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ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

INITIAL TESTING AND OPERATION OF THE
ARGONNE LOW POWER REACTOR (ALPR)

Reactor Engineering and Idaho Divisions

Edited by

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PREFACE

The contents of this report may be said to be a log of the major events during the tenure of the ALPR facility under Argonne National Laboratory. The period of plant testing and operation was limited, being from July 7, 1958 to February 5, 1959. The plant operating contractor, Combustion Engineering, Inc., thereafter took possession of the facility to initiate and conduct programs for the Army Reactors Branch, Division of Reactor Development, AEC. The facility is now identified as "SL-1," although the original designation "ALPR" is maintained for this report.

This report has been compiled by the combined efforts of the Reactor Engineering and Idaho Divisions of the Argonne National Laboratory.

The design and construction phases of the project, in addition to the component installation, preliminary systems testing, zero-power experiments and areas affected by the design parameters, were the responsibility of the Reactor Engineering Division.

The Idaho Division initially provided the reactor plant operators and was responsible for reactor operation, plant safety, and reactor operator training. The cooperative measures provided some areas of overlap that resulted in an effective program. The Idaho Division personnel contributing to this report were Messrs. W. R. Wallin, R. W. Thiel, J. D. Cerchione, and E. D. Graham.

A cadre of military personnel provided by the Army Reactors Branch were trained in reactor plant operation and maintenance. A large segment of the training period consisted of assisting the Laboratory personnel in meeting the program objectives.

E. E. Hamer
ALPR Project Coordinator
Reactor Engineering Division

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INITIAL TESTING AND OPERATION OF THE ARGONNE LOW POWER REACTOR (ALPR)

I. INTRODUCTION*

This report describes the major events of a program designed to test and operate the completed reactor power plant and associated equipment (Fig. 1). The contents are in chronological order, commencing with acceptance of the facility from the Atomic Energy Commission on July 3, 1958, and terminating with the transfer of the facility to the operating contractor on February 5, 1959.

FIG. 1
SCHEMATIC OF ARGONNE LOW POWER REACTOR

A. Facility Description

The Argonne Low Power Reactor (Fig. 2) is located at the National Reactor Testing Station, Idaho.

*C. R. Braun

FIG 2
ARGONNE LOW POWER REACTOR FACILITY

The ALPR is a prototype nuclear power plant designed⁽¹⁾ to supply electrical power and space heating at a proposed Arctic Auxiliary DEW (Distant Early Warning) Line radar station. The reactor is a 3-Mw(t), natural circulation boiling water reactor, enriched uranium-fueled, cooled and moderated with light water. The plant is designed to produce electric power (200 kw net plus 60 kw overload plus plant load) and space heat (400 kw net) consistent with (1) simplicity and reliability of operation and maintenance with minimum supervision; (2) minimum site construction; (3) maximum use of standard components; (4) limited water supply; (5) utilization of local gravel for reactor biological shielding; (6) transportability by airlift; and (7) nominal three-year fuel operation life-time per core loading. Salient plant operating characteristics and design criteria are summarized in Table 1.

B. Program Summary

The program consisted of equipment installation and reactor assembly, reactor zero-power experiments, plant startup, and plant

performance tests. Also included was a comprehensive training program for the military cadre and the operating contractor. Noteworthy in the program was the expedient component installation and reactor assembly.

Table 1

PLANT CHARACTERISTICS

Reactor Heat Output	3 Mw (nominal)
Steam Production	9020 lb/hr (nominal)
Operating Pressure	300 psig
Operating Temperature	421F (saturation)
Turbine - Generator Rating	300 kw
Generation Voltage	120/208 volt, 3 phase, 4 wire
Design Domestic Heat Load	400 kw
Design Ambient Temperature(a)	-60F to +60F
Design Maximum Wind Velocity	125 mph
Plant Factor	0.70
Design Core Life	3 yr (nominal)
Average Power Density in Core Coolant	17.5 kw/liter
Average Heat Flux	21,500 Btu/hr ft ²
Transportability of Components for Construction	Airlift
Elevation of Prototype Installation	5000 ft above sea level

(a)Not incorporated in building shell or insulation design

Simultaneously, with the reactor isolated from the system, all steam plant systems were checked at full temperature and pressure using a portable steam boiler. This work was accomplished three weeks after plant occupancy and one week later, on August 11, 1958, the first loading of fuel assemblies into the core structure was begun (see Experiments No. 1 through 7, Section V-C).

The entire zero-power experimental program was accomplished at the ALPR site. The 40-assembly reference core and a 59-assembly core for possible use in plants operating at higher powers were investigated over a 9-week period. On October 24, 1958, following disassembly of the zero-power experiment rig and final reactor assembly, the plant was started up. The first power run was achieved without notable difficulty, and full power operation was attained on that day.

The military cadre took an active part in the preceding phases of the program. The power operation training was essentially completed the following month. During this period various unusual operation conditions were imposed on the plant as a check on reactor safety.

The 500-hour full-power plant performance acceptance test was started on November 19, 1958, and was completed successfully on

December 11, 1958. Military cadre reactor operators under Argonne staff supervision conducted the first few days of the test run. The remainder of the test was conducted under cadre supervision.

Plant changes and improvements indicated by the test programs were carried out during the remainder of the Laboratory's responsibility, as well as additional training of military cadre and operating contractor personnel.

In this report data are presented on the various phases of the program, together with test results and recommendations for future plant improvements. The ALPR prototype has proven to be a stable and reliable nuclear power plant.

C. Organization

The ALPR organization during the program (see Section II) consisted of Argonne National Laboratory staff members and the military cadre assigned to the project (see Fig. 3). It should be pointed out that the performance of the cadre was outstanding in all respects and contributed significantly to the successful accomplishment of the program.

FIG. 3
ALPR OPERATIONS ORGANIZATION CHART

During the initial phases, all supervision was performed by Laboratory staff members; however, as the training program progressed, supervision responsibilities for various phases of plant administration and operation were delegated to military personnel. At the completion of the training program in December, 1958, the cadre had qualified for, and were performing all phases of, supervision and operation.

D. Acknowledgements

1. Personnel

The broad scope of the ALPR project makes it impossible to select all the names of individuals for special recognition since so many participated and made notable contributions. Argonne National Laboratory personnel at the site during phases of the program and assuming various roles at different times are as follows:

Idaho Division

J. D. Cerchione	Chief Operator Engineering Staff
E. D. Graham	Health Physicist
C. Miles	Chemist
M. Novick	Division Director
P. Stoddart	Health Physicist
A. M. Solbrig	Physics Staff
F. W. Thalgott	Assistant Division Director
R. W. Thiel	Chief Operator Engineering Staff
W. R. Wallin	Coordinator Chief Operator Engineering Staff

Reactor Engineering Division

C. R. Braun	Assistant Division Director Project Engineer
E. E. Hamer	Chief Operator Chief Loader Engineering Staff
A. Hirsh	Engineering Staff

H. H. Hooker	Coordinator Chief Operator Engineering Staff
W. J. Kann	Engineering Staff
W. C. Lipinski	Coordinator Chief Operator Engineering Staff
F. H. Martens	Coordinator Physics Staff
G. C. Milak	Engineering Staff
A. D. Rossin	Physics Staff
D. H. Shaftman	Chief Physicist
A. Smaardyck	Engineering Staff

The military cadre at the site assisting in the program are as follows:

Major D. A. Ayres	Cadre Chief
Captain R. A. Roberge	Training Officer
1/Lt. J. T. Artman	Operations Officer
WO-3 L. S. Hicks	Plant Superintendent
WO-3 R. E. Embleton	Chief Operator Mechanical Supervisor
WO-1 D. K. Brown	Chief Operator Electronics Supervisor
WO-1 W. C. Mathers	Chief Operator Electrical Supervisor
CPO H. J. Miller	Chief Operator Mechanical Technician
M/Sgt. H. M. Curran	Chief Operator Electrical Technician
M/Sgt. M. B. Hobson	Chief Operator Electrical Technician
M/Sgt. R. C. Lewis	Chief Operator Electronics Technician
M/Sgt. J. A. Parks	Chief Operator Mechanical Technician

M/Sgt. G. E. Paull	Health Physicist
T/Sgt. M. V. Sitton	Reactor Operator Electronic Technician
SFC R. M. Bishop	Reactor Operator Mechanical Technician
SFC P. J. Conlon	Reactor Operator Electronics Technician
SFC L. L. Dole	Health Physicist
SFC R. L. Honeycutt	Reactor Operator Mechanical Technician
SP-5 E. O'Neal	Reactor Operator Electronic Technician

2. Contractors

Under the direction of Argonne National Laboratory several private companies have 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
Worthington Corporation Wellsville, New York	Turbine-Generator Unit
Lasker Boiler & Engineering Corporation Chicago, Illinois	Pressure Vessel and Support Cylinder
Chicago Bridge & Iron Company Chicago, Illinois	Reactor Building Shell and Structural Steel
Modine Manufacturing Company Racine, Wisconsin	Air-cooled Condensers
Illinois Water Treatment Company Rockford, Illinois	Reactor Water Purification System
Leeds and Northrup Company Philadelphia, Pennsylvania	Process and Neutron Instrumentation
Nelson Electric Company Tulsa, Oklahoma	Metal Enclosed Low Voltage Switchgear
Bogue Electric Manufacturing Company Patterson, New Jersey	Emergency Power Supply

Ingersoll-Rand Company
Chicago, Illinois

Feedwater Pumps

Alco Products, Inc.
Schenectady, New York

Control Rod Drives

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

3. 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. PROGRAM SCHEDULE*

The plant was received from the Atomic Energy Commission on July 3, 1958. On February 5, 1959, the facility was transferred from the Argonne National Laboratory to the operating contractor, Combustion Engineering, Inc. The necessity to obtain early construction and operating data is reflected in the schedule outline below.

Table 2

PROGRAM SCHEDULE OUTLINE

1. Component Installation	July 7, 1958 to August 10, 1958
2. Preliminary Systems Testing	July 15, 1958 to August 10, 1958
3. Zero Power Experiments	August 11, 1958 to October 10, 1958
4. Power Operation Experiments	October 11, 1958 to December 11, 1958
5. Plant Performance Test	November 19, 1958 to December 11, 1958
6. Additional Plant Performance Tests	January 5, 1959 to January 9, 1959
7. Plant Modifications	December 12, 1958 to February 5, 1959
8. Training	
a. Military Cadre	July 7, 1958 to February 5, 1959
b. Operating Contractor	January 8, 1959 to February 5, 1959

*C. R. Braun

III. COMPONENT INSTALLATION (July 7, 1958 to August 10, 1958)

The work necessary to prepare the plant for the testing and operational program was extensive. One phase of the work was the installation of the components of the reactor and associated systems (Fig. 4). Another phase was the installation of instrumentation. The installations described in this section are for major components necessary to begin the program.

FIG. 4
REACTOR INSTALLATION, VERTICAL SECTION

A. Steam Boiler*

A portable, automatic, propane-fired, once-through (flash) boiler of 2,000-lb/hr steam capacity was installed on the operating floor of the reactor building (with fuel tanks outdoors) to provide a steam supply to the main steam line of the plant. Preliminary checks of the systems and equipment, including initial operation of the turbine-generator unit, main steam condenser, and other equipment in the cycle, were thereby accomplished with nonradioactive steam. Boiler feedwater was condensate with the initial and make-up requirements provided by the demineralizer for the plant raw water.

*R. W. Thiel

After the preliminary systems testing, the boiler was connected to a pair of temporary, custom-made heating coils installed in the reactor vessel to permit heating and pressurizing of the reactor during the zero-power experiments, instead of using the reactor core (nuclear heat) for this purpose. This allowed the fuel assemblies to be held at essentially zero power during most of the experiments.

The portable boiler was a temporary measure and proved to be a satisfactory steam source in this service. Its use resulted in a major savings in start-up time, since the largest part of the process systems checks had been completed by the time the reactor was ready for power operation. Additional time savings were achieved during the zero-power experiments because the virtually unirradiated fuel elements could be handled, worked upon, and stored without the need for time-consuming special procedures.

B. Control Circuits for the Second Steam Bypass Valve*

This work consisted of installing components in the Process Control Panel for controlling the second (0-10 000 lb/hr) main steam bypass valve in a manner essentially identical to that employed with the original (0-3,000 lb/hr) valve (Fig 5). The following new components were installed:

1. four push-buttons (open-close, manual-automatic);
2. two indicating lights (manual-automatic),
3. one valve-position indicator with calibrating resistor;
4. one electronic servo relay,
5. one precision set point setter,
6. four plug-in relays

In addition, the existing pressure-deviation indicator was modified by replacing the existing retransmitting slide wire with two new slide wires.

These items represent an addition to the equipment originally specified and time did not permit their inclusion in the Process Control Panel before shipment.

The system as installed performed satisfactorily. However, the control slidewire in the pressure-deviation indicator, which is associated with the automatic feature, has since been removed to make room for the slidewire associated with the automatic demand control system. Consequently, only manual control of the second bypass valve is used.

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FIG. 5
BYPASS STEAM VALVE NO.2 CONTROL CIRCUIT, SCHEMATIC

C. Reactor Water Level Instrument Wells*

The first components to be installed into the reactor pressure vessel were the primary and secondary water level instrument still wells (Fig. 6). The instruments are of the "float" type, requiring a device such as a still well in turbulent water.

The support brackets for the still wells are welded to the internal pressure vessel wall. Before welding, the brackets and attached still wells were aligned with the instrument nozzles on the pressure vessel head (see Fig. 9). After welding, all components, including the instruments, were assembled to insure proper alignment.

D. Core Support and Control Rods*

1. Core Support

Before placement in the pressure vessel, the core support was thoroughly examined for structural defects, misalignment, and other factors that would affect its performance. The unit was then degreased and washed. A four-point rope sling, fastened to the core support pads, was attached to the overhead crane and the unit was slowly lowered into position within the pressure vessel. After seating the core support on the support brackets on the pressure vessel thermal shield (Figs. 7 and 8), six special hold-down bolts were inserted to fasten the support to the brackets. Two guide pins, located nonsymmetrically, provided the proper orientation. The core support was checked for plumb and level. No shimming was required. Also checked was alignment of each control rod guide channel with its corresponding control rod drive nozzle in the pressure vessel head.

2. Control Rods

Before placement in the core support, the control rods were thoroughly examined for defects and then washed. Each control rod was then inserted into the proper control rod channel and checked for clearances and alignment (see Figs. 7 and 8).

E. Pressure Vessel Head**

After installation of the reactor liquid level still wells, core support, and control rods, the pressure vessel head was bolted to the pressure vessel. The control rod drives (Fig. 9) and liquid-level instruments were then assembled, and a hydrostatic test of 600 psig was performed on the pressure vessel and its appurtenances.

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STILL WELL FOR PRIMARY LEVEL TRANSMITTER
INSTALLATION IN REACTOR VESSEL

STILL WELL FOR SECONDARY LEVEL TRANSMITTER
INSTALLATION IN REACTOR VESSEL

FIG. 6
REACTOR WATER LEVEL INSTRUMENT STILL WELLS

FIG 7
 CORE PLAN, SECTION

0 2 4 6
SCALE-INCHES

FIG. 8
CORE, VERTICAL SECTION

FIG. 9
CONTROL ROD DRIVES (5)
AND TOP SHIELDING

The pressure vessel head is removed by means of a lifting plate bolted to the five inner control rod nozzle flanges. A three-chain sling is attached to the lifting plate and the head is raised by means of the bridge crane. Preparation for placement consisted of attaching the lifting plate to the head, raising the head off the floor, and then thoroughly cleaning the gasket surface.

FIG. 10
PRESSURE VESSEL HEAD BOLTING

The pressure vessel studs (Fig. 10) were removed from the pressure vessel flange and the stud holes were cleaned. The stud threads were cleaned and coated with a mixture of fine graphite and boiled linseed oil. The stud gage holes were also cleaned and the opening closed with tape to prevent re-entry of foreign matter. All studs were then returned to the pressure vessel flange.

Four guide pins were placed over four studs positioned nonsymmetrically (Studs Nos. 13, 25, 37, and 48) to guide the pressure vessel head into place.

Before placement of the head, the pressure vessel gasket grooves were cleaned and the gaskets inserted.

After proper preparation of the pressure vessel flange and head, the head was oriented and lowered into place guided by the guide pins. When the head was in position for bolting, the guide pins and lifting plate were removed. The head was then secured.⁽²⁾

After the hydrostatic test of the pressure vessel and its appurtenances, and after a "check-out" of assembly and operation of the control rods and drives, preparations were started for the zero-power experiments. This necessitated the removal of the pressure vessel head.

A subsequent removal of the pressure vessel head was necessary in order to remove the steam heating coils used during part of the zero-power experiments.

F. Control Rod Drives*

After the pressure vessel head was installed with the core support and control rods in place, nine control rod drives (Figs. 11 and 12) were installed⁽²⁾ on the pressure vessel head nozzles designed for that

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FIG. 11
 CONTROL ROD DRIVE

FIG. 12
TRANSMISSION AND POSITION INDICATOR

purpose (Figs. 13 and 14). It should be noted that nine drive locations are available for use and five drives are used with the reference 40-fuel assembly core (see Fig. 18).

FIG. 13
CONTROL ROD DRIVES (9)
AND TOP SHIELDING

FIG 14
CONTROL ROD DRIVE INSTALLATION
FOR "HOT" CRITICALS

After the test operation of the control rod drive mechanisms, the units were disassembled in preparation for the zero-power experiments. This preparation consisted of removing the reactor top shield blocks and pressure vessel head flange. A temporary rod drive support was placed on the pressure vessel flange and the drive mechanisms were completely reassembled except for the shielding blocks

G. Control Rod Drives Seal Water System*

Two headers, an inlet and an outlet, are provided to supply seal water to the control rod drive seals. Piping from the inlet header to the seal of each drive and from the seal outlet to the return header was installed (see Fig. 14). Since the drive has a pressure breakdown seal, the water

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for sealing is necessary. Inlet water is part of the feedwater loop and the leakage, or outlet water, is piped to the High-pressure (HP) Condensate Tank. Tubing for the seals can be removed easily and each inlet is provided with a valve to control flow.

Total water flow is measured before the supply, or inlet, header. The temperature of the water leakage, or outlet, from the seal is measured for each drive and regulated by varying the water flow from the inlet header.

H. Reactor Liquid Level Instrument Transmitters*

The primary and secondary liquid-level transmitters are both installed on the reactor pressure vessel head with displacers extending into still wells inside the vessel (see Figs. 6 and 9). The still wells, including their supporting and locating brackets, were placed inside the pressure vessel (see Section III C). The transmitters and displacers were mounted on the corresponding pressure vessel head nozzle and connected after the head had been secured in place. These latter components must be removed and reinstalled each time the vessel cover is removed and replaced.

I. Control Rod Drives Temporary Support**

In order to perform some of the zero-power experiments more conveniently, the pressure vessel head was removed and a temporary rod drive support was installed. The support was a system of 6 ft 6 in. channel iron supports welded horizontally to four angle iron legs. A base plate welded to each of the legs was drilled to fit over specific pressure vessel stud bolts. After proper shimming to obtain leveling, a threaded bushing was used to fasten the support firmly to the vessel. Necessary through-bolt holes were located in each of the channel iron pairs to coincide with the control rod drive flange holes.

J. Zero-power Experiments Instrumentation†

Although the monitoring instrumentation (Fig. 15) on the reactor was designed to cover the entire range of power of the completely loaded reactor, it was necessary to install special instrumentation (Fig. 16) for the zero-power experiments. The instrumentation was designed to give a significant reading with a fairly powerful antimony-beryllium neutron source (Fig. 17) in a fully loaded core (Fig. 18). During the zero-power experiments, the core was not fully loaded during most of the program. It was also very undesirable to have the very strong neutron source present. If

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FIG. 15
NUCLEAR INSTRUMENT BLOCK DIAGRAM

the source were left in during the experiment, the flux level, when the reactor was just critical, would have been extremely high - so high, in fact, that the fuel could not have been handled easily by the experimenters in changing fuel loadings, because of the high level of induced radioactivity.

FIG. 16
ADDITIONAL INSTRUMENTATION FOR
ZERO-POWER EXPERIMENTS

FIG. 17
ALPR SOURCE ASSEMBLY

In a number of the experiments an accurate measurement of small changes in reactivity was required. To accomplish this without removing the source from the core, the power level would have to be raised far above that induced by the multiplied source in order to mask the effect of the source. A qualitative illustration is provided in Fig. 19, showing the dependence of control rod position for apparent criticality at different flux levels. It may be seen that the true criticality is not achieved until the flux level is orders of magnitude above the source level.

The presence of a source also interferes with the determination of the true asymptotic period obtained by adding a positive reactivity to the reactor, requiring extremely long observation times and high powers to achieve good results.⁽³⁾

FIG. 18
CORE LOADING CONFIGURATION

FIG. 19
SOURCE EFFECT ON CONTROL ROD POSITION

1. "Cold" Criticals

For the "cold" critical experiments, a simple arrangement to remove the antimony rod from the beryllium block when criticality was approached was provided. A stainless steel flexible cable was passed over a series of pulleys and terminated in the lower part of the building stairwell. Two hooks were located to hold the looped end of this cable, one giving the "source-in" position and the other the "source-out" position. When it was desirable to remove the source, an operator slowly moved the cable from one position to the other.

additional instruments were provided so that there would still be significant instrument readings with the reactor operating at this much lower power. Sealed aluminum tubes, which could fit into the core structure in vacant fuel assembly locations, were provided so that the detectors could be located close to the core. The tubes were of sufficient length to reach above the top of the water, almost to the temporary control rod drive support structure. They were provided with clamps which could be adjusted to hold them at the desired position in the reactor pressure vessel. The tube and detector arrangement which was used from the initial loading until after the 32-fuel assembly loading is shown in Table 3.

The permanent instrumentation was revised and certain

Table 3

TUBE AND DETECTOR LOCATIONS USED WITH 32-ASSEMBLY LOADING

Tube	Fuel Location	Detectors	
		Upper	Lower
1	57	Kiethley No. 1	Channel No. 4
2	47	A.P.V.R.E.	Channel Nos. 5 and 6
3	37	Kiethley No. 2	Channel No. 2
4	77		Channel No. 3

When the loading was increased beyond 32 assemblies, instrument tubes Nos. 1, 2, and 3 were moved to fuel locations Nos. 68, 38 and 28, respectively (see Fig. 20). The tubes remained there until after the work on the 40-fuel assembly loading was completed. Before the core loading was increased beyond 40 fuel assemblies, the tubes were removed and the detectors placed in the permanent instrument wells (see Fig. 4).

Almost all of the permanent instrumentation was used. The only channel not used was channel No 1, the uncompensated boron chamber providing a high-level scram. Channels Nos. 2, 3, 4, 5, and 6 were used in their permanent form except that the chambers were located in the temporary instrument tubes until their relocation in the permanent instrument wells.

Three additional channels were provided. Two of these used compensated ion chambers feeding into Kiethley Model 410C dc amplifiers. These amplifiers are sensitive enough to detect currents as low as 10^{-13} amp. As such, they are capable of detecting the very lowest source levels. Both of these instruments were provided with trip circuits to scram the reactor whenever the flux level exceeded full-scale reading. One of these, designated Kiethley No. 1, was connected to the linear flux-level recorder. The other, Kiethley No. 2, was connected to a Brush recorder, on which fast transients, such as would occur with a rod drop, could be recorded.

The third instrument was an uncompensated ion chamber connected to an Applied Physics Corporation Vibrating Reed Electrometer. This channel could also "see" the very lowest flux levels. The channel was supplied with a recorder. During much of the work, this instrument was kept on the reactor operating floor to provide a direct indication of flux level as loading changes were being made. Figure 16 shows diagrammatically the complement of temporary instruments as used during the "cold" zero-power experimentation.

2 "Hot" Criticals

For the "hot" critical experiments the pressure vessel head was re-installed. To permit motion of the antimony rod within and out of the beryllium block, the antimony rod was attached to the extension arm of a "tee" shim rod drive (see Fig. 14).

The instrumentation used for the "cold" criticals was used for the "hot" criticals except for the Applied Physics Corporation Vibrating Reed Electrometer. During the "hot" criticals, the instruments were located in the permanent reactor instrument wells.

LOADING NO. _____

DATE: _____
 LOADING NO. _____
 EXPERIMENT NO: _____

TIME: _____
 START _____
 FINISH _____

DESCRIPTION OF EXPERIMENT:

APPROVAL: _____ (COORDINATOR) _____ (CHIEF PHYSICIST)
 REVIEWED: _____ (CHIEF OPERATOR) _____ (CHIEF LOADER)

FUEL ASSEMBLY INVENTORY *

26-mil B STRIP	BLUE	
21-mil B STRIP	RED	
NO B STRIP	GREEN (BLACK)	
(20-mil B STRIP	GREEN	)

CONTROL ROD INVENTORY

	CROSS					TEE			
ROD NUMBER	1	3	5	7	9	2	4	6	8
ROD POSITION (in)									

WATER TEMPERATURE (°F) _____ H_3BO_3 CONCENTRATION (g/liter OF H_2O) _____
 PRESSURE (psig) _____ POWER LEVEL (lb. steam/hr.) _____

REMARKS: * FOR 59-ASSEMBLY LOADINGS CONTAINING 20-mil B STRIPS, THE LOADING CHART WAS REVISED TEMPORARILY SO THAT A GREEN LINE DENOTED A 20-mil FULL LENGTH OR HALF-LENGTH B (BORON) STRIP.

FIG. 20
 ALPR FUEL LOADING CHART

K. Reactor Neutron Source*

The antimony-beryllium neutron source assembly (see Fig. 17) consists of two separate parts that were installed separately. The beryllium block is mounted in a dummy fuel assembly which was handled in the same manner as the regular dummy fuel assemblies. After the beryllium block was installed in its proper location in the core, the antimony rod was added. The irradiated antimony rod was transported in the fuel assembly coffin (Fig. 21) because of its high radioactivity, estimated at approximately 1000 curies.

The coffin with the antimony rod suspended from the coffin gripper was brought over the reactor pressure vessel and lowered until the bottom of the coffin reached the water level in the pressure vessel. The coffin gate was opened, and the antimony rod was lowered into the water and positioned in the beryllium block with the help of a manipulator.

L. Reactor Top Shielding**

The reactor top shield is composed of five pie-shaped concrete segments circumventing the control rod drives (see Figs. 9 and 22). The shield segment over the instrument wells (Segment No. 1, Fig. 9) is composed of two sections, to facilitate removal of the instruments without removal of the complete segment. Supported by the aforementioned shield segments is a top shield plate arrangement of laminated steel and masonite (Fig. 23) that provides a central cavity directly over the reactor pressure vessel for the rod drives and liquid level instruments.

Until the top shielding segments were needed on the operating floor, they were stored outside of the reactor building. By means of a special lifting device, each segment was raised to the reactor operating floor by the monorail hoist and then transferred to the bridge crane and placed. The key shield blocks were inserted after positioning each segment.

The laminated top shield (Fig. 23) was placed by a special lifting device. The lower masonite block was positioned integral with the lower stepped steel block (Segment No. 1), and the composite was centered with the control rod drives in alignment with holes in the masonite. The other laminates were then placed in sequence.

Top shield disassembly not requiring outside storage was accomplished by removing a specific shield segment in the area of the reactor top cavity to which access was wanted. This was done without moving the laminated shield and the shield segment was stored on the operating floor (Fig. 24).

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FIG. 21
FUEL ASSEMBLY COFFIN

FIG. 22
REACTOR TOP SHIELDING

FIG. 23
REMOVABLE TOP SHIELD PLATE ARRANGEMENT

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 (BELOW FLOOR)
16. EQUIPMENT DOORS
17. WASTE STORAGE TANKS (BELOW FLOOR)
18. FEED WATER LINE FILTER
19. PURIFICATION SYSTEM PANEL
20. PURIFICATION SYSTEM AREA
21. CONTAMINATED WATER STORAGE TANK (BELOW FLOOR)
22. BORON STORAGE TANK
23. OPEN EMERGENCY STAIRWAY
24. CONTROL ROD DRIVE MOTORS
25. CONCRETE SHIELD
26. SUPPORT FACILITIES BUILDING WITH CONTROL ROOM

FIG. 24
OPERATING FLOOR PLAN

IV. PRELIMINARY SYSTEMS TESTING (July 15, 1958 to August 10, 1958)

No effort had been made, or was possible to operate the power-plant systems until a source of steam was made available (see Section III-A). The general construction contractor had terminated his contract with systems hydrostatic and electrical continuity tests. Concurrent with installation of components (see Section III) and after installation of the portable steam boiler, preliminary testing of systems (Figs. 25 and 26) commenced. A typical log of plant operating data was reproduced as Appendix C.

A. Main Steam System (Fig. 27)*

The main steam system consists of the steam supply line to the turbine throttle-trip valve, to the bypass steam line junction and to the main safety valve to the condenser. The portable steam boiler was connected to the main steam line at the 4 x 2¹/₂-in. reducer and the line to the reactor was blanked off. This excluded the reactor pressure vessel, reactor pressure instrumentation, and the main safety valve to atmosphere with its instrumentation from the tests.

The steam system, being a short and simple system required only the adjustment of the main steam flow nozzle transmitter (see Section P-5) and a check on the rating of the main steam pressure relief valve to the condenser (360 psig).

B. Bypass Steam Systems (Fig. 27)**

The bypass steam system consists of the steam line junctioning at the main steam line near the turbine with one branch terminating at the simulated heat load heat exchanger and the other branch terminating at the turbine exhaust steam line. The major testing was performed on the (1) steam flow nozzle and instrumentation, (2) control valves and (3) pressure-relief and safety valves.

1. Steam Flow Nozzle and Instrumentation

The bypass steam flow nozzle indicator disagreed with the main steam flow nozzle indicator. Adjustment of the transmitter was made and is discussed in Section P-5. The other instrumentation was checked and adjustments made where necessary (see Section P).

2. Control Valves

Two single-ported lever-operated steam bypass valves are used in this system. Each valve-operating lever is connected to the corresponding

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FIG. 25
ALPR MASTER FLOW DIAGRAM

MAIN STEAM SYSTEM		SHIELD COOLING SYSTEM		REACTOR WATER AND CONTAMINATED WATER (RETENTION) SYSTEM (CONTD.)	
SYMBOLS	DESCRIPTION	SYMBOLS	DESCRIPTION	SYMBOLS	DESCRIPTION
MS-1	REACTOR SHUTOFF	SC-1	SHIELD COOLER PUMP INLET SHUTOFF	RP-17	MIXED BED DEMINERALIZER OUTLET TO CONTAMINATED WATER (RETENTION) TANK SHUTOFF
MS-2	TURBINE TRIP-THROTTLE	SC-2	SHIELD COOLER PUMP AND TANK DRAIN	RP-18	DEMINERALIZERS OUTLET TO CONTAMINATED WATER (RETENTION) TANK HOSE CONNECTION SHUTOFF
MS-3	STEAM BYPASS REGULATING 0-3000 LB/HR (1" VALVE)	SC-3	SHIELD COOLER PUMP DISCHARGE SHUTOFF	RP-19	CONTAMINATED WATER (RETENTION) TANK PUMP DISCHARGE HOSE CONNECTION SHUTOFF
MS-4	STEAM BYPASS REGULATING 0-10,000 LB/HR (1½" VALVE)	SC-4	SHIELD COOLER PUMP CHECK TO AUXILIARY COOLER	RP-20	FEEDWATER FILTER VENT SHUTOFF
MS-5	STEAM BYPASS BACK PRESSURE REGULATING (40 PSIG)	SC-5	SHIELD COOLING COIL #1 THROTTLING AND SHUTOFF	RP-21	FEEDWATER FILTER DRAIN SHUTOFF
MS-6	STEAM BYPASS PRESSURE RELIEF (50 PSIG)	SC-6	SHIELD COOLING COIL #2 THROTTLING AND SHUTOFF	RP-22	PURIFICATION SYSTEM FILTER VENT SHUTOFF
MS-7	REACTOR PRESSURE RELIEF TO CONDENSER (350 PSIG)	SC-7	SYSTEM DRAIN (MANUAL)	RP-23	PURIFICATION SYSTEM FILTER INLET DRAIN SHUTOFF
MS-8	REACTOR PRESSURE SAFETY TO ATMOSPHERE (385 PSIG)	SC-8	SYSTEM BYPASS AND DRAIN (SOLENOID)	RP-24	PURIFICATION SYSTEM FILTER DRAIN SHUTOFF
MS-9	TURBINE EXHAUST SAFETY TO ATMOSPHERE (5 PSIG)	SC-9	SYSTEM PRESSURE RELIEF	RP-25	CONTAMINATED WATER (RETENTION) TANK PUMP INLET HOSE CONNECTION SHUTOFF
AUXILIARY STEAM SYSTEM		SC-10	SYSTEM EXPANSION TANK CHECK TO SHIELD COOLER	RP-26	CONTAMINATED WATER (RETENTION) TANK PUMP INLET SHUTOFF
AS-1	AIR EJECTORS STEAM LINE SHUTOFF	SC-11	SHIELD COOLER PUMP AND SYSTEM FILTER BYPASS CHECK	RP-27	CONTAMINATED WATER (RETENTION) TANK PUMP DISCHARGE FOR PURIFICATION SYSTEM PUMP PRIMING SHUTOFF
AS-2	CONDENSER AIR EJECTOR STEAM SHUTOFF	MAKE-UP WATER SYSTEM		RP-28	CONTAMINATED WATER (RETENTION) TANK PUMP DISCHARGE SHUTOFF
AS-3	GLAND AIR EJECTOR STEAM SHUTOFF	MW-1	RAW WATER DEMINERALIZER SOLENOID VALVE BYPASS	RP-29	CONTAMINATED WATER (RETENTION) TANK PUMP DISCHARGE HOSE CONNECTION SHUTOFF
AS-4	TURBINE GLAND STEAM SHUTOFF	MW-2	RAW WATER DEMINERALIZER TANK INLET SHUTOFF	RP-30	FEEDWATER LOWER SPRAY RING SHUTOFF
AS-5	SPACE HEATING STEAM SHUTOFF	MW-3	RAW WATER DEMINERALIZER TANK BYPASS	RP-31	FEEDWATER UPPER SPRAY RING SHUTOFF
AS-6	SPACE HEATING STEAM REGULATING	MW-4	RAW WATER DEMINERALIZER OUTLET SHUTOFF	RP-32	EDUCTOR VALVE
AS-7	SPACE HEATING STEAM SUPPLY LINE DRAIN	MW-5	RAW WATER DEMINERALIZER DRAIN	BORON PUMP SYSTEM	
AS-8	SPACE HEATING HEAT EXCHANGER VENT	MW-6	RAW WATER DEMINERALIZER TANK VENT	BP-1	BORON TANK DRAIN SHUTOFF
AS-9	SPACE HEATING HEAT EXCHANGER VENT AND CONDENSATE DRAIN AT FLASH TANK SHUTOFF	MW-7	RAW WATER DEMINERALIZER INLET SHUTOFF (SOLENOID)	BP-2	BORON TANK DRAIN HOSE CONNECTION SHUTOFF
AS-10	CONDENSATE FROM L.P. CONDENSATE TANK	MW-8	RAW WATER DEMINERALIZER INLET SHUTOFF (MANUAL)	BP-3	BORON PUMP BYPASS HOSE CONNECTION SHUTOFF
FEEDWATER SYSTEM		MW-9	WATER STORAGE TANK TO CONDENSER HOTWELL SHUTOFF	BP-4	BORON PUMP DISCHARGE SHUTOFF
FW-1	FEEDWATER PUMP #1 INLET SHUTOFF	MW-10	WATER STORAGE TANK TO SHIELD COOLING EXPANSION TANK SHUTOFF	BP-5	BORON PUMP SYSTEM CHECK
FW-2	FEEDWATER PUMP #1 OUTLET CHECK	REACTOR WATER AND CONTAMINATED WATER (RETENTION) SYSTEM		BP-6	BORON PUMP SYSTEM LOWER SPRAY RING SHUTOFF
FW-3	FEEDWATER PUMP #1 OUTLET SHUTOFF	RP-1	FEEDWATER SYSTEM SHUTOFF	BP-7	BORON PUMP SYSTEM UPPER SPRAY RING SHUTOFF
FW-4	FEEDWATER PUMP #2 INLET SHUTOFF	RP-2	FEEDWATER FILTER INLET SHUTOFF	<div>FIG. 26</div> <div>VALVE DESIGNATIONS AND DESCRIPTIONS</div>	
FW-5	FEEDWATER PUMP #2 OUTLET CHECK	RP-3	FEEDWATER FILTER BYPASS SHUTOFF		
FW-6	FEEDWATER PUMP #2 OUTLET SHUTOFF	RP-4	PURIFICATION SYSTEM INLET SHUTOFF		
FW-7	FEEDWATER REGULATING (MOTOR OPERATED)	RP-5	MIXED BED DEMINERALIZER OUTLET REGULATING		
FW-8	FEEDWATER LINE DRAIN	RP-6	FEEDWATER FILTER OUTLET SHUTOFF		
FW-9	CONTROL ROD DRIVE COOLING WATER REGULATING	RP-7	PURIFICATION SYSTEM OUTLET SHUTOFF		
COOLING WATER SYSTEMS (CONDENSATE)		RP-8	MIXED BED DEMINERALIZER INLET SHUTOFF		
CW-1	CONDENSATE CIRCULATING PUMP INLET SHUTOFF	RP-9	CATION BED DEMINERALIZER INLET SHUTOFF		
CW-2	CONDENSATE CIRCULATING PUMP DISCHARGE SHUTOFF	RP-10	CONTAMINATED WATER (RETENTION) TANK-CATION BED DEMINERALIZER INLET SHUTOFF		
CW-3	CONDENSATE CIRCULATING LINE TO SHIELD COOLER DRAIN	RP-11	CATION BED DEMINERALIZER OUTLET SHUTOFF		
CW-4	CONDENSATE CIRCULATING LINE FROM SHIELD COOLER TO FLASH TANK SHUTOFF	RP-12	CONTAMINATED WATER (RETENTION) TANK-MIXED BED DEMINERALIZER INLET SHUTOFF		
CW-5	CONDENSATE CIRCULATING LINE FROM AIR EJECTOR PRECOOLER TO FLASH TANK SHUTOFF	RP-13	PURIFICATION SYSTEM PUMP BYPASS SHUTOFF		
FEEDWATER PUMPS GLAND SEAL VALVES CARRY NO SEPARATE DESIGNATION		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

FIG. 27
STEAM SYSTEMS AND SPACE HEATING FLOW DIAGRAM

motor-operated drive mechanism crank by means of a connecting link. The connecting links are provided with springs to permit slight overtravel of the drive mechanisms, with resultant tight closure of the valve. Preliminary adjustment consisted of adjusting the linkage so as to coordinate full valve stroke with full travel of the drive mechanism, and to provide approximately $\frac{3}{16}$ in. compression of the link springs with the drive mechanism operated to its limit in the closed direction.

