APAE-18

REACTORS—POWER

UNITED STATES ATOMIC ENERGY COMMISSION

INITIAL OPERATION AND TESTING OF THE ARMY PACKAGE POWER REACTOR APPR-1

Edited by
J. L. Meem

August 9, 1957

Alco Products, Inc.
Schenectady, New York

Technical Information Service Extension, Oak Ridge, Tenn.

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INITIAL OPERATION AND TESTING

OF THE

ARMY PACKAGE POWER REACTOR

APPR-1

Contract No. AT(11-1)-318

J. L. Meem, Editor

August 9, 1957

Alco Products, Inc.
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Schenectady, New York

SUMMARY

This report describes the startup and initial operation of the APPR-1 and summarizes the results of the test program on the reactor. The latter program began with testing of the vapor container for leaks, checking the integrity of the entire primary and secondary systems, and testing of individual components. Most of this testing was done during plant construction.

In March of 1957, pre-nuclear testing of the primary system was performed. The system was operated at full temperature and pressure using dummy fuel elements in the core. The reactor was brought critical on April 8, and was generating electrical power one week later on April 15. Numerous low and high power performance tests were performed and the reactor reached its full rated power at 10 megawatts of heat on April 20.

A formal dedication ceremony was held on April 29. On June 2, the plant was started on an endurance test. By July 2 the plant had operated for 700 hours with a down time of 7 hours and 24 minutes during this period due to equipment malfunctioning.

Particularly noteworthy during the testing was the response of the reactor to load demand, as the stability of the APPR-1 exceeded all expectations. The transients are described in Chapter IX of this report.

The temperature coefficient of the reactor was measured as $-2.3 \times 10^{-4}/^{o}F$. and the pressure coefficient as 2 to 4×10^{-6}/psi at the operating conditions of $450^{o}F$ and 1200 psi. Measurements were made to demonstrate that the reactor would be cooled by thermal convection in the event of loss of primary coolant flow.

Data is presented on xenon buildup, shielding, and overall plant heat balance, and the control of water purity in both the primary and secondary system is described.

The APPR-1 has proven itself to be a reliable nuclear power plant with remarkable stability and it has operated so as to meet all design requirements.

ACKNOWLEDGMENTS

In a project such as the APPR-1 it is impossible to select the names of individuals for special recognition, since so many made notable contributions.

Alco wishes to acknowlege the excellent cooperation and assistance of its subcontractors, particularly the Stone and Webster Engineering Corporation and the Minneapolis Honeywell Regulator Company. Invaluable assistance was rendered by the Oak Ridge National Laboratory.

Appreciation is expressed for the cooperation of the Army Reactors Branch of the AEC, the Schenectady Operations Office of the AEC, and the Nuclear Power Branch of ERDL.

INITIAL OPERATION AND TESTING

OF THE

ARMY PACKAGE POWER REACTOR

APPR-1

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CHAPTER I - INTRODUCTION

(J. L. Meem)

1. General

The Army Package Power Reactor was designed, constructed and is being operated by Alco Products, Inc. under contract with the Army Reactors Branch of the A. E. C. at Fort Belvoir, Virginia. The APPR-1 is a prototype of a reactor designed to meet the requirements and site conditions of a remote military base. Since the prototype reactor is constructed at a site in the United States, some of the design requirements were changed to meet these needs. In particular, containment of the maximum credible accident is provided. The reactor is to be used as a training facility for troops and specialists who might eventually be required to operate and service remote plants. The requirement that all components be transportable by air still exists even though the site is not remote.

2. Description (G. E. Cash)

The APPR-1 is a 10,000 kilowatt pressurized water reactor delivering 1,825 net kilowatts of electricity with 2.5 inches of mercury back pressure (85^{o}F. condenser cooling water). Where lower temperature condenser water is available, back pressure can be reduced, permitting delivery of 1,925 net kilowatts at 1.5 inches of mercury back pressure. The fuel elements are similar to those in the MTR but are clad with stainless steel rather than aluminum, and in addition to the fissionable material contain a burnout poison in the form of boron.

The reactor operates at a pressure of 1,200 psia and an outlet temperature of 450^{o}F. at full power. Two primary coolant pumps in parallel are provided either of which will provide the design flow rate of 4,000 gpm. The water flows from the reactor to a steam generator, where heat is transferred to the secondary (steam) system and from the generator the water flows back through the pump to the reactor inlet. This entire primary loop is installed inside a vapor container 32 feet in diameter and 60 feet high (See Figure II-1). The enclosure will contain the energy released from all of the steam generated by flashing of the superheated primary and secondary system water volumes when the maximum amount of heat has been stored in these volumes.

Outside the vapor container is located the spent fuel pit providing storage for spent fuel elements from the core. In the adjacent quadrant is located the concrete demineralizer room containing the demineralizers, vapor container ventilation equipment, etc. Also adjacent to the vapor container is the main building which houses the laboratory, instrument repair room, general maintenance area, pump room and electrical room on the ground floor. Above this area are located the offices, classroom, turbine room and control room.

Design details of the APPR-1 have been published in previous reports (See references 1-7). A list of references may be found at the end of this report.

3. History and Objectives

The concept of the APPR-1 originated at the Oak Ridge National Laboratory.[8] The objective of the design concept was to provide a nuclear power plant which would meet the power requirements of a remote military base. All components were to be light and compact enough so that they could be carried in existing air cargo ships, and the reactor was to be assembled at the site, hence the term "package" reactor. The design was based on known technology so far as practical consistent with reasonable cost.

