to safe levels within a few days after shutdown. Residual activities consist of traces of iron, cobalt, and other materials. To reduce the activity levels on the operating floor, additional gravel was placed in the region around the vault which contains the resin beds. It will be necessary to monitor the resultant levels after a long period of operation to evaluate the effectiveness of this additional shielding. If excessive levels are observed, additional shielding must be added.

If the radioactive resin is pumped out of a bed and into storage, the line carrying this material will be an intense source of radiation. Such operations should be conducted with extreme caution and under direct monitoring by the health physics representative.

4. Simulated Heat Load Heat Exchanger

The first radiation survey of the operating reactor and plant disclosed the presence of a "hot spot" on the upper surface of the simulated heat load heat exchanger near its southeast end (nearest the building wall). On contact, a Juno survey meter indicated levels as high as 300 mr/hr at the point marked "X" in Fig. 47. At other points on the heat exchanger shell the levels measured from 5 to 50 mr/hr. At a distance of two feet, where a man stands to read the turbine control panel, the levels were 15 to 20 mr/hr.

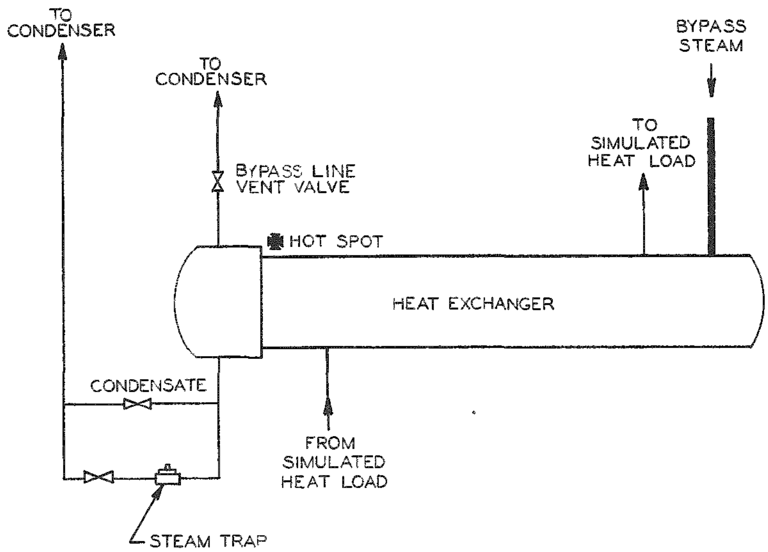


FIG. 47
SIMULATED HEAT LOAD HEAT EXCHANGER

When steam flow through the bypass steam system is stopped, the level drops rapidly, indicating that the activity is carried in the steam, and that whatever activity is accumulated near the "hot spot" has a short half-life. Although no quantitative measurement of the half-life was obtained, the logical isotope is clearly N^{16}, formed from the fast neutron reaction $O^{16}(n, p)N^{16}$, which decays with a 7.4-sec. half-life and is always observed in water systems of reactors. The nitrogen is carried with the steam, but as the steam is condensed in the shell, the nitrogen collects at the top near the exit end resulting in a localized "hot spot." Some rough data was taken to determine the behavior of this activity level as a function of main steam and bypass steam flow rates. The experimental conditions were not well stabilized, and measurements were taken hurriedly as flow rates were being changed. Although the data points showed appreciable scatter, a set of curves were fitted by eye and are shown in Fig. 48. The solid-line curves show a primary dependency of the level on the rate of bypass steam flow (which tends to level off above the 1500-lb/hr capacity of the simulated heat load heat exchanger). A secondary dependency is observed on the main steam flow rate. This can be attributed to the increase in transit time from reactor core to heat exchanger for a unit mass of steam and N^{16} as the rate of main steam flow is decreased. The holdup time in the steam dome increases and velocities in the steam pipe decrease. Thus more N^{16} decay takes place before the heat exchanger is reached, and the "hot spot" is reduced accordingly. Quantitative interpretation of these curves was not deemed advisable, so a repeat of the experiment was attempted.

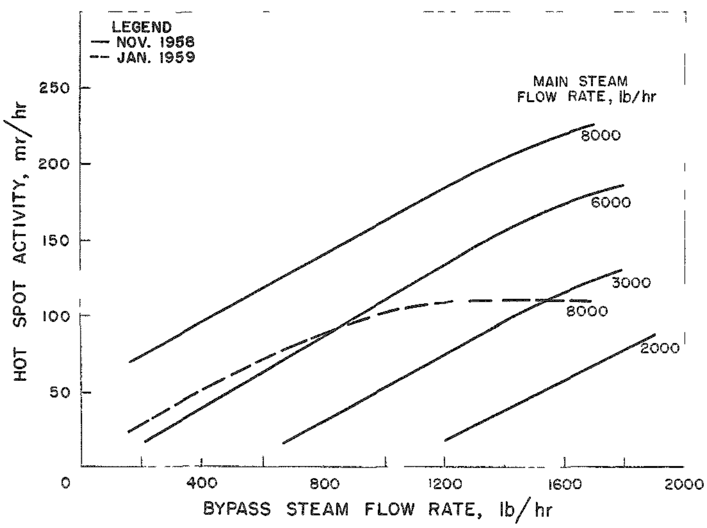


FIG. 48
HOT SPOT ACTIVITY AT HEAT EXCHANGER

Although the "hot spot" did not result in higher activity levels at any point in the plant which was accessible to personnel (it apparently contributed little to the dose at the turbine control panel which varied little with changes in bypass steam flow rate), an effort was made to eliminate it without the use of additional shielding. The vertical vent pipe at the end of the heat exchanger exhausts directly to the condenser. If it were opened, gaseous constituents would be swept from the heat exchanger into the condenser. This effect was clearly observed in practice when the valve on this pipe was opened. The record of the activity level is represented in Table 15.

Since this test (January 29, 1959), the "hot spot" activity has never risen above 130 mr/hr. No explanation for this behavior has been proposed. Nevertheless, an attempt was made to repeat the activity level versus steam flow rate experiments. The results are shown as the dashed curve in Fig. 48. The data were so scattered on runs at lower rates of steam that no curves could be drawn.

Table 15

SIMULATED HEAT LOAD HEAT EXCHANGER "HOT SPOT" RADIATION LEVELS VERSUS VENT VALVE POSITION (JANUARY 29, 1959)

Valve Position	"Hot Spot" (mr/hr)
Closed	240
$\frac{1}{4}$ turn open	65
Closed	140
$\frac{1}{8}$ turn open	105
$\frac{1}{4}$ turn open	50
Open	25
Closed	115
Closed (after one day operation)	110

Another obvious variable was evident in this experiment. The activity level would hold for several seconds and drop abruptly by 70 or 80% regardless of the vent valve position, and then slowly rise and suddenly drop again without showing a reproducible pattern. These fluctuations are attributed to the operation of the steam trap which intermittently passes condensate out of the heat exchanger system. Apparently this action affects the accumulation of noncondensibles in the heat exchanger.

To keep the "hot spot" activity level down, the valve on the vent line was set open $\frac{1}{8}$ turn. After an hour, the activity level was observed to be between 30 and 50 mr/hr at 8000-lb/hr main steam flow and 1500-lb/hr bypass steam flow rate. A nominal amount of steam is lost through this line, but not enough to affect the operation of the simulated space heat load system.

VIII. FUEL-HANDLING SYSTEM

A. Fuel Handling*

The procedure and equipment described herewith are for the purpose of unloading a "hot" reactor core and loading with new ("cold") fuel assemblies. The initial core is installed by the use of a long rod to which a fuel assembly gripper mechanism has been attached, and the assembly is lowered into position in the reactor core. All of the core fuel cells can easily be reached since the pressure vessel cover has not been installed. The subsequent reloading of a reactor core is accomplished without removing the pressure vessel head, thereby leaving the gasketed seal intact. Additional equipment for loading and unloading is required because the openings to the reactor core are limited to the control rod drive nozzles.

1. Procedure

It is anticipated that the entire reactor core will be replaced at one shutdown and at a time when the expected three-year core life has elapsed. Unloading of the spent fuel assemblies is accomplished with the pressure vessel head in place by using the control rod drive nozzles as passage ways (see Fig. 49). After depressurizing the reactor to atmospheric pressure with the water at a corresponding temperature, the temperature is further reduced by filling the pressure vessel with demineralized cold water from the contaminated-water-(retention)-storage tank (see Fig. 50). This also serves to attenuate core radiation during the fuel-unloading procedure. Water is taken from the discharge side of the contaminated-(retention)-water pump through a hose to an ion exchange bed in the waste-storage sump (see Fig. 6). The water is then pumped through another hose to the hot well, where the feedwater system pumps it to the reactor. The temperature of the water in the reactor vessel can be adjusted by circulating the water through the purification system. Additional cold water is obtained from the 1000-gal plant makeup water-storage tank. With a water height of 9 ft above the reactor core, the dose rate at the pressure vessel head is less than 2 mr/hr at a time 2 hr after reactor shutdown, except directly in the nozzle openings, where a dose rate of 140 mr/hr is calculated. After flooding of the core, the control rod drives are removed and the extension shafts detached from the control rods. This leaves an unobstructed region above the core, with the control rods remaining fully inserted.

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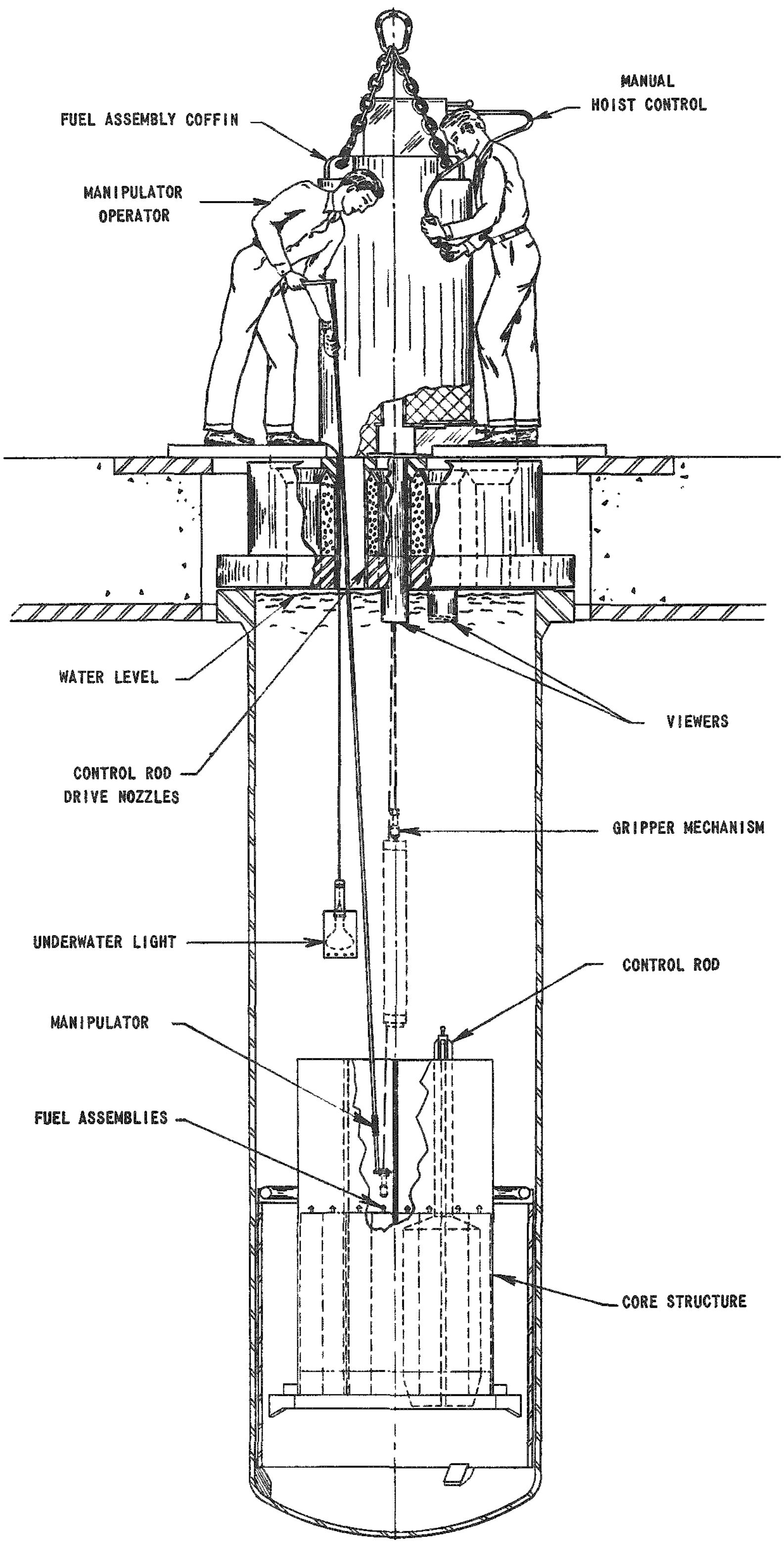

FIG. 49
FUEL ASSEMBLY TRANSFER SYSTEM

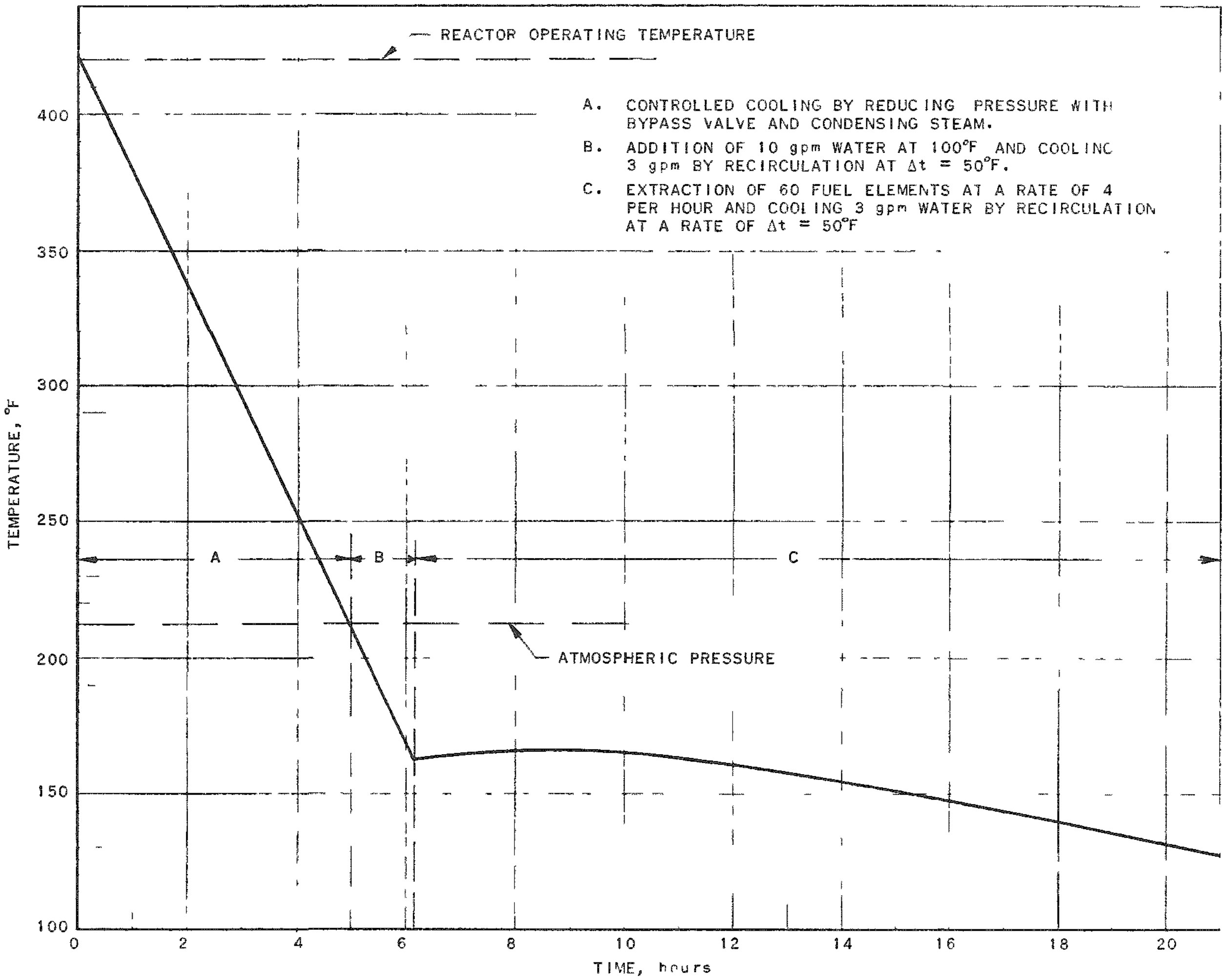

FIG. 50
REACTOR SHUTDOWN TEMPERATURES

The fuel assembly transfer coffin (see Fig. 51) is lowered over the top of one of the control rod drive nozzles, while the other nozzles are used for the insertion of the fuel gripper manipulator, viewers, and lighting equipment (see Fig. 49). Caps, or plugs, are placed on the nozzles not in use.

Two viewers (see Fig. 49) are used to observe the handling of the fuel assemblies through the nozzle openings. They may be located in any combination of nozzles which would permit sufficient view for both the manipulator and hoist control operators. The viewer is filled with water and is designed so that the lower end of the viewer is immersed at least 2 in. into the water. This prevents steaming of the lens at the lower end of the viewer and at the same time creates a solid column of water up to the top of the nozzle. The radiation at the point of viewing is attenuated by the column of water within the viewer (approximately 24 in.) so that the dose rate will be less than 7.5 mr/hr, even with a fuel assembly 2 ft above the reactor core.

The hoist control operator extracts a fuel assembly from the core by lowering the coffin fuel assembly gripper into the reactor pressure vessel by a manually operated hoist until the gripper is immediately above the core structure.

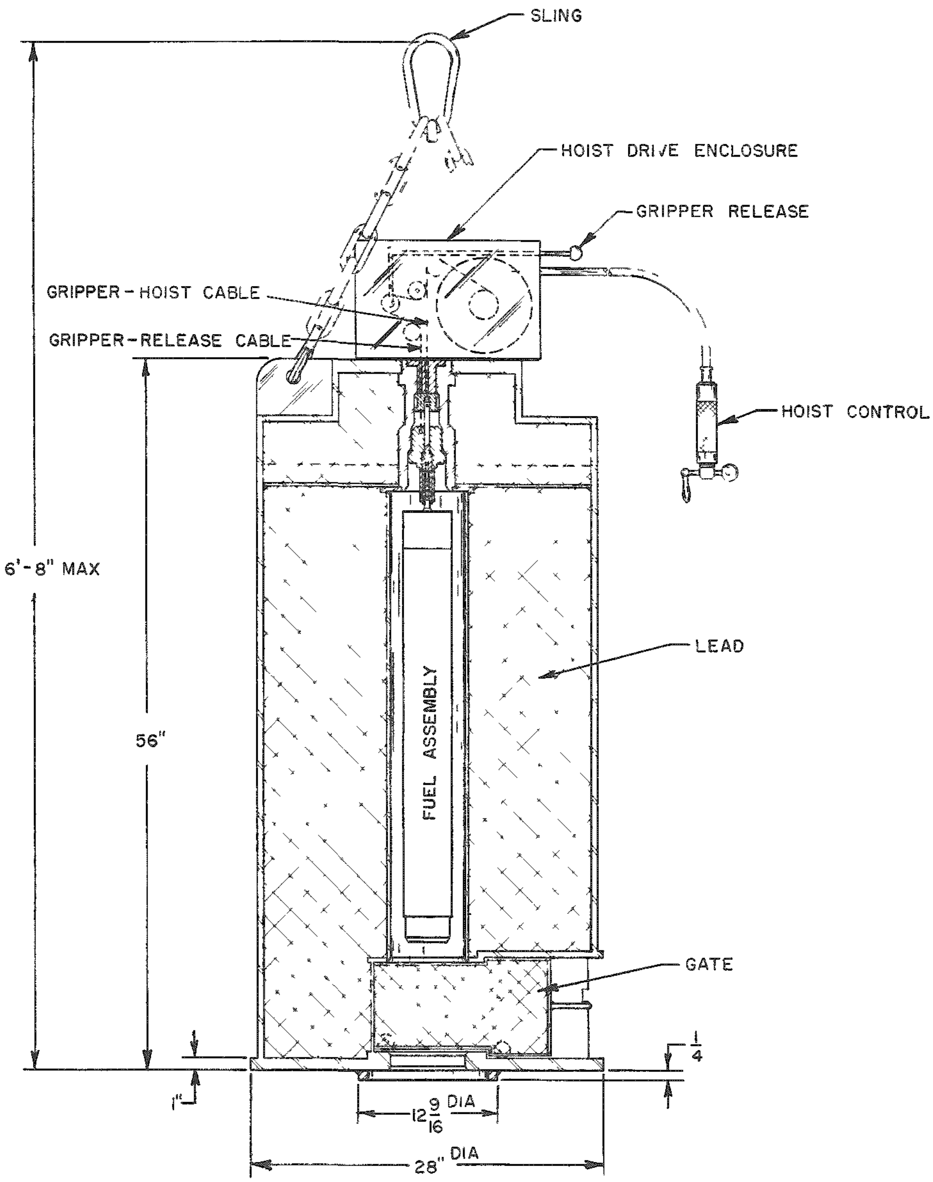

FIG. 51
FUEL ASSEMBLY COFFIN

The manipulator operator then takes hold of the coffin gripper mechanism (similar to that illustrated in Fig. 52), guides it in the direction required until over the designated fuel assembly, and attaches it. As the assembly is raised slowly by the hoist operator, the manipulator operator maintains control of the fuel assembly and guides the coffin gripper and the attached assembly until it swings free of the core structure. The manipulator is detached and the fuel assembly now rests in direct line with the coffin opening. At this point in the procedure, the personnel stand back. The hoist control operator then begins to raise the fuel assembly into the coffin. Indication that the fuel assembly is in the coffin is made by means of an indicator pin at the top of the coffin, which is forced up when the fuel gripper reaches the top of the coffin. Personnel then return only after the coffin gate has been closed and locked in position.

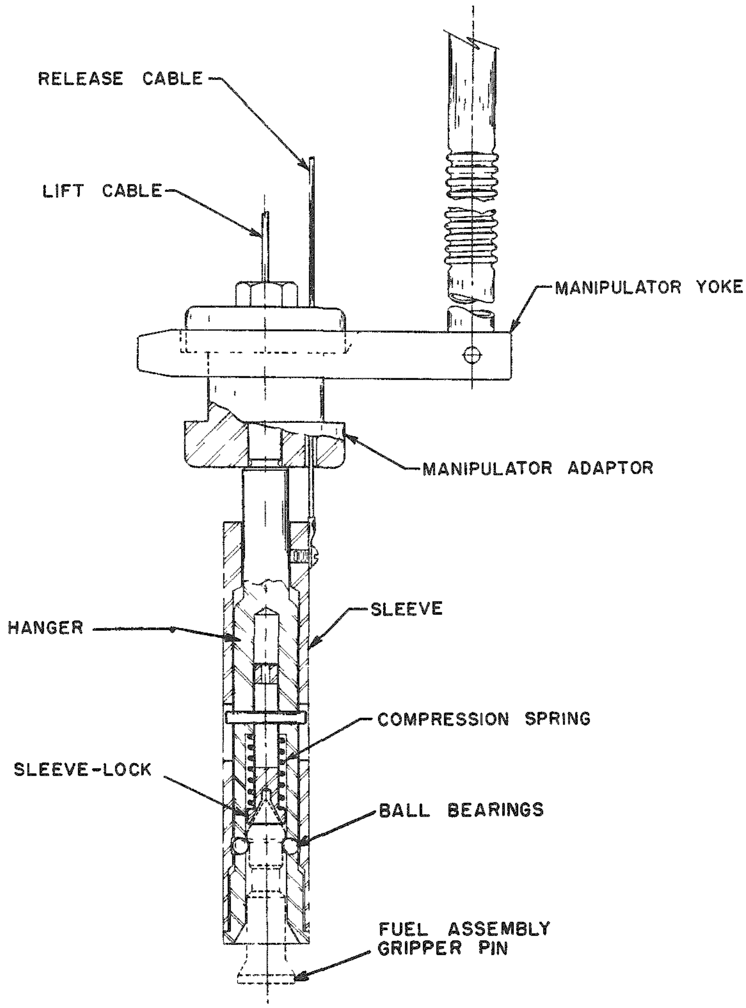

FIG. 52
FUEL ASSEMBLY GRIPPER MECHANISM

The coffin is then transported to the fuel-storage well by means of the bridge crane. The coffin is lowered slowly onto the top of the storage well loading port. The gate of the coffin is opened and the fuel assembly is lowered slowly into the well. The coffin is then returned to the reactor for another fuel assembly. The above procedure is repeated until all the fuel assemblies designated for removal are in the storage well.

The loading of "cold" fuel assemblies is accomplished by the above procedure but without the use of the coffin. (At this point "in-core" instrumentation is needed in addition to the existing "out-of-core" instrumentation.(19)) The "cold" fuel assembly is attached to a fuel gripper mechanism (Fig. 52) fixed on the end of a long rod, the end three feet being flexible tubing. It is lowered slowly into the reactor pressure vessel through one of the nozzles until it is immediately above the core structure. The manipulator operator then takes hold of the fuel assembly gripper and guides the assembly into the designated position. The assembly gripper is released and removed, and the procedure is repeated for subsequent "cold" fuel assemblies.

After reloading the core, the units disassembled (shielding blocks, rod drive connectors, etc.) are reassembled and the water level in the reactor pressure vessel is brought down to the normal operating level.

2. Fuel-handling Equipment

a. Viewer

The viewer (see Fig 49) is flanged on one end and consists primarily of a 16-gauge Type 304 stainless steel sheet rolled to $5\frac{1}{2}$ in. OD. It is seam welded to be water tight and is approximately 27 in. in length. The upper flanged end supports the unit and is designed to contain a plate glass lens with a watertight seal. On the bottom end, a slightly canted plate glass lens is watertight sealed and fastened The entire tube is filled with distilled water. "Weep" holes drilled into the tube wall directly below the bottom lens provide for automatic bleed-off of air bubbles which may rise into the viewing area. Each viewer is equipped with a handle attached to the flange and, although fitting closely in the nozzle, can be easily moved.

b. Fuel Assembly Gripper Mechanism

The fuel assembly gripper mechanism (see Fig. 52) consists of a hanger, sleeve, sleeve-lock, spring, and adaptor. The mechanism is fabricated from stainless steel, with the exception of the spring, which is of Inconel.

In the open, or pin-receiving, position, the gripper sleeve is in a raised position, held in place by the stainless steel ball bearings which, in turn, are held in place by the sleeve-lock shoulder, since the sleeve-lock is in the down position with the compression spring relaxed. When the gripper mechanism is lowered over a fuel assembly gripper pin, the weight of the mechanism is sufficient to compress the spring. This pushes the sleeve-lock upward, and the ball bearings are permitted to fall in place, locking the fuel assembly gripper pin in position at the machined undercut. When the ball bearings fall into position, the sleeve drops and locks the ball bearings in place. The fuel assembly is now ready to be moved.

Releasing the fuel assembly can be accomplished only when the load is completely removed and when the gripper mechanism is in line with the fuel assembly. Any "cocking" will not permit the seating of the gripper mechanism which is necessary to grasp or release the fuel assembly gripper pin With the fuel assembly in a full-rest position and the gripper mechanism in vertical alignment, the sleeve is raised by means of the release cable. After the sleeve has been raised to its maximum position, the ball bearings fall into the machined recess of the sleeve, which then releases the fuel assembly gripper pin. As the gripper mechanism is lifted, the sleeve-lock is forced downward to hold the ball bearings in place and thereby retain the sleeve in position to receive another fuel assembly gripper pin.

c. Fuel Gripper Manipulator

The distance from the top of the core to the opening of the control rod drive nozzles is such that an extremely long retrieving tool is required (see Fig. 49) However, since the overhead clearance of the operating floor is small, the pistol-grip manipulator is designed as two sections, each approximately 7 ft long. The lower half consists of a "Y"-type yoke connected to a short section of flexible metal tubing which, in turn, is silver-soldered to a $\frac{1}{2}$-in diameter stainless steel tube (see Fig. 52). This section of the manipulator is fastened to the pistol-grip section by means of a Ball-Lok pin button to form a semirigid, 14-ft tool. A short section of small safety chain is fastened between both sections to prevent the dropping of the lower section into the vessel. The "Y" yoke is designed to fit around and beneath an undercut slot on the fuel gripper mechanism. The flexible tubing permits additional maneuverability.

d. Underwater Light

The underwater light is approximately $14\frac{1}{2}$ in. long by $5\frac{1}{2}$ in. in diameter and is designed to fit through the nozzle openings. The light is shock-shielded by a stainless steel jacket and windowed by a Lucite lens. The jacket is perforated at the top and bottom to permit water circulation and cooling. The light socket can be used with either a special 1000-watt water-cooled bulb or a standard 150-watt outdoor reflector-type spotlight.

B. Fuel Transfer Coffin

1. Description*

A cylindrical, lead-shielded coffin (see Fig. 51) is used to transfer irradiated fuel assemblies, or the "hot" antimony rod from a neutron source, between the reactor and the fuel storage wells or the outside of the building The coffin is 27 in. in OD with a 6-in. ID stainless steel pipe at the center for the fuel assembly. The space between the outside wall ($\frac{1}{2}$-in.-thick steel) and the 6-in. pipe is filled with lead. The top of the pipe is capped by the fuel assembly gripper assembly and the bottom is closed by a movable lead-filled gate.

The top-mounted gripper assembly has a gripper similar to that described in section A-2-b. The gripper is attached to cables that are run through a pulley train and operated by a crank at the end of a flexible cable This allows the operator to stand back at a distance when the irradiated fuel assembly passes up and into the coffin, since the entrance zone of the coffin is thinner in shielding than the main part. An indicator pin raises and shows when the gripper is at the top. An audible indicator could easily be installed if desired.

The bottom gate is a steel box that is stepped and filled with lead. It is rolled into place after the coffin contains the fuel assembly.

The bottom of the coffin is equipped with a centering ring to locate the coffin when resting it on a reactor rod drive nozzle or the fuel storage well loading port. The coffin is lifted by a tri-sling fastened to three top-mounted lifting plates, or, where height is critical, it can be lifted directly, in a slightly tipped position, by one lifting plate.

2. Shielding Efficiency**

After fabrication, the coffin was checked for radiation streaming, voids or cracks, and to estimate the shielding effectiveness by means of a radioactive source. The coffin is designed to limit the dose from the most highly activated fuel assembly to 100 mr/hr at the outside surface at a time 1 day after reactor shutdown. A 5-in. long, gamma-emitting (Na^{24})

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**A. D. Rossin

source rod of approximately 300 curies was placed at the fuel assembly position in the coffin. This source corresponds to approximately one-fifth the calculated activity of the most active fuel assembly after 3 yr of operation at full power and after allowing 1 day of shutdown cooling. By moving the 300-curie Na^{24} source axially in the coffin, the following dose values were observed:

Average activity level at surface	30 mr/hr
Maximum activity level at surface	60 mr/hr
Gate (gate closed)	100 mr/hr

A second check was possible during the transfer of the 1000-curie, gamma-emitting (Sb^{124}) reactor startup source rod from EBR-I storage to the ALPR reactor.(32) The antimony rod is 12 in. long, and the center of the rod assumed a position in the coffin corresponding to the center of the fuel zone of an assembly. Dose values observed were:

Beam through bottom (gate closed)	160 mr/hr
Average at surface	<1 mr/hr

3. Fuel Assembly Heating*

After the reactor is shutdown, energy will continue to be generated in fuel assemblies because of the decay of accumulated fission products. When the assemblies are removed from the reactor pressure vessel for transfer to the fuel storage wells, there will be no water surrounding the spent fuel to carry this heat away. The situation pertaining to a fuel assembly remaining in the transfer coffin was investigated by solving some simple limiting situations.

(A) Problems

(1) Time required for the fuel assembly to reach melting temperature, assuming no heat dissipation.

(2) Temperature rise in the coffin, assuming all heat that is dissipated from the fuel assembly goes to the coffin for the length of time in item (1).

(3) Heat dissipation rate and temperature differential required, using time in item (1).

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(B) Conditions

(1) One fuel assembly in coffin

(2) One day after reactor shutdown

(3) In coffin indefinitely

(C) Calculations

Melting point of aluminum = 660°C

Temperature of fuel assembly at removal from reactor = 100°C

Allowable temperature rise = 560°C

Mass, aluminum	per assembly
fuel plates	9 x 200 gm
side plates	2 x 200 gm
uranium	9 x 40 gm
	2560 gm (neglecting end fittings)

Specific Heat (aluminum) = 0.215 cal/(gm)(°C).

Assume specific heat of uranium is the same as that of aluminum.

(A1) Heat required to reach melting temperature:

$2560 \times 560 \times 0.215 = 3.08 \times 10^5$ gm cal = 358 watt hours

Heat generation rate during operation:

$$\frac{3 \times 10^5}{40} = 75 \times 10^3 \text{ watts/assembly.}$$

At one day after shutdown, the rate of release of energy from the fission products is approximately 0.005 watt per watt of operating power, and this rate is not changing rapidly over a period of a few hours. The total rate of energy release per fuel assembly is

$$75 \times 10^3 \times 0.005 = 375 \text{ watts.}$$

Thus, if no heat is dissipated and all radiation is absorbed within the fuel assembly, melting temperatures would be attained in

$$\frac{358}{375} \times 60 = 57 \text{ min.}$$

Actually, about 40% of the energy is self-absorbed and the balance is dissipated in the lead coffin; thus the estimate of the time at which melting will occur can be extended to $2\frac{1}{2}$ hr.

(A2) If all the heat were dissipated in the lead coffin for one hour, the resulting temperature rise would be computed as follows:

Mass of lead = ~10,000 lb = ~4,500 kg

Specific Heat (lead) = 0.031 kcal/(kg)(°C)

375 watt-hours = 322 kcal

$$\Delta T = \frac{322}{4500 \times 0.031} = 2.3°C$$

Thus, the coffin will not get hot due to heat generation in the spent fuel assembly.

(A3) With the above information, it is clear that before melting temperatures could be reached a temperature differential of several hundred degrees centigrade would be developed between the assembly and the coffin. The condition of no heat dissipation cannot exist. Assuming heat transfer by radiation alone and approximately 2 ft^2 of effective heat transfer surface area, a temperature differential of 200°C is sufficient to dissipate all of the heat by radiation to the coffin. Since this effectively limits the maximum temperature rise of the fuel assembly, it can be predicted that no melting of aluminum and resulting escape of fission products would take place if an assembly were to be stuck in the coffin for an extended period of time.

C. Fuel Storage Wells*

1. Description

Fuel assemblies are stored in three identical fuel storage wells located within the shielding gravel at a point approximately 11 ft from the axis of the reactor (see Figs. 6 and 53). The wells are stainless steel cylinders, $19\frac{1}{2}$ in. ID by 15 ft 9 in. long, suspended from a flanged top and fitted into a stationary bottom-expansion cylinder. Inside each well is a cluster of seven 6-in.-diameter tubes which serve as containers for the fuel assemblies. In each tube three fuel assemblies may be stacked,

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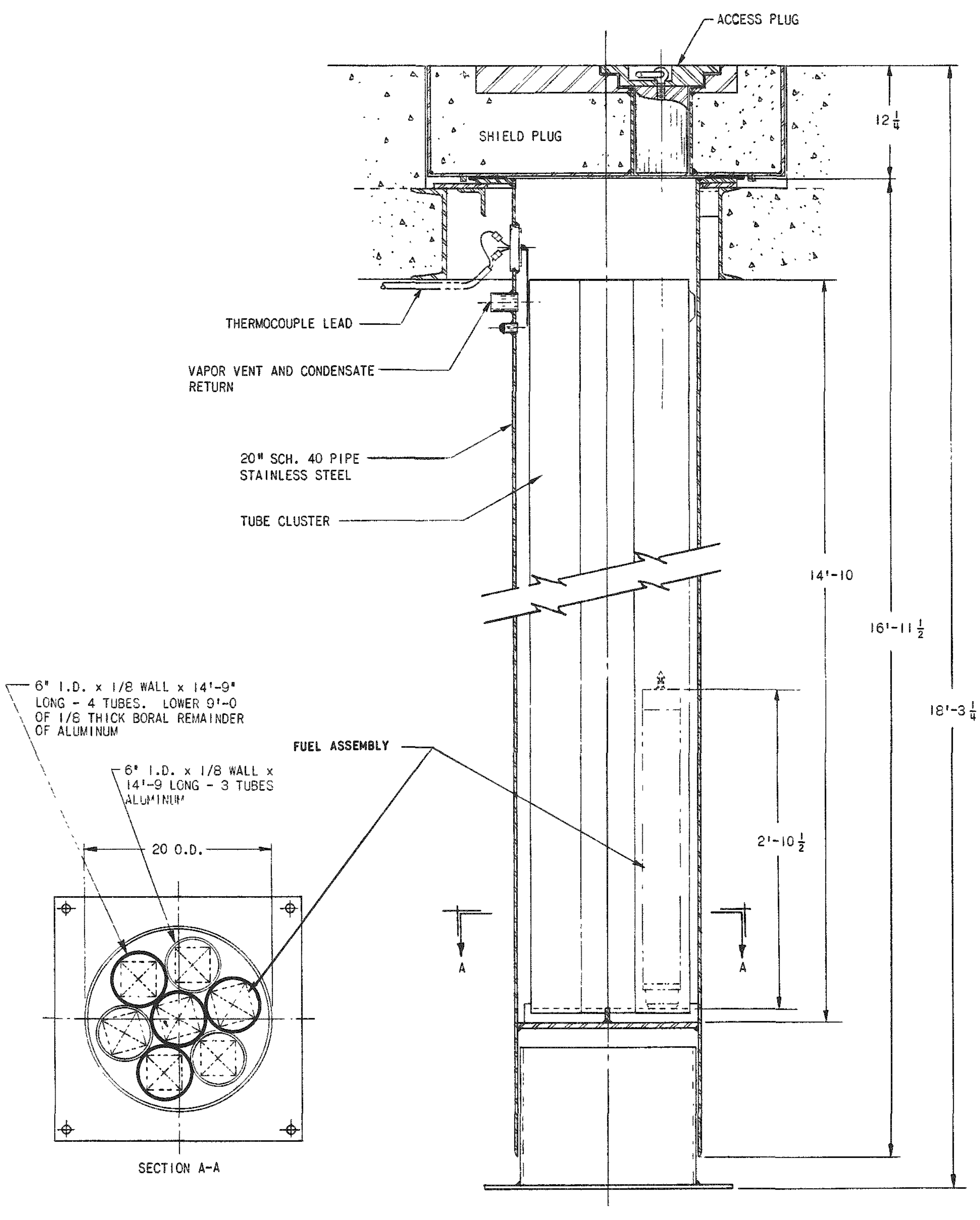

FIG. 53
FUEL ASSEMBLY STORAGE WELL

one above the other, for a total capacity of 21 assemblies per well. The design of the storage well is such that under no circumstances shall any tube contain more than 3 fuel assemblies.

In the region containing the fuel assemblies (the bottom 9 ft), the central storage tube and three alternate tubes around it are rolled from $\frac{1}{8}$-in.-thick Boral. The remaining three tubes are rolled from $\frac{1}{8}$-in.-thick (1100-series) aluminum. This system was designed to keep the storage wells, filled to design capacity, safely below criticality. Above the storage zone, to a total height of 14 ft 9 in., all the tubes are fabricated of aluminum. An aluminum-to-aluminum weld of the Boral tube to the aluminum extension tube was made possible by a special, picture-frame construction of the Boral tube. (Earlier attempts to weld the aluminum tube section directly to the exposed edge of a Boral sheet proved to be unsatisfactory.) The tubes are extended to within $10\frac{1}{2}$ in. of the storage well cover so as to act as guides for the insertion or removal of fuel assemblies.

The 12-in.-thick circular well covers are made of steel and concrete. Together with the water above the fuel assembly zone in the wells, these covers provide shielding. Neoprene gaskets are provided for both the cover and the plug to prevent the escape of water vapor from the well. The weight of the cover itself (approximately 1850 lb) provides a gasket pressure of 9 psi. A $\frac{3}{4}$-in.-square bar forms a protecting ring for the gasket when the cover is placed on a surface other than the well flange. The cover has an eccentrically located loading port which can be positioned over any outside tube by rotating the cover. This has been provided to limit the amount of radiation to personnel during the loading and unloading procedure. To load fuel assemblies into the central tube or to unload assemblies from that tube, the cover must be removed. The central tubes are to be used only when more than 54 fuel assemblies are to be stored. If it should be necessary to use the central tubes, they should be loaded first and unloaded last because of the reduced amount of shielding.

The tube cluster is centered and held in place by an alumina structure supported by lava insulator-separators which provide paths around the sides and at the bottom for the natural circulation of the water. Heated by decay heat from the fuel assemblies, the water flows upward in the tubes, then out through a series of 2-in. holes in the tubes above the 9-ft level, and downward on the outside of the tubes to the bottom, completing the natural-circulation cycle. The insulating separators act also to prevent galvanic corrosion between the aluminum components and the stainless steel tank.

Two thermocouples are used to measure the coolant temperature in the well, one at the bottom and the other at a point near the top, above the water level. Water evaporated in the cooling process is condensed by one of three finned-tube radiators (one per storage well) located in the coolant-air stream on the fan floor. The condensed water is then returned to the same storage well by gravity flow. Thirteen fuel assemblies, representing a cross section of the 40-fuel-assembly core that has been operated at 3 Mwt to equilibrium fission product heating, will evaporate ~50 gal of water per day, initially. After 25 days of storage, the rate of evaporation of water will have decreased to ~20 gal/day (see Fig. 54).

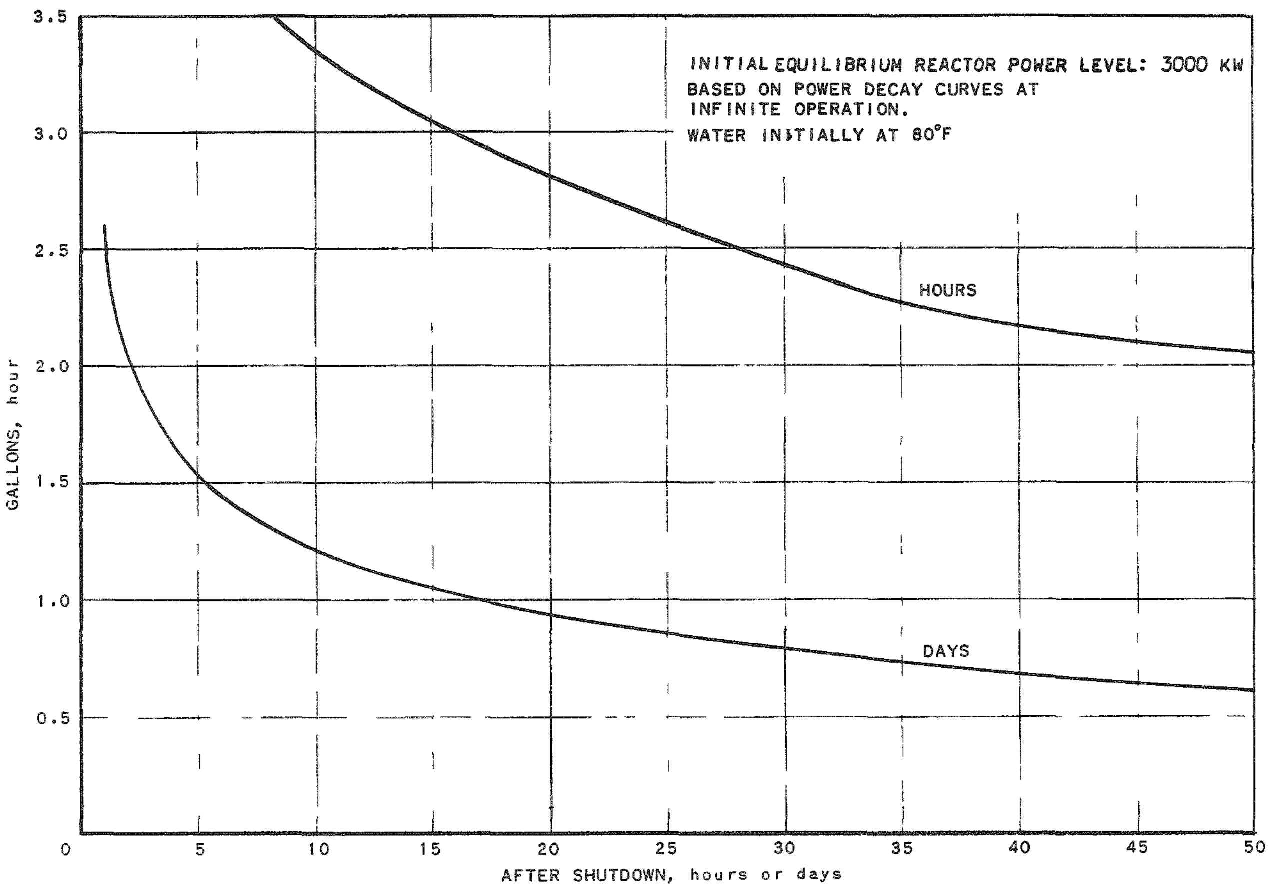

FIG. 54
EVAPORATION FROM FUEL STORAGE WELL (1/3 OF AVERAGE REACTOR CORE OPERATED AT 3 Mwt TO EQUILIBRIUM FISSION PRODUCT HEATING

The fuel storage wells may be empty and dry or they may contain water with or without fuel assemblies. No measurable heat is conducted into the fuel storage wells from the gravel as the result of radiation heating (see section VII-C). Therefore, any water remaining in a storage well not containing spent fuel assemblies will be at approximately ambient temperature. If the well contains fuel assemblies, the fission product heating probably will have raised the temperature of the water to a value higher than ambient. Therefore, it is necessary to withdraw approximately 25 gal of the hot water and replace it with cold water before inserting any additional fuel assemblies, so that no vapor will escape to the personnel area during the fuel transfer. If no fuel assemblies are in the storage well, the water level should be 51 in. below the operating floor level when at room temperature. This allows for water expansion due to temperature rise and displacement by the maximum loading of 21 fuel assemblies in the well.

2. Reactivity of the Loaded Fuel Storage System

The reactivity of a storage well filled with fuel assemblies was estimated as follows. The arrangement of Boral is such that only a small area of each of three alternate outer tubes is not covered by a thermally black absorber. These areas face a thin region of water and steel and are separated from the next storage well by at least $1\frac{1}{2}$ ft of a poorly reflecting, poorly moderating medium of gravel. Reactivity interaction effects between storage wells and with the reactor are small. A cell calculation was made to determine the fraction of thermal neutrons available which are captured by the Boral and the water. Assuming that the fuel assemblies are undepleted (350 grams U^{235} per assembly) and unpoisoned, and that the burnable-poison strips are missing from all assemblies, the calculation of the cold cell showed that $k_\infty < 0.9$; that is, even the most reactive infinite array of such tubes would be subcritical. Estimating the aluminum-to-water ratio to be at least 0.2 in the "core," using a radius of 23 cm ($\approx$9 in.) as an overestimate of the effective "core" size, and assuming a radial reflector savings of 8 cm, the neutron nonleakage factor is found to be smaller than 0.8. Thus $k_{eff} < 0.7$ is obtained, which allows a large margin of error for uncertainties of calculation and of possibly larger future loadings of U^{235} in the individual fuel assemblies.

The computed temperature and void coefficients of reactivity are negative. If burnable-poison strips are attached to the side plates of the fuel assemblies, the reactivity is reduced by perhaps 10% to 12% (~14 dollars to 17 dollars). Since Boral partially surrounds the fuel, the reflector savings is smaller than the 8 cm assumed. Thus, hot or cold, and with or without burnable-poison strips present, the fully loaded array of fuel assemblies in the three fuel storage wells is well below criticality.

A partial confirmation of the computed subcriticality of the loaded fuel storage wells was obtained experimentally when 14 fresh assemblies without burnable-poison strips were loaded into the middle well, forming one complete layer of seven fuel assemblies, a second layer of six fuel assemblies, and a fourteenth fuel assembly in a 3-high stack. (The second tier of assemblies was complete except for one assembly in a peripheral tube used to hold the source.) The array of 14 fuel assemblies in demineralized water was observed to be highly subcritical, in view of the small variation in count rate during the loadings.

IX. POWER PLANT COMPONENTS

A. Turbine-Generator Unit*

The turbine-generator unit (see Fig. 55) is rated at 300 kw and consists of a 4514-rpm, multistage impulse turbine (see Fig. 56) coupled to a 1200-rpm synchronous generator through a reduction gear. The exciter is direct-connected to the generator. A summary of turbine-generator data is presented in Table 16.

The design requirements(26) were based upon a unit that would provide the maximum reliability consistent with steam-consumption economy. (Single-stage units comprise the class for highest reliability and steam-consumption rate, and multistage units sacrifice some reliability to obtain improved steam-consumption rates.) For the nominal three years of continuous operation, a nine-stage unit manufactured by the Worthington Corporation was selected. The performance guarantee for the rated conditions is presented in Table 17.

Air inleakage at the turbine seals results in a higher-than-normal plant water loss as a saturated noncondensible gas through the air-ejector system. Because of the small size of the unit, commercial practice restricted the seals to carbon rings. Figure 57 illustrates the steam sealing system used in conjunction with the carbon rings. Figure 57 also illustrates the Air Ejector System which exhausts the inleakage and prevents radioactive steam from entering the personnel area.

In order to meet the electrical frequency and voltage requirements (see Appendix V), several schemes were considered (see section X-B-2). The selected scheme was predicated on the basis that the turbine governor could maintain the turbine-generator unit speed within the required frequency and voltage limits. The governor selected was the Woodward UG-8, directly connected to the turbine shaft, but self-contained (see Fig. 58). The same model of governor is used on turbines that drive the calender rolls in the paper industry.

The unit is on a single base plate which simplifies field installation and alignment. Although the unit can be furnished with automatically or manually operated steam nozzle valves, the valves were fixed in the full-open position. Their value lies in increased steam economy during variable power operation, although their use in the prototype was questionable with respect to reliability, ease of maintenance, and avoidance of potential steam leakage out through the valve stem packing glands. The sentinel valve on the turbine exhaust casing was also removed, since sufficient instrumentation was available to detect overpressure. In addition, personnel safety considerations prohibit the venting of radioactive steam to operating areas. Reclamation of such vented steam is economically impractical.