3. Pressure-relief and Safety Valves

The bypass pressure-relief valve was set at 40 psig in order to maintain this pressure on the steam supply to the simulated heat load heat exchanger. It was noted that the set pressure fluctuated with the operation of the bypass control valves. This fluctuation was considered to be permissible unless the pressure were to increase beyond the safety valve pressure rating or to decrease so much as to provide an inadequate supply of steam to the simulated head load heat exchanger. The setting, therefore, was left at 40 psig, and no adjustment was made.

The bypass safety valve was checked and found to agree with the set pressure rating of 50 psig.

C. Turbine Exhaust Steam System (Fig. 27)*

The turbine exhaust steam system consists of the piping and instrumentation between the exhaust end of the turbine and the main steam condenser header. Checks were primarily made of the instrumentation (see Section P) and the pressure-relief valve.

The "actuation" of the pressure-relief valve was not achieved on the first two attempts. On the first attempt the steam boiler was "unloaded" into the system with no equipment in operation while maintaining a temperature check on the condenser in order not to exceed the design maximum. It appeared that the air ejector aftercondenser vent did not have enough pressure drop to induce a pressure increase.

The second attempt was performed with the aforementioned vent blocked. Again the valve could not be actuated. This time it was noted that the turbine seal glands were venting the system sufficiently to limit the pressure maximum. It was decided to defer the rating of the safety valve until the reactor was in operation and a sufficient supply of steam was available to operate, in addition to having the turbine in operation with the glands sealed by steam.

*E.E. Hamer

D. Cooling Water Systems (Condensate)*

This system consists of two closed circuits and one open circuit. The closed circuits are (1) coolant for the air ejector aftercondenser and vapor precooler (Fig. 28), and (2) secondary coolant to the shield cooling system (Fig. 29). The open circuit supplies water to the feedwater pumps seal gland (see Fig. 28).

Balancing the closed circuits is accomplished by throttling valves CW-4 and CW-5 located at the flash tank. The shield cooling secondary circuit could not be balanced at this time because the primary circuit was not absorbing any heat. This system was balanced after the reactor was in operation. The air ejector aftercondenser and vapor precooler circuit were balanced by throttling valve CW-5 and observing a temperature difference between instruments TI-232 and TI-223. A 25F maximum temperature differential is considered satisfactory provided that the aftercondenser outlet temperature is not above the atmospheric boiling temperature of approximately 205F.

It was noted that the condensate circulating pump cavitated when valves CW-4 and CW-5 were in the full open position. The throttling of valve CW-2 was investigated as one way to eliminate the cavitation. Although valve CW-2 will do this, the throttling of valves CW-4 and CW-5 is less sensitive.

The seal water supply to the feedwater pumps needed no adjustment. After the Plant Performance Test (see Section VIII-D), a pressure-reducing valve and gauge were installed. The pressure to the pump seal glands was then maintained at the recommended 3 to 5 psig.

E. Low-pressure (LP) and High-pressure (HP) Condensate Return Systems (Figs. 28 and 30)**

The LP condensate return system collects condensate and vapor from the turbine exhaust piping, turbine seal glands, and turbine drains. The HP condensate return system collects condensate from the air ejectors aftercondenser, vapor precooler, the rod drive water seal effluent, and pressure vessel gasket leak-off area. Both systems discharge into the flash tank.

Both systems were tested for capacity. The HP system discharge rate was sufficient for all conditions imposed upon it. The LP system was tested with all turbine drain valves open and the turbine operating. The system operates at condenser vacuum. It was noted that the vacuum was not destroyed, indicating that the discharge rate was sufficient.

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FIG. 28
FEED WATER, COOLING WATER (CONDENSTATE), AIR
EJECTORS, CONDENSER, ROD DRIVE COOLING AND
DRAINS SYSTEMS FLOW DIAGRAM

FIG. 30
MISCELLANEOUS DRAINS AND VAPOR
LEAK-OFF FLOW DIAGRAM

The HP system discharges the water from its condensate return tank by means of a steam trap and pump. The LP system condensate return tank water is discharged through a pipe to the turbine exhaust line and pump. Each condensate return tank is provided with a two-stage, float-operated switch mechanism which incorporates both a condensate return pump control switch and an alarm switch. The operation of these circuits was tested by manually operating the switch units.

F. Air Ejector System*

Two air ejectors (Fig. 31), operating on independent systems, discharge into a common water-cooled aftercondenser (see Section IV-D). One ejector is for the main steam condenser (see Fig. 28) and the other is primarily for the turbine gland seal (see Fig. 30). The discharge from the water-cooled aftercondenser is through an air-cooled aftercondenser on the fan floor and then to the atmosphere (see Fig. 28).

FIG. 31
AIR EJECTORS, AFTERCONDENSER AND GLAND VAPOR PRECOOLER

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1. Systems Tests

Both air ejectors, when operating, functioned properly. The "roughing" was fast although no time was noted. A leak test on the condenser system using freon gas and a "halogen" detector indicated a tight system. A similar test on the turbine seal glands indicated excessive air in-leakage but none was detected entering the condenser system. Turbine seal gland air in-leakage does not impair performance but does increase the total operating plant water loss.

Water was noticed in the water-cooled aftercondenser discharge flowmeter. This was presumed to be due to the temperature of the discharge gases being higher than the ambient room temperature, therefore causing some condensation from the water saturated gases.

2. Smoke Test

The air ejector system noncondensable gases discharge to the atmosphere from the condenser air circulation system exhaust duct. A smoke test was performed in order to insure that the building interior would not be contaminated by radioactive products which would be emitted in the event of a fuel plate rupture.

Smoke was introduced at flowmeter FI-116 (see Fig. 28) and observed going to atmosphere during changes of condenser fan speed and damper positions. Short-circuiting to the condenser inlet duct was noted. The discharge pipe of the air ejectors system was therefore changed to a position which would discharge the gases to the atmosphere directly through the power plant building roof (Fig. 32).

G. Feedwater System (Fig. 28)*

This system provides water to the pressure vessel and functions as part of the control network in maintaining a desired water level in the reactor. The feedwater is also used as the secondary heat transfer fluid to the reactor purification system and also provides the coolant for the control rod drive seals.

1. Pumps

Circuits are provided for automatically starting the standby feedwater pump if the discharge pressure of the operating pump should fall below a set point. This action is initiated by electrical contacts in one of two pressure gauges, depending upon which pump is running. These contacts were originally arranged so as to be closed at normal running pressure, and to open with a drop in pressure. Early tests of the circuits, however,

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showed considerable vibration of the gauge needles, and consequent severe arcing at the contacts and chattering of the associated relays. Gauge snubbers were installed in the gauge lines, but these did not reduce the needle vibration sufficiently. Therefore, the circuit was modified so that the contacts are open at normal pump discharge pressure and close with a fall in pressure. With this change, the pump transfer circuits perform satisfactorily. Additional pushbuttons were installed adjacent to the pump discharge pressure gauges to facilitate testing the automatic pump transfer circuits.

During the early tests of feedwater pump No. 1, a considerable amount of lubricating oil escaped from the lower pump motor bearing (see Section VII-D-3). Investigation of this indicated that overfilling of the lower bearing reservoir had resulted in overflow of the oil into the space between the shaft and lower bearing cap and then down the motor shaft. Careful attention to the maximum filling level appears to have corrected this condition.

2. Control Valve

A single-ported, lever-operated valve is employed for feedwater flow control. A spring link is used to connect the valve lever to the motor-driven valve operator so that tight valve closure may be obtained. The linkage was adjusted so that full drive unit travel corresponded with full valve stroke, except that slight drive unit overtravel in the "close" direction compresses the spring link to provide tight closure of the valve.

H. Control Rod Drive Seal Cooling (Figs. 28 and 30)*

The control rod drive seal has two connections - an inlet and an outlet. The inlet water is supplied by the feedwater pump (see Section IV-G). It provides cooling for the seal and drive components, and assures that clean water is used to flush the seal and keep dirt particles to a minimum. Leakage or outlet water is returned to the HP condensate tank (see Section IV-E). A flowmeter is provided just after the feedwater pump to measure the total flow of water supplied to the drives. An inlet manifold is also provided with individual valves to control the flow to each drive. A thermocouple is attached to each seal outlet and the water temperature is relayed to the multipoint indicator in the control room.

A hydrostatic pressure test was performed on the high-pressure (inlet) part of the system and an adequate flow rate of 0.1 gpm per drive was established. (Subsequent monitoring of temperatures, see Section VII-D-2, indicated the temperatures to be within the normal operating range.)

*W. J. Kann

I. Reactor Water Purification System (Fig. 33)*

The reactor water purification system is comprised of two systems. The primary system maintains reactor water quality. The secondary system is the contaminated water (retention) system which facilitates the hydraulic removal and replacement of the primary system resin. A typical log of the system operating data is reproduced in Appendix C.

1. Initial Systems Tests

The initial tests performed were accomplished at room temperature. The major tests included a circuitry check, flow through each resin column, flow through columns in series and parallel, resin removal, and resin addition. Operation proved to be satisfactory in all cases.

2. Subsequent Tests

After the initial tests, continuous operation of the system commenced.

a. Purification System Pump

During a continuous operational test, the pump began to lose suction. Priming by use of the contaminated water (retention) pump was necessary to re-start the pump each time. Subsequently, the addition of an eductor in the suction line adjacent to the pressure vessel and extension of the suction line within the vessel overcame the problem.

b. Contaminated Water (Retention) System Pump

After repeated and prolonged use of the pump, a reduction in lift capacity was noted. This was traced to air in-leakage at the packing gland. A water seal and lantern rings were incorporated.

c. Resin Removal

The first series of test indicated efficient hydraulic removal of the resin from the columns to the disposal tanks by use of hoses. During subsequent tests (see Section VII-D-9), the resin became packed and removal was impossible when limited to the pump pressure of 50 psig. It was noted that to avoid a re-occurrence of the situation, rubber hoses would not be safe for use at a higher pressure. Temporary steel piping was incorporated to replace the rubber hoses.

Later tests also indicated that the last resin to be discharged was extremely high in activity and that the permanent piping installation should be shielded.

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FIG. 33
REACTOR WATER PURIFICATION SYSTEM FLOW DIAGRAM

d. Resin Addition

Later tests during resin addition indicated a decrease in the efficiency of the contaminated water (retention) pump in removing water from the resin columns. Water removal was considered necessary to prevent classification of the mixed bed resin. It was proven by resin efficiency tests that addition of mixed bed resin in a column full of water did not decrease the resin efficiency. Since the wet method of resin addition consumed more time than the dry method and since the working area may have a radiation level above the specified AEC tolerance level for continuous occupancy, it was recommended that an eductor be incorporated at the contaminated water (retention) pump to provide a continuously dry resin column during resin addition (see Appendix A-H-4). It is assumed that one reason for the failure of the pump to maintain a dry column was due to air inleakage at the system safety valve.

J. Boron Injection System (Figs. 4 and 29)*

This system is used to inject a boric acid solution into the reactor by a hand pump or by gravity flow. Either the upper or the lower spray ring can be selected as the outlet in the pressure vessel. Gravity flow may be used only at atmospheric reactor pressure conditions. The test results are noted in Table 4.

Table 4

BORON INJECTION SYSTEM TEST RESULTS

Test	Results
1. Hand pump - 65 strokes/min	2.3 gpm
2. Gravity through pump - lever down	1.5 gpm
3. Gravity through pump - lever up	1.5 gpm
4. Gravity through hose	4.6 gpm
5. Time for boric acid to reach upper spray ring from commencement of hand pumping	60 sec
6. Time for boric acid to reach upper spray ring by gravity flow through pump	60 sec
7. Time for boric acid to reach upper spray ring by gravity flow through hose	12 sec

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K. Simulated Heat Load System (Fig. 27)*

The simulation of space heating is by means of a closed water-loop system. Steam from the reactor plant bypass line is diverted to a heat exchanger to heat water. The heat in the water is dissipated to the outside air by means of a finned-tube cooler and fans.

The temperature-regulating valve was set to maintain a heat exchanger shell pressure in the range from 8 to 15 psig. The cooler louvers were also preset for the outside ambient temperature conditions. By varying the water flow and/or air flow through the finned-tube cooler, various power outputs and water temperatures were achieved, including the design requirement of 400 thermal kilowatts.

L. Reactor Shield Cooling System (Fig. 29)*

The shield cooling system carries away the heat from the reactor external lead thermal shield. The heat is then transferred to a secondary water system (see Section IV-D). During normal reactor operation and shutdown, forced circulation of water coolant is used. Two emergency coolant circulation systems are available for reactor shutdown conditions in case of pump failure. One system is a "natural circulation" type, which could not be tested because reactor operation is necessary. (Subsequent tests, see Appendix B-F, indicated that the system performs satisfactorily.) The other system is a "gravity flow" type, which passes water from the plant storage tank through the cooling coils in the lead thermal shield and drains to the plant contaminated water (retention) tank.

The forced circulation system tests indicated that the two double coil circuits in the thermal shield have the same pressure drop, thereby eliminating the need for circuit balancing. The independent flow through each of the two circuits was 10 gpm, with the total flow through the system (both coil circuits in parallel) being 15 gpm. The water was at room temperature and the pump pressure was 48.5 psig.

The gravity flow system proved to be inadequate because of the restriction in the solenoid dump valve. The $\frac{1}{16}$ -inch valve port was replaced by a $\frac{3}{8}$ -inch port resulting in adequate flow for shutdown cooling.

M. Plant Water Make-up System (Fig. 29)*

The plant water make-up system filters and demineralizes the unchlorinated site water for use in the plant. The recommended flow rate of 3 gpm was used at all times with the commercial mixed bed resin TM-1 (Illinois Water Treatment Company designation).

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The modifications necessary were the additions of (1) a plant water supply pump bypass line and (2) a water storage tank high level float switch. These additions eliminated the necessity of constant water level surveillance and possible water overflow. The float switch signal closes the system water supply solenoid valve.

N. Reactor Shield and Instrument Well Exhaust System (Fig. 34)*

Three functions are provided by this system: (1) the air cooling of the nuclear instruments (ion chambers) in the instrument tubes adjacent to the reactor pressure vessel, (2) the air cooling of the reactor top shielding area, and (3) the maintenance of a negative pressure on the bulk reactor biological gravel shield.

The radiological filter in the line to the biological shield was blanked off. No system test was made because of anticipated work within the shield gravel thereby destroying the top mastic seal and thus invalidating the results of the test.

The air flow through the instrument tubes was balanced against the reactor top shield air flow by manipulation of the dampers. Measurement of air flow to the instrument tubes was taken at the system intake air register. (The balance was verified later by reactor operation, see Section VII-D-13.)

O. Condenser Air Circulation System (Figs. 35 and 36)**

The condenser air circulation system consists of (1) a fan with speed regulation by condenser outlet temperature; (2) damper with position regulation by condenser inlet temperature; (3) manually positioned dampers; and (4) the necessary ductwork. Outside air is used in direct contact with the main steam condenser and other air-cooled heat exchangers when the outside ambient air temperature is 40F or above. For outside temperatures below 40F, condenser outlet air is recirculated and mixed with the outside air by means of the mixing chamber in order to maintain an average 40F air temperature at the condenser inlet.

1. Smoke Test

In order to evaluate the effect of the relative positions of the inlet and outlet ducts on the condenser inlet temperature, a smoke test was performed and photographed. The smoke generator was positioned in the outlet duct and the fan speed varied from zero to 100%. The conclusions indicated a necessity for duct modification since recirculation was evident. A temporary plywood structure to direct the outlet air stream upward and obtain the inlet air from below was fabricated. This arrangement resulted in a greater separation of the two ducts. The present duct configuration is the result of these tests (see Section VIII-A-1).

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FIG. 34
 PLANT VENTILATION FLOW DIAGRAM

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

FIG. 35
FAN FLOOR PLAN

FIG. 36
CONDENSER AIR CONTROL BLOCK DIAGRAM

2. System Control

Remote control for the condenser air circulation system is effected by two motor-operated damper assemblies and a motor-operated adjustment of the fan drive fluid coupling. The drive mechanisms and coupling linkages are similar to those used with the bypass steam valves (see Section IV-B-2), except that rigid connecting links are used. The linkages were adjusted, and the control systems were tested as described in Section IV-P.

3. System Balance

Surveys to determine the air volume, velocity and temperature distribution were made. A typical set of test data taken with the plant in operation is reproduced in Appendix C.

P. Process Systems Instrumentation (Figs. 37 and 38)*

The process systems instrumentation includes local indicators with or without control contacts, transmitters with either continuously variable or on-off type signal outputs, and process panel indicators, recorders and alarms. The preliminary testing and calibration procedures followed in the various cases are described below.

1. Pressure Gauges

The pressure gauges are of the Ashcroft type. The scale calibration of these gauges was checked against a standard gauge using a cylinder of nitrogen with a pressure regulator and suitable valving as a source of test pressure. The standard gauge was an Ashcroft gauge having a full-scale range of 600 psi, that had been previously calibrated using a dead weight tester to within ± 2 psi over the full scale. Where calibration points at 15 psig and below were necessary, a mercury manometer was substituted for the standard gauge. Alarm or interlocking contacts, where used, were set on the basis of gauge scale calibration and were tested to see that the proper sequence of events followed opening and closing of the contacts.

2. Pressure Transmitters

The pressure transmitters are all Manning, Maxwell, and Moore Microsen units, having a bourdon tube drive and a balanced-beam electronic transducer. These were calibrated using the same method as above, but against transmitter output current and deflection of the corresponding indicator on the process panel.

In one case, that of the pressure-deviation indicator, the procedure was more involved. This transmitter has a suppressed-zero range of from 230 to 350 psig. The indicator on the process panel has a full-scale range of 40 psi, calibrated from -20 to 0 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 pressure-deviation indicator reads zero. The zero setting is accomplished by adding a constant (but adjustable) voltage to the transmitter output.

As a consequence, the calibration of this instrument chain consisted of coordination of transmitter output as a function of pressure, auxiliary voltage supply, zero-setting potentiometer, and indicator response.

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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 ppH)
30. PROCESS PANEL TEMPERATURE SELECTOR
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. 37
PROCESS PANEL

3. Pressure Switches

Several pressure-actuated switches are used for alarm and interlocking functions. The operating points of these were calibrated using the nitrogen cylinder and standard gauge arrangement also (see Item 1 on page 68).

4. Combination Pressure-vacuum Indicators, Transmitters, and Switches

These devices were calibrated and tested in the same manner as the pressure devices except that calibration points below atmospheric pressure were obtained with a vacuum pump and a mercury manometer. Since these devices are sensitive to small pressure changes, it is important that, when the connecting lines are disconnected, they be filled to the final equilibrium water levels before reconnecting.

5. Flow Transmitters

The three flow transmitters employed in the plant (feedwater, main steam, and bypass steam) consist of differential pressure transmitters which are actuated by the pressure drop across the respective flow nozzles. Condensate chambers are interposed in the piping between the steam flow nozzles and the associated transmitters. The transmitters are Microsen units which are driven by Barton differential pressure cells and include integral square root extraction.

The feedwater flow transmitter was calibrated against the corresponding process panel recorder by applying gas pressures corresponding to various flow rates to the high-pressure side of the differential pressure cell with the low pressure side vented to the atmosphere. The pressure thus applied was measured with a mercury manometer. The steam flow transmitters were calibrated in a similar manner, except that the condensate chambers and connecting piping were first filled with water to the operating level. The test pressure was then applied to the upstream condensate chamber with the downstream chamber vented.

Some difficulty was experienced in maintaining the flow indication at or near zero with zero flow. This was not appreciably improved by painstaking re-calibration of the transmitter-indicator chains. Since the transmitter output is proportional to the square root of the actuating differential pressure, a five per cent indicated flow corresponds to a 0.25% of full range of the actuating pressure. It follows that these flow recorders should not be depended upon for flows less than approximately 10% of the full scale range.

6. Flow Indicators

No attempt was made to check the calibration of the various local-indicating float, flapper, and bulls-eye type flowmeters.

7. Liquid Level Transmitters

The hot well liquid level transmitter consists of a Barton-Microsen unit similar to the flow transmitters, but without the square root extraction feature. The transmitter output current drives a milliammeter-type indicator on the process panel. Since the maximum differential pressure corresponds to minimum water level in the hot well, the indicator is provided with an inverse scale. The Barton differential pressure cell bellows displaces approximately 1.5 cu in. for full-range travel. In order to make the resultant variation in height of the static water leg negligible, a small surge tank is installed in this leg so that the height of water in it varies less than $\frac{1}{8}$ in. over the operating range. This instrument was calibrated in a manner similar to that used with the flow transmitters (see Item 5 above). Two electronic alarm relays, which are also connected in series with the output of this transmitter, were adjusted to trip at the desired levels.

The reactor water level primary instrument chain consists of three elements. Water level is detected by a displacer-torque-tube assembly which positions the core of a linear differential transformer transducer. The transmitter output signal is fed to a servo receiver in the process panel which automatically adjusts a second differential transformer to balance the first, and transmits a dc signal to the water-level recorder. Because of the

variation of water density and torque tube stiffness with temperature the calibration of this instrument varies with temperature.

The transmitter and servo receiver were calibrated in a laboratory test facility for both room-temperature and 300-psig conditions prior to installation in the plant. A second room-temperature calibration was run after installation. This second calibration, together with the previously determined relationship between the room-temperature and hot calibrations, determined the hot calibration as installed. As calibrated, the instrument indicates water level directly when the reactor is at 300 psig. A calibration curve is used to convert instrument readings to the actual levels when the reactor water is at room temperature.

The servo receiver is equipped with high and low switches for reactor shutdown actuation. These have been found to have an undesirably large amount of hysteresis in their action. The level recorder is also equipped with two switches which are used for alarm actuation.

The reactor water level secondary instrument consists of a similar displacer-torque-tube assembly which is coupled to a pair of micro-switches. This instrument was also calibrated for hot and cold conditions prior to installation in the reactor pressure vessel. Alarm-switch settings were made at room temperature after installation to produce the desired alarm-trip points for the hot pressurized condition. This device has been found to be somewhat difficult to adjust accurately and to display considerable hysteresis in the switch action.

8. Local Temperature Indicators

With the exception of two Fenwal, Type 541, temperature-indicating controllers, which are used to detect steam relief and safety-valve flow, the local temperature indicators are Manning, Maxwell, and Moore mercury-actuated dial thermometers. In most cases, preliminary checking of these instruments consisted of setting and testing alarm contacts for operation at the desired temperatures.

9. Remote Temperature Instrumentation

Both iron-constantan thermocouples and resistance temperature detectors are employed as remote temperature-sensing elements. These are predominantly factory-manufactured units. Preliminary checking consisted of lead polarity verification, although thermocouples made up at the site were calibrated in a water bath against a mercury thermometer. The 48-point thermocouple selector pushbuttons were checked for correct correspondence between the various thermocouples and the selector pushbutton labeling. Several errors in lead polarity were found and corrected, particularly in the case of the condenser air temperature measurement where groups of 12 couples are connected in parallel.

10. Resistivity

Water resistivity is measured at two points in the purification system by means of temperature-compensated flow cells and a single-point indicator in conjunction with a selector switch. Preliminary testing consisted only of a wiring check. After plant startup, the resistivity indications were checked against determinations with a portable resistivity meter. The results were in good agreement.

The raw water demineralizer employs a resistivity bridge controller for actuation of a motor-operated valve. This system was tested to assure proper valve operation.

11. Process Panel Instrumentation - General

In addition, and prior to the specific system tests described above, all process panel instrumentation was checked to insure proper operating condition of local batteries, vacuum tubes, standardizing circuits, etc. All instrument-controlled positioning systems for valves, etc., were tested for correct direction of drive versus control parameter variation, correct remote position-indicator operation, proper total-drive travel, etc. All interlocking relay and manually controlled circuits were tested for proper operation. Operation of the annunciator was tested by opening the various alarm contacts to assure the proper sequence of annunciator alarm actions.

Q. Nuclear Instrumentation (Figs. 39 and 40)*

The nuclear instrumentation and reactor control systems were given preliminary tests as outlined below. In Fig. 15 is given a block diagram of the neutron flux and air ejector exhaust fission product activity instrumentation.

1. Instrument Channels I and II

A separate battery-powered, standard current source was used to test the ion chamber current meters and the high-flux shutdown relays. The ion chamber high-voltage supplies were checked for proper operation of the high-voltage interlock relay. The built-in test current circuits were tested for proper sequence of operation of the high-flux shutdown circuits. The calibration of the amplifier-type micromicroammeter used with Channel II was checked by applying known test currents to its input circuit.

2. Instrument Channels III and IV

The negative compensating voltage for the compensated ion chambers used with these channels was adjusted so that the ion chamber

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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 VII 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. CHANNEL 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

FIG. 39
NUCLEAR INSTRUMENTATION PANEL

1. CHANNEL III HIGH VOLTAGE SUPPLY
2. CHANNEL IV HIGH VOLTAGE SUPPLY
3. REACTOR SHUTDOWN PUSH BUTTON
4. CHANNEL V HIGH VOLTAGE SUPPLY
5. CHANNEL VI HIGH VOLTAGE SUPPLY
6. CHANNEL VII HIGH VOLTAGE SUPPLY
7. CHANNEL V, VI PREAMPLIFIERS
8. COOLING FAN

FIG. 40
NUCLEAR INSTRUMENTATION -
HIGH-VOLTAGE SUPPLY PANEL

output current was zero with the chambers exposed to the reactor source antimony rod portion only. This was done with the chambers located in temporary instrument wells inside the reactor pressure vessel. The log amplifier and indicator calibration settings were made using the built-in test current source. The reactor period amplifier and indicator were tested using a period simulator. Calibration of the various current ranges of the linear level indicating amplifier used with Channel IV was checked using the separate standard current source. The log power and linear power recorders were adjusted to produce instrument pen deflections in agreement with the associated indicating meters.

3. Instrument Channels V and VI

With the BF_3 counter tubes located in temporary instrument wells in the reactor pressure vessel and with the reactor source antimony rod in place in its beryllium block, a plot was made of counting rate versus applied high voltage. The voltage was then set slightly above the point at which the counting rate ceased to increase with applied voltage. The antimony rod was then withdrawn from the beryllium block, and the pulse

height selector in the linear amplifier was set at a point slightly above the point where gamma ray counts were eliminated.

4. Instrument Channel VII

This channel measures the activity of the air ejector exhaust gases for the purpose of detecting the presence of gaseous fission products. The scintillation crystal, photomultiplier tube and preamplifier are located adjacent to the air ejector aftercondenser vent line.

As a result of initial tests with a cesium-137 gamma-ray source, a coaxial signal cable was installed to replace the signal conductor in the multiconductor power-supply cable.

To permit identification of the source of any radioactivity detected by this instrument during operation of the plant, the pulse-height analyser was calibrated. Using the 0.66 Mev gamma rays emanating from the cesium source, plots were made of count rate versus base-line setting over the operating range of the pulse-height analyser with various amplifier gain and band width settings. The final settings yield a range of from zero to 1 Mev, corresponding to pulse-height analyser base lines from zero to 100 volts.

5. Area Radiation Monitors

This system employs five sensing units at various points in the plant with associated indicator and alarm units in the reactor control panel. One unit, located in the reactor water purification system vault (containing the resin columns and heat exchanger), is equipped with an integral calibration source. The remaining units were calibrated at 1.0 and 10 mr/hr, using a separate radioactive source.

6. Reactor Shutdown Circuits

Reactor shutdown may be initiated by pushbuttons, by power failure or by various abnormal conditions as detected by nuclear or process instrumentation. All instrument-initiated shutdowns are indicated on the shutdown annunciator. The shutdown circuits were individually tested by opening the initiating contacts and observing that the proper shutdown action resulted.

7. Control Rod Drive Motor Controls

The controls were tested for proper operation of the motor-starting contactors, and for proper interlocking among the manual control switches, control rod drive limit switches, and reactor shutdown circuits (see Fig. 41).

FIG. 41
CONTROL ROD DRIVE SCHEMATIC DIAGRAM

R. Control Rod Drives (Figs. 4 and 14)*

The control rod drives were assembled and disassembled with the pressure vessel head and the temporary support used during the zero-power experiments. Testing was also done on both mountings.

Each drive was checked as to electrical performance. All terminals of the drive were brought to a common plug-in with the control room cable. Indicator lights and the position-indicator transmitter were tested.

Each mechanism was checked for scram time. Two mechanisms required the removal of a retarding scram spring (negator spring) in order to comply with the scram time requirement of two (2) seconds. All other operational tests were satisfactory.

S. Emergency Power Supply System (Figs. 42 and 43)**

Alternating-current power for certain critical instrument and control functions is provided by a separate motor-generator unit consisting of an ac motor and generator, and a dc motor-generator. During normal operation, the ac generator is driven by the ac motor which is connected to the normal power distribution system. Concurrently, the dc machine functions as a generator to maintain the charge of the station battery. Upon failure of the ac supply voltage the dc machine becomes a motor and drives the ac generator. Operation of the system during a power failure was tested by opening the ac supply circuit breaker with the machine running normally. The restoration of power was simulated by re-closing the ac supply circuit breaker. Automatic operation was satisfactory.

The system also includes provision for switching the ac generator load to the normal distribution system. Phase rotation was checked to ensure correct rotation of motors energized by this supply.

T. Diesel-generator Unit (Fig. 44)**

The diesel-generator is a 60-kw unit manufactured for military use and loaned by the US Air Force. It is completely self-contained except for external fuel and exhaust line connections. Before running the unit, it was checked for proper lubrication and coolant water levels, starting battery charge, etc. After starting, the operations of the various engine instruments, speed governor, and voltage regulator were tested. The correctness of the external electrical connections was verified using the three synchronizing lamps supplied in the generator instrument panel.

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**H. H. Hooker

1. A.C. VOLTMETER
2. A.C. AMMETER
3. D.C. VOLTMETER
4. D.C. AMMETER
5. A.C. CONTROL SWITCH
6. FREQUENCY METER (A.C. OUTPUT)
7. MAINTENANCE STAND-BY SWITCH
8. D.C. VOLTMETER SWITCH
9. BATTERY CHARGE RHEOSTAT
10. D.C. CONTROL SWITCH
11. AUTO VOLT ADJUST RHEOSTAT
12. CHARGE INDICATOR LIGHT
13. STANDBY POWER INDICATOR LIGHT
14. BELL RESET PUSH BUTTON
15. DISCHARGE INDICATOR LIGHT
16. A.C. OVERLOAD RESET BUTTON
17. D.C. OVERLOAD RESET BUTTON
18. INPUT BREAKER
19. A.C. OUTPUT BREAKER
20. D.C. BREAKER

FIG. 42
EMERGENCY POWER SUPPLY PANEL

FIG. 43
EMERGENCY POWER SUPPLY SCHEMATIC DIAGRAM

With the unit running, a test was made to determine its ability to start the steam condenser air circulating fan. The fan drive motor is rated 75 hp and is the largest single auxiliary load in the plant. Since, after starting, the fan may run at light load during reactor plant startup, it was believed that the plant could be started with the aid of any electrical supply capable of starting the fan motor. (This later proved to be the case; see Experiment No. 147, Section VI-C.)

The first attempts to start the fan motor using the diesel-generator resulted in immediate actuation of the diesel-generator circuit breaker undervoltage trip device. When the undervoltage trip mechanism was subsequently blocked, the fan motor started successfully. During starting, the line voltage, as indicated at the generator, dropped from 120 to approximately 100 volts.

U. Turbine-generator Unit (Figs. 45 and 46)*

Although the maximum capacity of the portable steam boiler was 2000 lb/hr steam flow at 300 psig (see Section III-A), the testing of the turbine-generator unit was complete except for loads over 20 kw electrical and/or transient loads. The unit was operated with an electrical load of 30 kw for a short period of time. A typical log of operating data is reproduced in Appendix C.

1. Turbine and Auxiliaries

The turbine and auxiliaries noted herein, except for the oil cooler and oil temperature control valve, were tested under full load at the factory as part of the overall testing of the complete turbine-generator unit.

a. Turbine Bearings and Lubricating Oil System

All turbine and reduction gear bearings were removed from the casings, cleaned, and the bottom bearing halves were replaced. The reduction gears were manually rotated and cleaned with a brush and flushing oil. The turbine and gear casing sumps were thoroughly wiped and one 54-gal drum of flushing oil was added. The oil was heated to approximately 130F and the entire system (Fig. 47) was flushed by circulating the hot oil for 8 hours by use of the motor-driven auxiliary oil pump. (To heat the oil, the plant heating hot water boiler system was connected to the water side of the turbine auxiliary cooler.) Operation of the reduction gear oil spray nozzles was faulty, but became satisfactory upon removal of crud from inside the nozzles.