In 1954 it was decided that a prototype reactor would be built under the

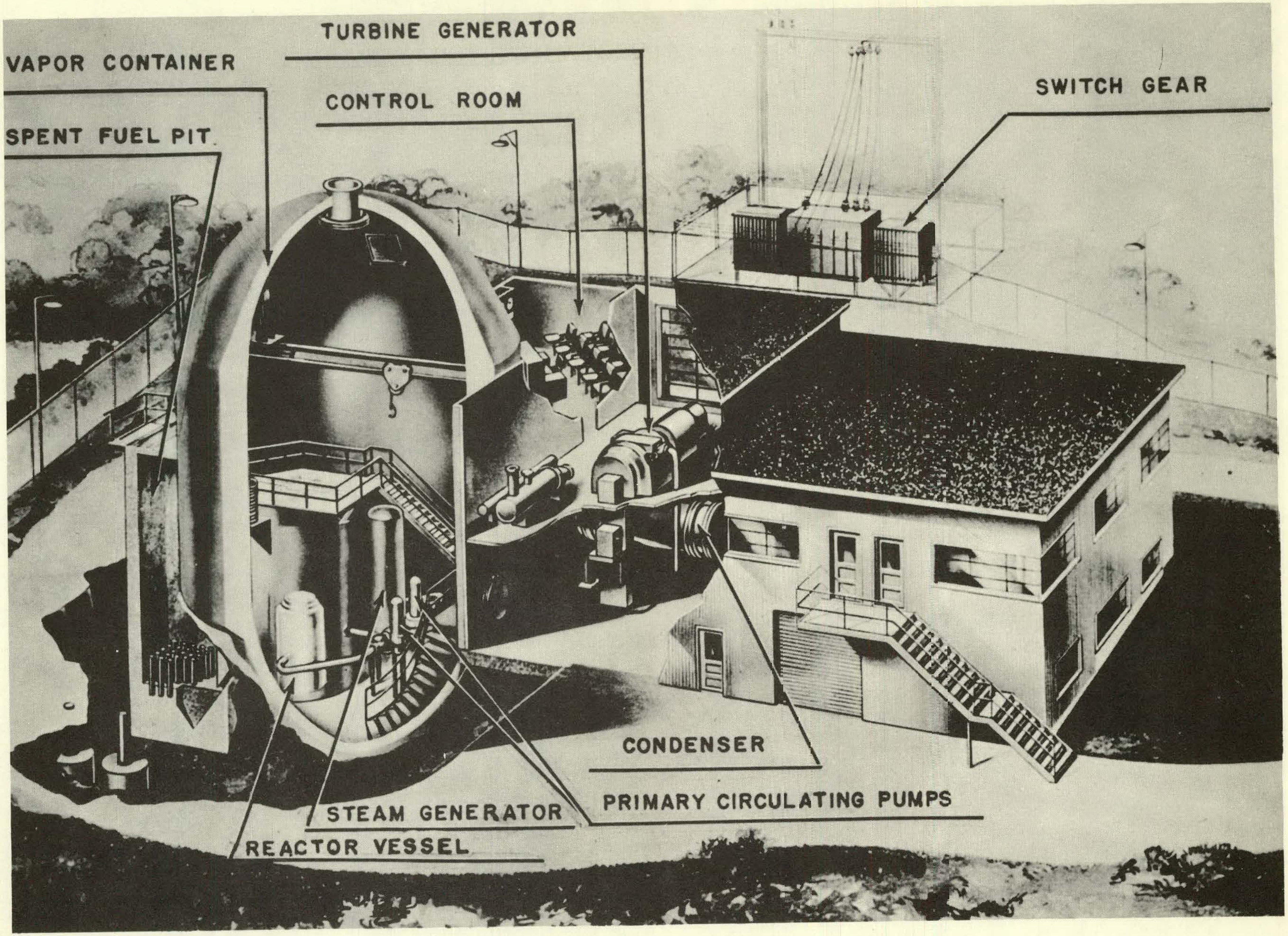


FIGURE I - 1
CUTAWAY VIEW OF THE APPR-1

joint auspices of the Atomic Energy Commission and the Department of Defense. A site was selected at the U. S. Army Engineer Research and Development Laboratories at Fort Belvoir, Virginia. A number of firms were invited to submit proposals for the design, construction, and test operation of the reactor, and the contract was awarded to Alco in December of 1954.

The major objectives of the APPR are to:

1. Solve technical construction and operation problems associated with a reliable nuclear power plant.

2. Provide firm cost information, operating parameters, and engineering test data necessary to adapt the system to a remote location and to improve operating characteristics.

3. Provide a training facility for troops and specialists who might eventually be required to operate and service remote plants.

4. Organization

The organization of the operating crew during plant startup is given in Fig. I-2. Those marked with an asterisk (*) comprised the military personnel who had been given training by Alco, as described in the next section.

It is to be noted that these men played key positions in the plant startup. The teamwork between military and civilian personnel was excellent.

In Fig. I-3 is shown the organization of the Alco Atomic Energy Engineering Department during the design and construction phases of the project. At various times as the plant went into operation and the inevitable problems with individual items of equipment arose, practically every individual in the department was called for specialized support.