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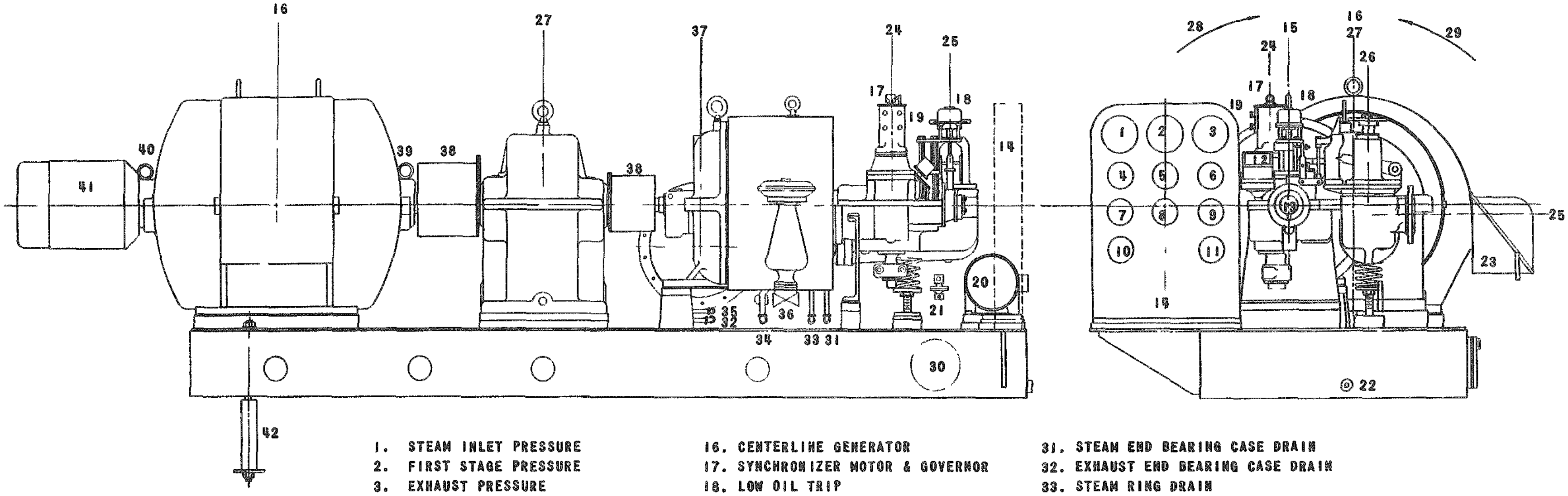
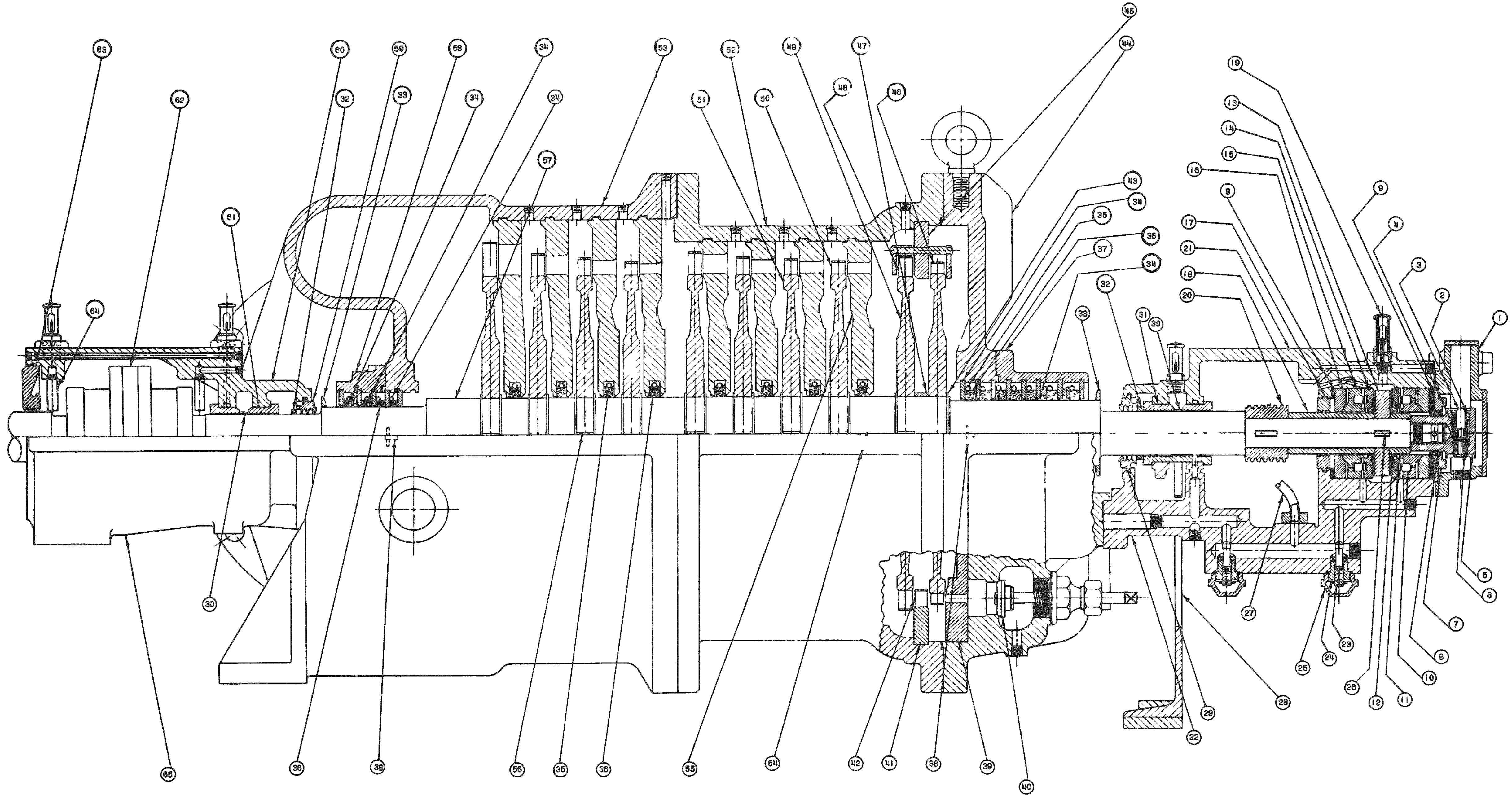
*E. E. Hamer and H. H. Hooker.



1. STEAM INLET PRESSURE
2. FIRST STAGE PRESSURE
3. EXHAUST PRESSURE
4. STEAM END BEARING THERMOMETER
5. BEARING OIL PRESSURE
6. EXHAUST END BEARING THERMOMETER
7. GEAR DRAIN THERMOMETER
8. OIL TANK THERMOMETER
9. OIL COOLER THERMOMETER
10. STEAM END GLAND PRESSURE
11. EXHAUST END GLAND PRESSURE
12. TACHOMETER
13. ELECTRIC TACHOMETER GENERATOR
14. GAUGE BOARD
15. CENTERLINE TURBINE
16. CENTERLINE GENERATOR
17. SYNCHRONIZER MOTOR & GOVERNOR
18. LOW OIL TRIP
19. SPEED ADJUSTMENT CONTROL KNOB
20. OIL PUMP AND MOTOR
21. SOLENOID VALVE - THROTTLE TRIP
22. OIL TANK DRAIN
23. GENERATOR TERMINAL BOX
24. CENTERLINE GOVERNOR
25. CENTERLINE STEAM INLET
26. CENTERLINE TRIP VALVE
27. CENTERLINE GEAR BOX
28. ROTATION OF TURBINE AND PINION
29. ROTATION OF GENERATOR AND GEAR
30. OIL TANK CLEAN OUT
31. STEAM END BEARING CASE DRAIN
32. EXHAUST END BEARING CASE DRAIN
33. STEAM RING DRAIN
34. CASING DRAIN
35. EXHAUST END CASING DRAIN
36. SEALING STEAM SUPPLY REGULATING VALVE
37. CENTERLINE STEAM EXHAUST
38. SHAFT COUPLING GUARD
39. INBOARD GENERATOR BEARING THERMOMETER
40. OUTBOARD GENERATOR BEARING THERMOMETER
41. EXCITER
42. TYPICAL ANCHOR

FIG. 55
TURBINE-GENERATOR UNIT



1. OVERSPEED TRIP HOUSING
2. OVERSPEED TRIP WEIGHT ADJUSTING SCREW
3. OVERSPEED TRIP WEIGHT
4. OVERSPEED TRIP SPRING
5. OVERSPEED TRIP GOVERNOR CUP
6. OVERSPEED TRIP BUSHING
7. ADAPTER
8. OIL SEAL RING
9. THRUST BEARING SHIMS
10. THRUST BEARING BASE RING BUTTON
11. THRUST BEARING SHOE BUTTON
12. THRUST SLEEVE
13. THRUST COLLAR
14. THRUST BEARING SHOE
15. THRUST BEARING BASE RING
16. THRUST BEARING SEAT
17. THRUST BEARING OIL BAFFLE
18. THRUST BEARING SHAFT SLEEVE
19. OIL SIGHT GAUGE
20. WORM GEAR
21. STEAM END BEARING CASE CAP
22. STEAM END BEARING CASE
23. NEEDLE VALVE
24. NEEDLE VALVE CAP
25. NEEDLE VALVE BONNET
26. SHAFT KEY
27. WORM AND GEAR OIL SPRAY TUBE
28. STEAM END SUPPORT
29. STEAM END BEARING CASE OIL BAFFLE
30. OIL RING
31. STEAM END BEARING
32. BEARING OIL SLINGER
33. SHAFT OIL SLINGER
34. PACKING CASE CARBON RING SPACER
35. CARBON RING SPRING
36. CARBON RING PACKING
37 STEAM END PACKING CASE
38. PACKING CASE CARBON RING STOP
39. STEAM INLET NOZZLE RING
40. STEAM INLET HAND VALVE
41. GUIDE BLADING RING
42 GUIDE BLADING RING BUCKET, SPACER AND SHROUD
43. SHRINK RING DISC
44. STEAM END HOUSING
45. WINDSHIELD
46 FIRST ROW CURTIS WHEEL BUCKET, SPACER AND SHROUD
47. SECOND ROW CURTIS WHEEL DISC SPACER
48 SECOND ROW CURTIS WHEEL BUCKET, SPACER AND SHROUD
49. SECOND ROW CURTIS WHEEL AND DISC
50. RATEAU WHEEL BUCKET, SPACER AND SHROUD
51. RATEAU STAGE WHEEL AND DISC
52 DIAPHRAGM COVER
53. EXHAUST END HOUSING
54. DIAPHRAGM PACKING RING STOP
55. DIAPHRAGM
56 RATEAU WHEEL DISC KEY
57. TURBINE SHAFT
58. EXHAUST END PACKING CASE
59. EXHAUST END INSIDE OIL BAFFLE
60. BEARING CASE CAP
61. EXHAUST END BEARING
62. SHAFT COUPLING
63. EXHAUST END OUTSIDE OIL BAFFLE
64. OIL SPRAY TUBE
65. EXHAUST END BEARING CASE

FIG. 56
LONGITUDINAL SECTION THROUGH
MULTI-STAGE STEAM TURBINE

Table 16

TURBINE-GENERATOR DATA

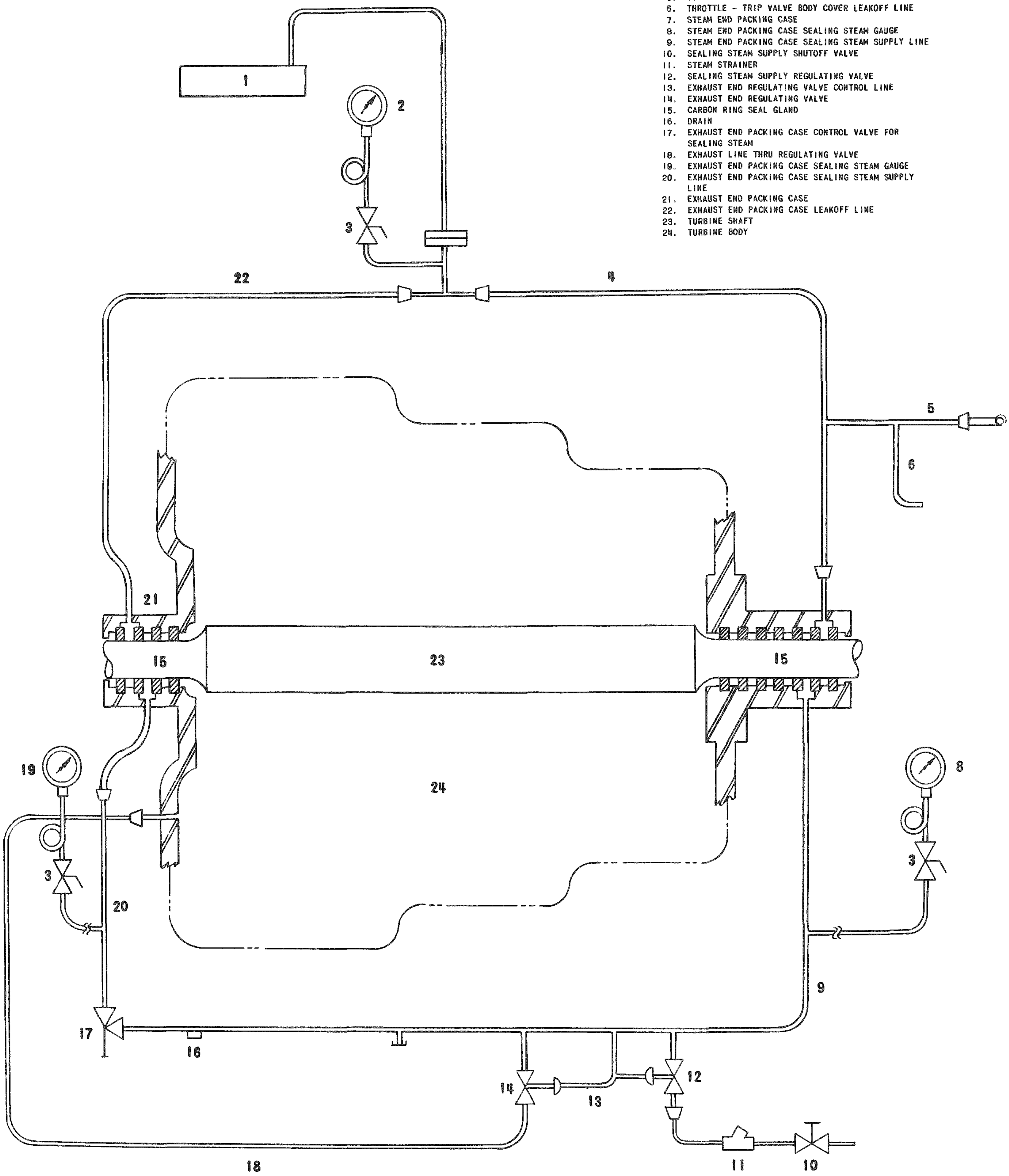
Turbine:	
Rating	300 kw
Capability	375 kw
Speed	4514 rpm
Staging	1 curtis - 8 rateau
Steam Conditions:[a]	
Throttle pressure	275 psig
Throttle temperature	0^{o}F superheat
Exhaust pressure	5 to 8 in. Hg abs
Steam Flow Rate at 300 kw[a]	7500 lb/hr
Governor	Woodward UG-8, direct-acting
Generator:	
Speed	1200 rpm
Rated output at 0.8 p.f.	300 kw
Voltage	120/208 volts, 3 phase
Frequency	60 cycles
Synchronous reactance	110% (375 kva base)
Transient reactance	23% (375 kva base)
Subtransient reactance	13% (375 kva base)
Negative sequence reactance	13% (375 kva base)
Zero sequence reactance	8% (375 kva base)
Short-circuit ratio	1.0
Exciter:	
Type	Direct-connected, shunt wound
Rating	5 kw, 125 volts
Turbine Materials of Construction:	
Casing	
Steam end	Steel
Exhaust end	Cast iron
Rotor forgings	Low alloy steel
Diaphragms	Cast iron
Valves	Stainless steel
Valve seats	Stainless steel
Nozzles	Stainless steel
Wheels	Steel
Bladings	Stainless steel
Bearings	Bronze and babbit
Turbine Gland Seals	Carbon rings and steam

[a]See Table 3, section IV.

Table 17

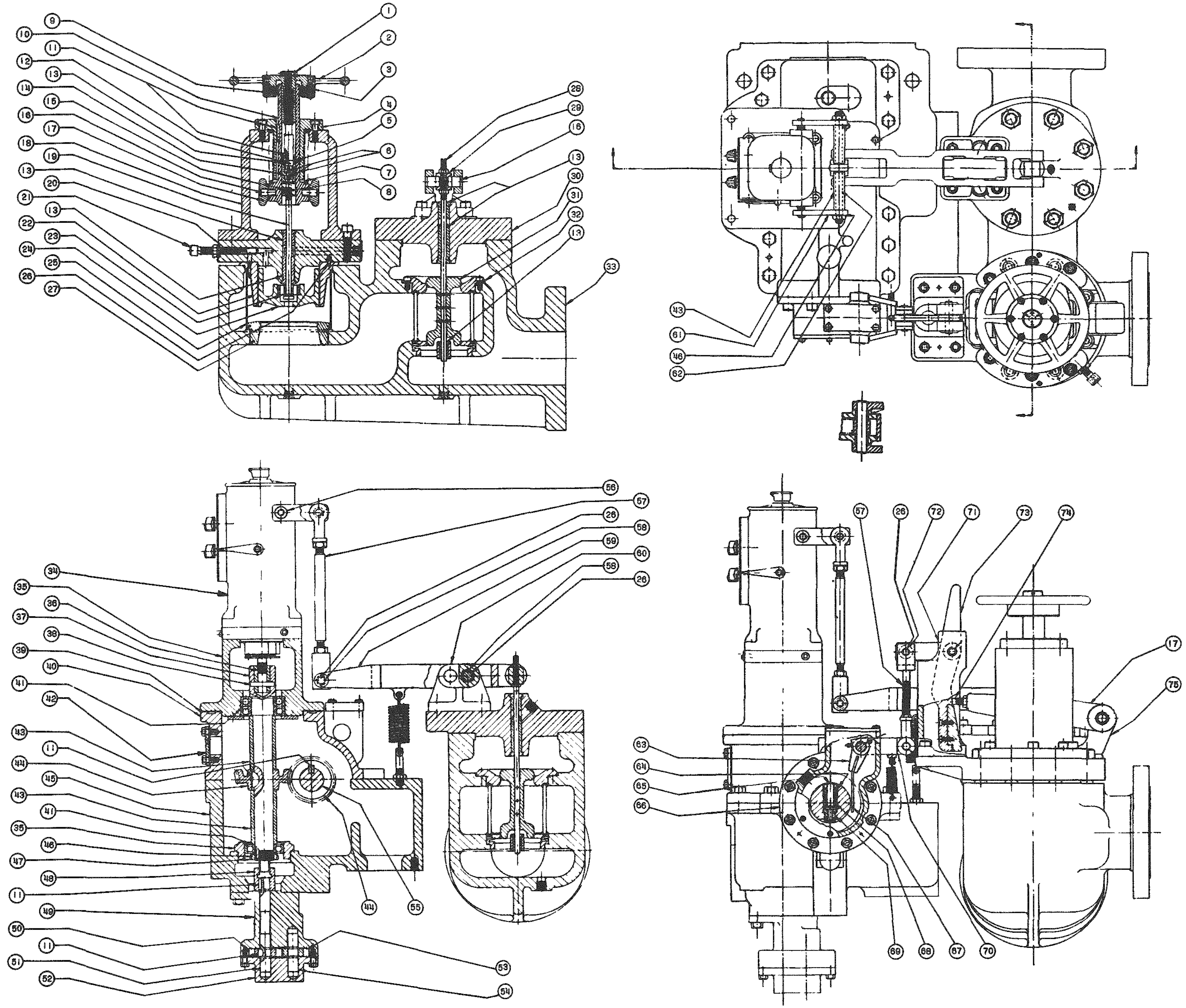
TURBINE-GENERATOR PERFORMANCE GUARANTEE AT RATED CONDITIONS

Load (%)	Load (kw)	Power Factor	Steam Consumption [lb/(kw)(hr)]
25	75	0.80	30.8
50	150	0.80	24.6
75	225	0.80	22.1
100	300	0.80	20.65
125	375	1.0	19.8



1. HIGH PRESSURE CONDENSATE RETURN TANK & AIR EJECTOR SYSTEM
2. GLAND LEAKOFF GAUGE
3. GAUGE COCK
4. STEAM END PACKING CASE LEAKOFF LINE
5. GOVERNOR VALVE BONNET LEAKOFF LINE
6. THROTTLE - TRIP VALVE BODY COVER LEAKOFF LINE
7. STEAM END PACKING CASE
8. STEAM END PACKING CASE SEALING STEAM GAUGE
9. STEAM END PACKING CASE SEALING STEAM SUPPLY LINE
10. SEALING STEAM SUPPLY SHUTOFF VALVE
11. STEAM STRAINER
12. SEALING STEAM SUPPLY REGULATING VALVE
13. EXHAUST END REGULATING VALVE CONTROL LINE
14. EXHAUST END REGULATING VALVE
15. CARBON RING SEAL GLAND
16. DRAIN
17. EXHAUST END PACKING CASE CONTROL VALVE FOR SEALING STEAM
18. EXHAUST LINE THRU REGULATING VALVE
19. EXHAUST END PACKING CASE SEALING STEAM GAUGE
20. EXHAUST END PACKING CASE SEALING STEAM SUPPLY LINE
21. EXHAUST END PACKING CASE
22. EXHAUST END PACKING CASE LEAKOFF LINE
23. TURBINE SHAFT
24. TURBINE BODY

FIG. 57
TURBINE-GENERATOR GLAND SEAL AND LEAK-OFF SYSTEM



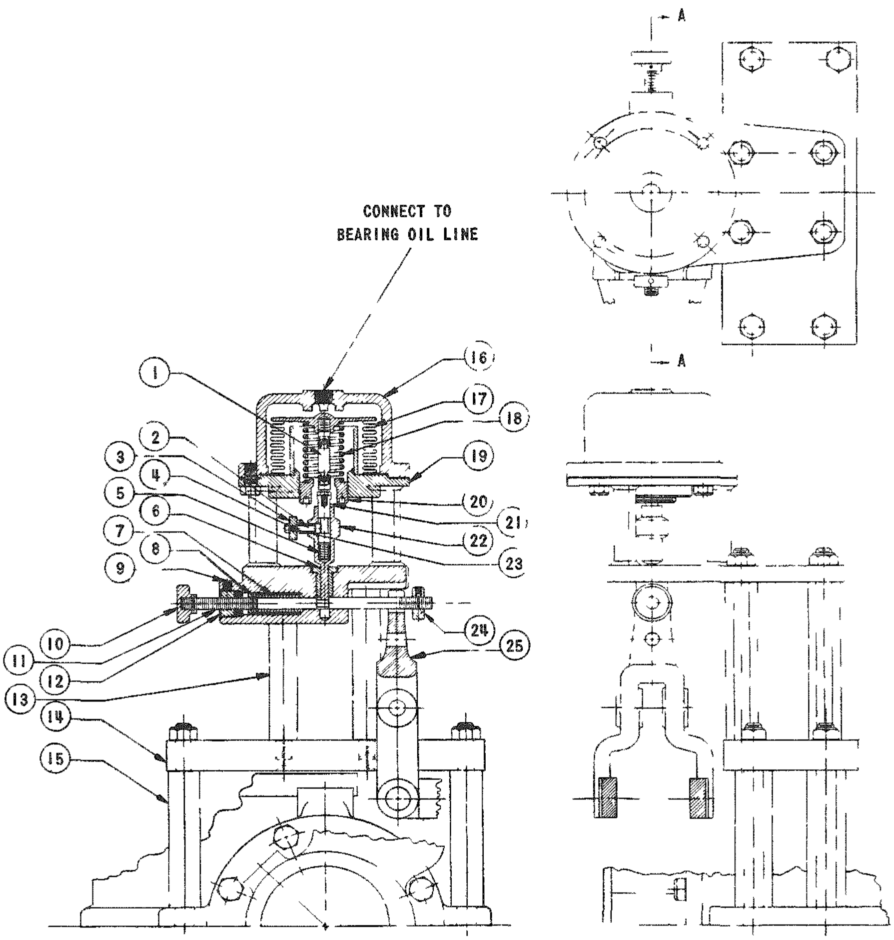
1. TRIP VALVE STEM STOP
2. TRIP VALVE HANDWHEEL
3. BEARING CAP
4. UPPER VALVE STEM GUIDE
5. VALVE STEM STOP BUSHING
6. BUTTON
7. TRIP VALVE SPRING
8. SET SCREW AND BRASS PLUG
9. VALVE STEM BALL BEARING
10. UPPER VALVE STEM
11. KEY
12. GOVERNOR VALVE LIFTING STEM
13. VALVE STEM BUSHING
14. VALVE STEM CONNECTING BLOCK
15. BUSHING
16. VALVE LEVER SLIDING BLOCK
17. TRIP VALVE LEVER
18. GUIDE BRACKET
19. TRIP VALVE STEM
20. TRIP VALVE BODY COVER
21. NEEDLE VALVE
22. TRIP VALVE GUIDE
23. VALVE PLUG
24. TRIP VALVE
25. STEAM STRAINER
26. PIN
27. TRIP VALVE SEAT
28. GOVERNOR VALVE STEM
29. GOVERNOR VALVE STEM BLOCK
30. GOVERNOR VALVE BODY COVER
31. GOVERNOR VALVE
32. GOVERNOR VALVE SEAT
33. GOVERNOR VALVE BODY
34. GOVERNOR (WOODWARD)
35. GOVERNOR BEARING HOUSING
36. GOVERNOR BUSHING
37. GOVERNOR DRIVE SHAFT
38. GOVERNOR BUSHING PIN
39. SHIMS
40. BEARING CASE CAP
41. GOVERNOR DRIVE SHAFT BALL BEARING
42. INSPECTION HOLE COVER
43. SHAFT SPACER SLEEVE
44. WORM GEAR
45. BEARING CASE
46. WASHER
47. LOCKNUT
48. OIL PUMP SHAFT COUPLING
49. OIL PUMP CASE
50. OIL PUMP DRIVE GEAR
51. OIL PUMP DRIVE SHAFT
52. OIL PUMP CASE COVER
53. OIL PUMP IDLER GEAR
54. OIL PUMP IDLER SHAFT
55. TURBINE SHAFT
56. SPLINED BUSHING
57. CONNECTING ROD
58. BEARING
59. GOVERNOR LEVER
60. GOVERNOR LEVER BRACKET
61. ROCKER ARM
62. ROCKER ARM SHAFT
63. INNER TRIP LEVER
64. OVERSPEED TRIP ADJUSTING SCREW
65. OVERSPEED TRIP WEIGHT
66. TRIP LEVER HOUSING
67. TRIP FINGER
68. OVERSPEED TRIP SPRING
69. OVERSPEED TRIP BUSHING
70. OVERSPEED TRIP LEVER
71. LEVER BRACKET
72. BEARING
73. TRIP LEVER
74. TRIP LEVER LATCH
75. TRIP LEVER BRACKET

FIG. 58
SECTION THROUGH GOVERNOR MECHANISM, EMERGENCY TRIP ASSEMBLY AND OIL PUMP

Operation of the unit[17,18] is uncomplicated and commences locally on the plant operating floor. After a turbine warm-up period of 20-30 min, the speed of the unit is increased until the governor takes over. The unit is then controlled from the plant control room. Turbine trip-out is accomplished by closure of the trip-throttle valve by the following methods:

1. Manually
2. Turbine overspeed
3. Low oil pressure
4. Electrical signal to the oil-bypass solenoid valve.

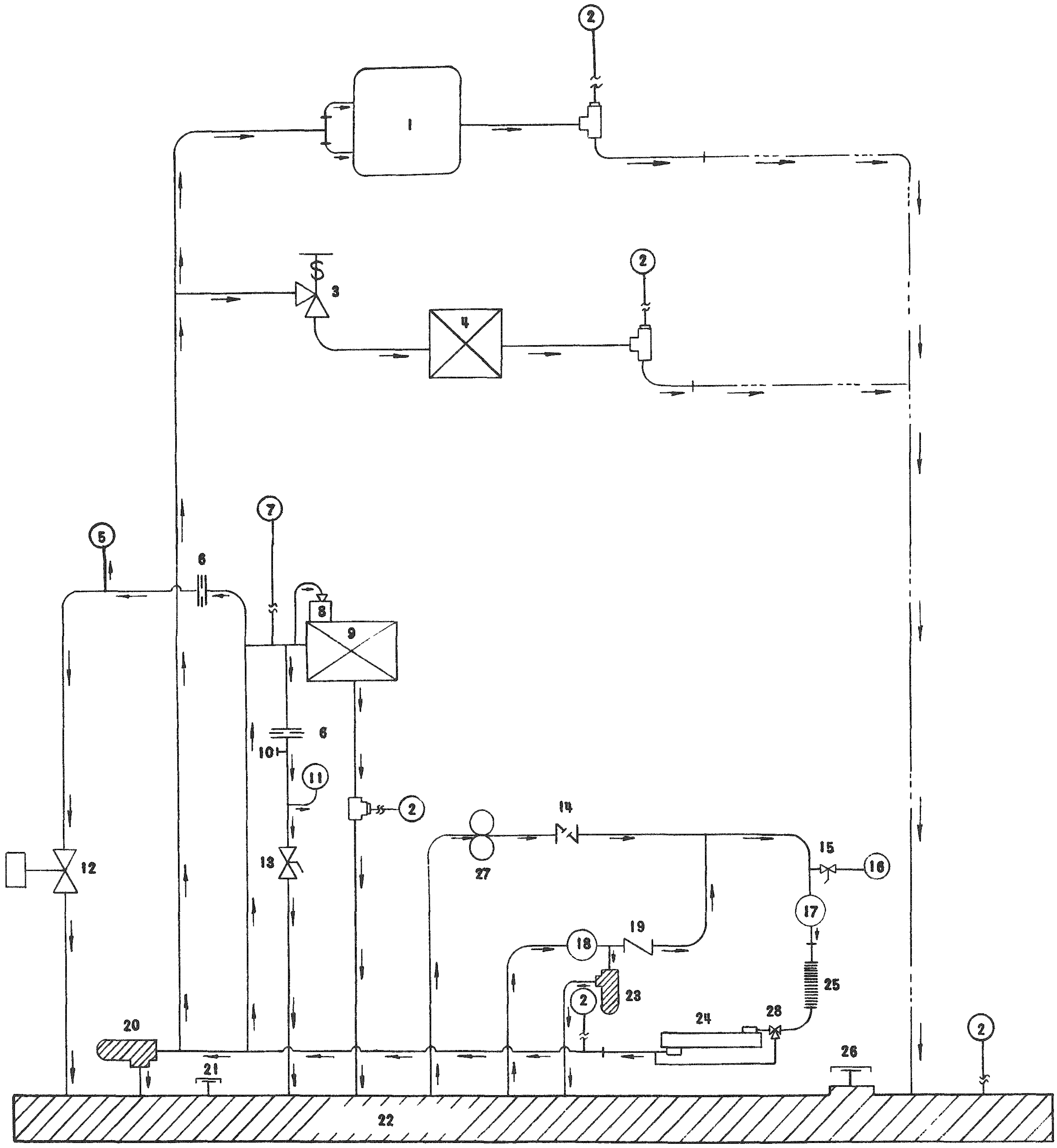
Figures 58, 59, and 60 illustrate the components and methods for tripping the throttle valve.



1. UNIVERSAL JOINT
2. STOP-STEAM TRIP
3. STOP HANDLE-TRIP LEVER
4. STOP BRACKET-TRIP LATCH
5. SPRING-TRIP SLEEVE
6. BUSHING
7. SPRING-PLUNGER ROD
8. SPRING PLATE
9. SET SCREW
10. HANDLE-PLUNGER ROD
11. PLUNGER ROD
12. ADJUSTING SCREW
13. SPACER ROD
14. ADAPTER
15. SPACER STUD
16. BELLOWS HOUSING
17. BELLOWS
18. SPRING-BELLOWS
19. BRACKET
20. NUT-SPRING ADJUSTING
21. STEM-BELLOWS
22. TRIP SLEEVE
23. SPRING-TRIP LATCH
24. STOP-HAND TRIP MECHANISM
25. TRIP LEVER-OUTER

FIG. 59
SECTION THROUGH LOW OIL PRESSURE TRIP ON TURBINE

The maximum steam flow at rated conditions is limited to 7500 lb/hr and is due to the restrictive capacity of the steam nozzles. Steam quality as low as 97% is permissible, but with a penalty of greater blade erosion and a decrease in the steam rate of 2% for each 1% decrease in quality. The steam quality, to date, from the reactor has been measured to be above 99.5% under normal operating conditions (see section V-C-1-a).



1. GEAR CASE
2. THERMOMETER ON GAUGEBOARD
3. SIGHT FEED OILER
4. EXHAUST END BEARING
5. LOW OIL PRESSURE TRIP
6. ORIFICE PLATE
7. BEARING OIL GAGE ON GAUGEBOARD
8. GOVERNOR BEARING HOUSING
9. STEAM END BEARING
10. PLUG FOR TEST GAUGE
11. PRESSURE SWITCH - AUXILIARY OIL PUMP
12. SOLENOID DUMP VALVE
13. TEST VALVE
14. ORIFICE CHECK VALVE
15. GAGE COCK
16. EXCESS PRESSURE GAUGE
17. OIL STRAINER
18. MOTOR DRIVEN STAND-BY OIL PUMP
19. CHECK VALVE
20. LOW PRESSURE RELIEF VALVE
21. OIL TANK AIR VENT
22. OIL TANK
23. STAND-BY OIL PUMP RELIEF VALVE
24. AUXILIARY WATER COOLED OIL COOLER
25. AIR COOLED OIL COOLER
26. OIL GAGE & FILLING HOLE
27. TURBINE OIL PUMP
28. AUXILIARY WATER COOLED OIL COOLER BYPASS VALVE

FIG. 60
TURBINE LUBRICATING OIL SYSTEM

The generator is a standard self-cooled synchronous machine manufactured by the Electric Machinery Manufacturing Company. The generator rotor is equipped with an amortisseur winding for stability. One generator bearing is electrically insulated from the frame to eliminate shaft and bearing currents. The exciter is overhung on the end of the generator shaft opposite the turbine. In addition to the characteristics included in Table 16, the generator reactive capability is shown in Fig. 61.

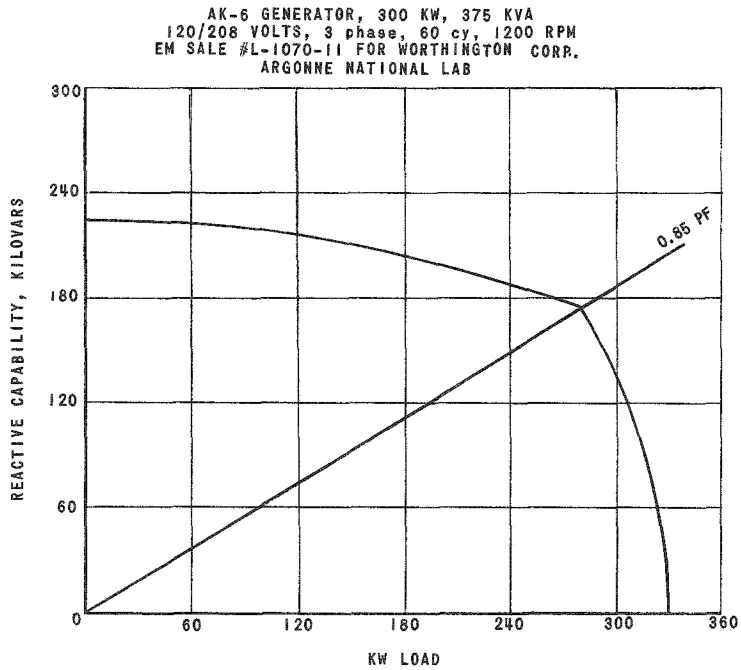


FIG. 61
GENERATOR REACTIVE CAPABILITY CURVE

B. Steam Condensing System*

The steam condensing system consists essentially of (1) the air-cooled condenser sections[26] in which the actual condensing takes place, (2) the air-circulation system, including flow dampers for recirculation of air, a mixing chamber for tempering cold incoming air, and a fan used for moving air over the condenser heat transfer surfaces, and (3) the air-ejector system for removal of noncondensable gases.

1. Air-cooled Condenser Sections

There are five air-cooled condenser sections, assembled in parallel to form one unit and installed in a configuration shown schematically in Fig. 7. One of the condenser sections is shown in Fig. 62.

*A. Smaardyk.

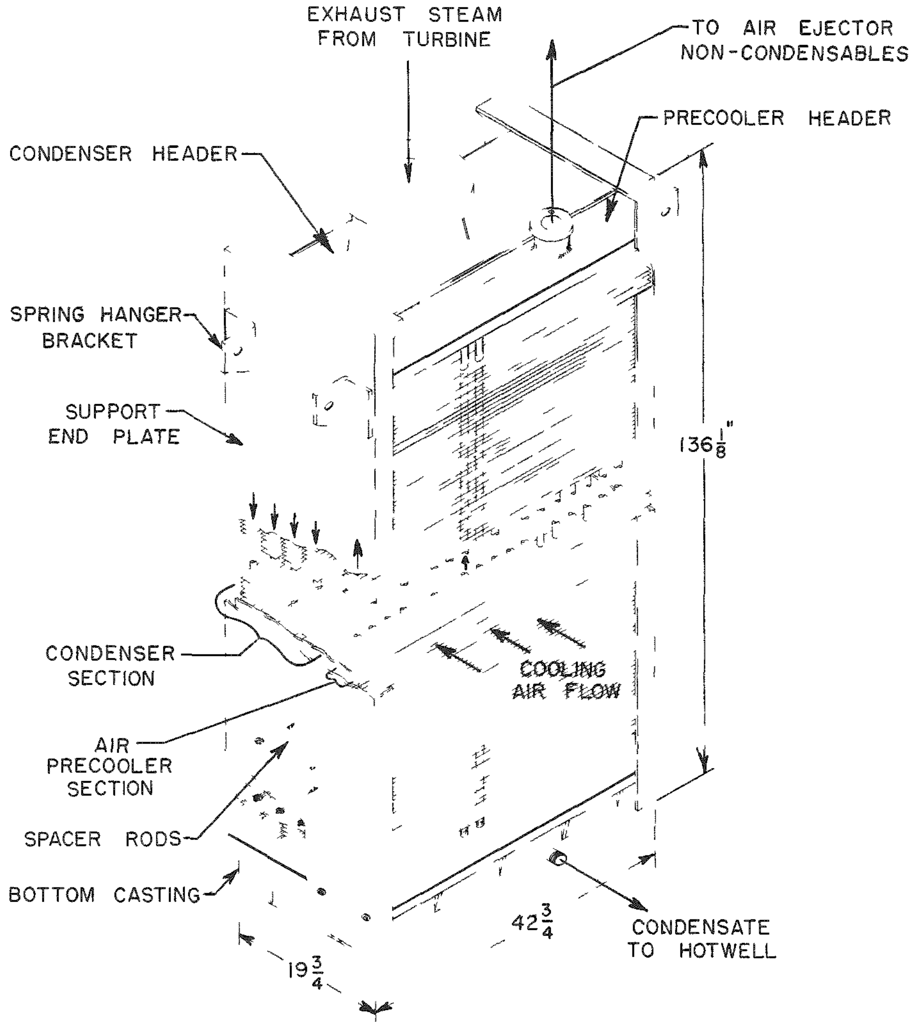


FIG. 62
CONDENSER SECTION

Exhaust steam from the turbine flows into a manifold from which the steam is directed to the five condenser sections. This steam, containing products of dissociation and other noncondensable gases, flows downward and condenses within four vertical rows of condenser tubes. The condensate is collected in the lower condenser casting. Saturated non-condensable gases rise within another row of tubes, called the condenser precooler, and are withdrawn by means of the air ejector.

The water collected in the lower condenser castings is conducted through pipes and seal loops to the plant hot well on the operating floor (see Fig. 18).

The condenser sections are constructed entirely of type 6061-T6 aluminum. There are 95 tubes, of 1-in. OD and 0.083-in. wall, per condenser section. The number of cooling fins per unit tube length is varied from tube row to tube row in order to equalize heat absorption approximately. Table 18 lists this variation and also the metal-to-air heat transfer areas in the various condenser tube regions.

Table 18

CONDENSER HEAT TRANSFER AREAS

Per Section				Per 5 Sections	
				Total Heat Transfer Area (ft^2)	
Row Designation	Number of Fins per Inch	Number of Fins per Row	Total Heat Transfer Area (ft^2)	Air Side	Steam Side
Precooler	4	480	458	2290	220
First Row	7	840	764	3820	220
Second Row	7	840	764	3820	220
Third Row	7	840	764	3820	220
Fourth Row	9	1080	968	4840	220
			Total	18590	1100

The condenser tubes were rolled in the tube headers and then seal welded. All welds and castings were inspected for cracks and pinholes by the dye-penetrant method. Aluminum inserts were shrunk in at the inlet of each tube to reduce the effects of erosion by steam flow. Then the completed units were tested at the factory by complete immersion in water and with nitrogen inside the condenser sections at a pressure of 50 psig. These procedures of construction and testing were followed in order to produce a tight condenser of utmost integrity.

The condenser is designed to remove 7,500,000 Btu/hr from the exhaust steam when the air flow rate is 115,740 SCFM and the air inlet temperature is 60°F.

2. Condenser Air Circulating Fan and Air Damper System

Air, drawn from the outside of the reactor building through the intake duct, flows through an air-mixing chamber and is guided by baffles into the condenser sections (see Fig. 7). The air is heated as it passes through the condenser, and some or all of this heated air is discharged by the condenser fan to the atmosphere through a duct located above the air intake.

The condenser system has been designed for a condenser air inlet temperature of 40°F. Temperatures below freezing should be avoided to preclude icing within the heat transfer coils. However, satisfactory tests have been run with a condenser air inlet temperature of 25°F that indicate a considerable margin in the design parameter.

Generally when the outside air temperature is below freezing, a portion of the heated air leaving the condenser is recirculated and mixed with the fresh incoming air to keep the temperature of the air entering the condenser above the freezing point. Uniform mixing is effected by means of a mixing chamber, consisting of an assembly of equally spaced airfoil-shaped ducts. The recirculating or heated air is discharged from these ducts into the stream of the fresh incoming air flowing over the ducts. Regulation of the recirculation of heated air is accomplished by means of (1) air-mixing dampers located at the trailing edges of the airfoil-shaped ducts and (2) exhaust dampers located in the fan discharge. These dampers are adjusted by means of damper drive motors and are controlled from the control room, either manually or automatically. In the case of automatic control, a temperature recorder-controller measures the average air temperature by means of a group of twelve thermocouples located and distributed at the inlet face of the condenser. The damper controls are designed such that the travel ratio and bias of the dampers are adjustable. The travel ratio, defined as the relative movement between the exhaust and mixing dampers, can be adjusted to a desired setting between 0.5 to 4.5. The "per cent bias" is a variable setting, adjustable between -50 and +50, which setting corresponds to the per cent offset between the dampers in degrees when the mixer damper is in mid-position. When the mixer damper is in a different position than mid-position, the offset is affected by the travel ratio.

A set of manual dampers is installed in the recirculating duct for additional flexibility of control. These dampers are closed when the outside temperature is well above 40°F to minimize recirculation leakage through the mixing dampers.

The condenser air-circulating fan is driven by a 1760-rpm, 75-hp motor, hydraulic coupling, and V-belts. The speed of the fluid coupling output shaft is adjustable from approximately 350 to 1728 rpm. Since the diameter ratio of the output shaft pulley to that of the fan pulley is 9.6 to 44, the maximum fan speed is 377 rpm.

The speed of the fan is varied by regulating the quantity of oil in the working circuit of the fluid coupling. Adjustment of the fan speed is done from the control room, either manually or automatically, by means of a temperature controller. The controlled temperature is sensed by a group of twelve thermocouples distributed over the outlet face of the condenser. For design conditions at full power and at condenser air inlet and outlet temperatures of 60°F and 120°F, respectively, the fan rotates at its full speed of 377 rpm. At this speed and these air temperatures the static differential air pressure has been measured to be ~2 in. WG at 25.37 in. Hg barometric pressure. According to data supplied by the manufacturer, this pressure drop corresponds to an air flow of 128,000 cfm or 7450 lb/min for the same conditions but at sea level.

3. Condenser Heat Transfer and Pressure Drop

The overall heat transfer rate per condenser section is given by

$$Q = 1.085 \times SCFM \times (T - t) \times F \quad ,$$

where

- Q - heat transfer rate for each of the five condenser sections, Btu/hr
- T - entering steam temperature, °F
- t - air temperature at inlet face of condenser, °F
- SCFM - standard cubic feet per minute; cooling air flow referred to standard atmospheric conditions at 60°F and sea level.
- F - factor depending upon the flow rate,

SCFM	F
15000	0.820
20000	0.783
23100	0.748
25000	0.733

This equation is generally applicable for heat transfer at steam pressures down to atmospheric conditions. Its accuracy is then within plus or minus 7%. Its use was extended to the ALPR condenser since it was considered that the equation could be used for pressures below atmospheric.

The heat flow rate to the condenser can be calculated as follows:

$$H_c = W_s h_s - \frac{H_t}{e} - H_s \quad ,$$

where

- H_c - heat flow rate to the condenser, Btu/hr
- W_s - total steam flow rate leaving the reactor vessel, lb/hr
- h_s - enthalpy of steam leaving reactor, Btu/lb
- H_t - turbine output, Btu/hr
- e - overall turbine efficiency, 0.92
- H_s - simulated heat load output, Btu/hr.

The rate of energy input to the condenser may be related to an equivalent flow rate of dry steam. Then the steam pressure drop is given by the following equation:

$$\Delta P_s = 0.520 \times \left(\frac{W_c}{1500}\right)^{1.72} \times \left(\frac{V_c}{181}\right) \times P_s \quad ,$$

where

ΔP_s - steam pressure drop in condenser, in. Hg abs

W_c - equivalent flow rate of dry steam, lb/hr

V_c - specific volume of saturated steam at approximately 3°F less than entering steam temperature, ft^3/lb

P_s - steam pressure at condenser inlet, in. Hg abs.

The air-side pressure drop ΔP_a of the cooling air flowing through the condensers may be obtained from the equation

$$\Delta P_a = 1.30 \times \left(\frac{SCFM}{23100}\right)^{1.76} \times \frac{0.0748}{\rho} \quad ,$$

where

ρ - air density in lb/ft^2 at average air temperature through condenser.

The calculated condenser performance based on the above equations is shown in Fig. 63. Correction factors for altitude and air temperature have been applied.

4. Condenser Performance

Actual condenser performance can be determined by the following three methods:

a. Fan Performance Method

The fan pressure drop and fan inlet and outside air temperatures are measured. Then the fan pressure drop is corrected to the barometric pressure and temperature assumed in calculating the performance curve. The heat transferred can then be calculated by the usual method.

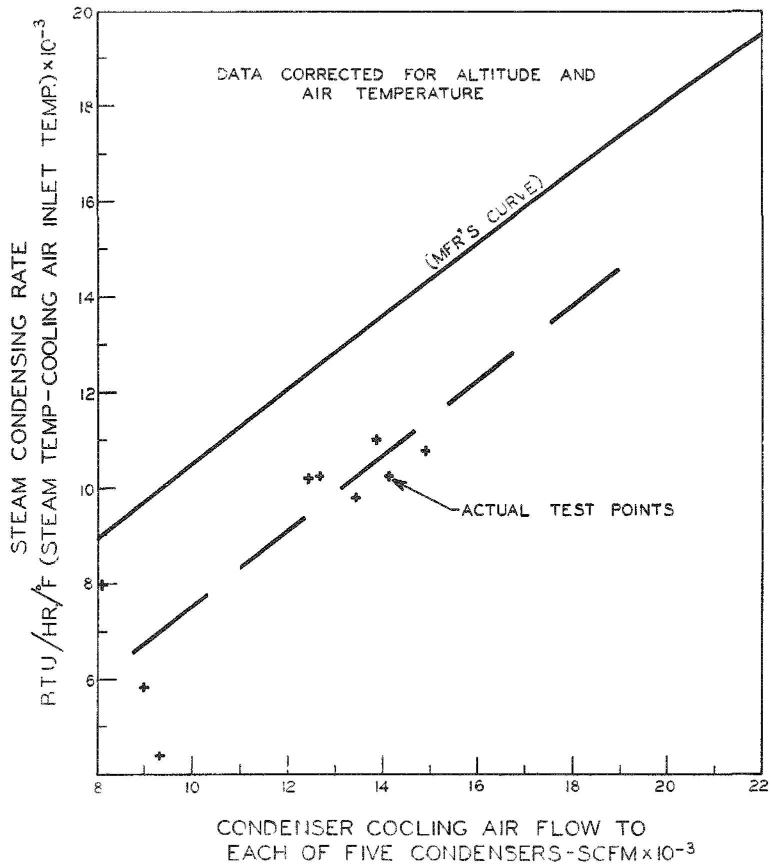


FIG. 63
CONDENSER PERFORMANCE

b. Anemometer Method

Anemometer readings taken across the condenser surface are averaged and multiplied by the condenser face surface area to obtain the air flow rate through the condenser. From this information and the average air temperatures at the inlet and outlet faces of the condenser, the heat transferred can be calculated.

c. Condenser Pressure Drop Method

From the measured condenser air pressure drop, the air flow can be estimated by means of laboratory test data obtained by the manufacturer from representative condenser sections. Then, as in (b), the heat transferred can be calculated.

Actual condenser performance test results have been obtained during plant tests, using the above three methods.[8,11] Complete agreement among the above methods has not been found, probably because of the difficulty and inaccuracy in measuring air flow, pressure drop, and air temperatures. However, the tests have established the following:

(1) a delicately inbalanced steam distribution in the five individual condenser units;

(2) a higher back pressure than predicted; and

(3) a steam-condensing rate, for a given flow of cooling air, which is roughly 20% below manufacturer's predictions (see Fig. 63).

Various means, such as the use of orifice plates to each condenser and throttling of the flow from the precoolers to the air ejector, have been tried in an effort to correct the steam distribution at the ALPR installation, it being assumed that an improvement in distribution would also improve the condenser performance. These efforts were unsuccessful.

The ALPR condenser is unconventional and the first of its kind, and is, hence, experimental in nature. The design has been based on data established for steam-to-air heat exchangers much smaller in size and operating at atmospheric and higher pressures. It is believed that heat transfer coefficients for conditions at atmospheric pressure are applicable to conditions at pressures below atmospheric, so that it is now believed that the condenser performance may be related, firstly, to the geometric shape or size, and secondly, to the manifolding of the condenser system. It is thought that the distribution of steam through long vertical tubes may be significantly different from the distribution of steam over broad interconnected areas as in conventional surface condenser design.

5. Air-ejector System

The function of the air-ejector system is two-fold: (1) to remove air and other noncondensable gases from the condenser and hotwell; and (2) to maintain a partial vacuum on the turbine glands, the leak-off space between the pressure vessel gaskets, and the high-pressure condensate system, thus preventing leakage of radioactive vapor into the atmosphere.

The air-ejector system (Fig. 64) consists of one gland steam precooler, one gland steam ejector, one condenser air ejector, one after-condenser, and one air-cooled after-condenser. The air ejectors are single-stage, steam-jet air pumps using steam to induce a vacuum and to discharge a mixture of air or noncondensables and steam. The discharged mixture is piped into the after-condenser at atmospheric pressure. There the steam is condensed and the noncondensable gases are separated and vented to the atmosphere. The degree of separation is governed by the temperature of the condensate used for coolant in the after-condenser. In order to reduce the water vapor carryover, an additional air-cooled after-condenser is installed in the line venting to the atmosphere.

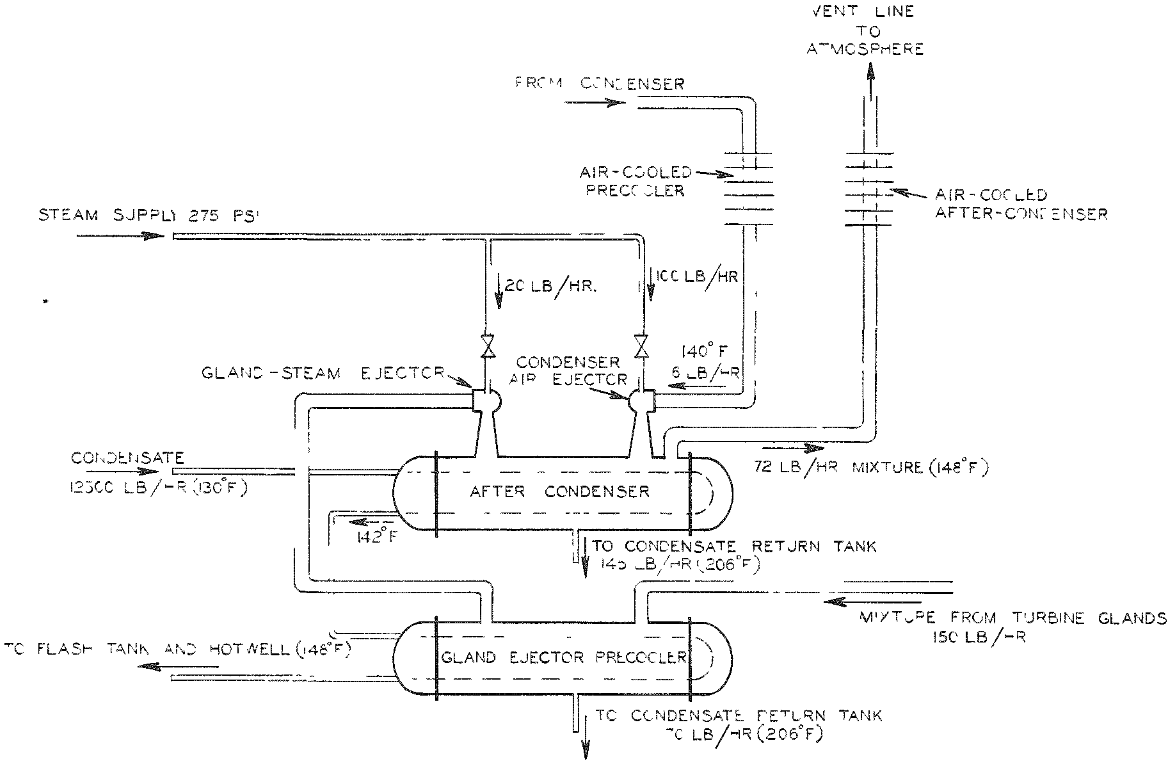


FIG. 64
AIR EJECTOR SYSTEM

The gland-steam ejector maintains 3 in. Hg vacuum and the condenser 25 in. Hg vacuum, referred to operation at sea level. Since the gland-steam ejector operates at a pressure close to atmospheric, most of the condensable vapors can be removed at suction pressure. This is done by a gland-steam precooler which effectively reduces the size of the ejector and the steam consumption.