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FIG. 45
TURBINE-GENERATOR UNIT

FIG. 46
LONGITUDINAL SECTION THROUGH
MULTI-STAGE STEAM TURBINE

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. 47
TURBINE LUBRICATING OIL SYSTEM

The dirty flushing oil was completely drained from the oil sump, oil coolers, and all piping. The turbine and gear casing oil sumps were wiped clean. All bearing caps and the gear casing cover were removed. The lower halves of all bearings were rolled out, thoroughly cleaned, and the entire unit was then re-assembled and made ready for operation. Approximately 70 gal of lubricating oil were added to the turbine sump tank and the oil system was tested satisfactorily.

b. Turbine Alignment

Immediately after installation, the turbine-generator unit was aligned by the general construction contractor. An alignment check prior to anticipated operation revealed that realignment was required. It is assumed that the addition of the biological gravel shielding material after the initial turbine alignment produced a movement of the building and turbine-generator bedplate. The unit was realigned in the presence of the Worthington Corporation representative and with the use of jacking bolts and shim stock.

c. Reduction Gear

The turbine was slowly brought to rated speed for the first time. After approximately 20 min of warming-up operation, the gear casing cover adjacent to the main reduction gear bearing thrust face became excessively hot and noisy. The unit was immediately shut down and the reduction gear bearings disassembled for inspection. Both thrust faces were "wiped" on the main reduction gear bearings and slight "wiping" was found on the bottom half of the generator end main reduction gear bearing. All of the "wiped" bearing metal was scraped from the bearings and the unit was re-assembled. Thrust clearance readings were then taken and found to be 0.009 in., which was less than the required 0.012 in. Both thrust bearing faces were again removed, machined, reinstalled, and the clearance was found to be 0.015 in. The next turbine operation test proved that all bearings performed satisfactorily.

d. Turbine Operation

The second turbine test indicated a tendency for the turbine to overspeed. The governor valve (Fig. 48) was disassembled and the double-seated valve was lapped in the valve body seat. Also, the two Mason-Neilan turbine gland seal steam control valves (Fig. 49) were adjusted during this test, after which the system operated in a satisfactory manner.

The third turbine test again indicated faulty governor-valve operation by overspeeding the turbine. The governor valve and seat cage were again removed and lapped. A "bluing" check indicated excellent seating contact of the valve. A check of the governor valve stem indicated a

FIG. 48
SECTION THROUGH GOVERNOR MECHANISM,
EMERGENCY TRIP ASSEMBLY AND OIL PUMP

FIG. 49
 TURBINE-GENERATOR GLAND SEAL
 AND LEAK-OFF SYSTEM

"bowing" of approximately 0.010 to 0.012 in. A new stem was ordered. Subsequent tests indicated that replacement of the stem was not necessary.

The fourth turbine test again indicated overspeeding of the turbine. Further inspection revealed the turbine steam drains to be improperly installed and without individual isolating valves, making it possible for steam to bypass the governor valve. The drain piping was modified to correct this deficiency and the governor valve then controlled turbine speed satisfactorily.

e. Turbine Lubricationg Oil Pump

During subsequent turbine tests at rated speed, the auxiliary oil pump idled in a reverse direction, indicating a leaking check valve. The check valve was disassembled and was found to have a $\frac{3}{16}$ -in. hole drilled through the valve body from the discharge to the inlet side. This hole was plugged with a tapered pin to stop the backflow of oil. The Worthington drawing requires two 1-in. check valves in the system, one each on the discharge side of the main and auxiliary pumps. The check valve on the gear-driven main pump has an orifice. It is assumed that the two valves were interchanged when installed at the factory. During the next test the check valve operated satisfactorily.

The turbine lubricating oil system requires the auxiliary oil pump to start at a pressure of 5 psig and stop at 6 psig. Operation was at approximately $5\frac{1}{2}$ psig. The operating pressure is maintained by an adjustable pressure-relief valve on the discharge side of the auxiliary oil pump. The relief valve had to be set almost totally closed in order to build up an oil pressure of approximately 25 psig on the discharge side of the pump to overcome the loss in head in the long run of oil piping to the oil coolers and to obtain an oil pressure of 5 to $5\frac{1}{2}$ psig at the turbine bearings. The system performed well except on shutdown, when the turbine speed decreased, the main oil pump could not maintain a bearing oil pressure of 5 psig, and the auxiliary oil pump started. With both pumps operating the pressure quickly reached 6 psig and the auxiliary oil pump stopped thereby causing the oil pressure to decrease. This cycling of the auxiliary oil pump was overcome by increasing the upper limit of the pressure control range to 7 psig.

f. Air-cooled Oil Cooler

The air-cooled oil cooler was found to be installed in an inverted position, thereby positioning the flanges on the lower side instead of the upper side, as indicated on the drawings. It was believed that air was trapped in the cooler, thus decreasing its efficiency. This belief was substantiated by the temperature rise in the oil and also by the "gurgling" sound in the cooler while in operation. The cooler was taken out and a vent

installed. Approximately 30% of the air passage on the oil cooler was plugged with building insulation and tar. The air fins were cleaned and the cooler reinstalled. To avoid modification of the existing piping, the cooler was again installed in the inverted position and with the newly installed vent on top. The system was then tested and operated in a satisfactory manner.

g. Oil Temperature Control Valve

The Fulton Sylphon temperature controlled valve was calibrated to close at 115 to 120F in order to force oil through the oil coolers. Automatic operation of this valve during startup allows the high-pressure cold oil to pass directly from the pump discharge to the oil header, thereby bypassing the coolers. This causes the auxiliary oil pump to stop on high-pressure cut-out and results in cycling. The $\frac{1}{4}$ -in. stroke of the temperature-controlled valve was decreased to approximately $\frac{1}{16}$ in., so that the increased pressure drop through the valve resulted in a pressure of 6 psig in the header. The pump cycling was thereby eliminated.

2. Generator and Exciter (Electrical Tests)

The generator and exciter were also tested under full load conditions at the factory as part of the overall testing of the complete turbine-generator unit.

a. Preliminary Tests

Before running the generator, the operation of the following devices was tested:

1. throttle trip solenoid valve;
2. speed-changer motor;
3. main generator circuit breaker, field circuit breaker, bus tie circuit breaker, generator transfer switch, and bus tie switch, including operation of the various key interlocks;
4. reactor shutdown actuation of generator circuit breaker trip circuit.

b. Observations during Turbine Warmup

While the turbine was being gradually brought up to operating speed, the following observations were made:

1. The remote tachometer indicator was observed to indicate.

2. The operation of the auxiliary turbine lubricating oil pump automatic cutout was observed with rising oil pressure.
3. The exciter voltage was observed to build up.

c. Tests at Operating Speed

Before paralleling the generator with the utility system, the following tests and observations were made (see Figs. 50, 51, and 52):

1. The exciter field rheostat was set to produce an open-circuit excitation voltage of 150 volts, and this setting was marked.
2. After closing the generator field circuit breaker, the generator voltage regulator was tested for ability to maintain generated voltage at 120 volts ac.
3. Remote control of turbine speed was tested by operating the speed changer switch.
4. At approximate synchronism, the generator output voltage was compared with the utility line voltage at the generator circuit breaker terminals for phase rotation, and for consistency with the synchroscope and synchronizing light indications. The generator frequency meter was observed to indicate approximately 60 cycles.

d. Parallel Operation

The first attempts to operate the generator in parallel with the utility system resulted in immediate tripping of the generator circuit breaker, due in turn to the tripping of one of the generator differential relays. The operation of these three protective relays depends upon a comparison of the currents at the two ends of each of the three generator armature windings. Two sets of three current transformers are used to obtain instrument currents proportional to generator-winding currents. It is necessary that the connections between each differential relay and its two associated current transformers be correct both as to phase and polarity.

Since previous stages of the testing indicated that no ground faults existed in the generator and generator bus, it was concluded that the relays were connected incorrectly to the current transformers. Accordingly, the generator was connected to the isolated equipment bus and was loaded with one 10-kw artificial load unit. The resulting load current was insufficient to trip the differential relays, but permitted a measurement to be made

FIG. 51
GENERATOR CONTROL SCHEMATIC WIRING DIAGRAM

FIG. 52
GENERATOR EXCITATION SCHEMATIC

of the current transformer secondary currents and their vector summations at the relay terminals. These measurements indicated that all connections to one set of three current transformers were reversed and that connections to two of the transformers were interchanged. After correcting these connections, the generator was successfully paralleled with the utility system.

It was anticipated that the above changes in current transformer connections might result in incorrect operation of the generator wattmeter, watthour meter, and power factor meter. This was confirmed later during reactor power operation when higher load currents were available. Appropriate corrections were made at that time.

V. ZERO-POWER EXPERIMENTS (August 11, 1958 to October 10, 1958)*

This section of the report describes the organization and program for the zero-power experiments. The organization described herein was used not only for the experiments, but also functioned during all subsequent reactor tests. The organizational chart is illustrated in Fig. 3.

It is not intended that the program of the zero-power experiments be detailed here.⁽⁴⁾ Instead an outline of the mechanics of the program and of the experiments performed, as extracted from the plant log book and core loading charts, is presented. It should be noted that reference to a five-control rod configuration always refers to the center five cross-type control rods. Use of a nine-control rod configuration required the addition of four tee-type rods in the outer region. The control rod and fuel assembly locations are illustrated in Fig. 20. A photograph of the control room is reproduced as Fig. 53.

FIG 53
CONTROL ROOM -
ZERO POWER EXPERIMENTS

Experiments No. 1 thru 98 were performed with all nine control rods in the pressure vessel. During experiments requiring the use of only five control rods, the tee rods were clamped in a full-out position. For experiments carried out at above atmospheric pressures, tee rod and drive No. 4

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were used to pull the antimony rod from the beryllium block. Therefore, the tee rod was in full-out position when the source was in a "pulled" position. It should be noted that the 40-assembly reference core refers to the availability of the five cross control rods for control.

A. Operating Instructions for Nuclear Tests

The following instructions were used during the experimentation and also during subsequent operations.

1. Organization and Responsibilities

A Coordinator will have overall responsibilities for the nuclear tests to be performed. Responsible staff people will be notified when the approved alternate is temporarily in charge.

A Chief Operator will be responsible for all electrical systems and instrumentation used in performing nuclear tests. He will be in charge of the control room and all operations performed there. Reactor Operators who perform manipulations and take data in the control room will be supervised by the Chief Operator.

A Chief Physicist must be present at all times when alterations are being made to the reactor core or when operations which could affect reactivity are being performed in the control room or on the reactor. Approved alternates may serve in the absence of the Chief Physicist.

All fuel for the reactor will be under the control of a Chief Loader. The Chief Loader will be responsible for an accurate inventory of the fuel at all times and for making all changes to the reactor core.

An independent check of alterations made to the reactor core will be made by a staff member designated by the Coordinator to verify that all fuel loadings are exactly in accordance with the approved loading plan. This check must be completely independent of the one made by the Chief Loader.

Before any fuel is loaded into the reactor or any alterations made to fuel loadings already in the reactor, a loading plan must be prepared by the Chief Physicist and reviewed by the Coordinator, the Chief Operator, and the Chief Loader. Only after this loading plan has been approved by the Coordinator and Chief Physicist will the Chief Loader take any action on the new experiment.

Approvals of the Chief Operator and Coordinator are required for any changes in the neutron source, neutron detectors, safety and alarm interlocks, or nuclear instrumentation.

Laboratory personnel will be responsible for insuring that proper precautions are taken against radiation or other reactor hazards. A Health Physicist will be responsible for conformance with all AEC regulations pertaining to radiation safety. He will be responsible for maintaining records for personnel exposure, area surveys, waste disposal, and other matters relating to health physics.

Keys and key switches are used to lock out certain critical items of equipment on the control panel to prevent unauthorized manipulation. The keys will be controlled by the Coordinator.

These keys will be issued by the Coordinator. A record will be made when the keys are issued and to whom. At the termination of activities on a given day, all keys must be returned to the Coordinator.

2. General

The security control of the site will follow the practice of the Idaho Division of the Laboratory. The entire site will be under security control during the period when nuclear tests are being performed. Access to the building will be controlled from the site entrance. An approved list of people who may enter the site at any time will be supplied by the Coordinator and modified as necessary to keep the list as short as possible without seriously hampering the work. If persons not on this list wish to enter the site, the guard will contact the Coordinator. If the Coordinator feels that conditions in the building at the time are such that it is impossible or inconvenient to admit the person, entry will be denied.

Personnel selected by the Coordinator will be stationed in the Support Facilities Building to prevent entry of personnel to the control room and to the operating floor. Upon authorization of the Coordinator, construction and other personnel, if necessary, will be admitted to do types of work that previously have been reviewed and approved by the Laboratory.

A list of approved personnel, Reactor Operators, and alternates for each of the major areas of responsibility will be posted in the control room log. The list will be expanded from time to time as other personnel gain experience in performing tests of this kind.

3. Routine Checks

a. Daily Checks

During the period of nuclear testing the daily checks will include the following:

1. Inspect the control room and reactor building. Report any abnormalities. Maintain good house-keeping and remove all unnecessary equipment.
2. Check the diesel-generator oil supply, start the diesel-generator, operate the unit for a few minutes, and shut off.
3. Check to see that the building ventilation system is operating.
4. Check the solution level in the boric acid tank. Also check for leakage.
5. Check the setting of the period meter.
6. Check the setting of the high-flux trip circuits.
7. Pull the antimony rod out of beryllium block and take readings on the nuclear instruments. A significant change should occur in all readings of nuclear instruments. Restore the antimony rod to the beryllium block and verify that the nuclear instrument readings return to their former values. Care must be exercised to insure that adequate shielding is above the antimony source as it is being withdrawn.
8. Check to see that the emergency door to the outside stairway can be opened easily.
9. All pertinent operations connected with the preparation of experiments are to be recorded in the log book kept in the control room by the Chief Operator.

b. Pre-startup Checks

Before starting any experiment the following steps are to be carried out:

1. Make sure that only approved personnel are in the control room and the reactor building.
2. Verify that power is connected to the plant transformer and energize all electrical buses.

3. Check to insure that the emergency power system is operating.
4. Check reactor water level to verify that it is at the desired level.
5. Check the radiation monitors to verify that all radiation levels are normal.
6. Check reactor water quality. Determine if boric acid is present, and, if so, the quantity by chemical analysis.
7. Determine that the dummy fuel assembly containing the neutron source is in the proper core location and that the antimony rod is inside the beryllium block.

4. Control Tests

Prior to beginning multiplication or critical experiments on any given day, operation of the control rods is to be checked. Each rod is to be raised to the all-out position and dropped by pushing the "rod drop" button at the control console. The resulting insertion time should be less than two seconds. If any rod requires more than two seconds, the test should be repeated twice. If the rod drop time still exceeds two seconds on either of these two tests, the rod should be considered defective. If any rod is defective, no multiplication or critical tests are to be performed until the trouble is corrected.

5. Multiplication or Critical Experiments

All experiments will be performed in the following manner:

1. Two qualified Reactor Operators will be present.
2. A fuel-loading plan will be prepared by the Chief Physicist and reviewed by the Coordinator, Chief Operator, and Chief Loader.
3. The approved fuel-loading plan will be given to the Chief Loader.
4. The Chief Operator will announce on the building communication system that an experiment is about to begin and will turn on power to the control rod drives.

5. Rods will be raised to a predetermined configuration chosen so that enough control is available to shut down the reactor by insertion of the central control rod alone, or by the insertion of at least two off-center cross control rods.
6. The Chief Loader will proceed to load the fuel assemblies, one at a time, into the water-filled core until the presented loading is achieved. The Chief Operator will be notified over the communication system as each fuel assembly enters the core and when each assembly is in place. The loading will be checked to verify that it is exactly according to the approved loading plan.
7. An announcement will be made over the communication system that all personnel must evacuate the reactor building. The Coordinator will check the reactor building to make certain that evacuation is complete.
8. The Chief Operator will withdraw the control rods slowly, while listening to the counters and watching the linear flux recorder. A Reactor Operator stationed next to the Chief Operator is responsible for changing ranges on nuclear instruments as ordered. The control rod drive motors must be stopped while instrument ranges are being changed. If a significant change in the counting rate occurs, control rod motion is stopped while accurate instrument readings are taken and recorded.
9. When necessary, the source and instrument tube locations will be changed by the Chief Loader in accordance with instruction by the Coordinator.
10. All pertinent data are to be recorded in the log book. The time when the operations were performed are to be recorded together with any significant observation of the results of the operations.
11. Experiments are to be numbered consecutively and are to be described clearly in the log book. Other pertinent data, such as fuel-loading charts, counting rates, and nuclear calculations, are to be neatly filed under the respective experiment or run numbers.

B. Preparations for General Cold Criticals

The reactor core shroud, control rods, pressure vessel head, shield blocks, and control rod drives were installed during the week of July 21, 1958 (see Section III). The entire assembly was checked for clearances, and the control rods were operated and drop times were measured to insure satisfactory operation. The shielding, rod drives, and pressure vessel head were then removed.

The antimony-beryllium neutron source was installed in the reactor core on August 5, 1958 (see Section III-K).

The temporary I-beam control rod drive support and rod drives were installed on August 5, 1958 (see Section III-F-3). The temporary instrument tubes were placed in the reactor pressure vessel, and the nuclear instruments were installed in these tubes and checked with the source (see Section III-J).

C. List of Experiments

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
1-7	Approach to minimum critical loading (10 fuel assemblies with no boron strips. Experiment No. 7 attained criticality).	8/11/58
8	Relative rod worth in 10-assembly core (no boron strips).	8/12/58
9-18	Determination of maximum controllable core loadings.	8/13/58 8/14/58
19-25	Approach to initial 40-assembly core with boron poison strips and 5 control rods	8/14/58 8/19/58
26	Relative rod worth in initial 40-assembly core (Loading No. 19).	8/20/58
27-30	Water hole experiments in 40-assembly core.	8/21/58
31-37	Preliminary determination of worth of 26-mil boron strip in various core locations in initial 40-assembly core.	8/21/58

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
38	Preliminary determination of worth of 21-mil boron strips in central cell of initial 40-assembly core.	8/21/58
39-44	Approach to initial 59-assembly core with boron poison strips and nine control rods (Loading No. 38).	8/22/58 8/26/58
45	Relative rod worth in initial 59-assembly core.	8/27/58
46-47	Reactivity effect of assembly orientation in initial 59-assembly core.	8/27/58
48-53	Reactivity effects of heating initial 59-assembly core to various temperatures below atmospheric boiling.	8/27/58 9/ 2/58
54	Intermediate loading of 51 assemblies with five control rods.	9/ 2/58
55-63	Control rod calibrations in initial 59-assembly core, with nine control rods, using various concentrations of boric acid in the reactor water.	9/ 2/58 9/ 6/58
64-66	Intermediate loading of 48 assemblies with five control rods.	9/ 6/58 9/ 9/58
67-83	Control rod calibrations in initial 40-assembly core using various concentrations of boric acid.	9/ 9/58 9/13/58
84-85	Worth of 21 and 26-mil boron strips on four central fuel assemblies in initial 40-assembly core with control rods withdrawn to anticipated operating position.	9/13/58
86	Worth of 21-mil boron half-strips on four central fuel assemblies in initial 40-assembly core with control rods withdrawn to anticipated operating position.	9/13/58
87-90	Worth of 26-mil strips in various locations in initial 40-assembly core.	9/15/58 9/16/58

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
91-95	Preliminary measurement of reactivity effects of heating a 40-assembly core (Loading No. 53) to various temperatures below atmospheric boiling with control rods withdrawn to anticipated operating position by regulating the concentrations of boric acid in the reactor water.	9/16/58 9/17/58
96	Worth of 26-mil strip in a single core location; completion of program started in Experiments Nos. 87-90.	9/17/58
97-98	Criticals on 40-assembly loading containing one full-length boron strip per assembly plus half-strips on the central sixteen assemblies (Loading No. 56).	9/17/58

D. Preparations for 40-Assembly Reference Core Hot Criticals and 59-Assembly Reference Core Cold Criticals

In preparation for the next series of experiments personnel:

1. unloaded the fuel;
2. welded boron strips to the fuel assemblies;
3. inserted gold wires into fuel assemblies for flux plots in the 40-assembly reference core;
4. removed all rod drives and temporary mounting framework;
5. modified control rod drive No. 4 and extension rod to serve as a source-puller with control rod in place as a guide (attached gripper to extension rod and disabled drive scram mechanism);
6. removed all control rods.
7. removed follower extensions from rods No. 3 and 7, and closed off ends on control rods Nos. 1, 3, 5, and 7;
8. installed the steam coils;
9. installed rods Nos. 1, 3, 4, 5, 7, and 9;

10. loaded the 40-assembly reference core;
11. installed pressure vessel head;
12. installed rod drives Nos. 1, 3, 4, 5, 7, and 9;
13. installed the water level-indicating float and transmitter;
14. welded the steam coil piping to boiler supply;
15. performed concurrently miscellaneous and electrical work;
and
16. installed blind flanges on unused rod drive locations.

E. List of Experiments for 40-Assembly Reference Core Hot
Criticals.

<u>Experi- ment No.</u>	<u>Description</u>	<u>Date</u>
99-100	Check criticals on final (reference) 40-assembly core loading (Loading No. 57); final boron strip loading <u>welded</u> to side plates of fuel assemblies: one full-length 26-mil strip per assembly plus one half-length 21-mil strip on bottom half of other side plate in central 16 assemblies.	9/23/58 9/24/58
101-102	Rod calibrations in 40-assembly reference core, varying the concentration of boric acid in the reactor water.	9/29/58 9/30/58
103-112	Reactivity effects of heating 40-assembly reference core loading to various temperatures up to 417F. Heating was accomplished by use of a steam boiler with steam coils in the outer reflector of the core. Control rods were raised to the anticipated position for 3-Mw operation, using boric acid for extra control. Gold wires were irradiated for flux mappings in the hot (417F) reactor.	10/1/58 10/3/58

F. List of Experiments for 59-Assembly Reference Core Cold Criticals

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
113-116	Criticals leading to selection of boron-strip loading in final 59-assembly core with five control rods; check critical on this reference 59-assembly core (Loading No. 60). Each of the central 16 assemblies contained $1\frac{1}{2}$, 20-mil boron strips, each strip nominally containing 0.8 gram of boron-10. Each of the 16 assemblies in the next ring contained one full 26-mil strip and one half-length 21-mil strip. Each of 24 assemblies in the outer ring of the core contained one 26-mil strip. Three assemblies without strips and one source assembly filled out the core.	10/6/58
117-118	Measurement of reactivity controlled by the five rods in the cold reference 59-assembly core.	10/7/58

G. Preparation for 59-Assembly Reference Core Hot Criticals

In preparation for the next series of experiments, personnel:

1. removed control rod drive No. 1 and blind flanges from rod positions Nos. 2, 6, and 8;
2. removed the fuel assemblies;
3. welded 0.8-gm boron strips on sixteen of the remaining fuel assemblies;
4. inserted copper wires into the fuel assemblies for flux mapping;
5. loaded the fuel assemblies into the reactor vessel;
6. re-installed rod drive No. 1; and
7. re-installed blind flanges in rod positions Nos. 2, 6, and 8.

H. List of Experiments for 59-Assembly Reference Core Hot
Criticals

<u>Experi- ment No.</u>	<u>Description</u>	<u>Date</u>
119-122	Reactivity effects of heating 59-assembly reference core to operating temperature (417 F). Control rods were withdrawn to anticipated operating position.	10/8/58 10/9/58

VI. POWER OPERATION EXPERIMENTS (October 11 1958 to December 11, 1958)*

After having explored the reactor at essentially zero power and having achieved the desired core loading for power operation, the plant was made ready to operate with steam from the reactor. Preliminary tests had been performed with the use of a portable steam boiler (see Section IV).

This section encompasses (1) operating instructions for the power tests in addition to those instructions already set forth (see Section V-A), (2) preparation, and (3) the list of the major experiments and results. Some experiments were recorded by a Brush Electronics Company instrument and are reproduced as Appendix E.

A. Operating Instructions for Power Tests

The following steps indicate how the approach to power is to be made. (These instructions were in addition to the instructions set forth in Section V-A and were also used during subsequent operation.)

1. The nuclear instruments will be calibrated to read approximate power level by operating the cold reactor at very low power level and comparing the readings of nuclear instruments with the thermal power level as calculated from the measured rate of temperature rise of the reactor.
2. The high-flux trips will be set to allow reactor power level to be raised to approximately $\frac{1}{2}$ Mw.
3. Valves in the plant will be set to bypass excess steam in accordance with the more detailed procedure.^(2,5) The reactor power level will be regulated to provide for a gradual increase in pressure to 300 psig. The reactor power level will be raised to about $\frac{1}{2}$ Mw and held while all equipment and instruments are being checked.
4. The power level will be raised in increments of from $\frac{1}{2}$ to 1 Mw. After each increment the power will be leveled off to note overall operation. This will be continued until the 3-Mw power level is reached or indications of abnormal operation are noted. Any abnormalities which would have an adverse effect on safety must be resolved before proceeding to higher power levels. The safety circuits will be advanced in steps to correspond to power increase.

* W. C. Lipinski

5. The power generating equipment will be placed in operation.(2,5)
6. Scrams and alarms are outlined in Table 5.

Table 5

OPERATING CONDITIONS RESULTING IN REACTOR
SHUTDOWN OR ANNUNCIATION

- NO x YES

Item	Operating Conditions	Conditions Causing Annunciator Operation	Conditions Causing Automatic Reactor Shutdown
1	Reactor neutron flux, high - Channel I	x	x
2	Reactor neutron flux, high - Channel II	x	x
3	Reactor neutron flux ion chamber high-voltage supply, Channel I, deenergized	x	x
4	Reactor neutron flux ion chamber high-voltage supply, Channel II, deenergized	x	x
5	Reactor water level, high or low	x	x
6	Reactor steam pressure, high	x	x
7	Main steam pressure-relief valve, steam flow	x	x
8	Main steam safety valve, steam flow	x	x
9	Condenser pressure, high	x	x
10	Reactor period, short	x	x
11	Actuation of manual "Scram" pushbutton	-	x
12	Control power, deenergized	-	x
13	Rod drive clutch rectifier, failure	-	x
14	Bypass steam flow, high	x	-
15	Bypass valve discharge pressure, high	x	-
16	Condenser outlet air temperature, high	x	-
17	Condenser inlet air temperature, low	x	-
18	Feed pump discharge pressure, low	x	-
19	Purification water temperature, high	x	-
20	Air ejector after condenser temperature, high	x	-
21	Shield cooler outlet water temperature, high	x	-
22	Hotwell water level, high or low	x	-
23	Feed pump, automatic switchover	x	-
24	Reactor pressure, low	x	x
25	High-pressure condensate tank water level, high	x	-
26	Low-pressure condensate tank water level, high	x	-
27	Turbine oil inlet temperature, high	x	-
28	Thermal shield temperature, high	x	x
29	Shield cooler pump discharge pressure, low	x	-

B. Preparation for Nuclear Power Operation

In preparation for the next series of experiments, personnel:

1. removed all control rod drives;
2. cut the steam coil piping to boiler;
3. removed the pressure vessel head;
4. unloaded the core;
5. removed the source;
6. removed control rod No. 4;
7. removed the steam coils;
8. installed purification system piping changes (see Section VI-I-2a);
9. cleaned the pressure vessel, core support structure, and control rods;
10. installed the source;
11. loaded the core;
12. installed the pressure vessel head;
13. installed rod drives Nos. 1, 3, 5, 7, and 9; and
14. hydrostatically tested the pressure vessel.

Concurrent with the work noted above, the portable steam boiler was removed and the reactor pressure vessel was connected to the rest of the plant.

C. List of Experiments and Results (reference 40-assembly core loading)

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
123-125	Check criticals on reference 40-assembly core and measurement of cadmium ratio of U ²³⁵ foils. ⁽⁴⁾	10/23/58

Experiment No.	Description	Date																				
126-130	Series of critical experiments at various water temperatures using nuclear heat to increase reactor water temperature. ⁽⁴⁾	10/23/58 10/24/58																				
131	At 300 psig, reactor power was increased in steps of 1000 lb/hr to 9150-lb/hr steam flow rate. Check the feedwater regulating valve, feedwater flow rate versus valve position.	10/24/58																				
	<table><tr><th>Valve (% open)</th><th>Feedwater Flow Rate (lb/hr)</th></tr><tr><td>0</td><td>0</td></tr><tr><td>17</td><td>1750</td></tr><tr><td>20</td><td>2200</td></tr><tr><td>26</td><td>2800</td></tr><tr><td>34</td><td>3750</td></tr><tr><td>39</td><td>4500</td></tr><tr><td>43</td><td>5350</td></tr><tr><td>49</td><td>6600</td></tr><tr><td>58</td><td>7950</td></tr></table>	Valve (% open)	Feedwater Flow Rate (lb/hr)	0	0	17	1750	20	2200	26	2800	34	3750	39	4500	43	5350	49	6600	58	7950	
Valve (% open)	Feedwater Flow Rate (lb/hr)																					
0	0																					
17	1750																					
20	2200																					
26	2800																					
34	3750																					
39	4500																					
43	5350																					
49	6600																					
58	7950																					
132	Full-power run of 40-hr to establish equilibrium xenon. Boric acid was added in increments in order to withdraw control rods to the anticipated equilibrium xenon full-power rod position. Boric acid was gradually removed by the purification system to hold power constant without moving control rods. ^(1,6)	10/28/58 10/29/58 10/30/58																				
133	Held reactor pressure at 300 psig with nuclear power and no steam flow from reactor pressure vessel for 8 hours and then increased reactor power to 8000-lb/hr steam flow rate.	10/30/58																				
134	"Popped" condenser safety valve at 4 psig.	10/30/58																				
135	Shutdown with reactor high-pressure trip at 332 psig.	10/30/58																				
136	Shutdown with condenser high-pressure trip at 3.1 psig.	10/30/58																				
137	Shutdown with reactor low-pressure trip at 251 psig.	10/30/58																				

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
138	Shutdown with reactor water low-level trip and re-set to minus 15 in.	11/3/58
139	Made preliminary check of 3-element feedwater control operation. Feedwater regulating valve oscillated, but satisfactory for use until it could be conveniently corrected.	11/3/58
140	Measured steam flow versus center rod position starting with a steam flow rate of 600 lb/hr and rod bank at 17.6 in.	11/3/58

Rod No. 1, 3, 5, and 7 at 17.6 in.

<u>Rod No. 9 (in.)</u>	<u>Steam Flow Rate (lb/hr)</u>
17.6	600
18.0	1600
18.5	2350
19.0	2950
19.5	3350
20.0	3900
21.0	4800
22.0	5520
23.0	6050
24.0	6800
25.0	7400
26.0	7900

141	Measured steam flow versus center rod position, starting with steam flow rate of 4000 lb/hr and rod bank at 19.0 in.	11/3/58
-----	--	---------

Rod No. 1, 3, 5, and 7 at 19.0 in.

<u>Rod No. 9 (in.)</u>	<u>Steam Flow Rate (lb/hr)</u>
19.0	4000
18.0	2750
17.0	1420
16.0	500
20.0	6000
21.0	7000
22.0	7950

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>																
142	Measured steam flow versus center rod position, starting with a steam flow rate of 8000 lb/hr and rod bank at 20.2 in.	11/3/58																
	<u>Rod No. 1, 3, 5, and 7 at 20.2 in.</u>																	
	<table><tr><th><u>Rod No. 9 (in.)</u></th><th><u>Steam Flow Rate (lb/hr)</u></th></tr><tr><td>20.1</td><td>7950</td></tr><tr><td>19.0</td><td>6300</td></tr><tr><td>18.0</td><td>4500</td></tr><tr><td>17.0</td><td>3150</td></tr><tr><td>16.0</td><td>1800</td></tr><tr><td>15.0</td><td>350</td></tr></table>	<u>Rod No. 9 (in.)</u>	<u>Steam Flow Rate (lb/hr)</u>	20.1	7950	19.0	6300	18.0	4500	17.0	3150	16.0	1800	15.0	350			
<u>Rod No. 9 (in.)</u>	<u>Steam Flow Rate (lb/hr)</u>																	
20.1	7950																	
19.0	6300																	
18.0	4500																	
17.0	3150																	
16.0	1800																	
15.0	350																	
143	Measured steam flow versus control rod No. 1 position, starting with a steam flow rate of 8050 lb/hr and control rod No. 1 at 30.0 in. and control rod Nos. 3, 5, 7, and 9 at 19.6 in.	11/3/58																
	<u>Rod Nos. 3, 5, 7, and 9 at 19.6 in.</u>																	
	<table><tr><th><u>Rod No. 1 (in)</u></th><th><u>Steam Flow Rate (lb/hr)</u></th></tr><tr><td>30.0</td><td>8050</td></tr><tr><td>25.0</td><td>7300</td></tr><tr><td>20.0</td><td>5600</td></tr><tr><td>15.0</td><td>3650</td></tr><tr><td>10.0</td><td>1630</td></tr><tr><td>5.0</td><td>800</td></tr><tr><td>0.0</td><td>800</td></tr></table>	<u>Rod No. 1 (in)</u>	<u>Steam Flow Rate (lb/hr)</u>	30.0	8050	25.0	7300	20.0	5600	15.0	3650	10.0	1630	5.0	800	0.0	800	
<u>Rod No. 1 (in)</u>	<u>Steam Flow Rate (lb/hr)</u>																	
30.0	8050																	
25.0	7300																	
20.0	5600																	
15.0	3650																	
10.0	1630																	
5.0	800																	
0.0	800																	
144	Diverted feedwater flow from lower spray ring to upper spray ring with reactor operating at 4580 lb/hr; unsuccessful because of steam hammer in lines (see Fig. 73).	11/3/58																
145	Increased reactor pressure. Main steam safety valve opened at 380 psig and temperature trip indicated 220 F with trip setting of 230 F.	11/3/58																

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
146	Shutdown reactor with reactor water level low at minus 12 in. Set shutdown at minus 15 in. and plus 18 in.	11/3/58
147	Started reactor using diesel-generator and transferred station load from diesel-generator to turbine-generator successfully.	11/4/58
148	Feedwater transient. Closed feedwater valve, starting with equilibrium conditions of 4000 lb/hr of feedwater and of steam. No significant effects were observed other than a 30% decrease in neutron flux and eventual shutdown due to low water level of reactor.	11/4/58
149	Feedwater transient. Increased feedwater flow to 8000 lb/hr, starting with equilibrium conditions of 4000 lb/hr of feedwater and of steam. No significant effects were observed other than a 30% increase in neutron flux and eventual shutdown due to high level of reactor water.	11/4/58
150	Steam flow transient. Increased steam flow to 7000 lb/hr, starting with equilibrium conditions of 4000 lb/hr of feedwater and of steam. Neutron flux decreased 40%. Feedwater flow increases as reactor pressure decreases. Eventual shutdown on low pressure.	11/4/58
151	Steam flow transient. Decreased steam flow to zero, starting with equilibrium conditions of 4000 lb/hr of feedwater flow and steam flow. Neutron flux decreased 40%. Feedwater flow increases as reactor pressure decreases. Eventual shutdown on low pressure.	11/4/58
152	Feedwater transient. Decreased feedwater flow to zero, starting with equilibrium conditions of 8000 lb/hr of feedwater and of steam. Neutron flux level decreased 35%. Eventual shutdown due to low level of reactor water.	11/4/58

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
153	Steam flow transient. Decreased steam flow to zero, starting with equilibrium conditions of 8000 lb/hr of feedwater and of steam. The experiment was terminated when the reactor flux level had increased 70%. Shutdown occurred by high-flux trip before the reactor high-pressure trip point (332 psig) was reached.	11/4/58
154	Feedwater transient. Started second feedwater pump with initial conditions of 8000 lb/hr of feedwater and of steam flow. The feedwater flow increased to 8900 lb/hr and the neutron flux level showed a slight increase.	11/4/58
155	Operated main steam safety valve at 385 psig. The shutdown temperature switch operated at 242 F after a slight delay in heating.	11/4/58
156	Operated main steam pressure-relief valve at 348 psig. Shutdown temperature switch operated at 230 F after a 3-min delay in heating.	11/4/58
157	Preliminary testing of automatic power demand control system based on steam bypass control system. Oscillations in reactor power were excessive.	11/4/58
158	Operated main steam pressure-relief valve at 350 psig. Alarm indicating main steam pressure-relief valve flow occurred 45 sec later. Reactor shutdown 135 sec after main steam pressure-relief valve opened.	11/4/58
159	Calibrated steam bypass valve No. 2. Steam flow versus valve position	11/5/58

<u>Valve (% open)</u>	<u>Steam Flow Rate (lb/hr)</u>
20	1120
30	1680
39	2520
49	3750
56	4600
62	5380

Experi-
ment
No.