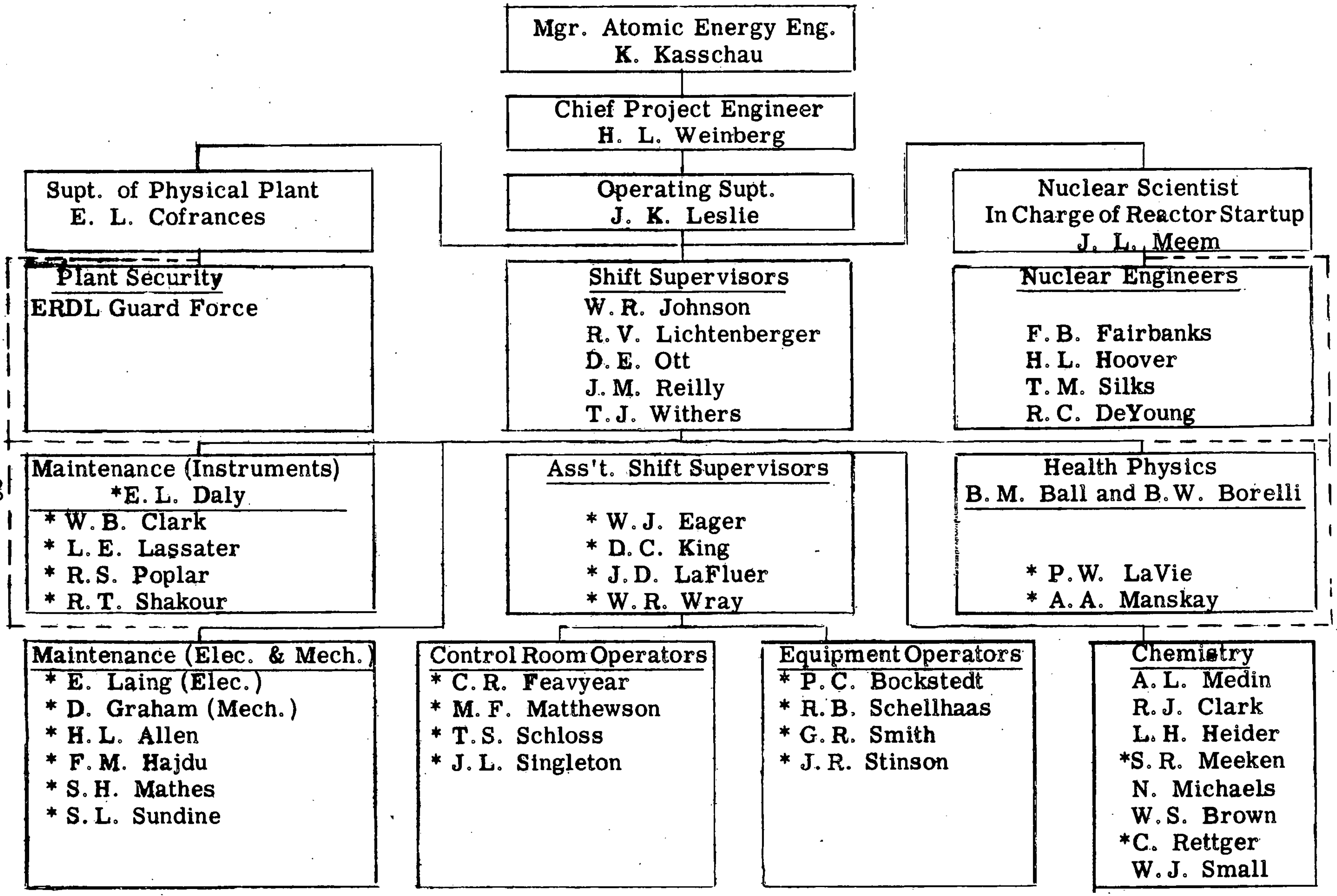


FIG. I-2 - APPR-1 START-UP ORGANIZATION CHART
* Denotes Military Personnel

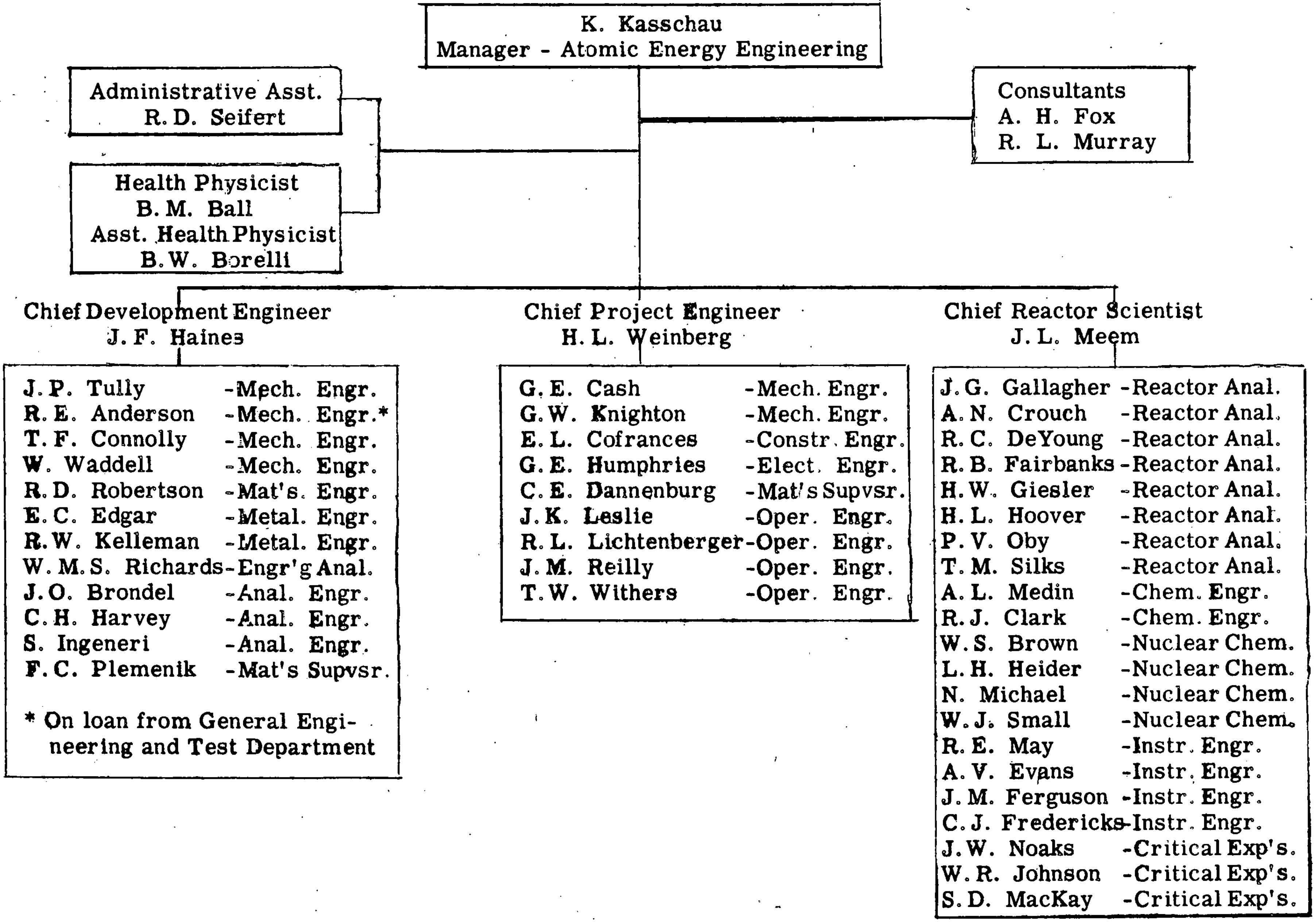


FIG. I-3 ORGANIZATION CHART OF THE ALCO ATOMIC ENERGY ENGINEERING DEPARTMENT DURING DESIGN AND CONSTRUCTION OF APPR-1

5. Crew Training (G. E. Humphries)

5.1 General

A cadre of military personnel was provided to Alco for training in the operation of the reactor prior to plant startup. These men had basic nuclear training at the University of Virginia during the summer of 1956 and were then sent to Alco for training first in Schenectady and then at Fort Belvoir. The plant was operated by these men from the start under Alco supervision.

5.2 Training at Schenectady

The classwork dealt with a review of some basic engineering principles, the application of these principles to the APPR-1 systems and equipment, study of the characteristics of the component pieces of equipment, the integration of the equipment into component systems, the interrelation of the component systems in the overall plant, and plant operations. The health physics-chemistry technicians were given also a more intensive lecture and laboratory course in water treatment chemistry and health physics. Four officers and four non-commissioned officers were given additional training in reactor operation at the Alco Criticality Facility. The officer and non-commissioned officers of the instrumentation group were given supplementary instrumentation training at Minneapolis-Honeywell Instrument Company and Foxboro Instrument Company plus several months' experience on the Criticality Facility instrumentation and studying the APPR-1 instrumentation.

5.3 Training at Fort Belvoir

The operating systems were reviewed and studied to orient the operators from the standpoint of physical location and appearance. The emphasis was placed on operating procedures, techniques, and philosophy, and certain primary operations were studied in detail and simulated as close as practical. Check lists and safety procedures were covered. Maintenance schedules, routine maintenance procedures, extraordinary maintenance problems, and maintenance tips were presented, and demonstrated where practical. To a large degree the army personnel were used in test and pre-operational work.

The health physics-chemistry personnel were given a separate course in water treatment, radiochemistry and health physics tests, analysis, and operating procedures and practice. The instrumentation group was responsible for instrument check-out as the equipment was installed.

6. Chronology

Following is a chronology of the significant operations starting with the prenuclear operations and continuing through the completion of the 700-hour test. Numerous down times were encountered. These were almost all due to troubles and maintenance in conventional equipment which seems to be the typical pattern in the startup of all reactors to date.

March 12	- Started Loading of Dummy Fuel Elements
March 13	- Completed Loading of Dummy Fuel Elements - Performed Demonstration of Changing Rod Drive Seals
March 16	- Started Heat-up of Primary System for Dummy Core Test
March 17	- Reached Operating Temperature and Pressure in Primary System
March 22	- Interrupted Dummy Core Test
March 29	- Resumed Dummy Core Test

April 2	- Finished Dummy Core Test
April 5	- Started Unloading of Dummy Fuel Elements
April 6	- Completed Unloading of Dummy Fuel Elements
April 7	- Started Loading of Live Fuel Elements
April 8	- Reactor Critical
April 9	- Reactor Core Fully Loaded
April 10	- Began Measurements on Temperature and Pressure Coefficients
April 13	-Raised Reactor Power Level to 100 KW of Heat Equivalent to One Percent of Full Power
April 15	- Turbine Generator on the Line Generating 500 KW of Electricity. Reactor Power Level at About 2-1/2 Megawatts of Heat.
April 20	- Reached Full Power Level of 10 Megawatts of Heat and 2200 Gross Kilowatts of Electricity
April 29	- Dedication Ceremony for APPR-1
June 2	- Start of 700 Hour Test
July 2	- 700 Hour Test Completed

CHAPTER II - PLANT CONSTRUCTION AND EQUIPMENT TESTING

(G. E. Cash and G. W. Knighton)

1. Plant Construction

Site clearing and excavating started in the early fall of 1955, and the plant was completed in March of 1957.

Initial work included excavation for the vapor container which required a retaining wall of sheet piling and driving of temporary piling for the construction of the circulating water structure. Work then progressed on the structure foundations. The work on both the building and vapor container, (See Figure II-1) commenced with the erection of the steel shell of the vapor container, structural steel, siding and roofing. The vapor container was tested under the ASME Code for Unfired Pressure Vessels and additional tests which included 100 percent radiographing and a helium mass spectrometer leak test. Upon completion of testing, the concrete outside of the container required for shielding was poured. Simultaneously, the two-foot steel reinforced concrete liner for the steel shell which is required for missile protection and assisting in shielding requirements was poured.

During this period of time some of the major equipment was moved into the building such as the turbine room crane, turbo generator, evaporator, feed-water heater, pumps, and electrical switchgear components. Upon completion of the containment vessel, the primary equipment including the reactor vessel, steam generator, pressurizer and primary pumps were lowered through the upper equipment access opening and mounted on their respective supports. Figure II-2 shows the reactor vessel being lowered into the vapor container, and Figure II-3