The discharge line from the ejector system is instrumented for the measurement of the energy level and intensity of radiation of the noncondensable gases (see Fig. 18). Thus, rupture of a fuel plate may be identified by an increase in the level of radiation having energies corresponding to the energies of fission product gases.

The air-ejector system has been designed to remove as much as 16.6 lb/hr of vapor and noncondensables from the condenser at 5 in. Hg abs pressure and simultaneously to remove as much as 150 lb/hr of vapor-air mixture from the turbine glands and pressure vessel leak-off gasket space. The capacity of the condenser steam jet air ejector varies with vacuum conditions (see Fig. 65). Actual tests have shown that the noncondensable decomposition rate of the ALPR is of the order of 16 cc O_2/liter of condensate flow.(12) This corresponds to a rate of removal of noncondensable gas of approximately 0.1 ft^3/min at full power, which is well below the capacity of the condenser air ejector. Therefore, there is a considerable excess capacity available to remove air at startup. Because of the small condenser volume, estimated at roughly 30 ft^3, including some exhaust piping, the rate of removal of air at startup is more than adequate to reach a satisfactory vacuum within minutes from the time the ejectors are started.

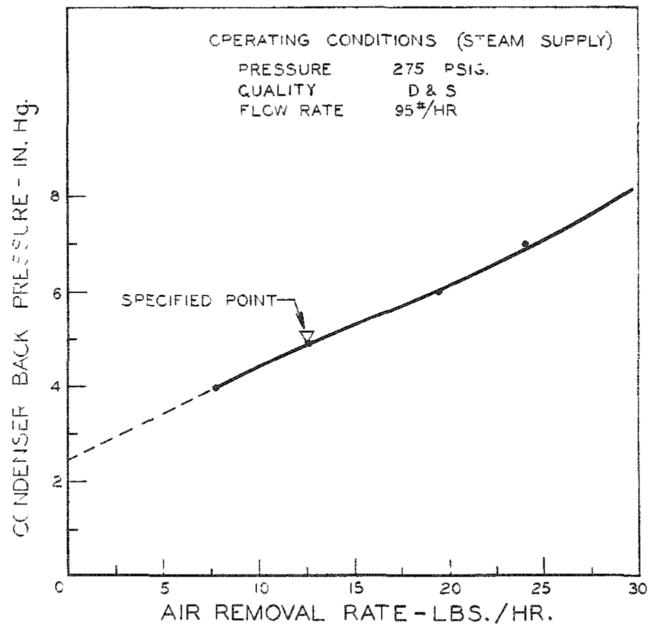


FIG. 65
CONDENSER AIR EJECTOR PERFORMANCE

Air inleakage has been thoroughly checked during plant operation by means of the halogen gas detection method. No evidence of leakage was found.

Table 19 is a summary of pertinent equipment design data.

C. Reactor Feedwater Pumps*

There are two vertically arranged centrifugal-type pumps,[26] each one of which is capable of delivering the full-load feedwater requirements of approximately 20 gpm. Only one pump is normally used while the other is on standby. Automatic change-over is actuated by an interlocking pressure-sensitive switch and occurs upon a loss of feedwater pressure.

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*A. Smaardyk.

Table 19

AIR-EJECTOR SYSTEM DESIGN DATA

Condenser Air Ejector	
Vacuum	25 in. Hg bar
Steam flow required	100 lb/hr
Steam pressure and temperature	275 psig, saturation
Noncondensables from condenser	10 lb/hr
Mixture from condenser	16.6 lb/hr
Temperature of mixture	110°F
Gland-Steam Ejector	
Suction pressure	27 in. Hg abs
Noncondensables from glands	50 lb/hr
Mixture from glands (70 lb/hr is condensed in the precooler between the gland exhaust and the ejector)	80 lb/hr
Steam pressure and temperature	275 psig, saturation
Steam flow required	20 lb/hr
Gland Ejector Precooler	
Mixture entering	150 lb/hr
Vapor condensed	70 lb/hr
Operating pressure	27 in. Hg abs
Capacity	68,300 Btu/hr
Heat transfer surface	23.5 ft^2
Number of tubes and size	(80) - $\frac{3}{4}$ in. OD x 22 BWG
Number of passes	4
After-condenser	
Mixture entering	
from glands	100 lb/hr
from condenser	117 lb/hr
Vapor condensed	145 lb/hr
Operating pressure	atmospheric
Heat transfer surface	46 ft^2
Capacity	148,900 Btu/hr
Number of tubes and size	(116) - $\frac{3}{8}$ in. OD x 22 BWG
Number of passes	4
Air-cooled After-condenser	
This condenser is a dip-brazed aluminum plate-type heat exchanger, one channel deep and 18 channels wide.	
Mixture entering	72 lb/hr
Air flow	535 SCFM
Capacity	10,525 Btu/hr
Heat transfer surface	
Metal to air	102 ft^2
Mixture to metal	70 ft^2
Pressure drop - mixture side	8.88 in. Hg
Pressure drop - air side	1.5 in. WG

The feedpump takes condensate from the hot well (collected from the condenser, air ejectors, and space-heating system) and returns it to the reactor. Six centrifugal stages, or pump impellers, are used to develop the hydraulic head necessary to pump water from the condenser operating at 5 in. Hg abs to the reactor operating at 300 psig. A 1-in. bypass line with a $\frac{1}{8}$-in.-diameter orifice is installed between the pump discharge and the hot well. This provides a continuous bypass flow to prevent the pump from overheating should the feedwater valve be closed completely. A double set of conventional packing rings is used to seal the pump shaft. Water, from the condensate circulating pump, is supplied to the packing rings for cooling and sealing. Since the pump suction pressure at the gland is below atmospheric, the water supply to the gland prevents air inleakage at this point. Conventional packing was selected over mechanical seals because the rate of leakage of cooling water normally increases gradually with the packing and can be adjusted, while a breakdown of the mechanical seal could involve a large and sudden loss of cooling water. Also, replacement of a mechanical seal involves the dismantling of the motor from the pump.

The pumps are commercial units designed for applications with corrosive liquids. All parts in contact with the working fluid are constructed from a 13% chromium steel. Castings are ASTM-A296, CA-15 and other parts are AISI-type 416, both free-machining and hardened as required. The lower bearing bushing, which is lubricated and cooled by the working fluid, is fabricated from ASTM-type 501, hardened 5% chromium, 0.5% molybdenum, stainless steel.

The pumps (see Fig. 66) are driven by 15-hp, 3515-rpm, 208-volt, 3-phase, 60-cycle, synchronous, vertically mounted motors.

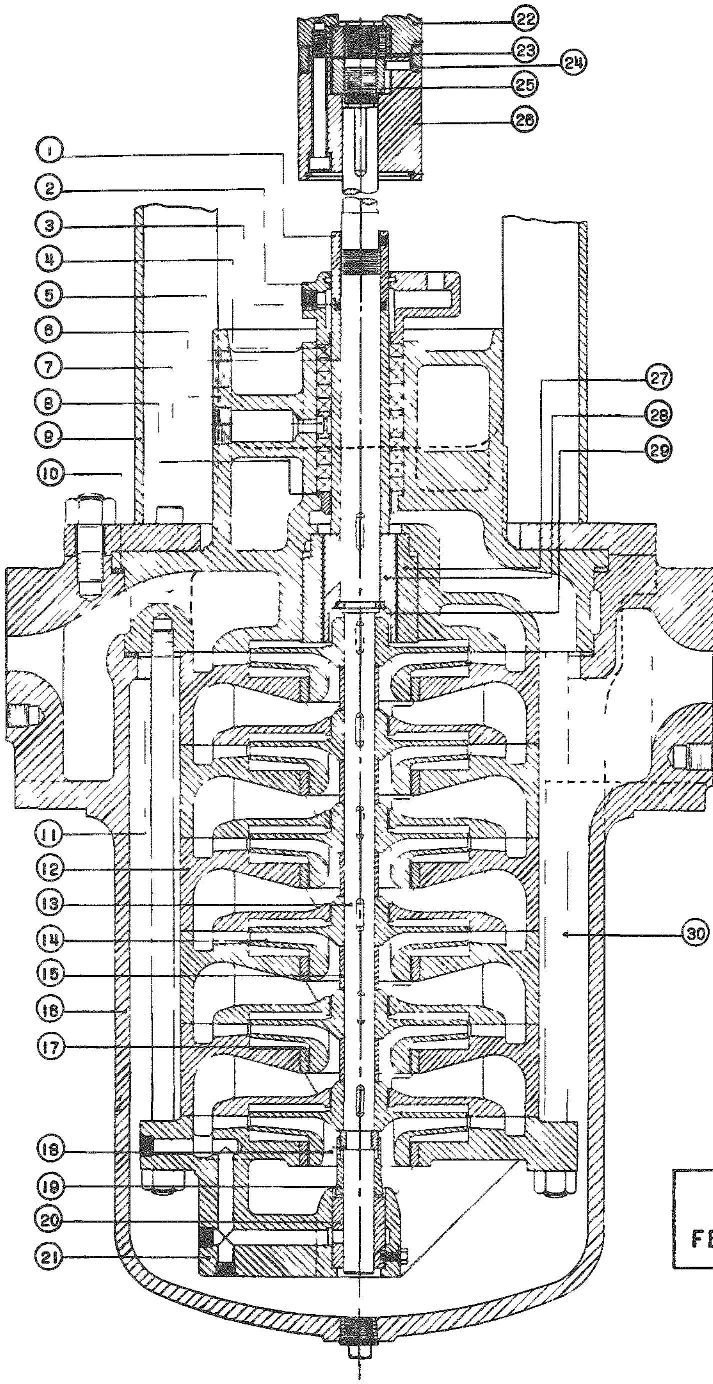
The pump characteristics are shown in Fig. 67.

D. Primary Water Makeup System*

The primary water makeup system provides demineralized water for conditions of plant makeup. Raw (unchlorinated) well water is supplied to the demineralizer and the demineralized water is stored in the adjacent and overhead 1000-gal plant water storage tank (see Fig. 6).

The system consists of a $1\frac{1}{2}$-ft^3 resin capacity ion exchange tank, pre- and post-filters, water meter, flow indicator, an effluent conductivity cell with a valve-controlling-type conductivity bridge, motor-operated shut-off valve, and appropriate piping. The motor-operated valve functions when the conductivity of the demineralized water exceeds a preset (on the conductivity bridge) value (see Fig. 19). The resin used to demineralize the well water is Illinois Water Treatment Co.-type TM-1, commercial grade. This resin is a mixture of a cation resin in the hydrogen form and an anion resin in the hydroxyl form.

*E. E. Hamer



1. SHAFT SLEEVE NUT
2. GLAND-TWO HALVES
3. SECONDARY GLAND PACKING-SHAFT SEAL
4. PACKING-SHAFT SLEEVE
5. SHAFT SLEEVE
6. DISCHARGE HEAD
7. LIQUID SEAL CAGE
8. STUFFING BOX BUSHING
9. SUPPORTING HEAD
10. GASKET-OUTER
11. GASKET-INNER
12. CHANNEL RING
13. SHAFT
14. IMPELLER
15. INTERMEDIATE SLEEVE
16. CASING
17. CASING RING
18. SPLIT RING
19. RETAINING RING
20. BEARING BUSHING-LOWER
21. LOWER BEARING HOUSING
22. COUPLING-DRIVER HALF
23. COUPLING NUT-DRIVER SHAFT
24. ALIGNING RING
25. COUPLING NUT-PUMP SHAFT
26. COUPLING-PUMP HALF
27. THROTTLING BUSHING
28. THROTTLING SLEEVE
29. SPLIT RING
30. ROTOR ASSEMBLY BOLT

FIG. 66
FEED-WATER PUMP CUTAWAY SECTION

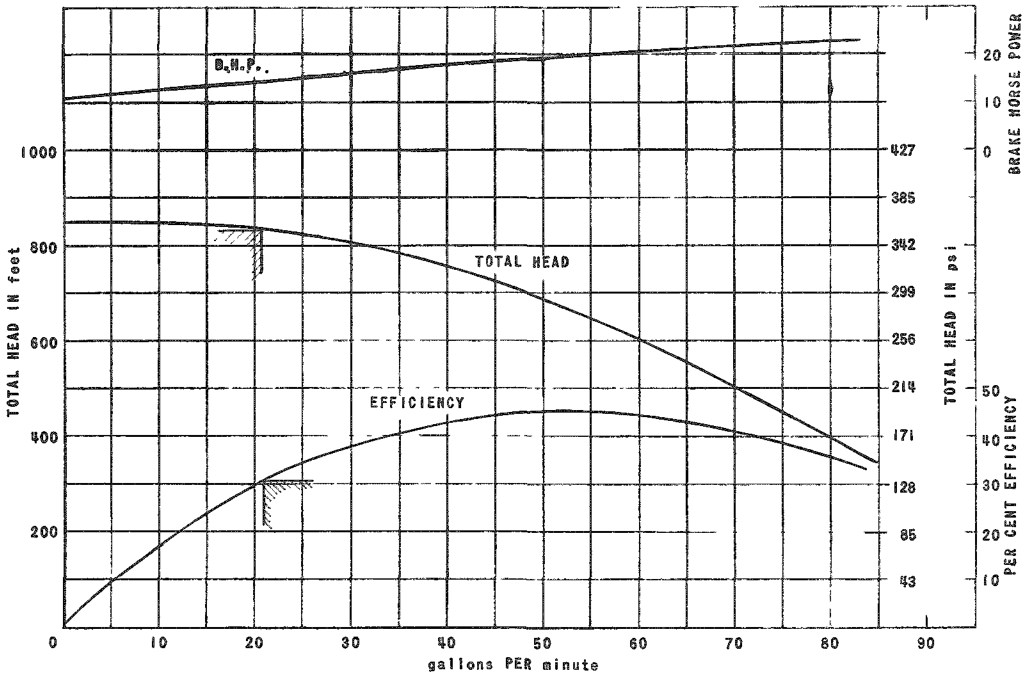


FIG. 67
FEED-WATER PUMP CHARACTERISTIC CURVE
(FEED WATER AT 135°F)

No regeneration of resin is performed. When expended, the resin is replaced. By not regenerating the resin the possibility of contaminating the demineralized water in the storage tank due to operator error during the regeneration procedure is eliminated. Also, in a proposed remote installation, no chemicals need to be transported by air. With floor area in the plant (or any floor area at a remote arctic site) at a premium, the present smaller installation without regeneration equipment is preferred. If so desired, the resin regeneration can be performed elsewhere.

E. Space-heating System*

The space-heating system simulates the approximate conditions of output such as would be required for a proposed arctic installation. An outdoor air-water heat exchanger dissipates the heat and allows variations in heat demand to be controlled manually.

Figure 17 contains a flow diagram of the simulated space-heating system. The heat source is steam, from which the latent heat of evaporation is extracted, and is on the shell side of the heat exchanger. The resultant condensate is returned to the plant hot well via the flash tank. Water on the tube side of the heat exchanger is heated and pumped through the finned-tube air-water outdoor heat exchanger, from which the heat is rejected to the atmosphere (see Fig. 4).

*E. E. Hamer

The steam supply to the heat exchanger is regulated by a water temperature-regulating steam valve with the sensing element in the heat exchanger effluent water line, thus maintaining a constant preset outlet water temperature. The steam pressure is maintained at approximately 20 psig at all times during operation. The system is designed to generate and dissipate 400 kw (1,350,000 Btu/hr) of heat. The remote arctic DEW Line stations presently have a hot water-heating system which supplies hot water at 170-180°F from the plant diesel-engine exhaust gases and oil-fired boilers. With a constant temperature of the water supply, the ALPR space-heat load is varied by changing the mass flow rate through the steam-water heat exchanger by means of a manually operated 3-way bypass valve.

X. ELECTRICAL GENERATION AND DISTRIBUTION SYSTEM*

A. General

The ALPR is intended to serve as a prototype for small power plants for use at remote military installations which do not have access to electric utility lines. Electric power for reactor plant startup and for site operation during reactor shutdown at such intallations is expected to be furnished by diesel-engine driven generators. Consequently, during normal plant operation, the turbine-generator is isolated from other power sources. Parallel operation with the standby diesel-generators occurs only with the transfer of load incidental to plant startup and scheduled shutdown.

As a matter of convenience and to facilitate testing, connections to an electric utility line are provided at the prototype site. For similar reasons the prototype installation does not include a full complement of standby diesel-generators.

Plant design requirements are included as Appendix V.

B. Description of System

The electrical generation and distribution system is shown diagrammatically in Fig. 68. As shown, the main turbine-generator feeds the generator bus, which in turn may be connected to either the utility (main) bus, or to the equipment (auxiliary) bus. Similarly, the standby diesel-generator and the plant auxiliaries may also be connected to either bus. The external utility line (NRTS distribution system) supplies power to the site deep well pump and may be connected to the utility bus. Depending upon switching connections, power for the site support facilities may be obtained either from the utility or from the plant turbine-generator (via the utility bus).

1. Turbine-Generator Unit

The main generator is an Electric Machinery Manufacturing Company salient pole machine which is driven by a Worthington Corporation turbine through a reduction gear. The generator data are included in Table 16.

The generator is connected through a 1200-amp electrically operated air circuit breaker to the generator bus. Per cent differential relays are used for protection against faults between the generator winding neutral tie point and the load side of the generator circuit breaker. A 1200-amp dead front knife switch permits connecting the generator bus to either station bus (utility bus or equipment bus). Key interlocks prevent operation of this switch unless the generator circuit breaker is open.

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*H. H. Hooker

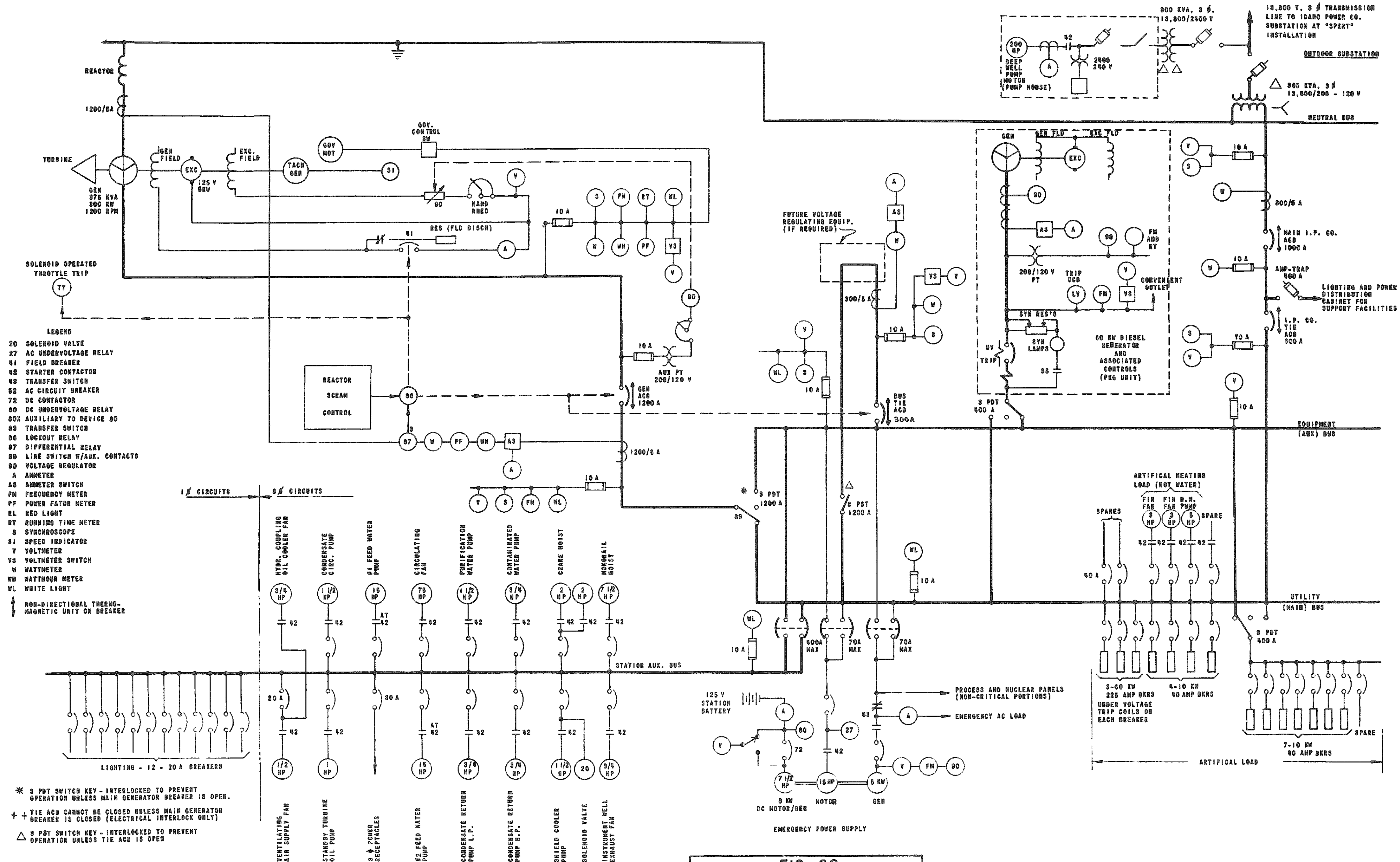


FIG. 68
SINGLE LINE POWER DIAGRAM

Generator voltage regulation is effected by means of a direct-acting rheostatic-type regulator which is connected in the exciter field circuit. The generator excitation circuit is shown schematically in Fig. 69.

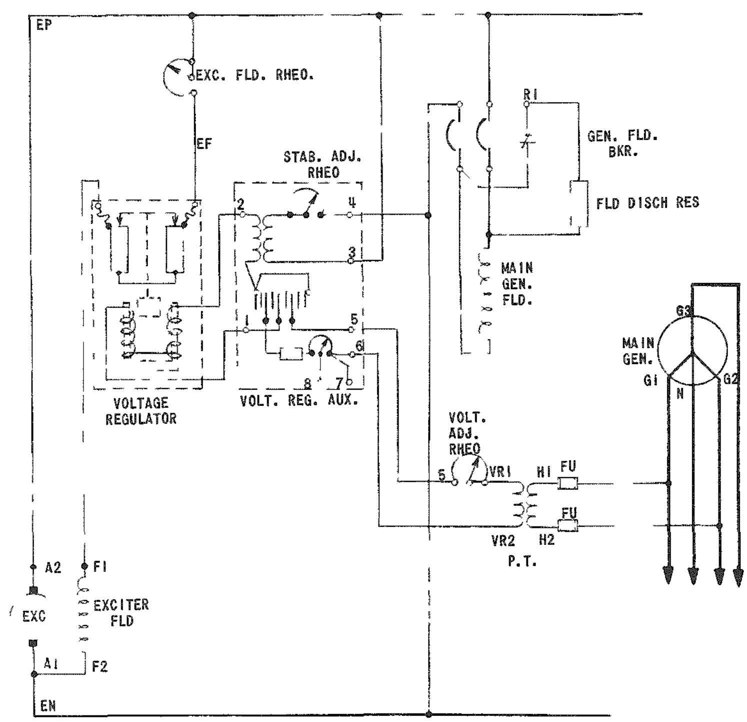


FIG. 69
GENERATOR EXCITATION SCHEMATIC

Turbine speed is controlled by an isochronous Woodward governor which is equipped with a motor-driven speed changer. During the brief periods when the turbine-generator is operated in parallel with the external utility line, stable operation requires that the governor be temporarily adjusted to provide positive speed droop with load. The speed changer motor is energized from the generator leads and is remotely operated from the control room. The generator circuit breaker and speed changer control circuits are shown in Fig. 70.

2. Station Bus Arrangement

The electrical system at a DEW Line Auxiliary Station features two separate station buses. Feeding these buses are four 60-kw diesel-engine driven generators, each of which may be connected to either bus.

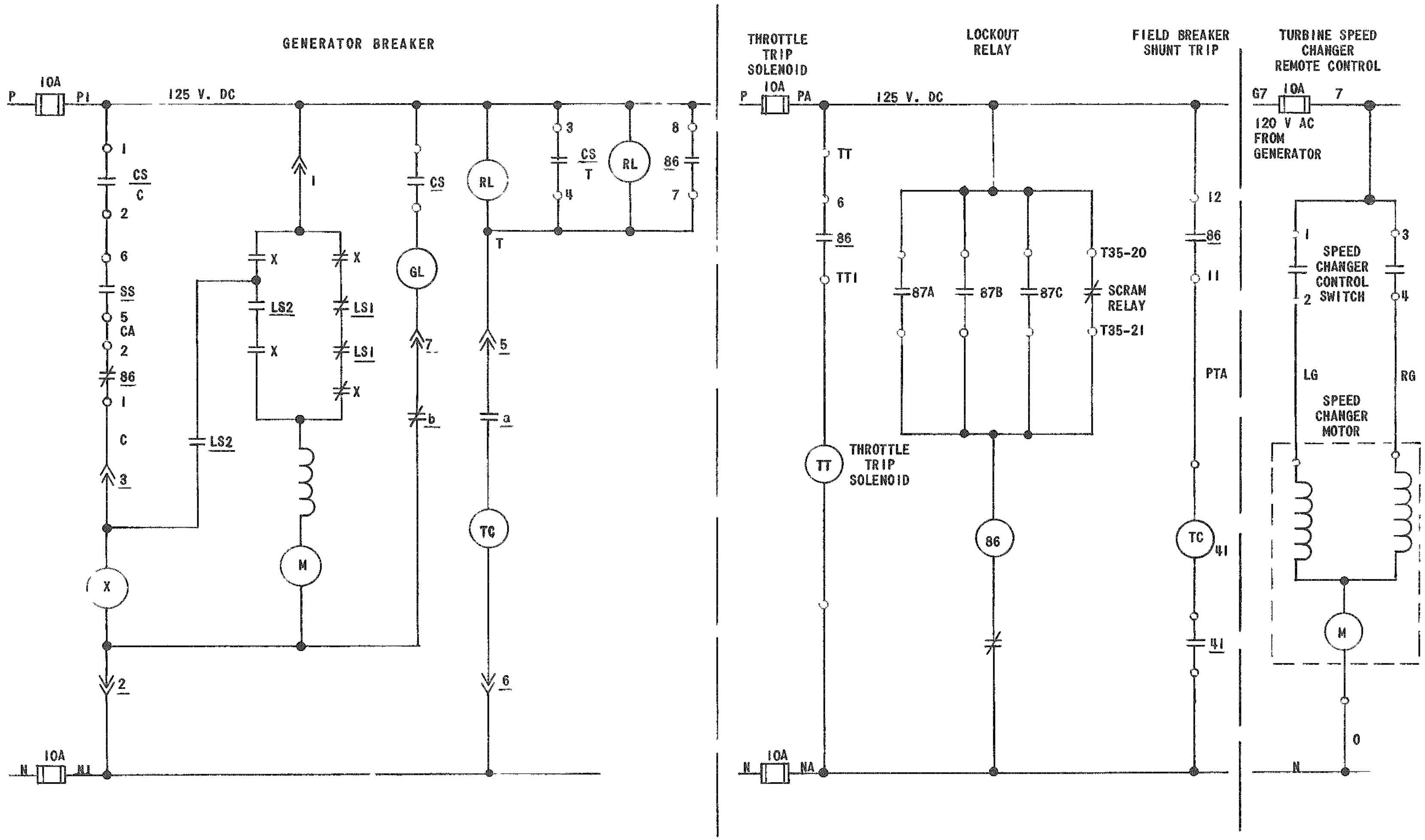


FIG. 70

GENERATOR CONTROL SCHEMATIC WIRING DIAGRAM

The system is normally arranged so that a single generator feeds the equipment bus, to which is connected the sensitive radar and other electronic equipment. The remaining station loads are connected to the utility bus, which is energized by the remaining three generators as required. Thus the electronic equipment is isolated from line disturbances caused by motor starting, etc.

During the design of the prototype plant, consideration was given to various means for affording similar isolation of the buses. The arrangements considered were:

(1) two identical full-capacity turbine-generators, each connected to one of the buses;

(2) a synchronous motor-alternator set, driven from the utility bus, and supplying power to the equipment bus; and

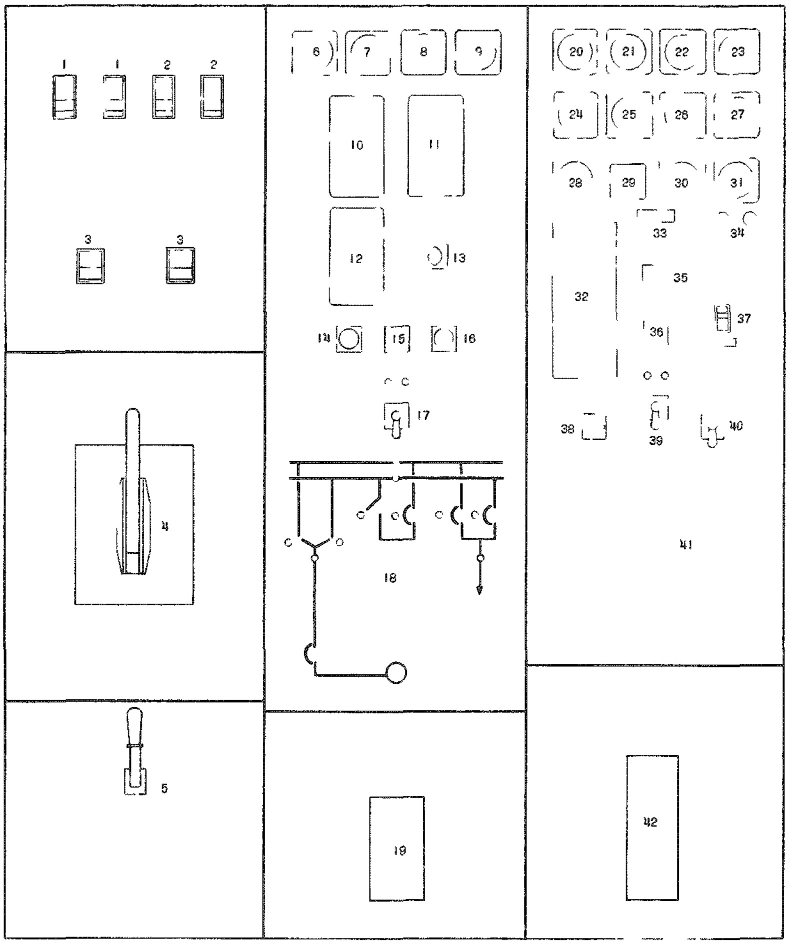
(3) an arrangement consisting of a reactance-compensating-transformer combination for reducing the transfer of transient disturbances on the utility bus to the equipment bus, in combination with a motor-driven induction regulator to compensate for relatively slow voltage changes.

The first scheme was abandoned as being too expensive and requiring too much space on the plant operating floor. Of the other two schemes, the last appeared to be considerably lower in cost and required no continuously rotating machinery. However, an attempt to obtain bids on the compensating transformer equipment from various suppliers was unsuccessful. Since the actual sensitivity of the electronic load to transient line disturbances was not specifically defined, and since the single main turbine-driven generator showed promise of meeting the voltage and frequency-regulation requirements, the decision was made to eliminate any special voltage-regulating equipment from the initial design. Accordingly, the two buses may be tied together through a disconnect switch and manually operated air circuit breaker.

In order to preserve the flexibility of distribution provided at the DEW Line stations, the plant auxiliary loads and the standby diesel-generator may be connected to either station bus. The support facility load, external utility line connections, and the dummy electrical load are peculiar to the prototype installation, and as such are not provided with the same flexibility of switching. The arrangement of the generator control panel and the main distribution panel are shown in Figs. 71 and 72.

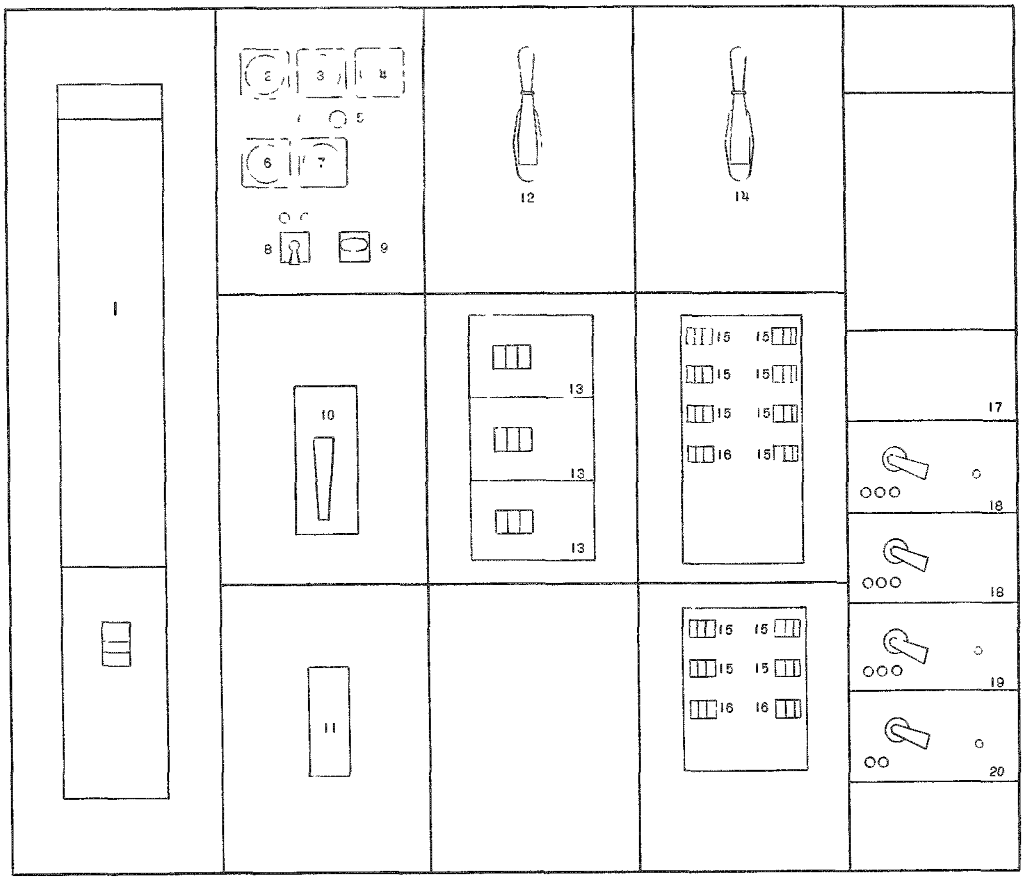
3. Utility Line Connections

With reference to Figs. 68 and 73, the site is served by a 13.8-kv overhead transmission line. The transmission voltage is stepped down to 2400 volts for the site deep well pump, and to 120/208 volts for general distribution, using two 300-kva transformers. The transmission line termination and the transformers are located in an outdoor substation.



1. MOTOR GENERATOR SET SUPPLY BREAKERS
2. NON ESSENTIAL INSTRUMENTS & CONTROL SUPPLY BREAKERS
3. STATION AUXILIARIES SUPPLY BREAKERS
4. GENERATOR TRANSFER SWITCH
5. BUS TIE SWITCH
6. BUS TIE AMMETER
7. BUS TIE WATTMETER
8. BUS TIE VOLTMETER
9. EQUIPMENT BUS VOLTMETER
10. DIFFERENTIAL RELAY
11. DIFFERENTIAL RELAY
12. DIFFERENTIAL RELAY
13. LOCKOUT RELAY
14. BUS TIE AMMETER TRANSFER SWITCH
15. SYNCHROSCOPE SWITCH (BUS TIE BKR.)
16. BUS TIE VOLTMETER TRANSFER SWITCH
17. BUS TIE BREAKER CONTROL SWITCH
18. MIMIC BUS & INDICATOR LIGHTS
19. BUS TIE BREAKER
20. GENERATOR AMMETER
21. GENERATOR WATTMETER
22. GENERATOR BUS VOLTMETER
23. GENERATOR VOLTMETER
24. GENERATOR FREQUENCY METER
25. GENERATOR BUS FREQUENCY METER
26. EXCITER VOLTMETER
27. EXCITER AMMETER
28. GENERATOR SPEED
29. RUNNING TIME METER
30. POWER FACTOR METER
31. SYNCHROSCOPE
32. WATTHOUR METER
33. GENERATOR VOLTAGE CONTROL
34. SYNCHRONIZING LIGHTS
35. GENERATOR VOLTMETER TRANSFER SWITCH
36. SYNCHROSCOPE SWITCH (GEN. BREAKERS)
37. FIELD BREAKER
38. GENERATOR AMMETER TRANSFER SWITCH
39. GENERATOR BREAKER CONTROL SWITCH
40. TURBINE GOVERNOR CONTROL SWITCH
41. EXCITER FIELD RHEOSTAT
42. GENERATOR BREAKER

FIG. 71
GENERATOR CONTROL PANEL



1. LIGHTING AND POWER DISTRIBUTION PANEL
2. UTILITY BUS VOLTMETER
3. SYNCHROSCOPE
4. EQUIPMENT BUS VOLTMETER
5. SYNCHRONIZING LIGHTS
6. IDAHO POWER COMPANY VOLTMETER
7. IDAHO POWER COMPANY WATTMETER
8. TIE CIRCUIT BREAKER CONTROL SWITCH
9. SYNCHROSCOPE SWITCH
10. MAIN CIRCUIT BREAKER
11. TIE CIRCUIT BREAKER
12. DIESEL GENERATOR TRANSFER SWITCH
13. 60 KW SIMULATED LOAD CIRCUIT BREAKER
14. LOAD TRANSFER SWITCH
15. 10 KW SIMULATED LOAD CIRCUIT BREAKER
16. SPARE
17. SPARE
18. SIMULATED HEAT LOAD FAN STARTER
19. SIMULATED HEAT LOAD WATER PUMP STARTER
20. SPARE MOTOR STARTER

FIG. 72
MAIN DISTRIBUTION PANEL

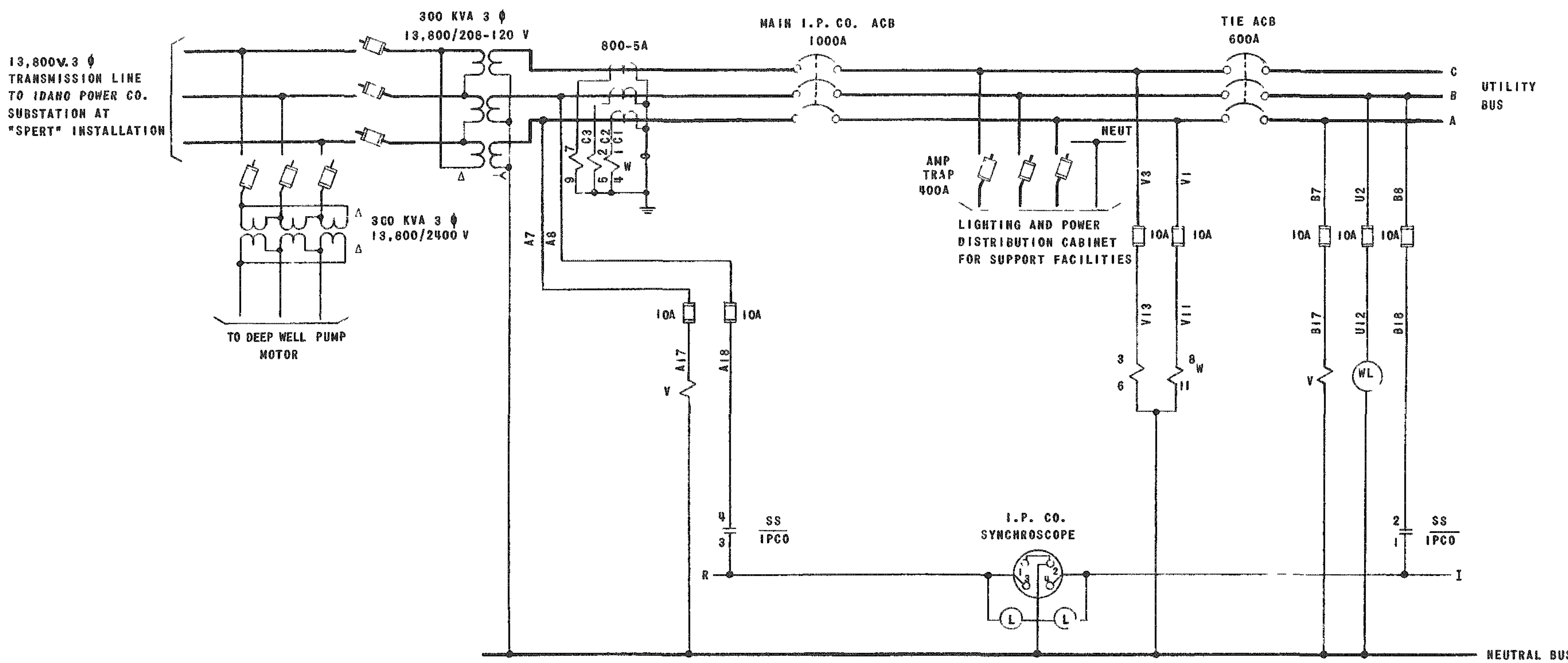


FIG. 73
IDAHO POWER FEED AND TIE CIRCUIT SCHEMATIC DIAGRAM

The 120/208-volt secondary circuit is connected to a manually operated 1000-amp air circuit breaker in the main distribution panel. The support facility building load is normally fed from the load side of this circuit breaker. An electrically operated tie circuit breaker, rated at 600 amp, is also provided so that the utility line may serve reactor plant power requirements when the turbine-generator is not operating. Means are provided for synchronizing and paralleling the plant with the utility system across this circuit breaker.

4. Diesel Generator

The 60-kw diesel-generator was included in the plant primarily for operator training purposes. It is similar to the diesel power units used at the DEW Line stations and is provided with similar connections to the two station buses (see Fig. 74). Tests have shown, moreover, that the reactor plant can be started using only the single 60-kw diesel-generator. This is possible largely because the 75-hp condenser cooling fan operates at greatly reduced load during plant startup and because control power for the associated motor starter operating coil is obtained from the station battery. Engine cranking power is obtained from a 24-volt storage battery which is an integral part of the unit. The diesel engine is arranged to be started under manual control only. Synchronizing lights and a manually operated circuit breaker are included in the unit control panel.

5. Emergency Power Supply

Although reactor shutdown is automatically initiated in the event of failure of reactor plant electric power, it is important that uninterrupted power be available to certain equipment. The affected devices are listed below:

(1) Process Instrumentation

a. Main steam pressure indicator
b. Main steam flow recorder
c. Reactor steam pressure recorder
d. Bypass steam flow recorder
e. Bypass valve drive circuits
f. Condenser vacuum recorder
g. Feedwater flow recorder
h. Feedwater valve drive circuits
i. Reactor water level recorder
j. Multipoint temperature indicator

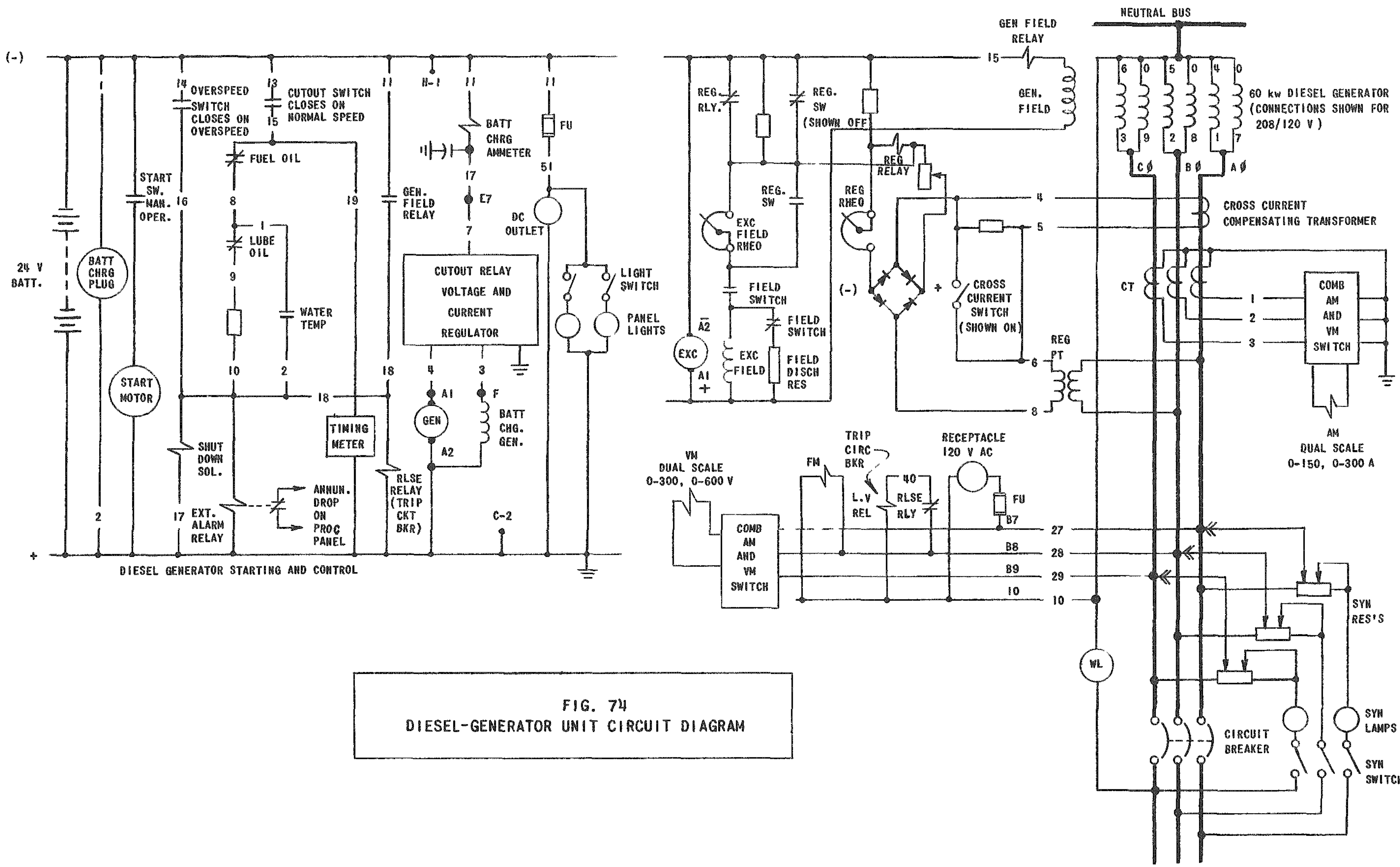


FIG. 74
DIESEL-GENERATOR UNIT CIRCUIT DIAGRAM

(2) Nuclear Instrumentation and Control

a. Channel I high-voltage supply
b. Channel II high-voltage supply
c. Channel II amplifier
d. Control rod drive motors

(3) System Valves (drive closed upon reactor scram)

a. Steam bypass valves
b. Feedwater control valve

With reference to Fig. 75, the continuous (emergency) power supply consists of a three-unit motor-generator set and its associated control panel. The motor-generator set consists, in turn, of a synchronous motor, an alternator, and a dc motor-generator. Under normal conditions the synchronous motor drives the alternator, which supplies power to the critical equipment, and the dc machine, which operates as a generator to maintain charge on the station battery. Upon failure of normal ac supply voltage, the synchronous motor is disconnected from the line, and the alternator is driven by the dc machine without interruption.

Once the unit is started, the necessary switching actions and excitation adjustments automatically follow a failure of normal power. Upon restoration of power for a definite time, the unit reverts automatically to its normal mode of operation.

6. Station Battery

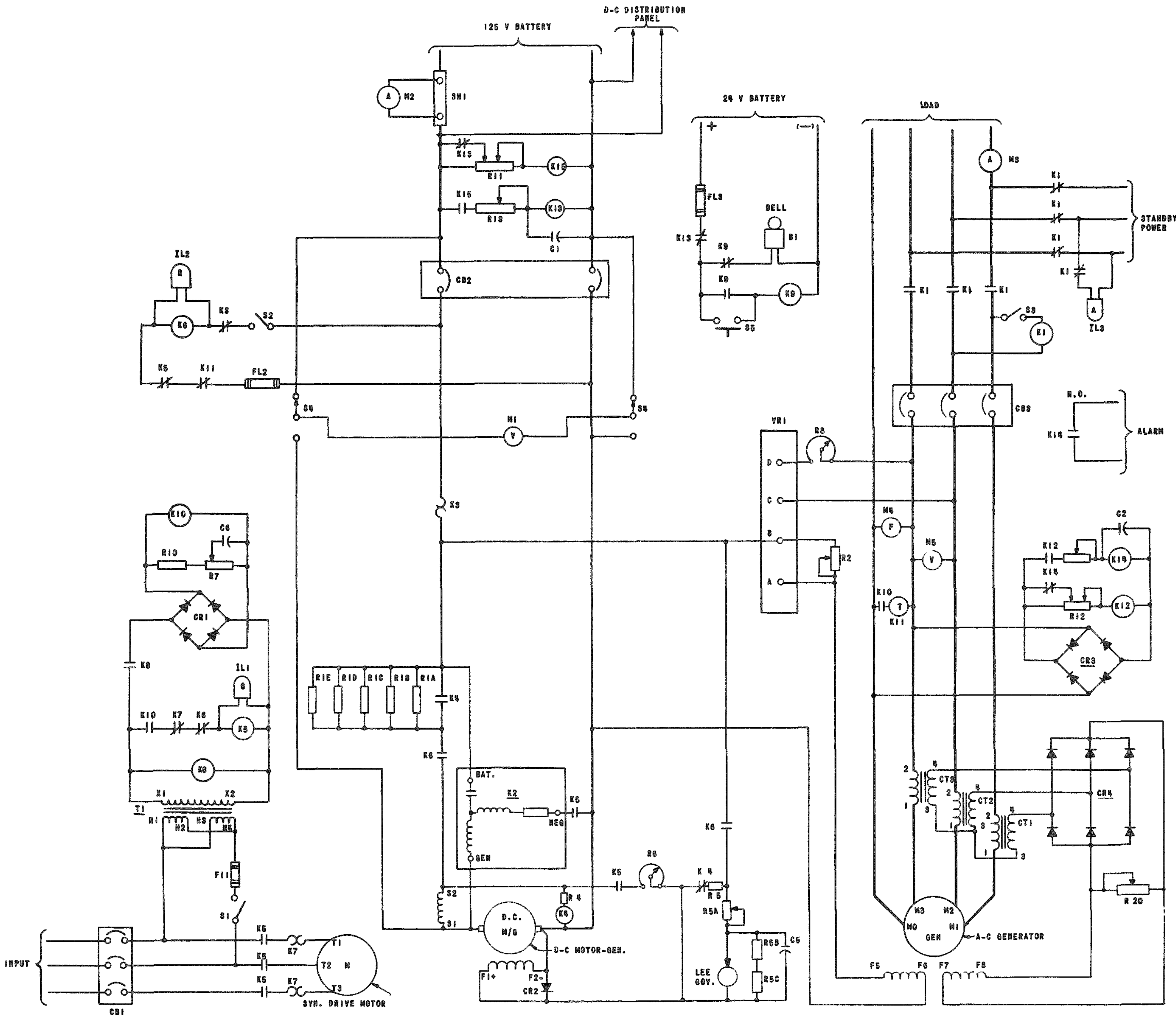
A 92-cell, 82-ampere-hour, 125-volt nickel-cadmium storage battery is used to supply power for the normal operation of the following:

(1) circuit breaker closing and shunt trip coils;
(2) condenser cooling fan motor starter coils;
(3) annunciators;
(4) miscellaneous indicating lights, etc.

Following a failure of normal power, the battery also drives the continuous ac power supply.

7. Dummy Electrical Load

In order to permit loading the plant generator to its full capacity without resorting to operation in parallel with the utility system, a dummy load bank is provided. The load bank is constructed using 10-kw cast iron grid resistor units. It is cooled by air convection and is located outdoors.



LEGEND

M1	DC VOLTMETER
M2	DC AMMETER
M3	AC AMMETER
M4	FREQUENCY METER
M5	AC VOLTMETER
S1	AC CONTROL SWITCH
S2	DC CONTROL SWITCH
S3	MAINTENANCE STANDBY SWITCH
S4	DC VOLTMETER SWITCH
S5	BELL RESET PUSHBUTTON
R6	BATTERY CHARGE RHEOSTAT
R8	AUTO VOLT ADJUST RHEOSTAT
IL1	CHARGE INDICATOR LIGHT
IL2	DISCHARGE INDICATOR LIGHT
IL3	STANDBY POWER INDICATOR LIGHT
K3	DC OVERLOAD RELAY
K7	AC OVERLOAD RELAY
CB1	INPUT BREAKER
CB2	DC BREAKER
CB3	AC OUTPUT BREAKER

FIG. 75
EMERGENCY POWER SUPPLY SCHEMATIC DIAGRAM

This load totals 290 kw, and is subdivided into eleven 10-kw segments and three 60-kw segments (the plant design requirement is 260-kw net). Individual circuit breakers permit switching the load, and provide over-current protection for each segment.