Description

Date

159
(Con't)

<u>Valve (% open)</u>	<u>Steam Flow Rate (lb/hr)</u>
68	5850
74	6620
81	7200
90	7500
98	7850
100	8200

160

Calibrated steam bypass valve No. 1. Steam flow versus valve position.

11/5/58

<u>Valve (% open)</u>	<u>Steam Flow Rate (lb/hr)</u>
12	630
16	950
32	2020
43	3450
51	4280
55	4800
60	5250
73	5950
83.5	6220

161

Tested automatic power demand control system based on steam bypass control: four 60-kw steps on and off at approximately 3.5-min intervals with the 4-rod bank (Rod Nos. 1, 3, 5 and 7) at 18.0 in., travel ratio of 4.5, and a dead band of ± 1.0 psi.

11/5/58

<u>Time</u>	<u>Data</u>
1747	Added 60-kw load
1748	Reactor pressure - 300 psig
	Main steam flow - 4400 lb/hr
	Bypass steam flow - 1270 lb/hr
	Rod No. 9 - 21.4 in.
1750	Added 60-kw load
	Main steam flow - 5400 lb/hr
	Bypass steam flow - 1230 lb/hr
	Rod No. 9 - 22.0 in.

Experi-
ment
No.

Description

Date

161
(Con't)

Time

Date

1754	Added 60-kw load		
1757	Main steam flow	- 6200 lb/hr	
	Bypass steam flow	- 900 lb/hr	
	Rod No. 9	- 23.0 in.	
1758	Added 60-kw load		
1801	Main steam flow	- 7360 lb/hr	
	Bypass steam flow	- 800 lb/hr	
	Rod No. 9	- 24.95 in.	
1802.5	Removed 60-kw load		
1804	Main steam flow	- 7400 lb/hr	
	Bypass steam flow	- 1560 lb/hr	
	Rod No. 9	- 24.95 in.	
	Rod No. 9 failed to drive in on this load change		
1805.5	Removed 60-kw load (Rod No. 9 driving in)		
1806.5	Main steam flow	- 6470 lb/hr	
	Bypass steam flow	- 1800 lb/hr	
	Rod No. 9	- 24 in.	
1808	Rod No. 9 driving to 23.5 in with no load change		
1808.5	Removed 60-kw load (Rod No. 9 driving in)		
1811	Main steam flow	- 5250 lb/hr	
	Bypass steam flow	- 1850 lb/hr	
	Rod No. 9	- 22.5 in.	
	Turbine-generator load	- 130 kw	

Experi-
ment
No.

Description

Date

161

(Con't)

Time

Data

1812	Removed 60-kw load	
1814	Main steam flow	- 4150 lb/hr
	Bypass steam flow	- 1950 lb/hr
	Rod No. 9	- 21.6 in.
1814.5	Rod No. 9 driving in with no load change	
1816	Main steam flow	- 3760 lb/hr
	Bypass steam flow	- 1700 lb/hr
	Rod No. 9	- 21.1 in.
	Turbine-generator load	- 60 kw

The dead band on the automatic rod drive was reduced to ± 0.5 psi.

162

Same as Experiment No. 161 except with a ± 0.5 -psi dead band.

11/5/58

Time

Data

1827	Added 60-kw load	
1830.5	Main steam flow	- 4250 lb/hr
	Bypass steam flow	- 1100 lb/hr
	Rod No. 9	- 21.5 in.
	Turbine-generator load	- 120 kw
1831.5	Added 60-kw load	
1834	Main steam flow	- 5750 lb/hr
	Bypass steam flow	- 1400 lb/hr
	Rod No. 9	- 22.8 in.
	Turbine-generator load	- 180 kw
1834.5	Added 60-kw load	

Experi-
ment
No.

Description

Date

162

(Con't)

Time

Data

1836 Main steam flow - 6600 lb/hr
 Bypass steam flow - 1230 lb/hr
 Rod No. 9 - 23.8 in.
 Turbine-generator load - 240 kw

1837 Added 60-kw load
 Main steam flow - 7600 lb/hr
 Bypass steam flow - 1050 lb/hr
 Rod No. 9 - 25.45 in.
 Turbine-generator load - 295 kw

1840 Removed 60-kw load
 Main steam flow - 7000 lb/hr
 Bypass steam flow - 1170 lb/hr
 Rod No. 9 - 24.5 in.
 Turbine-generator load - 245 kw

1844.5 Removed 60-kw load

1847 Main steam flow - 6100 lb/hr
 Bypass steam flow - 1530 lb/hr
 Rod No. 9 - 23.5 in.
 Turbine-generator load - 190 kw

163

Same as experiment No. 162 except with a 2.0 travel
ratio

11/5/58

1907 Main steam flow - 3800 lb/hr
 Bypass steam flow - 1250 lb/hr
 Rod No. 9 - 21.2 in.
 Turbine-generator load - 145 kw

1907.5 Added 60-kw load
 (Rod No. 9 driving out)

1910 Main steam flow - 4850 lb/hr
 Bypass steam flow - 1460 lb/hr
 Rod No. 9 - 22.0 in.
 Turbine-generator load - 145 kw

Experi-
ment
No.

Description

Date

163
(Con't)

<u>Time</u>	<u>Data</u>
1911	Added 60-kw load
1914	Main steam flow - 6000 lb/hr Bypass steam flow - 1650 lb/hr Rod No. 9 - 23.3 in. Turbine-generator load - 200 kw
1918	Main steam flow - 7000 lb/hr Bypass steam flow - 1600 lb/hr Rod No. 9 - 24.6 in. Turbine-generator load - 240 kw
1918.5	Added 60-kw load
1921	Main steam flow - 8200 lb/hr Bypass steam flow - 1460 lb/hr Rod No. 9 - 26.5 in. Turbine-generator load - 300 kw
1922	Removed 60-kw load
1923.5	Main steam flow - 7350 lb/hr Bypass steam flow - 1440 lb/hr Rod No. 9 - 25.1 in. Turbine-generator load - 250 kw
1924	Removed 60-kw load
1926.5	Main steam flow - 6200 lb/hr Bypass steam flow - 1500 lb/hr Rod No. 9 - 23.8 in. Turbine-generator load - 200 kw
1927	Removed 60-kw load Main steam flow - 4960 lb/hr Bypass steam flow - 1340 lb/hr Rod No. 9 - 22.3 in. Turbine-generator load - 145 kw

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
163 (Con't)	<div> <div><u>Time</u></div> <div><u>Data</u></div> </div>	
	1930 Removed 60-kw load Main steam flow - 3850 lb/hr Bypass steam flow - 1460 lb/hr Rod No. 9 - 21.5 in. Turbine-generator load - 95 kw	
	The travel ratio on the automatic rod drive was here- after increased to 2.5.	
164	Same as Experiment No. 162 except with a 2.5 travel ratio.	11/5/58
	2047 Added 60-kw load to check automatic operation.	
	2049 Removed 60-kw load.	
	2052 Rod No. 9 - 21.5 in. Rods Nos. 1, 3, 5, and 7 - 18.0 in. Main steam flow - 3900 lb/hr Bypass steam flow - 1540 lb/hr Turbine-generator load - 95 kw	
	2053 Added 60-kw load	
	2055 Main steam flow - 5100 lb/hr Bypass steam flow - 1820 lb/hr Rod No. 9 - 22.7 in. Turbine-generator load - 145 kw	
	2055.5 Added 60-kw load	
	2057 Main steam flow - 6100 lb/hr Bypass steam flow - 1890 lb/hr Rod No. 9 - 27.7 in. Turbine-generator load - 200 kw	
	2057.5 Added 60-kw load	

<u>Experiment No.</u>	<u>Description</u>		<u>Date</u>
164 (Con't)	<u>Time</u>	<u>Data</u>	
	2059	Main steam flow - 6400 lb/hr Bypass steam flow - 1750 lb/hr Rod No. 9 - 24.8 in. Turbine-generator load - 251 kw	
	2102	Removed 60-kw load	
	2104	Main steam flow - 6300 lb/hr Bypass steam flow - 1780 lb/hr Rod No. 9 - 24.0 in. Turbine-generator load - 200 kw	
	2106	Removed 60-kw load Main steam flow - 5250 lb/hr Bypass steam flow - 1690 lb/hr Rod No. 9 - 22.85 in. Turbine-generator load - 150 kw	
	2107.5	Removed 60-kw load Main steam flow - 4250 lb/hr Bypass steam flow - 1690 lb/hr Rod No. 9 - 21.8 in. Turbine-generator load - 97 kw	
165	Same as Experiment No. 164 except the interval between load changes was 50 sec. Observation indicated the operation to be satisfactory.		11/5/58
166	Same as Experiment No. 164 except with a 120-kw step increase and decrease in load. No overshoot was observed.		11/5/58
167	Same as Experiment No. 164 except with a 180-kw step increase and decrease in load. A reactor pressure drop of approximately 5 psig was noted for the step increase and a 10-psig rise was noted after the step decrease.		

Experi-
ment
No.

Description

Date

168	Measured the variation in the bypass steam flow rate (automatic steam bypass control) with respect to ± 0.5 -psi pressure dead band and travel ratios of 1.0, 1.4, 2.0, 2.5, 3.0, 4.0, and 4.5. Calibrated travel ratio of valve No. 1 bypass steam flow rate.	11/5/58																									
	<table><tr><td>Travel ratio</td><td>1.0</td><td>2.0</td><td>4.0</td><td>4.5</td></tr><tr><td>Dead band (psi)</td><td>± 0.5</td><td>± 0.5</td><td>± 0.5</td><td>± 0.5</td></tr><tr><td>Gain</td><td>-----</td><td>----</td><td>Max</td><td>-----</td></tr><tr><td>Valve position (% open)</td><td>35</td><td>35</td><td>35</td><td>35</td></tr><tr><td>Steam flow rate (lb/hr)</td><td>± 140</td><td>± 250</td><td>± 440</td><td>± 580</td></tr></table>	Travel ratio	1.0	2.0	4.0	4.5	Dead band (psi)	± 0.5	± 0.5	± 0.5	± 0.5	Gain	-----	----	Max	-----	Valve position (% open)	35	35	35	35	Steam flow rate (lb/hr)	± 140	± 250	± 440	± 580	
Travel ratio	1.0	2.0	4.0	4.5																							
Dead band (psi)	± 0.5	± 0.5	± 0.5	± 0.5																							
Gain	-----	----	Max	-----																							
Valve position (% open)	35	35	35	35																							
Steam flow rate (lb/hr)	± 140	± 250	± 440	± 580																							
	<table><tr><td>Travel ratio</td><td>4.5</td><td>3.0</td><td>2.5</td><td>2.0</td></tr><tr><td>Dead band (psi)</td><td>± 0.5</td><td>± 0.5</td><td>± 0.5</td><td>± 0.5</td></tr><tr><td>Gain</td><td>Max</td><td>Max</td><td>Max</td><td>Max</td></tr><tr><td>Valve position (% open)</td><td>25</td><td>25</td><td>25</td><td>25</td></tr><tr><td>Steam flow rate (lb/hr)</td><td>± 425</td><td>± 285</td><td>± 280</td><td>± 235</td></tr></table>	Travel ratio	4.5	3.0	2.5	2.0	Dead band (psi)	± 0.5	± 0.5	± 0.5	± 0.5	Gain	Max	Max	Max	Max	Valve position (% open)	25	25	25	25	Steam flow rate (lb/hr)	± 425	± 285	± 280	± 235	
Travel ratio	4.5	3.0	2.5	2.0																							
Dead band (psi)	± 0.5	± 0.5	± 0.5	± 0.5																							
Gain	Max	Max	Max	Max																							
Valve position (% open)	25	25	25	25																							
Steam flow rate (lb/hr)	± 425	± 285	± 280	± 235																							
	<table><tr><td>Travel ratio</td><td>1.4</td><td>1.0</td></tr><tr><td>Dead band (psi)</td><td>± 0.5</td><td>± 1.0</td></tr><tr><td>Gain</td><td>Reduced</td><td>Less</td></tr><tr><td>Valve position (% open)</td><td>25</td><td>25</td></tr><tr><td>Steam flow rate (lb/hr)</td><td>± 155</td><td>± 245</td></tr></table>	Travel ratio	1.4	1.0	Dead band (psi)	± 0.5	± 1.0	Gain	Reduced	Less	Valve position (% open)	25	25	Steam flow rate (lb/hr)	± 155	± 245											
Travel ratio	1.4	1.0																									
Dead band (psi)	± 0.5	± 1.0																									
Gain	Reduced	Less																									
Valve position (% open)	25	25																									
Steam flow rate (lb/hr)	± 155	± 245																									
169	Shutdown reactor on low water level at minus 15 in	11/5/58																									
170	With control rods Nos. 1, 3, 5, and 7 at 18.0 in. and No. 9 at 30.0 in. steam flow is 9250 lb/hr.	11/6/58																									
172	Same as Experiment No. 164 except with bias 30% lag. Observed satisfactory operation.	11/6/58																									
173	Same as Experiment 172 except with ± 0.25 -psi dead band. Automatic control oscillated excessively.	11/6/58																									
174	Same as Experiment 172 except with ± 0.5 -psi dead band and travel ratio of 2.0. Excessive number of overshoots. Believed to be underdampened.	11/6/58																									

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
175	Recorded reactor noise (flux) at 5750 lb/hr (see Fig. 74)	11/6/58
176	Shutdown reactor when water level was raised to plus 18.5 in.	11/6/58
177	Preliminary investigation of voltage and frequency regulation of the turbine-generator with respect to load changes.	

Turbine-generator droop 10

Time	1435	1437	1439	1440
Load added/removed (kw)	----	fan full speed	+60	+60
Operating load (kw)	55	90.1	148	205
Frequency (cps)	60	60	59.95	59.4
Voltage (volts)	120	121	120	119

Time	1441	1442	1443	1444
Load added/removed (kw)	+60	+60	-60	-60
Operating load (kw)	260	310	255	200
Frequency (cps)	59.4	59.5	60.5	60.5
Voltage (volts)	119	118.5	119.5	122

Time	1445	1446
Load added/removed (kw)	-60	-60
Operating load (kw)	145.5	95
Frequency (cps)	60.4	60.6
Voltage (volts)	122	123

Turbine-generator droop 5

Time	1508	1509	1510	1511
Load added/removed (kw)	+60	+60	+60	+60
Operating load (kw)	150	205	250	310
Frequency (cps)	59.2	58.9	59.4	58.45
Voltage (volts)	119.5	119.5	118.8	118
Recovery time (sec)	32	32.4	35	39

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
177 (Con't)	<u>Turbine-generator droop 5 (Con't)</u>	
	Time 1512 1513 1514 1515	
	Load added/removed (kw) -60 -60 -60 -60	
	Operating load (kw) 255 200 150 90	
	Frequency (cps) 60.45 60.5 60.45 60.46	
	Voltage (volts) 120.5 121.5 122.3 123	
	Recovery time (sec) 23.8 32.6 32.4 39	
	<u>Turbine-generator droop - zero</u>	
	Time 1520 1521 1522 1523	
	Load added/removed (kw) +60 +60 +60 +60	
	Operating load (kw) 150 205 260 310	
	Voltage (volts) 121 120.8 119.5 118	
	Recovery time (sec) 60 60 32.2 32.1	
	Time 1524 1525 1526 1527	
	Load added/removed (kw) -60 -60 -60 -60	
	Operating load (kw) 260 205 150 90	
	Frequency (cps) 60.42 60.42 60.42 60.45	
	Voltage (volts) 120.5 121.5 122.5 122.3	
	Recovery time (sec) 26.1 30 33 37.6	
178	Generator loaded with four 60-kw steps at 10-sec intervals. Reactor shutdown by high-flux trip at 120% of full power. Reset trips to 150%.	11/7/58
179	Generator loaded with four 60-kw steps at 10-sec intervals. Observed satisfactory operation.	11/7/58
180	Generator loaded with four 60-kw steps at 30-sec intervals. Observed satisfactory operation.	11/7/58
181	Investigated automatic power demand control system with zero bypass steam flow. System oscillated.	11/7/58
182	Transient response of steam bypass control system to pressure changes was investigated with various travel ratio settings. Confirmation of Experiment No. 168.	11/7/58

<u>Experi- ment No.</u>	<u>Description</u>	<u>Date</u>
183	Blind startup experiment at 165 F water temperature and a 50-sec average period. It took 7 min to reach 1% of full power. Peak power was 4.4% of full power.	11/10/58
184	Blind startup experiment at 170 F water temperature and a 22-sec average period. It took 180 sec to reach 1% of full power. Peak power was 10% of full power.	11/10/58
185	Blind startup experiment at 177.5 F water temperature and a 12.2-sec average period. It took 90 sec to reach 1% of full power. Peak power was 19% of full power.	
186	Blind startup experiment at 184 F water temperature and a 6-sec minimum period. It took 32.5 sec to reach 1% of full power. Peak power was 25% of full power.	11/10/58
187	Plant Performance Test (see Section VII)	11/19/58 12/11/58
188	Preliminary investigation of automatic power demand control system in which center rod position is proportional to pressure error. A Leeds & Northrup, Series 60, controller was temporarily connected into the rod drive control circuit and a 100% slidewire was installed in the P - P ₀ reactor pressure indicator. Various settings of proportional band and reset were used and the load was incrementally changed in 60-kw steps and finally in 180-kw steps. The control system performed satisfactorily. Plans were then made for a future permanent installation (see Section IX-B).	12/ 4/58

VII. PLANT PERFORMANCE TEST (November 19, 1958 to December 11, 1958)*

The 500-hr plant performance acceptance test was started on November 19, 1958 to ascertain whether or not the plant met the design requirements. Based upon continuous operation, a down-time of 10% was allowed. The test terminated on December 11, 1958 after a total elapsed time of 518 hr, of which $503\frac{1}{4}$ hr had been continuous operation. Chargeable down-time was $11\frac{1}{2}$ hr, or 2.35% (see Section VII-C).

A. Design Requirements

On March 9, 1956, the Army Reactors Branch issued the design requirements for the present ALPR prototype plant. The following are the design requirements applicable to the operation of the plant.

TECHNICAL CHARACTERISTICS ARGONNE LOW POWER REACTOR (ALPR)

March 9, 1956

(Including change No. 1, dated May 1, 1956
and change No. 2, dated June 6, 1956)

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 volt, 3-phase, 4-wire, wye connected
4. Power Factor:
0.8
5. Plant Factor:
0.7

*W. R. Wallin, R. W. Thiel, J. D. Cerchione

Operational Requirements

1. Environmental Conditions

Outside ambient air temperature minus 60F to plus 60F.

The plant should be designed for 60-kw electric overload at 40F outside ambient air temperature (see Item 3).

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 electrical and/or heating loads, parallel operation with the conventional systems will be established, the load shifted to the nuclear plant and the conventional systems 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.

Whenever required and foreseen the shutdown of the nuclear plant will be accomplished manually in accordance with the standard operating procedures.

3. Plant Overload

This plant must be capable of a 60-kw (electrical) overload for a period of one hour or less in each 24-hr cycle. Such a condition 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.

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.

4. Stability and Regulation:

The plant must be stable throughout its operating range, which is defined as any heating load up to 400 kw plus an 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 by more than $\pm 5\%$ from the design rated value with load varying from 20 to 260 kw. Load may be applied in increments as much as 60 kw.
- b. Frequency must not vary by more than $\pm 5\%$ from the design rated value with load varying from 20 to 260 kw.
- c. Voltage may not vary by more than $\pm 2\%$ from the steady-state value at any given load within the operating range.
- d. Frequency may not vary by more than ± 0.1 cps from the steady-state value at any given load within the operating range.
- e. Transient voltages may not vary by more than $\pm 5\%$ from steady-state values. Steady state voltage must be re-established within 5 sec.
- f. Transient frequencies may not vary from steady-state values by more than ± 0.5 cps. Steady-state frequency must be re-established within 5 sec.

If a bypass steam system is utilized to meet the requirement for control, 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 to 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, then 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.

B. Test Conditions

1. Outside Ambient Temperatures

The design requirements specified an outside ambient temperature range of minus 60F to plus 60F. It was expected that the prototype site conditions could not duplicate this temperature range. The anticipated outside ambient temperature range during the plant performance test was 0 F to plus 60F. The actual temperature range during the test is illustrated in Table 6.

Table 6
PERFORMANCE TEST: DAILY OUTSIDE AMBIENT TEMPERATURES
(degrees F)

NOVEMBER			DECEMBER		
Date	Minimum	Maximum	Date	Minimum	Maximum
19	35	39	1	20	47
20	29	45	2	17	44
21	28	50	3	22	55
22	21	48	4	25	41
23	20	40	5	11	38
24	28	47	6	20	41
25	18	39	7	25	36
26	8	47	8	24	36
27	9	31	9	24	30
28	6	29	10	22	34
29	13	38	11	24	30
30	16	41			

2. Plant Load

The design requirements specified an electrical load of 200 kw net with a 60-kw overload for one hour in each 24-hr day when the outside ambient temperature was 40F or below. The 400-kw net space heating load was to be applied continuously. In addition, load change requirements were set forth (see Section VII-A). The test conditions demonstrating the design requirements were as follows:

- a. The plant electrical load was maintained at maximum capacity for 23 hr of each 24-hr day. For one hour of each 24-hr day, the plant electrical load was reduced to station load plus 10 kw. The intent was to demonstrate the ability of the plant to accommodate load changes.
- b. The plant space heating load was maintained at 400 kw net.
- c. Wherever standby equipment is provided the equipment was alternated daily.

During the Power Operation Experiments (see Section VI), it became evident that the experimental air-cooled steam condenser would not be able to maintain the full power capacity expected at an installation in arctic areas. An allowance was made for the difference in condenser performance, and, consequently, plant capacity, due to the difference in elevation of a proposed installation at sea level and the prototype installation at 5000 ft above sea level. New electrical load requirements, therefore, were established for the test.

To accommodate the maximum electrical load of 360 kw (200 kw net plus 100 kw plant auxiliaries plus 60-kw overload) at a sea-level installation, the air-cooled steam condenser would be required to dissipate 7.26×10^6 Btu/hr. At the prototype installation (5000 ft above sea level) the air density factor is 0.832; therefore, the equivalent (to sea level) maximum heat load is 6.04×10^6 Btu/hr for the condenser, or 286 kw plant electrical. Table 7 compares the design requirements with the prototype corrected test conditions.

C. Plant Performance

Although the plant had logged only one continuous operating cycle of 40 hr (Experiment No. 132, Section VI-C), schedule commitments made it desirable to start the 500-hr plant performance acceptance test as soon as possible. The test was begun on November 19, 1958, and was concluded on December 11, 1958, for a total elapsed time of 518 hr. The chargeable plant shutdown time for this period was $11\frac{1}{2}$ hr, or 2.35%.

Table 7

DESIGN REQUIREMENTS (SEA-LEVEL INSTALLATION) VERSUS
ALPR INSTALLATION TEST CONDITIONS
(5000 FT ABOVE SEA LEVEL)

	Winter -60F to +40F		Summer 40F to 60F	
	Sea Level	5000 ft	Sea Level	5000 ft
Turbine-Generator (kwe)	360	286	300	240
Space Heating (kwt)	400	400	400	400
Steam Flow Rate (lb/hr)				
Turbine	7200	6000	6200	5135
Bypass for space heat	1400	1400	-	-
Air ejectors and turbine gland seals	100	100	100	100
Total	8700	7500	6300	5235

The plant electrical and space heat loads were kept as close as possible to 300 kw and 400 kw, respectively. For one hour each day, the plant electrical load was reduced to station load plus 10 kw of dummy load. The record of the electrical load read on the plant wattmeter during the test is presented in Fig. 54. The record of the space heat load is illustrated in Fig. 55. It should be noted that as the outside ambient temperature increased, the air-cooled steam condenser vacuum decreased, thereby necessitating a reduction in the plant electrical load. Also, deviations in the space heat load were the direct result of variations in the plant steam bypass flow rate effected by plant electrical load changes (see Section VII D-5).

A simplified plant heat balance diagram, constructed in part from typical instrument readings and in part from design data, is illustrated in Fig. 56. Instrument charts for one day and log data sheets for one shift (eight hours) of the same day, are reproduced as Appendix D.

The plant shutdowns experienced during the test were as follows:

1. Condenser high pressure (unscheduled shutdown)

The air-cooled condenser fan motor starter tripped on overload. The fan stoppage resulted in a loss of condenser vacuum. Down time was 2 hr.

LEGEND

LETTER	SHUTDOWN	TIME (hr.)
A	CONDENSER HIGH PRESSURE	2
B	REACTOR WATER LEVEL HIGH	1½
C	CONDENSATE CIRCULATING SYSTEM LEAK	7
D	CONDENSER FAN BEARING	1¼
E	DEDICATION CEREMONY	3¼

• STATION AUXILLARY LOAD +10 kw DUMMY LOAD

FIG. 54
ELECTRICAL LOAD DURING PLANT PERFORMANCE TEST

FIG. 55
 BYPASS STEAM RATE AND SPACE HEATING LOAD DURING
 PLANT PERFORMANCE TEST

FIG. 56
TYPICAL ALPR HEAT BALANCE FOR PLANT PERFORMANCE TEST AT 300 KW(e)
NOVEMBER 30, 1958

2. Reactor water level (unscheduled shutdown)

The reactor water level instrument transmitter apparently became inoperative by sticking. This resulted in a false signal and an excess of water was added to the reactor. When the instrument became operative again, the shutdown resulted. Down time was $1\frac{1}{2}$ hr.

3. Condensate circulating system leak (scheduled shutdown)

The reactor power was reduced and the steam system was secured in order to repair a water leak in the line to the feed-water pump gland seal. Also, a valve was added to the line. Down time was 7 hr.

4. Condenser fan bearing (scheduled shutdown)

The reactor power was reduced and the steam system was secured in order to check the cause of noise in the condenser fan pulley jack-shaft bearing. The noise was the result of a chipped bearing race (see Section IV-D). Down time was $1\frac{1}{4}$ hr.

5. Facility dedication ceremony (scheduled shutdown)

The reactor power was reduced and the turbine-generator unit was operated at no-load conditions. Down time was $3\frac{1}{4}$ hr. This time was not charged as plant down time.

The preceding plant shutdowns demonstrated that an average of $1\frac{1}{2}$ hr is required to return the plant to full power operation after a plant shutdown from power. In order to place the plant into operation from cold conditions to full power, a total of 4 hr is required. This time consists of:

- (1) One hour for a plant check
- (2) Two hours for heating the reactor and plant systems to operating temperature
- (3) One hour for testing the control rod operation (rod-drops) at operating temperature and pressure.

D. Systems Performance

1. Reactor Control and Instrumentation

No difficulty was encountered with the flux-level measuring instruments during the test

The instruments for radiation monitoring indicated changes in activity level satisfactorily. Initially, there was some interaction between radiation monitoring channels when calibrating. This was traced to a wire shorted to an instrument chassis.

Two apparent difficulties were encountered with the process system instrumentation.

- a. The malfunction of the reactor water level instrument transmitter produced an unscheduled reactor shutdown (see Section VII C-2). The instrument apparently became inoperative by sticking and thereby producing a false signal.
- b. The main steam pressure indicator transmitter isolating valve had to be closed due to a steam leak at the valve bonnet. This rendered the instrument inoperative for the balance of the test.

The condenser fan and damper occasionally required manual operation. This was necessary because air was entering the condenser at temperatures higher than outside ambient temperature as the result of air recirculation (see Section VII D-6). By lowering the condenser outlet temperature recorder set point during the time of high outside ambient temperature, and with the system operating on automatic, poor condenser vacuum was improved, thereby producing better plant performance. The lowering of the condenser outlet temperature recorder set point increased the condenser fan speed, resulting in an increased air flow which, in turn, decreased the temperature differential across the condenser.

Of the two bypass steam control valves in the system, only the smaller valve (0-3000 lb/hr) was used. Automatic operation of the bypass valve to control the reactor pressure was satisfactory.

The feedwater control valve occasionally drove to the full open position and remained long enough to require the switching from automatic to manual control in order to regain normal operation. Following the plant performance test, valve operation became satisfactory after adjustment of the controller.

The "demand control" system could not maintain the steam bypass flow rate within limits (dead band) narrow enough to prevent control rod drive oscillations.

Following the plant performance test, readjustment of the "demand control" system resulted in excellent system performance.

2. Control Rod Drives

In preparation for the plant performance test the drive mechanism for control rod No. 7 was replaced. This was done because the required maximum drop time of two seconds for the rod was exceeded. (Investigation indicated a binding rack backup roller assembly which was repaired by cleaning.) Also, one each of two constant velocity "neg'ator" springs was removed from rod drive units No. 5, 7, and 9. Previously one "neg'ator" spring had been removed from rod drive units No. 1 and 3.

Rod-drop tests conducted on November 19, prior to the commencement of the plant performance test, were satisfactory. During the entire plant performance test, the rod drive cooling water flow was maintained at approximately 60 gph. Each of the five isolating (and regulating) needle valves in the individual cooling water supply lines to the rod drive seals was open four to five turns. This flow rate resulted in control rod drive seal temperatures averaging from 110F to 140F, as measured on the 48-point temperature indicator in the control room.

Once each day the turbine-generator electrical load was reduced to near station electrical load. This necessitated a five-rod bank movement of approximately one inch. These movements, plus others (including shutdowns), were accomplished without incident.

3. Feedwater Pump

The reactor feedwater pumps presented no serious operating problems during the performance test, although it did become necessary to shut down the plant once to repair piping leaks in the gland seal cooling water system (see Section VII C-3).

The feedwater pumps were alternated at least once each day, in order to provide for equal operating time.

The feedwater pump No. 2 gland seal was repacked once during the test and both pumps had above-normal water leakage out of the glands. It was apparent that this was due to high gland seal water pressure. It was also noted that the gland jacket cooling water system did not increase the pump or pump gland seal performance. Revisions to the feedwater pump gland seal water supply and gland jacket cooling water systems are noted in Section VIII-D.

A minor oil leak developed at the lower motor bearing oil reservoir of feedwater pumps No. 2 during the test. This was later rectified (see Section IV-g).

4. Turbine-Generator Unit

The turbine-generator unit produced no serious self-induced operating problems during the plant performance test and operated satisfactorily with a minimum of attention.

A loss of vacuum in the turbine gland seal air ejector system (see Section VII D-7 and D-8) resulted in the emission of radioactive steam from the turbine gland seals. It became necessary to install an exhaustor and duct near the turbine exhaust gland in order to maintain activity levels within tolerable limits on the plant operating floor. Upon restoration of the vacuum, the gland seals performed satisfactorily.

Occasionally it became necessary to place the water-cooled turbine auxiliary oil cooler in service. This was during periods when the condenser fan was at low speed, the turbine was at idle, and all the plant auxiliaries were operated from the Idaho Power Company supply line.

It was noted that the turbine speed response to load changes did not meet the design requirements. The lag was not objectionable from an operating standpoint. This deficiency was later rectified (see Section IX-A).

5. Simulated Heat Load

The simulated heat load system presented no serious operating problems during the performance test.

It was noted that the three-way water-regulating valve is of limited value in maintaining full load in cold weather. Better control of the return water temperature was achieved by the on-off operation of each of the two finned-tube cooler fans and the adjustment of the cooler dampers.

It was also noted that the 400-kw full load space heat could not be maintained when the bypass steam flow rate was less than 1400 lb/hr or the heat exchanger shell steam pressure was less than 20 psig.

Cooler fan No. 1 and No. 2 motors tripped out twice and once, respectively, apparently because of electrical overload.

A steam leak observed at the steam flow-regulating valve was the result of inadequate tightening of the flanges.

6. Main Steam Condenser and Air Distribution System

The condenser back pressure was found to be roughly 5 to 7 in. Hg higher than anticipated from the original specifications. At ambient air temperatures above 40F, the plant load was reduced accordingly to

limit the air temperature leaving the condenser. This was done by disconnecting some of the electrical load grids (dummy load) but maintaining the test conditions specified in Section VII A.

"Cold spots" were observed near the bottom of one or more of the condenser sections, indicating "plugging" of the sections by either water or air.

Prior to test of plant performance, performance data on the condenser system indicated that condenser performance was approximately 20% below manufacturer's predictions. The analysis of the condenser performance is presented in the Design Report.⁽¹⁾ Details of the test runs for the purpose of eliminating the "cold spots" were made after the plant performance test run and are reported in Section IX-C.

During the first week of the plant performance test, one of the condenser fan pulley jack-shaft bearings became defective, as indicated by noise. Greasing this bearing every few hours with a high-temperature grease made it possible to continue the test.

The motor-operated mixing chamber dampers did not shut tightly, thereby allowing some air recirculation. It was necessary to operate the manual recirculation dampers in addition to the aforementioned motor-operated mixer dampers to maintain proper air temperatures. Gaskets were installed on each damper blade after the plant performance test, and temperature control is now possible using only the motor-operated dampers.

7. Air Ejectors

The condenser air ejector system functioned properly during the test of plant performance.

The turbine gland air ejector system lost vacuum after a condensate return system steam trap became defective (see Section VII, D-4 and D-8). Repair of the steam trap restored the vacuum to normal.

Appreciable quantities of condensate were observed to be present in the air ejector after condenser vent line. Due to this situation, the air flowmeter was inoperative and some radioactive water was observed being discharged to the atmosphere with the vent gases.

8. Condensate Systems

A malfunction in the high-pressure condensate return system resulted in a serious operating and health physics problem during the plant performance test (see Section VII D-4). A loss of vacuum in the

turbine gland air ejector system was traced to a faulty steam trap in the drain line from the high-pressure condensate return tank.

The susceptibility of the condensate circulating pump to cavitation required close observation and the careful balancing of the water flow rates in order to obtain adequate cooling throughout the condensate systems.

Other than the aforementioned, the condensate systems performed satisfactorily.

9. Reactor Water Purification System

On November 6, 1958, a mixed resin NR-11 (Illinois Water Treatment Co. resin designation) and cation resin NR-1 were charged to the mixed bed and cation columns. Reactor water was frequently tested for quality. On November 10, the water resistivity measured 1.3 megohm-cm at pH 6.6.

After the first day (November 19) of the plant performance test, the reactor water quality became poor and on November 20 measured 0.34 megohm-cm at pH 5.6. The NR-11 and NR-1 resins were replaced with new resins on November 20 and 21. Removal of the spent resin was made difficult because of resin obstructions in the pipe lines and resin removal hoses. The obstruction and also the failure of the recirculating line pressure relief valve to open properly caused a momentary pressure buildup and fracture of the contaminated water (retention) flowmeter glass (F-114, Fig. 33) on the 1600-2400 shift of November 20, and subsequently, on the 0000-0800 shift after glass replacement. The flowmeter was then bypassed, using temporary hose connections and blind flanges.