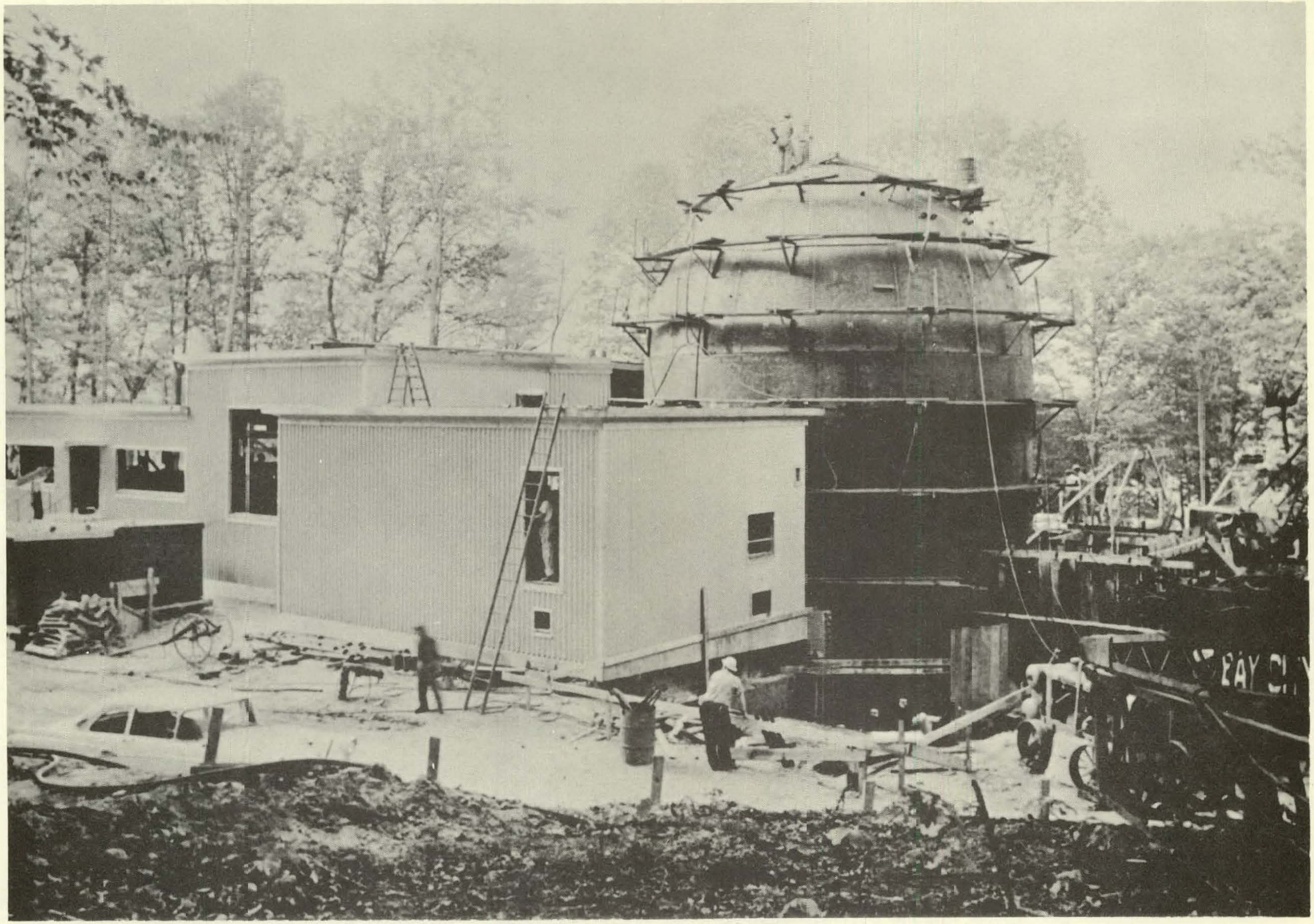


FIG. II - 1 VIEW OF VAPOR CONTAINER AND BUILDING UNDER CONSTRUCTION

FIG. II - 2 REACTOR VESSEL BEING LOWERED INTO VAPOR CONTAINER

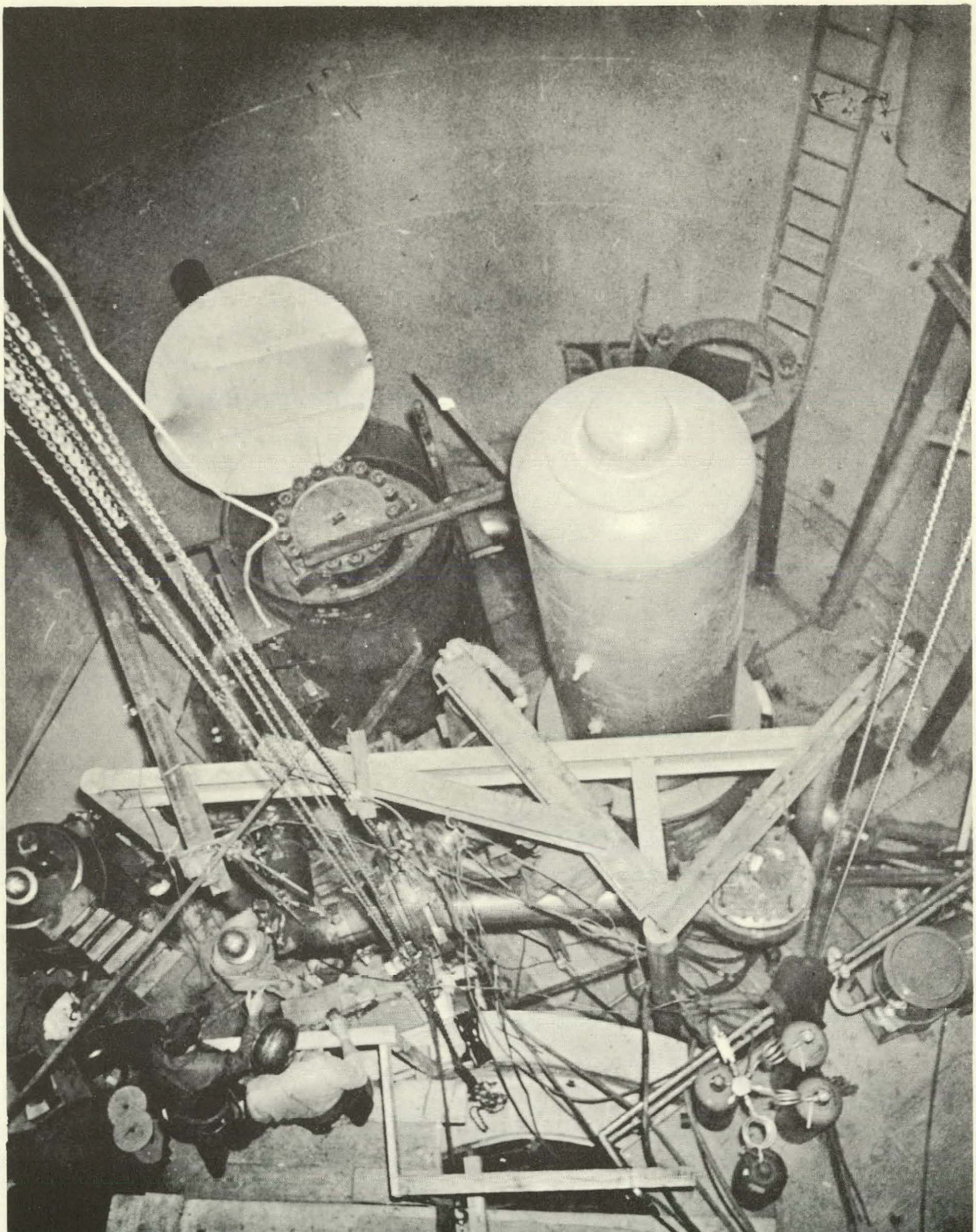


FIG. II - 3 REACTOR VESSEL AND STEAM GENERATOR IN PLACE shows the vessel in place before the primary shielding was added. To the right of the reactor vessel is the steam generator.

Upon completion of the installation of the primary coolant piping, the primary loop was given radiographic, hydrostatic and helium tests after which the primary shielding and shield tank and miscellaneous other equipment was installed to complete the construction work in the vapor container. During this later phase of work in the vapor container, the remainder of the equipment including instrumentation, water treatment equipment, facilities, components, etc., were installed and connected in the buildings surrounding the vapor container.

Figure II-4 is a view of the turbine room with the main steam line and turbine in the foreground, enclosed generator in the background, and the evaporator and feedwater heater (hidden) located to the left of the turbine. Construction was nearing completion at this time.

Figure II-5 is a view from the top of the shield tank looking down into the reactor core, and Figure II-6 shows the completed plant. The Control Room is shown in Figure II-7. Immediately in front of the console is the graphic panel with the nuclear panel on the right and the electrical panel on the left.

2. Component and System Testing (G. W. Knighton)

2.1 General

Reliable and safe operation of APPR-1 was achieved by requirements set forth in the fabrication and testing of all primary system components and major secondary system components. Strict inspection by Alco representatives assured fulfillment of design criteria.