C. Normal System Startup and Operation

Prior to reactor plant startup, the transformer substation and 13.8-kv utility line are assumed to be in operation. Referring to Fig. 68, the Main Idaho Power external utility circuit breaker is assumed to be closed. The Idaho Power tie circuit breaker is then closed. These operations result in energizing the utility bus. If it is desired that the equipment bus also be energized, the bus tie switch and bus tie circuit breaker are closed. Circuit breakers are then closed (normally to the utility bus), energizing the station auxiliary bus, emergency power supply feeder, and instrumentation feeder. Plant equipment is started as needed.[17,18] At any time prior to closing the turbine-generator circuit breaker, the generator bus switch is closed, normally to the utility bus.

When the turbine-generator has been brought to approximately synchronous speed, using the trip-throttle valve, and the governor has assumed control, the following steps are taken:

(1) The governor speed changer is operated from the control room to produce synchronism with the station bus frequency.

(2) Generator voltage is adjusted by means of the voltage regulator control.

(3) At synchronism, the generator circuit breaker is closed.

(4) Using the speed changer control switch, load is transferred to the generator until the Idaho Power wattmeter indicates only the site support facilities load. During this operation the voltage regulator is manipulated to maintain a power factor of approximately 0.8.

(5) The Idaho Power tie circuit breaker is opened.

At this point the generator is supplying power to the plant auxiliaries plus any portions of the dummy load that may be connected. Support facilities power is being obtained from the external utility line. Plant shutdown involves the reverse of these operations.

D. System Operation under Abnormal Conditions

1. Failure of Utility Power during Plant Operation

Upon failure of utility power during plant operation, the support facilities load may be picked up by the plant turbine-generator.

To accomplish this, it is necessary only to open the manually operated main Idaho Power circuit breaker and close the electrically operated Idaho Power tie circuit breaker. If it is desired to transfer the support facilities load to the utility system without interruption upon restoration of utility power, it is necessary to synchronize the plant turbine-generator with the utility system across the manually operated main Idaho Power circuit breaker. Alternatively, the support facilities load may be dropped temporarily by opening the Idaho Power tie circuit breaker before closing the manually operated circuit breaker.

2. Failure of the Turbine-Generator during Plant Operation

Failure of the turbine-generator results in immediate automatic reactor and plant shutdown, except that the critical loads are maintained by the emergency power supply. Simultaneously, the turbine trip-throttle valve and generator circuit breaker are automatically tripped. All motor starters not connected to the emergency power supply drop out. The station buses may then be re-energized by closing the Idaho Power tie circuit breaker. Alternatively, the diesel-generator may be started and connected to whichever station bus is connected to the plant auxiliary loads.

3. Plant Startup in the Absence of Utility Power

As mentioned previously, the diesel-generator is capable of furnishing the power required by the plant auxiliaries during plant startup. To accomplish this, it is necessary that the condenser cooling fan be started first, since the resulting starting current causes a substantial drop in the diesel-generator output voltage. Next, one of the reactor feedwater pumps is started, after which the emergency power supply motor-generator set is started. Following this, the remaining auxiliaries are started as required.

XI. REACTOR AND POWER PLANT INSTRUMENTATION AND CONTROL SYSTEMS

A. Instrumentation*

1. Nuclear

Measurement of instantaneous reactor power is accomplished by means of six channels of neutron-detecting instrumentation, shown diagrammatically in Fig. 76. (In addition to the above six channels, Channel VII, which is employed to monitor air-ejector exhaust activity, is shown.) In Fig. 77 is shown the effective range of each channel. Figure 77 is based on an ionization chamber sensitivity of 4×10^{-14} amp per unit neutron flux, a counter tube sensitivity of 4.5 counts per second per unit neutron flux, and a flux at the detectors of 4×10^9 (n)(cm)/(cm^3)(sec) at a reactor power of 3 Mw.

With reference to Figs. 9 and 13, the neutron detectors are positioned in four vertical stainless steel instrument wells which extend down from the operating floor into the gravel shield adjacent to the reactor pressure vessel. Aluminum cages are employed to support the detectors centrally in the wells at the desired height. Cooling air drawn from the operating floor passes from a manifold up through the wells and out through another manifold, from which it is eventually exhausted outside the building. A drain is provided at the bottom of each well.

The instrument wells are numbered clockwise around the reactor top (see Fig. 31). In addition to the reactor pressure vessel and instrument well walls, the following amounts of metal are interposed between the detectors and the reactor core:

Instrument wells Nos. 1, 2, and 3: $1\frac{3}{8}$ in. steel and $5\frac{1}{4}$ in. lead.

Instrument well No. 4: $2\frac{3}{8}$ in. steel and $1\frac{1}{4}$ in. lead.

The extra lead for gamma filtering was omitted from in front of instrument well No. 4 to permit comparisons in instrument response.

a. Channel I

An uncompensated, boron-lined ionization chamber is connected so that its output current drives a panel-type microammeter and a sensitive moving coil relay in series. This channel provides an indication of reactor power from approximately 1% to 150% of full power. In addition, it serves to trip the reactor shutdown circuits on abnormally high flux without requiring an amplifier.

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*H. H. Hooker

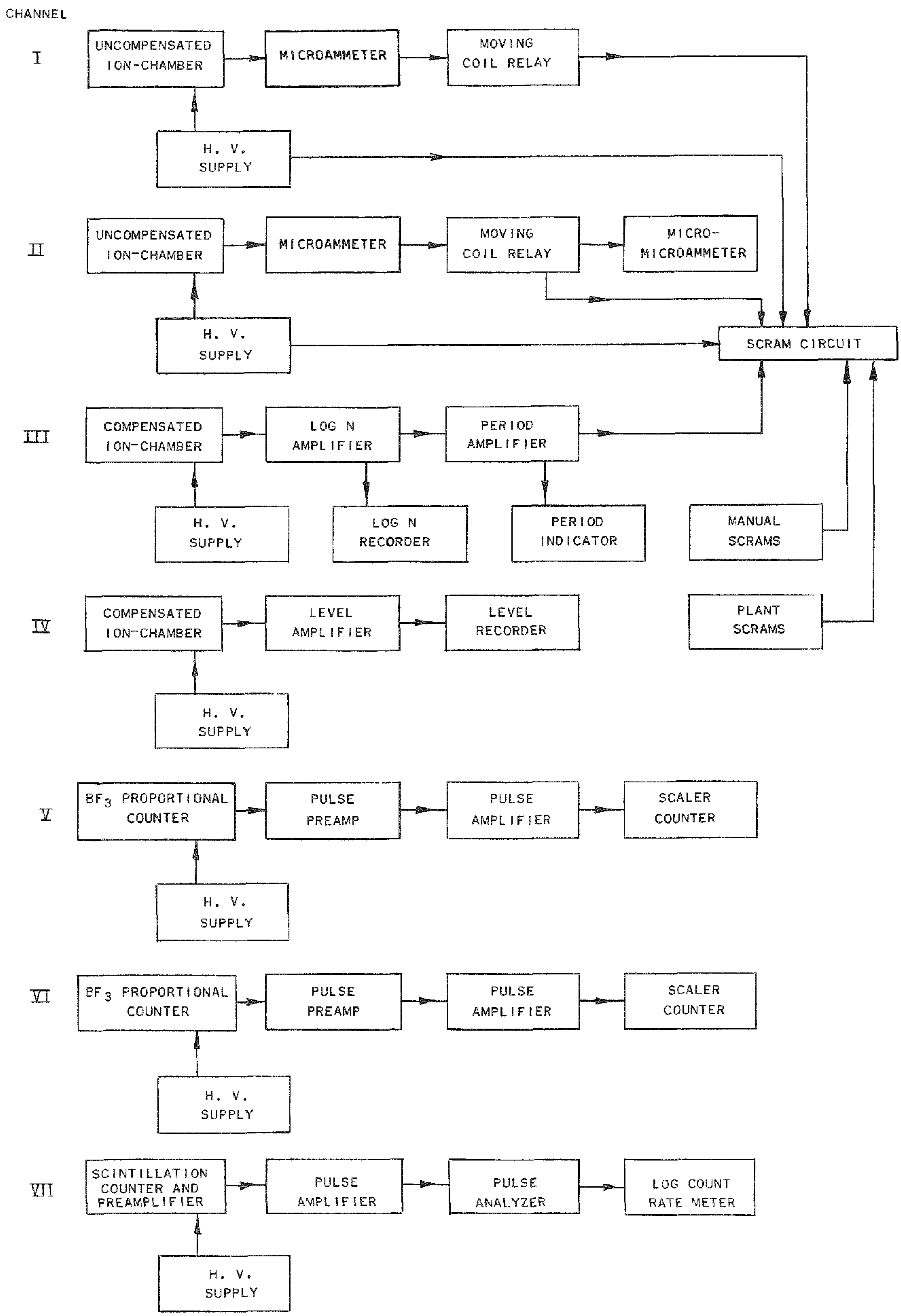


FIG. 76
NUCLEAR INSTRUMENT BLOCK DIAGRAM

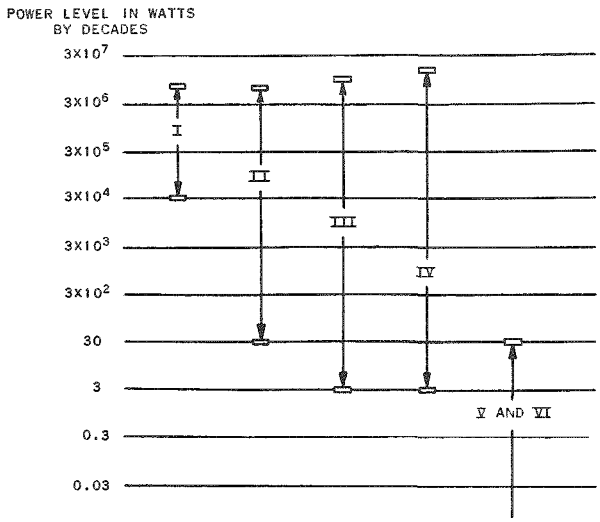


FIG. 77
POWER LEVELS CORRESPONDING TO OPERATING RANGES OF INSTRUMENTATION CHANNELS I TO VI

Referring to Fig. 78, it will be seen that increasing ion chamber current to the trip setting of relay K1 causes this relay to close its contacts, which short circuit the coil of auxiliary relay K2. A pair of contracts on K2, in turn, initiates reactor shutdown. Adjustable shunts are provided for adjustment of the microammeter full-scale sensitivity and the sensitive relay-tripping current. In practice, an ionization chamber current of approximately 1.6 x 10^{-4} amp is observed with a reactor power of 3 Mw. Failure of the chamber high-voltage supply, or an uncoupled connector in the high-voltage or signal coaxial cables, also causes relay K2 to become de-energized, resulting in reactor shutdown. A trip test circuit is incorporated in the indicator chassis; by

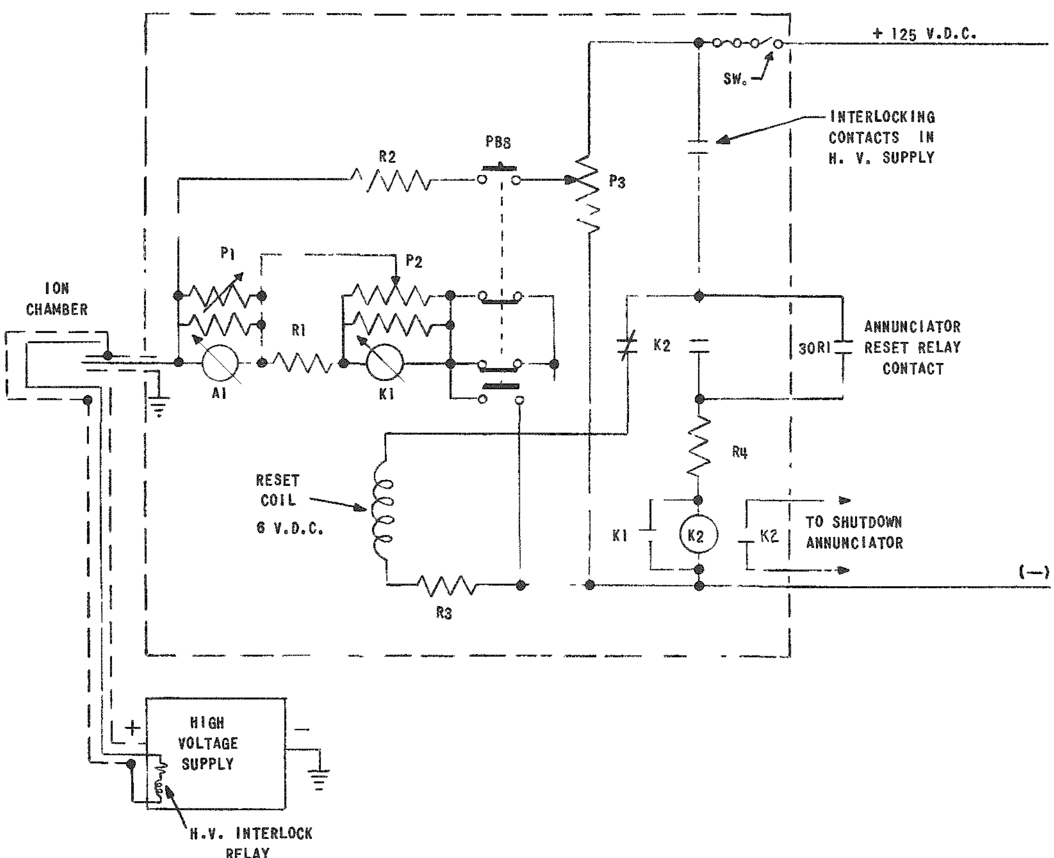


FIG. 78
CHANNEL I
SCRAM/INDICATOR SYSTEM

means of this circuit a preset test current may be passed through the microammeter and sensitive relay, enabling testing of the entire circuit. With the exception of the ion chamber, all components are located in the Nuclear Panel (see Fig. 79).

b. Channel II

In addition to a duplication of the components comprising Channel I, Channel II includes a chopper-type micromicroammeter having a maximum full-scale sensitivity of 1×10^{-9} amp, and a minimum full-scale sensitivity of 5×10^{-4} amp. This channel is useful from about five decades below full power to 150% full power. This channel duplicates the high-flux trip functions of Channel I. Either channel may trip the reactor shutdown circuits independently of the other.

c. Channel III

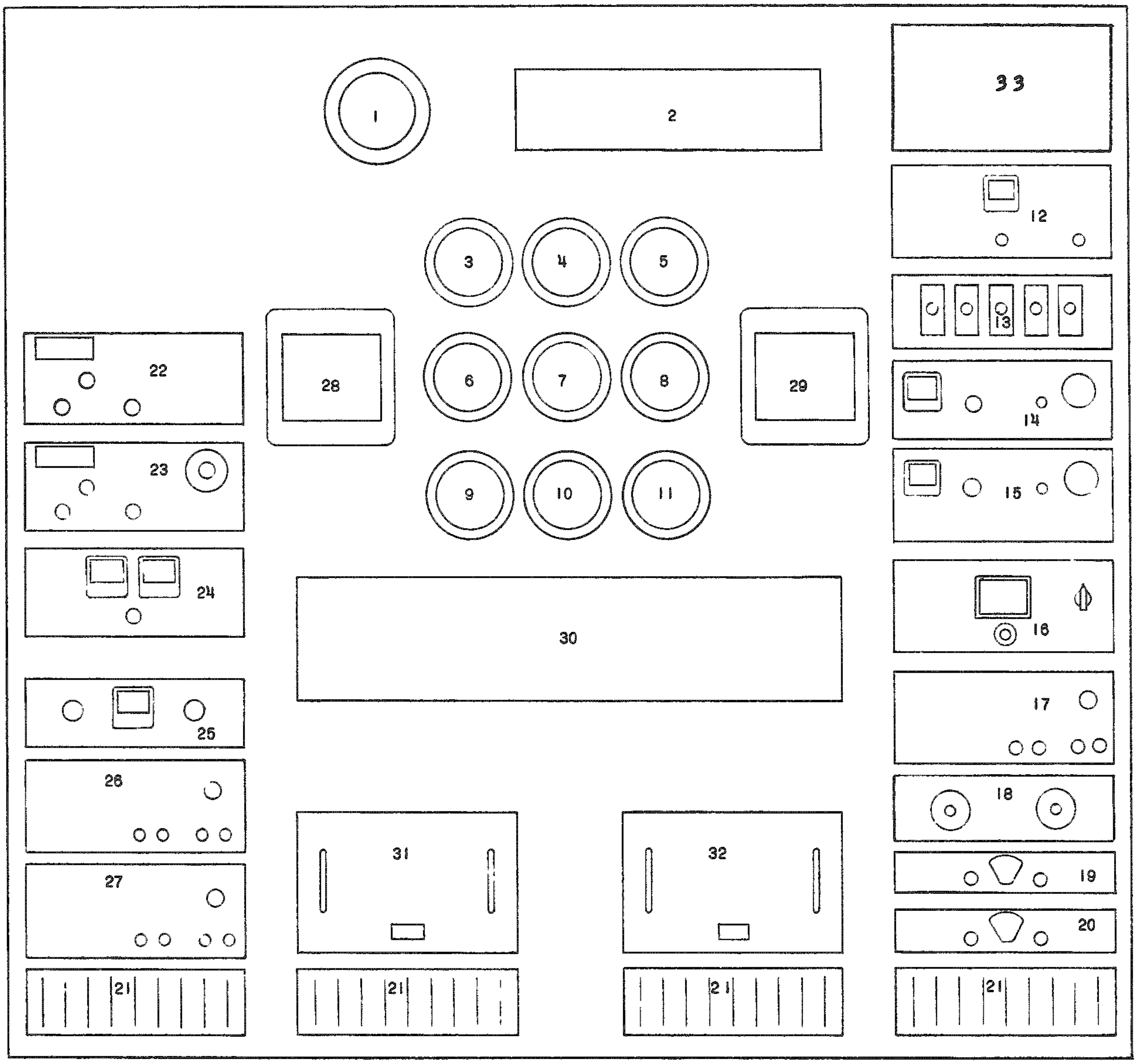
This channel consists of a gamma-compensated ionization chamber with its associated power supplies, log and period amplifiers, and a log power recorder. The log amplifier is calibrated with the log power indicator and recorder for a range from 0.0001% to 300% nominal full power. The period circuit indicates reactor period in the range of from -30 to infinity to +3 sec, and trips the reactor shutdown circuits in the event of a period shorter than preset amount. The period circuit is most useful during reactor startup when a positive period can be maintained in the subcooled reactor. A key-operated switch (see Fig. 80) is provided so that the period trip circuit may be bypassed when the reactor is boiling.

d. Channel IV

In this channel, a sensitive, direct-coupled amplifier, having a full-scale sensitivity from 3×10^{-13} to 1×10^{-3} amp, is used to measure the output current of a gamma-compensated ionization chamber. The ion chamber current is indicated on a panel meter and recorded on the linear power recorder. This channel provides information on reactor power at operating power and below over a range of approximately seven decades.

e. Channels V and VI

Channels V and VI are essentially identical, and each consists of a BF_3 proportional counter and associated high-voltage supply, a pulse preamplifier, a linear pulse amplifier, and a scaler-counter. The preamplifiers are located in a panel in the reactor building in order to reduce the length of high-impedance signal circuit. The high-voltage supplies are also located in this panel. The remaining linear amplifiers and scaler-counters are installed in the Nuclear Panel in the control room.

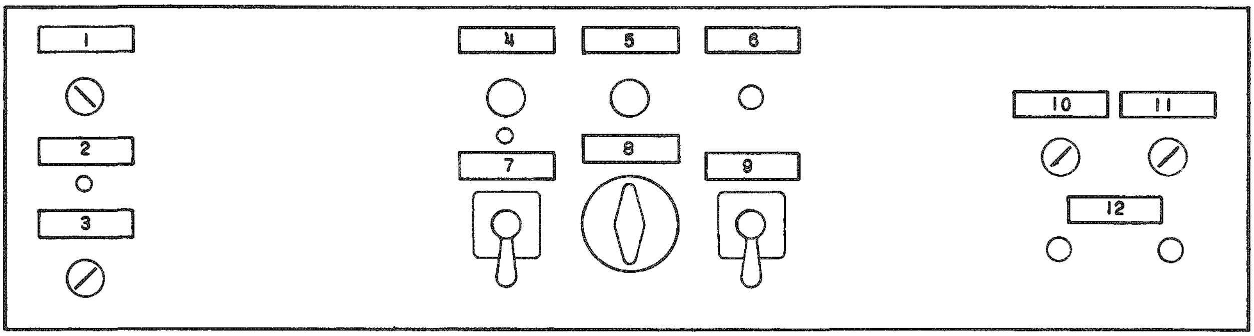


1. 24 HOUR SYNCHRONOUS CLOCK
2. NUCLEAR PANEL ANNUNCIATOR
3. ROD 4 POSITION INDICATOR
4. ROD 5 POSITION INDICATOR
5. ROD 6 POSITION INDICATOR
6. ROD 3 POSITION INDICATOR
7. ROD 9 POSITION INDICATOR
8. ROD 7 POSITION INDICATOR
9. ROD 2 POSITION INDICATOR
10. ROD 1 POSITION INDICATOR
11. ROD 8 POSITION INDICATOR
12. CHANNEL VII LOG COUNT RATE AMPLIFIER PANEL
13. CHANNEL VIII REMOTE AREA MONITOR PANEL
14. CHANNEL II SCRAM-INDICATOR SYSTEM PANEL
15. CHANNEL I SCRAM-INDICATOR SYSTEM PANEL
16. CHANNEL II D.C. AMPLIFIER PANEL
17. CHANNEL VII LINEAR PULSE AMPLIFIER PANEL
18. CHANNEL VII PULSE HEIGHT ANALYSER PANEL
19. CHANNEL I HIGH VOLTAGE POWER SUPPLY PANEL
20. CHANNEL II HIGH VOLTAGE POWER SUPPLY PANEL
21. COOLING FAN
22. CHANNEL V SCALER DECADE PANEL
23. CHANNEL VI SCALER DECADE PANEL
24. CHANNEL III LOG N-PERIOD AMPLIFIER PANEL
25. CHANNER IV ELECTROMETER PANEL
26. CHANNEL V LINEAR PULSE AMPLIFIER PANEL
27. CHANNEL VI LINEAR PULSE AMPLIFIER PANEL
28. LOG FLUX LEVEL RECORDER
29. LINEAR FLUX LEVEL RECORDER
30. REACTOR CONTROL PANEL
31. ELECTRIC VOLTAGE REGULATOR PANEL
32. ELECTRIC VOLTAGE REGULATOR PANEL
33. POWER DEMAND CONTROL SYSTEM

FIG. 79

NUCLEAR INSTRUMENTATION PANEL

A switch is provided in the reactor control panel so that the high-voltage supplies may be de-energized when the reactor is operating at a power above the counting range. This prolongs the useful life of the counter tubes.



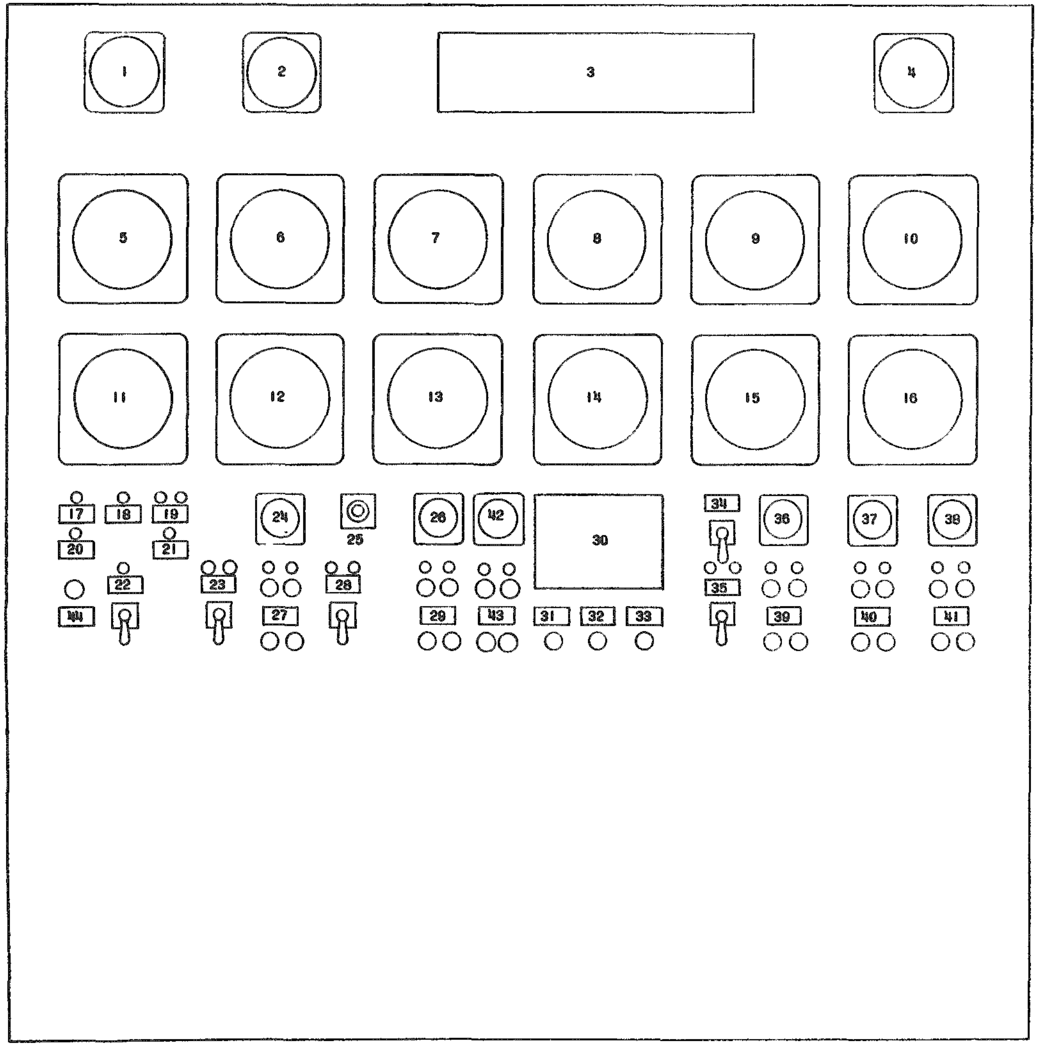
1. COUNTER HIGH VOLTAGE CONTROL SWITCH
2. CONTROL POWER "ON" INDICATING LIGHT
3. CONTROL POWER KEY SWITCH
4. CENTER ROD DROP PUSH-BUTTON
5. REACTOR SHUTDOWN PUSH-BUTTON
6. PERIPHERAL ROD DROP PUSH-BUTTON
7. CENTER ROD CONTROL SWITCH
8. ROD DRIVE SELECTOR SWITCH
9. PERIPHERAL ROD CONTROL SWITCH
10. PERIOD SHUTDOWN BYPASS KEY SWITCH
11. LOW PRESSURE SHUTDOWN BYPASS KEY SWITCH
12. ANNUNCIATOR TEST AND RESET PUSH BUTTONS

FIG. 80
REACTOR CONTROL PANEL

2. Plant Parameters

Instrumentation of plant parameters consists of indicators mounted locally near the various points of measurement, and indicators or indicator-recorders, located in the control room, which are actuated by remote transmitters. Except for a small panel containing the resistance-thermometer indicator and related selector switch, all control room indication and recording of plant parameters is concentrated on the Process Panel (see Fig. 81). An annunciator is provided on this panel for indicating various abnormal conditions (see Fig. 82). Wherever practicable, the annunicator obtains its operating signals directly from contacts mounted in the local indicators.

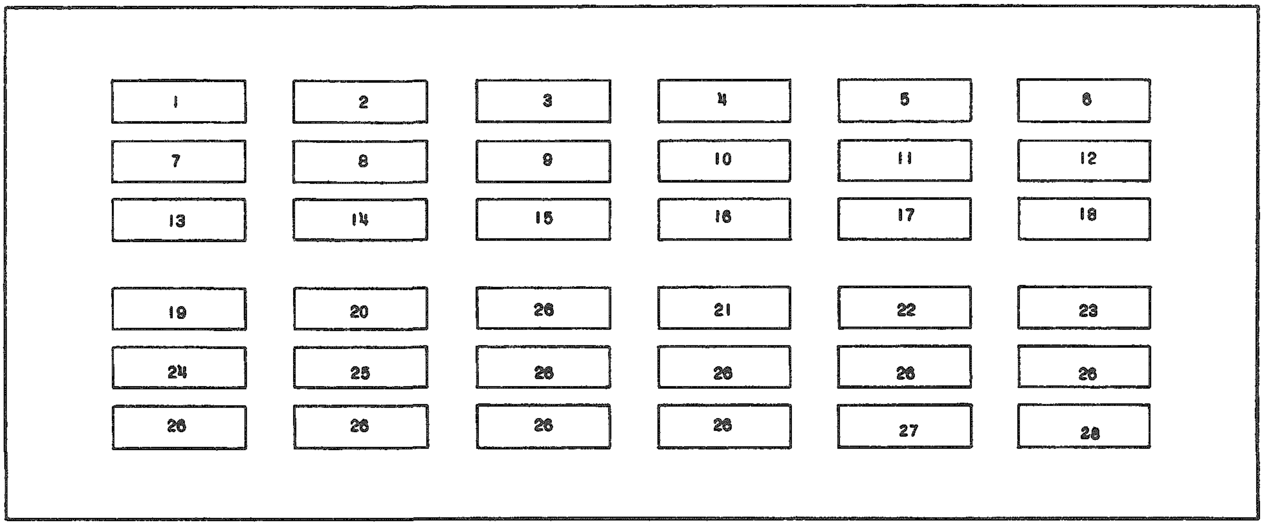
The considerations governing the instrumentation design were as follows: (1) the system should be as simple and reliable as possible, consistent with the commercial availability of proven components; and (2) transmission of signals between the various components must be accomplished electrically.



1. FEED-WATER PRESSURE INDICATOR
2. MAIN STEAM PRESSURE INDICATOR
3. PROCESS PANEL ANNUNCIATOR
4. HOTWELL WATER LEVEL INDICATOR
5. FEED-WATER FLOW RECORDER
6. REACTOR STEAM PRESSURE RECORDER
7. MAIN STEAM FLOW RECORDER
8. BYPASS STEAM FLOW RECORDER
9. CONDENSER VACUUM RECORDER
10. CONDENSER AIR OUTLET TEMPERATURE RECORDER
11. FEED-WATER TEMPERATURE RECORDER
12. REACTOR WATER LEVEL RECORDER
13. MAIN STEAM PRESSURE DEVIATION INDICATOR
14. SYSTEM TEMPERATURE INDICATOR
15. RESISTIVITY INDICATOR
16. CONDENSER AIR INLET TEMPERATURE RECORDER
17. HIGH PRESSURE CONDENSATE RETURN TANK PUMP INDICATOR LIGHT
18. LOW PRESSURE CONDENSATE RETURN TANK PUMP INDICATOR LIGHT
19. CONDENSATE CIRCULATION PUMP INDICATOR LIGHTS
20. SHIELD COOLER PUMP INDICATOR LIGHT
21. INSTRUMENT WELL EXHAUST FAN INDICATOR LIGHT
22. WATER PURIFICATION PUMP CONTROL SWITCH & INDICATOR LIGHT
23. FEED-WATER PUMP #1 CONTROL SWITCH & INDICATOR LIGHTS
24. FEED-WATER VALVE POSITION INDICATOR
25. SET POINT SETTER (FOR ITEM 13 RANGE)
26. BYPASS STEAM REGULATOR VALVE POSITION INDICATOR (0-3000 ppH)
27. FEED-WATER VALVE TRANSFER & CONTROL PUSH BUTTONS & INDICATOR LIGHTS
28. FEED-WATER PUMP #2 CONTROL
29. BYPASS STEAM VALVE CONTROL & TRANSFER PUSH BUTTONS & INDICATOR LIGHTS (0-3000 ppn)
30. PROCESS PANEL TEMPERATURE SELEÇTOR
31. VALVE CONTROL SHUTDOWN RESET
32. ANNUNCIATOR RESET PUSH BUTTON
33. ANNUNCIATOR TEST PUSH BUTTON
34. RESISTIVITY INDICATOR SELECTOR
35. FAN MOTOR OFF-ON CONTROL SWITCH & INDICATOR LIGHTS
36. CONDENSER AIR CIRCULATING FAN AMMETER
37. MIXER DAMPER POSITION INDICATOR
38. AIR EXHAUST DAMPER POSITION INDICATOR
39. CONDENSER AIR CIRCULATING FAN CONTROL PUSH BUTTONS & INDICATOR LIGHTS
40. MIXER DAMPER TRANSFER AND CONTROL PUSH BUTTONS & INDICATOR LIGHTS
41. EXHAUST DAMPER DRIVE TRANSFER & CONTROL PUSH BUTTONS & INDICATOR LIGHTS
42. BYPASS STEAM REGULATOR VALVE POSITION INDICATOR (0-10,000 pph)
43. BYPASS STEAM VALVE CONTROL & TRANSFER PUSH BUTTONS & INDICATOR LIGHTS (0-10,000 pph)
44. SHIELD COOLER SOLENOID VALVE OPEN PUSH BUTTON

FIG. 81

PROCESS INSTRUMENTATION PANEL



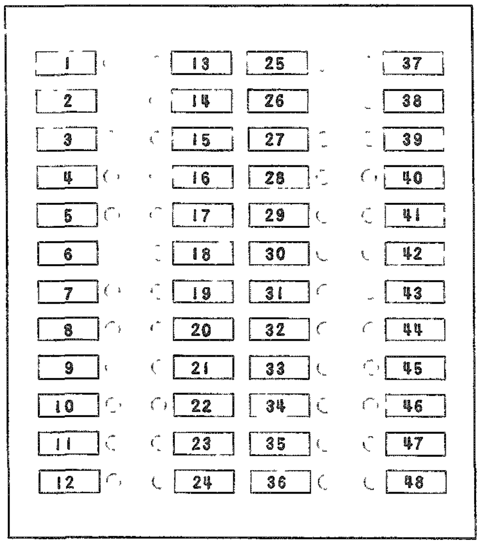
1. CONDENSER VACUUM - LOW
2. BYPASS STEAM FLOW - HIGH
3. BYPASS VALVE DISCHARGE PRESSURE - HIGH
4. REACTOR PRESSURE - LOW
5. CONDENSER OUTLET AIR TEMPERATURE - HIGH
6. CONDENSER INLET AIR TEMPERATURE - LOW
7. FEED PUMP DISCHARGE PRESSURE - LOW
8. PURIFICATION WATER TEMPERATURE - HIGH
9. REACTOR WATER LEVEL - LOW
10. AIR EJECTOR AFTER CONDENSER TEMPERATURE - HIGH
11. SHIELD COOLER OUTLET TEMPERATURE - HIGH
12. HOTWELL WATER LEVEL - HIGH OR LOW
13. REACTOR WATER LEVEL - HIGH
14. FEED PUMP AUTOMATIC SWITCHOVER
15. MAIN STEAM PRESSURE RELIEF VALVE FLOW
16. REACTOR PRESSURE - HIGH
17. HIGH PRESSURE CONDENSATE TANK LEVEL - HIGH
18. LOW PRESSURE CONDENSATE TANK LEVEL - HIGH
19. TURBINE OIL INLET TEMPERATURE - HIGH
20. MAIN STEAM SAFETY VALVE FLOW
21. SHIELD COOLER PUMP DISCHARGE - LOW
22. DIESEL GENERATOR TROUBLE
23. SHIELD COOLER BYPASS VALVE OPEN
24. TURBINE OIL PRESSURE - LOW
25. EMERGENCY POWER TROUBLE
26. SPARE
27. REACTOR WATER LEVEL (SECONDARY)-LOW
28. REACTOR WATER LEVEL (SECONDARY)-HIGH

FIG. 82
PROCESS PANEL ANNUNCIATOR

Instrumentation of the prototype plant is intended to provide all information necessary for an evaluation of the performance of the plant and its suitability for the intended application. Thus it is to be expected that future plants of this type will employ less instrumentation, and that indicating instruments will be used in many cases where indicator-recorders are presently employed in the prototype.

a. Local Temperature Indicators

The local temperature indicators (see Fig. 83) are of three general types: (a) mercury-actuated dial thermometers; (b) liquid-actuated temperature indicators, other than those using mercury; and (c) bimetal-actuated dial thermometers. Stainless steel separable sockets are used wherever the instrument bulbs are inserted in the piping system. Certain of these indicators are equipped with control and/or alarm contacts for actuation of the process annunciator or reactor shutdown circuits.



1. WATER PURIFICATION (A)
2. AIR PRE-COOLER VAPOR OUTLET (B)
3. WATER FROM THERMAL SHIELD COOLING COIL (C)
4. REACTOR WATER (D)
5. AIR EJECTOR AFTER COOLER VAPOR OUTLET (E)
6. OPERATING FLOOR (F)
7. OPERATING FLOOR (G)
8. GRAVEL (W)
9. GRAVEL (I)
10. GRAVEL (J)
11. GRAVEL (K)
12. SUB-REACTOR SHIELD (L)
13. SUB-REACTOR SHIELD (M)
14. REACTOR SHIELD (CIRCUIT #1)(H)
15. REACTOR SHIELD (CIRCUIT #2)(O)
16. SPARE
17. FUEL STORAGE WELL #1, UPPER (V)
18. FUEL STORAGE WELL #1, LOWER (U)
19. FUEL STORAGE WELL #2, UPPER (T)
20. FUEL STORAGE WELL #2, LOWER (S)
21. FUEL STORAGE WELL #3, UPPER (R)
22. FUEL STORAGE WELL #3, LOWER (O)
23. CONCRETE SHIELD (N)
24. CONCRETE SHIELD (X)
25. INSTRUMENT WELL #1 (P)
26. INSTRUMENT WELL #2 (Y)
27. INSTRUMENT WELL #3 (Z)
28. INSTRUMENT WELL #4 (AA)
29. CONTROL ROD #1 (BB)
30. CONTROL ROD #2 (CC)
31. CONTROL ROD #3 (DD)
32. CONTROL ROD #4 (EE)
33. CONTROL ROD #5 (FF)
34. CONTROL ROD #6 (GG)
35. CONTROL ROD #7 (HH)
36. CONTROL ROD #8 (II)
37. CONTROL ROD #9 (JJ)
38. TO 48. SPARES (KK-TT)

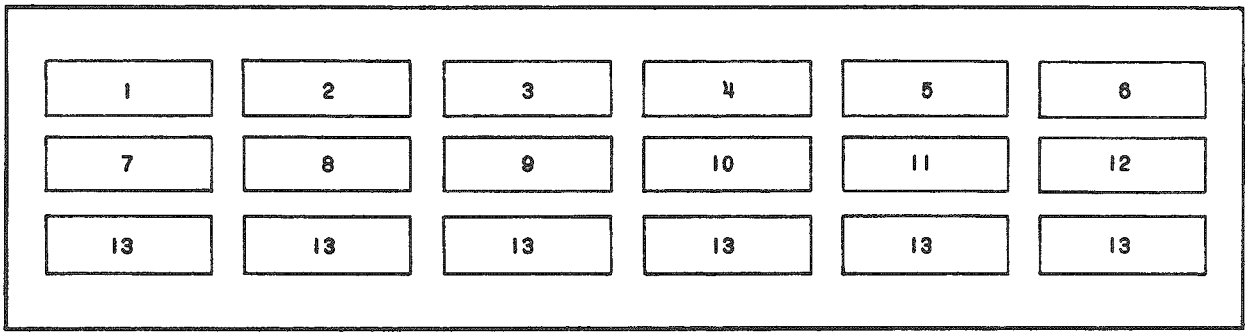
NOTE: LETTER ( ) IDENTIFIES THE INSTRUMENT TI-16 ( )

FIG. 83
PROCESS PANEL TEMPERATURE SELECTOR

The temperatures causing actuation of these circuits are listed in Figs. 82 and 84. The local temperature indicators are listed in Table 20.

b. Control Room Temperature Indicators and Recorders

Iron-constantan thermocouples are used as temperature detectors for the temperature instruments located on the Process Panel. All thermocouple extension wire running from the thermocouples to the instrument terminals is iron-constantan, and the instruments incorporate automatic cold-junction compensation. Two groups of 12 thermocouples each are used for measurement of condenser inlet and outlet average air temperature. The thermocouples in each group are connected in parallel through swamping resistors to eliminate the effect of unequal individual resistances.



1. CHANNEL I FLUX - HIGH
2. CHANNEL II FLUX - HIGH
3. CHANNEL III PERIOD - SHORT
4. REACTOR WATER LEVEL - LOW
5. REACTOR WATER LEVEL - HIGH
6. REACTOR STEAM PRESSURE - LOW
7. REACTOR STEAM PRESSURE - HIGH
8. MAIN STEAM PRESSURE RELIEF VALVE FLOW
9. MAIN STEAM SAFETY VALVE FLOW
10. CONDENSER PRESSURE - HIGH
11. THERMAL SHIELD CIRCUIT NO. 1 TEMPERATURE - HIGH
12. THERMAL SHIELD CIRCUIT NO. 2 TEMPERATURE - HIGH
13. SPARE

FIG. 84
NUCLEAR PANEL ANNUNCIATOR (REACTOR SHUTDOWN)

Table 20

LOCALLY MOUNTED TEMPERATURE INDICATORS

Instrument No.	Measurement	Reference Flow Diagram Figure No.
127	Main Steam Relief Valve Discharge	17
128	Air-ejector After-condenser	18
129	Turbine Oil Cooler Oil Outlet	19
131	Main Steam Safety Valve Discharge	17
220	Purification System Water	40
221	Retention Tank	40
223	Condensate to Precooler	18
224	Condensate from Precooler	18
225	Turbine Gland Seal Leakage After Precooler	18
226	Shield Cooler Outlet, Circuit No. 1	19
227	Shield Cooler Outlet, Circuit No. 2	19
229	Shield Cooler Inlet	19
230	Water to Space-heating System	17
231	Water from Space-heating System	17
232	Condensate from Hotwell	18
233	Steam End Turbine Bearing	17
234	Exhaust End Turbine Bearing	17
235	Inboard Generator Bearing	17
236	Outboard Generator Bearing	17
237	Turbine Gear Drain Sump	17
241	Thermal Shield, Curcuit No. 1	19
242	Thermal Shield, Circuit No. 2	19
249	Turbine Oil Tank	19
250	Turbine Oil Cooler Outlet	19

Resistance temperature detectors are used with an indicator and selector switch to measure condenser inlet air temperature at five specific points. This instrument was installed as an aid to initial balancing of the air flow through the condenser. The control room temperature instruments are listed in Table 21.

Table 21

CONTROL ROOM TEMPERATURE INDICATORS AND RECORDERS

Instrument No.	Measurement	Fig. 81 Detail No.	Reference Flow Diagram Figure No.
7	Feedwater (Recorder, 0-400°F)	11	18
9	Condenser Inlet Air (Recorder, 0-300°F)	16	18
11	Condenser Outlet Air (Recorder, 0-300°F)	10	17
16	System Temperatures (48-point Indicator, 0-500°F, see Fig. 83)	14,30	18,19,21
137	Condenser Inlet Air (Indicator, -60 to +60°F)	-	18

c. Local Pressure Indicators and Switches

The local pressure indicators are standard bourdon tube powered gauges, certain of which are equipped with pointer-actuated contacts for control and alarm functions. However, the pressure-actuated reactor-shutdown circuits utilize bourdon tube powered mercury switches. These were selected because of their greater mechanical ruggedness and immunity to vibration. The local pressure indicators and switches are listed in Table 22.

d. Process Panel Pressure Indicators and Recorders

The pressure (and vacuum) transmitters which furnish signals to the process panel indicators and recorders are of the force balance type. They consist of a bourdon tube pressure-sensitive element which is mechanically coupled to a balanced beam. The beam in turn is electromagnetically coupled to an electronic oscillator which comprises one arm of a Wheatstone bridge. Deflection of the bourdon tube with pressure deflects the beam and detunes the oscillator. The output current due to the resulting bridge unbalance varies uniformly with pressure over a range from 0.5 to 5.0 ma. This current rebalances the beam in a slightly different position and serves as the input signal for the associated indicator or recorder. This type of transmitter was selected because of its compatibility with both types of indicator and recorder (see below), and because the dc output signal is much less subject to interference caused by electromagnetic coupling with nearby power cables.

In order to expand the scale calibration in the normal operating range and to increase the control sensitivity, the Pressure Deviation Indicator (Instrument No. 5, Table 23) utilizes a modified input circuit.

Table 22

LOCALLY MOUNTED PRESSURE INDICATORS AND SWITCHES

Instrument No.	Measurement	Reference Flow Diagram Figure No.
123	Main Steam Bypass Valve Discharge	17
125	Feedwater Pump No. 1 Discharge	18
126	Feedwater Pump No. 2 Discharge	18
132	Condenser (Pressure Switch)	17
133	Reactor Steam	17
136	Air Filter Differential (Gravel Vent)	21
140	Reactor Steam (Pressure Switch)	17
201	Purification System Water	40
202	Retention Tank Pump Discharge	40
203	Main Steam	17
205	Steam to Condenser Air Ejector	17
206	Steam to Gland Air Ejector	17
209	Hot Water Heater Shell	17
210	Turbine Steam Inlet	17
211	Turbine First Stage	17
212	Turbine Exhaust	17
213	Turbine Bearing Oil	19
214	Turbine Steam End Gland	17
215	Turbine Exhaust End Gland	17
243	Shield Cooler Pump Discharge	19
247	Purification System Pump Discharge (Pressure Switch)	40

Table 23

PROCESS PANEL PRESSURE INDICATORS AND RECORDERS

Instrument No.	Transmitter No.	Measurement	Fig. 81 Detail No.	Reference Flow Diagram Figure No.
1	101	Feedwater (Indicator, 0-400 psig)	1	18
3	102	Main Steam (Indicator, 0-400 psig)	2	17
4	103	Condenser (Recorder, 30 in. Hg Vac. to 0 to 30 psig)	9	17
5	122	Main Steam ($P-P_0$, -20 to +20 psi)	13	17
12	106	Reactor Steam	6	17

The associated transmitter (Transmitter No. 122, Table 23) has a suppressed-zero operating range from 230 to 350 psig. The indicator has a full-scale range of 40 psi, calibrated from -20 to +20 psi. A manually operated zero-setting potentiometer, calibrated from 270 to 310 psig, is provided on the Process Panel for setting the pressure at which the indicator reads zero. This is accomplished by adding a constant (but adjustable) voltage to the transmitter output. Control slidewires in this instrument furnish control signals for main steam bypass valve and control rod drive automatic control circuits.

The Process Panel pressure indicators and recorders are either suitably calibrated direct-reading milliammeters or self-balancing potentiometer-type millivoltmeters. These instruments are listed in Table 23.

e. Locally Mounted Level Indicators and Switches

The level-measuring devices are of several different types.

Standard sight glass gauges are provided for measuring water level in the:

(1) hot well;
(2) water-storage tank;
(3) space-heating surge tank; and
(4) shield cooler surge tank.

Retention tank level is measured by using a hand-pumped bubbler-type instrument.

The condensate return tank level switches, used for pump control and level alarms, are two-stage, float-operated mercury switches.

The reactor water level alarm switch device consists of a displacer-torque-tube assembly, the output shaft of which is coupled to two adjustable alarm switches. The device initially considered for this application involved sampling the fluid in the reactor at three different levels by discharging it through an orifice to the condenser. A pressure gauge connected to the high-pressure side of the orifice would thus permit identification of the fluid being discharged as water, mixture, or steam. At best, this would permit only a determination of water level between limits. In addition, such a device requires manual operation each time a reading is desired.

The present instrument, although it automatically furnishes an alarm when water level is outside the preset limits, has proven to be difficult to adjust accurately, and has been found to exhibit considerable hysteresis in switch actuation.

The local level instruments and switches are listed in Table 24.

Table 24

LOCALLY MOUNTED LEVEL INDICATORS AND SWITCHES

Instrument No.	Measurement	Reference Flow Diagram Figure No.
110	Reactor Water	18
118	Retention Tank	40
119	High-pressure (HP) Condensate Return Tank	18
120	Low-pressure (LP) Condensate Return Tank	20
248	Water Storage Tank	19

f. Process Panel Level Instruments

Hot well and reactor water level measurements are transmitted to the Process Panel.

The hot well level transmitter is of the beam balance type. It is similar to the pressure transmitters described previously, but utilizes a differential pressure bellows as a primary sensing element. The Process Panel indicator is a milliammeter.

The reactor level transmitter consists of a displacer-torque-tube mechanism which is coupled to a movable core differential transformer. Although the beam balance transmitter is considered to be generally more suitable for reasons outlined previously, it was not used here for the following reasons:

(1) The transmitter is located on the reactor vessel head inside the concrete top shield (see Fig. 31). Any maintenance, such as replacement of a vacuum tube, would then require moving one or more shielding blocks.

(2) The ambient temperature at the transmitter may at times approach 400°F. This is considerably higher than ordinary components of electronic circuits can withstand without damage.

The transmitting differential transformer is connected to a servo receiver in the Process Panel, which contains a second differential transformer. This unit detects any unbalance between the two transformers and repositions the core of the receiver transformer to re-establish balance. In so doing, a retransmitting slidewire is repositioned. The dc output signal derived from the slidewire serves as an input to the reactor water level recording instrument. The servo receiver is equipped with high and low-level reactor shutdown contacts. The level recorder is equipped with high and low-level alarm contacts, and provides one of the three inputs to the feedwater control system.

Additional information on these instruments is listed in Table 25.

Table 25

PROCESS PANEL LEVEL INSTRUMENTS

Instrument No.	Transmitter No	Measurement	Fig. 81 Detail No.	Reference Flow Diagram Figure No.
14	108	Reactor Water (Recorder, -6 ft 6 in. to 5 ft 6 in.)	12	18
15	109	Hot well (Indicator)	4	18

g. Locally Mounted Flow Instruments

Except for the simulated space-heating system water flow meter, which is a mercury manometer, all local flowmeters are of the tapered tube-float type These are listed in Table 26.

Table 26

LOCALLY MOUNTED FLOW INSTRUMENTS

Instrument No.	Transmitter Measurement	Reference Flow Diagram Figure No.
114	Retention Water Pump Discharge	40
115	Purification System Water	40
116	Air Ejector System Exhaust	18
117	Water and Space-heating Heat Exchanger	17

h. Process Panel Flow Instruments

For the flow measurements which are transmitted to the control room, flow nozzles are used in conjunction with differential pressure cells and beam balance transmitters. The transmitters are similar to the previously described pressure transmitters, but differ in that the output current signal is proportional to the square root of the input pressure difference. The associated Process Panel instruments are all self-balancing potentiometer-type recorders. The feedwater and main steam flow recorders also furnish input signals to the feedwater control system. These instruments are listed in Table 27.

Table 27

PROCESS PANEL FLOW INSTRUMENTS

Instrument No.	Transmitter No.	Measurement	Fig. 81 Detail No.	Reference Flow Diagram Figure No.
6	104	Feedwater (0-10,000 lb/hr)	5	18
8	105	Bypass Steam (0-3,000 lb/hr)	8	17
13	107	Main Steam (0-10,000 lb/hr)	7	17

i Water-resistivity Measurement

A water-resistivity measurement of the raw water demineralizer effluent (Instrument No. 121, Fig 19) is made locally. This instrument is connected to a motor-operated valve in the demineralizer inlet line so that the valve automatically closes when the effluent resistivity falls below a preset value

Two temperature-compensated cells are used to measure water resistivity at the purification system influent and effluent, respectively (Instrument Nos RI-19A and 19B, Fig. 40). These cells are connected through a selector switch to the Process Panel resistivity indicator (Instrument No 19, Fig 81, Detail No. 15).

j Position Indication

Except for the control rod drives, all position indication is accomplished by means of rheostats, mounted in the drive units, which are connected to ac voltmeters on the Process Panel. The indicator systems are energized from the 120-volt ac instrument supply. The indicators are listed in Table 28.

Table 28

PROCESS PANEL POSITION INDICATORS

Position Measurement	Fig. 81 Detail No.
Feedwater Valve	24
Bypass Valve "A" (0-3,000 lb/hr)	26
Bypass Valve "B" (0-10,000 lb/hr)	42
Mixer Damper	37
Exhaust Damper	38

k. Condenser Fan Motor Current

An ammeter is provided on the Process Panel to indicate condenser fan motor current. A 200/5-amp current transformer, located in the Motor Control Center (see Fig. 6) supplies instrument current proportional to motor line current.

l. Annunciator

A 36-point annunciator is installed on the Process Panel to indicate various abnormal conditions in the plant. All alarm points are connected to control contacts which are closed under normal operating conditions. The annunciator indicates visually and audibly when a pair of control contacts opens. The various plant conditions causing annunciation are listed in Fig. 82.