A portable assembly of one-inch piping to extend from the H-1 hose connection at the purification system panel to a sight flow glass placed in the waste storage sump, was constructed. From this point, a hose was used to transfer the spent resin to the resin disposal tank located in the waste storage sump.

A recommendation for minor piping modifications to the purification system to reduce the amount of handling, and total length, of hoses is noted in Appendix A-H-2.

After the resin change, a flow of 3 gpm was maintained through the mixed bed resin starting at 1000 on November 21. At 1015 the resistivity of the reactor water was 0.46 megohm-cm and the purification effluent resistivity was 1.6 megohm-cm. At 1530, the corresponding readings were 0.64 and 1.28 megohm-cm at 6.6 and 6.2 pH, respectively, and during the remainder of the plant performance test satisfactory water conditions were maintained.

On November 21, a water leak at the feedwater filter plug was repaired by welding the plug to the filter body.

The reactor water resistivity before and after ion exchange was read in the control room. The water quality, pH, and resistivity were also measured at least once daily on water samples by means of portable instruments. The results of these measurements were recorded in a special log book retained in the chemistry laboratory. The resistivity measurements thus taken agreed well with those read in the control room.

The feedwater filter was changed November 11 and on December 2, and the purification water filter on December 3.

On December 8, depletion of the mixed bed resin was indicated when the resistivity of the reactor water was measured to be 0.9 megohm-cm at pH 6.3 and the resistivity of the purification system effluent 0.75 megohm-cm at pH 6.5. The resin (NR-11), located in the mixed bed column, was replaced with ease on December 9, using the portable piping assembly.

At 0050 on December 10, failure of the purification system water circulating pump caused inoperation of the purification system. After measuring abnormal pump stator winding resistances, it was reasoned that system water had entered the winding cavity, probably through a minute defect in the stator-can.

The life of the NR-11 resin bed was considerably shorter than the anticipated life of approximately three months. Subsequent chemical analyses of raw water supply, demineralized water supply and reactor water showed excessive amounts of oil. This oil, inadvertently introduced in the raw water supply from an oil-lubricated bearing (see Appendix A-D-16), was deposited on the resin and found to be the cause of the reduced resin life.

10 Plant Make-up Water System

The raw water demineralizer required one resin change during the plant performance test. Previous experience, plus experience during this test, indicated that approximately 2000 gal of 8-grain demineralized water could be made per $1\frac{1}{2}$ cu ft of resin.

An operating error in valve manipulation allowed the addition of raw water to the plant demineralized water storage tank. The raw water was turned on to supply cooling water to the turbine water-cooled auxiliary oil cooler and the demineralizer bypass valve had not been secured. The result was poor quality water in the storage tank. This was corrected by the use of a portable pump recirculating the storage tank water back through the demineralizer.

Figure 57 illustrates the rate of water consumption by the plant during the plant performance test. The average use was 72 gal per day. During the second week of operation, excessive water leakage from the feedwater pump gland seals was noted. A more normal indication of plant water requirements is the average of 55 gal per day used during the first and third weeks of the plant performance test.

11. Electrical Auxiliaries

The diesel-generator was operated each day to assure its availability. Experiments prior to the plant performance test had shown that the plant could be brought to full power using the 60-kw diesel-generator only (see Experiment No. 147, Section VI-C) providing that the undervoltage trip device on the generator breaker was made inoperative while starting the condenser fan.

The Emergency Power Supply functioned as required.

The switchgear and motor-control equipment required routine maintenance only. Two overload trips on the operating floor motor control center were due, in part, to excessive room temperature. The turbine-generator breaker required nearly two seconds to close, which made synchronization difficult. This is inherent in the design of the circuit breaker. The generator voltage varied as much as 3%. Adjustment of the voltage regulator and governor subsequent to completion of plant performance test resulted in satisfactory voltage regulation (see Section IX-A).

The failure of the purification system pump is described in Section VII-D-9.

The electrical equipment performed satisfactorily in all other respects.

12. Shield Cooling System

There was a wide variation in the shield cooling system temperatures. The maximum temperature reached was 200F. The high temperatures occurred during the first week of operation. They were due to improper adjustment of the condensate flow rate to the heat exchanger. After adjustment of this flow rate, the temperatures of the shield cooling water and the lead thermal shield of the pressure vessel ranged from 130F to 170F. The primary cause of this variance apparently was the change in condensate temperature due to changes in the main steam condenser vacuum. The temperature rise across the shield cooling loops ranged from 3F to 10F. Assuming an average flow rate of 10 gpm from the shield cooling pump, it was determined that 5 to 15 thermal kw was being dissipated by the shield cooling system.

WATER ADDED TO HOTWELL (1"=8.3 GALS)

LETTER	GALLONS	INCHES	LETTER	GALLONS	INCHES
A	27.9	3.00	H	49.8	6.00
B	29.0	3.50	I	58.1	7.00
C	33.2	4.00	J	66.4	8.00
D	35.3	4.25	K	70.5	8.50
E	37.3	4.50	L	83.0	10.00
F	41.5	5.00	M	99.6	12.00
G	45.6	5.50	N	107.9	13.00

FIG. 57
WATER COMSUMPTION DURING PLANT PERFORMANCE TEST

On one occasion the shield cooling pump was secured for approximately four hours while the cause of a motor starter overload trip was investigated. No appreciable rise in the shield cooling water outlet temperature was observed, indicating that the natural convection cooling system may be adequate for all cooling needs of this system (see Appendix B-F).

13. Reactor Shield and Instrument Well Ventilation System

This system functioned satisfactorily. Instrument well air temperatures are dependent upon the operating floor air temperatures and the exit air temperature is dependent primarily on the inlet air temperature. The temperature rise in the instrument wells was not more than 5F.

14. Building Heating and Ventilation

The building ventilation fan did not deliver sufficient air at high condenser fan speeds. The building ventilation fan takes air from the inlet duct of the main condenser fan and/or recirculated air from the building ventilation system mixing chamber by means of dampers. The fan was unable to overcome the differential pressure between the operating floor and the condenser fan duct when the condenser fan speed was greater than 90% of full speed. When the outside ambient air temperature rose above 80F it was necessary to open the building freight doors and also the two personnel doors in order to maintain operating floor room temperatures below 100F. When the outside ambient air temperature was below 60F the operating floor room temperature could be kept at satisfactory levels by the ventilating system (see Appendix A-G).

15. Shielding and Health Physics

A typical log of health physics data taken during plant operation is reproduced in Appendix D. Radiation activity levels at various locations in the operating plant were measured hourly. Two plant areas of high radiation levels were found: (1) the floor area near the purification system, and (2) one end of the space heat heat exchanger. The high radiation level at the purification system was due to insufficient shielding in the upper area around the purification vault, i.e., between the vault lid and the normal level of the gravel biological shield. (The purification vault contains the resin columns and heat exchanger.) The activity level was reduced by the use of additional gravel locally (see Appendix A-E-3). The high radiation level at one end of the space heat heat exchanger was due to the accumulation of radioactive gases in the shell section. The activity level was reduced by venting the heat-exchanger shell to the plant flash tank. Shielding was adequate to maintain radiation activity levels in other normal personnel occupancy areas below 10 mr/hr (see Appendix A-E-2).

Contamination of the floor area at the purification system panel resulted from a "spill" during the changing of resin (see Section VII D-9). The area was cleaned by removing the contaminating particles with a vacuum cleaner and wiping the area with rags.

Radiation exposure of personnel was kept well below allowable tolerance levels.

During one period of plant operation, air contamination resulted from steam leaks at the turbine gland seals. The area activity level was made tolerable by installing an exhauster until the situation was remedied by the repair of the malfunctioning equipment (see Section VII D-4).

E. Plant Supervision and Operation

A minimum of three men per shift is deemed necessary for normal plant operation. The military officers in charge requested that two additional men be assigned for training. Therefore, a total of five military personnel were on duty per shift, i. e., one Chief Operator, two Reactor Operators, and two Trainees. Because the time was not available for qualifying Reactor Operators and Chief Operators before the start of the plant performance test, a Laboratory staff engineer was present on each shift during the first week of operation, for supervision and training.

When sufficient military personnel had been trained to function in shifts, they were given full responsibility, although the Laboratory staff engineers remained available by telephone. During the final two weeks, the plant was operated by military personnel.

It must be noted that these military people had specific related qualifications before entering the program, i. e., they had experience on other reactors (see Section X-A-1) and they were closely associated with the ALPR plant throughout its development and construction. These factors contributed heavily to their ability to take charge of plant operation with a minimum of on-site operating experience.

Two deficiencies were apparent in the availability of expert personnel of a particular specialty:

- 1) For proper maintenance it is imperative to have a part-time welder experienced in work with stainless steels.
- 2) Only one military health physics specialist was assigned to the project. In order for one man to fulfill the needs of the prototype plant, more training should be given to operating personnel and closer attention must be given to regulations set down for health and safety. For a

future remote installation all operating personnel would need to know safe practices and possible hazards involved in working with reactors and possible radioactive contamination.

The Plant Performance Test demonstrated the need for plant improvements and further thought in several areas (see Section VIII, and Appendices A and B). This was not unexpected, however, in view of the very limited operating background that had been accumulated prior to the performance test. Under these conditions, a down time of only 2.35% is considered to be excellent.

An important consideration in securing this operating record was the fact that despite the variety of troubles encountered the plant was kept in service.

A number of plant improvements have been accomplished already and an additional number have been proposed. The improvements accomplished have put the plant in condition such that it meets design requirements except for the main steam condenser (see Section VIII)..

Another point important to the projected end use of this prototype plant was also demonstrated, in part, during the performance test, i. e., the plant appears to be suited for the use of virtually complete automatic control, with only part-time attendants required. These attendants could be personnel having other primary responsibilities. The number of nuclear and power plant specialists needed would be quite small.

VIII. PLANT MODIFICATIONS (December 12, 1958 to February 5, 1959)*

Prior to and during the Plant Performance Test (Section VII), plant deficiencies were noted. The items necessitating immediate action are reported in the preceding sections. Other items were deferred for correction and/or incorporation at this time. The following are the additions, changes and/or corrections made in the plant after the plant performance test and until the termination of the Laboratory's operation of the ALPR facility. Plant maintenance is not discussed.

A. Condenser Air Handling and Distribution System

1. Condenser Air Handling Duct

An extension to the condenser air handling system air intake and exhaust ducts (Fig. 58) was installed in order to separate the two air flows. This modification diminished the air "short circuiting" experienced during earlier plant operation. A temporary plywood structure was used until the permanent modification could be made.

2. Condenser Mixing Chamber Dampers

A series of stiffening members and weather strip gaskets were installed on the motor-operated mixing chamber dampers in order to reduce the recirculation of air while the dampers are in the closed, or a "no recirculation," position. Tests performed during four days of operation, January 28 through January 31, 1959, proved the gaskets to be effective. The condenser inlet temperature was maintained at the outside ambient air temperature with the motor-operated mixing chamber dampers closed and the manual air recirculation dampers open.

3. Condenser and Condenser Precooler

Valves were installed at each of the condenser precooler air ejector connections and, also, in the main branch of the condenser air ejector line. This work was in preparation for the condenser performance test (see Section IX-C). By throttling the valves a balanced heat distribution within each condenser section, and also for the condenser as a unit, was believed possible.

4. Condenser Fan

The condenser fan jack-shaft pulley with a pitch diameter of 10.6 in. was replaced by a pulley with a pitch diameter of 10.2 in. The purpose of this change was to provide for a built-in limit on fan speed and to increase operating band width of the fan speed controller.

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FIG. 58
CONDENSER AIR HANDLING DUCT EXTENSION

B. Air Ejector Aftercondenser Vent

1. Flowmeter

A moisture separator was installed ahead of the air ejector aftercondenser vent flowmeter and drainage was provided both above and below. This installation was to provide accurate flow measurement by eliminating the condensate within the flowmeter. The installation did not perform satisfactorily. Consequently, the flowmeter, moisture separator, and incidental piping were removed. The air ejector aftercondenser vent was then connected directly to the fission product detection hold-up tank. Recommendations for future revision are noted in Appendix A-D-3.

2. Radiological Filter

A housing for the air ejector aftercondenser vent radiological filter was constructed and installed. Monometer connections are provided for checking purposes.

C. Control Rod Drives

1. Support

The rod drive motor and transmission support for drive No. 9 was equipped with anchor bolts to eliminate observed "rocking" and vibration.

2. Guards

Sheet metal enclosures, with hinged covers, were installed on each rod drive transmission unit to protect the transmission, limit switches, selsyns, and other integral parts.

D. Feedwater Pump Gland Seals

The gland seal water supply system for the feedwater pumps was modified to incorporate a pressure-reducing valve. The purpose was to reduce the amount of pump gland seal water leakage without decreasing the pump seal efficiency.

Also, the pump gland cooling system piping was observed to be unnecessary and was eliminated.

E. Reactor Water Purification System

1. Contaminated Water (Retention) Flowmeter

The flowmeter, F-114, was replaced with a sight flow indicator originally in the feedwater pump gland cooling system. Experience indicated the advisability of replacing the flowmeter by a higher pressure unit (see Section VII-D-9).

2. Resin Removal

The employment of hoses during the removal of the purification system resins was unsatisfactory because of leakage, spillage, and plugging (see Section VII-D-9). Most of the hoses were replaced by temporary steel piping. Recommendations for permanent installation are noted in Appendix A-H-2.

3. Hose Rack

A hose rack was fabricated and installed on the building wall, adjacent to the freight door, and behind the purification system panel. This was done in the interest of good housekeeping and to identify hoses according to their customary usage.

4. Purification System Pump

The cooling water jacket of the purification system pump was connected to the reactor thermal shield cooling system to reduce the stator winding temperature, thereby increasing the winding lifetime.

5. Shielding

Additional gravel shielding was placed around the upper end of the purification system vault (see Section VII-D-15). The gravel was then resurfaced with the mastic seal "Nataz." During the four-day operational test, January 28 through January 31, 1959, the maximum radiation level observed in the personnel area was 20 mr/hr (see Appendix A-E-3).

F. Shield Cooling System

The shield cooling system solenoid-operated bypass and drain valve, SC-8 (Fig. 29), was disconnected from the plant power supply and connected to the emergency power supply. This change eliminates the unnecessary opening of the valve and resultant loss of water during incidental plant power failures.

G. Tool Board

A small tool board for a limited number of hand tools was installed adjacent to the plant make-up water supply system. The tools are for the convenience of operators during the "tours" of the reactor operating and fan floors.

H. Condenser, Building Ventilation and Reactor Shield and Instrument Well Exhaust Fans

The fan controls were revised to stop on loss of electrical power and to remain so after power is restored. This arrangement meets the needs and conditions of emergency shutdown and plant evacuation.

I. Motor Control Center

The pushbutton stations on the motor-control center, located on the reactor operating floor, were mounted on the doors of the starters. This provides for a more satisfactory operation of the pushbuttons than when the units were mounted on the fixed portion of the control panel and protruded through the starter door.

J. Reactor Automatic Control System

Modifications were made to the reactor automatic control system, producing a satisfactory operating system (see Section IX-B).

K. Simulated Heat Load System

A permanent water make-up line was installed between the plant water supply system and the water side of the simulated heat load system. This was done as an operating convenience.

L. Freight Door

Door batteners were installed along the sides of the plant freight door. The intent was to reduce air leakage into the plant. While an appreciable improvement was noted, an unsatisfactory air leakage rate still exists. Recommendations for further improvement are noted in Appendix A-D-14.

M. Supports

1. Turbine Seal Steam Supply

An angle iron brace was installed between the turbine seal steam supply pipe and turbine instrument panel to eliminate the observed vibration of the pipe.

2. Bypass Steam Flowmeter Sealing Chambers

An angle iron brace was installed between the support bracket of the bypass steam flowmeter sealing chambers and the one-inch bypass valve drive support to eliminate the observed vibration of the sealing chambers.

N. Guards

Guards were fabricated and installed over all the exposed machine couplings.

IX. ADDITIONAL PLANT PERFORMANCE TESTS (January 5, 1959 to January 9, 1959)

Several major areas of plant performance needed further investigation and/or correction as the result of information acquired during the Plant Performance Test (Section VII). Because the time schedule dictated an early plant performance test, some experimentation was deferred until after the test. The turbine-generator frequency and voltage regulation, reactor automatic control, and condenser performance tests are discussed in this section.

A. Turbine Generator Frequency and Voltage Regulation*

The following paragraphs (1) through (6) are excerpted from "Technical Characteristics, Argonne Low Power Reactor" (see Section VII-A), and pertain to the turbine generator voltage and frequency regulation.

1. Voltage may not vary by more than $\pm 5\%$ 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.
2. Frequency must not vary by more than $\pm 5\%$ from the design rated value with load varying from 20 to 260 kw.
3. Voltage may not vary by more than $\pm 2\%$ from the steady-state value at any given load within the operating range.
4. Frequency may not vary by more than ± 0.1 cps from the steady-state value at any given load within the operating range.
5. Transient voltages may not vary by more than $\pm 5\%$ from steady-state values. Steady-state voltage must be re-established within 5 sec.
6. Transient frequencies may not vary from steady-state values by more than ± 0.5 cps. Steady-state frequency must be re-established within 5 sec.

Paragraphs (3) and (4) refer to variations occurring under constant electrical load conditions, while paragraphs (5) and (6) refer to variations occurring as a result of load changes of as much as 60 kw. The load referred to in these paragraphs is net electrical load. A range from 20 to 260 kw in net load corresponds approximately to a range from 90 to 350 kw in gross generator load.

*H. H. Hooker

In order to test the ability of the turbine governor to respond to changes in load, recordings were made of turbine speed (in terms of frequency) during a series of stepwise changes in electrical load of approximately 60 kw (unity power factor) each. The measurement of speed was made using the circuit shown in Fig. 59.

FIG. 59
FREQUENCY-MEASURING CIRCUIT

The first such recording showed that, while the initial transient frequency deviation following each load change was not excessive, an excessive time was required for the recovery of steady-state frequency.

An inspection of the turbine governor valve and linkage revealed that the valve stem was binding within its guide. This was evidently due in part to a dark reddish coating on both the stem and guide, and to a slight bending of the stem. Upon removing the coating and straightening the stem, the valve and linkage operated freely. The governor response test described above was then repeated, but again did not yield satisfactory results.

The turbine governor was then removed and another installed in its place. After minor adjustments were made, the oscillograph recordings shown in Fig. 61 and 62 were obtained. These figures also include a simultaneous recording of generator output voltage. The voltage measurement was made using the arrangement shown in Fig. 60.

FIG. 60
VOLTAGE-MEASURING CIRCUIT

FIG. 61
EFFECT OF ELECTRICAL LOAD CHANGES ON ALPR GENERATOR VOLTAGE AND FREQUENCY

FIG. 62
EFFECT OF ELECTRICAL LOAD CHANGES ON ALPR GENERATOR VOLTAGE AND FREQUENCY

These recordings indicate that, under the conditions of the test, the voltage regulation falls well within the limits specified in paragraphs (1), (3), and (5) above, and that the frequency regulation falls within those specified in paragraphs (2) and (4). However, when the generator is more heavily loaded, as when load increments (3) and (4) in Figure 61 were applied, the transient frequency regulation did not meet the specification in paragraph (6). Additional testing is required to show whether this is a question of further adjustment of the governor or whether it is inherent in the present system.

Although the 60-kw load increment contemplated in the specifications represents a unity power factor heating load, another possible source of appreciable transient voltage disturbances exists in the plant itself. If the discharge pressure of the feedwater pump in use should drop below a set pressure, the standby pump will start automatically. These pumps are driven by 15-hp motors which are equipped with reduced-voltage starters. Although the motor starting current is probably less than the current corresponding to a 60-kw resistance load, the starting power factor is low and the resultant voltage drop (to be compensated by the voltage regulator) is relatively greater. Voltage regulation under these conditions has not as yet been measured (see Appendix A-C-5).

B. Reactor Automatic Control*

After preliminary investigation of the automatic power demand control system in which the position of the center control rod (rod No. 9) is proportional to the reactor pressure error (see Experiment No. 188, Section VI-C), a permanent installation of the Leeds & Northrup, Series 60 controller in the rod drive control circuit was made. Also installed was a 20% slidewire in the P-P₀ reactor pressure indicator, which then nullified the automatic operation of steam bypass valve No. 2. This valve is normally closed and deletion or relocation is recommended (see Appendix A-D-1).

The experiments listed below are experiments performed with the aforementioned automatic control system. Included as the first two experiments (Experiment No. 189 and 190) are rod calibration and generator frequency and voltage experiments, respectively. Results of the experiments are noted or referenced. The Brush recorder charts for most of the experiments, including (1) rod position, (2) rod (motion) direction, (3) (steam) pressure, (4) steam flow rate, and (5) (neutron) flux level, are reproduced as Appendix E.

*W. C. Lipinski

Experiment No.	Description	Date														
189	Measured steam flow versus center rod position; the 5-rod bank was at 19.3 in. for a steam flow rate of 6000 lb/hr.	1/6/59														
	<div><div>Rod No. 1, 3, 5, and 7 at 19.3 in.</div><table><tr><th>Rod No. 9 (in.)</th><th>Steam Flow Rate (lb/hr)</th></tr><tr><td>16.0</td><td>2400</td></tr><tr><td>17.0</td><td>3900</td></tr><tr><td>18.0</td><td>5200</td></tr><tr><td>19.0</td><td>6200</td></tr><tr><td>20.0</td><td>7300</td></tr><tr><td>21.0</td><td>8300</td></tr></table></div>	Rod No. 9 (in.)	Steam Flow Rate (lb/hr)	16.0	2400	17.0	3900	18.0	5200	19.0	6200	20.0	7300	21.0	8300	
Rod No. 9 (in.)	Steam Flow Rate (lb/hr)															
16.0	2400															
17.0	3900															
18.0	5200															
19.0	6200															
20.0	7300															
21.0	8300															
190	Adjusted turbine-generator unit governor and recorded generator frequency and voltage variation with load changes. Results are noted in Section IX, A and Figs. 61 and 62.	1/9/59														
191	Step response of the automatic power demand system in which the center rod position is proportional to the reactor pressure error. The proportional band was set at 200% with reset at zero. A slight overshoot of neutron flux and a steady-state pressure error resulted (Fig. 75).	1/9/59														
192	Same as Experiment No. 191 except with a proportional band setting of 100%. A large overshoot and dampened oscillation of neutron flux resulted. Also, there was a steady-state pressure error (Fig. 76).	1/9/59														
193	Same as Experiment No. 191 except with a proportional band setting of 50%. A large overshoot and dampened oscillation of neutron flux resulted. Also there was a steady-state pressure error (Fig. 77).	1/9/59														
194	Same as Experiment No. 191 except with a reset of 0.1. Result was that reactor pressure slowly returned to 300 psig (Fig. 78).	1/9/59														
195	Same as Experiment No. 191 except with a reset of 0.3. Again the reactor pressure very slowly returned to 300 psig (Fig. 79).	1/9/59														

Experi- ment No.	Description	Date
196	Same as Experiment No. 191 except with a reset of 1.0. The pressure returned rapidly to 300 psig without overshoot (Fig. 80).	1/9/59
197	Same as Experiment No. 191 except with a reset of 2.0. This resulted in the pressure overshooting 300 psig slightly (Fig. 81).	1/9/59
198	Same as Experiment No. 191 except with a reset of 3.0. The pressure overshoot 300 psig and showed a slight tendency to oscillate (Fig. 82).	1/9/59
199	Same as Experiment No. 191 except with a reset of 6.0. The pressure oscillated as a large overshoot (Fig. 83).	1/9/59
200	Same as Experiment No. 191 except with a proportional band setting of 175 per cent and a reset of 1.0. (<u>This appears to be the optimum setting for system response.</u>) The neutron flux overshoot to restore the system energy lost in the initial interval when reactor power output was less than load demand (Fig. 84).	
201	Same as Experiment No. 191 except the amplifier gain was increased to 10 and gradually decreased until the oscillations stopped.	1/9/59
202	Same as Experiment No. 191 except with a proportional band setting of 175% and a reset of 1.0. Three 60-kw load steps were introduced at 60-sec intervals. The pressure was restored to 300 psig in 60 sec (Fig. 85). Note: The ROD DIRECTION chart provides erroneous information under certain conditions. The control slidewire in the P-P ₀ pressure indicator is a proportional control over ± 4 psi. Limit switches of ± 4 psi are used to run the rod "in" when the pressure is greater than 304 psig and to run the rod "out" when the pressure is less than 296 psig. When the pressure is returning to the normal control point the electronic controller will program a control rod reversal to restore normal pressure without overshoot or undershoot. If the pressure is outside the ± 4 psi limits, a conflict can arise where the controller commands reverse rod direction. When	1/9/59

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
202 (Cont'd.)	the controller calls for a rod direction opposite to that called for by a limit switch, the chart record of rod direction will be recorded in the "in" direction four (4) chart divisions below the "off" position. The true "in" indication is six (6) divisions below the "off" position. The rod direction record is ambiguous at four (4) divisions below "off" and the rod position record should be for the true direction. Records of Experiments Nos. 203, 204, 205, 206, 207, 208, 209, 210A, and 210B also show this ambiguous condition.	
203	Same as Experiment No. 202 except load increments at 50-sec intervals. The pressure was approximately 299 psig after a load increase and 303 psig after a load decrease, in 50 sec (Fig. 86).	1/9/59
204	Same as Experiment No. 202 except load increments at 40-sec intervals. The pressure reached a minimum of 294.4 psig after the three 60-kw step increases in load at 40-sec intervals and a maximum of 305.2 psig after similar step decreases (Fig. 87).	1/9/59
205	Same as Experiment No. 202 except load increments at 30-sec intervals. The pressure reached a minimum of 292.4 psig after three 60-kw step increases in load at 30-sec intervals and a maximum of 308 psig after similar step decreases (Fig. 88).	1/9/59
206	Same as Experiment No. 202 except load increments at 20-sec intervals. The pressure reached a minimum of 286.8 psig, a neutron flux peak of 130% of equilibrium value, and maximum rod withdrawal to 21.6 in. After three 60-kw step decreases in load, pressure reached a maximum of 314.4 psig, followed by an undershoot to 295.2 psig. The neutron flux decreased to 25% of the equilibrium value followed by an overshoot to 138%. At maximum insertion the center control rod was at 16.2 in. (Fig. 89).	1/9/59
207	Same as Experiment No. 205 except no limit switches at ± 4 psi.	1/9/59

<u>Experiment No.</u>	<u>Description</u>	<u>Date</u>
207 (Cont'd.)	The control slidewire in the $P-P_0$ pressure indicator is a 20% slidewire and will maintain proportional control over ± 4 psi. In the previous experiments limit switches had been used to start the control rod "out" at full velocity if the pressure was greater than 304 psig. The purpose of this experiment was to investigate system performance without the switches at ± 4 psi. As shown in Fig. 90, there is excessive control rod starting and stopping action outside the ± 4 psi limits, but the tendency for the neutron flux to overshoot and undershoot the equilibrium values is decreased.	
208	Same as Experiment No. 206 except no limit switches at ± 4 psi. Excessive control rod starting and stopping was noted (Fig. 91).	1/9/59
209	Same as Experiment No. 202 except with 120-kw load increments. Following a single 120-kw step increase in load, the pressure reached a minimum value of 289.6 psig, neutron flux overshoot to 132% of equilibrium value, and the control rod maximum withdrawal was to 20.3 in. After a single 120-kw step decrease in load, the pressure maximum was 311.2 psig, neutron flux undershot to 37.5% of equilibrium value, and the control rod at maximum insertion was 16.4 in. (Fig. 92).	
210A	Same as Experiment No. 202 except with 180-kw load incremental increase. Following a single 180-kw step increase in load, the pressure reached a minimum value of 278.4 psig, neutron flux overshoot to 139% of equilibrium value, and the control rod at maximum withdrawal was 22.1 in. (Fig. 93).	1/9/59
210B	Same as Experiment No. 202 except with 180-kw load incremental decrease. Following a single 180-kw step decrease in load, the pressure reached a maximum value of 324 psig, neutron flux undershot to 29.4% of equilibrium value, and the control rod at maximum insertion was 15.5 in. (Fig. 94).	

C. Main Steam Condenser-Precooler Tests*

A test run was made to study the effect of throttling the condenser precooler section outlets to the air ejector. This was an effort to improve suction on the precoolers and eliminate "cold spots" (Section VII-D-6) in the condenser sections (Figs. 25 and 63). It was also

FIG. 63
ALPR CONDENSER SECTION

*A. Smaardyk

assumed that a more uniform distribution of condenser and precooler temperatures would increase the condenser performance and the condenser vacuum. Test data covering the result of precooler valve changes are shown in Tables 8 and 9. From these data, the following conclusions and/or observations were made:

1. It was possible to eliminate a "cold spot" in the condenser and warm up a particular segment but only at the expense of moving this "cold spot" into another portion of the condenser; hence, a "cold spot" could be shifted around from segment to segment by throttling the precooler valves.
2. All the condenser faces and precoolers could not be made equally hot at the same time.
3. With all the precooler valves throttled to an opening of one-fourth to one-half turn, it required only a very slight turning on any one valve to vary temperatures on the precoolers.
4. Overall vacuum remained constant during the throttling of the valves except when the valves were sharply throttled to the shut-off point; then the vacuum tended to decrease.
5. It appeared that all suggested changes which might correct or aid in the steam distribution to the segments had been explored, and that the problem was of basic condenser design (see Section VII-D-6).

Table 8

MAIN STEAM CONDENSER - PRECOOLER TEST NO. 1 (JANUARY 7, 1959)

Step No.	Changes	Item	Condenser Segments ⁽¹⁾					Remarks
			1	2	3	4	5	
1.	All valves open	Valve turns ⁽²⁾ Condenser face	5 bottom cold	5 hot	5 hot	5 bottom cold	5 hot	Full power
2.	At this time the valves on segments No. 2, 3, and 5 were throttled in small increments until they were as in step 3.							
3.	Throttled valves ⁽³⁾ No. 2, 3, and 5	Valve turns Condenser face	5 bottom cold	$\frac{1}{2}$ hot	$\frac{1}{2}$ hot	5 bottom cold	$\frac{1}{2}$ hot	
4.	With the valves in position (Step 3) it was still impossible to warm up segments No. 1, and 4 and at this time the reactor was shut down. No further testing was done at power this date. However, after the shutdown with no change in valves, segments No. 1 and 4 became very warm and segments No. 2, 3, and 5 became cold, as in step 5.							
5a.	After shutdown no valve changes	Valve turns Condenser face	5 became hot	$\frac{1}{2}$ cold	$\frac{1}{2}$ cold	5 hot	$\frac{1}{2}$ cold	
5b.	To verify this (Step 5a), valve No. 4 was then closed to $\frac{1}{2}$ turn and this segment quickly cooled off as noted in step 6a.							
6a.	Closed valve No. 4 to $\frac{1}{2}$ turn	Valve turns Condenser face	5 hot	$\frac{1}{2}$ cold	$\frac{1}{2}$ cold	$\frac{1}{2}$ became cold	$\frac{1}{2}$ cold	
6b.	Valve No. 4 was then opened to 5 turns (See step 5a) and it became hot again as noted in step 7.							
7.	Opened valve No. 4	Valve turns Condenser face	5 hot	$\frac{1}{2}$ cold	$\frac{1}{2}$ cold	5 hot	$\frac{1}{2}$ cold	

(1) Condenser segment No. 1 is located in the center of the building and No. 5 at the building exterior wall.

(2) Valve turns are numbered from a closed position.

(3) Throttle valves are identified with their respective condenser segments.

Note: The above tests indicate that an entirely different effect results due to valve position when the power is changed. It was difficult to warm up segments No. 1 and 4 at full power. However, when the power was reduced to a very low value, these two segments immediately became hot, while segment No. 5 became cold with only a $\frac{1}{2}$ -turn valve opening.

Table 9

MAIN STEAM CONDENSER-PRECOOLER TEST NO. 2 (JANUARY 8, 1959)

Step No.	Changes	Item	Condenser Segments(1)					Remarks
			1	2	3	4	5	
1.	All valves open Start of test	Valve turns(2) Condenser face Condenser precooler	5 bottom cold cold	5 hot cold	5 hot cold	5 hot cold	5 hot cold	Warming up turbine
2.	Throttle valves(3) No. 2, 3, and 5	Valve turns Condenser face Condenser precooler	5 hot cold	1 bottom 1' cold cold	1 hot cold	5 hot cold	1 hot ---	Warming up turbine
3.	Opened valve No. 2	Valve turns Condenser face Condenser precooler	5 cold bottom 3' cold	5 cold bottom 5' cold	1 hot ---	5 hot ---	1 hot ---	Warming up turbine
4.	Closed valves No. 3, 4, and 5	Valve turns Condenser face Condenser precooler	5 bottom 1½' cold cold	5 bottom 8' cold cold	1/2 hot hot	1 hot hot	1/2 hot hot	Warming up turbine
5.	Closed valves No. 3, 4, and 5	Valve turns Condenser face Condenser precooler	5 bottom 1' cold cold	5 bottom 3' cold cold	3/8 hot hot	7/8 hot hot	3/8 hot hot	Warming up turbine
6.	No change Increasing load	Valve turns Condenser face Condenser precooler	5 hot cold	5 bottom 1' cold cold	3/8 hot hot	7/8 hot hot	3/8 hot hot	Warming up turbine
7.	No change Increasing load	Valve turns Condenser face Condenser precooler	5 bottom 2' cold cold	5 bottom 4' cold cold	3/8 hot hot	7/8 hot hot	3/8 hot hot	Loading tur- bine to about 4000 lb/hr
8.	Closed valves No. 3, 4, and 5	Valve turns Condenser face Condenser precooler	5 All faces gradually became hot as load increased.	5 3/8	5/8	5/8	3/8	Increasing load
9.	Closed valves No. 3, 4, and 5	Valve turns Condenser face Condenser precooler	5 bottom 2' cold cold	5 bottom 2' cold cold	1/4 hot hot	1/2 hot hot	1/4 hot hot	Increasing load

Table 9 (Cont'd.)
MAIN STEAM CONDENSER-PRECOOLER TEST NO. 2 (JANUARY 8, 1959)

Step No.	Changes	Item	Condenser Segments ⁽¹⁾					Remarks
			1	2	3	4	5	
10.	No change	Valve turns (2) Condenser face Condenser precooler	5 bottom 2' cold cold	5 bottom 3' cold cold	1/4 hot hot	1/2 hot hot	1/4 hot hot	Decreased load to 60 kw
11.	No change	Valve turns Condenser face Condenser precooler	5 All faces became cold	5 hot quickly cold	1/4 warm	1/2 hot	1/4 hot	Increasing load to 8000 lb/hr
12.	Closed valve No. 4(3)	Valve turns Condenser face Condenser precooler	5 hot cold	5 hot cold	1/4 hot warm	1/4 hot hot	1/4 hot hot	Increasing load to 8000 lb/hr
13.	Closed valves No. 4 and 5 Opened valve No. 3	Valve turns Condenser face Condenser precooler	5 bottom 1' cold cold	5 hot hot	3/8 hot hot	1/8 hot hot	1/8 hot hot	at 8000 lb/hr
14.	Closed valve No. 2	Valve turns Condenser face Condenser precooler	5 All faces hot cold	2 hot	3/8 hot	1/8 hot	1/8 hot	at 8000 lb/hr
15.	Closed valve No. 2 and 3	Valve turns Condenser face Condenser precooler	5 All faces hot cold	1-1/2 hot	1/4 warm	1/8 warm	1/4 lukewarm	at 8000 lb/hr
16.	Closed valve No. 2 Opened valve No. 5	Valve turns Condenser face Condenser precooler	5 All faces hot cold	3/4 hot	1/4 warm	1/8 warm	1/4 warm	at 8000 lb/hr
17.	Closed valve No. 2 to 1/2 turn	Valve turns Condenser face Condenser precooler	5 hot cool	1/2 hot warm	1/4 bottom cool cool	1/8 hot warm	1/4 hot hot	at 8000 lb/hr
18.	Opened valve No. 3 to 1/2 turn, closed valve No. 5 to 1/8 turn	Valve turns Condenser face Condenser precooler	5 hot cold	3/8 hot warm	1/8 hot hot	1/8 hot cool	1/8 hot cool	at 8000 lb/hr

Table 9 (Cont'd.)