2.2 Primary Components

All vessels were designed and tested with the ASME Code for Unfired

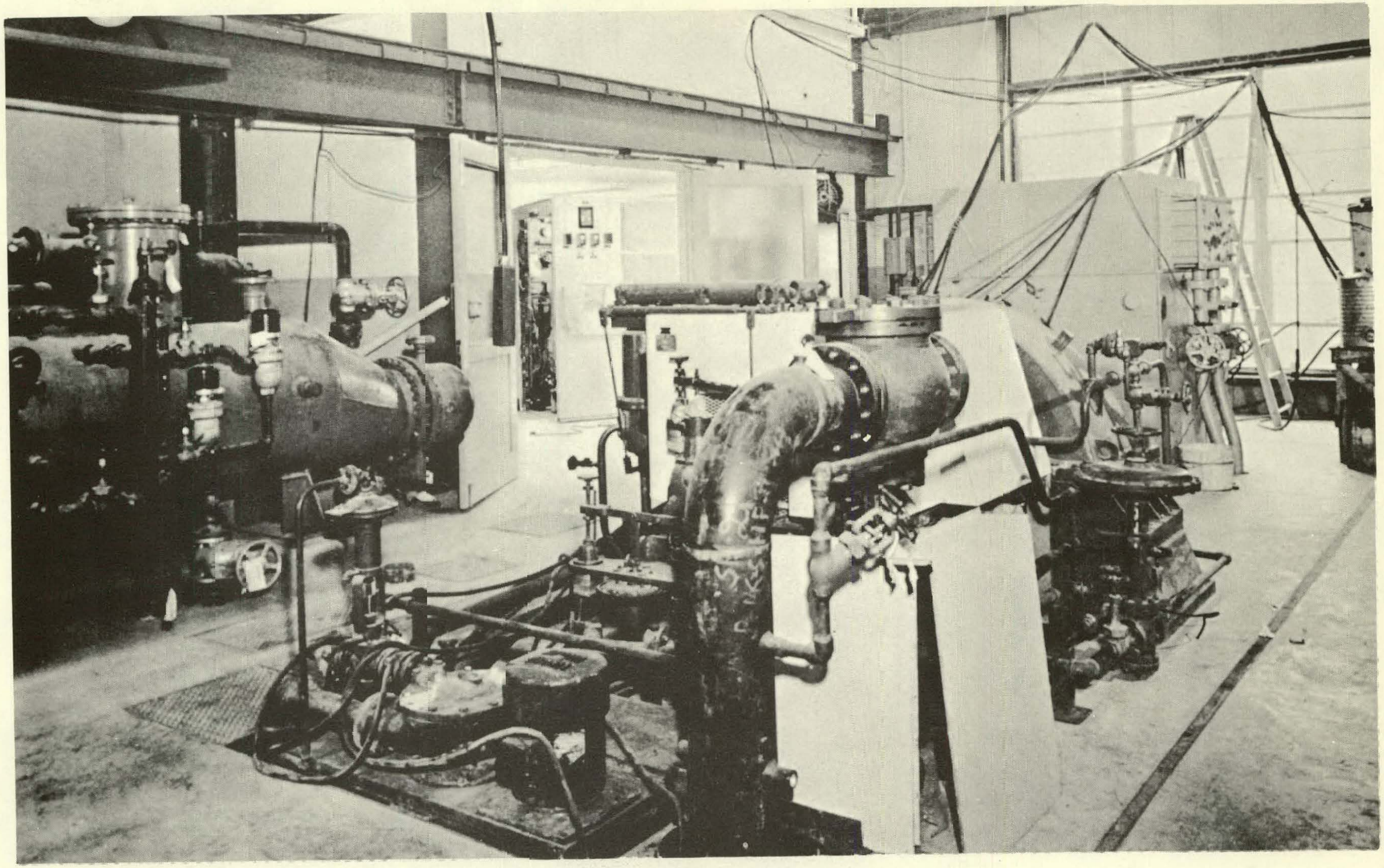


FIG. II - 4 VIEW OF TURBINE ROOM NEAR COMPLETION OF CONSTRUCTION

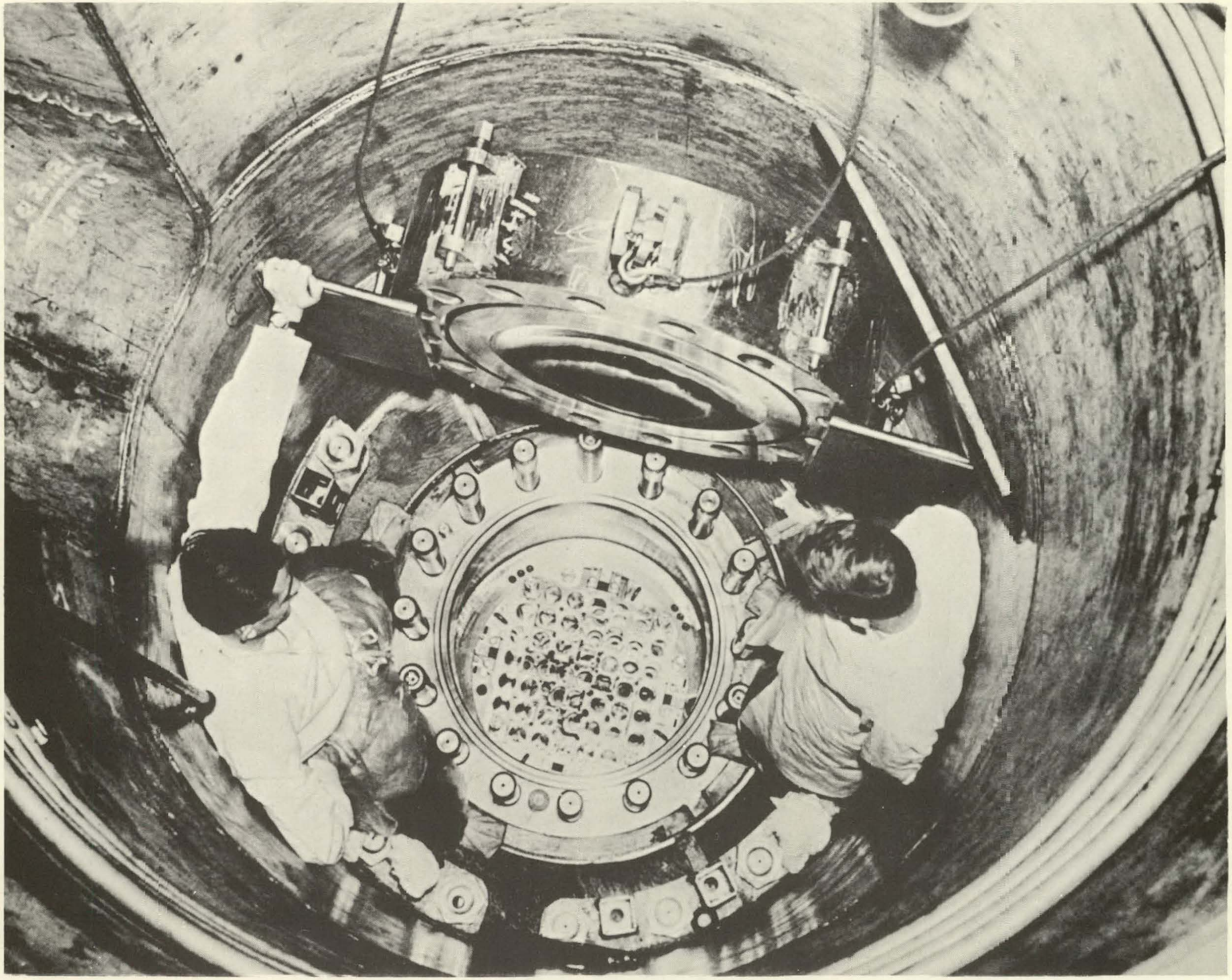


FIG. II - 5 REACTOR CORE AS VIEWED FROM TOP OF SHIELD TANK

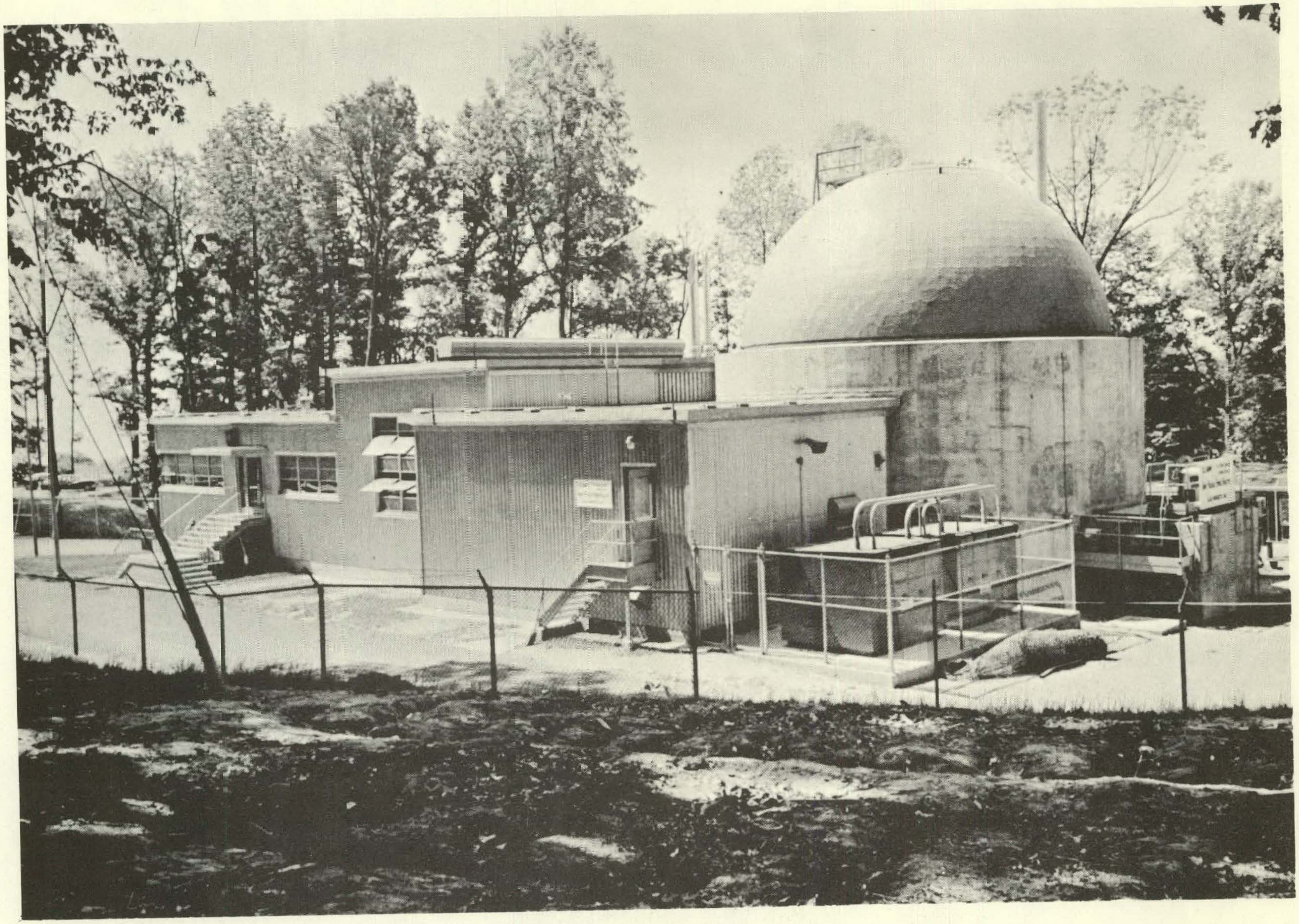


FIG. II-6 GENERAL VIEW OF PLANT

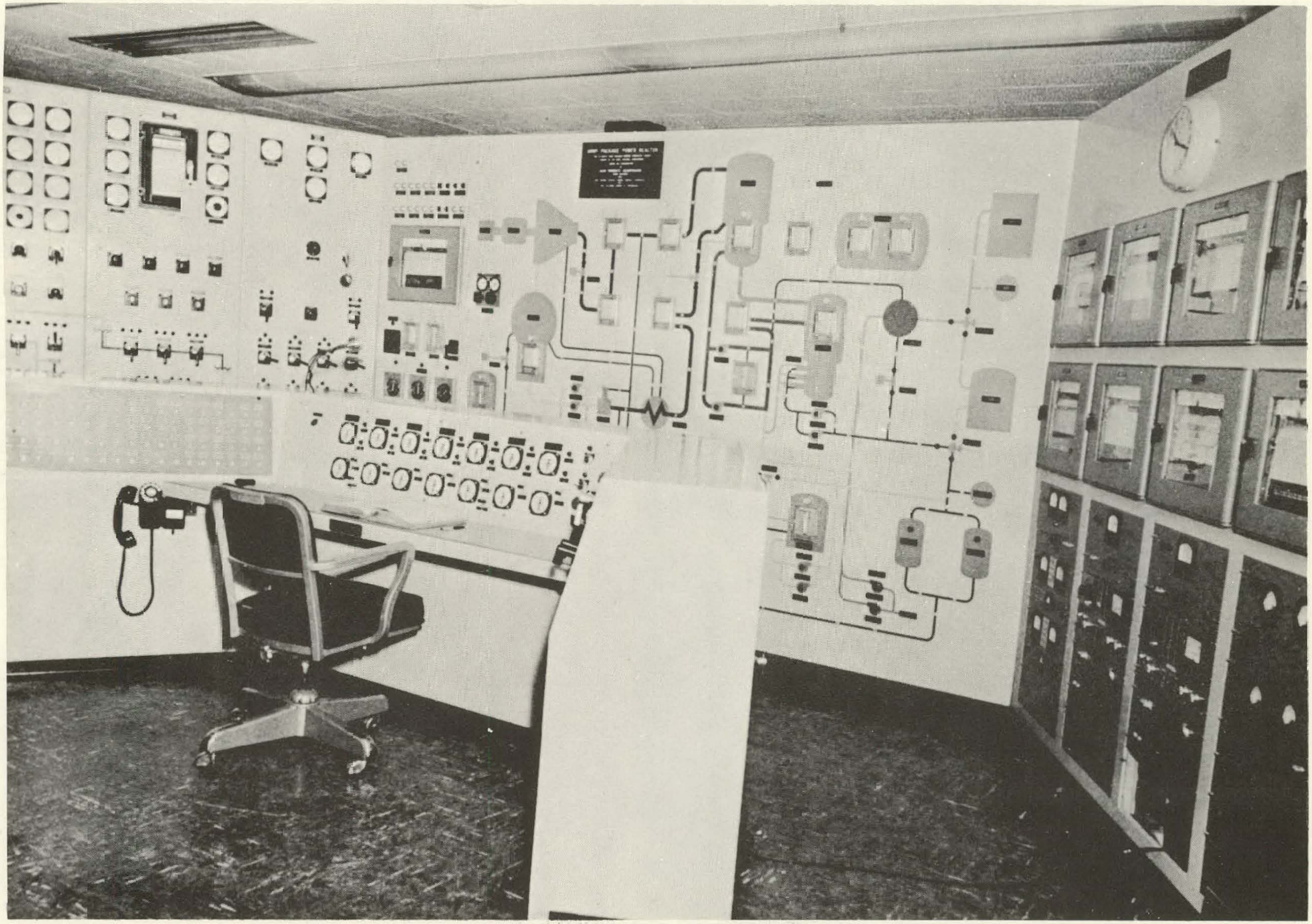


FIG. II - 7 CONTROL ROOM

Pressure Vessels and all other applicable codes as a minimum requirement. Stringent cleanliness requirements and 125 rms interior surface finish was maintained. In accordance with requirements each component was hydrostatically tested. In addition, each vessel was leak checked by a helium mass spectrometer technique. This test was performed by the standard vacuum method with the sensitivity of 1×10^{-5} cc/sec. No leaks were found in any of the components even at the gasketed joints. Each component, including piping, was shipped with all openings sealed with expandable rubber plugs and filled with inert gas to 4-5 psig pressure to insure cleanliness on arrival at the site. All valves were pressure tested by the manufacturer prior to shipment.

2.3 Secondary Components

The major components of the secondary system were fabricated with strict inspection by Alco representatives. They were shop tested prior to shipment to insure proper operation. The condenser was hydrostatically tested and given a Freon leak check prior to shipment. This unit was shipped by rail with special shock absorbing mounts to prevent any damage to the tube-to-tube sheet joints. Prior to installation under the turbine pedestal a second Freon test was made to insure leak tightness.

2.4 Primary System Testing

Upon completion of installation of the vessels, the piping was welded in and the joints radiographed to insure leak tightness. The vessels, piping and valves were then subjected to a helium mass spectrometer leak test to guarantee the leak tight integrity of the system. The sensitivity of the detector was 1×10^{-5} cc/sec. In addition, a hydrostatic test of the system was performed at

1.5 times the design pressure.

2.5 Secondary System Testing

Erection of all equipment was made with careful check for alignment and support. Rotation of all rotating equipment was checked to prevent damage to equipment. Piping was hydrostatically tested in accordance with ASME Code requirements.

2.6 Subsequent Systems Testing

Upon completion of component and system testing a portable oil-fired steam generator was attached to the secondary system and was used to heat the primary system to temperature and pressure for the non-nuclear test described in Chapter III. All instrumentation was checked for proper operation. During the non-nuclear test a larger portable oil-fired steam generator was installed to permit operation of the turbine generator set, thus checking out the operation of the secondary system and its instrumentation at low generator loading.