3. Radiation Monitoring

The following components are used to measure radiation levels in the plant:[31]

(a) Area radiation monitor
(b) Air ejector exhaust activity monitor
(c) Continuous air monitor
(d) Portable survey meters

a. Area Monitoring System

The area radiation-monitoring system utilizes five sensing units, one of which is mounted at or near each of the following points in the plant:

	Location	Instrument Range
(1)	Hot well	0.1 to 100 mr/hr
(2)	Feedwater filter	0.1 to 100 mr/hr
(3)	Main steam line	0.1 to 100 mr/hr
(4)	Turbine	0.1 to 100 mr/hr
(5)	Purification system vault	10 mr/hr to 10 r/hr

Each sensing unit contains an ionization chamber and an electrometer tube, the current through which is proportional to the logarithm of the intensity of the incident radiation. Five microammeters in the control room indicate the output current of the sensing units, and are equipped with adjustable alarm contacts which may be set at any point in the three-decade range. Individual indicating lights and a common Process Panel Annunciator point are actuated upon closing the alarm contacts.

b. Air-ejector Exhaust Activity Monitor

The purpose of this monitoring system is to detect failure of the fuel plate cladding by a continuous measurement of the gaseous fission product activity in the air-ejector exhaust stream. Referring to Fig. 76, a gamma-sensitive scintillation counter with its associated pulse preamplifier is mounted adjacent to the air-ejector exhaust stack (see Fig. 18). Pulses from this unit are amplified by a linear amplifier. The amplified pulses are then fed to a single-channel differential pulse-height analyzer which selects the pulses having an energy corresponding to the decay of xenon-135 and transmits them to an indicating logarithmic count rate meter. Experience with the EBWR indicates that a certain amount of xenon-135 activity, proportional to reactor power, is to be expected in the air-ejector exhaust. A sudden substantial rise in the measured activity will be considered indicative of failure of a fuel plate cladding.

c. Continuous Air Monitor

An air-particulate monitor is provided. This continuously monitors a filter through which ambient air is drawn. The unit includes an integral alarm and, in addition, is connected to the annunciator on the Process Panel. Automatic range switching is provided, with full-scale ranges of 0-2000, 0-10,000, and 0-20,000 counts per minute.

d. Portable Survey Meters

Portable radiation survey meters are used for periodic measurements of radiation intensity at various specific points in the plant during operation. These instruments are also used for assuring personnel safety during maintenance operations involving radioactive materials.

B. Control Systems

1. Reactor

a. Startup and Shutdown Interlocks*

Reactor power is controlled by positioning five cadmium control rods in the core. (Although provision is made for nine control rods, five cross and four tee, the four tee rods located at the edge of the core are not used with the present 40-fuel assembly core loading; see Fig. 12). The rods are driven by 3-phase induction motors, which are coupled to rack-and-pinion-type drive mechanisms through combination electromagnetic and freewheeling mechanical clutches. The motors are equipped with integral brakes to eliminate coasting. Thus, interruption of power to the magnetic clutches permits the rods to fall into the reactor shutdown position while the mechanical clutch permits positive "drive-in" in the event of sticking. Fast shutdown of the reactor is effected by de-energizing the electromagnetic clutches. Simultaneously, the control rod drive motors are energized in a timed sequence to drive in. Based on the design for nine control rods, they are energized in groups of four, three, and two, with starting time delays of zero, $\frac{1}{2}$ sec, and 1 sec, respectively. This was necessary in order to avoid imposing all rod drive motor-starting currents on the emergency power supply simultaneously with the consequent severe voltage drop.

Actuation of the reactor shutdown interlocks automatically causes the following additional events:

(1) The automatic demand control system is switched off "automatic."

(2) The turbine throttle valve is tripped.

(3) The generator main circuit breaker and field circuit breaker are tripped.

(4) The feedwater valve and both steam bypass valves are tripped off "automatic" and are driven closed. (Operation of a reset push-button is necessary to restore manual control.)

(5) The condenser air system damper and fan speed controls are switched from "automatic" to "manual."

Figure 85 shows the circuit arrangement of the reactor startup and shutdown interlocks. The contacts shown as 30A1, etc., are shutdown annunciator relay contacts. These are related to the shutdown contacts in the various instruments, as shown typically in Fig. 86.

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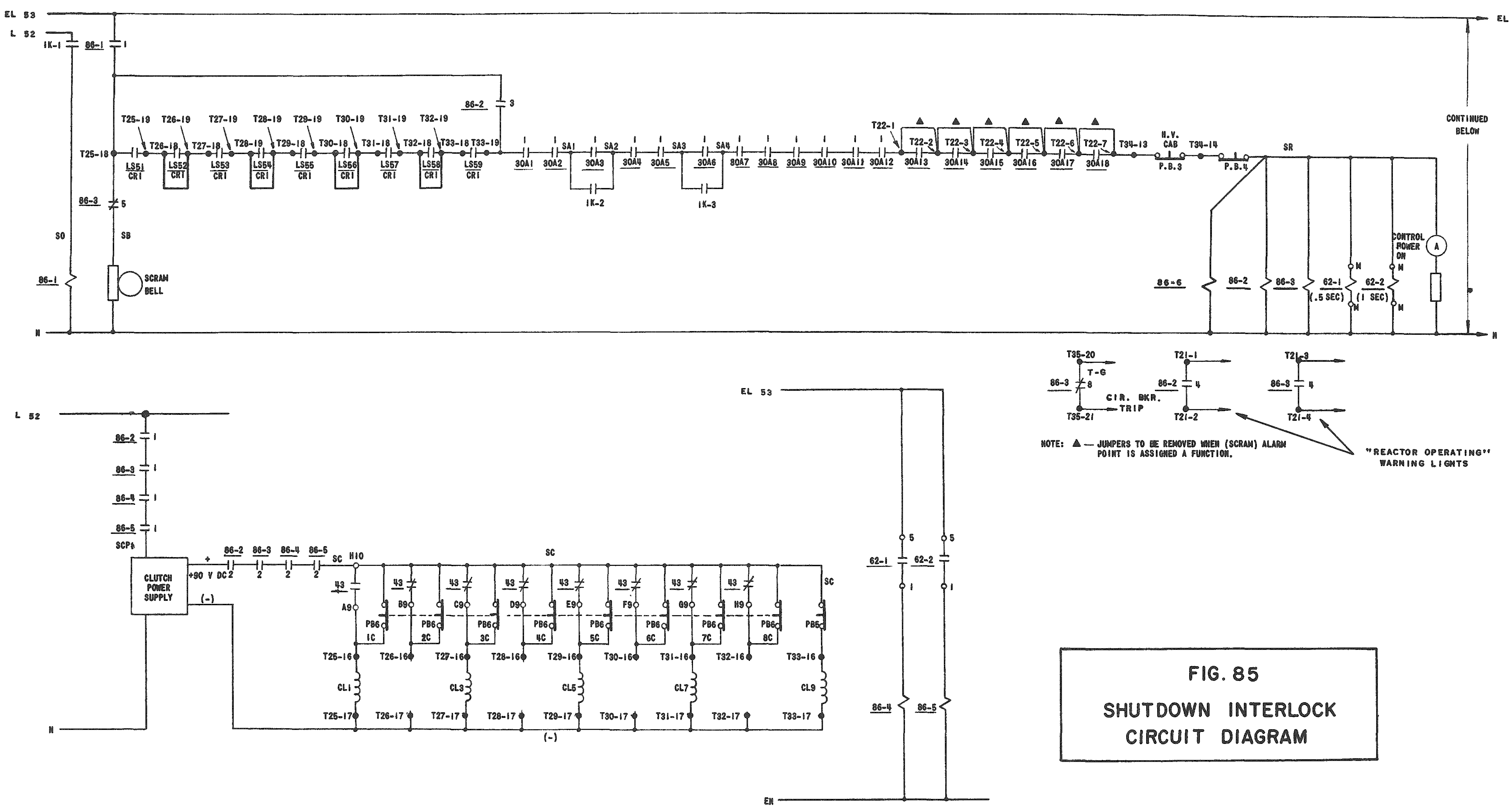

FIG. 85

SHUTDOWN INTERLOCK CIRCUIT DIAGRAM

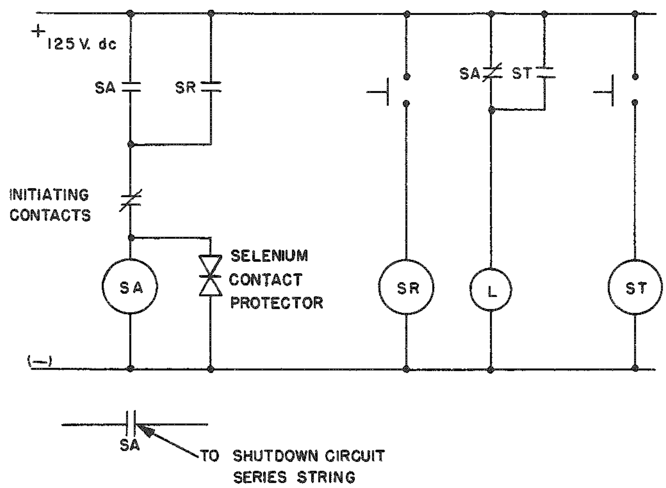

FIG. 86
TYPICAL SHUTDOWN
ANNUNCIATOR CIRCUIT

This circuit arrangement provides an identifying visual signal on the shutdown annunciator, which persists until manually reset, even though the initiating contacts may have reclosed. All instrument-actuated shutdown interlocks are connected to the shutdown annunciator. The reactor and plant conditions which cause actuation of the various contacts are listed below.

(1) LS51 to LS59, inclusive: "Rod in" limit switches (shown for nine control rods). These switches are mounted on the control rod drives and are closed when the respective control rods are completely inserted.

(2) 30A1: Channel I neutron flux, high; reactor power greater than approximately 4 Mw (see Fig. 78).

(3) 30A2: Channel II neutron flux, high; reactor power greater than approximately 4 Mw.

(4) 30A3: Channel III reactor period, short-contacts set for periods of the order of 10 sec. The period circuit does not yield meaningful information during steady-state operation above ~10% of full power. To prevent spurious shutdowns, the period shutdown contacts are bypassed in this region by means of key switch 1K-2.

(5) 30A4: Reactor water level, low; less than 2 ft 9 in. above reactor core.

(6) 30A5: Reactor water level, high; more than 5 ft 3 in. above reactor core.

(7) 30A6: Reactor steam pressure, low; less than 250 psig. The purpose of this interlock is to guard against possible instability accompanying boiling operation at full power and low pressure. Key switch 1K-3 is used to bypass the interlock during reactor startup.

(8) 30A7: Reactor steam pressure, high; more than 340 psig.

(9) 30A8: Main steam pressure-relief valve, flow. Flow through this valve is sensed by a temperature detector on the discharge side of the valve. The valve opens at 350 psig.

(10) 30A9. Main steam safety valve, flow. Flow through this valve is also sensed by a temperature detector. The valve opens at 385 psig.

(11) 30A10: Condenser pressure, high; greater than 3.5 psig.

(12) 30A11: Thermal shield (circuit No. 1) temperature, high; above 400°F.

(13) 30A12: Thermal shield (circuit No. 2) temperature, high; above 400°F.

(14) P.B.3: Reactor shutdown pushbutton, located on the High Voltage Supply Panel on the operating floor.

(15) P.B.4: Reactor shutdown pushbutton, located on the Reactor Control Panel in the control room.

Before control rods can be withdrawn following a shutdown:

(1) All rods must be fully inserted. This condition is indicated by individual "Rod In" lights on the Nuclear Panel (see Fig. 79).

(2) All instrument-actuated shutdown circuits, excepting Reactor Pressure, Low, must be satisfied as indicated on the shutdown annunciator (see Fig. 84).

(3) The Reactor Pressure, Low; shutdown bypass key switch must be closed (see Fig. 80)

(4) The control power key switch must be closed. Control power is taken from the emergency ac supply circuit so that operation of the shutdown alarm bell is independent of the normal ac supply. The control power relay (86-1, Fig. 85), however, is energized from the normal ac line so that a power failure will cause a reactor shutdown.

b. Manual Control*

Control rods Nos. 1, 3, 5, 7 and 9 are used with the 40-fuel assembly core. Rod No. 9 is the center cross rod, and rods Nos. 1,3,5, and 7 are the remaining four cross rods (see Fig. 12). Figure 87 shows the rod drive motor control circuits for control rods Nos. 1 and 9. The circuit shown for control rod No.1 is typical also of that used with Nos. 3, 5, and 7. Power for all control rod drive motors is supplied by the emergency power system. Thus electric power is continuously available for the automatic drive-in feature described in the preceding section (a). Standard mechanically and electrically interlocked combination reversing magnetic starters are used. These are centralized in a rod drive motor control center in the control room.

A single-turn synchro transmitter is geared to each rod drive pinion shaft and is connected electrically to a companion synchro receiver in the Nuclear Panel for rod position indication. The indicator pointer rotates approximately 270 degrees for a control rod travel of 30 in., and is accurate to approximately 0.1 in.

Two rod drive control switches are provided on the Reactor Control Panel. Referring to Fig. 87, one switch (1-1) may be connected to control any one of the peripheral rod drives by means of a selector switch (43). Additional contacts on this switch energize the "Selector Rod" indicating light beneath the appropriate rod position indicator. Interlocking contacts in the control switches prevent the simultaneous withdrawal of the center rod and any of the peripheral rods.

Contacts in the reactor shutdown circuit (86-2) automatically operate the rod drives in the "In" direction following a shutdown signal.

Limit switches (LS11, LS21, etc.), which are mounted on the rod drive mechanisms, stop the drive motors and energize "Rod In" and "Rod Out" indicating lights at the extremes of travel. These lights are also located immediately beneath the rod position indicators.

Shown in Fig. 88 are the control contacts associated with the automatic demand control system (see section c).

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*H. H. Hooker

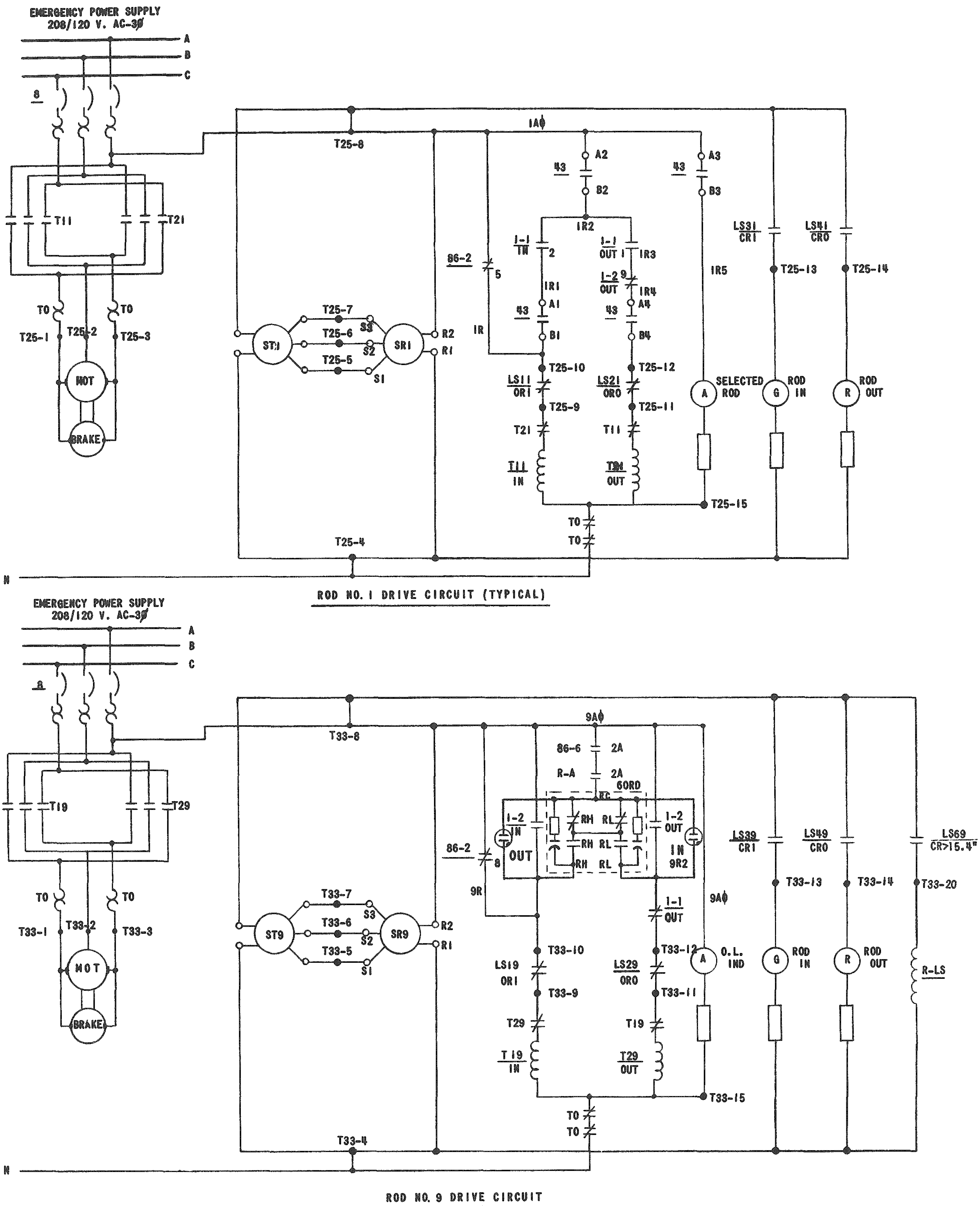

FIG. 87
CONTROL ROD DRIVE SCHEMATIC DIAGRAM

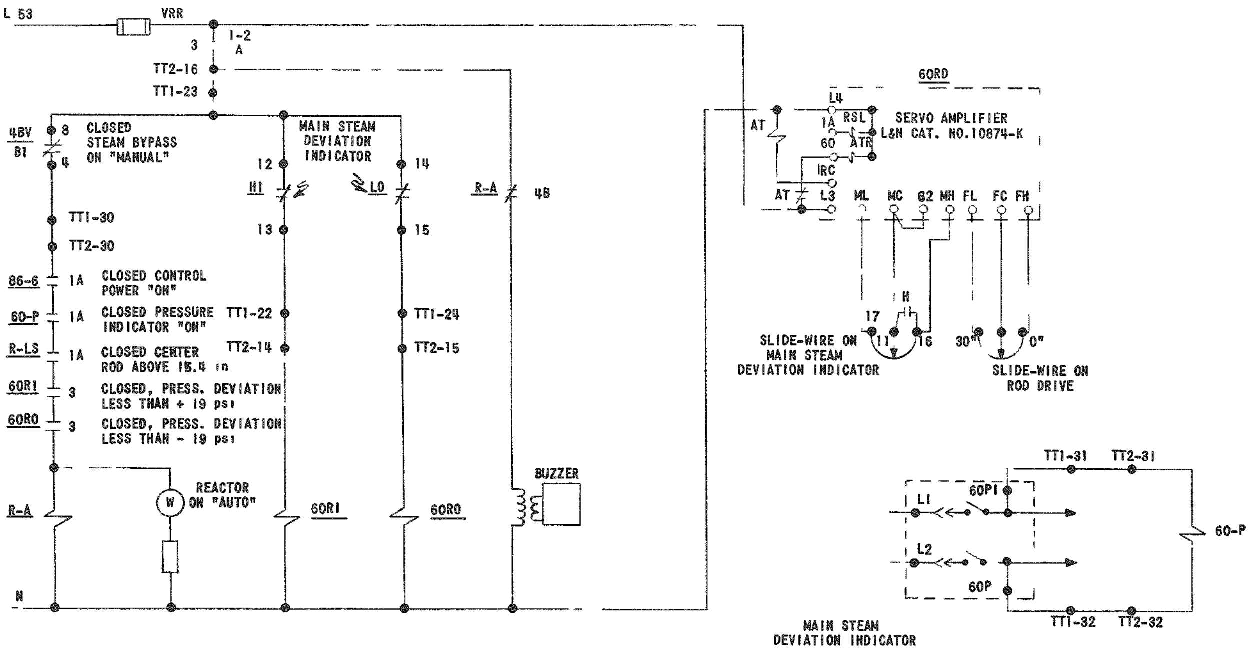

FIG. 88
AUTOMATIC ROD DRIVE CONTROL CIRCUITS

Two "Rod Drop" pushbuttons are provided on the Reactor Control Panel. The action of these pushbuttons is different from the "Reactor Shutdown" pushbutton (see section a) in several respects. With reference to Fig. 85, it will be seen that one pushbutton (PB6) is shunted by contacts on switch No. 43 for all but the selected control rod (No. 1 is shown in the figure). Thus depressing pushbutton No. PB6 will de-energize clutch coil No. CL1 and permit control rod No. 1 to drop. Similarly, pushbutton No. PB5 is connected to clutch coil No. CL9 for control rod No. 9. These pushbuttons are used in conjunction with the measurement of the time required for each rod to drop from a given position in the "rod-drop tests." The action differs further from that of the reactor shutdown circuits in that the affected control rod falls only while the pushbutton is depressed. None of the auxiliary shutdown switching action, described in the preceding section (a) takes place here.

The control rod drive clutches are supplied 90 volts dc by a rectifier. The rectifier, in turn, is energized from the normal ac line. Failure of the normal power source will de-energize the clutches, if for some reason relay No. 86-1 does not drop out.

c. Automatic Power Demand*

The design requirements (see Appendix V) specified that the reactor plant have the ability to adjust the power generation, and hence the steam output, to meet the electrical and space-heat system load demand placed upon the power plant. Also specified was reliable operation for prolonged periods of time with the minimum of supervision.

All previous boiling reactors required manual operation of control rods in order to change reactor power and used an auxiliary steam-bypass system to control reactor pressure. An auxiliary steam-bypass system can handle small load increases up to the amount of steam being bypassed or load decreases up to the full capacity of the bypass system. Any design based on steam bypass which is expected to handle load increases requires steam to be continuously passed through the bypass in the amount of the anticipated load increase. Where only electrical generation is required, the steam being passed through the bypass wastes power and lowers plant efficiency.

The ALPR is the first boiling reactor designed with automatic control of reactor power and does not require a steam-bypass system to control reactor pressure. The design provides for automatic adjustment of reactor power generation to meet load demand in any combination of space-heating or electrical load, including zero space-heat load. The ability to operate only the turbine-generator unit, without waste bypass steam, increases the plant efficiency and consequently extends the reactor core life. The automatic feature eliminates the necessity of full-time attendance of a reactor operator to follow load changes by moving control rods manually, and fulfills the design requirements of minimum supervision.

The ability of the reactor to automatically follow changes of electrical and space-heating load permits independent operation of a governor system to determine the amount of steam required by the turbine-generator unit, and a temperature-control system to determine the amount of steam required for space-heating purposes.

Figure 89 is a block diagram of the plant power demand control system. The portion of the diagram enclosed by the dashed line and referred to as "Reactor Dynamics," includes all effects taking place within the reactor pressure vessel. The feedwater control system and the automatic power demand system are shown as loops external to the reactor pressure vessel.(33)

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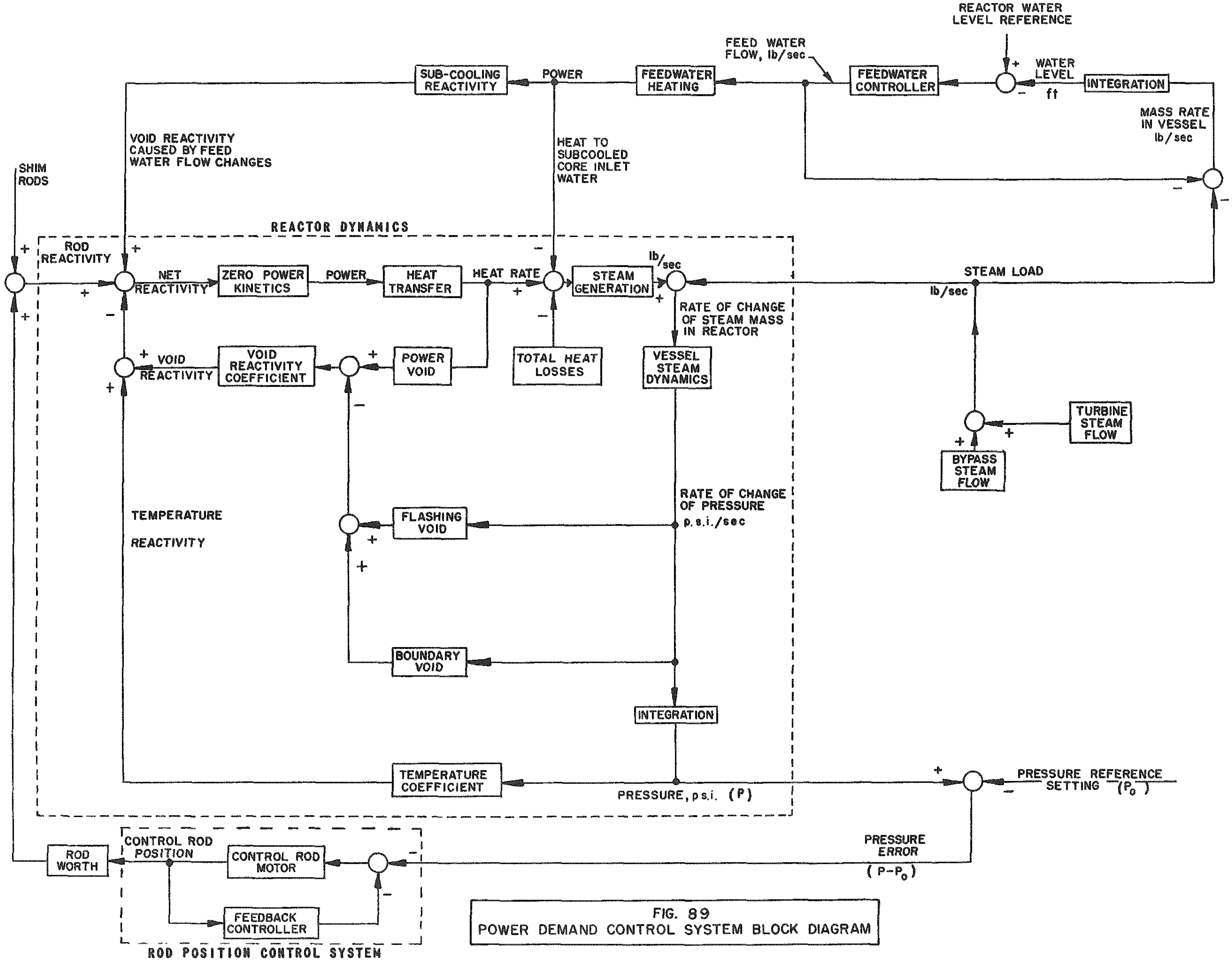

FIG. 89
POWER DEMAND CONTROL SYSTEM BLOCK DIAGRAM

Any differential between the steam load (turbine-generator unit plus space-heating system) and reactor steam generation results in a rate of change of pressure. The difference between the actual pressure and the pressure reference results in a pressure error. If the pressure is rising, the center control rod (rod No. 9) is inserted in proportion to the pressure change to decrease reactor power.

The rod position control system positions the center rod in proportion to the pressure error. Existing equipment used for manual control rod operation, consisting of motor starter, drive motor, and drive mechanism, is incorporated in an on-off-type position-control system. A commercially available relay amplifier with adjustable gain and reset rates is used for error detection between control rod position and pressure.

The ratio of control rod stroke to pressure error is adjusted to establish equilibrium reactor power without overshoot in minimum time with respect to a load change. With proportional action only, the reactor pressure will change and is directly proportional to reactor power under steady power operation. Reactor pressure is maintained at 300 psig at all loads by introducing reset action. The rate of reset is adjusted to restore reactor pressure (after a load change) to 300 psig as rapidly as possible without overshoot.

Supervisory interlocks are used to transfer the center rod control from automatic to manual in the event some malfunction develops in control equipment. The following conditions result in immediate transfer to manual operation and are audibly annunciated:

(1) Reactor Control Panel power off.

(2) $P-P_0$ pressure indicator power off.

(3) Reactor pressure error less than -19 psi.

(4) Reactor pressure error greater than +19 psi.

(5) Center control rod position less than 15.4 in.

(6) Center control rod position more than 22.9 in.

(7) Steam bypass control system in "automatic."

2. Reactor Pressure*

During manual power operation, the reactor operates at a constant power level determined by the setting of the control rods. The steam from the reactor flows either to the turbine-generator unit or through a

*W. C. Lipinski

bypass valve which admits steam to the space-heating heat exchanger (see Fig. 11). Reactor steam pressure is controlled by automatically adjusting the rate of flow of bypass steam in proportion to a pressure error, as shown in Fig. 90.

Any differential between the steam load requirement and reactor steam generation results in a rate of change of pressure. The difference between the actual steam pressure and the pressure reference results in a pressure error. If pressure is rising, the bypass valve is opened in proportion to the pressure change, the rate of flow of bypass steam is increased, the power differential is decreased, and the pressure will stop rising. Under equilibrium steam flow conditions, reactor pressure is proportional to reactor power.

The bypass valve is motor operated and is positioned by an electronic controller. By means of a control bridge circuit (which incorporates a slidewire in the $P-P_0$ pressure indicator and another slidewire on the valve drive mechanism) and an electronic relay unit, the bypass valve position is made proportional to the pressure error. A travel-ratio adjustment permits setting the ratio of movement of the bypass valve mechanism with respect to the $P-P_0$ slidewire, which has a total span of 40 psi. The minimum travel-ratio setting of 0.5 will stroke the valve through 50% of its travel with a pressure change of 40 psi, whereas the maximum travel ratio setting of 4.5 will full stroke the valve with a pressure change of 8.9 psi. A bias adjustment is provided to allow selection of the pressure at which the valve is fully closed. The fastest response and smallest pressure deviation in following load changes is obtained with a maximum travel-ratio setting of 4.5.

Control rods must be positioned manually to insure that the bypass valve is never closed or fully open in order to maintain control of pressure.

The pressure-control system is interlocked to transfer from automatic to manual operation and to drive the bypass valve closed in the event of reactor shutdown. The bypass valve can be opened manually, following a shutdown, after pressing a reset button.

3. Reactor Water Level*

The feedwater control system is designed to regulate the flow of water into the reactor vessel at a rate which will hold the level fixed at a set point (three-element control) or allow the level to deviate from the set point enough to hold water flow equal to steam flow (proportional control). The latter is a variation of the three-element control and is shown in Fig. 89.

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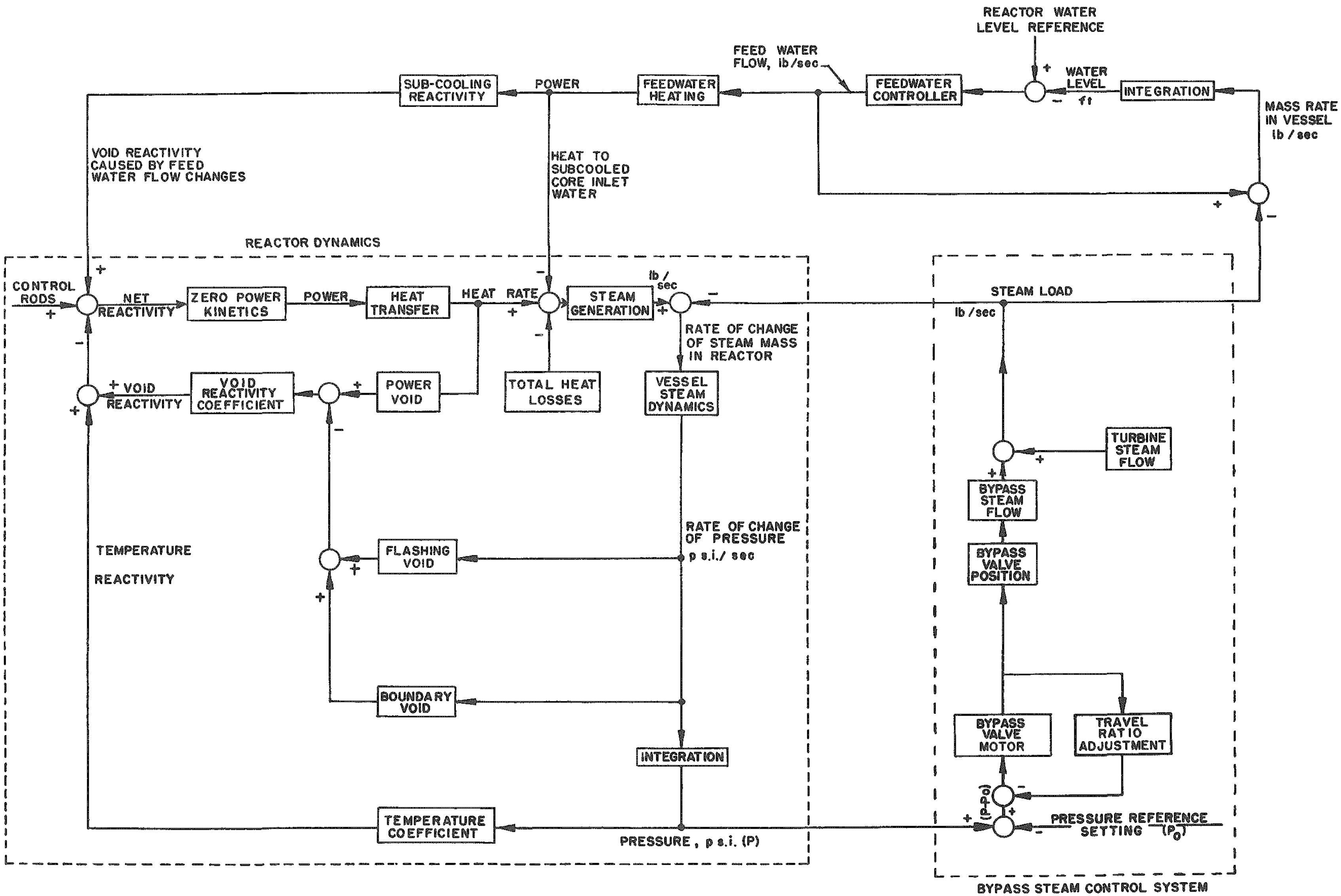

FIG 90
PLANT BLOCK DIAGRAM WITH
BYPASS STEAM CONTROL SYSTEM

Figure 91 illustrates the feedwater control system. The level and flow transducers have electrical signal outputs which are displayed on circular chart recorders located on the Process Panel in the control room. The potentiometers are retransmitting slidewires attached to the circular chart recorders.

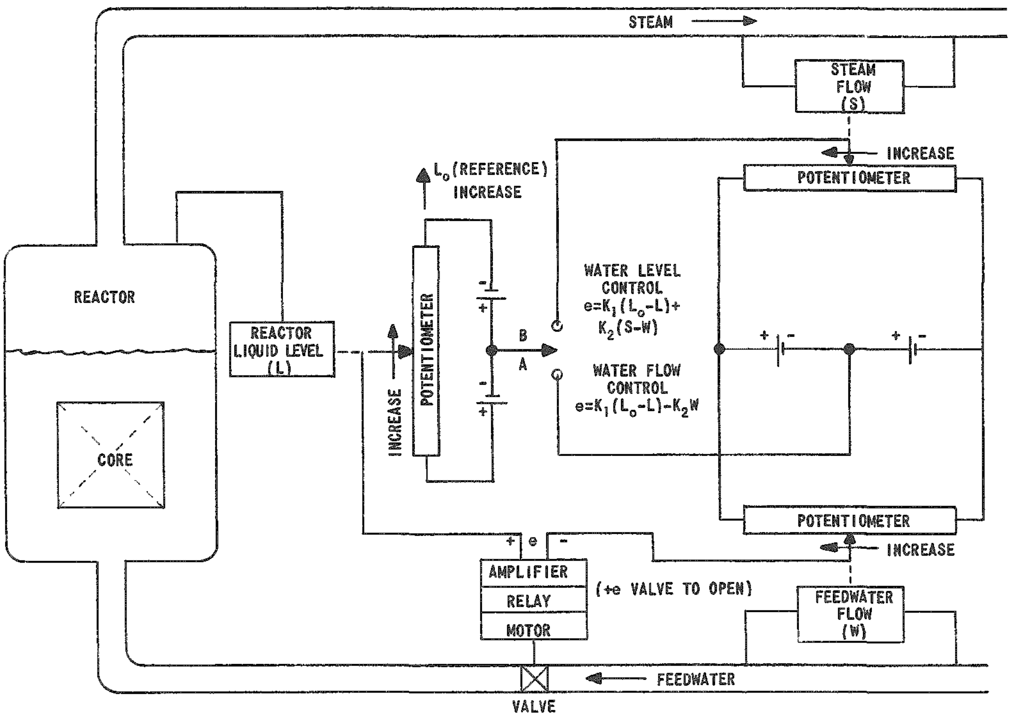

FIG. 91
FEEDWATER CONTROL SYSTEM

The control system is of the on-off type. A constant-velocity motor is energized by relays in a polarity-sensitive amplifier.

Referring to Fig. 91, with the switch in position "A" (proportional level control) the following equation represents the system error:

$$e = K_1(L_0 - L) - K_2W \quad , \qquad (1)$$

where

L_0 = reference level
L = reactor level
W = feedwater flow
K_1 = constant
K_2 = constant .

Under equilibrium conditions the error is equal to zero, the water flow rate is equal to the steam flow rate, and the feedwater flow rate is proportional to the water level deviation from the set point.

In switch position "B," three-element control is used, and

$$e = K_1(L_0 - L) + K_2(S - W) \quad , \tag{2}$$

where

S = reactor steam flow rate .

Under equilibrium conditions the error is equal to zero, the water flow rate is equal to the steam flow rate, and the water level of the reactor is equal to the set point. The level is related to steam flow rate and water flow rate by the following equation:

$$L = K_3 \int (W - S)\, dt \quad , \tag{3}$$

where

$K_3 = 0.00128$ ft/lb .

The error (e) determines the dead band of the system and the accuracy with which level is maintained. The gain of the relay amplifier, which is adjustable, determines the error necessary to operate the control relays. The control system design includes circuits for introducing integral and derivative control.

4. Condenser Air Outlet Temperature*

During periods of light plant load, the heat input to the main steam condenser is reduced. Consequently, if cooling air having a sufficient low temperature is available, condenser vacuum can be maintained with less than the maximum flow of cooling air. Since the condenser fan motor represents a large part of the auxiliary electrical load, plant efficiency can be increased by reducing the speed of the fan during these periods. To accomplish this, fan speed is automatically regulated as a function of condenser outlet air temperature.

Figure 92 is a diagrammatic representation of the condenser air control system. The fan is driven by its drive motor through a variable hydraulic coupling. The stiffness of the coupling is controlled by varying

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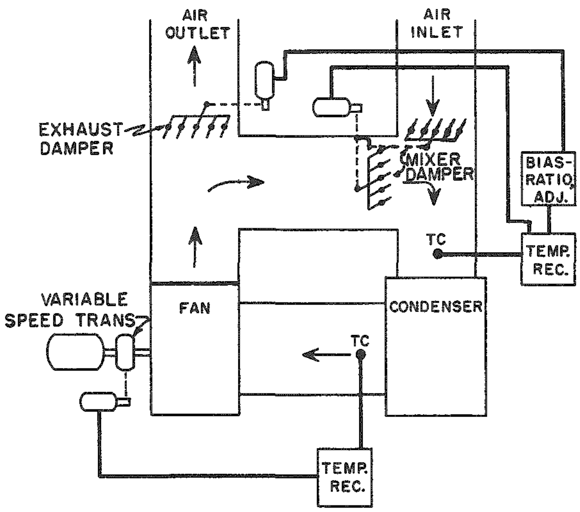

FIG. 92
CONDENSER AIR CONTROL BLOCK DIAGRAM

the position of an internal oil scoop tube, which in turn varies the amount of oil circulating between the driving and driven members of the coupling.

An array of 12 thermocouples is distributed across the outlet face of the condenser. These are connected in parallel through swamping resistors to yield a signal proportional to average outlet air temperature. This temperature is indicated and recorded on the Process Panel (item 10, Fig. 81) by the condenser outlet air temperature recorder (item 10, Fig. 81). A control slidewire in this instrument furnishes a signal proportional to the temperature deviation from a set point (normally 120°F) to a relay amplifier. The relay amplifier controls the scoop tube drive unit in an on-off fashion so as to increase the fan speed when the temperature is above the set point, and conversely. The scoop tube drive unit includes a retransmitting slidewire which is connected to the relay amplifier as a position feedback. The amplifier provides proportional, integral, and derivative control action. In the event of a reactor shutdown, the system reverts to manual pushbutton control.

5. Condenser Air Inlet Temperature*

Although plant efficiency increases as the main steam condenser cooling air temperature decreases, the condenser cannot be allowed to freeze. Consequently, when the outside air temperature is below freezing it is necessary to temper the condenser inlet air by recirculating a portion of the outlet air and mixing it with the incoming stream of outside air. With reference to Figs. 7 and 92, the relative amounts of recirculated and fresh air are controlled by positioning dampers in the exhaust duct and in the mixing chamber.

The automatic control system acts to control the dampers so as to maintain a constant condenser inlet air temperature slightly above freezing when the outside air is below the control temperature. A second array of 12 thermocouples (see section 4) is used to sense the average condenser inlet air temperature. This temperature is also indicated and recorded on the Process Panel (item 16, Fig. 81). The recorder includes a control slidewire which functions in conjunction with a relay amplifier and damper

*H. H. Hooker

drive unit to position the mixer damper. The mode of operation is identical to that described for the fan speed control (see section 4). Thus the mixer damper opens to admit more recirculated air as the condenser inlet air temperature falls below the control point.

The exhaust damper control relay amplifier derives its control signal from a retransmitting slidewire in the mixer damper drive unit. It is balanced by a position feedback from a slidewire in the exhaust damper drive unit. The exhaust damper relay amplifier incorporates "travel ratio" and "bias" adjustments. The travel ratio may be set at any value between 0.5 and 4.5, and determines the ratio of exhaust damper movement to mixer damper movement. The bias may be set at any point from 50% lag to zero to 50% lead. This setting determines the position of the exhaust damper when the mixer damper is at the midpoint of its travel. The directions of travel are such that the exhaust damper closes as the mixer damper opens.

Switching is provided so that both dampers may be positioned automatically, each may be positioned manually independently of the other, or the mixer damper may be positioned manually with the exhaust damper following automatically. The automatic control system is interlocked with the reactor shutdown circuits so that damper control automatically switches to "manual" in the event of a reactor shutdown.

Appendix I

EVOLUTION OF THE REACTOR CORE DESIGN AND REACTOR PHYSICS CHARACTERISTICS OF THE FRESH REFERENCE 3-Mwt SYSTEM*

A. Evolution of the Reactor Core Design

1. Introduction (cf. section V: Reactor Components)

In several important respects, the ALPR reactor is similar to the BORAX-II and BORAX-III reactors of Argonne National Laboratory.(34) All three are boiling water reactors: (a) fueled with uranium, highly enriched in U^{235}, in aluminum-clad fuel plates; (b) cooled and moderated by H_2O circulating within the pressure vessel under natural convection; and (c) controlled by a system of control rods comprised of cadmium confined within sheets of aluminum. Indeed, at the time of inception of the ALPR project, the BORAX reactors were the only boiling water reactors with plate-type fuel assemblies that had been operated at elevated temperatures and at power. Features of the reactor core design that were considered to have had important influences on the stability of operation of the BORAX reactors were essentially retained in the ALPR core.(2,3) In particular, in view of the proved stability of power operation of the aluminum-U-H_2O BORAX cores, and since no experience had been garnered with boiling water reactors fueled with steel-clad fuel plates or rods in a steel core structure, aluminum was used as the basic metal constituent of the core. (Moreover, it was estimated that the use of aluminum would result in a reactor core less expensive than a steel system.)

Control rods consisting of cadmium contained by aluminum sheets functioned well in the BORAX-III reactor, at temperatures and pressures close to the corresponding design conditions of the ALPR, whereas other control materials (e.g., hafnium) were expensive, or had not yet been proved at the time of selection of the basic control rod design (e.g., stainless steel containing 2 w/o boron), or appeared to offer little advantage over cadmium alone. Similar rods were used for ALPR.

In the following paragraphs of this section, the evolution of the design of various reactor core components is discussed, primarily in the context of the reactor physics of the system, and comparisons with the BORAX components are presented. To some extent there is duplication of some statements in the detailed description in section V-A (Reactor Components : Core) in an effort to maintain the continuity of the discussion. A general reference for this section and for much of the material in Appendix I is ANL-6078 (Zero-power Experiments on the Argonne Low Power Reactor).

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* D. H. Shaftman

2. The Fuel Assembly

To lengthen the useful life of the fuel assembly, a newly developed, corrosion-resistant alloy of aluminum with 1 w/o nickel, type X8001, was selected as cladding for the fuel plate and as the basic structural material of the fuel assembly.(35) Also, in contrast with the 0.020-in-thick fuel plate clad of BORAX-III, a cladding thickness of 0.035 in. was selected. To retain an overall metal-to-water ratio in the reactor core intermediate to the ratios in the early BORAX cores, and to avoid a high uranium content of the fuel-plate core ("meat"), a fuel-plate-core thickness of 0.050 in. was chosen, in contrast with a thickness of 0.020 in. in BORAX fuel plates. The fuel-plate core is comprised of an alloy of uranium enriched to ~93 w/o U^{235} and aluminum containing 2 w/o nickel. The nickel was added to improve the resistance of the alloy to corrosion by hot water, an extra margin of safety in the event of a localized breach of the clad. The use of thick fuel plates made it feasible to dispense with the strengthening center web structure used in the BORAX-III fuel assembly, and, instead, the fuel plates were flanged and spot welded to two aluminum-nickel (type X8001) side plates. The active core length of 25.8 in. of BORAX-III was retained in ALPR. A fuel plate separation of 0.310 in. was chosen to attain an overall metal-to-water ratio of 0.5 in the core. The corresponding coolant channel dimension in BORAX-III was 0.264 in., but it was not expected that the somewhat larger channel would have a strong influence on the stability of the power operation.

The outcome of these design decisions was a fuel assembly containing a total of 350 gm U^{235} in nine clad fuel plates, flanged and welded to two side plates. The assembly is $3\frac{7}{8}$ in. square and has an active length of 25.8 in. (For further details, see section V-A-1: Reactor Components - Fuel Assembly.)

3. Reactor Core Size

A 3-Mwt reference core with forty fuel assemblies was chosen. With this loading, at 3 Mwt, an average power density of ~17.5 kwt per liter of coolant fluid in the core was obtained. This level appeared to be conservative in comparison with known stable power densities achieved in BORAX-III.(34) However, for a variety of reasons, provision was made for a maximum core loading of fifty-nine fuel assemblies plus one source assembly. Of these motivations, the most important were: (a) the operation with thicker fuel plates and larger coolant channels than had been used in BORAX, and the anticipation that an automatic demand-control system would be utilized to adjust reactor heat output to load needs, with the attendant uncertain effects on the stable power output of the reactor core; (b) a flexibility in core loading to permit reactivity adjustments shown to be desirable as a result of zero-power or power experiments; and (c) the desirability of having available a reactor core from which significantly higher power could be got, for possible use in another plant. The average fuel loading of each assembly (~350 gm U^{235}) was determined by theoretical analysis for the forty-fuel-assembly, 3-Mwt core, and for a nominal core life of three years at average power. In Fig. 12 a schematic of the reference core loading for 3-Mwt operation is shown, including forty fuel assemblies, one source assembly, and ten dummy fuel assemblies used to keep the fuel assemblies and the source assembly vertically aligned in the core.

4. Control Rods

The locations, the number, and the size of the cross-shaped control rods were determined by the combination of the size of the rod drive mechanisms, the limited effective (radial) area of the forty-assembly reference core, the possibility that a still larger core might be used with only the cross-shaped rods for control, and the reactivity hold-down requirements of the cold reactor without xenon.

It was calculated that upon withdrawing the five reference cross-shaped control rods from zero to 30 in. the net change in the total reactivity controlled by the control rods and by a burnable poison distributed uniformly in the fuel-plate core would be 14% to 15% (~20 dollars) in the cold fresh reactor.(36) (In this reactor, one dollar corresponds to approximately 0.7% reactivity.) With an allowance for the uncertainty of the calculation, it was clear that the rod effectiveness could not be reduced and still meet the requirements of rod control, including: (a) compensation of the positive reactivity effects of loss of steam voids, cooling of the reactor water, and loss of equilibrium xenon; (b) the control of the margin of excess reactivity in the fresh operating reactor; and (c) a shutdown margin large enough to permit the full withdrawal of at least one off-center cross control rod in the subcritical system. In the reference design, it was calculated that the cold fresh reactor could not be shut down with the center rod at 30 in. Therefore, alternative designs were considered, including a system of four rods in a diamond pattern, and a system of nine control rods with five rod drives. The four-rod design was rejected because it was calculated that the cold fresh system probably could not be shut down with <u>any</u> cross rod stuck out of the core. The nine-rod design required that each of four control rod drives move two rods, while the fifth drive served for the central control rod. This concept was rejected because of the difficulty of driving two control rods vertically within narrow channels with a rod drive mechanism that was not centered over either rod.

Four tee-shaped control rod channels were made available for additional control by tee-shaped rods in cores larger than the system of forty fuel assemblies. The tee-shaped rods were not intended for use in the reference 3-Mwt system since they are not very effective in that loading and it was highly improbable that their presence could offset removal of the central cross-shaped control rod and keep the cold fresh reactor shut down. (It was determined experimentally that the final 59-fuel-assembly loading could be controlled by the five cross control rods alone, but it would be advisable to use the tee rods also to increase the margin of shutdown to the point where possibly one off-center cross control rod could be withdrawn to 30 in., leaving the cold reactor subcritical.)

Control rods consisting of cadmium sheet confined between sheets of aluminum had been used successfully in BORAX at similar operating temperatures. A control rod of the type proposed was relatively inexpensive. There were no problems of gas formation as a result of neutron absorption by the cadmium. And finally, in view of the thermal "blackness" of the 0.060-in-thick cadmium, it was possible to operate such a rod for at least two core cycles without a significant change in rod effectiveness, assuming that the only loss of Cd^{113} was by neutron absorption. Therefore a reference system of five cross-shaped control rods, each with an absorbing span of 14 in. and an absorbing length of 34 in., and a system of four tee-shaped rods were designed, consisting of cadmium contained within sheets of X8001 alloy. When the control rods are positioned at their indicated zero position, the cadmium overlaps the bottom of the active core by $3\frac{1}{8}$ in. and the top of the active core by $5\frac{1}{16}$ in. (Additional details of the composition and fabrication of the control rods are presented in section V-A-4: Reactor Components - Control Rods.)

5. Burnable Poison

In the early phases of the core design, the decision was made to supplement the control of reactivity by the system of cadmium rods through the incorporation of a burnable poison in the fuel-plate cores, in the form of boron essentially fully enriched in B^{10}. Theoretical analysis indicated that 16 gm of B^{10} would control $\sim$11% in the cold fresh reactor when uniformly mixed with the U^{235}. In accordance with a request by the Army Reactors Branch that the fuel assemblies be suppliable commercially, research and development contracts were let for a technique of fabricating fuel plates with B^{10} in the "meat." These contracts did not lead to a method for producing fuel plates of the specified quality of integral bonding of cladding to the fuel-plate core. To expedite the procurement of fuel plates from a commercial fabricator, it was decided not to require that the fuel plate contain boron and, instead, to fusion-weld thin strips, extruded from an aggregate of powders of aluminum-nickel (X8001) and of boron enriched in B^{10}, to one or both side plates of selected fuel assemblies. "Poison" strips of a similar type had been used in BORAX-III to assist rod control and had provided satisfactory service.(37) Two lots of burnable-poison strips were obtained, forty strips 0.026 in. thick and containing nominally 0.5 gm B^{10}, and forty strips 0.021 in. thick and with $\sim$0.4 gm B^{10} in each full-length (25.8-in.) strip. In the reference 3-Mwt core, one full-length 0.5-gm-B^{10} strip is attached to each fuel assembly. In each of the central sixteen fuel assemblies one half-length strip, containing $\sim$0.2 gm B^{10}, is welded to the other side plate, in the lower half of the active core region.