MAIN STEAM CONDENSER-PRECOOLER TEST NO. 2 (JANUARY 8, 1959)

Step No.	Changes	Item	Condenser Segments ⁽¹⁾					Remarks
			1	2	3	4	5	
19.	Closed valve No. 2 ⁽³⁾ and 3	Valve turns ⁽²⁾ Condenser face Condenser precooler	5 hot cold	3/8 hot warm	1/8 hot hot	1/8 hot cool	1/8 hot cool	at 8000 lb/hr
20.	Opened valve No. 3, 4, and 5	Valve turns Condenser face Condenser precooler	5 bottom 1' cold cold	3/8 hot hot	3/8 hot hot	1/4 hot warm	1/4 hot warm	at 8000 lb/hr
21.	Closed valve No. 2 and 3	Valve turns Condenser face Condenser precooler	5 hot hot	1/4 bottom cool cold	1/4 hot warm	1/4 hot cool	1/4 hot hot	at 8000 lb/hr
22.	Closed valve No. 1 and 5	Valve turns Condenser face Condenser precooler	2 hot hot	1/4 plus hot cool	1/4 hot hot	1/4 plus hot cool	1/8 plus hot hot	at 8000 lb/hr
23.	Closed valve No. 1 Opened valve No. 2 and 4	Valve turns Condenser face Condenser precooler	1 hot ---	3/8 hot ---	1/4 hot ---	3/8 bottom cool ---	1/8 hot ---	at 8000 lb/hr
24.	Closed valve No. 1	Valve turns Condenser face Condenser precooler	3/4 hot hot	3/8 hot warm	1/4 hot cool	3/8 warm cool	1/8 hot hot	at 8000 lb/hr
25.	Closed valve No. 1 Opened valve No. 3 and 4	Valve turns Condenser face Condenser precooler	1/2 hot hot	3/8 hot cool	3/8 hot hot	1/2 hot warm	1/8 hot cool	at 8000 lb/hr
26.	Closed valve No. 1 and 3, Opened valve No. 2 and 5	Valve turns Condenser face Condenser precooler	3/8 --- ---	1/2 --- ---	1/4 --- ---	1/2 --- ---	1/4 --- ---	at 8000 lb/hr

(1) Condenser segment No. 1 is located in the center of the building and No. 5 at the building exterior wall.

(2) Valve turns are numbered from a closed position.

(3) Throttle valves are identified with their respective condenser segments.

X. TRAINING*

A comprehensive training program for the military cadre and the operating contractor, Combustion Engineering, Inc., was a significant part of the overall ALPR program. At the time of transfer of responsibility for the operation of the plant to the operating contractor, on February 5, 1959, qualified operating personnel among the military cadre consisted of eight Chief Operators and five additional Reactor Operators.

A. Military Cadre

1. Previous Experience

Initially fourteen men were listed as cadre for the ALPR. During the calendar year 1957 these men were assigned to various reactors for operating experience: eight to EBWR, five to BORAX and two to MTR, for periods of six months to one year.

Beginning late in 1957, and for about six months during the construction of ALPR, these people, under the guidance of their military officers, were engaged in classroom training in the various items of equipment to be a part of the plant. They used drawings and instruction books available at the time. This classroom training program has been continued for newly arrived military and civilian operating contractor personnel.

As construction of the plant was approaching completion, the cadre worked under Laboratory personnel supervision in testing and checking out plant equipment. This included the calibration of instruments, testing of the steam system with an auxiliary portable boiler and the operation of pumps with their associated piping systems. The installation of core support structure and rod drives was also accomplished by the cadre. All areas of work were accomplished, or assisted, by the military cadre.

2. Operation Training

During the beginning of zero-power experiments the cadre worked under the supervision of Laboratory personnel in operating the control console. They also assisted in loading fuel assemblies into the core, mounting burnable poison strips on fuel assemblies and making many modifications of core loading.

*W. R. Wallin

At the completion of the zero-power experiments, the plant was prepared for power operation. Following the first experiment at power, crews of military personnel operated the plant under the direct supervision of a Laboratory staff engineer. After approximately two weeks of on-the-job instruction and observation by Laboratory staff engineers, it appeared that the military crews were sufficiently well qualified to run the plant on their own. From the end of November to the time of the ALPR facility transfer, the military cadre accepted full responsibility to the Laboratory for operation of the plant with the Laboratory staff engineers acting as advisors.

A written examination was given to the cadre following the Plant Performance Test. This examination was in three parts, titled Plant Operator, Reactor Operator, and Chief Operator. To qualify for a rating, the individual had to pass the test at the appropriate level in addition to carrying out his duties in an acceptable manner. The ratings of the various cadre personnel are listed in Table 10.

Table 10

REACTOR OPERATION RATINGS OF THE MILITARY CADRE

Name	Reactor Operator	Chief Operator
WO-3 R. E. Embleton	X	X
WO-1 D. K. Brown	X	X
WO-1 W. C. Mathers	X	X
CPO H. J. Miller	X	X
M/Sgt. H. M. Curran	X	X
M/Sgt. M. D. Hobson	X	X
M/Sgt. R. C. Lewis	X	X
M/Sgt. J. A. Parks	X	X
T/Sgt. M. V. Sitton	X	
SFC R. M. Bishop	X	
SFC T. J. Conlon	X	
SFC R. L. Honeycutt	X	
SP-5 E. O'Niel	X	

3. Training of Student Personnel

In the late summer of 1958, eight military personnel arrived at the site and were trained in the classroom program. At the completion of that program, they participated in operation and maintenance of the plant. It is anticipated that with additional experience these men can be qualified as Plant Operators.

B. Operating Contractor Training

In preparation for transference of the operation of the ALPR to Combustion Engineering Inc., the new operating contractor, a brief training program was conducted. The Combustion Engineering personnel were given a short classroom course by the military cadre. They performed zero-power experiments under the supervision of Laboratory staff engineers and qualified military Chief Operators. The operating personnel for Combustion Engineering worked with the military crews during four days of continuous plant operation in the last week of January. During the two weeks in which Combustion Engineering personnel were in training, frequent discussions and inspections were conducted in which Laboratory staff engineers transmitted detailed information on operation of the plant equipment.

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APPENDIX A

RECOMMENDATIONS FOR IMPROVED
PLANT OPERATION AND CONSTRUCTION

Since the Laboratory's tenure of ALPR plant operation was limited, the plans for plant improvement were also limited. These plans included experiments, surveys, and modifications. The areas experimental in nature (research and development) are noted in Appendix B. Recommendations for improved plant operation and construction confined to surveys and modifications are noted below. These recommendations are based upon experience, observation, and analysis during the facility construction and the period of operation by the Laboratory.

A. Automatic Power Demand Control System*

1. A 20% control slidewire is presently used in the $P-P_0$ pressure indicator. The control system will function with normal proportional and reset action within the limits of ± 4 psi, which is 20% of the ± 20 psi range of the $P-P_0$ indicator. A single 60-kw load change will not produce a pressure change greater than ± 4 psi, provided pressure has been allowed to return to zero (which takes about 45 sec) before application of another 60-kw load change. If load changes are made at intervals shorter than 45 sec, the pressure will exceed ± 4 psi (plus 4 psi on load decrease and minus 4 psi on load increase), and the control slidewire will not be able to transmit information as to the actual pressure. Loss of pressure information will delay the operation of the control rod for corrective action, and will result in a larger pressure deviation. To improve the operation of the control system by restoring pressure to normal as quickly as possible, it is recommended that a 100% slidewire be installed in the $P-P_0$ pressure indicator, or limit switches be installed, which close at plus 4 and minus 4 psi, and connected in parallel with RH and RL contacts of the electronic relay amplifier to actuate the center rod motor starter directly. The switch which closes at plus 4 psi is to drive the control rod in, and the switch which closes at minus 4 psi is to drive the control rod out. The limit switches are open between minus 4 and plus 4 psi, and the electronic relay unit will control the center rod position to restore pressure to normal.

2. A relay designated ATR in the electronic relay amplifier is energized and used to balance the electronic amplifier in preparation for automatic operation when the power demand control is not in use.

*W. C. Lipinski

It is recommended that a contact, which is closed in manual, be added to the center rod control switch and wired in series with the ATR relay. When the center rod control switch is pulled out for automatic operation the ATR relay will be de-energized.

The present wiring arrangement of the ATR relay is not failsafe. A contact in the center rod control switch, closed during automatic operation, was used to energize an intermediate relay. A normally closed contact of the intermediate relay was used to energize the ATR relay. If the intermediate relay burned out while in automatic power demand operation, the ATR relay would be energized and would zero the electronic relay amplifier preventing any control rod operation.

The recommended addition of a contact in the center rod control switch, closed during manual operation, would provide failsafe operation of the ATR relay.

3. Two limit switches should be added to the present switch assembly on control rod No. 9 and connected in series in the automatic power demand interlock circuit. The switches should be easily adjustable in order to change the range of automatic operation of the center rod over the life of the core. One switch would open when the center rod ran in to a selected limit, and the other switch would open when the center rod ran out to a selected limit. When either of these switches opened, the power demand control system would be placed in manual operation and an alarm would sound.

The limit switch which is temporarily being used as an "in" limit interlock should be rewired in series with the present center rod "out" limit switch, and both switches should be set to open at 30 in.

An "in" limit interlock is necessary to protect the reactor in the event some malfunction would simulate a high-pressure condition and result in center rod insertion. If the center rod were inserted far enough, the reactor would become subcritical. If the condition causing simulation of high pressure should clear, providing other conditions (such as low pressure) did not shut down the reactor in the meantime, the center rod would be withdrawn at constant velocity, a short period would be developed (since the period shutdown circuit would normally be bypassed for power operation), and reactor power would increase until the pressure returned to normal. The limits of how far the control rod would run "in" and "out", the shortest period developed, and the maximum power reached would be a function of the steam load at the time of the malfunction calling for rod "in" and of the duration of the malfunction before the control rod started "out." In view of the unanswered questions

as to magnitude of the peak power and period, it is considered desirable to set an "in" limit on center rod operation in automatic operation.

An "out" limit is required in automatic operation because continued withdrawal of the center rod would result in power production higher than normal and shutdown on reactor high pressure. The "out" limit switch provides additional time prior to a pressure shutdown and enables an operator to reduce reactor power or to open a valve to reduce pressure.

4. If the two limit switches are added to the $P-P_0$ pressure indicator as outlined in Item 1, the "in" and "out" indicating lights located in the face of the electronic relay amplifier will not function properly. The lights as presently wired are both "on" in automatic operation, and, during a period of automatic control rod motion, the light which is "on" indicates the direction of rod travel. With the addition of the ± 4 psi limit switches, it is possible for both lights to be out even though the control rod is in motion. The both lights "off" condition corresponds to manual operation. This ambiguity can be corrected by wiring the lights directly in parallel with the center rod motor starter coils. The indicating lights then would be normally "off" or "on" when the appropriate motor starter coil is energized. The indicating lights would indicate direction of rod travel during manual manipulation of the center rod. At present the lights give reverse indications during manual operation.

B. Instrumentation*

1. Better insulators in Channel II high neutron flux trip test circuit to enable the microammeter to read down to 10^{-11} amp are needed. The test voltage is 125v dc; therefore the switch and wiring resistance should be greater than 10^{-13} ohm.

2. Reactor area monitors should be recalibrated with a more powerful source held at greater distances from the sensing units to provide a more uniform radiation field over the sensing units.

3. An alarm for the fission gases monitor, Channel VII, to indicate audibly a fuel element rupture, is recommended. This can be accomplished by the installation of an additional relay in the amplifier chassis and connected to a spare process panel annunciator point presently containing the required relay.

4. A low condenser vacuum alarm is currently sounded at $1\frac{1}{2}$ psig and the reactor is shut down automatically at $3\frac{1}{2}$ psig. The alarm point should be reset to 10 in. Hg vacuum. It is apparent that this is the point at which corrective action should be taken upon the condition of

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diminishing vacuum. This new setting will be particularly desirable when the reactor plant is operating without immediate supervision.

5. Response time of safety and relief valve alarm devices should be reduced by employing a type of sensing bulb that does not require the protection of a separable socket.

6. To measure reactor water temperature it is recommended that a thermocouple and well be installed in one of the unused control rod locations. The present thermocouple requires water circulation to the reactor purification system, and does not give a rapid, direct measurement of reactor water temperature.

7. Addition of selenium contact protectors is recommended to relays 30R1 through 30R9 and 30T1 through 30T9 in the shutdown annunciator circuit. These protectors will reduce arcing at the associated pushbutton contacts and protect the relay coils against surge voltage breakdown.

8. The present tachometer mounted on the turbine has a range of 3000 to 5000 rpm and is of the vibrating reed type. As an aid to training personnel in the starting of the turbine, an electric tachometer indicator can be connected to the control room electric tachometer system. The indicator should have the same range as, and be in parallel with, the control room indicator in order to permit observation of the turbine starting speed locally.

9. A pressure gauge-type level indicator is needed in the pump house for the 50,000 gal water storage tank. This gauge can use the same line presently connected to the automatic deep well pump controls.

10. A resistance thermometer bulb presently on hand at ALPR should be installed outside the building to read ambient temperature. The present thermocouple installation does not permit reading below zero degrees Fahrenheit. The resistance thermometer is calibrated to minus 60F.

11. The high-pressure (HP) and low-pressure (LP) condensate return tanks condensate levels were not known at times when this information was required during plant operation. The monitoring of the condensate level is necessary to confirm the satisfactory operation of condensate discharge from the tanks. It is recommended that a sight gauge glass be installed on each tank.

12. The reactor water level indicating instruments: While there is no evidence that the existing indicating devices are inoperative or inaccurate, there is also no indication that they are continually accurate. It is recommended that future pressure vessels of the ALPR type be equipped with a side-arm water level indicating sight gauge glass. The lower connection from the gauge glass could penetrate the pressure vessel at approximately one foot below the normal cold reactor water level, not impractically low. The upper connection could be made at a point approximately two feet above the normal hot reactor water level. The recommended gauge glass can be placed in a well, or pit, to the side of the vessel and can be shielded with a removable cover. A light source within the well, or pit, together with a system of mirrors or a simple periscope will permit observation as required.

13. Reactor water and steam conditions: Experience to date indicates that it is desirable at times to monitor water and steam conditions to a greater extent than was anticipated. Therefore, contingent upon the installation of the pressure vessel water level indicating sight gauge glass, Item 12, it would be desirable to install a sample tap at the upper and lower gauge glass connections. These taps may be piped, if necessary, to a water-cooled sample condenser. Alternatively, the sample tap installation may also be accomplished by the insertion of tubes to the desired levels through one of the spare nozzles in the pressure vessel head.

C. Electrical*

1. Removal of the interlock which presently requires the main generator breaker to be closed before the bus tie breaker can be closed is recommended. In the present plant, this interlock serves no useful purpose and reduces flexibility of switching.

2. The power line neutral should be separate from the station grounding system. The present connection permits 120-volt unbalanced load currents to flow partially through building grounding system.

3. A condenser fan motor starter back contact should be connected to the "slow" circuit in the fan speed control so that when this fan is stopped it automatically drives the speed control to the extreme slow position.

4. Additional mimic bus sections should be installed showing (a) Idaho tie breaker, (b) support facilities feeder, and (c) main Idaho breaker. Also, indicating lights to correspond with present mimic bus lights should be considered.

*H. H. Hooker, R. W. Thiel

5. Main generator voltage regulator transient response measurements (by recording line voltages) during feedpump transfer operation are recommended. Although the feedpump drive motors start at reduced voltage, the starting power factor is low. This may result in a more severe demand on the voltage regulator than would be encountered in the case of a 60 kw load change at unity power factor.

6. Emergency power supply output line voltage measurements during simulated power outage are recommended. This can be done by manually opening the ac line breaker to the emergency supply to initiate transfer to the battery drive. The breaker should then be reclosed to initiate transfer back to the ac drive.

7. Station battery capability in terms of hours of driving the emergency power supply should be determined. This should be done with as nearly a normal load on the supply as possible. It should probably include driving all five rods in from the normal operating position.

8. Accurate plant auxiliaries load measurements should be made for different net power outputs, using the bus tie wattmeter.

9. Several false fire alarms have been sounded by the ADT fire protection system. Most of these alarms have been due to abnormally high temperatures existing in the fan loft during shutdown operations. The existing thermocouples are rated at 160F and are located directly over the turbine, on the fan floor. These should be replaced by thermocouples rated at 230F.

D. Mechanical*

1. Moving or removing the steam bypass valve No. 2 is recommended. The second bypass valve was to have a capacity of 10,000 lb/hr. It is presently installed after the bypass steam flow nozzle which has a 200-in. water gauge pressure drop at 3000 lb/hr and 400 psig. This nozzle limits the maximum amount of steam which can be passed through the second bypass valve.

2. The air ejectors aftercondenser vent, as presently installed, discharges to the atmosphere through the plant roof. Condensation and possibly icing occur when the warm moist air comes in contact with the cold roof. It is recommended that the vent discharge be relocated so as to discharge into the exhaust of the main condenser fan. The piping from the radiological filter to the roof should be replaced with piping from the filter to the one-inch opening in the plenum section of the condenser air exhaust duct. This duct opening is now blanked off with a blind flange. The radioactive vent gases then would be diluted by the condenser exhaust stream and would be directed upward (see Section VIII-A-1).

*E. E. Hamer, W. R. Wallin, W. C. Lipinski, R. W. Thiel, J. D. Cerchione,, G. L. Jorgenson, E. D. Graham, W. J. Kann

In order to be assured that no radioactive gases enter the building, it is also recommended that a smoke test be made and photographed. With smoke coming from the aftercondenser vent discharge of the air ejector, the condenser fan should be operated through its entire speed range including "off" and all dampers should be operated in various combinations with each selected fan speed.

If there is any indication of possible contamination of the inside of the building, it is recommended that a small duct (6 to 10 in. in diameter) be installed on the fan exhaust duct, penetrating the roof as a short stack at the present location of the aftercondenser vent of the air ejectors. The vent should then be installed to discharge into the duct. The duct should be provided with a back-draft damper between the fan exhaust duct and vent discharge. In this manner, vent gases will be discharged to the atmosphere through the roof and diluted with fan exhaust air. Possible condensate should be drained from the duct. Another photographed smoke test is suggested after this installation. The back-draft damper should be closed at zero fan speed.

3. The air ejector aftercondenser vent flowmeter was originally installed adjacent to the aftercondenser and removed because of faulty measurement of the noncondensable gases in the presence of water. It is recommended that a pitot tube replace the original float-type flowmeter and that it be located in the air ejector aftercondenser vent discharge line between the air-cooled aftercondenser on the fan floor and to the atmosphere. The pitot tube should be connected to a differential pressure water manometer which can be located on the operating floor for convenience. A temperature-sensing element (TI-16E) is now present at the proposed pitot tube location. From these measurements, calculations can be made of total dry air flow and water loss.

4. A temporary plywood structure was used during the Plant Performance Test to separate the condenser air intake and exhaust air flows in order to reduce the existing air recirculation. This structure was attached to the ducts on the outside of the reactor building. Immediately prior to the termination of the Laboratory's operation of the facility, a permanent sheetmetal structure was fabricated and installed. It was not possible to obtain extensive data on the efficiency of the installation. It is recommended that this be done and the degree of recirculation be measured to determine the advisability of redesign or modification of the installation.

5. Another area of condenser air system recirculation (see Item 4) was the air leakage past the mixing chamber motor-operated dampers. Gasket material was placed on the leading edges for positive shut-off. Again, since it was not possible to obtain conclusive information as to the effectiveness of this installation, it is recommended that data be obtained to establish the degree of recirculation present.

6. A valve and a vacuum gauge should be installed in the vacuum line of the turbine gland seal-air ejector. This would reduce the quantity of air handled by the air ejector system, reducing the water loss through the discharge stack of the air through the air ejector aftercondenser. The valve should be installed between the turbine and the present air ejector vacuum gauge. The vacuum gauge should be installed between the valve and turbine to indicate the vacuum at the turbine gland seals. The valve should be throttled, but not below a vacuum reading of less than 1 in. Hg under operating conditions.

7. The condensate circulating pump is essential to the operation of the feedwater pump and the feedwater pump is necessary to maintain the reactor water level. It is recommended that a second (standby) condensate circulating pump be permanently installed in the system.

8. A filter with a bypass is recommended for installation in the control rod drive cooling water supply line as a precautionary measure. Its capacity should be the same as the capacity of the flowmeter, (180 gph) and the filter should remove all particles over 10 microns in size.

9. The boral shield plates, mounted on the control rod drive plugs and presently located in the spare rod drive pressure vessel nozzles, should be removed. These plates were added for shielding purposes and are now deteriorating. Operating experience has shown that the radiation levels are low enough without these plates.

10. The Fulton Sylphon temperature-controlled valve, in the turbine lubricating oil cooling system, is calibrated to close at 115 to 120F in order to force oil through the oil coolers. When on "automatic," and during start-up of the turbine, this valve bypassed the high-pressure oil directly from the pump discharge to the oil header, causing the turbine auxiliary oil pump to stop on a signal from the high-pressure "cut-out." Cycling resulted. The $\frac{1}{4}$ -in. stroke of the temperature-controlled valve was decreased to approximately $\frac{1}{16}$ in. so that the increased pressure drop through the valve resulted in 6 psig in the header, and the pump ceased to cycle. It is recommended that an orifice be installed at this point in the system to give the desired pressure drop instead of using the temperature-control valve for this purpose.

11. The bridge crane hook speed is too slow and time-consuming for much of the service required. It would be advantageous to have available, in addition, a higher speed for each of the two hooks.

12. Positioning of the heavy shield blocks was cumbersome because of the manual operation of the chain-fall-type crane bridge and hoist carriage. It would be advantageous to motorize the bridge and carriage.

13. The reactor biological shield gravel vent system incorporates a barometric damper located in the air space between the gravel and the operating floor. At present it serves no specific purpose and the elimination of this damper may make available more cooling air for the reactor top shield area and instrument wells.

14. The reactor plant freight door seals allow more than acceptable air inleakage. Door batteners have provided a partial solution. It is recommended that the existing rubber door gasket be replaced with soft rubber having a $1\frac{1}{2}$ -in. projection.

15. Structural Grade A-283d steel plate is used in the prototype power plant building shell and is adequate for the Idaho installation. However, a high stress steel, such as U. S. Steel "T-1," may be required in the arctic installation for suitability in minus 60F ambient air temperature. The highest stressed shell area is the ring around the reactor biological shielding gravel. It may be informative to make stress checks in this area. The use of strain gauges is suggested.

16. Undesirable quantities of oil in the plant water systems have been noted. The oil comes from the oil-lubricated bearings of the deep well pump. It is recommended that these bearings be replaced with the manufacturer's water-lubricated bearings.

17. The furnace room is equipped with one small ventilating louver opening into the machine shop. When the furnace room doors are closed, this louver is inadequate and room temperatures become excessive. There has been one false fire alarm from this room to date. It is recommended that a second adjustable louver be installed in the outside building wall as an additional ventilating device.

18. The reactor plant water drainage system receives water at the reactor operating floor near the auxiliary turbine oil cooler and drains to a site drainage pond. The system does not function properly in freezing weather. This is due to the formation of ice within the pond and the adjacent piping. It is recommended that a grader be employed to enlarge the pond and dike it so as to allow proper drainage of the piping.

It is also recommended that this area be fenced and access to it limited. Presently, a temporary hose drain line has been installed to allow the water to drain to an area northeast of the plant.

19. The underground static pressure line between the outdoor water storage tank and the deepwell pump controller in the pump house becomes frozen during below-zero weather. It is recommended that additional fill be added over the line to a height of approximately three feet near the building wall where the line comes close to the ground surface.

20. The pump drip drains in the pump house presently extend horizontally through the pump house wall and spill to the open ground. During cold weather, the drains freeze, causing water to accumulate on the pump house floor. The pump drains should be run vertically from the pumps down through the building floor to about five feet below grade and then to a dry well.

21. The experimental main steam condenser performance did not meet the specifications as set forth for a ultimate sea level installation. Further investigation is necessary to determine its adequacy.

22. After the initial alignment of the turbine-generator unit the reactor biological gravel shielding was installed and, the unit required re-alignment. It is recommended that for future installations the gravel and other shielding, and heavy building components, be placed before alignment of the turbine-generator unit. Also, assurance that the building is firmly supported by the piers, or piles, is necessary.

23. During application of the building shell insulation many difficulties were encountered. It is recommended that consideration be given to the application of the insulation on the exterior building wall and the possible use of block type with a fabric-mastic covering. This type of installation may also reduce the building stress in the area of the reactor biological gravel shield.

E. Shielding*

1. Reactor top shield radiation measurements made during the last week in January 1959 indicated that the reactor may be operated at 3 Mw with the top two segments of the top shield (Fig. 23) removed, without resulting in unduly high radiation levels anywhere in the plant. Levels directly above the remaining top shield segments will be two to five times standard tolerance (40 hr/wk), but this area is normally not

*A. D. Rossin

entered by personnel. Levels on the fan floor are less than two to three times tolerance. Radiation levels with various top shield arrangements are noted in Table 11.

Table 11

RADIATION LEVELS WITH VARIOUS TOP SHIELD ARRANGEMENTS

No.	Configuration (Bottom to Top)	Dosage above Top Shield			On Operating Floor Gamma (mr/hr)	Segment No.
		Gamma (JUNO) (mr/hr)	Neutrons/cm ² -sec			
			Fast	Thermal		
1	No top shields	200	5500	1700	10	---
2	7 in. Masonite + 1 in. steel	100	150	170	5	1
3	As above directly above 3 in. hole in 7 in. Masonite block	40	130	60	---	1
4	7 in. Masonite + 1 in. steel					1
	4 in. Steel					2
	2 in. Masonite	16	40	15	3	2
5	7 in. Masonite + 1 in. steel					1
	4 in. Steel					2
	2 in. Masonite					2
	4 in. Steel					3
	2 in. Masonite	3	15	8	3	3
6	Full Shield (Fig. 23)	1-2	10	2	3	1,2,3,4*

*Segment No. 4 is comprised of 5 in. steel.

2. The simulated heat load heat exchanger shell vent line valve (Fig. 64) was initially opened on January 29, 1959. The localized "hot spot" reading, possibly arising from accumulated N¹⁶, dropped from 290 to approximately 130 mr/hr and has never returned to 290 mr/hr. If the valve is permanently left open $\frac{1}{4}$ turn, the "hot spot" activity level does not exceed 50 mr/hr. The dose one foot radially from the shell of the heat exchanger is approximately 15 mr/hr on the average. This theoretically allows 20 hours per week working time at this location. Table 12 illustrates the initial valve manipulations made to reduce the "hot spot" activity level.

Table 12

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

FIG. 64
SIMULATED HEAT LOAD HEAT EXCHANGER

3. The radiation levels at the purification system panel and resin column areas were noted to be increasing during the extended reactor operational period. The source of the radiation was found to be the resin in the resin columns located within the purification system vault. Additional gravel shielding material was placed around and in the region of the vault. It is recommended that the panel and maintenance areas be checked periodically to evaluate the effectiveness of the additional shielding.

4. The pressure vessel support cylinder and external thermal shield incorporate a lead gamma filter in the area between the reactor and three of the four nuclear instrument wells and the reactor. When comparing the responses between the filtered and unfiltered instruments in the wells during 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 recommended that such a filter shield be omitted in a future design.

F. Neutron Source*

It is apparent that the ALPR antimony-beryllium source, as designed, delivered its anticipated yield. It does not appear economical to

*A. D. Rossin

have a source yield at a level where it can be seen by ion chambers before reactor startup; hence it should be possible to reduce the cost of the source. Two recommendations are noted:

1. The beryllium block is the biggest item of fabrication cost. It can be designed as a rectangular block, lying entirely between the rod and the core, thereby eliminating at least half the quantity of beryllium required, and also the machining of the cylindrical hole.

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

G. Reactor Building Ventilation System*

The reactor building ventilation system, as now installed, consists of two supply ducts to a mixing chamber with a motor-controlled adjustable damper in each. One supply duct takes hot air from the condenser air exhaust duct, the other takes cold air from the condenser air inlet duct. The ventilating air supply fan takes air from the mixing chamber and delivers it to the operating floor of the reactor building where it is distributed through two registers. The modulation of the dampers in the supply ducts are controlled from a thermostat on the operating floor.

Field test information indicates that there are two areas for improved performance. They are as follows:

1. The delivered air volume decreases as the condenser air circulating fan speed increases with the hot air supply blanked off. The supply of ventilating air is approximately 10% of that required by the specifications when the condenser fan is operating at maximum speed. Fan performance data from the manufacturer indicates that for the measured pressure differential, the fan should be delivering more air.

Recognizing that the ventilation requirements were based upon assumptions with respect to the heating in the gravel below the operating floor and its influence on the floor temperature, it is recommended that the adequacy of the present air flow be determined under warm weather conditions with the plant at full load. If the ventilation air supply does prove to be deficient under actual conditions, a larger fan should be installed as required on the basis of temperature data taken from the operating plant. This should not interfere with plant operation for additional ventilation can be provided by opening the freight and personnel doors and by using portable fans until the new fan installation is made.

*E. E. Hamer

2. Because of the differential pressure between the condenser air exhaust duct and the condenser air inlet duct, the mixing chamber receives too great a supply of hot air.

It is recommended that the linkage between the controller and damper on the hot air supply duct be adjusted or revised so that the movement of this damper is limited to a very small opening and tight shut-off. An alternate solution would be to permanently close some of the blades on this damper and have the controller actuate only one or two blades.

H. Reactor Water Purification System*

1. The use of a porous type-1 strong base anion exchange resin to achieve maximum quality of the reactor water as opposed to the specified NR-11, which contains a weak base anion exchange resin, is recommended (see Section VII-D-9). This mixture of strong base porous anion resin and the 8% cross-linked strong base cation is sold under Illinois Water Treatment Company designation NR-14. The company recommends this mixture for operation at 170F and the resin will be able to withstand higher temperatures up to the specified 200F without damage, particularly since the resin is used only once.

It should be noted that regardless of the type of exchange resin used, none of the resins will operate satisfactorily in the presence of more than 1 ppm of oil, according to the Illinois Water Treatment Company. The replacement of the deep well oil-lubricated bearings with water lubricated bearings is recommended (see Appendix A-D-16).

2. The original disposal of resin from the system columns was accomplished by the use of hoses. This method was unsatisfactory and temporary piping and tubing were installed. This proved to be satisfactory, and it is recommended that permanent piping be installed along with certain modifications as follows.

The resin removal lines from each bed should be out at a point 32 inches above the operating floor level, and valves RP-14 and RP-17 relocated and installed at the upper ends of these lines (see Fig. 33). Each valve should then be piped upward to a common line which extends horizontally in the direction of the Waste Storage Sump (see Fig. 24) and at an elevation of $38\frac{1}{2}$ in. above the operating floor. Valve RP-18 should be removed from its present location and installed in the horizontal line. In this manner, connection H-1 is relocated and will appear at the downstream end of the line and the edge of the

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purification system panel. It is also recommended that H-1 be changed from a hose connection to a tubing connection. Connection H-7, similar to H-1, should be installed at the same end of the purification system panel, but at an elevation of 2 in. above the floor. A line should be run from this connection behind the panel for a distance of approximately 8 ft 6 in., where it can be joined to the existing piping leading back to the contaminated water (retention) tank. The tubing leads from the spent resin disposal tanks can then be connected directly to the relocated connections H-1 and H-7. The sight flow gauge should still remain below floor level in the storage sump during the resin disposal operation. The shielding of the effluent line should be dictated by radiation surveys.

The dimensions quoted above are recommended because they refer to present areas which are relatively open and accessible for the installation.

3. Venting of the lines connecting the two resin columns to valves RP-8 and RP-9 is recommended. Vents in the form of a plugged tee at the highest elevation should be useful in removing air from the columns after resin addition.

4. The contaminated water (retention) tank pump has lost priming when used to remove water from either the cation or mixed bed demineralizer during resin addition. It is believed that this difficulty is due to air inleakage at the pressure relief valve.

It is recommended that the Pemberthy brass ejector, presently at the ALPR site, be installed so that the suction side of the ejector is connected to hose connection H-2 (see Fig. 33). Hose connection H-9 should be attached to the upstream pressure side of the ejector and the discharge of the ejector should be attached to hose connection H-7. During normal operation valves RP-26, RP-29, and RP-19 should be open as well as valves RP-10, or RP-11, and RP-12, depending upon which resin bed is to be replaced. Then the contaminated water (retention) tank pump should be started, pumping from the tank through valve RP-29 to the high pressure side of the ejector. The suction side of the ejector will pull a vacuum through valve RP-19 and the discharge from the ejector will return through hose connection H-7 to the drain header of the tank. This should eliminate the resin column water removal problem assuming that there is not too great an air inleakage around the valve stems and the pressure-relief valve.

5. The present purification system installation is designed for continuous service at normal plant operation. During frequent startups and shutdowns of the reactor, the reactor water temperature is mostly between 170F and boiling. The purification system is inoperative in this temperature range.

It is recommended that equipment be installed which would allow continuous operation of the purification system and thus maintain high-quality reactor water at all times. Figure 65 and Figure 66 represent the recommendations for continuous cooling of the reactor purification water. It is a modification to the present purification system and is installed across valve RP-32.

To operate the cooler, close valve RP-32, open valves RP-33 and 34. Regulate the cooling water to the tube side of the heat exchanger until the desired purification water flow is passed through the Spence temperature regulating valve, as observed on the purification system flowmeter. The addition of a local temperature gauge in the outlet line of the cooler (purification water side) would be desirable, although not absolutely necessary.

FIG. 65
PURIFICATION SYSTEM SHUTDOWN COOLER FLOW DIAGRAM

1. The decomposition of the water was found to be $16.1 \text{ cm}^3 \text{ O}_2/\text{liter}$ of condensed steam. This rate was essentially independent of the reactor power level (see Table 13). Consequently, the rate of gas production is approximately a linear function of reactor power over this range. The concentration of dissolved oxygen in the reactor water was slightly less than 7.0. The reactor feedwater contained 0.2 ppm of dissolved oxygen. The total gas flow rate from the air ejectors was found to be $3.4 \pm 0.14 \text{ cfm}$ at one atmosphere and 32°F.

This section includes the data, results, and recommendations obtained from a chemical investigation during the first week of the plant performance test. The tests and therefore this discussion are preliminary in nature.

I. Water Chemistry*

FIG. 66
PURIFICATION SYSTEM SHUTDOWN COOLER

Table 13

REACTOR WATER DECOMPOSITION DATA

Date	Time	Reactor Steam Flow (lb/hr)	Condensed Steam (cc gas/l)	ppm O ₂ in Condensate	Gas analysis (%)			Condensed Steam (cc O ₂ /l)
					O ₂	H ₂	Inert	
11-20-58	15:30	4,000	51.0	3.6	31.3	66.8	1.9	16.2
11-21-58	13:30	7,850	51.1	3.4	29.1	69.4	1.5	15.2
11-21-58	20:50	7,600	48.5	4.3	29.8	68.1	2.1	15.4
11-22-58	10:30	7,800	49.3	4.1	31.1	68.2	0.7	16.1
11-22-58	15:15	7,550	50.7	4.1	30.7	68.5	0.8	16.3
11-22-58	18:30	5,580	50.0	4.2	31.1	68.1	0.8	16.3
11-23-58	11:15	8,000	49.2	4.4	30.9	68.6	0.5	16.2

To observe the effect of hydrogen addition on the retardation of decomposition, the following experiment was run. Bottled hydrogen gas was metered through a high-pressure rotameter into the reactor feedwater line at the point where the effluent from the ion-exchange cleanup system and the feedwater meet. The oxygen content of the steam was suppressed to essentially zero with the addition of 32.8 cm³ H₂/liter of feedwater. The data from the hydrogen addition are shown in Figure 67 and Table 14. As the hydrogen was added, the steam system activities increased as expected due to an increase in the N¹⁶ activity in the steam phase. This can be seen in Table 15. The results obtained during the hydrogen addition parallel those obtained from BORAX⁽⁶⁾ and EBWR⁽⁷⁾ for similar experiments.