CHAPTER III - PRE-NUCLEAR OPERATION

(J. L. Meem)

1. General Description

Upon completion of construction and prior to loading of uranium into the reactor core, a non-nuclear test program was conducted. A set of dummy fuel elements was provided which were identical to the actual fuel elements except that the fuel plates were of stainless steel throughout. The reactor core was loaded with these non-fuel-bearing fuel elements and the primary system was heated to 450^{O}F and the pressure raised to 1200 psi. The system was operated under these conditions for several days. A number of tests were conducted during this operation as follows:

1. Demonstration of loading and unloading the reactor core.
2. Demonstration of changing a rod drive seal.
3. Demonstration of techniques in maintaining control of primary water purity.
4. Test the action of the control rod drives under operating conditions.
5. Determine the effect on the fuel elements of short time operation at temperature.

The primary system was heated by passing steam from an auxiliary boiler through the secondary system into the main steam generator. With a primary coolant pump in operation this heat was then distributed throughout the reactor vessel and entire primary system. Primary system pressure could be controlled either with the pressurizer or by running the primary makeup pumps and restricting the rate of blowdown.

While not a part of the dummy core tests, it is worthwhile mentioning that while running those tests, time was available to connect an additional auxiliary boiler of larger capacity to the turbine and check out the turbine generator equipment and the electrical switch gear. The electrical generator was synchronized with Virginia Electric Power Company (VEPCO) and put on the line at several hundred kilowatts during this period.

In Figure III-1 is shown a plot of temperature vs. time during the dummy core tests. The system was at a temperature of 430°F or greater for 134 hours and above 400°F for 169 hours. System pressure is plotted for the same period.

The period of shutdown from March 22 to March 29 was devoted to equipment repair and maintenance.

2. Demonstration of Core Reloading (T. F. Connolly)

During the final construction and start-up period, demonstrations were held on the various operations connected with fuel handling. The times of these various operations were recorded in the operating log. The core was loaded with dummy fuel and the cover installed before the dummy core test. After the dummy run this fuel was unloaded to the spent fuel pit. These operations were all performed using inexperienced crews and it is felt that many of the times could be shortened with experienced crews.

It is assumed that some preliminary work could be done during the reactor cool down period. The vapor container could be opened, the shield tank covers removed, the railings installed and the tank lights installed. Perhaps the spent fuel tube seal could be removed, but the time is included in the following table.