B. Reactor Physics Characteristics of the Fresh Reference (3-Mwt) Reactor at Room Temperature (Zero-power Experimentation)[38]

1. Introduction (cf. section V: Reactor Components)

In summary, the reference 3-Mwt reactor (loading No. 57)[38] includes forty fuel assemblies, one Sb-Be source assembly, and ten dummy fuel assemblies. The fuel assemblies are arranged so as to approximate a right circular cylinder, presenting an active core region 25.8 in. long and with an effective diameter of ~31.4 in. The reactor contains a total of 14.0 kg U^{235}, in the form of uranium enriched to ~93 w/o U^{235}. Each fuel assembly is $3\frac{7}{8}$ in. square and contains nine fuel plates comprised of a center "meat" portion of Al-Ni-U alloy clad with an Al-Ni alloy (X8001). The reactor is moderated and cooled by H_2O circulating by natural convection within the pressure vessel. The basic metal constituent of the reactor core is aluminum; the overall metal-to-H_2O volume ratio in the core is ~0.5. Five cross-shaped control rods are included. Each rod has an absorbing span of 14 in. and an absorbing length of 34 in. The neutron absorber is 0.060-in.-thick cadmium, confined between two sheets of X8001. The central cross control rod has a X8001 follower, 17 in. long; the followers for the other rods are 5 in. long. All five control rods are moved, one at a time, for shim control and for reactor shutdown. (The center cross rod is moved to regulate the reactor heat output in connection with an automatic demand-control system.) Supplementary control of the reactivity required in the fresh reactor for a nominal reactor core life of three years is provided by thin burnable-poison strips containing an aggregate of X8001 and boron highly enriched in B^{10}. One full-length (25.8-in.) strip containing ~0.5 gm B^{10} is fusion-welded to one side plate of each fuel assembly, and one half-length strip containing ~0.2 gm B^{10} is welded to the bottom half of the other side plate of each of the central sixteen fuel assemblies.

2. Reactivity Effects of Control Rod Motion

At 94°F, the reference reactor was critical with the five control rods banked at an indicated position of ~12.6 in., or with the central rod (No. 9) at 19.1 in. and the other rods at zero. (At indicated "zero," the cadmium in each control rod extends $3\frac{1}{8}$ in. below and $5\frac{1}{16}$ in. above the active core region.) At room temperature, the reactor was subcritical with any one off-center cross rod at 30 in. and the other rods at zero, and it was subcritical by approximately 5 dollars when all control rods were inserted fully.

The control system was calibrated, in terms of the motion of control rods from a critical banked position, by measuring the positive asymptotic period corresponding to a slight withdrawal of the control rod(s). During these calibrations, the antimony source rod was removed from the beryllium block to eliminate interference from the Sb-Be neutron source.

The requisite additional control was obtained by the incremental addition of boric acid to the reactor water. Figure 93 shows the relationship between the asymptotic reactor period and the (positive) excess reactivity (in dollars), computed on the basis of the Keepin-Wimett-Ziegler measurements[39] of the parameters of the delayed neutron groups. Curves showing the differential reactivity effects of moving rod No. 9, or the four off-center cross rods from the five-rod bank are given in Fig. 94. An alternative fit to the curve for rod No. 9 at indicated positions between 15 in. and 20 in. is shown by a dashed line in Fig. 94. It is believed that the solid line is a better fit, but the alternative curve section is admissible.

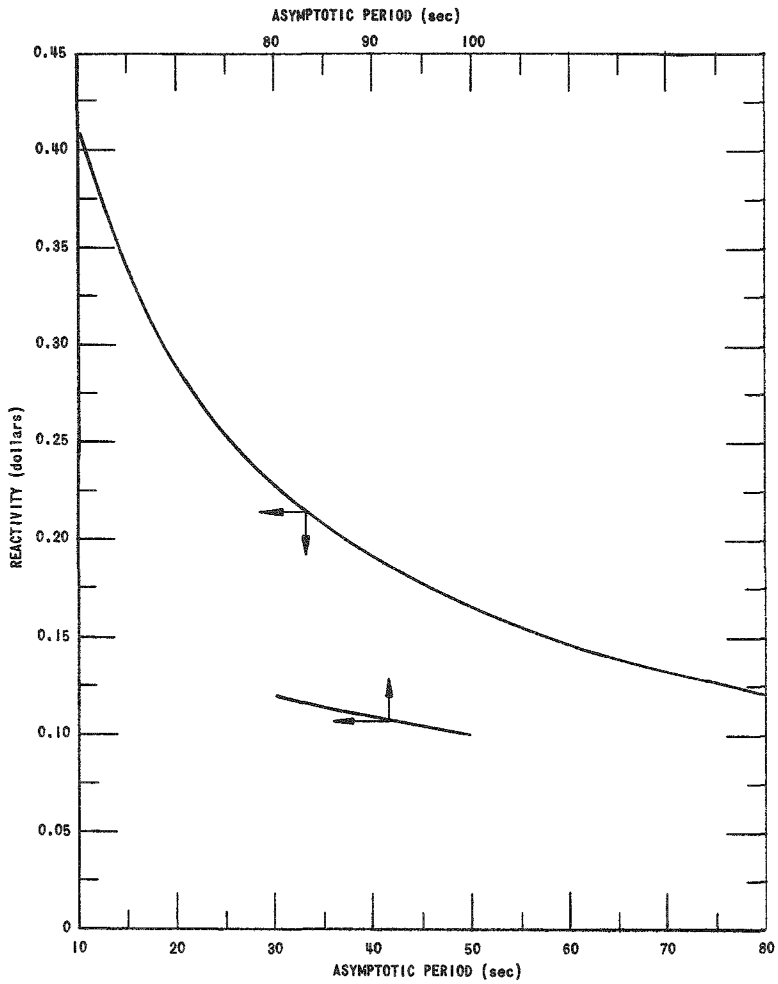


FIG. 93
REACTIVITY vs. ASYMPTOTIC REACTOR PERIOD
(β=0.006398; ℓ^*=5x10^{-5}sec)

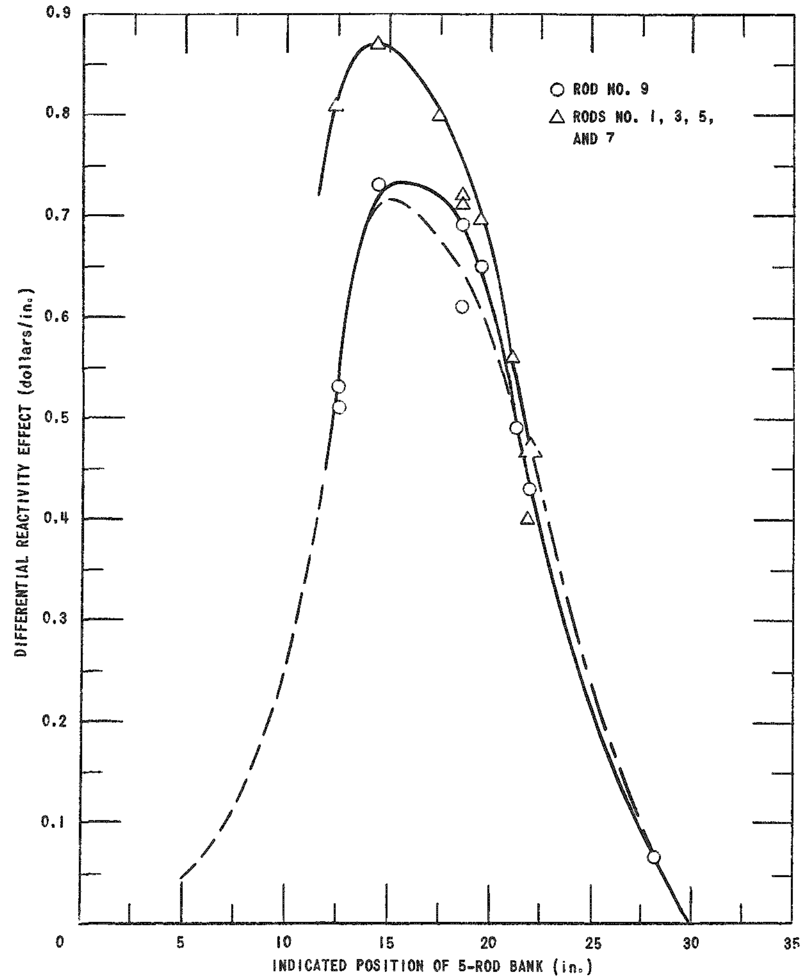


FIG. 94
DIFFERENTIAL REACTIVITY EFFECT OF MOVING CONTROL RODS FROM 5-ROD BANK (LOADING NO. 57 AT ROOM TEMPERATURE)

Integration of the curves in Fig. 94 from the generic indicated position Z to 30 in. yields the curves shown in Fig. 95. Note that Fig. 95 provides an estimate of 16.5 dollars (~11.5%) for the net reactivity effect of withdrawing the five control rods to 30 in. in the cold fresh reactor unexposed to fuel burnup. An alternative formula used to estimate the total reactivity effect, $\rho(Z_0 \rightarrow Z_n)$, of withdrawing the five rods from Z_0 to Z_n is [38,40]

$$\rho(Z_0 \rightarrow Z_n) \sim 1 - \exp\left(-\int_{Z_0}^{Z_n} \frac{d\rho}{dZ}\, dZ\right) \quad ,$$

where $\frac{d\rho}{dZ}$ is the differential reactivity effect of motion of the 5-rod bank at position Z, in units of %/in. This formula yields an estimate of 10.9% available excess reactivity.

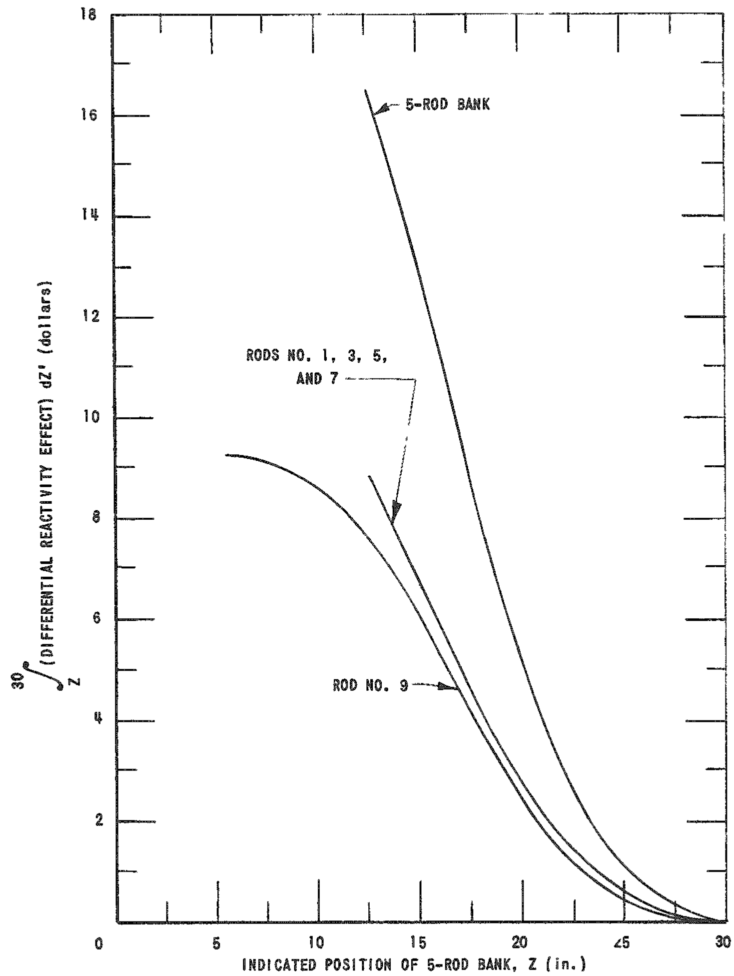


FIG. 95
INTEGRAL OF DIFFERENTIAL REACTIVITY EFFECT OF MOVING RODS FROM 5-ROD BANK IN COLD FINAL REFERENCE 40-ASSEMBLY CORE (LOADING NO. 57)

3. Reactivity Effect of Boric Acid Addition

Boric acid poisoning of the reactor water was calibrated in three different core loadings. In two loadings, an excellent fit to the total worth of boric acid addition by a function linear in the H_3BO_3 concentration in the reactor water was observed and an average value of $\sim$2.4 dollars/(gm H_3BO_3/gal) inferred.(38) In the cold reference core, however, it was more difficult to assign an average worth. Possibly there was a real variation in the worth of H_3BO_3 because of the effect of the half-length strips. Possibly the use of a less direct procedure of water sampling, required during these experiments, introduced a spurious variation. The measured worths varied between 2.4 dollars/(gm/gal) and 2.7 dollars/(gm/gal), over the range of boric acid concentrations encountered, with the reactor water at room temperature. The average worth is approximately 2.5 dollars/(gm H_3BO_3/gal).

4. Reactivity Controlled by the Burnable-poison Strips (Control Rods Banked at ~20 in. Indicated Position)

Strictly speaking the burnable-poison strips were not calibrated in the 3-Mwt reference core, for during most of the calibration experiments only the four assemblies at the axis of the core had the half-length poison strips rather than all of the central sixteen assemblies. However, it is believed that the strip calibration in the actual reference system would not be much different. It is believed that the absence of the additional twelve half-length strips introduced an effect of underestimation of the worth of the other strips in the central sixteen assemblies. Starting with core loading No. 45 in the sequence of zero-power experiments (essentially identical to the 3-Mwt reference loading except that there were no half-length burnable-poison strips in loading No. 45), strips were calibrated with the five control rods banked at ~20 in. The results are summarized in Table 29. There is a significant reduction in the reactivity worth of the full-length strips relative to their worth in a core loading where rods are inserted farther.[38] A total strip worth of ~18 dollars is indicated in the cold reference core with the control rods banked at 20 in.; of this, ~5 dollars is contributed by the 16 half-length strips.

The strips were calibrated in units of four to reduce the reactivity uncertainty arising from an uncertainty of ±0.05 in. at each indicated position of the control rod.

5. Reactivity Effect of Fuel Assembly Omission

The procedure proposed to effect a loading change at an ultimate site is to remove or to introduce assemblies through control rod nozzles in the pressure vessel lid. It is possible, though unlikely, that a fuel assembly location could be left unfilled until the loading was otherwise completed. It was of interest, therefore, to ascertain the effect on reactivity of inserting a fuel assembly into an otherwise-full core loading, containing a total of thirty-nine fuel assemblies. The actual experiment was performed in loading No. 19, the initial reference forty-fuel-assembly core, with only the one 0.5-gm-B^{10} strip per assembly; it consisted of measuring the change in the critical position of rod No. 9 upon removal of a fuel assembly from the 40-assembly core, as a function of the radial location of the assembly. The four off-center control rods were at indicated zero throughout this series of experiments. The results are given in Table 30 uncorrected for the effect of assembly removal on the calibration of the control system. It is believed that with half-length strips in place on the central sixteen fuel assemblies, the removal of a fuel assembly from those sixteen locations would result in a smaller negative reactivity effect because the half-length strip would be removed as well. Some of this effect would be counteracted by the increased worth of the control rod adjacent to that half-length strip.

Table 29

REACTIVITY CALIBRATION OF BURNABLE-POISON STRIPS "IN" LOADING NO. 45
(BASE ROD BANK ~20.12 IN.)

Fuel Assembly Locations[a]	Strip Modifications[b]	Estimated Reactivity Change[e] (dollars)
44; 45; 54; 55	0.5 gm $B^{10} \longrightarrow$ 0.4 gm B^{10}	+0.38
44; 45; 54; 55	0.5-gm-B^{10} strips removed.	+2.8
44; 45; 54; 55	Half-length "poison" strip containing ~0.2 gm B^{10} added to bottom half of the other side plate.[c]	-1.5
43; 46; 53; 56	0.5-gm-B^{10} strips removed.	+2.2
33; 36; 63; 66	0.5-gm-B^{10} strips removed.	+2.0
42; 47; 52; 57	0.5-gm-B^{10} strips removed.	+0.85
32; 37; 62; 67	0.5-gm-B^{10} strips removed.	+0.55
	Twelve more half-length strips each containing ~0.2 gm B^{10} added to remainder of the central sixteen assemblies.[d]	-3.5 to -3.9

[a] See Fig. 12

[b] Strip-equivalence relationships may be observed in terms of the following assembly locations:

43 = 35 = 56 = 64	42 = 74 = 57 = 25
34 = 53 = 46 = 65	41 = 84 = 58 = 15
44 = 45 = 54 = 55	51 = 85 = 48 = 14
33 = 36 = 63 = 66	23 = 62 = 76 = 37
32 = 73 = 67 = 26	24 = 52 = 75 = 47

[c] These four half-length strips were left in position for the remainder of the strip calibrations. Boric acid solution was added to lower the critical rod bank to ~20.2 in.

[d] This configuration is loading No. 57, the core composition of the final reference forty-assembly reactor.

[e] One dollar ≈ 0.7%

Summary:		
	Forty full-length 0.5-gm-B^{10} strips	12.7 dollars
	Sixteen half-length 0.4-gm-B^{10} strips	5.0 to 5.4 dollars
	TOTAL -	~ 17.9 dollars

It should be noted that the experiment could not be duplicated in the (final) reference loading since the critical position of rod No. 9 in this loading is ~19.1 in. instead of the value of ~12.9 in. observed with the initial 40-assembly core (see Table 30).

Table 30

REACTIVITY LOSS UPON REMOVAL OF A FUEL ASSEMBLY FROM THE OTHERWISE-FULL INITIAL REFERENCE FORTY-ASSEMBLY CORE (LOADING No. 19)

Fuel Assembly Location[a]	Change in Indicated Critical Position of Central Control Rod[b] (in.)	Estimated Reactivity Effect[c] (dollars)
47	1.1	-1.4
36	1.55	-1.9
46	3.2	-3.6
45	6.7	-6.6

[a] See Fig. 12

[b] Initial critical rod configuration: Rods No. 1, 3, 5 and 7 at zero; rod No. 9 at 12.90 in.

[c] By positive-period measurements, the differential reactivity effect of withdrawing rod No. 9 with the other four rods at zero was determined to be ~1.3 dollars/in. at 12.9 in., and 0.66 dollar/in. at 19.6 in. The net differential reactivity effect was assumed to vary linearly between these two values in obtaining the numerical values tabulated in this column.

Note: One dollar ≈0.7%.

C. Reactor Physics Characteristics of the Fresh Reference (3-Mwt) Reactor at Elevated Temperatures

1. Reactivity Effects of Reactor Water Heating (Zero Power)

It is important to distinguish between: (1) the change in reactivity of a fixed ALPR core configuration when the temperature is changed, and (2) the net change in the excess reactivity available to the system when the temperature is changed. In the fresh 3-Mwt reference core loading, it was necessary to move the control rod bank to higher indicated positions to maintain criticality during heating of the reactor water. In the sense of (1) above, this represented a sizable negative reactivity effect of heating. However, with an increase in water temperature, the control system became more effective, i.e., the reactivity controlled with the rods at given position increased with temperature. Thus, in the sense of (2), there was a (numerically) smaller change in the net excess available reactivity.

During the program of zero-power reactor physics measurements an attempt was made to determine the "average temperature coefficient(s)" of reactor heating, primarily in the sense of (2) above, with the control rods banked at ~21 in. An auxiliary steam boiler, connected to temporary steam coils located in the radial "downcomer" region of the core, was used to heat the reactor water to ~417°F, (essentially) the operating temperature of the water in the pressure vessel.(7) Boric acid was used to poison the reactor water in the cold system to compensate for the withdrawal of the control rod bank to 21 in. At higher temperatures it was necessary to withdraw rods farther to achieve criticality, in spite of the positive reactivity effect of the reduction in the concentration of boric acid in the core accompanying the reduction in water density. Thus, even with boric acid in solution, if the rods had been left in place the reactor would have become more and more subcritical as the heating progressed. From time to time, clean water was pumped into the pressure vessel to reduce the H_3BO_3 concentration so as to retain a critical rod bank position of ~21 in.

It was discovered later that the procedure of taking samples of the reactor water at elevated temperatures was inadequate. Therefore, the variations in concentration of boric acid in the reactor water during these experiments were computed from recorded data of dilutions plus reductions in water density corresponding to measured changes in its saturation temperature. The analysis of a reactor water sample taken at 150°F supplied the base point of the computation. The reactor was critical at 150°F with the control rods banked at ~21.2 in. and with a reactor water concentration of ~5.35 gm H_3BO_3/gal. Upon heating of the water to 287°F, in conjunction with a deliberate addition of clean water, the critical rod bank position was changed to 21.6 in. A positive-period calibration of the rods at this position yielded an estimate of ~1.5 dollars/in. for the five-rod bank. The boric acid concentration in the reactor water had been reduced by an estimated 1.2 gm/gal. Thus, <u>assuming that the boric acid poisoning effect is the same as in the cold reactor</u>, namely, ~2.5 dollars/(gm H_3BO_3/gal), it is estimated that, if the boric acid concentration had been unchanged during the heating, the reactor would have become subcritical by ~3.6 dollars upon heating from 150°F to 287°F, or an average of ~-2.6×10^{-2} dollar/°F. Taking into account the increased effectiveness of rods at the higher temperature, it is estimated that the net loss in available excess reactivity was ~1.7 dollars, corresponding to an average reactivity change of ~-1.3×10^{-2} dollar/°F from 150°F to 287°F.

By similar methods, it is estimated that the net change in available excess reactivity in heating from 90°F to 421°F is $-3.3(\pm 0.5)$ dollars with control rods banked at ~21 in., or an average of $-1.0(\pm 0.2) \times 10^{-2}$ dollar/°F. (As discussed in Appendix I, section C-4, following, it is conceivable that the boric acid reactivity effect is reduced somewhat at higher temperatures. This effect has not been taken into account in these computations.)

At room temperature, the critical rod bank position in the unpoisoned core was ~12.6 in. With rods inserted this far, a uniform heating of the reactor water would lead to a large negative reactivity effect, in part because of the resulting increase in control system effectiveness. When the reactor was first taken to temperature by nuclear heating, the critical position of the five-rod bank was increased by ~1.2 in. when the reactor water temperature was raised from 70°F to 170°F. If, then, the rods had been lowered to their initial, 70°F, position, the reactor would have become subcritical by ~1.8 dollars.

2. Reactivity Effects of Control Rod Motion

The differential reactivity effect of moving control rods from the five-rod bank was measured at zero power, at various temperatures, and at various banked rod positions. The calibration procedure adopted was to heat the reactor water with the auxiliary steam boiler circuit, wait until the water temperatures measured above and below the core were equalized, determine the critical position of the rod bank, and then withdraw rod(s) slightly for positive-period measurements. The accuracy of the calibration was checked by repeating the period measurement. The positive reactivity effect of reactor cooling introduced some uncertainty, for usually one set of calibrations required 15 to 30 min., during which the reactor water temperature dropped 2 to 5°F. However, it was apparent that the differential reactivity effect of rod motion from a given critical rod bank position increased considerably as the temperature was raised. There was no detailed calibration of the control system at the operating temperature, but various isolated local calibrations are available that corroborate this statement. For example, at a critical rod bank position of ~21.8 in.,

$$\text{Rod No. 9:} \begin{cases} 0.43 \text{ dollar/in.} & \text{at } 95^\circ\text{F} \\ 0.54 \text{ dollar/in.} & \text{at } 225^\circ\text{F} \\ 0.65 \text{ dollar/in.} & \text{at } 285^\circ\text{F} \end{cases} .$$

The measured change in "rod worth" does not appear to arise simply because of changes in the shape of the calibration curve, for the measured effectiveness of rod No. 9 at 397°F, and at a critical rod bank position of ~24.5 in., was ~0.35 dollar/in. in comparison with the value ~0.21 dollar/in. (indicated in Fig. 94) at 95°F and at the average position of 24.9 in. However, it may be that the curve is not sufficiently accurate at 24.9 in. to permit one to draw the obvious conclusion.

A discussion of the importance of this temperature dependence in estimations of the reactivity effect of reactor heating is presented in Appendix I, section C-1, above.

3. <u>Activation Plots of Bare Gold Wires in the Core (~412°F; Zero Power; Control Rods Banked at 22.2 in.)</u>

Bare gold wires were positioned vertically in various fuel assemblies in one quadrant of the reference core for "neutron flux" mapping. At a temperature of 412°F, and with the five control rods positioned at 22.2 in., the wires were irradiated at low power. Figure 96A shows the approximate locations of these wires and the numbers assigned to them.

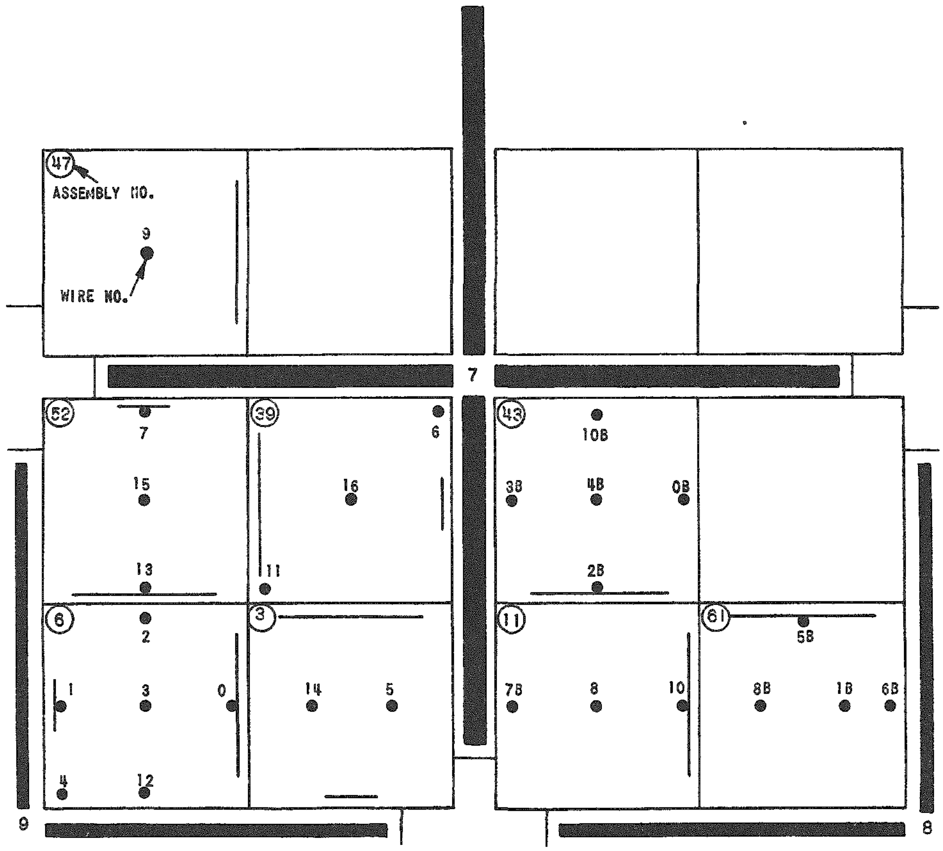


FIG. 96A
POSITIONS OF GOLD WIRES FOR "FLUX PLOT" IN LOADING NO. 57 (~420°F)

(The letter "B" assigned to some of the wires represents the coding used to insure that the wires could not be falsely identified when they were removed from the reactor.) The axial distributions of the gold activity are presented in Figs. 96B, 96C, and 96D. A curve through the datum points from wire No. 0B would almost coincide with the curve shown for wire No. 4B. Similarly, the curve through data from wire No. 8 also fits the normalized activation plot of wire No. 9. The half-life of the decaying gold isotope in question (Au^{198}) is 2.7 days. All counting data have been normalized by correcting for the decay of the gold activity relative to a single, fixed time.

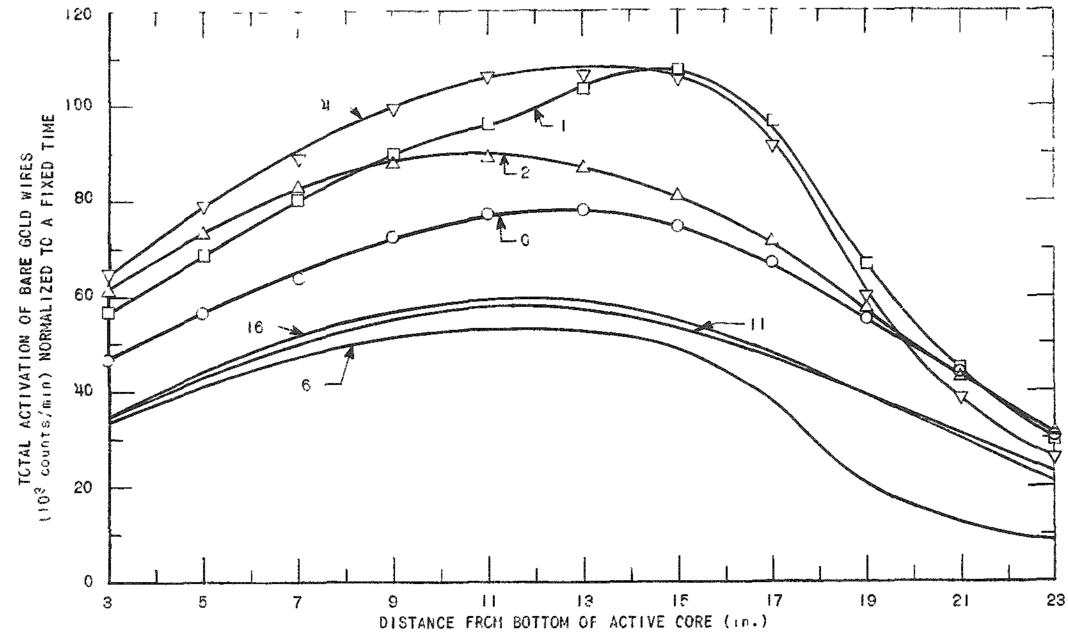


FIG. 96B

AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX"
IN LOADING NO. 57 (~420°F)
(5-ROD BANK AT INDICATED POSITION OF 22.2 in.)

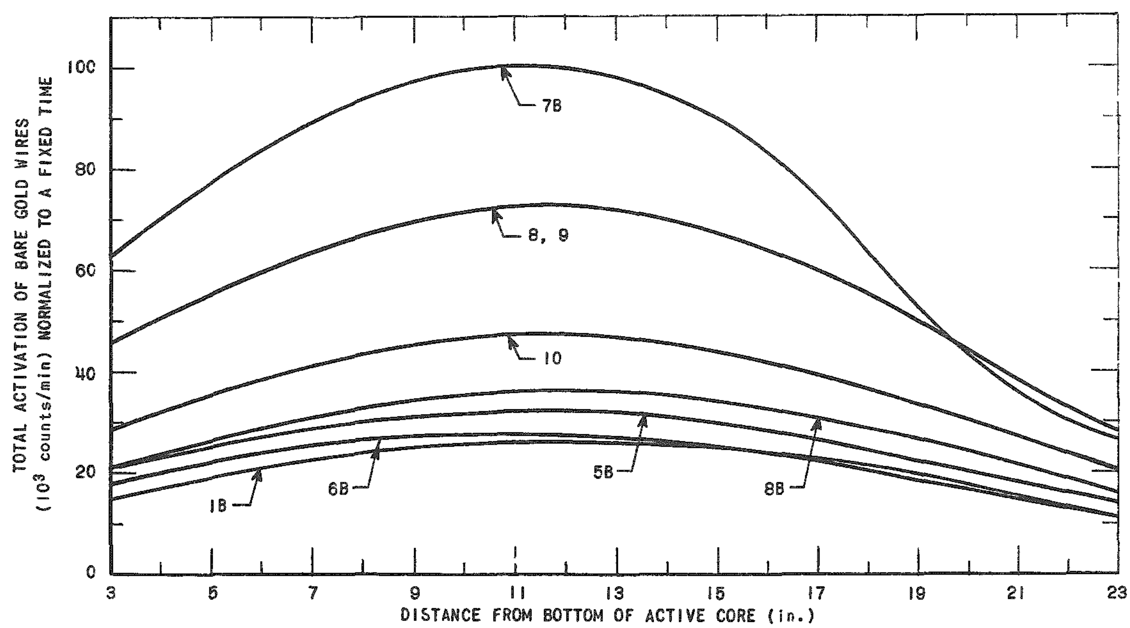


FIG. 96C

AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX"
IN LOADING NO. 57 (~420°F)
(5-ROD BANK AT INDICATED POSITION OF 22.2 in.)

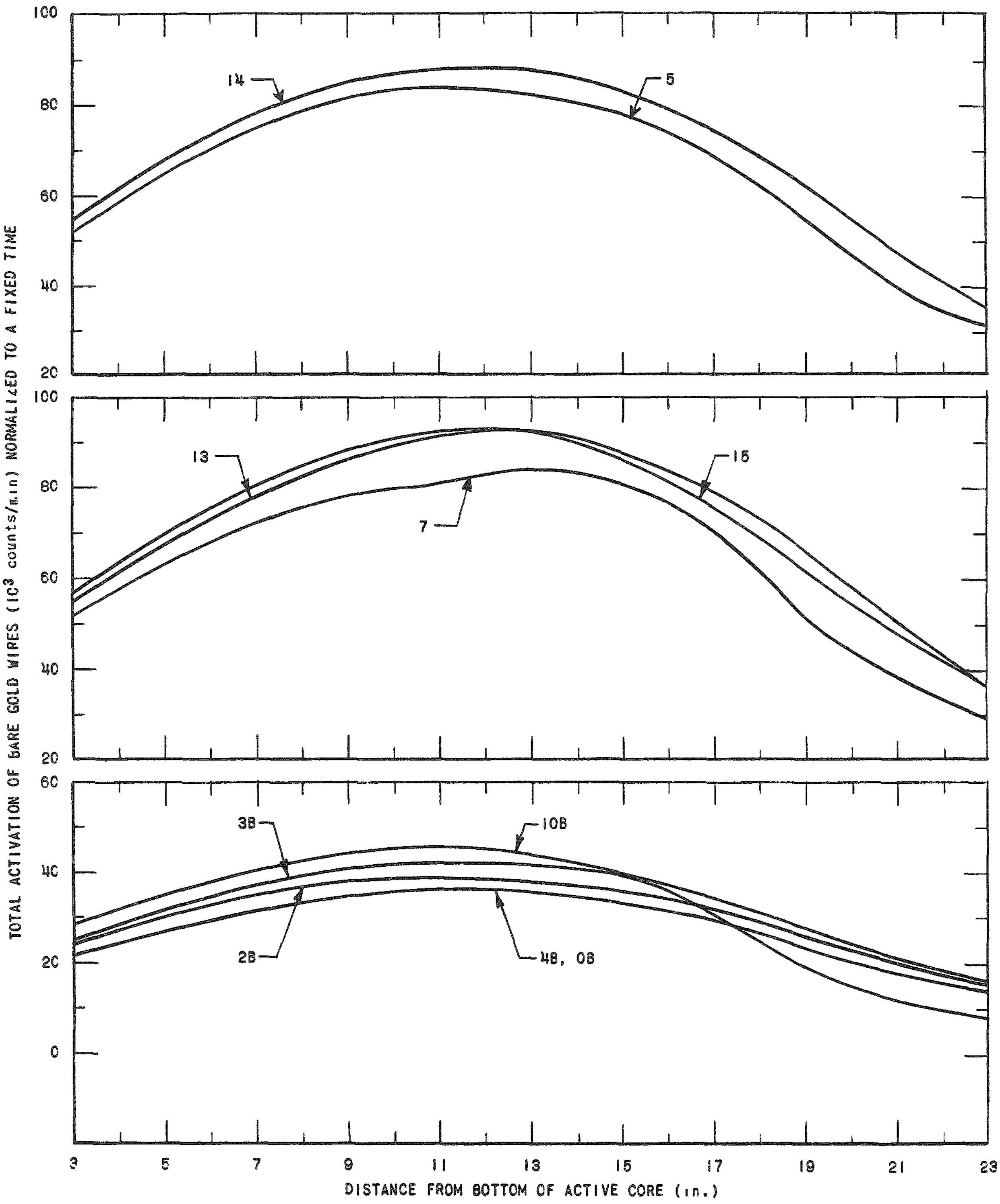


FIG. 96D
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX"
IN LOADING NO. 57 (~420°F)
(5-ROD BANK AT INDICATED POSITION OF 22.2 in.)

4. Buildup to Equilibrium Xenon (Xe^{135}) Concentration

Shortly after the reactor was brought to power for the first time, it was subjected to a sustained period of operation at ~8000 lb steam/hr in an experiment devised to follow the buildup of the Xe^{135} concentration to equilibrium distribution and level. Enough boric acid was added to the reactor water to withdraw the five-rod bank to a critical position of 22.95 in., and the rods were held at this position for approximately 35 hours by incremental dilution of the H_3BO_3 solution. At the beginning of the experiment the reactor was essentially free of xenon, and a sample taken of the reactor water was found to contain 2.61 gm H_3BO_3 per gallon of cold reactor water. At the end of 35 hours, the boric acid concentration was essentially constant, determined to be ~1.35 to 1.40 gm/gal of cold water. The difference, ~1.25 gm/gal, corresponds to an average decrement of ~1 gm/gal in the operating system. If the differential reactivity effect of boric acid measured in the cold reactor may be applied, equilibrium xenon controls ~2.5 dollars, or ~1.7%, at 8000 lb steam/hr. Because of the reduction in fluid density in the coolant channels and in the control rod channels, the relative neutron flux in the boron-containing regions of the core drops, and it is believed that a more reasonable estimate of the reactivity loss to equilibrium xenon is 2.3 dollars, or ~1.6%. At 3 Mwt, the reactivity controlled by equilibrium xenon would be ~2.5 dollars, or ~1.7%. Figure 97 shows the reactivity controlled by xenon as a function of time after operation at 8000 lb steam/hr.

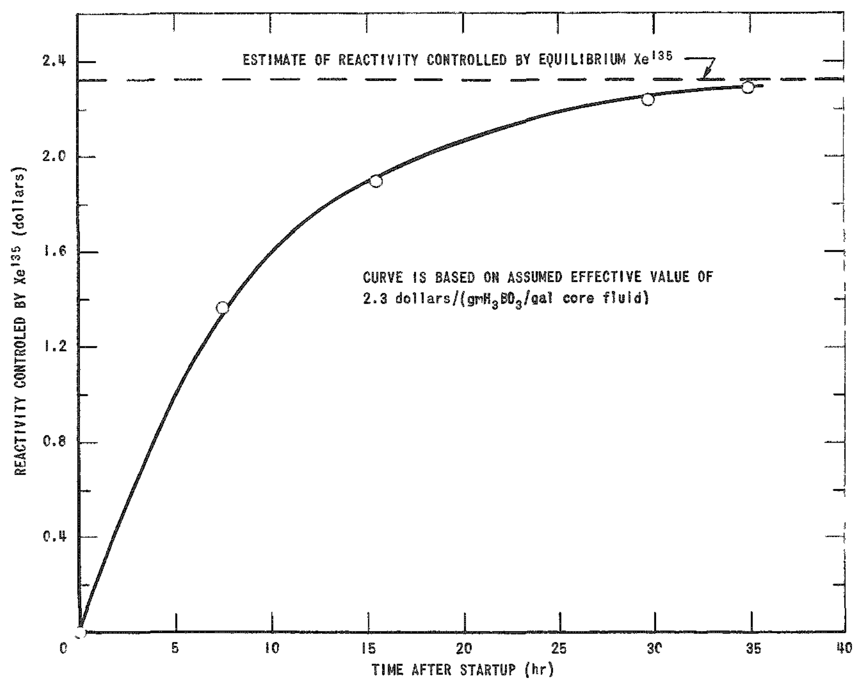


FIG. 97
BUILDUP TO EQUILIBRIUM XENON (Xe^{135}) POISONING IN FRESH REFERENCE
3 Mwt REACTOR OPERATING AT 8,000 lb steam/hr
(5-ROD BANK AT 23.0 in.)

(The experimental data of boric acid concentration are subject to an uncertainty due to the procedure used to sample the reactor water.)

5. Decay of Xenon (Xe^{135}) after Shutdown from Equilibrium Operation at 8000 lb Steam/hr

At the completion of the 500-hour plant performance test, a measurement of the change in the critical position of rod No. 9 was made as a function of time after "shutdown" from equilibrium operation at 8000 lb steam/hr. The power was dropped essentially to zero with respect to the variation of the xenon concentration; the operating pressure of 300 psig was maintained during the shutdown. The data of critical rod position during the first few hours after shutdown are anomalous in that, at first, additional rod control was required (albeit this was a very small effect), and then, with the four off-center rods banked at 19.6 in., rod No. 9 was raised slightly before xenon decay required the lowering of rod No. 9. It is believed that there was a lag period during which steam bubbles adhering to the metal surfaces in the core diffused out of the core. Clearly, the change in xenon concentration was small during the first few hours. It is estimated that maximum xenon poisoning exceeded the equilibrium poisoning level by less than 0.4 dollar (probably by less than 0.3 dollar). The critical position of rod No. 9 as a function of time after shutdown is shown in Fig. 98, starting at $3\frac{1}{2}$ hours after the end of the 500-hr test.

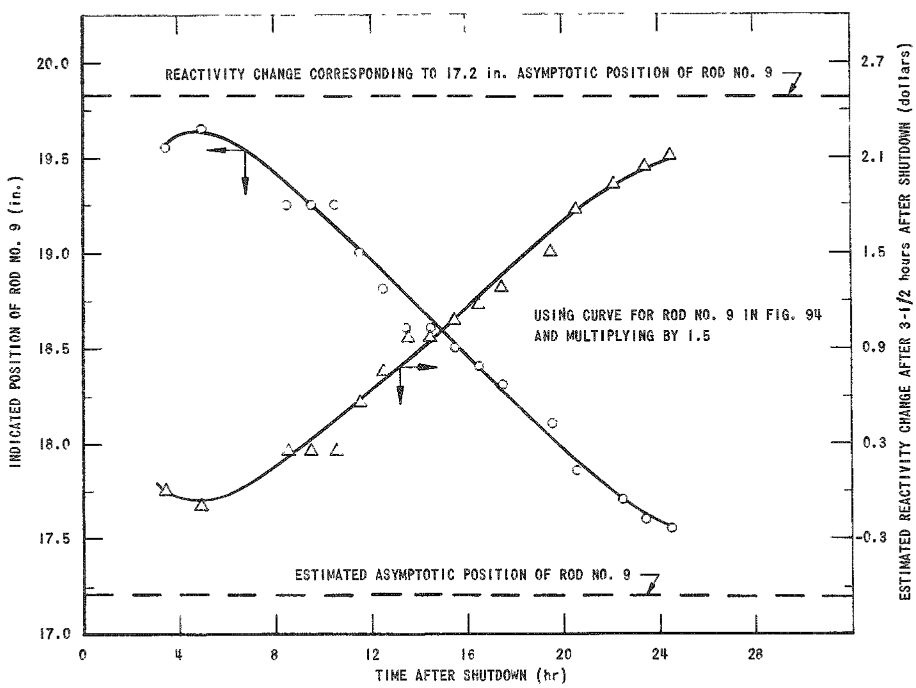


FIG. 98

DECAY OF XENON (Xe^{135}) FOLLOWING SHUTDOWN FROM EQUILIBRIUM OPERATION OF FRESH REFERENCE 3 Mwt REACTOR AT 8,000 lb steam/hr

(CONTROL RODS NO. 1, 3, 5, AND 7 AT 19.6 in.; REACTOR PRESSURE MAINTAINED AT 300 psig)

If one assumes that the shape of the curve for differential reactivity effect of the motion of rod No. 9 from the five-rod bank in the cold reactor is the same in this situation also, where rod No. 9 is lowered past the other cross rods, and if one applies a factor of 1.5 for the ratio of rod "worth" in the hot reactor to that in the cold reactor, the second curve in Fig. 98 is obtained.

6. Control-rod-bank Position as a Function of Reactor Water Temperature ("No" Xenon)

There was no attempt made to determine the position of the five-rod bank at various temperatures in the complete absence of xenon. However, when the reactor was first taken to power, the critical position of the five-rod bank was measured, at essentially zero power, as a function of water temperature. During each incremental change in water temperature, the reactor was operated at ~0.5 Mwt, and the effect of xenon build-up on the position of the rod bank was small; it has not been taken into account in the curve shown in Fig. 99.

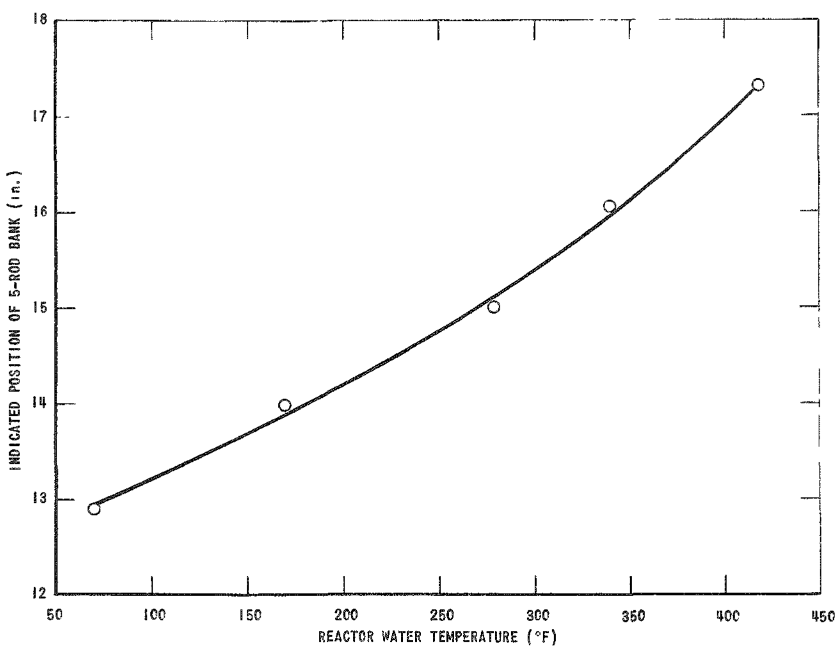


FIG. 99
ESTIMATED POSITION OF 5-ROD BANK IN FRESH REFERENCE 3 Mwt REACTOR VS. REACTOR WATER TEMPERATURE
(ZERO POWER; ESSENTIALLY ZERO XENON)

7. Control Rod Positions vs. Reactor Power Level

After the reactor had been shut down for approximately $2\frac{1}{2}$ days, it was operated again at low power until the operating temperature was attained. Then, within a period of four hours, three series of power calibration experiments were completed. In each case, the four off-center cross control rods were left at a preassigned banked position, and the reactor power level (in lb steam/hr) was measured as a function of the position of rod No. 9. It is estimated that, at the beginning of this four-hour interval, the reactivity controlled by xenon was ~0.3 dollar and that the control in xenon varied by less than 0.5 dollar during the three sets of experiments. The varying bias would be reflected more prominently in the first set, where, with rods No. 1, 3, 5, and 7 at 17.6 in., rod No. 9 was moved to 26 in. at 7900 lb steam/hr; in this region of small differential worth, the indicated position of rod No. 9 is sensitive to the slowly increasing level of Xe^{135}. However, within any one series of calibrations, there was relatively little change in the xenon concentration.

The three calibration curves are shown in Fig. 100 together with the datum points. No corrections have been attempted for the variation in xenon level.

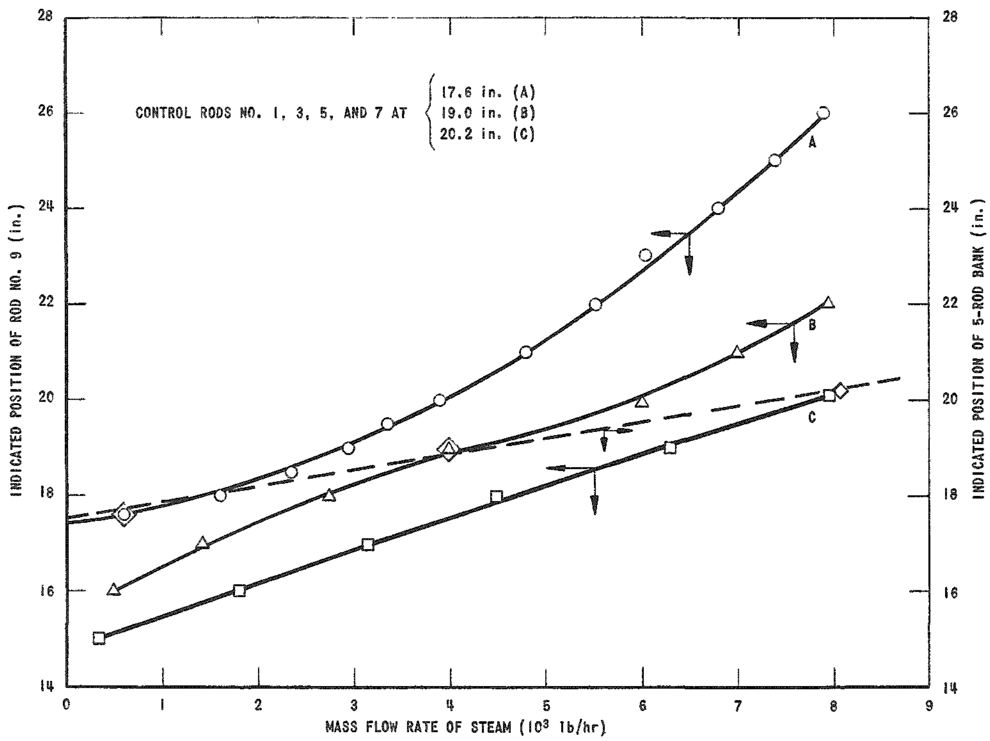


FIG. 100
INDICATED POSITIONS OF CONTROL RODS VS. POWER LEVEL OF FRESH REFERENCE 3 Mwt REACTOR (ESSENTIALLY NO XENON PRESENT)

A fourth curve has been plotted, the position of the 5-rod bank versus steam mass flow rate, using data from the other three curves; in view of the uncertainties of the calibration, a linear fit was chosen to represent these data. The rod-bank position, ~20.2 in., at 8000 lb steam/hr may be compared with the value of ~22.3 in. reported for equilibrium operation at ~8000 lb/hr.[13]

8. Estimate of the Net Excess Reactivity Controlled by Steam Voids[14]

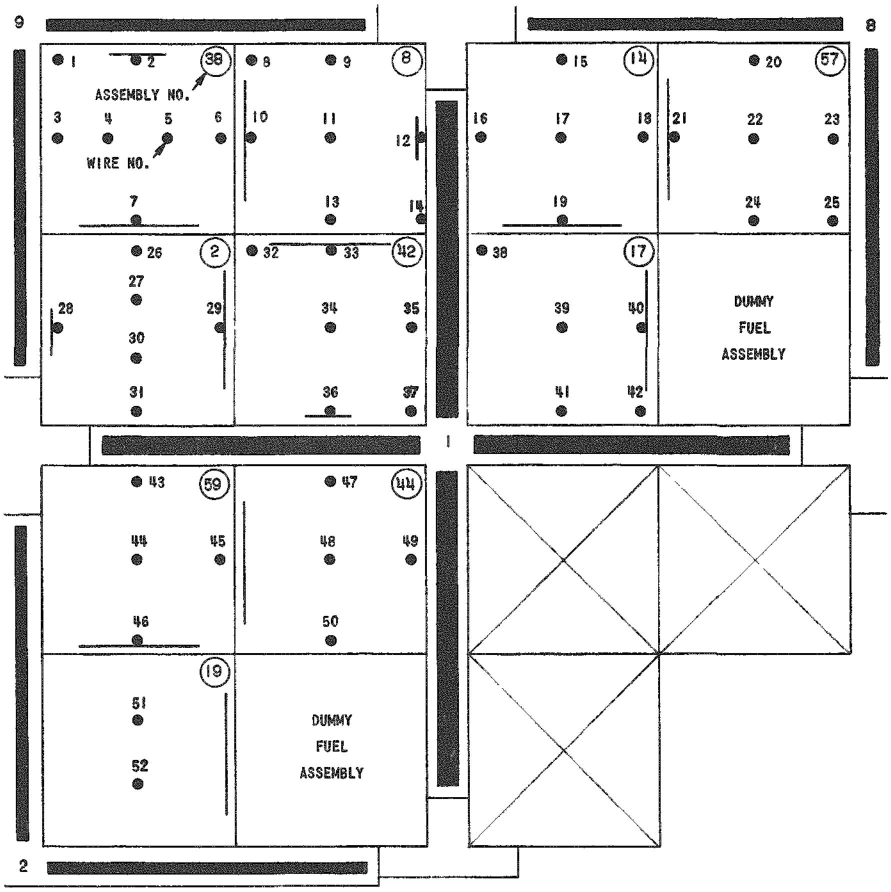
The analysis of any experiment intended to yield the net reactivity loss engendered in the production of steam voids in the reactor is complicated by the presence of control rods, since any change in power results in a change in the reactivity controlled by the rods. Measurement of differential rod "worth" by the positive-period technique is ruled out in the operating reactor because the increase in steam void quickly cancels the positive reactivity effect of control rod withdrawal. Analysis of the data in Fig. 100 would not yield accurate results because the net reactivity available with the control rods at 20.2 in. and with a steam mass flow rate of 8000 lb/hr is not known. Nor, for that matter, is the net reactivity available in the hot system at zero power and with the rods at 17.3 in. known, because of the lack of time to calibrate rods completely in the hot reactor prior to the 500-hr plant performance test or subsequently.[15]

A calibration of reactor power relative to the amount of boric acid in the reactor water was attempted, with rods left in position. Although the measured reductions of the equilibrium rate of flow of steam corresponding to the additions of small amounts of boric acid solution were somewhat inconsistent, an estimate of 3 to 4.4 dollars has been inferred for the reactivity loss due to steam voids at 3 Mwt. The data were not corrected for the change in the reactivity controlled by the rods, arising from the change in power, estimated to lead to an ~10% reduction, i.e., to 2.7 to 4.0 dollars, or ~3.3($\pm$0.6) dollars. The data and the uncertainties of the analysis do not warrant a more reliable estimate.