FIG. 67
REACTOR WATER DECOMPOSITION VERSUS HYDROGEN ADDITION
NOVEMBER 22, 1958

Table 14

HYDROGEN ADDITION DATA

Date	Time	Reactor Steam Flow (lb/hr)	H ₂ Rate (cc/liter)	Condensed Steam (cc gas/l)	ppm O ₂ in Condensate	Gas analysis (%)			Condensed Steam (cc O ₂ /l)
						O ₂	H ₂	Inert	
11-25-58	15:00	8,100	32.8	30.1	0.05	0.0	100	---	0.04
	16:30	8,000	26.1	24.3	1.4	8.5	90.8	0.6	2.69
	18:30	7,900	4.18	43.4	3.6	29.0	71.0	---	13.3
	21:45	7,700	11.7	34.0	3.4	29.9	72.1	---	10.5

Table 15

PLANT ACTIVITIES DURING HYDROGEN ADDITION

(mr/hr)

H ₂ Rate (%)	Feedwater Pumps		Simulated Heat Load Exchanger		Main Steam Line		Main Steam Line (Behind panel)	Auxiliary Steam Line	Turbine	
	No.1	No.2	Pt.1	Pt.2	Pt.1	Pt.2			Pt.1	Pt.2
0	1.5	1	120	1.2	7	3	26	6.5	3.5	5
32.8	2.2	1.5	300	1.5	15	10	35	12	8.5	9
32.8	4	3.5	700	+	+	17.5	50	+	+	18
26.1	4	4	100	2	8	4	22	6	4	5
51.0	1	2	265	2.5	18	13	35	15	10	10
4.18	4	1	100	1	7	4	25	8	4	5
11.7	1	1	140	1.5	10	6	24	9	5	7
11.7	2	2	350	2	16	9	35	16	7	8
43.0	2	2	225	2	16	9	35	15	7	9

2. The decontamination factor for Na²⁴ (the ratio of the concentration of the given isotope in one phase as compared to the concentration in another phase) between the reactor water and the steam was found to be approximately 1000. This is comparable to the values in BORAX and EBWR. The decay curves for reactor water and steam can be found in Figs. 68 and 69, respectively. Notice that at this time the major long-lived activity in the reactor water is Na²⁴, corresponding to 6.27×10^5 dpm/ml.

The decontamination factor across the ion-exchange cleanup loop is essentially infinite for all nongaseous activities except for what is believed to be 2.6-hr Si³¹. A decay curve for the effluent of the mixed bed is shown in Figure 70.

FIG. 68
REACTOR WATER DECAY CURVE
NOVEMBER 21, 1958

FIG. 69
REACTOR STEAM DECAY CURVE
NOVEMBER 21, 1958

FIG. 70
DECAY CURVE OF MIXED BED DEMINERALIZER
EFFLUENT WATER
NOVEMBER 21, 1958

3. Fission product activities in the steam, the effluent gases from the condenser, and the reactor water were investigated by gamma spectroscopy. The following gaseous activities were detected in a sample of condensed steam: Xe^{133} , Xe^{135} , Xe^{138} , Xe^{140} , and Kr^{88} .

The emission rates of fission product gases Xe^{138} , Xe^{133} , and Kr^{88} in the exhaust gases leaving the reactor system are shown in Table 16.

Table 16

EMISSION RATES OF FISSION PRODUCT GASES

Gas	Emission Rate		Theoretical Ratio of Gas to Xe^{133} if All Gas Is Recoiled into Steam	Observed Ratio of Gas to Xe^{133}
	Curies/Day	Half-Life		
Xe^{138}	57×10^{-2}	17 min	392	41
Xe^{133}	1.5×10^{-2}	5.3 days	1	1
Kr^{88}	7×10^{-2}	168 min	25	5

One iodine analysis of the reactor water was made. The result obtained was 1.26×10^3 dpm/ml of I^{131} . Assuming that all of the iodine I^{131} produced is released upon fission of U^{235} and, correcting for the ion exchange cleanup, one finds that approximately 1.4 mg of U^{235} is required to produce this quantity of I^{131} . This is in agreement with a value of approximately 1 mg of U^{235} to produce the observed Xe^{138} . However, approximately 11 mg of U^{235} are required to produce the observed Xe^{133} .

In order to formulate any conclusions, a continuing program of analysis is necessary.

4. It is recommended that the following determinations be made on a periodic basis, during all periods when the reactor is running.

- a. pH of reactor water and reactor feed water once each shift.
- b. Oxygen content of reactor water and reactor feed water once per day.
- c. Total solids in reactor water and reactor feed water once per month.

- d. Water decomposition rate
 - (1) Two times per week until level is established, then
 - (2) One time per week until operating conditions are changed.
- e. Total beta and, alpha count and decay curve of reactor water, ion exchanger effluent, and steam two times per week. Decay curves should be plotted until the activity has reached background levels or for 3 to 4 days. After behavior is established, drop to one time per week.
- f. Xe^{138} , Xe^{135} , Xe^{133} , and Kr^{88} in air ejector gas, two times per week until level is established, then once per week unless an incident should occur.
- g. Crud in reactor water and reactor feed water. Crud filter samples should be taken two times per week until level is established, then once per week.
 - (1) Total crud, in ppm
 - (2) Chemical and radiochemical analysis of crud.
- h. Activities in reactor water, ion exchange effluent, and condensed steam; one time per week for each until level is established, then reduced frequency of determinations for those that do not change, (unless an incident should occur):
 Na^{24} , Al, Co^{58} , Co^{60} , Mn^{56} , Mn^{54} , I^{131} , I^{135} , Mo^{99} , Ba^{140} ,
 Si , F^{18} , N^{13} .
- i. Meter survey of components of water and steam systems once per week to follow buildup of activity due to deposits.

APPENDIX B

RECOMMENDATIONS FOR RESEARCH AND
DEVELOPMENT PROGRAMS

Since the Laboratory's tenure of ALPR plant operation was limited, no research and development programs were initiated. Many areas of experimentation are open, of which a few are noted here. It is impossible to present the details for each recommendation; therefore, only an outline is presented for most of the recommendations.

A. Automatic Power Demand Control System*

1. Investigate the stability of the control system with only the limit switches set at ± 4 psi in operation and the relay control amplifier made inoperative by energizing the ATR relay. This should be done at several different power levels. Of importance are: divergent or convergent oscillations, maximum or minimum limits reached by control rod, maximum and minimum neutron flux values, maximum and minimum pressure values, and elapsed time to reactor shutdown or transfer to manual operation by rod position or pressure limits.

2. Investigate the stability of the control system with zero steam flow from the reactor vessel.

3. Measure time lapse to shutdown, and reactor neutron flux and reactor pressure versus time when the control rod is run out manually at maximum velocity starting from various power levels and 300 psig. These experiments would show what would happen during automatic operation if an upper position limit was not placed on the center rod and a malfunction should occur which would call for control rod withdrawal.

4. Investigate improving speed of response of the automatic power demand control system to load changes by using faster rod withdrawal rates.

B. Reactor Startup*

Continue instrumentation startup experiments (see Section VI-C, Experiments Nos. 183 through 186). The purpose of these experiments is to determine the extent of instrumentation necessary to place a boiling reactor into operation.

C. Reactor Power Extrapolation*

1. Measure the transfer functions of reactor flux to rod position and reactor pressure to rod position at various pressures and power levels. This information would be used in predicting the stability of operation at higher powers.

*W. C. Lipinski

2. Operate reactor at higher power levels with the 40-assembly and a 59-assembly core. Changes to plant equipment (feedwater system, steam dump, etc.) are necessary. If a larger feedwater return system were used, and if both feedwater pumps were used simultaneously, it should be possible to maintain the water level in the vessel at powers significantly above 3 Mw for at least five minutes. The present condenser system cannot handle the greater steam mass flow rates, and the extra power would have to be dissipated in some other fashion. With the use of a high stack it might be feasible to blow the extra steam to the atmosphere. With the appropriate plumbing, the 1000-gal water storage tank could be used to make up this water loss for a short period of time.

D. Reactor Control*

1. Conduct experiments to determine the feasibility of reactor control by feedwater injection into the steam dome (see Experiment No. 144, Section VI-C). Some redesign may be necessary.

2. Investigate the use of an additional control system which can be used to pressurize the reactor and regulate the rate of rise of temperature automatically. Such a control system would relieve an operator and would bring the reactor up to pressure more quickly by maintaining maximum allowable rate of rise of temperature.

3. If the pressure control system (see Item 2) adequately controls reactor power at zero steam flow from the reactor vessel, it is then possible to keep the reactor operating on loss of generator power. At the present time the reactor is shut down on loss of generator power. With reactor pressure controlled by the control rod position, it is possible to operate the reactor at a low neutron level to maintain the 300-psig steam pressure with no steam leaving the reactor pressure vessel; thus no electrical power is required to condense steam. A tenfold increase in rod insertion velocity (motor driven) would be required to limit the rise in reactor pressure to less than the shutdown value of 340 psig occurring with the loss of full power at a steam rate of 8000 lb/hr. At the present rod velocity (motor driven), reactor pressure would rise to a theoretical value of 780 psig on loss of load at 8000 lb/hr.

The present motor-driven rod speed would handle a shutdown from 2600 lb/hr and limit the final reactor pressure to 340 psig. The speed of rod insertion required to limit the pressure rise is proportional to the square of the reactor power level at the time of load loss. Doubling the present speed of rod insertion would handle a loss of load at 3600 lb/hr.

The purpose of trying to keep the reactor from being shut down on loss of load is to reduce the time required to get back to power. The five-rod bank is at approximately 20 in. at 8000 lb/hr. At present 27 min

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is required simply to raise the five rods from zero to 20 in., not counting the delay in approaching critical, establishing a period, and increasing power.

Another approach that can be used to handle load losses at 8000 lb/hr is to drop the center rod partially. The center rod need not be fully inserted. At the present five-rod bank position a center rod insertion of 7 in. is sufficient to change power from 8000 lb/hr to zero lb/hr. On loss of load, the clutch power to the center rod could be turned off and when the rod had fallen to 15.0 in., a limit switch could be used to turn on clutch power and to stop the center rod at slightly less than 15.0 in. The automatic power demand control system would then function to maintain reactor pressure at 300 psig, and the reactor would be ready to supply steam when needed. The emergency power system would have to supply electrical power to nuclear instruments, shutdown annunciator, clutch power supply, control rod motors, and automatic power demand control circuits. Suitable experiments would have to be conducted to determine the feasibility of such a proposal.

E. Reactor Instrumentation*

The power channels (I-IV) should be calibrated over the power range of the operating reactor because of the variation of the mean density of the core fluid as power level is changed.

F. Reactor Physics Statics of the Present 40-Assembly Core**

1. Reactivity Effect of Steam Void Several different experimental methods have been applied to determine the reactivity controlled by steam void at 3 Mw of heat production. In one approach, increments of boric acid solution were pumped by hand from the boric acid tank into the pressure vessel through the lower feedwater spray ring. At the beginning and end of each addition, the equilibrium mass flow rate of steam from the vessel was adjusted to approximately 8000 lb/hr by withdrawal of control rods to new positions. By chemical analyses of reactor water samples for boric acid concentration before and after the addition, and by a measurement of the change in height of the boric acid solution in the tank, the change in the boric acid concentration in the vessel was estimated. It was discovered later that the sampling technique was not adequate for the precision required. The scatter in the data from three separate additions of boric acid solution is sufficiently large as to make it advisable to repeat the experiment, this time with a more accurate sampling method and with a metered addition of the solution.

As a suggestion, a "standard" solution of boric acid could be made up - e.g., approximately 100 gm H_3BO_3 /gal, and chemically analyzed. Then 4 gal (approximately 400 gm H_3BO_3) of this solution could be measured

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**D. H. Shaftman

into the empty, clean boric acid tank. Prior to the initiation of this experiment, the lines from this tank to the lower feedwater spray ring should be flushed with clean water to ensure the accuracy of the addition. After a check critical at a mass steam flow rate of 8000 lb/hr, the hand pump could be used to pump the 4 gal of solution into the reactor through the lower feedwater spray ring and the lines again would be flushed by pumping an additional amount of clean water. As before, it is recommended that the experiment consist of three incremental additions of boric acid, and that both the power level at the initial clean rod position and the rod positions (as a bank) for 8000-lb/hr operation be determined. At the same time, and without extra additions of boric acid, similar data should be obtained at a lower power level - e.g., 6000 lb/hr. Finally, enough boric acid should be added to permit the full withdrawal of rods while producing 8000 lb/hr. By adding more boric acid to this system, again a correlation between the change in boric acid poisoning and power production could be obtained, proceeding as above. In this way the reactivity effect of power production may be deduced for the reactor without rods, as well.

2. Calibration of Control Rods at the Operating Temperature

The control rods have been calibrated in the cold reactor, but the calibration at temperature was deferred in order to make early preparations for the 500-hr plant performance test. After the reactor was readied for the first rise to operating temperature using nuclear heat, and with the source immobile in its core location, a coincidental calibration was attempted at temperatures above atmospheric. Positive period calibrations at fluxes high enough to mask the effect of the fixed source yielded erratic results, possibly because of a local formation of voids, and these calibrations were discontinued. It would be desirable to design and use a source-puller mechanism to remove the potent source while performing rod calibrations at various fixed temperatures including the operating temperatures. As was done in the cold reactor, boric acid solution could be added to the reactor water to poison the system incrementally and to permit withdrawal of rods for differential worth measurements at various vertical locations and at fixed temperatures. The worth curve for the four-rod bank, the five-rod bank, and the regulating rod could be determined.

A technique was developed for the Experimental Boiling Water Reactor (EBWR) tests to calibrate rods while temperature and xenon concentration were slowly changing.⁽⁸⁾ This method should be evaluated for its applicability to ALPR experimentation.

3. Cadmium Ratios of U²³⁵ and Gold Foils. Both in the cold and hot reactors, cadmium ratios should be measured at various core locations to obtain more information about the neutron flux spectrum. It is recommended that U²³⁵ foils and gold foils be used, and possibly other elements (for example, dysprosium).

G. Reactor Statics of the 59-Assembly Core*

1. General Comments: The present selection of a 59-assembly reference core is based upon using assemblies each with the same fuel loading as in the 40-assembly core. In order to get by with only five rods, and to get the rods farther out during operation, burnable poison strips of greater B^{10} content were used in the central 16 fuel assemblies. With the exception of three extra assemblies, the remainder of the 59-assembly core is simply the 40-assembly core pushed out by the insertion of the central 16 assemblies.

A hot flux plot (using bare copper wires) has been made in the reference 59-assembly core, showing that the average flux in most of the outer assemblies is not unreasonably low in comparison with the average flux in the centrally located fuel assemblies.

An experimental flux plot in a hot and/or cold 59-assembly core containing only the more lightly-loaded B^{10} strips and nine control rods would provide comparison data in determining the relative merits of the initial 59-assembly reference core (9 rods; 0.5-gm and 0.4-gm B^{10} -strips) and the present reference 59-assembly system.

Because of the high activity level of the irradiated fuel assemblies, a system for inserting and holding wires or foils without direct, close access would have to be devised. The advisability of getting these flux plots must be considered in terms of a decision as to how useful such data would be versus the time and effort to be expended in obtaining the data.

In the Laboratory's effort, the 40-assembly core was the important reference system. Enlargement to a 59-assembly core was offered only secondary consideration. However, if emphasis is to be shifted to a larger core, additional theoretical analysis would be required to tailor a fuel and B^{10} loading to the changed requirements. Very likely, a smaller fuel loading than that of the present reference 59-assembly loading would be selected, as well as a different B^{10} loading. Consideration should be given again to the possibility of incorporating a burnable poison in the meat of the fuel assemblies, and, in fact, of varying the amount of burnable poison in various radial locations.

2. Other Specific Experiments for the 59-Assembly Reference Core

a. Determine the reactivity effect of steam void in the operating system, no boric acid in the water.

b. Determine the reactivity effect of steam void at various powers, with boric acid in the water to permit essentially complete withdrawal of rods from the core.

*D. H. Shaftman

c. Calibrate the rod bank in terms of equilibrium bank position versus power. Also, with the off-center control rods held at a pre-determined banked position, calibrate the center (regulating) rod versus steam flow from the vessel.

d. Observe the readings of the various neutron detectors versus the reactor power.

H. Reactor Dynamics of 40-Assembly and 59-Assembly Cores*

1. Temperature Profile of the Reactor Water during Transients:

One of the major uncertainties of the analysis of slow reactor transients, e.g., of the pressure and temperature variations during adjustment of the reactor power to the steam load, is the problem of how to compute the temperature profile of the water entering the core. An important step in providing this information would be to instrument the reactor water with a system of thermocouples, e.g., sensitive differential thermocouples to measure temperature differences at various positions in the downcomer, and above and below the core. An accuracy of approximately 0.2F is required.

Once such instrumentation becomes available, measurements should be made during the reactor response to changes in load, both increases and decreases in load.

2. Use of Boric Acid for Shim Control In the EBWR and in BORAX reactors the physics feasibility of shim control with boric acid in the reactor water was tested successfully.⁽⁹⁾ If such a control scheme could be adopted, perhaps only the central (regulating) rod would be needed during power operation. A simple method of diluting or concentrating the boric acid solution would be essential. One (or more) of the present system of off-center control rods could be held cocked as a safety rod. The stability of power operation under these circumstances must be checked.

3. Effect of Height of Top Reflector on Stability of Power Operation SPERT experiments have shown that the height of the water level above the core might have a significant effect on the stability of operation at a fixed power level. The influence of water level is apparent in theoretical predictions of fluid flow parameters in the vessels, such as the velocity of the fluid entering the core, because of the difference in driving head for different water levels. Experiments to determine the stability effects of high and low water levels in the ALPR not only would add to the as yet small store of data of this type, but also would produce information about the safety of permitting a greater variation in water level during routine operation, and about possible improvements in operating stability to be gained by changing the present zero setting of water level.

*D. H. Shaftman

At present, there is an automatic rod scram if the water level is raised or lowered beyond designated points. It would be advantageous to change these scram levels temporarily to meet the specific requirements of a subexperiment while retaining the safety features of such an automatic reactor shutdown. At the same time, the volume of water in the hot well could be adjusted so as to retain the annunciation signal indicating low or high hot well level. Each change in operating water level should be small, and the reactor should be operated at each level long enough to provide both a good measure of stability and an indication of the safety of continuing to the next level. The shrouds extending above the active core do not completely bound the periphery of the maximum core loading; therefore, if the water level should fall or be dropped below the top of these shrouds, there would still be access to the downcomer region and some degree of natural convection would be provided. Thus there is little likelihood of a shift from smooth operation to a divergent pattern of flux oscillations following a single incremental change in water level. Under these circumstances, it is not expected that chugging would develop so rapidly as to preclude the safe termination of an experiment by rod scram. The experiments should be discontinued if signs of incipient chugging should be observed.

4. Extrapolation to Stable Higher Power Levels. Measurement of the reactor transfer function at power has been suggested elsewhere in this report (see Appendix B-C-1). In BORAX IV analysis,⁽¹⁰⁾ the autocorrelation function technique was used with reasonable success in obtaining the power-frequency spectrum. Thus it may be possible to evaluate the stability of operation at higher power levels with some confidence by a careful analysis of the flux patterns at the present full power level.

I. Long-term Effects of Energy Production*

As the core cycle progresses, the rod-bank position at 3 Mw will be adjusted to control the net excess reactivity remaining. A careful record should be kept of the positions of the rods versus the total heat energy (Mwd) produced by the reactor core.

Since the differential worth curve of the rods will change as the core fuel and burnable poison are depleted, it is necessary to re-calibrate rods from time to time so as to obtain a curve of the variation of the net excess reactivity available as a function of the total Mwd production.

J. Magnetic-jack Control Rod Drive**

The Laboratory has available for experimentation a magnetic-jack type control rod drive (Fig. 71) with associated instrumentation and controls.⁽¹¹⁾ The unit was designed and fabricated for use with the ALPR

*D. H. Shaftman

**W. J. Kann

DRAWING LIST - JACK #27

Part No.	Part
1	Pressure Shell Body Flange Tube Extension Flange Pipe Plug
2	Holder Weldment End Disc
3	Armature Weldment
4	Nonmagnetic Sleeve
5	Nonmagnetic Washer
6	Shim
7	Cushion Washer
8	Drive Rod
9	P. I. Rod
10	Move Coil Assembly End Plate Ring Coil
11	Grip Coil Assembly Shell Coil Terminal Post
12	Flange Ring
13	Flanged Tube
14	Flanged Plate
15	Coil Clamp Plate
16	Coil Spacer Ring
17	S Connector
18	C Connector
19	L Connector
20	Gasket - Flexitallic #RL-6J
21	Socket Hd. Sc.-3/4 - 10 x 1-1/2 Lg. Steel
22	Socket Hd. Sc.-1/4 - 20 x 7/8 Lg. SST
23	Round Hd. Sc.-#6-32 x 5/16 Lg. SST
24	Hex. Nut - #6 - 32 SST
25	Lock Washer - #6 SST

FIG. 71
MAGNETIC JACK TYPE
CONTROL ROD DRIVE

reactor. It is recommended that a research and development program be initiated, substituting one magnetic-jack rod drive for an existing rack and pinion type rod drive in one of the shim-rod positions. Subsequently the use of the magnetic-type drive for all the rod drives positions should be considered. The advantages of the magnetic-jack type drive over the existing rack and pinion type drive are (1) that it is hermetically sealed, (2) that there is a minimum of moving parts, and (3) that it operates completely within the reactor top removable shield, thereby eliminating the shaft extensions through the shield blocks and the motor-transmission units on the plant operating floor.

K. Shield Cooling System*

From January 28 through January 31, four days of continuous plant operation, the shield cooling system pump was secured, thereby using natural water circulation as the means of heat removal from the pressure vessel external thermal shield (see Fig. 29). The maximum shield water temperatures were 145F inlet, 177F outlet circuit No. 1 and 175F outlet circuit No. 2. The temperatures varied during the period of operation because of plant power level changes; however, the temperature difference between the shield water inlet and outlet remained between 30F and 35F. During the final two days of plant operation, the water temperature difference remained constant (30F). It appears, in view of the preceding operating experience, that the shield cooling system pump may be eliminated and the system operated by natural convection. This would reduce the condensate circulating pump load by eliminating one of its water circuits. Further testing is recommended, particularly at higher operating floor temperatures and with additional temperature monitoring points.

L. Reactor Biological Shield Experimental Hole**

It is recommended that further consideration be given to possible experiments to be done in the experimental hole (Fig. 72). 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.

M. Instrumentation[†]

1. Reactor water level - high and low scram switches. Investigate the feasibility of substantially reducing the hysteresis associated with the action of the reactor shutdown contacts in the reactor water level primary instrument. These contacts, situated in the differential transformer receiver unit in the Process Panel, presently require a large differential in water level for their actuation and reset.

*W. R. Wallin

**A. D. Rossin

[†] H. H. Hooker

FIG. 72
REACTOR BIOLOGICAL SHIELD EXPERIMENTAL FACILITY

2. Secondary reactor water level instrument. Attempt to find a more accurate and reliable device for the reactor water level secondary alarm function. The present device is too difficult to adjust and is not sufficiently accurate (see Section IV-P-7).

N. Mechanical*

1. Investigate means for providing cooler ambient conditions for the plant ventilating fan and instrument well exhaust fan drives located on top of the fan floor air-lock. These motors are now exposed to condenser outlet air at high temperature and are expected to show a reduced insulation life as a result.

*H. H. Hooker

2. Item 1 also will apply to the condenser fan drive motor as soon as a condenser section No. 1 operates properly with consequent high-temperature outlet air. This motor is presently cooled considerably by the cool air discharged from the adjacent condenser section No. 1.

O. Corrosion and Radiation*

Examination of fuel assemblies at intervals of approximately 3000 Mw-hr, should reveal useful information. Also, a complete analysis of corrosion and radiation damage of a fuel assembly from the high-flux region once each calendar year should be undertaken.

*N. R. Grant

APPENDIX C

PRELIMINARY SYSTEMS TESTING DATA

A typical log of plant operating data is reproduced as this appendix. Data from three separate tests performed during the preliminary systems testing phase (see Section IV) of the plant testing and operating program are represented. These tests are:

1. Systems Steam Test - without the turbine-generator unit (July 19, 1958).
2. Condenser Air Circulating Systems Test (July 19, 1958).
3. Turbine-Generator Unit Test (July 25, 1958).

It should be noted that the plant operating conditions as recorded for the steam systems test are part of the operating data taken during the condenser air circulating system test.

Data Sheet #1

Signature E. E. HamerDate July 19, 1958Run No. 1Type Test SteamStart 0900Stop 1515PROCESS PANELSYSTEMS STEAM TEST

Time		0900	0915	0930	1000	1100	1200	1300	1400	1500
Instru- ment Number	Instrument	Data	Data	Data	Data	Data	Data	Data	Data	Data
PI-1	Feedwater Pressure psig	Out of service								
PI-3	Main Steam Pressure psig	0	305	300	300	300	302	305	302	305
LI-15	Hotwell Water Level Inches	Out of service								
PIC-133	Reactor Steam Pressure psig	Out of service								
FIR-6	Feedwater Flow lbs/hr	Out of service								
TIR-7	Feedwater Temperatures °F	Out of service								
VIR-4	Condenser Vacuum in. Hg	0	16	16½	17	15	15	16	16	15½
TRC-9	Condenser Air Inlet Temperature °F	64	68	72	78	83	85	94	98	82
TRC-11	Condenser Air Outlet Temperature °F	64½	90	91½	84	86	87	111	117	104
LIR-14	Reactor Water Level Inches	Out of service								
FIR-13	Main Steam Flow lb/hr	0	1700	1700	1750	1750	2050	2050	2050	2050
FIR-8	Bypass Steam Flow lb/hr	0	1300	1360	1300	1350	1450	1440	1440	1440
RI-19A	Demineralizer Influent Megohms	Out of service								
RI-19B	Demineralizer Effluent Megohms	Out of service								
RL-28	HP Condensate Return Tank Pump	Off	Off	Off	Off	Off	Off	Off	Off	Off
RL-29	LP Condensate Return Tank Pump	Off	Off	Off	Off	Off	Off	Off	Off	Off
RL-27	Condensate Circulating Pump	On	On	On	On	On	On	On	On	On
RL-30	Shield Cooler Pump	Out of service								
RL-35	Instrument Well Exhaust Fan	Out of service								
	Feedwater Valve Control	Out of service								
	Steam Bypass Valve Set Point Setler psig	300	300	300	300	300	300	300	300	300
	Ambient	64	-	-	71	73	75	82	86	80

REMARKS: Calm until 1300.

Wind S.W. 8-10 mph after 1300.

Condenser Thermocouple corrected 1445

Opening air ejector after condenser vent rotometer (flowmeter) shows continuous water drain.

Data Sheet #2

Signature E. E. HamerDate July 19, 1958Run No. 1Type Test SteamPROCESS PANELSYSTEMS STEAM TEST

		Time	0900	0915	0930	1000	1100	1200	1300	1400	1500
Instrument Number	Instrument	Data	Data	Data	Data	Data	Data	Data	Data	Data	Data
	Condenser Fan Ammeter amps	70	70	74	70	105	105	100	100	95	
RL-31	Condenser Air Circulating Fan Speed Control	Man.	Man.	Man.	Man.	Man.	Man.	Man.	Man.	Man.	
	Air Exhaust Damper Control	100	100	100	100	100	100	100	100	100	
	Air Mixer Damper Control	0	0	0	0	0	0	0	0	0	
RL-33	Water Purification Pump Control	Out of service									
RL-23	Feedwater Pump #1 Control	Out of service									
RL-24	Feedwater Pump #2 Control	Used intermittently to empty hotwell into storage tank									
PI-5	Main Steam Pressure Deviation Indicator psig	Out of service									
	Bypass Steam Valve Control (0-3000 lb/hr) %	Out of service									
	Bypass Steam Valve Control (0-10,000 lb/hr) %	Out of service									
TI-16A	Water Purification °F	Out of service									
	Bypass Valve Position	20	27	27	26	28	30	31	32	33	
TI-16B	Air Precooler Vapor Outlet °F	-	-	-	76	85	95	96	98	100	
TI-16D	Reactor Water °F	Out of service									
TI-16E	Air Ejector After Condenser Vapor Outlet °F	-	-	-	-	-	-	91	92	95	

REMARKS: Vacuum instrument not operating properly as yet.
 Bypass valve put on automatic and oscillates because of small steam volume system.
 Put fan on automatic - work needed.
 Bypass safety valve blown at 50 psig.
 Main steam to condenser safety valve blown at 360 psig.
 Checked out condenser air system.
 Need outside deflector plate to reduce air system short circuiting.
 Repair steam leak at air ejector strainers.
 Feedpump #2 leaks oil from motor oil reservoir.
 Raw water demineralizer meter needs repair.
 Feedpump #1 operated and needs gland seal water supply repiping.
 Air ejector after condenser steam trap will handle $\frac{1}{2}$ gpm to HP condensate return tank.
 LP and HP condensate return units now operating satisfactorily, but switch sticking.
 Condenser fan 40 amp = zero speed.
 160 amp = 100% speed.
 Clean strainers.
 Check pipe hangers
 Cap rod drive manifold
 Cap pressure vessel leak-off

Data Sheet #1

Signature E. E. HamerDate July 19, 1958MISCELLANEOUSSYSTEMS STEAM TESTRun No. 1Type Test Steam

Time		0900	0915	0930	1000	1100	1200	1300	1400	1500
Instru- ment Number	Instrument	Data	Data	Data	Data	Data	Data			
FI-135	Control Rod Cooling Flow gpm	Not installed								
PI-136	Radiological Filter Diff. Press. in. H ₂ O	Out of service								
	Water Storage Tank Level inches	Continuously changing and checked frequently								
	Hotwell Tank Level inches	Continuously changing and checked frequently								
	Shield Cooling System Tank Level inches	Same levels as water storage tank								
FI-117	Simulated Heat Load Sys- tem Flow gpm	0	-	-	-	50	134	-	-	-
TI-230	Simulated Heat Load to Heater Temp. °F	75	-	-	-	130	143	-	-	-
TI-231	Simulated Heat Load from Heater °F	62	-	-	-	173	165	-	-	-
PIC-123	Bypass Steam Pressure psig	35	47	48	37	29	27½	30½	31	31½
TIC-131	Main Steam Relief Valve Disch. Temp. °F	-	-	-	-	-	-	-	-	-
	Condenser Differential Pressure in. H ₂ O	Not installed								
	Condenser Fan Diff. Press. in. H ₂ O	Not installed								
	LP Condensate Return Tank Press. psig	Not installed								
FI-116	Air Ejector After Con- denser Air Flow cfm	Bouncing								
TIC-127	Safety Valve Temperature °F	40	99	150	150	149	135	130	130	150

REMARKS: Total Power from Simulated Heat Load - 424 kw

Flow 134 gpm

 $\rho = 61.1 \text{ lb/ft}^3$

Temperature in 143°F)

Av. 154°F

 $\frac{61.1}{7.48} = 8.17 \text{ lb/gal}$

Temperature out 165°F)

 $\rho = 8.17 \text{ lb/gal.}$

$$P = \frac{(134)(8.17)(165-143)(60)}{3413} = 424 \text{ kw}$$

Started Shield Cooling System

Turbine Generator unit not in service.

Data Sheet #1

Date July 19, 1958Start 0930Stop 1500Signature D. M. LeppkeCONDENSER AIR CIRCULATION SYSTEM TEST

Time ⁽¹⁾	0930	1000	1025	1200	1400	1500	
Location ⁽²⁾	Data	Data	Data	Data	Data	Data	Remarks
Outside Ambient ⁽³⁾ °F	70	71	75	75	86	80	

Mixing Chamber Outlet

Top Center °F	71	72	72	76	82	83	Prior to test, the temperature at the top center was 70°F
Top Right °F	72	72	73	78	81	84	
Center °F	70	-	-	-	78	82	
Bottom Left °F	70	70	72	-	78	82	
Bottom Center °F	70	-	-	77	-	-	
Bottom Right °F	70	71	73	-	79	84	
No. 1 (left) fpm	-	-	-	-	-	1648	Readings in the horizontal sections between each vertical support.
No. 2 fpm	-	-	-	-	-	1540	
No. 3 (center) fpm	-	-	-	-	-	800	
No. 4 fpm	-	-	-	-	-	900	
No. 5 fpm	-	-	-	-	-	628	

(1) Time indicated is the starting time for the readings in sequence shown. The last reading is approximately 25 minutes after start.

(2) Condenser sections are numbered as No. 1 in the center of the building and No. 5 adjacent to the building wall.

(3) Outside ambient temperatures were taken under the outside plant stairway and approximately 4 feet above grade.

Data Sheet #2

Date July 19, 1958Start 0930Stop 1500Signature D. M. LeppkeCONDENSER AIR CIRCULATION SYSTEM TEST

Time ⁽¹⁾	0930	1000	1025	1200	1400	1500	
Location ⁽²⁾	Data	Data	Data	Data	Data	Data	Remarks
Outside Ambient ⁽³⁾ °F	70	71	72	75	86	80	

Condenser Inlet

Section 1 - top	°F	71	71	-	-	82	-	Temperatures taken by dry bulb thermometer.
Section 2 - top	°F	71	72	-	-	-	-	
Section 3 - top	°F	71	72	-	-	81	-	
Section 4 - top	°F	72	72	-	-	-	-	
Section 5 - top	°F	72	72	-	-	80	-	
Section 1 - bottom	°F	70	72	-	-	80	-	
Section 2 - bottom	°F	70	72	-	-	-	-	
Section 3 - bottom	°F	71	72	-	77	81	-	
Section 4 - bottom	°F	71	72	-	-	-	-	
Section 5 - bottom	°F	71	72	-	-	80	-	

Data Sheet #3

Date July 19, 1958
 Start 0930
 Stop 1500
 Signature D. M. Leppke

CONDENSER AIR CIRCULATION SYSTEM TEST

Time ⁽¹⁾	0930	1000	1025	1200	1400	1500	
Location ⁽²⁾	Data	Data	Data	Data	Data	Data	Remarks
Outside Ambient ⁽³⁾ °F	70	71	75	75	86	80	

Condenser Outlet

Section 1 - top	°F	124	77	-	-	128	-	Measured at approximately 6 feet above the fan floor with an anemometer.
Section 3 - top	°F	122	118	-	-	126	-	
Section 5 - top	°F	123	124	-	-	125	-	
Section 1 - bottom	°F	74	74	-	-	82	-	
Section 3 - bottom	°F	74	74	-	-	82	-	
Section 5 - bottom	°F	100	74	-	-	108	-	
Section 1	fpm	-	-	-	-	-	572	
Section 2	fpm	-	-	-	-	-	528	
Section 3	fpm	-	-	-	-	-	552	
Section 4	fpm	-	-	-	-	-	544	
Section 5	fpm	-	-	-	-	-	528	

Miscellaneous

Condenser Air Circulation Fan Motor	°F	-	-	-	-	122	-	Temperatures are ambient.
Building Ventilation Fan Motor	°F	-	113	-	-	123	-	

Condenser Movement (cold to hot)
 approximately $\frac{1}{8}$ " horizontally
 bottom tube sheet moved downward noticeably

Data Sheet #1

Signature E. E. HamerDate July 25, 1958Run No. 1Type Test TurbineStart 1420Stop 1605Process PanelTURBINE-GENERATOR UNIT TEST

		Time	1430	1450	1515	1545		
Instrument Number	Instrument	Data	Data	Data	Data	Data	Data	Data
PI-1	Feedwater Pressure, psig	Out of service						
PI-3	Main Steam Pressure, psig	300	300	295	300			
LI-15	Hotwell Water Level, inches	Out of service						
PIR-12	Reactor Steam Pressure, psig	Out of service						
FIR-6	Feedwater Flow, lb/hr	Out of service						
TIR-7	Feedwater Temperatures, °F	Out of service						
VIR-4	Condenser Vacuum, in. Hg	3½	15½	15½	15½			
TRC-9	Condenser Air Inlet Temperature °F	81	84	86	124			
TRC-11	Condenser Air Outlet Temperature °F	81	104	108	88			
LIR-14	Reactor Water Level, inches	Out of service						
FIR-13	Main Steam Flow, lb/hr	1800	1800	1850	1850			
FIR-8	Bypass Steam Flow, lb/hr	675	700	500	500			
RI-19A	Demineralizer Influent, megohms-CM	Out of service						
RI-19B	Demineralizer Effluent, megohms-CM	Out of service						
RL-28	HP Condensate Return Tank Pump	Off	Off	Off	Off			
RL-29	LP Condensate Return Tank Pump	Off	On	Off	Off			
RL-27	Condensate Circulating Pump	On	On	On	On			
RL-30	Shield Cooler Pump	On	On	On	On			
RL-35	Instrument Well Exhaust Fan	Out of service						
	Feedwater Valve Control	Out of service						
	Steam Bypass Valve set point setler, psig	300	300	300	300			
	Ambient	81.5	81.5	-	-			

REMARKS: Roll Turbine at 1455

Air system on automatic at 1455 - not operating properly

Cond. vacuum not correct. See turbine gauge board.