9. Activation Plots of Bare Nickel-Cobalt Wires in the Core (~420°F; 8000 lb steam/hr; Control Rods Banked at 23.0 in.)

Before the 3-Mwt reference core was taken to power for the first time, 52 bare nickel wires containing ~0.05 w/o cobalt were aligned vertically in fuel assemblies in one quadrant of the core. Shortly after the first, brief testing of the reactor at nontrivial power levels, the reactor was operated at a steam mass flow rate of 8000 lb/hr for approximately forty hours, during which time a five-rod-bank position of 23.0 in. was maintained by varying the concentration of boric acid in the reactor water. (It was then that the data concerning the buildup to equilibrium xenon, shown in Fig. 97, were obtained.) The wires were removed from the reactor at the end of this time, and subsequently they were counted using a single-channel analyzer covering the Co^{60} gamma peak at 1.172 Mev. An n-p reaction with the nickel leads to Co^{58} gamma activity, primarily below 0.87 Mev; it is believed that the discrimination against this activity was satisfactory. One wire (No. 37) broke during preparations for counting, and the data for that position have been omitted.

In Fig. 101A are shown the positions of the wires in a plan schematic of the reactor core quadrant. The axial distributions of the Co^{60} activity are shown in Figs. 101B to 101Q (less letters "I" and "O"), assembly by assembly. Note that the datum points, and hence the labeling of the abscissa, refer to a base point such that 28.5 in. corresponds to the top of the fuel plate. Thus, for example, the distance "10 in." designates a point that is approximately 8.3 in. above the bottom of the active fuel plate.



SHORT LINES REPRESENT HALF-LENGTH $0.4gB^{10}$-STRIPS
LONG LINES REPRESENT FULL-LENGTH $0.5gB^{10}$-STRIPS

FIG. 101A
POSITIONS OF NICKEL-COBALT WIRES FOR "FLUX PLOT" IN THE REFERENCE 3 Mwt CORE
(~420°F; 8,000 lb steam/hr)

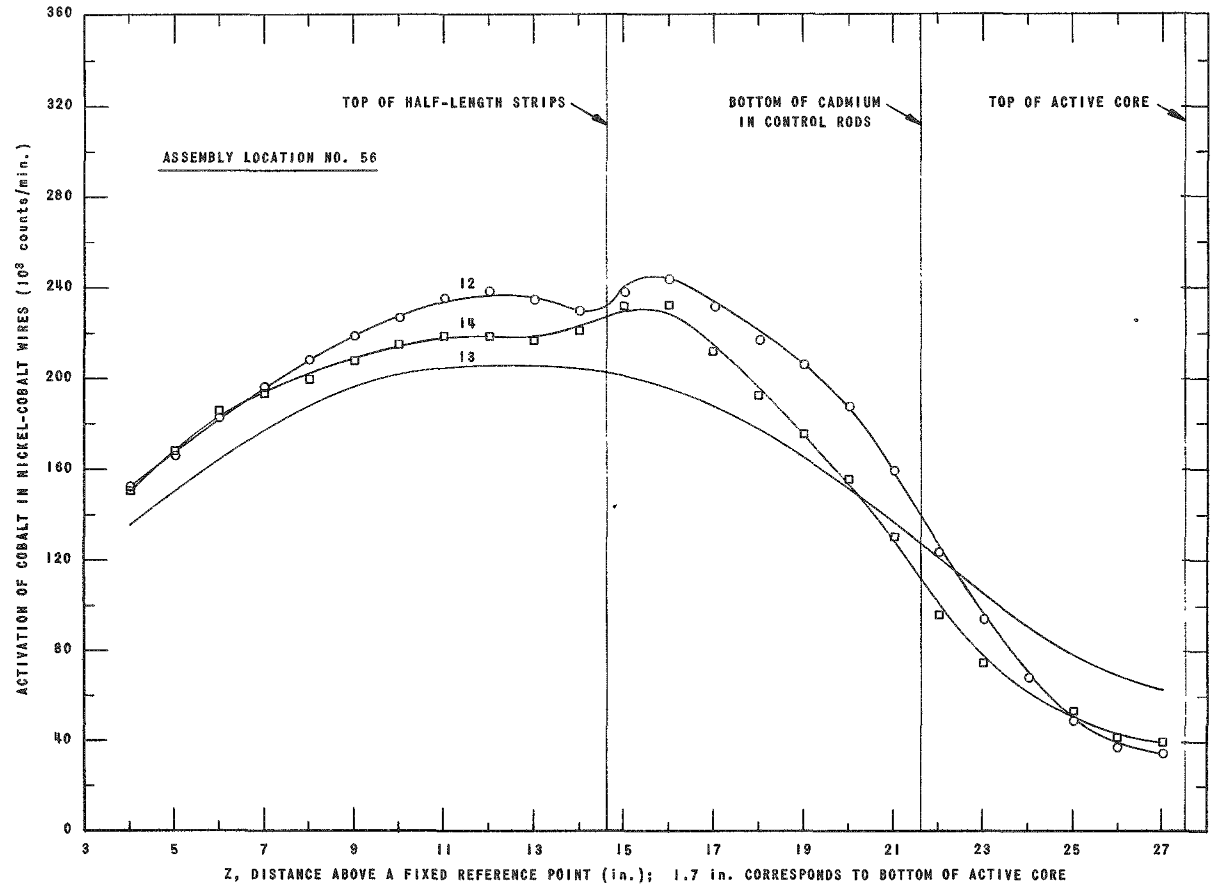


FIG. 101E
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

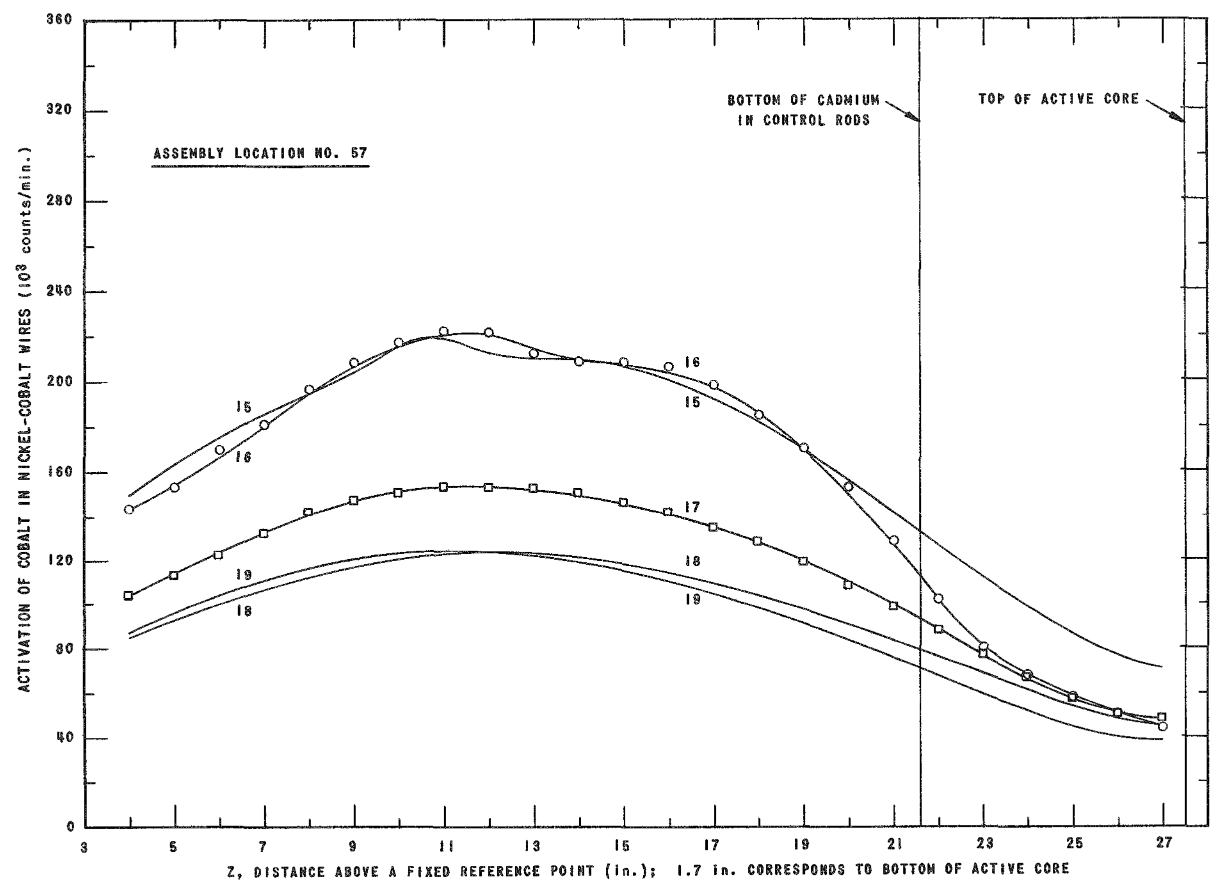


FIG. 101F
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

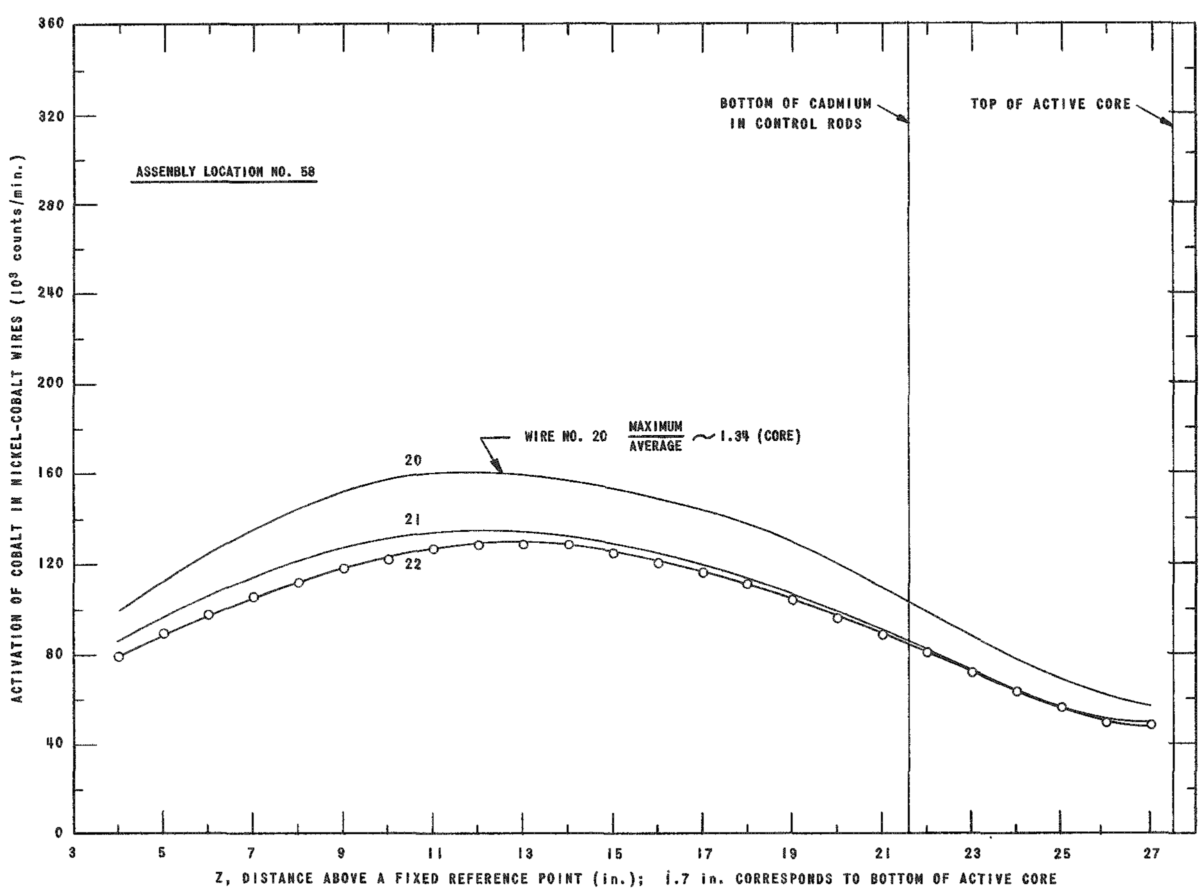


FIG. 101G
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

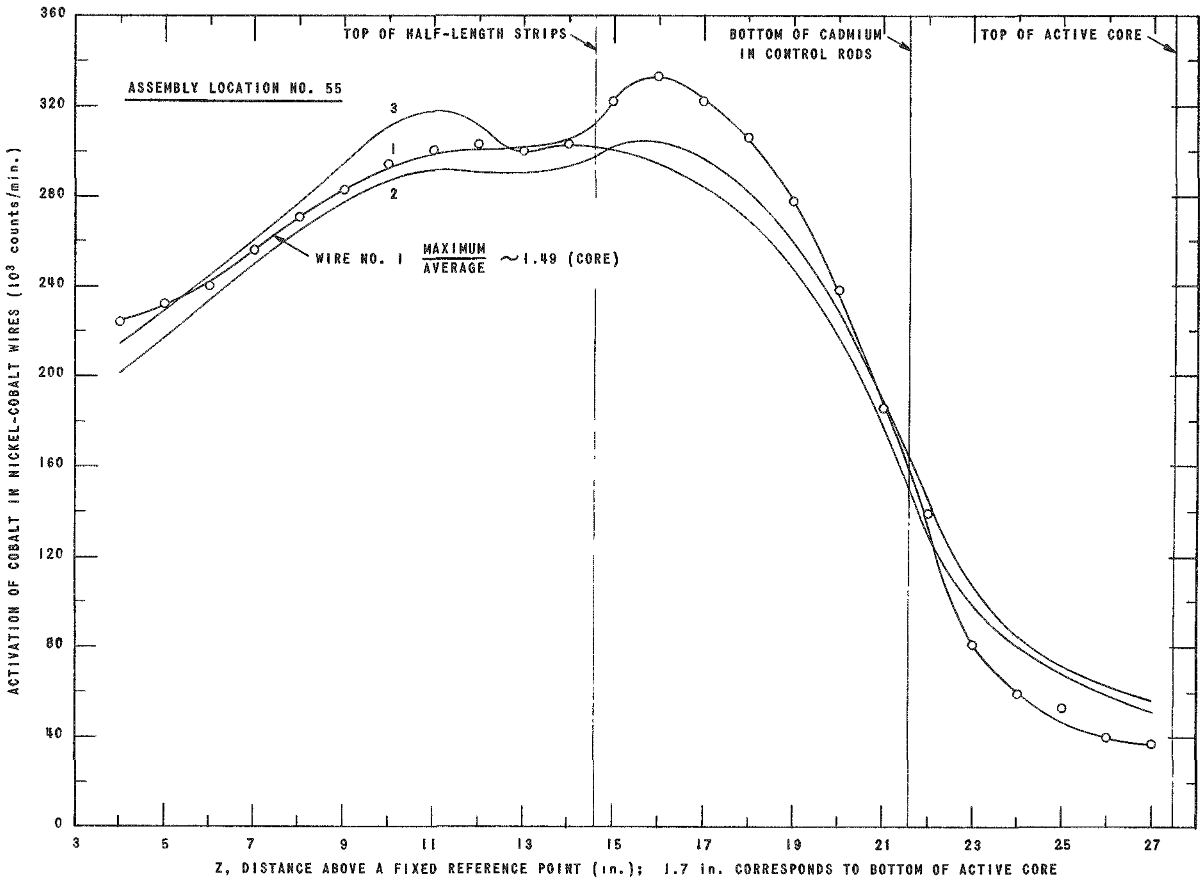


FIG. 101B
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

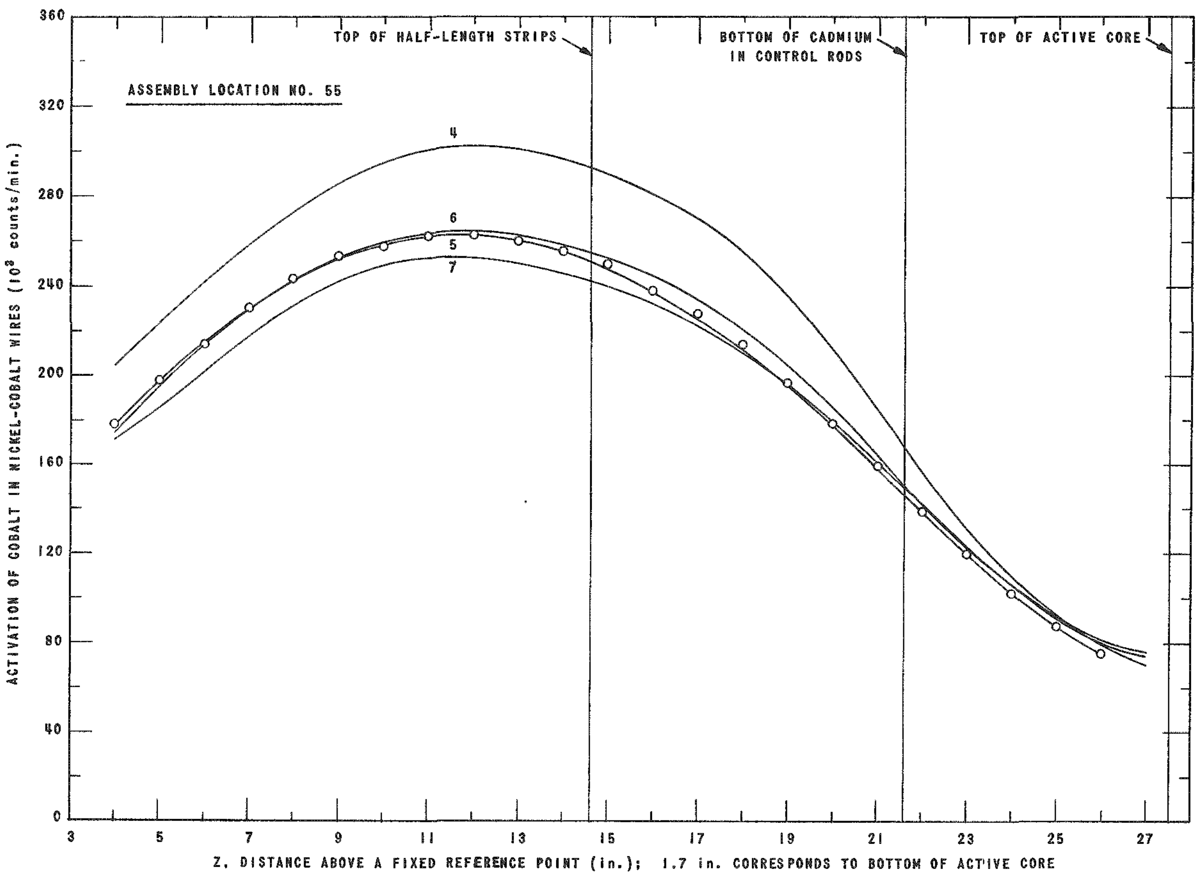


FIG. 101C
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

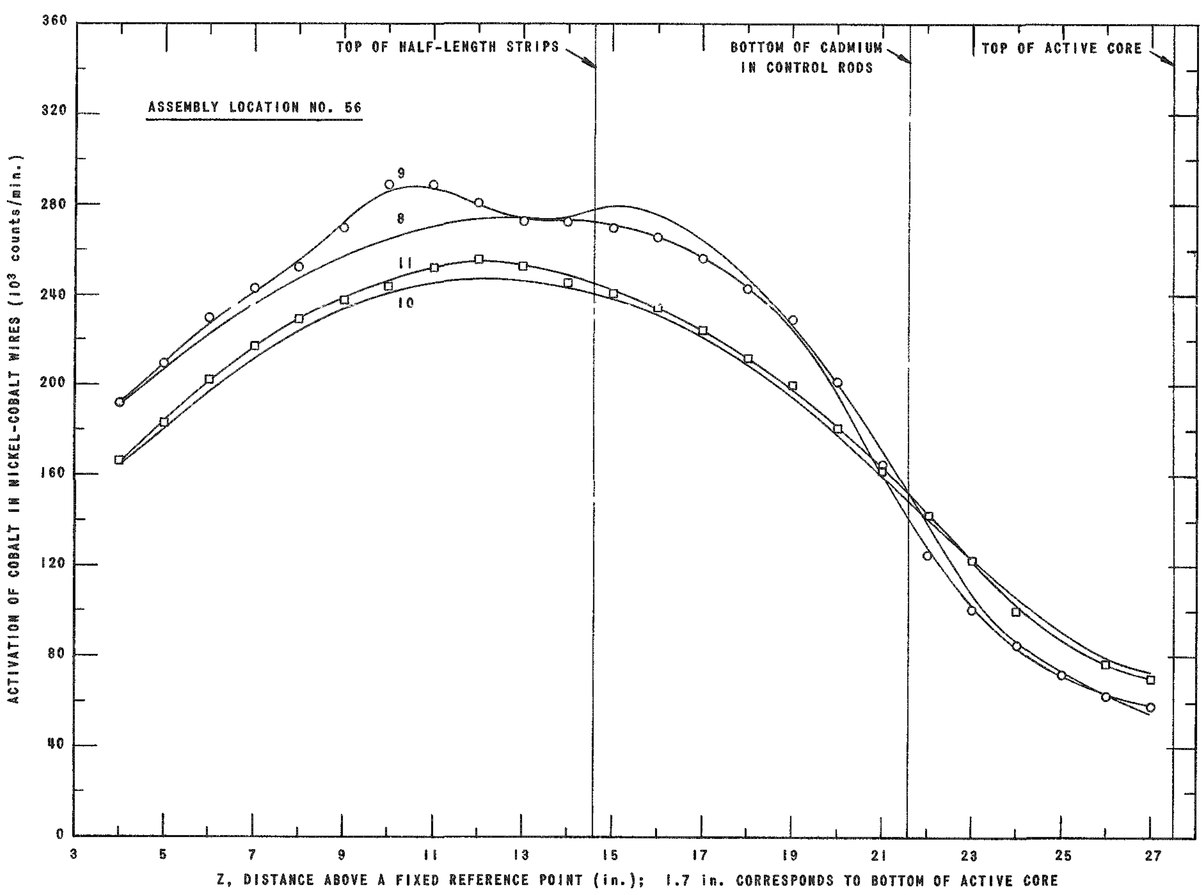


FIG. 101D
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

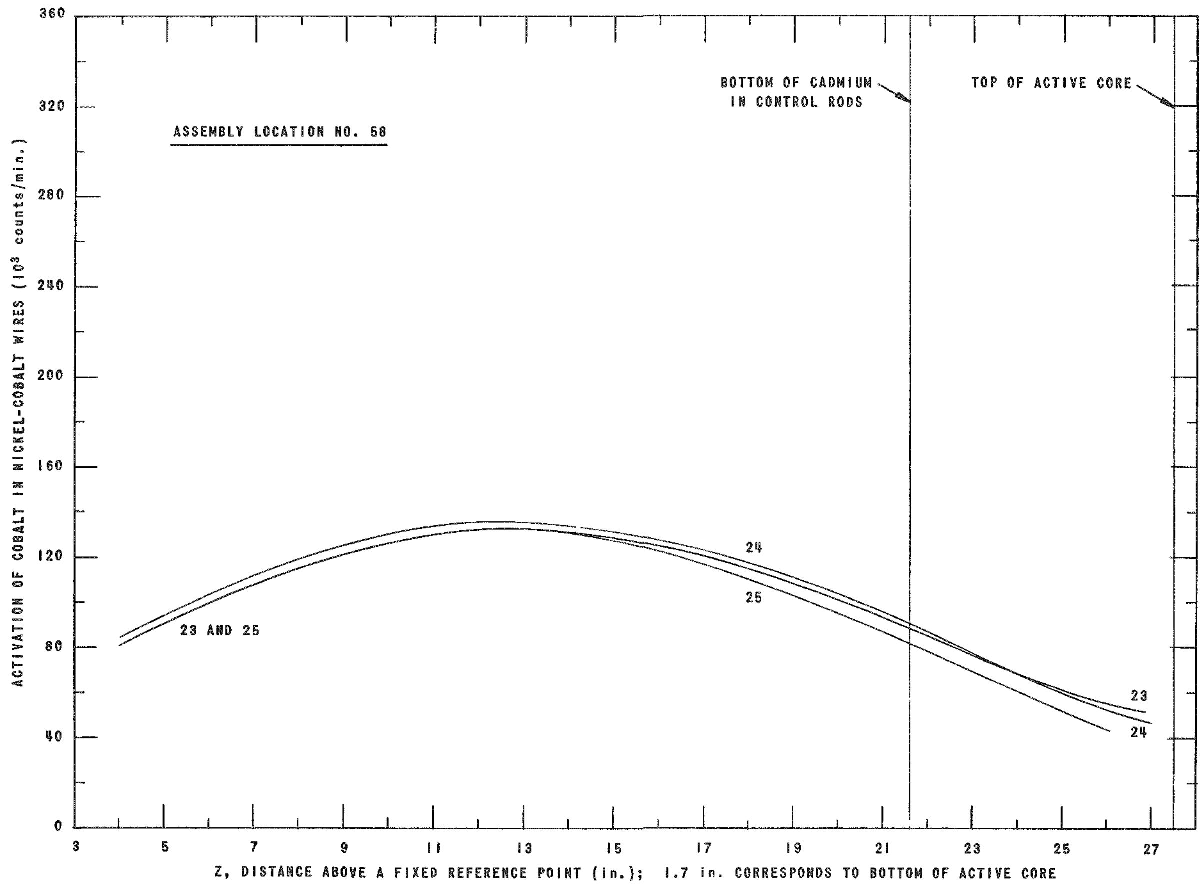


FIG. 101H
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

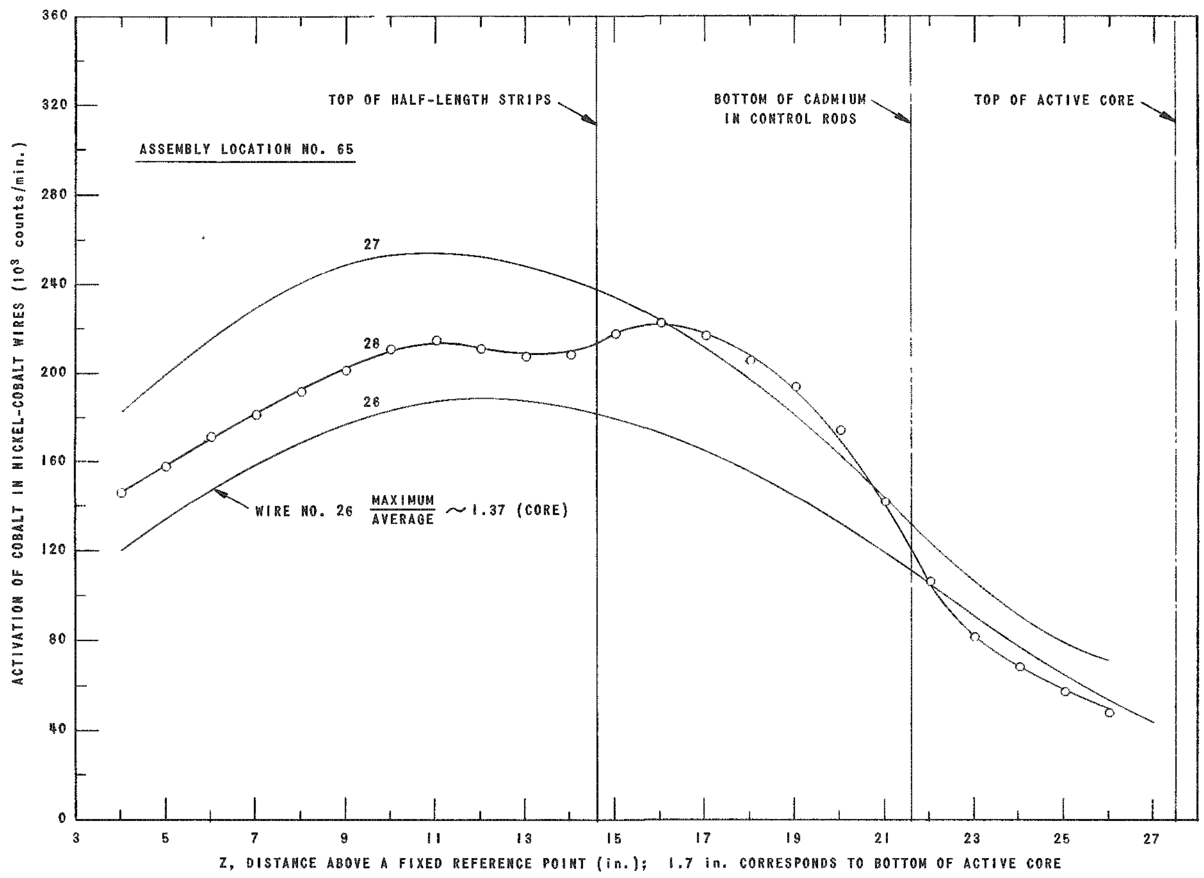


FIG. 101J
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

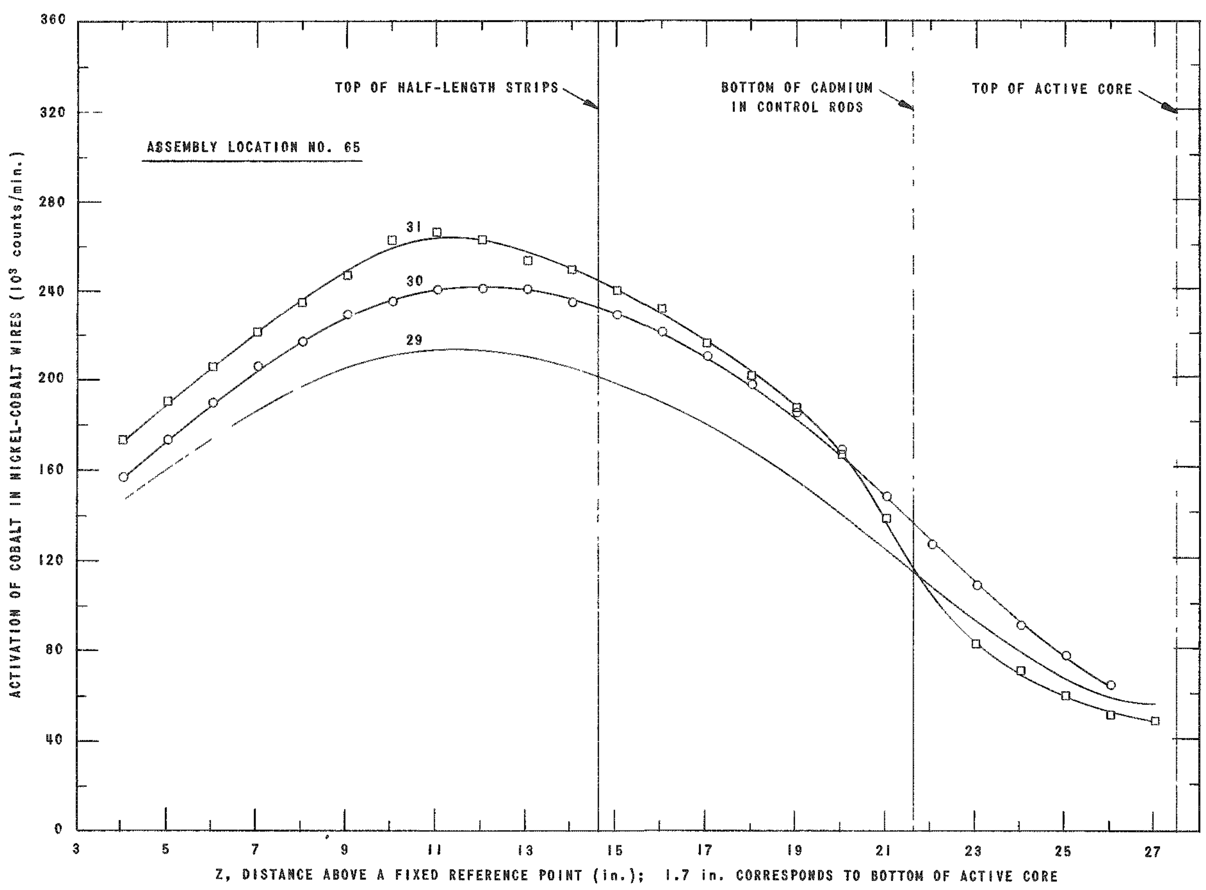


FIG. 101K
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

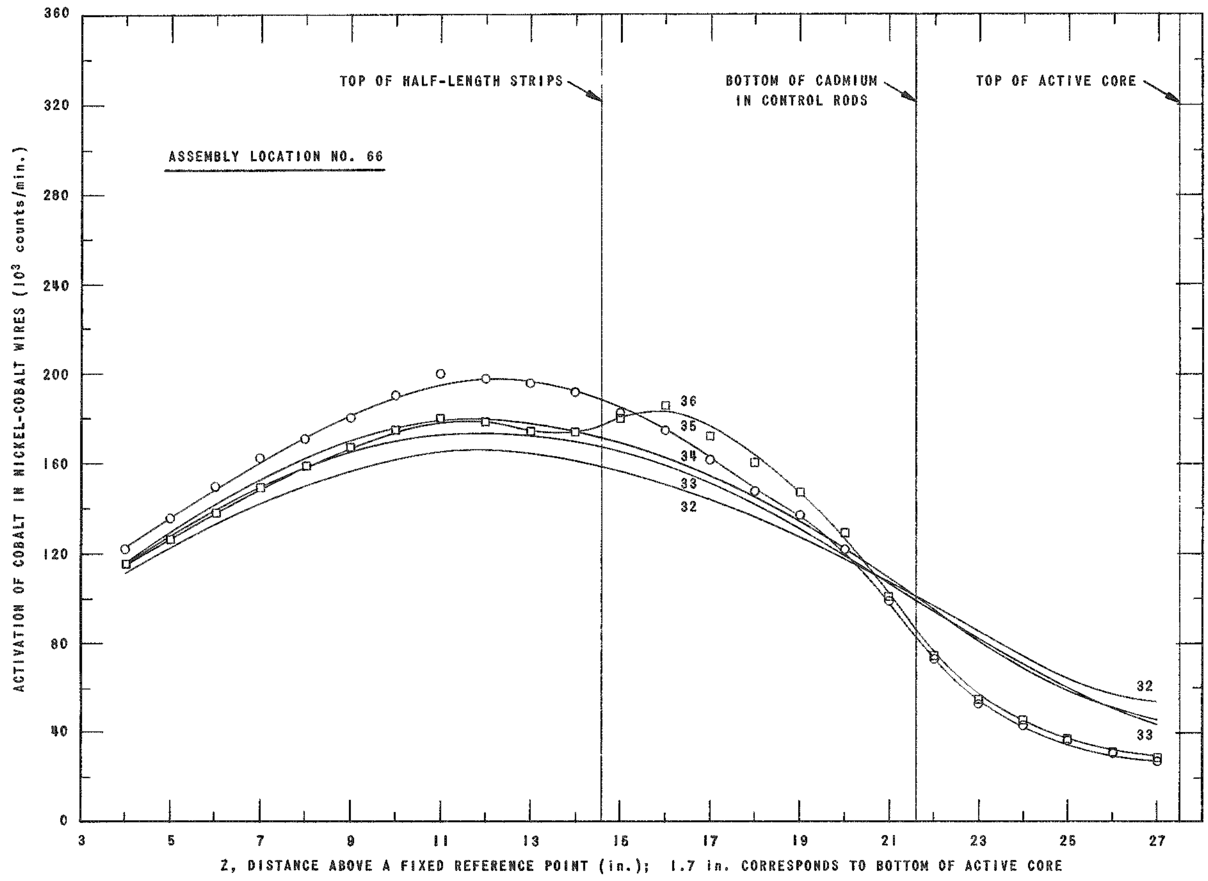


FIG. 101L
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

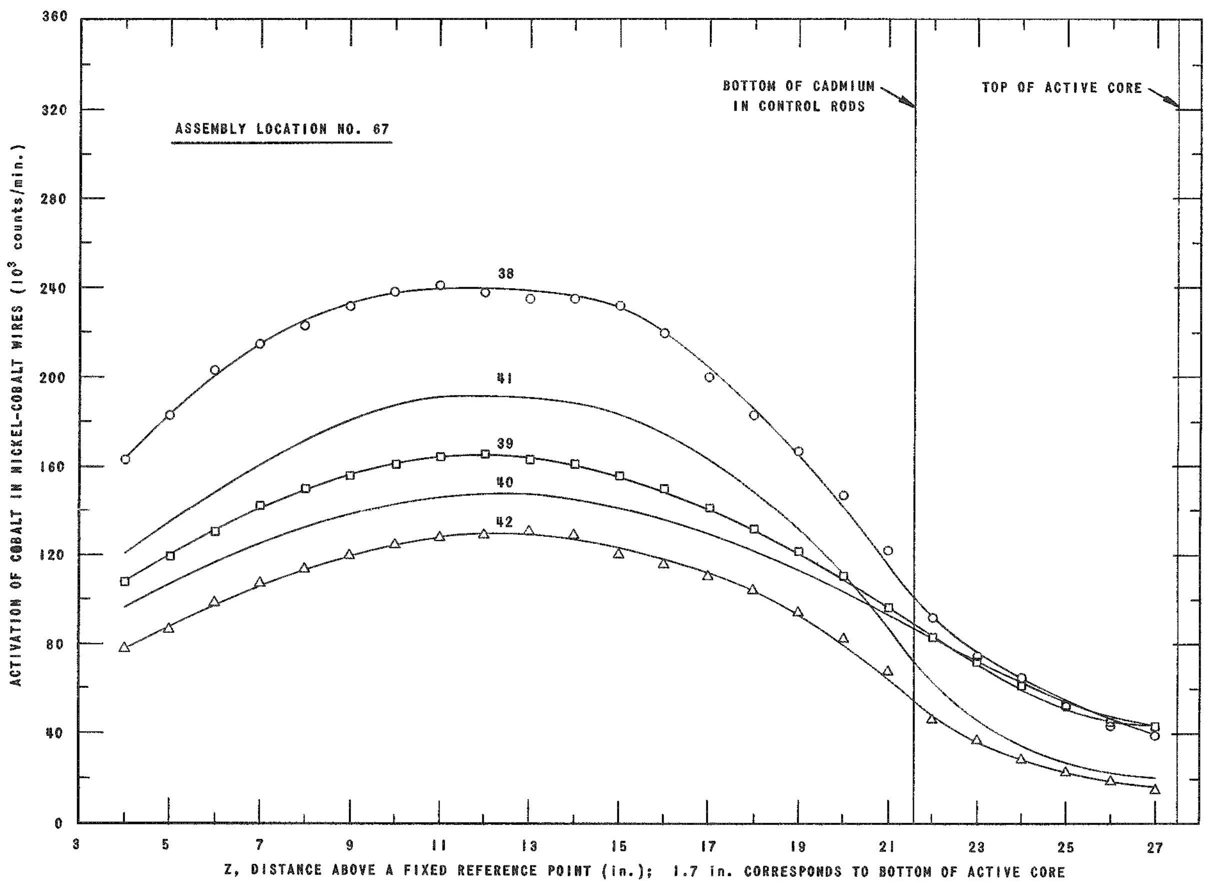


FIG. 101M
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

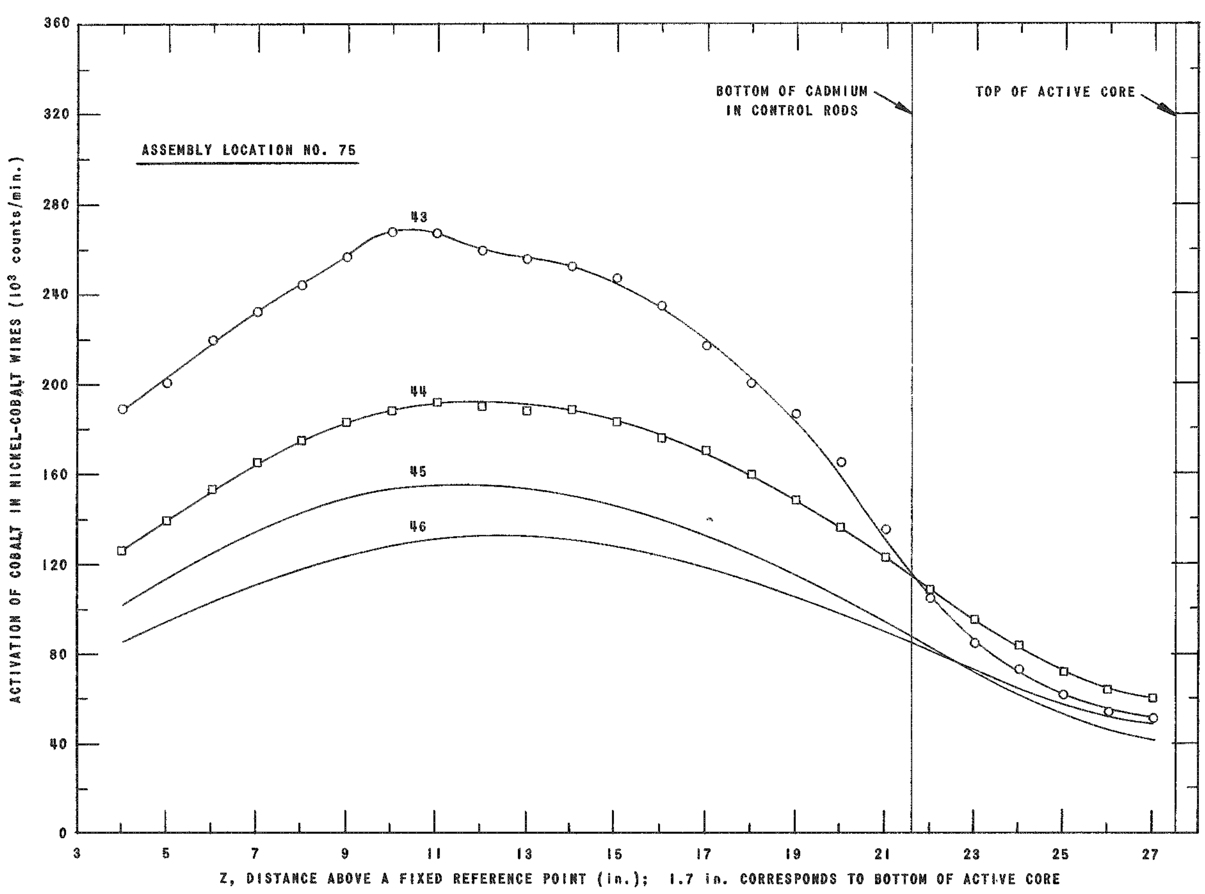


FIG. 101N
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

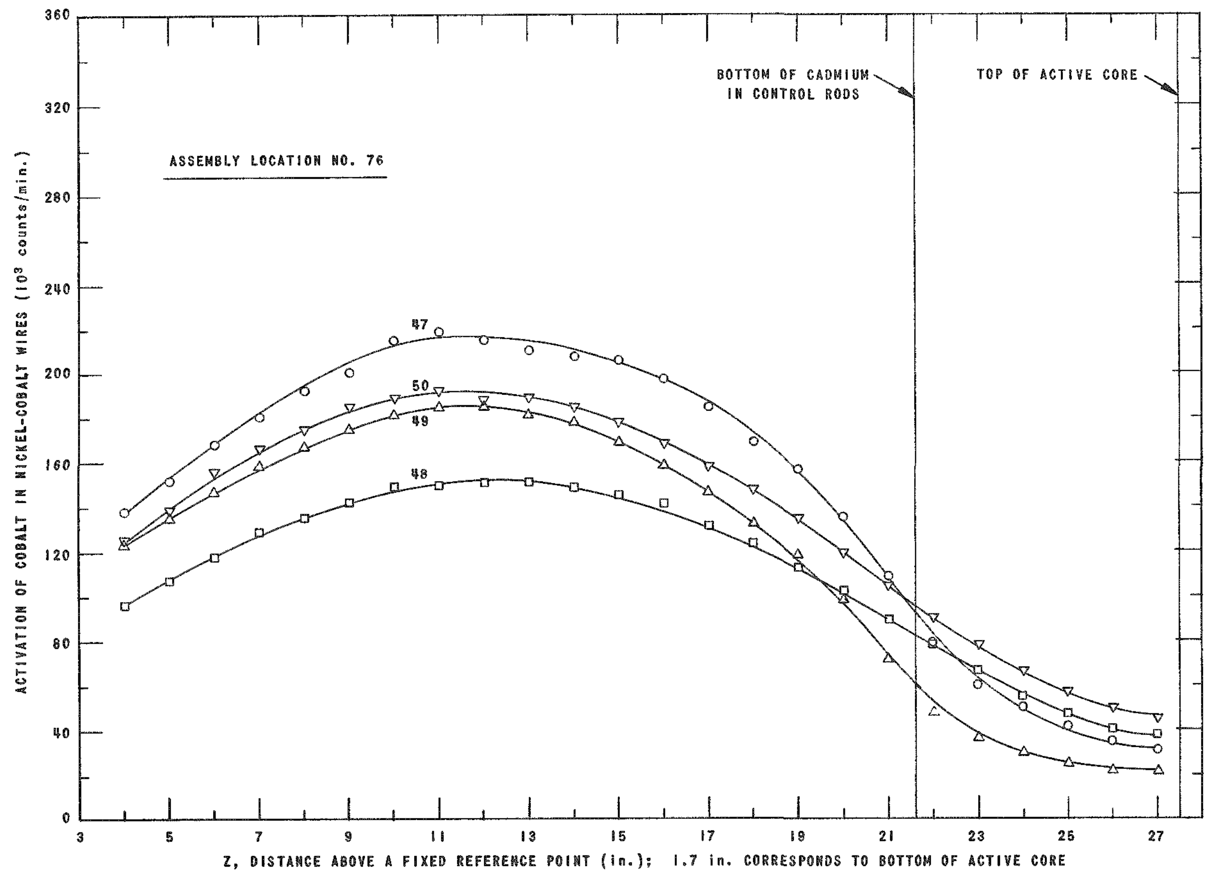


FIG. 101P
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

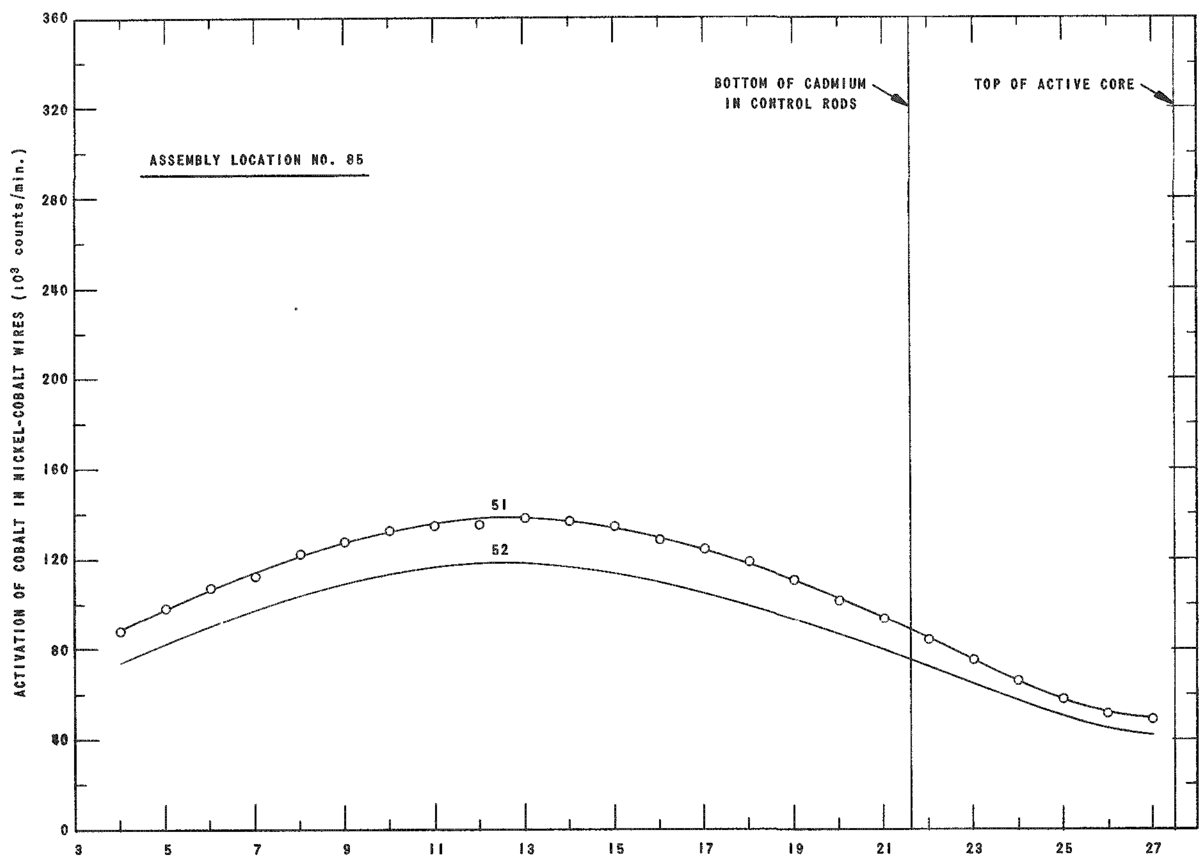


FIG. 101Q
AXIAL DISTRIBUTION OF "THERMAL NEUTRON FLUX" IN OPERATING 3 Mwt REFERENCE REACTOR
(REACTOR PRESSURE 300 psig; 8000 lb steam/hr; 5-ROD BANK AT 23.0 in. INDICATED POSITION)

D. Comparison with Reactor Physics Characteristics of the Initial Reference Forty-fuel-assembly Core (~Loading No. 45)

Since the half-length burnable-poison strips are less gray, and are positioned adjacent to control rod channels, in the bottom half of the core, these strips are subject to more rapid burnup in the higher thermal neutron fluxes encountered in these regions than are the full-length strips. Consequently, it is of some interest to point up the distortion of the curves of differential reactivity effect of rod motion from the 5-rod bank occasioned by the insertion of the half-length strips in what is otherwise the initial reference core of forty fuel assemblies. The corresponding differential curve for rod No. 9 in loading No. 19 (or, equivalently, in loading No. 45) at room temperature is presented in Fig. 102.

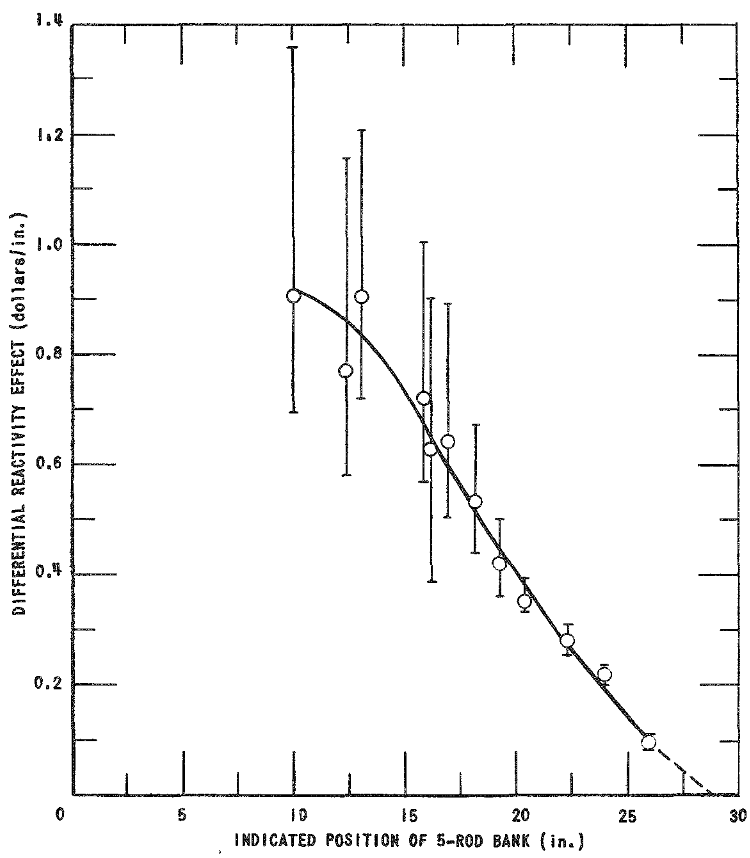


FIG. 102
DIFFERENTIAL REACTIVITY EFFECT OF MOVING CONTROL ROD NO. 9 FROM 5-ROD BANK (LOADING NO. 45 AT ROOM TEMPERATURE)

The curve for the bank of rods Nos. 1, 3, 5, and 7, moving from the 5-rod bank, is so similar in shape and in magnitude that it is not presented. By a comparison of Fig. 102 with Fig. 94, it may be seen that the half-length strips acted to reduce the effectiveness of control rod No. 9 and to shift the position of the peak differential reactivity effect. The off-center cross rods were affected also, but not so markedly.