Took over turbine in control room at 1450.

20 kw load 1505

30 kw load at 1520 for 3 minutes

Drop load at 1600

Data Sheet #2

Signature E. E. HamerDate July 25, 1958Run No. 1Type Test TurbineProcess PanelTURBINE-GENERATOR UNIT TEST

		Time	1430	1450	1515	1545		
Instrument Number	Instrument	Data	Data	Data	Data	Data	Data	Data
	Condenser Fan Ammeter, amps	60	165	115	65			
RL-31	Condenser Air Circulating Fan Speed Control	Manual	Automatic	Manual	Manual			
	Air Exhaust Damper Control	100	100	100	100			
	Air Mixer Damper Control	0	0	0	0			
RL-33	Water Purification Pump Cont.	On	On	On	On			
RL-23	Feedwater Pump #1 Control	Out of service						
RL-24	Feedwater Pump #2 Control	Used intermittently to empty hotwell into storage tank						
PI-5	Main Steam Pressure Deviation Indicator, psig	0	0	0	0			
	Bypass Steam Valve Control (0-3000 lb/hr) %	Manual 16	Manual 16	Manual 10	Manual 10			
	Bypass Steam Valve Control (0-10,000 lb/hr) %	Out of service						
TI-16A	Water Purification °F	85	85	85	85			
TI-16B	Air Pre-cooler Vapor Outlet °F	-	100	100	120			
TI-16D	Reactor Water, °F	85	85	85	85			
TI-16E	Air Ejector After Condenser Vapor Outlet, °F	-	90	90	96			

REMARKS: Annunciation set - shield pump disch. press - to high

shield cooler outlet temp. - to low

Shield cooler bypass valve not working

Steam leak - 3/8" elbow to bypass flow nozzle (high press. inst. line)

2 leak tub. throttle trip valve

Condenser Air Temperature Out is approx. 10% below set point when on automatic

Data Sheet #1

Signature L. S. HicksDate July 25, 1958Run No. 1Type Test TurbineLocal Panel #1TURBINE-GENERATOR UNIT TEST

		Time	1445	1515	1545			
Instrument Number	Instrument	Data	Data	Data	Data	Data	Data	Data
TIC-128	Air Ejector After Condenser Temperature, °F	132	137	145				
PI-205	Steam to Air Ejector Condenser Pressure, psig	277	270	260				
PI-206	Steam to Air Ejector Gland Pressure, psig	278	270	260				
VI-207	HP Condensate Return Tank Vacuum, in. Hg	4.8	4.5	4.3				
TI-225	Gland Seal Leakage Temperature °F	138	142	154				

Local Panel #2

PIC-125	Feedwater Pump #1 Pressure, psig	Out of service					
PIC-126	Feedwater Pump #2 Pressure, psig	0	0	365			
TI-223	Condensate to Pre-cooler °F	142	144	156			
TI-224	Condensate from Pre-cooler Temperature, °F	143	144	156			
TI-232	Condensate from Hotwell Temperature, °F	117	120	129			

Local Panel #3

TI-129	Turbine Oil Tank Oil Outlet Temperature, °F	90	97	104			
PI-209	Hot Water Heater Shell Pressure, psig	Out of service					

REMARKS: Check repair of turbine aux. oil pump check valve
 Check reset of aux. oil pump limit switches to relieve hunting.
 Vent air to oil turbine oil cooler.

Data Sheet #1

Signature L. S. HicksDate July 25, 1958Run No. 1Type Test TurbineMiscellaneousTURBINE-GENERATOR UNIT TEST

		Time	1445	1515	1545			
Instrument Number	Instrument		Data	Data	Data	Data	Data	Data
FI-135	Control Rod Cooling Flow, gpm		Not installed					
PI-136	Radiological Filter Diff. Press. in. H ₂ O		Out of service					
	Water Storage Tank Level, inches		Continuously changing and checked frequently					
	Hotwell Tank Level, inches		Continuously changing and checked frequently					
	Shield Cooling System Tank Level, inches		Same level as water storage tank					
FI-117	Simulated Heat Load System Flow, gpm		Out of service					
TI-230	Simulated Heat Load to Heater, °F		Out of service					
TI-231	Simulated Heat Load from Heater, °F		Out of service					
PIC-123	Bypass Steam Pressure, psig		21	14	14.5			
TIC-131	Main Steam Relief Valve Disch. Temp. °F		125	125	130			
	Condenser Differential Pressure, in. H ₂ O		.91	.92	.92			
	Condenser Fan Diff. Press. psig		.5	.35	.3			
	LP Condensate Return Tank Press. psig		0	0	0			
FI-116	Air Ejector After Condenser Air Flow, cfm		Bouncing	Bouncing	Bouncing			
TIC-127	Safety Valve Temp., °F		-	-	-			

REMARKS:

Data Sheet #1

Signature L. S. HicksDate July 25, 1958Run No. 1Type Test TurbineTurbine PanelTURBINE-GENERATOR UNIT TEST

		Time	1445	1515	1545			
Instrument Number	Instrument	Data	Data	Data	Data	Data	Data	Data
	Turbine Speed, rpm	4515	4500	4500				
VI-212	Exhaust Vacuum, in. Hg.	22	21.9	20.3				
PI-211	First Stage Press., psig	30" Hg	18" Hg	30" Hg				
PI-210	Inlet Pressure, psig	297	290	290				
TI-233	Steam End Bearing Drain Temperature, °F	98	117	114				
PI-213	Bearing Oil Pressure, psig	10.8	8.3	8.7				
TI-234	Exhaust End Bearing Drain Temperature, °F	91	100	106				
TI-237	Gear Drain Temp., °F	87	100	108				
TI-249	Oil Tank Temp., °F	86	94	102				
TI-250	Oil Cooler Outlet Temp., °F	86	97	104				
PI-215	Exhaust End Gland Press., psig	2.1	2.1	2.2				
PI-214	Steam End Gland Press., psig	1.8	1.9	2.1				
TI-235	In-Board Generator Bearing Temp., °F	100	115	122				
TI-236	Outboard Generator Bearing Temp., °F	93	122	172				
	Excess Oil Pressure, psig	53	50	50				

REMARKS: Check phasing, 10 kw load, voltage regulator operation;
synchroscope and light, etc.
Check on Recorder Vacuum Unit

<u>Turbine panel</u>	<u>Recorder</u>
12	9.5
17	12.5
20.5	14.5

Overspeed trip at 4875 rpm.

Data Sheet #1

Signature L. S. HicksDate July 25, 1958Run No. 1Type Test Purification
System and
Shield Cooler
FlowPurification System
PanelTURBINE-GENERATOR UNIT TEST

		Time	1430	1445	1500	1515	1530	1545
Instrument Number	Instrument	Data	Data	Data	Data	Data	Data	Data
FI-115	Purification System Flow, gpm	2.7	2.7	2.7	2.7	2.7	2.7	2.7
FI-114	Retention Tank Flow, gpm	0	0	0	0	0	0	0
TI-241	Thermal Shield #1 Temperature, °F	72	73	83	85	85	86	
TI-242	Thermal Shield #2 Temperature, °F	75	75	89	90	95	97	
PI-201	Purification System Pump Pressure, psig	40	40	40	40	40	40	
TIC-220	Purification System Temperature, psig	85	85	86	86	87	86	
PI-202	Retention Tank Pump Pressure, psig	0	0	0	0	0	0	
TIC-226	Shield Cooling Coils Cir. #2 Outlet Temperature, °F	80	96	105	105	108	110	
TIC-227	Shield Cooling Coils Cir. #2 Outlet Temperature, °F	78	96	102	103	105	108	
TIC-229	Shield Cooling Coils Inlet Temperature, °F	110	107	110	110	108	117	
LI-118	Retention Tank Water Level	3/8	3/8	3/8	3/8	3/8	3/8	
TI-221	Retention Tank Water Temperature, °F	75	79	84	85	87	88	
PIC-243	Shield Cooler Pump Pressure, psig	48.6	48.2	48	48	48	48	

REMARKS: Purification system in series flow → Cation → Mixed Bed
Shield Cooling flow is 15 gpm

Data Sheet #1

Signature E. E. HamerDate July 19, 1958LOCAL PANEL #1SYSTEMS STEAM TESTRun No. 1Type Test Steam

Instrument Number	Instrument	Time	0900	0915	0930	1000	1100	1200	1300	1400	1500
		Data	Data	Data	Data	Data	Data	Data	Data	Data	Data
TIC-128	Air Ejector After Condenser Temperature °F	75	143	145	145	157	165	152	154	160	
PI-205	Steam to Air Ejector Condenser Pressure psig	30	255	252	252	241	228	249	240	245	
PI-206	Steam to Air Ejector Gland Pressure psig	0	245	240	240	224	238	220	165	247	
VI-207	HP Condensate Return Tank Vacuum in. Hg	0	5	5½	5½	5	5	5	4	5½	
TI-225	Gland Seal Leakage Temperature °F	80	93	110	120	133	138	136	135	137	

LOCAL PANEL #2

PIC-125	Feedwater Pump #1 Pressure psig	Out of service									
PIC-126	Feedwater Pump #2 Pressure psig	0	0	0	0	360	360	360	0	0	
TI-223	Condensate to Pre-cooler, Temp. °F	80	145	147	148	159	166	153	154	161	
TI-224	Condensate from Pre-cooler, Temp. °F	80	144	145	147	158	166	153	154	160	
TI-232	Condensate from Hotwell Temp. °F	80	129	130	130	143	150	135	137	137	

LOCAL PANEL #3

TIC-129	Turbine Oil Tank Oil Outlet Temperature °F	Out of service									
PI-209	Hot Water Heater Shell Pressure psig	0	0	0	0	8½	8	0	0	0	

REMARKS: Remove Handle from Feedpump bypass valves.
 Insulate flash tank vent pipe.

APPENDIX DPLANT PERFORMANCE TEST DATA

A typical log of plant operation data taken during the Plant Performance Test phase of the plant testing and operating program (see Section VII) is reproduced as this appendix. Instrument charts for one day and log data sheets for one eight-hour shift on the same day (November 30, 1958) are included.

REACTOR STEAM PRESSURE
NOVEMBER 30, 1958
0-500 psig

REACTOR WATER LEVEL
NOVEMBER 30, 1958
-6 FT. ZERO +5 FT.

MAIN STEAM FLOW
NOVEMBER 30, 1958
0-10,000 lb/hr.

BYPASS STEAM FLOW
NOVEMBER 30, 1958
0-3000 lbs./hr.

CONDENSER VACUUM
NOVEMBER 30, 1958
30 in. Hg. ZERO 30psi

FEEDWATER FLOW
NOVEMBER 30, 1958
0-10,000 lbs./hr.

FEEDWATER TEMPERATURE
NOVEMBER 30, 1958
0-400°F

CONDENSER AIR TEMPERATURE IN
NOVEMBER 30, 1958
0-300°F

CONDENSER AIR TEMPERATURE OUT
NOVEMBER 30, 1958
0-300°F

Page 1

CONTROL ROOM LOG

DEVICES TO BE CHECKED	Time of Readings							
	0830	0930	1030	1130	1230	1330	1430	1530
	3×10^{-4}	3×10^{-4}	3×10^{-4}	3×10^{-4}	3×10^{-4}	3×10^{-4}	3×10^{-4}	3×10^{-4}
Linear Power Electrometer	4.3	4.4	4.2	4.2	4.2	4.2	1.9	4.0
Rod #1 Position	22.2	22.2	22.2	22.2	22.2	22.2	22.4	22.4
Rod #2 Position	--	--	--	--	--	--	--	--
Rod #3 Position	22.2	22.2	22.2	22.2	22.2	22.2	22.4	22.4
Rod #4 Position	--	--	--	--	--	--	--	--
Rod #5 Position	22.2	22.2	22.2	22.2	22.2	22.2	22.4	22.4
Rod #6 Position	--	--	--	--	--	--	--	--
Rod #7 Position	22.2	22.2	22.2	22.2	22.2	22.2	22.4	22.4
Rod #8 Position	--	--	--	--	--	--	--	--
Rod #9 Position	22.5	22.5	22.5	22.5	22.5	22.8	18.8	22.1
Log Power	17	18	18	18	18	18	8.5	19
Fission Break Monitor	55	55	60	55	58	60	30	49
Area Monitor #1	32	31	3.3	3.3	3.0	3.2	.51	3.0
Area Monitor #2	15	14	13	14	13	14	7.8	14
Area Monitor #3	4.1	4.1	4.1	4.1	4.1	4.1	4.0	4.1
Area Monitor #4	.72	.72	.72	.72	.72	.72	.71	.70
Area Monitor #5	--	--	--	--	--	--	--	--
Feedwater - Pressure psig	353	350	348	348	348	348	358	349
Feedwater - Flow lb/hr	7750	7950	7980	7980	7980	8300	3100	8100
Feedwater - Temperature °F	175	174	173	173	173	176	174	171
Main Steam Pressure psig	299	299	300	300	298	300	300	300
Reactor Pressure psig	303	303	304	303	303	304	297	303
Reactor Water Level ft.	-.2	0	0	0	+.25	0	0	0

Date November 30, 1958Signature J. A. Parks

Record readings once each hour. Report abnormal conditions to Shift Supervisor.

CONTROL ROOM LOG

DEVICES TO BE CHECKED	Time of Readings							
	0830	0930	1030	1130	1230	1330	1430	1530
Main Steam Flow lb/hr	8140	8120	8180	8200	8180	8400	4000	8150
P-P ₀ psi	-.5	-.3	-.3	-.4	-.5	-.5	-.2	-.4
By Pass Steam Flow lb/hr	1420	1500	1500	1450	1410	1475	1525	1500
Condenser Vacuum in. Hg	14.8	15.0	15.2	15.0	15.0	13.5	19.3	14.2
Water Conductivity Influent Megohm-cm	.78	.8	.82	.83	.84	.84	.94	.82
Water Conductivity Effluent Megohm-cm	1.35	1.4	1.36	1.36	1.38	1.35	1.55	1.4
Hotwell Level Inches	24	24.2	22.5	22.6	22.4	22.5	21.0	22.25
Condenser Air Temp. In °F	40	42	38	42	43	49	45	52
Condenser Air Temp. Out °F	117	117	117	120	122	129.5	84	132
Turbine Generator kw	300	298	300	300	300	300	90	300
Turbine Generator volts	122	121	120	120	120	120	120	120
Turbine Generator kw-hrs	0681	0684	0688	0690	0693	0696	0698	0700
Turbine Generator hours	331	332	333	334	335	336	337	338
Utility Bus volts	123	121	122	122	122	122	131	120
Utility Bus kw	33	32	26	30	25	30	22	20
Emergency Power a-c volts	208	208	208	208	208	208	208	208
Emergency Power a-c amps	5.8	5.8	5.8	5.8	5.8	5.8	5.8	5.8
Emergency Power d-c volts	132	132	133	132	132	132	132	132
Emergency Power d-c amps	0	0	0	0	0	0	0	0
Diesel Generator amps	0	0	0	0	0	0	0	0

Date November 30, 1958Signature J. A. Parks

Record readings once each hour. Report abnormal conditions to Shift Supervisor

Page 3

CONTROL ROOM LOG

DEVICES TO BE CHECKED	Time of Readings							
	0830	0930	1030	1130	1230	1330	1430	1530
Diesel Generator volts	0	0	0	0	0	0	0	0
Diesel Generator hours	66	66	66	66	66	66	66	66
Simulated Heat Load Flow gpm	132	128	128	128	126	127	127	127
Simulated Heat Load Temp. In °F	152	162	167	167	166	164	169	170
Simulated Heat Load Temp. Out °F	177	188	193	192	191	194	193	195
Dummy Load kw	-	-	230	230	230	230	10	230

Date November 30, 1958Signature J. A. Parks

Record readings once each hour. Report abnormal conditions to Shift Supervisor.

CONTROL ROOM LOG
TEMPERATURES (°F)

DEVICES TO BE CHECKED	Time of Readings							
	0830	0930	1030	1130	1230	1330	1430	1530
Water Purification	153	153	152	152	153	156	130	152
Air Precooler Vapor Outlet	145	145	144	144	146	154	99	147
Water from Thermal Shield Cooling Coil	159	159	159	158	159	161	152	149
Reactor Water	415	415	415	415	416	415	415	415
Air Ejector aftercondenser Vapor Outlet	58	59	54	57	60	66	57	68
Operating Floor #1	63	63	63	70	71	74	68	69
Operating Floor #2	64	65	66	68	72	74	77	76
Gravel - #1	82	82	82	82	83	83	84	84
Gravel - #2	82	82	82	82	82	82	82	81
Gravel - #3	69	69	69	68	68	69	69	69
Gravel - #4	58	58	58	57	57	58	58	59
Sub Reactor Shield - #1	74	74	74	73	73	74	74	73
Sub Reactor Shield - #2	74	73	73	72	72	72	73	72
Reactor Shield Circuit - #1	151	151	151	150	150	151	147	142
Reactor Shield Circuit - #2	156	157	157	156	156	158	152	149
Fuel Storage Well - #1	57	57	57	57	57	57	57	57
Fuel Storage Well - #2	73	73	74	73	73	74	74	74
Fuel Storage Well - #3	34	34	34	34	34	34	34	34
Fuel Storage Well - #4	56	56	56	56	56	56	56	56
Fuel Storage Well - #5	36	41	41	41	41	41	41	41
Fuel Storage Well - #6	54	54	54	53	53	53	53	54
Concrete Shield - #1	95	95	95	94	94	94	94	95
Concrete Shield - #2	108	108	108	108	108	108	109	109
Instrument Well - #1	65	66	67	68	70	72	74	74
Instrument Well - #2	64	65	67	68	70	72	74	76
Instrument Well - #3	64	65	67	68	70	72	74	74
Instrument Well - #4	63	64	66	67	69	71	73	73
Control Rod #1	114	114	114	114	115	117	114	115

Date November 30, 1958

Signature J. A. Parks

Record readings once each hour. Report abnormal conditions to Shift Supervisor

Page 2

CONTROL ROOM LOG
TEMPERATURES (°F)

DEVICES TO BE CHECKED	Time of Readings							
	0830	0930	1030	1130	1230	1330	1430	1530
Control Rod #2	-	-	-	-	-	-	-	-
Control Rod #3	111	110	111	111	112	114	108	109
Control Rod #4	-	-	-	-	-	-	-	-
Control Rod #5	103	103	112	103	104	106	102	105
Level Transmitter	124	125	125	125	126	127	127	126
Control Rod #7	119	119	119	119	120	122	121	120
Control Rod #8	-	-	-	-	-	-	-	-
Control Rod #9	126	123	123	123	124	131	109	130
Outside Air	16	22	29	34	38	39	41	41
Air Space above #1 Reactor	123	122	123	123	124	125	123	125
Air Space above #2 Reactor	96	96	96	96	95	98	97	97
Air Space above #3 Reactor	106	106	107	107	107	109	108	110
Reactor Water	392	392	392	393	393	392	392	392
Hotwell Water	150	150	149	148	151	157	113	154

Date November 30, 1958Signature J. A. Parks

Record readings once each hour. Report abnormal conditions to Shift Supervisor.

OPERATING FLOOR LOG

Instrument Number	Device to be Checked	Normal Full Power Readings	Time 1030	Time 1330
	Condensate Circ. Pump *	On	On	On
Sight Glass	Hotwell Water Level	18-24 in.	24	23
	Feedwater Pump #1 *	One Pump	On	On
	Feedwater Pump #2 *	Standby	Off-Auto	Off-Auto
Flow Sight Glass	Feed Pump Gland Seal	On	On	On
Flowmeter	Rod Drive Seal Water	120 gph (9 drives)	40	40

INSTRUMENT PANELS #1 AND #2

PI 205	Steam Press. to Cond. Air Ejec.	275 psig	282	282
PI 206	Steam Press. to Gland Air Ejec.	275 psig	270	270
VI 207	HP Cond. Return Tank Vacuum	3 in. Hg	4.5	3.5
TIC 128	Air Ejec. After Cond. Temp.	200°F	178	180
T1 225	Gland Seal Leak Temp.	150°F	172	177
T1 224	Cond. from Pre-cooler Temp.	148°F	174	178
T1 223	Cond. from Pre-cooler Temp.	142°F	175	179
T1 232	Cond. from Hotwell Temp.	135°F	154	159
PIC 125	Feedwater Pump #1 Pressure	If on 350 psig	355	355
PIC 126	Feedwater Pump #2 Pressure	If on 350 psig	0	0
F1 116	Air Ejec. After Cond. Disch. Flow	12 cfm	10	9.5
	Cond. Return HP* (Automatic)	On if running	Off	Off
	Fuel Storage Wells	-	-	-
	Fuel Storage Wells Water Level below Floor	33-45 in.	-	-
	Steam Leaks	-	-	-

WATER PURIFICATION PANEL

F1 115	Total Flow	3-5 gpm	1.7	1.2
F1 113	Mixed Bed Flow	3-5 gpm	0	0

* Observe bearings for temperature and vibration

Date November 30, 1958Signature J. A. Parks

Record readings once each 8 hour shift. Report abnormal conditions to the Shift Supervisor.

OPERATING FLOOR LOG

Instrument Number	Device to be Checked	Normal Full Power Readings	Time 1030	Time 1330
P1 201	Purif Pump Press.	340 psig	350	350
P1 202	Ret. Tank Pump Press.	0 psig	0	0
P1 243	Shield Cooler Pump	40 psig	36	36
T1 229	Shield Cooler Inlet Temp.	140-150°F	157	155
T1 221	Ret. Tank Temp.	40°F or amb.	48	49
T1C 226	Shield Coil #1 Outlet Temp.	180-190°F	166	157
T1C 227	Shield Coil #2 Outlet Temp.	180-190°F	164	156
T1 118	Ret. Tank Level	$\frac{1}{2}$ Full	$\frac{9}{16}$	$\frac{9}{16}$
T1 241	Lead Shield Temp.	150-190°F	155	147
T1 242	Lead Shield Temp.	150-190°F	165	158
238	Shield Cooler Pump*	On	On	On
239	Purification Water Pump*	On	On	On
240	Retention Tank Water Pump*	Off	Off	Off
T1 220	Purification System Temp.	150-170°F	169	159
Sight Glass	Shield Cooler Surge Tank	18-24 in.	about full	$\frac{7}{8}$ full
	Boron Storage Tank	Full	$\frac{3}{4}$ full	$\frac{3}{4}$ full
P1 133	Reactor Steam Press.	300 psig	305	305
T1C 131	Main Steam Safety Valve Discharge Temp.	120°F	190	190
P1 203	Main Steam Press.	300 psig	305	305

FAN FLOOR

	Condenser Fan and Drive	On	On	On
PC 136	Rad. Filter Diff. Press.	1.5 in. H ₂ O	.02	.021
RIC 123	Bypass Valve Disch. Press.	40 psig	44	45
TIC 129	Main Steam Relief Disch. Temp.	120°F	145	145
	Building Vent Fan	On	Off	Off
	Condensate Return LP* (automatic)	On if running	Off	Off

* Observe bearings for temperature and vibration.

Date November 30, 1958Signature J. A. Parks

Record readings once each 8 hour shift. Report abnormal conditions to Shift Supervisor.

OPERATING FLOOR LOG

Instrument Number	Device to be Checked	Normal Full Power Readings	Time 1030	Time 1330
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TURBINE GAUGE BOARD

298	295	275 psig	Steam Inlet	275 psig
37	37	275 psig	1st Stage Press.	275 psig
15.1	15.5	20 in. Hg	Exhaust Press.	15.1
105	106	120-129°F	Steam End Bearing Temp.	105
	9.75	8 psig	Bearing Oil Pres.	9.75
112	100	120-129°F	Exhaust End Bearing Temp.	112
103	101	120-129°F	Gear Drain Temp.	103
108	103	120-129°F	Oil Tank Temp.	108
106	96	120-129°F	Oil Cooler Temp.	106
	1.9	2.2 psig	Steam End Gland Press.	1.9
	2.75	0.25 psig	Exhaust End Gland Press.	1.8
	4550	4514 rpm	Turbine Speed	4550
	Normal	-	Turbine Bearing Vibration	Normal
118	114	110°F	Turbine Oil Tank Outlet Temp.	118
PI 209	19	20 psig	Hotwater Heater Shell Press.	15.5

Sight Glass	Water Storage Tank	18-24 in.	34	35
Flow Sight Glass	Raw Water Demin. Flow	2½ gpm	-	-
RIC 121	Raw Water Demin. Water Quality	.5 megohm-cm	2.2	-
Pushbutton Light	Raw Water Supply Pump (when demin. or oil cooler is on)	On	On-Off	Off

NUCLEAR INSTRUMENT PANEL

Voltneter	Channel II Voltage Positive	300 volts	295	295
Voltneter	Channel III Voltage Negative	40 volts	15	15
Voltneter	Channel IV Voltage Positive	300 volts	290	290
Voltneter	Channel IV Voltage Negative	40 volts	12	12
Voltneter	Channel V High Voltage	2000 volts	-	-
Voltneter	Channel VI High Voltage	2000 volts	-	-

Date November 30, 1958

Signature J. A. Parks

Record readings once each 8 hour shift. Report abnormal conditions to Shift Supervisor.

OPERATING FLOOR LOG

Instrument	Device to be Checked	Normal Full Power Readings	Time 1030	Time 1330
Voltmeter	Channel VIII High Voltage	1000	1000	1000

MOTOR CONTROL CENTER

Motor Control	Hydraulic Coupling Oil Cooler Fan	On running	On	On
Motor Control	Building Vent Fan	On running	Off	Off
Motor Control	Building Crane	Off	Off	Off
Motor Control	Mono Rail Crane	Off	Off	Off
Motor Control	Condenser Return Pump LP	Off-Auto	Off-Auto	Off-Auto
Motor Control	Instrument Well Vent Fan	On running	On	On
Motor Control	Shield Cooler Pump	On running	On	On
Motor Control	Retention Tank Pump	Off	Off	Off
Motor Control	Feed Pump #1	One pump	On	On
Motor Control	Feed Pump #2	On Standby	Off-Auto	Off-Auto
Motor Control	Standby Turbine Oil Pump	Off-Auto	Off-Auto	Off-Auto
Motor Control	Condenser Circulating Fan	On running	On	On
Motor Control	Condensate Circulating Pump	On running	On	On

Date November 30, 1958Signature J. A. Parks

Record readings once each 8 hour shift. Report abnormal conditons to Shift Supervisor.

ROUTINE OPERATORS HEALTH PHYSICS SURVEY LOG SHEET

Power Level 3 MWDate November 30, 1958Surveyor LeeOPERATING FLOOR

	Time	0815	1000	1130	1230	1440	1930
1. Average around shield		$\frac{1}{12}$	$\frac{2}{13}$	$\frac{2}{15}$	$\frac{2}{15}$	$\frac{1}{15}$	15
2. Feedwater pump No. 1		1	1	1	1	1	-
3. Feedwater pump No. 2		1.5	2	1.5	1.5	1.5	-
4. Simulated heat load heat exchanger		105	100	95	95	30	25
5. End, simulated heat load heat exchanger		2	2	2	2	1	-
6. Bypass steam line		6.5	8.5	9	9	4	-
7. Main steam line		7	7	7.5	8	3	-
8. Middle, turbine control panel		4	5	4	4	2	-
9. Turbine casing		5	6	5.5	5	2	10
10. Main steam line, by exit light		6	7	6	6	3	-
11. Purification system panel		7	9	8	8	5	25
12. Feedwater filter		24	25	26	26	21	37
13. Main steam line, behind purification system panel		28	28	29	28	22	40
14. Air ejector flow meter		5	5.4	5	5	2.5	-
15. Reactor top (center)		1	1	-	1	1	-

FAN FLOOR

1. Main steam line	-	2	-	-	-	-
2. Exhaust steam line	-	3	-	-	-	-
3. Bypass steam line	-	2	-	-	-	-
4. Condenser bank (first section)	-	13	-	-	-	-
5. Area background (wall near entrance)	-	1	-	-	-	-

MISCELLANEOUS

1. Control room, rear of panel	-	.07	-	.06	-	-
2. Lower landing, by plate	.06	.06	.07	.07	.06	-
3. Between lower and middle landing	.04	.04	.04	.04	.03	-
4. Middle landing	.02	.03	.02	.03	.03	-
5. Doorway, operating floor level	1	1	1	1	.5	-
6. Doorway, chest height	1	1	1.5	1.5	.5	.5

INSTRUCTIONS: All surveys will be made with the JUNO survey meter. Record results in mr (milliroentgens/hr). Surveys should be performed hourly, available time permitting.

WARNING: If the general background level on the reactor operating floor rises above 30 mr/hr, evacuate the operating floor immediately and contact Health Physics personnel.

WARNING: If the Constant Air Monitor switches to the high range (Red flasher and alarm bell on for more than 30 seconds), evacuate the reactor operating floor immediately and contact Health Physics personnel.

If either of the two warning conditions described above should occur and it is necessary for operating personnel to enter the reactor operating floor, personnel will wear Scott Air Paks, coveralls, shoe covers, and gloves. Such equipment will be put on at the foot of the stairs and will be removed at the same location after completion of the job. Under these conditions, personnel will spend an absolute minimum of time on the operating floor.

REMARKS: Lowered power - all dummy load off at 1430.

APPENDIX E

POWER OPERATION EXPERIMENTS AND ADDITIONAL PLANT PERFORMANCE TESTS DATA

In order to analyse the behavior and response time of various instruments and controls, and to determine overall performance of the automatic power demand control system, a Brush Electronics Company recorder was used. The resultant strip charts from various experiments are reproduced as this appendix.

Included are the charts for Experiments No. 144, (Neutron Flux During Partial Injection of Feedwater in Upper Spray Ring), and No. 175, (Variation in Neutron Flux at 5750 lbs/hr Steam Flow), as discussed in Section VI-C. The charts for the automatic power demand control system experiments are noted as Experiments No. 191 through 210B (Figs. 75 through 94) and are discussed in Section IX-B. Recorded, in each experiment, were, (1) rod position, (2) rod (motion) direction, (3) (steam) pressure, (4) steam flow rate, and (5) (neutron) flux level.

NOTE: REACTOR STEAM FLOW AND FEEDWATER FLOW WERE 4350 lb./hr. VALVE TO UPPER SPRAY RING WAS OPENED TWO TURNS WITH VALVE TO LOWER SPRAY RING OPEN FULL. STEAM HAMMER AND PIPE VIBRATION PREVENTED COMPLETE DIVERSION THROUGH UPPER SPRAY RING.

FIG. 73
NEUTRON FLUX DURING PARTIAL INJECTION OF FEED WATER IN UPPER SPRAY RING
EXPERIMENT NO. 144

FIG. 74
VARIATION IN NEUTRON FLUX AT 5750 lb./hr. STEAM FLOW. EXPERIMENT NO. 175

FIG. 75
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 191. PROPORTIONAL BAND 200%. ZERO RESET.

FIG. 76
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 192. PROPORTIONAL BAND 100%. ZERO RESET.

FIG. 77
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 193. PROPORTIONAL BAND 50%. ZERO RESET.

FIG. 78
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60 KW-STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 194. PROPORTIONAL BAND 200%. RESET 0.1.

FIG. 79
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 195. PROPORTIONAL BAND 200%. RESET 0.3.

FIG. 80
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 80-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 196. PROPORTIONAL BAND 200%. RESET 1.0.

FIG. 81
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 197. PROPORTIONAL BAND 200%. RESET 2.0.

FIG. 82
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 198. PROPORTIONAL BAND 200%. RESET 3.0.

FIG. 83
 TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 199. PROPORTIONAL BAND 200%. RESET 6.0.

FIG. 84
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 200. PROPORTIONAL BAND 175%. RESET 1.0.

FIG. 85
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND AT 60 SECOND INTERVALS. EXPERIMENT NO. 202. PROPORTIONAL BAND 175%. RESET 1.0.

FIG. 86
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND AT 50 SECOND INTERVALS. EXPERIMENT NO. 203. PROPORTIONAL BAND 175%. RESET 1.0.

FIG. 87
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND AT 40 SECOND INTERVALS. EXPERIMENT NO. 204. PROPORTIONAL BAND 175%. RESET 1.0.

FIG. 88
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND AT 30 SECOND INTERVALS. EXPERIMENT NO. 205. PROPORTIONAL BAND 175%. RESET 1.0.

FIG. 89
 TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND AT 20 SECOND INTERVALS. EXPERIMENT NO. 206. PROPORTIONAL BAND 175%. RESET 1.0.

FIG. 90
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND AT 30 SECOND INTERVALS. EXPERIMENT NO. 207. PROPORTIONAL BAND 175%. RESET 1.0. NO SLIDEWIRE LIMIT SWITCHES.

FIG. 91
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 60-KW STEP CHANGES IN ELECTRICAL DEMAND AT 20 SECOND INTERVALS. EXPERIMENT NO. 208. PROPORTIONAL BAND 175%. RESET 1.0 NO SLIDEWIRE LIMIT SWITCHES.

FIG. 92
TRANSIENT RESPONSE OF PLANT PARAMETERS TO 120-KW STEP CHANGES IN ELECTRICAL DEMAND. EXPERIMENT NO. 209. PROPORTIONAL BAND 175%. RESET 1.0

FIG. 93
 TRANSIENT RESPONSE OF PLANT PARAMETERS TO A 180-KW STEP INCREASE IN ELECTRICAL DEMAND. EXPERIMENT NO. 210A. PROPORTIONAL BAND 175%. RESET 1.0

FIG. 94
TRANSIENT RESPONSE OF PLANT PARAMETERS TO A 180-KW STEP DECREASE IN ELECTRICAL DEMAND. EXPERIMENT NO. 2108. PROPORTIONAL BAND 175%. RESET 1.0

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