Bare gold wires were activated in the hot (~420°F) reactor at essentially zero power. The axial distributions of the Au^{198} activity are presented in the report on the zero-power experiments with various ALPR core loadings.[38]

Appendix II

REACTOR THERMAL ANALYSIS*

The reactor has been designed for a thermal output of 3000 kw, corresponding to a steam flow rate of 9020 lb/hr. The actual maximum steam flow rate depends on condenser vacuum and, in turn, on ambient conditions of barometric pressure and temperature. Typically, at an actual steam flow rate of 8350 lb/hr, the following set of plant conditions were obtained:

	Observed	Design
Main steam pressure	304 psig saturated	300 psig
Temperature hotwell	143°F	130°F
Turbine load	310 kw	360 kw
Simulated heat load	412 kw	400 kw
Air to condenser	34°F	60°F
Air flow	69250 SCFM	86000 SCFM
Barometric pressure	24.86 in. Hg abs	sea level
Condenser pressure	8 in. Hg abs	4.5 in Hg abs

For comparison, the design conditions are also shown The difference between the design rate of flow of steam and the actual observed rate was due (1) to the difference in barometric pressure, causing a lower air mass flow rate, and (2) the lower turbine load, imposed by poor condenser vacuum (see section IX - B).

Feedwater at 175°F enters the reactor vessel from a spray ring located on a plane approximately level with the top of the core (see Fig. 23). The incoming water mixes with the circulating water in the downcomer, with the result that the downcomer water is subcooled of the order of 1.5 to 2°F. This subcooled water enters the reactor fuel channels where its temperature is raised to saturation in the first 25% of the core height, according to calculations. Thus, the full-load steam generation of 8350-9020 lb/hr occurs approximately within the upper 75% of the core.

The difference in the density of the mixture of heated water and steam within the core and that of the subcooled water within the downcomer produces a driving head responsible for the natural circulation within the core.

*A. Smaardyk

In the heat transfer calculations, a simplified model was assumed in which heat generation took place uniformly over the entire core. Further assumptions made in the evaluation are as follows:

1. Control rods are fully withdrawn.

2. Steam voids are not carried over into the downcomer.

3. Uniform distributed mixing occurs between incoming feedwater and recirculating water.

4. Uniform water temperature and velocity at the inlet to the fuel channels.

The procedure for calculating the heat transfer characteristics was then applied as follows:(41)

1. Assume an exit void fraction α_e which is the ratio of vapor phase flow area to total flow area at the exit of the fuel channels.

2. Solve the flow equation $\Sigma \Delta p = 0$ for all pressure drops within and over the core and around the downcomer.

3. Using an experimental value of slip ratio, calculate the steam quality from the equation

$$x = \frac{1}{1 + \left[\frac{V_f}{V_g}\right]\left[\frac{\rho_f}{\rho_g}\right]\left[\frac{1}{\alpha} - 1\right]},$$

where

x - steam quality, ratio of vapor flow rate to total mixture flow rate

V_f - specific volume of saturated liquid

V_g - specific volume of saturated vapor

ρ_f - density of saturated liquid

ρ_g - density of saturated vapor

α - steam volume fraction (ft^3 steam/ft^3 mixture).

4. Obtain the reactor power and subcooling by substituting known values of steam quality and feedwater inlet enthalpy h_0 in the equation

$$Q_t = W_t\ [(h_f - h_{in}) + Xh_{fg}] = W_g\ [(h_f - h_o) + h_{fg}] \quad ,$$

where

h_f - enthalpy of saturated liquid

h_{in} - enthalpy of subcooled water

h_{fg} - latent heat of vaporization

W_t - flow rate, total

Q_t - heat rate, total

W_g - steam flow rate.

The calculations were made for one case without a chimney or riser above the core and for another case with the benefit of a riser section. For purposes of comparison, the results are shown in Fig. 103.

Actually, a riser section is formed by the control rod channels except for approximately two-inch-wide vertical gaps between these channels (see Fig. 25). Likely, considerable portions of the circulating water bypass the rise section through these gaps. Also, bypassing is possible through eight vacant fuel assembly spaces located within the riser geometry. These spaces have been earmarked for insertion of fuel assemblies in a 59-assembly core. Presumably, the actual reactor performance lies between the values shown for the two cases in Fig. 103.

Other important heat transfer parameters used in, or concluded from the calculations are summarized in Table 31.

The fuel and clad temperatures were calculated using the published[42] boiling heat transfer data. For the fuel plate, the averaged temperature gradient through the metal was calculated to be of the order of 1°F. The temperature difference (10°F) of the fuel and clad was estimated for an arbitrarily assumed scale coefficient of $h_s \sim 2000$. This estimate was considered conservative for ALPR fuel environment of high-purity water.

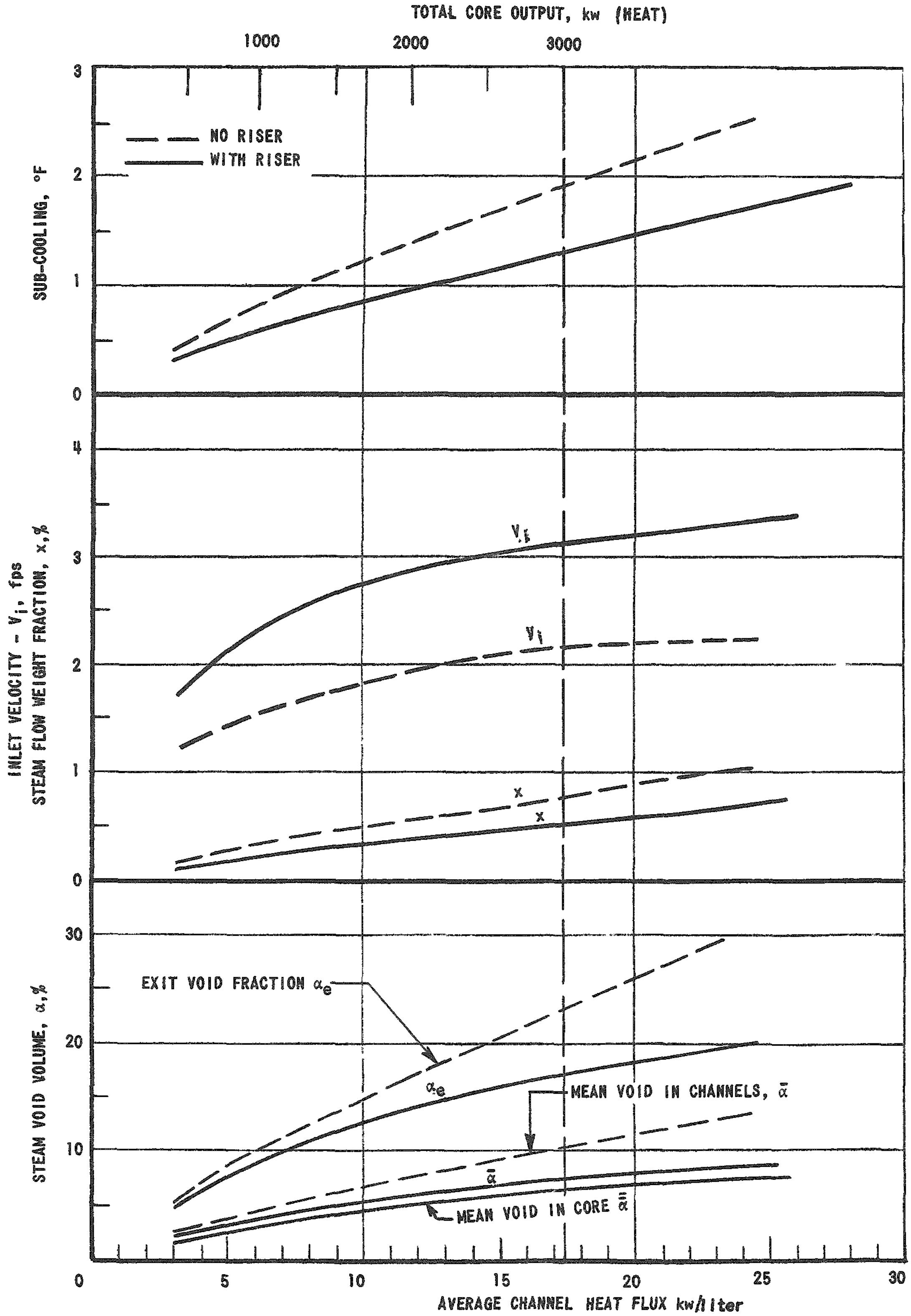


FIG. 103
BOILING PARAMETERS, 40 FUEL ASSEMBLIES
(175°F FEED-WATER)

Table 31

SUMMARY OF HEAT TRANSFER CHARACTERISTICS FOR 40-FUEL ASSEMBLY REFERENCE CORE AT 3 Mwt

Steam Flow	9,020 lb/hr
Average Heat Flux	21,500 Btu/(hr)(ft^2)
Maximum Heat Flux	75,500 Btu/(hr)(ft^2)
Average Temperature at Center Line of Fuel Plate	450°F
Maximum Temperature at Center Line of Fuel Plate	470°F
Average Temperature at Surface of Fuel Plate Cladding	440°F
Maximum Temperature at Surface of Fuel Plate Cladding	460°F
Moderator Temperature	421°F
Feedwater Inlet Temperature	175°F
Total Heat Transfer Area	475 ft^2
Average Boiling Length	20 in.
Average Power Density in Core Coolant	17.5 kw/liter
Coolant Velocity in Fuel Channels	2.2 ft/sec at entrance of core at full power without riser. 3.2 ft/sec at entrance of core at full power with riser
Average Steam Void in Fuel Assembly Channel	~10% without riser ~8% with riser
Average Steam Void in Core Moderator	~8% without riser ~6% with riser
Exit Steam Quality	~0.8% without riser ~0.6% with riser
Subcooling at Core Inlet	1.5 to 2°F
Water Recirculation Ratio (lb water per lb steam)	130

Appendix III

BIOLOGICAL GRAVEL SHIELDING CALCULATIONS*

A. Constants

In order to make the shielding calculations it was necessary to determine a set of appropriate nuclear parameters for the gravel. Analysis was made of several samples, and a typical composition was estimated. The important elemental constituents are given in Table 32.

Table 32

REFERENCE GRAVEL ELEMENTAL COMPOSITION AND CROSS SECTIONS

Element	Atoms/cc	Atomic Wt (approx)	[Atoms/cc] (Atomic Wt)	Removal σ_r (barns)	Removal Σ_r (cm^{-1})	Absorption σ_a (barns)	Absorption Σ_a (cm^{-1})
Fe	0.000078 × 10^{24}	56	0.00437 × 10^{24}	2.0	0.000156	2.53	0.000197
H		1		1.0		0.33	
O	0.0358	16	0.5728	1.0	0.0358	0.0002	0.0000716
Mg	0.00012	24	0.00288	1.2	0.000144	0.063	0.0000076
Ca	0.003	40	0.120	1.5	0.0045	0.43	0.00129
Si	0.012	28	0.336	1.[illegible]	0.0144	0.13	0.00156
Al	0.004	27	0.108	1.[illegible]	0.0048	0.23	0.00092
Na	0.00024	23	0.00552	1.0	0.00024	0.51	0.000122
K	0.0002	39	0.0078	1.1	0.00022	2.0	0.0004
Ti	0.0001	48	0.004[illegible]	[illegible]	0.0002	5.6	0.00056
Totals			1.162 × 10^{24} (a)		0.0606		0.00513

(a) The density of the reference gravel is as follows: $\frac{1.162 \times 10^{24}}{[illegible] \times 10^{[illegible]}}$ = 1.936 [illegible]

Constants for the mixture were then arrived at by simple summation of the contributions of the constituents. Constants were calculated for gravel(43) having a density of 120 lb/ft³. The resulting parameters chosen are given in Table 33, where μ_t is the total gamma-ray attenuation coefficient, μ_e is the energy absorption coefficient, and N (E) is the effective number of secondary gamma rays born at energy E per neutron capture.

Table 33

REFERENCE GRAVEL CONSTANTS FOR GAMMA-RAY CALCULATIONS

Gamma-ray Energy (Mev)	μ_t(cm^{-1})	μ_e(cm^{-1})	N (E)
1	0.1237	0.1113	-
2	0.0877	0.079	-
2.5	0.0770	0.069	0.64
4	0.0588	0.053	0.83
6	0.0520	0.047	0.7
8	0.0404	0.036	0.063

*A. D. Rossin

Actual test showed the prototype gravel to weight about 105 lb/ft^3. If one were to recalculate the shield, constants could be corrected simply by the ratio of the densities.

B. Experimental Determination of Flux in Gravel

In response to a request from the Army Reactors Branch, an experimental facility (Fig. 104) was added to the prototype in the belief that some quantitative data on gravel as a shielding material could be obtained.

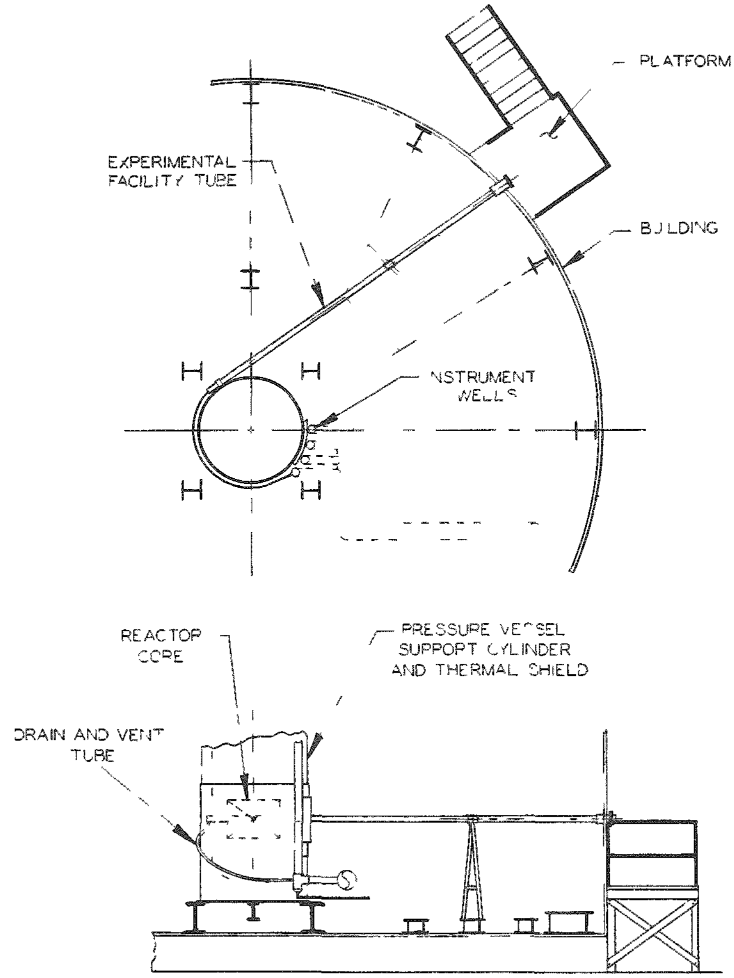


FIG 104
REACTOR BIOLOGICAL SHIELD EXPERIMENTAL FACILITY

A 2-in.-ID aluminum pipe extends from the outside wall of the plant building to a point tangent to the reactor external thermal shield and at the height of the reactor core midplane. The tube is normally kept filled with seven plugs. These are cans, each 29 in. long, filled with shield gravel, thus approximating the nuclear characteristics of the biological shield. Figure 105 shows the relation of the plugs and the distance in the experimental hole to the actual distance from the core centerline.

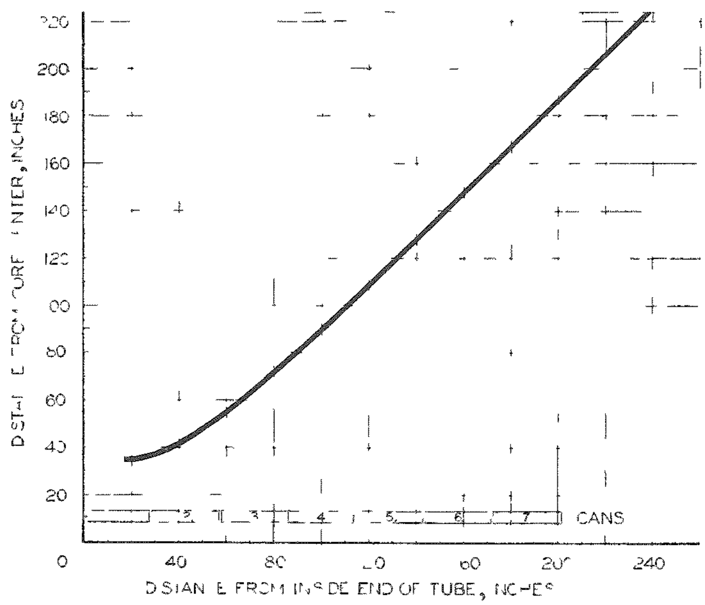


FIG 105
CAN LOCATIONS IN EXPERIMENTAL HOLE

Gold foils were attached to the plugs and irradiated over a short period of operation. During the Plant Performance Test(8) another set of foils was irradiated to saturation, permitting a better absolute flux determination. The results are shown in Fig. 106. When the count rate obtained from the cadmium-covered foils is subtracted from the bare-foil rate, the difference should give the thermal neutron activations. The flux thus measured is compared with calculated thermal neutron flux in Fig. 107. The calibration was based on a similar foil irradiated in the Laboratory's Standard Pile.

It is recommended that further consideration be given to possible experiments to be done in the experimental hole. Other foil materials may be used, and the plugs make it possible to irradiate samples of gravel or other materials. Detailed activation analysis of the gravel could give information on the long-lived activities which may be important for major maintenance operations, should such be necessary deep in the shield.

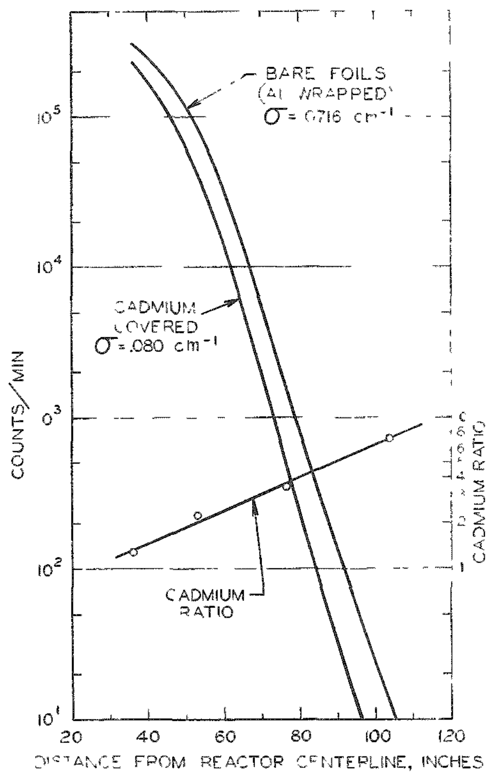


FIG 106
GRAVEL SHIELD
GOLD FOIL FLUX TRAVERSE IN EXPERIMENTAL HOLE

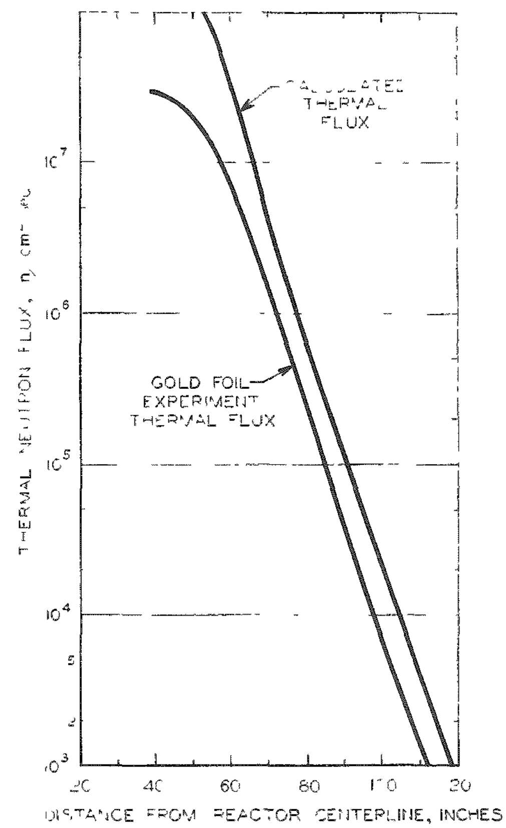


FIG 107
THERMAL NEUTRON FLUXES IN GRAVEL

Appendix IV

ALUMINUM-NICKEL ALLOY CORROSION TESTING*

During the period of initial design of the ALPR, a new aluminum alloy had been developed by the Laboratory's Metallurgy Division. By adding 1 w/o nickel to a standard type 1100 aluminum alloy, it was possible to avoid the disasterous intergranular attack at elevated temperatures that normally occurs in the type 1100 alloy.(35) Although evaluation of the new aluminum-nickel alloy, type X8001, was just beginning, it appeared that it might be useful for the ALPR operating conditions (fuel plate surface temperature of 460°F and coolant temperature of 421°F). Thus Reactor Engineering Division began a series of tests to determine the value of the material for this reactor.(4)

In the first test (to study the effect of boiling on the corrosion of X8001), a $\frac{3}{8}$-in.-OD stainless steel tube clad with X8001 aluminum was partially enclosed in a small pressure vessel. The vessel and tube were mounted vertically, and the tube was mechanically sealed to the vessel at both ends where it extended beyond the vessel. The vessel was partially filled with high-purity water. By passing 500-550°F water from a dynamic loop through the interior of the tube, boiling was obtained on the exterior of the tube. The heat flux decreased as the water temperature inside the tube decreased along the length of the tube, with a maximum value of 25,000 Btu/(hr)(ft^2) near the bottom of the tube. A bulk water temperature of 420°F was maintained by varying automatically the flow of cooling water in a separate tube in the steam zone of the vessel. A small amount of water from the vessel was bypassed through a cation resin exchange bed for a period of about three hours every working day. Thus a water resistivity from 130,000 to 520,000 ohm-cm and a pH from 6.5 to 7.5 were maintained.

Three tubes were tested in this manner for varying lengths of time. The oxide formed was removed by placing the tube in a saturated boric acid solution and passing a high-voltage ac from the sample to a stainless steel electrode. It was then possible to determine metal loss by weight difference. The results of tests on these samples, the longest of which was tested for 3700 hours, indicated a corrosion rate of 3.3 mils per year.(44)

Operation of another boiling corrosion rig was started during the previous test. This time X8001-clad cartridge heaters were used (see Fig. 108). The rig was designed to hold 24 samples at one time and was electrically wired in a manner so that one set of heaters could operate

*N. R. Grant

at full power [23,000 Btu/(hr)(ft^2)], one set at half power, one set at quarter power, and one set at zero power. Testing was again at a bulk water temperature of 420°F, but continuous flow was maintained through a mixed bed ion exchange resin by natural convection. This method produced water of high purity (over 1 megohm-cm) at all times. Temperature control of the rig was similar to that for the previous test.

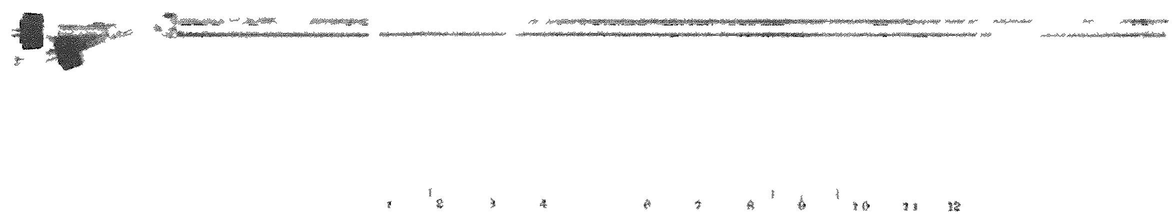


FIG. 108
X8001 CLAD BOILING CORROSION CARTRIDGE

Samples were withdrawn at intervals and the cladding removed. Weight-loss determinations were made by using an iodine-methanol solution to dissolve the aluminum and measuring the amount of oxide remaining. The longest test time was a little over one year. The rates of metal loss varied from 1.8 mils/yr for the full-power samples to 0.36 mil/yr for the zero-power samples. Most of the difference in weight loss may be attributed to the difference in surface temperature of the aluminum (see Fig. 109).(44)

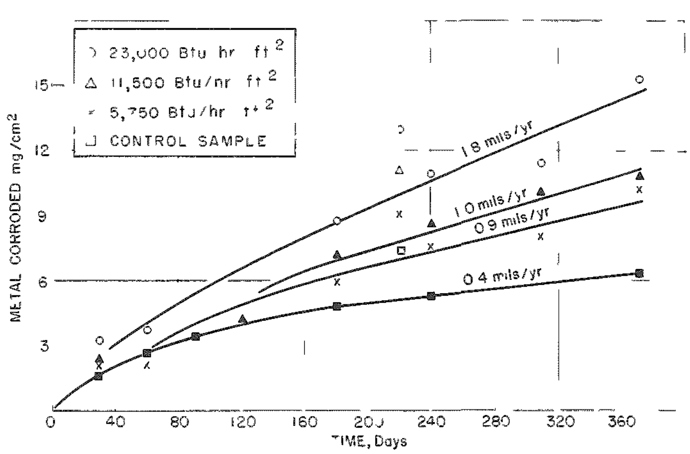


FIG 109
EFFECTS OF BOILING AND HEAT THROUGHPUT ON AQUEOUS CORROSION ON ALLOY X8001 AT 420°F

Various static autoclave tests were conducted at 420°F concurrently with the above boiling tests.(45) A static test was conducted on sandwiches of cadmium between two plates of X8001 aluminum, a structure that simulated a proposed control rod. Holes were drilled in the cadmium and the aluminum plates were spot welded together. No bonding around the edges was made in order to simulate the worst conditions of exposure of the cadmium to the high-purity water. Two types of fastening

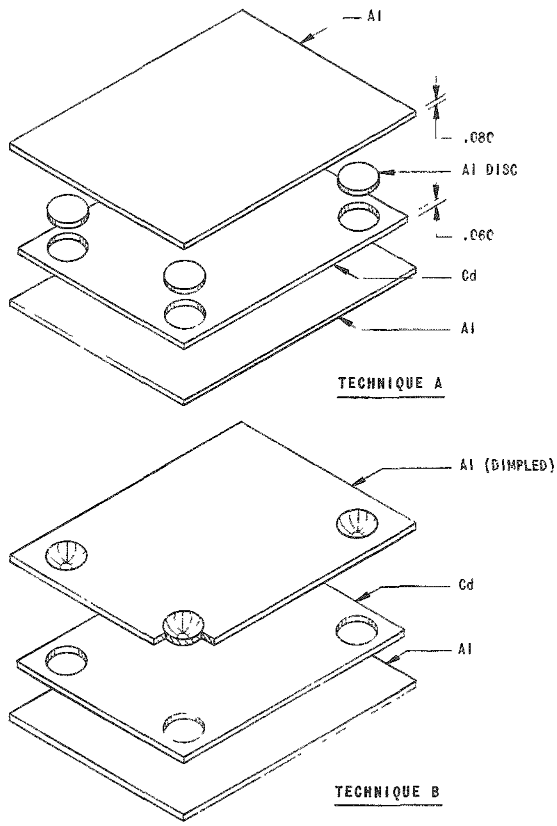
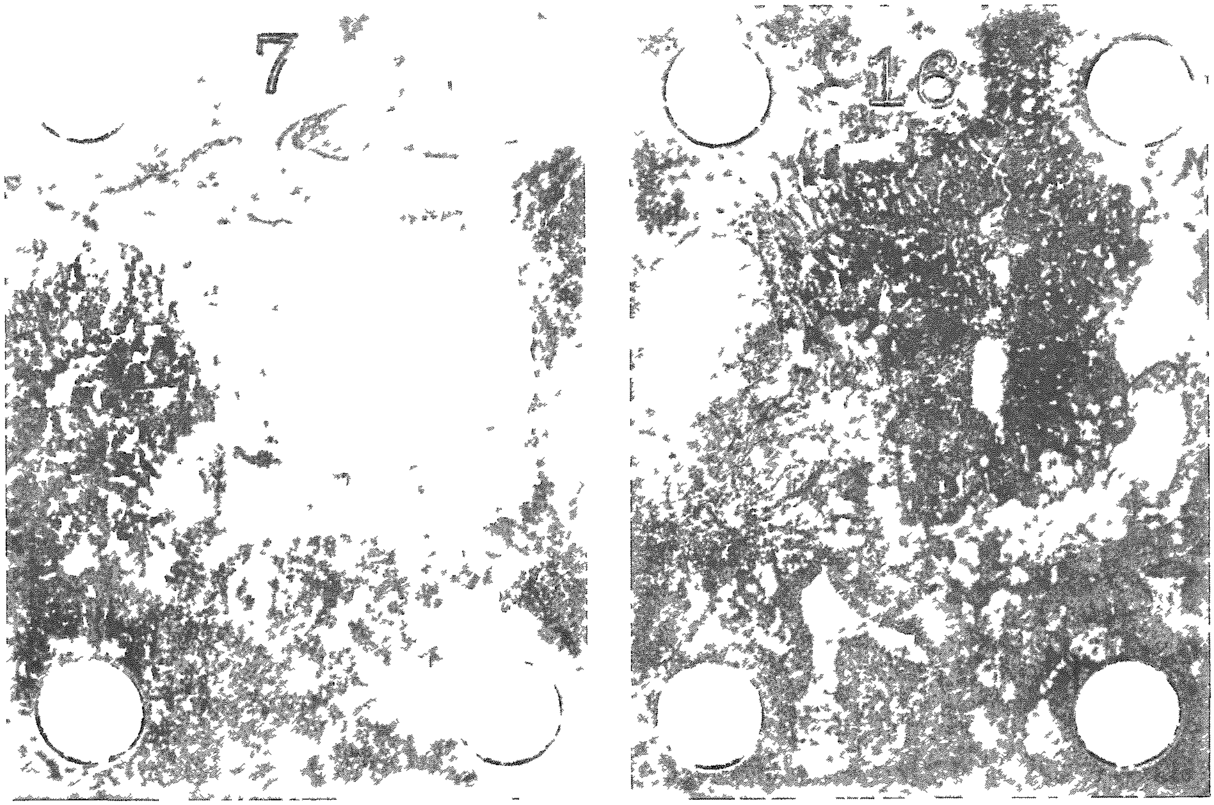


FIG. 110
PROPOSED TECHNIQUES FOR CONTROL ROD CONSTRUCTION

were tested (see Fig. 110): (1) one of the aluminum plates was dimpled so that it contacted the other during welding (technique B), and (2) an aluminum disc was used to separate the two plates (technique A). One sandwich of each type was removed from test every two weeks for a total of 18 weeks. The water from each period was chemically analyzed for cadmium. After the test the sandwiches were separated (see Fig. 111), the oxide dissolved from the surface of the cadmium, and weight losses determined. Unfortunately, the weight losses were extremely scattered and no rate could be established. However, it can be stated that the two methods yielded approximately the same weight losses and that most of the samples, regardless of time, had a weight loss between 0.6 and 1.2 mg/cm^2. Chemical analysis showed 2 to 5 micrograms cadmium per ml of water. The erratic behavior of the results may be due to the variation in how tightly the sandwiches were pressed together, thereby changing the amount of water that could circulate between the cadmium and the aluminum. Technique B was used for the manufacture of the control rods (see Fig. 27) because it is less expensive than technique A and produces a more reliable weld.

Autoclave tests at 550°F were made on fuel plates (powdered metallurgy process), obtained from the Sylvania Corp., some of which were known to be unbonded and some which had drilled holes for intentional defects. All but one of these plates had blistered to some extent by the end of about three months of testing. When the Metallurgy Division began making fuel plates (rolled ingot process), two of these plates were corrosion tested for one week, blister tested, and again corrosion tested for an additional week. Both plates proved satisfactory. Each fuel plate for the ALPR core passed the blister test at 1022°F for one hour before consolidation into a fuel assembly.

During fabrication of the fuel plates, a coupon was cut from the end of each plate for corrosion testing. These coupons contained no fuel but consisted of three layers of X8001 aluminum alloy clad material bonded during the manufacturing process. A total of 798 coupons were obtained.



TECHNIQUE "A" TECHNIQUE "B"

FIG. 111
APPEARANCE OF CADMIUM AFTER 2316 hours IN INITIALLY HIGH PURITY WATER AT 417°F

Before the corrosion test was started, the coupons were cleaned in acetone and every twentieth one was weighed to obtain metal-loss information. The corrosion test was carried out at 550°F in an autoclave using distilled deionized water. All samples were visually checked after a 30-day period to insure that no other aluminum alloy had been incorporated by mistake. All were X8001. Samples were removed and defilmed at times up to 7000 hours to obtain a corrosion curve (see Fig. 112). The results indicated a corrosion rate of about 1 mil per year. Blistering occurred on about half of the samples exposed the full test period.

Twenty-one fuel plates were tested for periods ranging from 287 to 309 days at 450°F. Four small blisters ($<\frac{1}{8}$ in. in dia) were found in the cladding near the edges in the "picture frame" area. No distortion or swelling was noted.

Eight fuel plates were tested for 316 days at 600°F. The dimensions of the plates changed, growing nearly 4% in length. A number of small blisters near the edges developed during the exposure.

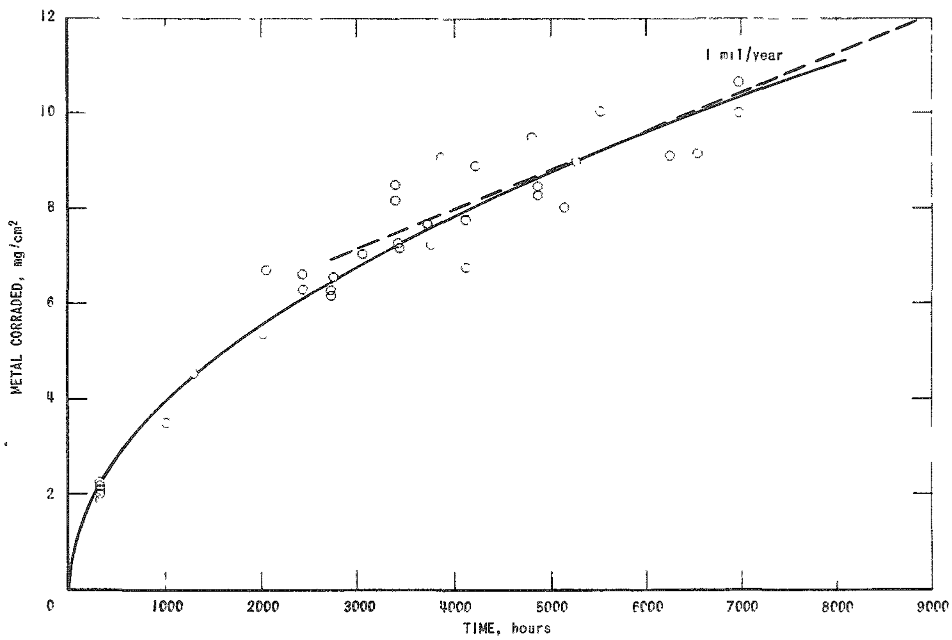


FIG. 112
CORROSION OF FUEL PLATE COUPONS AT 550°F

Numerous tests were made on the X8001 alloy at higher temperatures, both in- and out-of-pile. In two in-pile tests at 500°F, samples exposed in a neutron flux zone showed slightly less corrosion than the samples in the same water but out of the flux.(44)

One of the most important factors for controlling the corrosion of X8001 alloy is to maintain the highest possible purity of water, preferably of 1 megohm-cm resistivity or higher. If it is below this value, the pH must be maintained at 7.0 or slightly below. Another important factor is to have as much aluminum surface area in the reactor as possible. Dynamic loop tests have shown that the higher the corroding surface area per unit circulating water, the lower will be the corrosion (see Fig. 113).(45) The ALPR has over 300 cm^2 of X8001 per liter of water, which is higher by a factor of four than can be attained in the Reactor Engineering Division corrosion loops. Thus the reactor is well on the safe side in regard to this corrosion factor. The mechanism of this surface-to-water-volume ratio effect, and the relationship to the cleanup system operation, are still under investigation.(46)

Although the corrosion of the X8001 material is more difficult to predict at higher temperatures because of the factors noted above, the results at 420°F have been consistently good, and it is believed that no serious corrosion problems will arise from use of the X8001 aluminum in ALPR. The corrosion behavior of the X8001 clad fuel elements in Borax IV appears to support this view.(45)

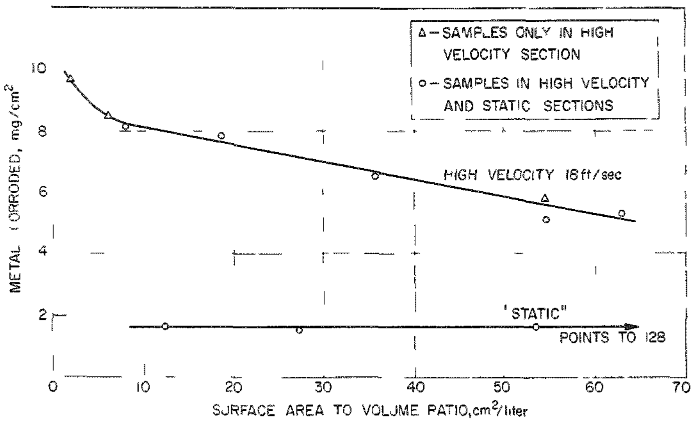


FIG. 113
EFFECT OF SURFACE AREA TO VOLUME RATIO ON THE CORROSION OF ALUMINUM ALLOY X8001 – ONE WEEK AT 500°F IN DISTILLED WATER

The boron poison strips made by extruding a mixture of elemental B^{10} powder and X8001 aluminum powder were also tested for corrosion resistance. Samples were held in water 100 days at 440°F, 114 days at 550°F, and 34 days at 660°F. Specimens tested at 660°F were blistered and in poor condition, but intact. Samples tested at the lower temperatures were in good condition at the conclusion of the tests. The amount of corrosion was approximately equal to that experienced by X8001 aluminum corroded for the same period.

Appendix V

TECHNICAL CHARACTERISTICS - ARGONNE LOW POWER REACTOR (ALPR)

(March 9, 1956)

Note: The following characteristics include Change No. 1 (May 1, 1956), Change No. 2 (June 14, 1956), and interpretation of the specifications ensuing from meeting between Army Reactors Branch and Argonne National Laboratory personnel on August 21,1956.

The Argonne Low Power Reactor Project (ALPR) has been established by the Atomic Energy Commission, at the request of the Department of Defense, as an ultimate member of a family of nuclear power plants to produce electric power and heat at remote military bases.

The ALPR should be designed to provide the best possible boiling heterogeneous nuclear power plant system that meets the requirements of an Auxiliary Station, DEW Line, and is compatible with arctic environmental conditions. The plant will include the structure housing and auxiliary systems. The prototype plant will be constructed at the National Reactor Testing Station, Idaho. The purpose of prototype construction is to obtain information and data on new features of the system. A major objective of the prototype design is to minimize the engineering modifications that need be made when adapting this design to a specific Auxiliary Station.

Construction of the prototype power plant system should include as many features peculiar to arctic conditions as is possible. For example, the air space required between grade and the bottom of the plant structure will be provided. Whether such air space is provided in the prototype structure by placing it on piling, as it must be in the arctic, or by some other construction design, may be determined by considerations of local economy.

Whenever it is necessary to deviate from arctic practices, it is desired that appropriate recommendations be made through the Chicago Operations Office to the Army Reactors Branch, Division of Reactor Development, Washington, D.C.

These Technical Characteristics are to be construed as design objectives for the prototype plant and not as performance guarantees.[47] Changes may be made to ensure the design of a simple, reliable nuclear power plant for the intended application. Recommendation for changes on modifications should be made to the Army Reactors Branch through the Chicago Operations Office.

The DEW Line is now under construction. Specific information on loads, load factors, and curves are not available. A specific site for the installation of the first operating unit has not been selected. Therefore, for many items in this specification, only "average arctic conditions" are given. As these and other application information becomes available, data will be furnished to Argonne National Laboratory.

It is desired that, insofar as possible, the materials, components, and systems of the nuclear power plant, especially for the reactor component, should be acceptance tested under simulated operation conditions before prototype construction.

Washington supervision of this project rests in the Army Reactors Branch, Division of Reactor Development, U. S. Atomic Energy Commission.

General Requirements

1. Capacity (Design)

   200 kw Electrical
   400 kw Heat, Net (approximately 1,300,000 Btu/hr)

2. Frequency

   60 cps

3. Voltage (Busbar)

   120/208, 3-phase, 4-wire, wye connected

4. Power Factor

   0.8

5. Plant Factor

   0.7

6. Standby Equipment

Full capacity, electrical and heating, in conventional diesel-electric power plants and/or oil-fired furnaces assumed.[a]

7. Transportability

Sealift, airlift, and overland transportation are available in the Arctic. Sealift is used only during the summer months, whereas airlift may operate the year round. The runways at some DEW Line sites become soft during the summer which prevents aircraft operation.

[a]The full-capacity electrical requirement is for an arctic installation. The prototype installation was approved with 60-kw diesel-electric.[48,49]

Aircraft operating to DEW Line stations are C-123 and C-124. Details on cargo preparation for these aircraft will be furnished separately.[a] The prototype plant must be designed to incorporate this air transportability feature, by components, prior to initial operation.

8. Personnel

Every man stationed at a remote arctic site increases the logistic support problem for that location. For this reason, the number of operating personnel to run this plant must be a minimum. It may be assumed that certain members of the organization will have some training in the operation of this plant to the extent of routine operation, inspections, and preventive maintenance. Such training will include basic steam technology, power plant operation, parallel operation of generators and instruction in reactor operation; health physics, nuclear instrumentation, etc. These individuals undoubtedly will be utilized in the discharge of the primary mission of the remote site.

The supervisory personnel at this remote site may be expected to have very little knowledge of reactor technology. The reliance for uninterrupted operation is placed upon the technological development of the plant components, the plant design and the operating personnel. Thus, a nuclear plant that will operate reliably for prolonged periods of time with the minimum of supervision and logistic support is required.

The military personnel assigned to remote arctic stations are relieved every 12 months, or less, and replaced by a new group. It should be assumed, therefore, the background and training of the incoming personnel is the bare minimum to satisfactorily accomplish their mission.

The present plans are to operate the DEW Line by contract. It should be assumed the initial operators will be civilians. Major maintenance, fuel changing and radioactive waste disposal will be accomplished by especially trained personnel, civilian and/or military.

9. Facilities

The buildings designed for the Auxiliary Station of the DEW Line consist of modules[b] approximately 16 ft x 28 ft x 16 ft. These modules are connected in one line so that the ultimate structure is 28 ft wide and up to 412 ft long, depending on the number of modules used. The radar antenna dome is above these modules. Each module, the assembly, and the radome is designed for winds of 125 mph, 2 in. of ice, and 30 lb/ft^2 of snow.

[a]The air transportability feature limited all "packages" to $7\frac{1}{2}$ ft x 9 ft x 20 ft, and 20,000 lb in weight.

[b]The module requirement is for an arctic installation. The prefabricated support facilities structure for the prototype was approved.[50]

The modules are designed to be compatible with arctic conditions:

(a) Comfortable, adequate, and flexible to meet personnel and equipment requirements.

(b) Resistant to fire, wind, cold, storm, and deterioration.

(c) Simple and economical to transport, construct, and maintain at arctic sites.

The building design is to emphasize maximum resistance to, prevention of, detection of, and control of fire, consistent with associated problems of arctic construction, maintenance, operations, economics and logistics.

The floor of the antenna in the radome over the building train is 50 ft above the local terrain. The highest point of the building housing this power plant should not be higher than 50 ft above ground. Should this height be exceeded, the Army Reactors Branch, AEC, should be so informed before the design is approved.

Outside access doors must not be encroached upon by building design lest the requisite area for snow removal by mechanized equipment be hampered.

The building housing this plant should conform to these general DEW Line requirements.

Operational Requirements

1. Environmental Conditions

Outside ambient temperature: -60°F to +60°F. The plant should be designed for 60 kw electrical overload at 40°F outside ambient air temperature. (see No. 4 below).

Winds: The specifications on the present structures require a building to withstand 125-mph gusts, 30 lb/ft^2 of snow, and 2 in. of ice.

Permafrost: Permafrost may be expected throughout the DEW Line region. Thawing of the permafrost introduces undesirable stresses in the buildings. Adequate protection must be provided to preserve the permafrost regime. Should refrigeration be required, this electrical load must be considered parasitic.

Annual precipitation: The arctic coast, in general, has very little precipitation. That section of the coast in Alaska proposed for the first installation receives approximately 4.5 in./yr.

Water availability: Many of the DEW Stations have available a source of fresh water, other than melted snow. This water may be expected to have a high mineral and organic material content. Detailed information will be made available at a later date.

Site materials: Information on this subject is limited. Gravel has been used at many DEW Line Stations. Detailed information will be furnished at a later date.[a]

2. Routine Operation

Upon successful completion of acceptance tests, the operational nuclear plant is expected to be placed in routine operation. The plant will be prepared for operation by an operator in accordance with standard procedures for starting up the plant. When ready to take the electric and/or heating loads, parallel operation with the conventional systems will be established, the load shifted to the nuclear plant, the conventional system shut down as prescribed by standard operating procedures. The plant will be adjusted to the load conditions and placed into "demand controlled operation" by the operator. When the plant is operating stably and to the satisfaction of the operator, further detailed and constant attention by operating personnel should be unnecessary. The operator will be performing other duties elsewhere. Routine inspections and preventive maintenance (such as keeping oil reservoirs filled) will be necessary.

Guidance on frequency and extent of inspections and maintenance is expected of Argonne National Laboratory, and the Architect-Engineer.

Whenever required and foreseen, the shutdown of the nuclear plant will be accomplished manually in accordance with standard operating procedures.

3. Restart

Normal startups will be under the control of an operator in accordance with standard procedures. This action will follow core replacements, scheduled shutdowns, and emergency shutdowns. The nuclear plant must be capable of restarting after a shutdown occurring anytime during the core life. Plant down time only reintroduces the petroleum logistics problem that nuclear plants are to relieve, hence, the need to restart at the earliest possible moment.

4. Plant Overload

This plant must be capable of a 60-kw (electrical) overload for a period of one (1) hour or less in each 24-hour cycle. Such a condition

[a]Gravel from a DEW Line station was examined and appraised as being acceptable for biological shield material.

may result from the simultaneous utilization of equipment with low diversity factor or the energizing of radome heaters. Under this condition of operation, the plant must be stable and respond readily and automatically to demand control signals; an operator must not be required.

This overload requirement may be relaxed if a non-standard turbine-generator size should be necessary.

When in this overload condition, the reactor must have sufficient control capability so that for steady operation, or for sudden reduction of load, the reactor will not present any unusual hazard.

5. Stability and Regulation

The plant must be stable throughout its operating range, which is defined as any heating load up to 400 kw plus a net electrical load from 20 to 260 kw. The maximum load of 260 kw includes a 60-kw overload. The following voltage and frequency conditions must be observed:

(a) Voltage may not vary more than ± 5% from the design rated value with load varying from 20 to 260 kw. Load may be applied in increments of as much as 60 kw.

(b) Frequency must not vary more than ± 5% from the design rated value with load varying from 20 to 260 kw.

(c) Voltage may not vary more than ± 2% from the steady-state value at any given load within the operating range.

(d) Frequency may not vary more than ± 0.1 cps from the steady-state value at any given load within the operating range.

(e) Transient voltages may not vary more than ± 5% from steady-state values. Steady-state voltage must be re-established within 5 sec.

(f) Transient frequencies may not vary more than ± 0.5 cps from steady-state values. Steady-state frequency must be re-established within 5 sec.

If a bypass steam system is utilized to meet the requirement for control and 60-kw load increments, up to approximately 1300 lb/hr of steam may be bypassed from the main steam line through a heat exchanger to the condenser. A dead band of approximately 200-300 lb/hr will be allowed to avoid excessive operation of the control system. It must be possible to adjust manually the amount of steam which is bypassed. The bypass steam flow rate must be re-established within 30 sec.

The voltage and frequency requirements listed above are dictated by the requirements of the radar-communications equipment at the DEW Line Station. The sensitivity of this equipment to frequency and voltage changes is being investigated and as detailed data becomes available this information will be furnished Argonne National Laboratory. In order to provide this high-quality electrical power to the electronics equipment, the conventional plant is so arranged that transient-producing utility type equipment is isolated from the radar by using (a) separate diesel-generator(s). Consideration should be given to the isolation of the radar-communications equipment load by some suitable arrangement. However, if this is not feasible, this regulation of voltage and frequency must be applied to the full plant load. If the sensitive electronics load can be effectively isolated, the voltage and frequency regulation for the balance of the station load need only conform to current practice of ± 5% voltage and ± 2% frequency variations. The characteristics of the nuclear plant must be such as to provide for stable parallel operation under all load conditions with the conventional diesel-electric equipment now used.

6. Instrumentation

Instrumentation of the nuclear plant should be a minimum consistent with the safety of the plant and personnel. The prototype of this plant may be instrumented as necessary to meet the AEC Advisory Committee on Reactor Safeguards (ACRS) requirements, but analysis should be made and experimentation should be conducted on the prototype to confirm the minimum requisite instrumentation. Provision should be made for audible warning when hazardous operating conditions are imminent.

Unless otherwise available, controls and instrumentation sufficient for satisfactory parallel operation with the conventional plant must be provided.

The complete control operation - startup, manual and/or automatic load changes, parallel operation and shutdown - should be performed from one control panel. All visual and recording instruments of the nuclear plant must be available at one location. Duplicate instruments may be provided at desired locations.

Power Plant Components

1. Reactor

The reactor of the boiling heterogeneous type is to have the inherent ability to adjust the power generation, and hence, the steam output, to meet the load demand placed upon the power plant. The load demand may be heating or electrical and any combination of the two. The reactor must be stable at all reactor power levels from a "zero power critical" condition through a power level equivalent to a plant electrical overload of 50% throughout the core life. Provisions must be incorporated in the design to provide a sub-critical core at all times during the core or fuel assembly replacement operation and the subsequent cooling period.

Minimum mechanical equipment should be within the biological shield of the reactor. Components with rotating or movable parts should be readily available for inspection and maintenance.

Unless the reactor design is adequate to accommodate the dissipation of heat after shutdown, cooling must be provided, including the cooling of spent cores in storage, if necessary.

2. Component Contamination

The conceptual design places the turbine in the primary loop and outside of the biological shield. This planning requires maximum water purity, minimum entrainment of radioactive particles in the steam, high fuel plate integrity, and a system for the detection of excessive radioactivity in the primary loop.

3. Core Life

The life of the reactor core must be at least two (2) years at full design load. It is expected that fuel changes will be made every three (3) years. The fuel plates must withstand the effects of burnup, corrosion, erosion, heat generation, and to retain the fission products for these periods.[a] This requirement is necessary to hold down the logistic support of this plant and to minimize the time required to reach the point of more economic operation than by conventional plants. The plant down time for any reason must be a minimum in order to keep the petroleum consumption low.

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[a]Since there was insufficient operating experience with aluminum type X8001-clad fuel plates in a boiling water reactor, a firm guarantee of a 3-year core life could not be provided. The Army Reactors Branch was made cognizant of this fact early in the design phases of the program.[5] An effort has been made to lengthen the operational life of the fuel plate by increasing the cladding thickness, and by using Al-Ni-U fuel plate "meat" that is almost as corrosion resistant in boiling water as the cladding material. It is believed that at the present (February 5, 1959) operating conditions of the reactor the assemblies of fuel plates should fulfill the design goal of core life.

4. Shielding

The shielding of the reactor component represents the greatest single volume of the plant, and emphasis should be placed upon realizing the smallest volume and weight of shield necessary.[a]

On the other hand, personnel at the remote site must not be exposed to a radiation dose in excess of 50 mr/7-day week. In the shield design, consideration may be given to the use of exclusion areas and/or shadow shielding, provided the above maximum dose rate is not exceeded. The personnel safety criteria for shielding the reactor apply to the shielding of the spent fuel storage pit.

5. Fuel Storage

Fuel storage for spent fuel assemblies of one core must be provided. The time spent in "cooling" is subject to economic evaluation. It may be assumed that a spent core will cool for one year, and possibly for one core life. Adequate safety provisions must be incorporated to maintain the spent fuel in a subcritical condition at all times during storage.

The fuel storage chamber should be so designed and situated with respect to the reactor and the power plant that removal of the spent assemblies is facilitated and the "cooling off" does not interfere with the operation and maintenance of the plant.

[a] The shielding criteria was based upon the necessity to transport all shield materials by air. The use of locally available gravel for shielding relaxed the requirement.

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