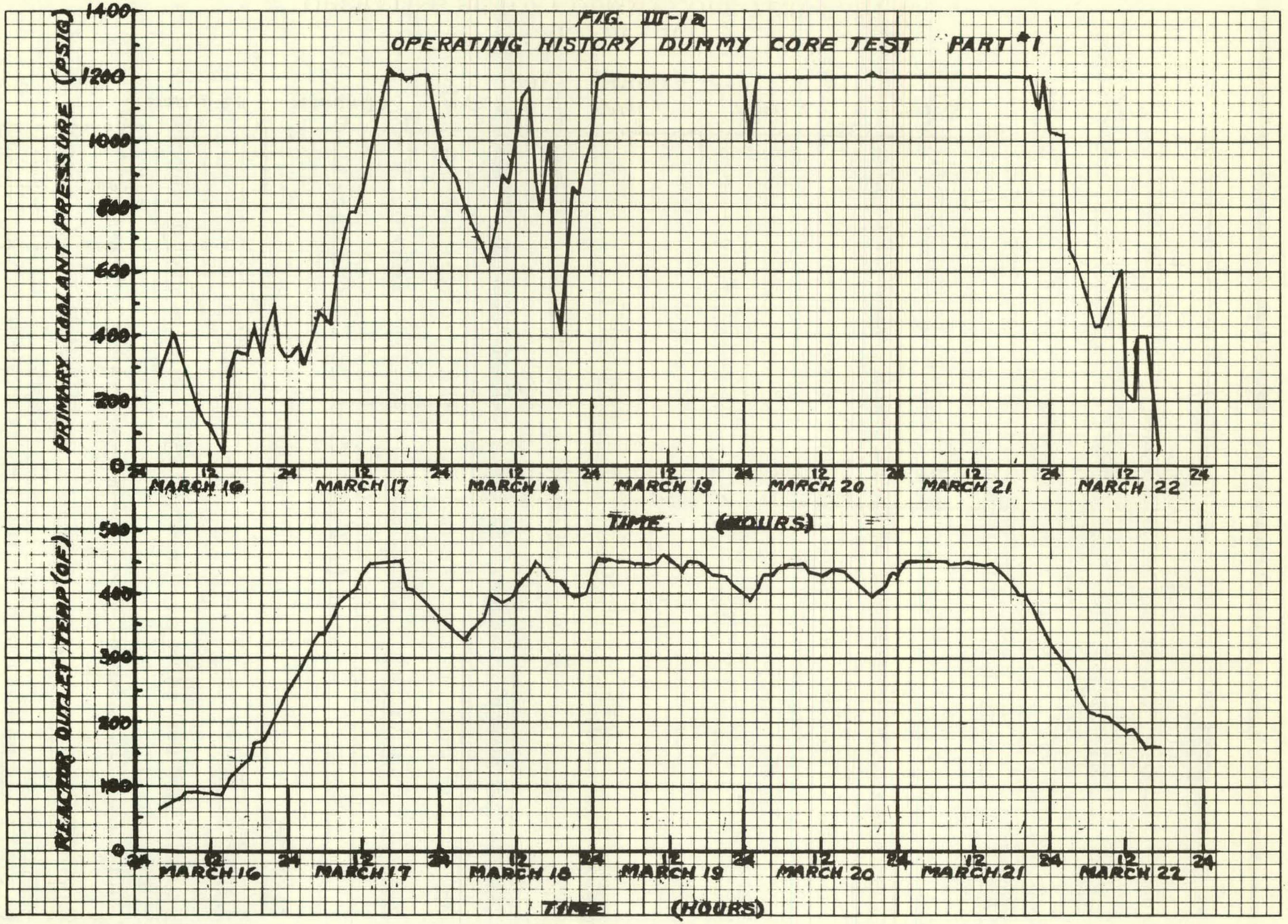


FIG. III - 1a
OPERATING HISTORY DUMMY CORE TEST PART #1

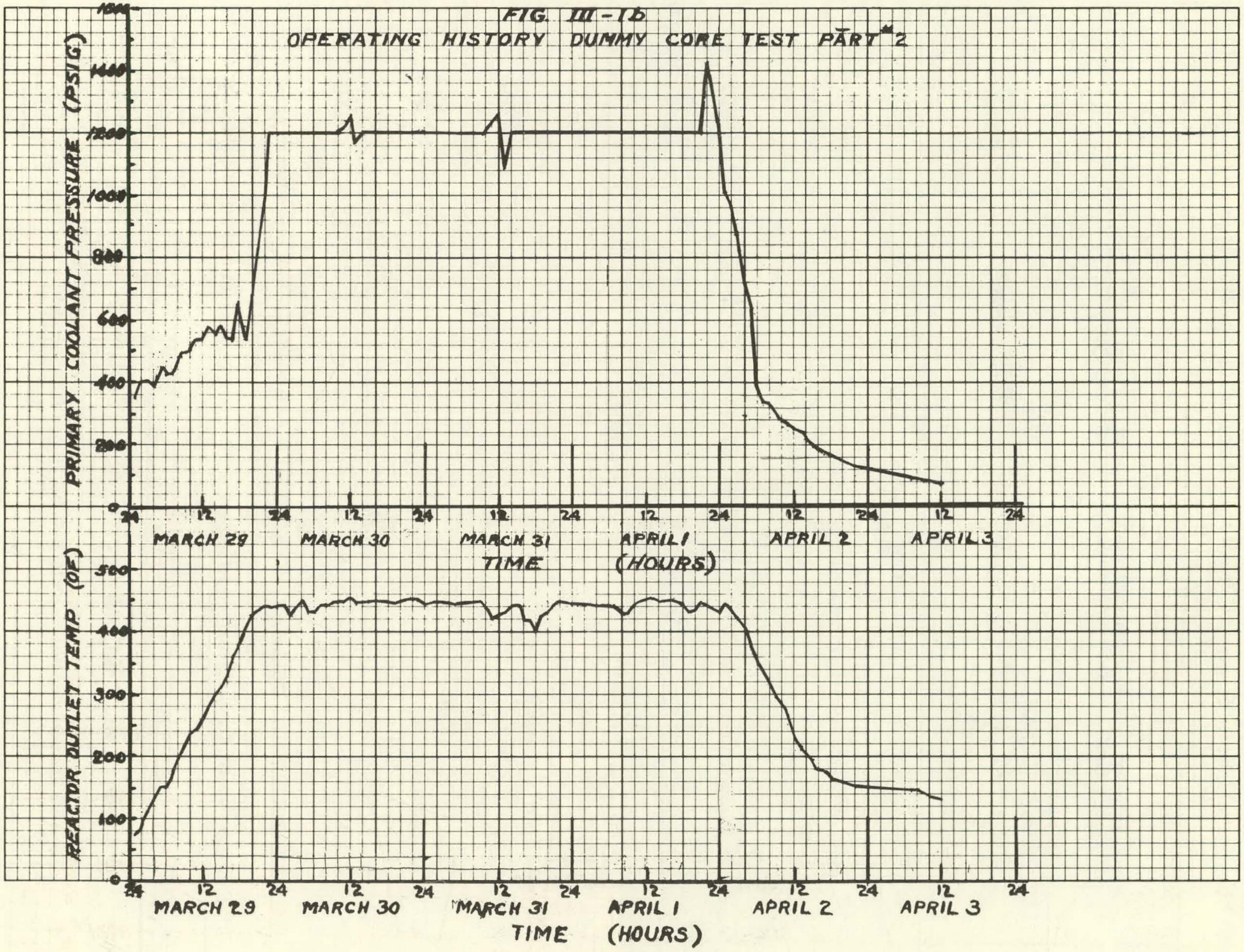


FIG. III - 1b
OPERATING HISTORY DUMMY CORE TEST PART #2

Insomuch as no accurate timing of closing the shield tank was taken, it is assumed that one hour would be sufficient time to perform this operation and is also included in the following table.

TABLE III - 1, DEMONSTRATION OF CORE RELOADING

Remove Spent Fuel Tube Seal		10 Min.
Loosen Vessel Nuts	72 Min.	
Remove & Store Nuts, Install Stud Caps	69 Min.	
Install Sling	34 Min.	
Store Cover	13 Min.	3 Hr. 8 Min.
Remove Control Rod Caps	14 Min.	
Unload & Load Control Rods	275 Min.	
Replace Control Rod Caps	14 Min.	5 Hr. 3 Min.
Open Core Structure Doors	12 Min.	
Unload Stationary Elements	309 Min.	
Load Stationary Elements	184 Min.	
Close Core Cover Doors	50 Min.	9 Hr. 15 Min.
Position Cover	16 Min.	
Remove Sling	32 Min.	
Place Nuts and Remove Stud Caps	56 Min.	
Tighten Nuts	190 Min.	4 Hr. 54 Min.
Replace Spent Fuel Tube Seal		10 Min.
Close Shield Tank		1 Hr.
Total		23 Hr. 40 Min.

3. Demonstration of Replacing Rod Drive Mechanism (T. F. Connolly)

The purpose of this demonstration was to show that a drive assembly could be removed and replaced in less than 3 hours. The crew consisted of four men. Three of these men had never performed this operation before. All rod drives were in place and in operating condition at the start of the demonstration. When the power to the motors, magnetic clutches and instrument brackets was shut off the demonstration commenced. The times for the operations were recorded in the operating log book.

The steps and times are as follows:

TABLE III-2 DEMONSTRATION OF REPLACING ROD DRIVE MECHANISM

I.	Remove Rod Drive			
	A.	Remove instrument bracket cover. Uncouple amphenol connections to motor, instrument bracket and magnetic clutch. Remove universal shaft. Remove motor.	4 Min.	
	B.	Remove magnetic clutch assembly	3 Min.	
	C.	Remove instrument bracket Close valve on seal feed line Seat ball valve Close drain line valve Remove drain line Remove seal feed line Remove seal assembly	14 Min.	
		Time to remove rod drive		21 Min.
II.	Install Rod Drive			
	D.	Install new flexitallic gasket Install water seal Install drain and feed lines Check for leakage Install instrument bracket (check position of nut in relation to micro switches)	14 Min.	
	E.	Install magnetic clutch	4 Min.	
	F.	Install motor Install universal drive shaft Hook up amphenol connectors Install instrument bracket Cover	7 Min.	
		Total installation time		25 Min.
	Total time, removal and installation			46 Min.

The rod drive was then checked out with the control room and found to be in satisfactory condition. The time for seven drives using the above numbers would be 5 hours 21 minutes. This time would be considerably reduced with experienced crews and performing steps A and B at the same time, and E and F at the same time.

4. Maintenance of Primary Water Purity (A. L. Medin)

Results of the various analytical determinations on the primary water during the dummy core test are given in Chapter XII. In summarizing these results it was clearly demonstrated that the oxygen concentration could be controlled to less than .03 ppm and at many times was essentially zero. The chloride concentration after the demineralizer was placed into operation was reduced essentially to zero. Prior to demineralizer operation, chloride concentration was generally below .5 ppm. The overall total solid concentration was also less than the specified 2 ppm. Resistivity of the primary make-up water generally exceeded 2,000,000 ohms whereas the water in the primary loop itself averaged around 500,000 ohms.

5. Control Rod Drive Performance (J. F. Haines)

A considerable amount of control rod drive testing was performed in conjunction with the dummy fuel element run. In the course of this shakedown running somewhat over 3500 total rod scrams were accumulated. At the end of this shakedown running the control rod drives were operating smoothly and consistently and were considered entirely satisfactory for power operation. Rod testing was discontinued as such at this point except for two runs which were made after the fuel had been loaded. The first of these was made before the reactor had run at power and a second check run made after some full power operation had been accumulated.

These runs and similar ones during the dummy testing period will serve as a reference point against which control rod drive performance can be evaulated during core life time.

Most of the runs made during the dummy core test consisted of raising and dropping all rods at once, recording the travel as a function of time of each rod individually. Thus, seven scrams were necessary to obtain one record on each of the seven rods. Each drop, therefore, counted for seven total scram cycles. A few runs were made in which each rod was raised and scrammed individually; the other six remaining down. This type of run was, of course, that used after the reactor core had been loaded with active fuel. Comparison of drop times between the two methods showed essentially identical results, since the variation in drop time between the two types of runs was well within the statistical spread of drop times for either type of run independently.

The average drop time for the first three inches of insertion of all rods for any one run fell within a range of 0.231 to 0.256 seconds drop including instrument and clutch delay time. The reason for selection of this distance will become apparent when the method of measurement is described. The spread of values, after all necessary adjustments had been made, was considerably less, the maximum time being 0.250 seconds in this case. On the first run after loading the reactor before going to power the maximum drop time (for Safety Rod No. 2) was 0.249 and the minimum time (for Shim 2) was 0.223 seconds for the first three inches of travel. Acceleration rates of the rods range from 0.40G to .46G under normal reactor operating conditions.

Drop times of the rods were measured by means of a Minneapolis-Honeywell

Visicorder. This instrument is basically a galvanometer type of oscillograph using a high intensity light source and photographic recording paper which requires only exposure to light to bring out the recorded trace. Although the Visicorder is a multi-channel device, only three channels were used for this test, one to record drop time of the rod being tested, one to show initiation of the scram cycle by recording safety amplifier input, and one to show breaking of the magnetic clutch circuit.

The testing technique consisted of applying the voltage from one phase of the fine position synchro-generator to the Visicorder input. This gave a 400 cycle AC signal, the amplitude of which was a direct indication of control rod position. Since the fine synchro makes one revolution for 3 inches of rod travel, one complete cycle of the signal amplitude represented 3 inches of travel and 1/2 cycle represented 1-1/2 inch.

A representative Visicorder trace is reproduced in Figure III-2. It is seen that the record appears as a modulated AC wave having a variable modulation frequency increasing as the rod accelerates. The constant amplitude section at the beginning of the trace represents the signal voltage with the rod in the full-up position. The point at which the clutch circuit is broken by the safety amplifier is marked on the curve as the point at which the clutch voltage starts to decay. The interval between these events (not shown on the trace) is less than 0.010 sec. 11 cycles after this (0.028 sec.) rod motion can just be detected as a departure of the amplitude envelope from the initial straight line. 51 cycles (0.127 sec.) later, the rod has reached 1-1/2 in. insertion as indicated by passing the initial amplitude after 1/2 envelope cycle. 73 cycles (0.182 sec.) later the synchro has made one complete revolution represented by one full cycle of the amplitude

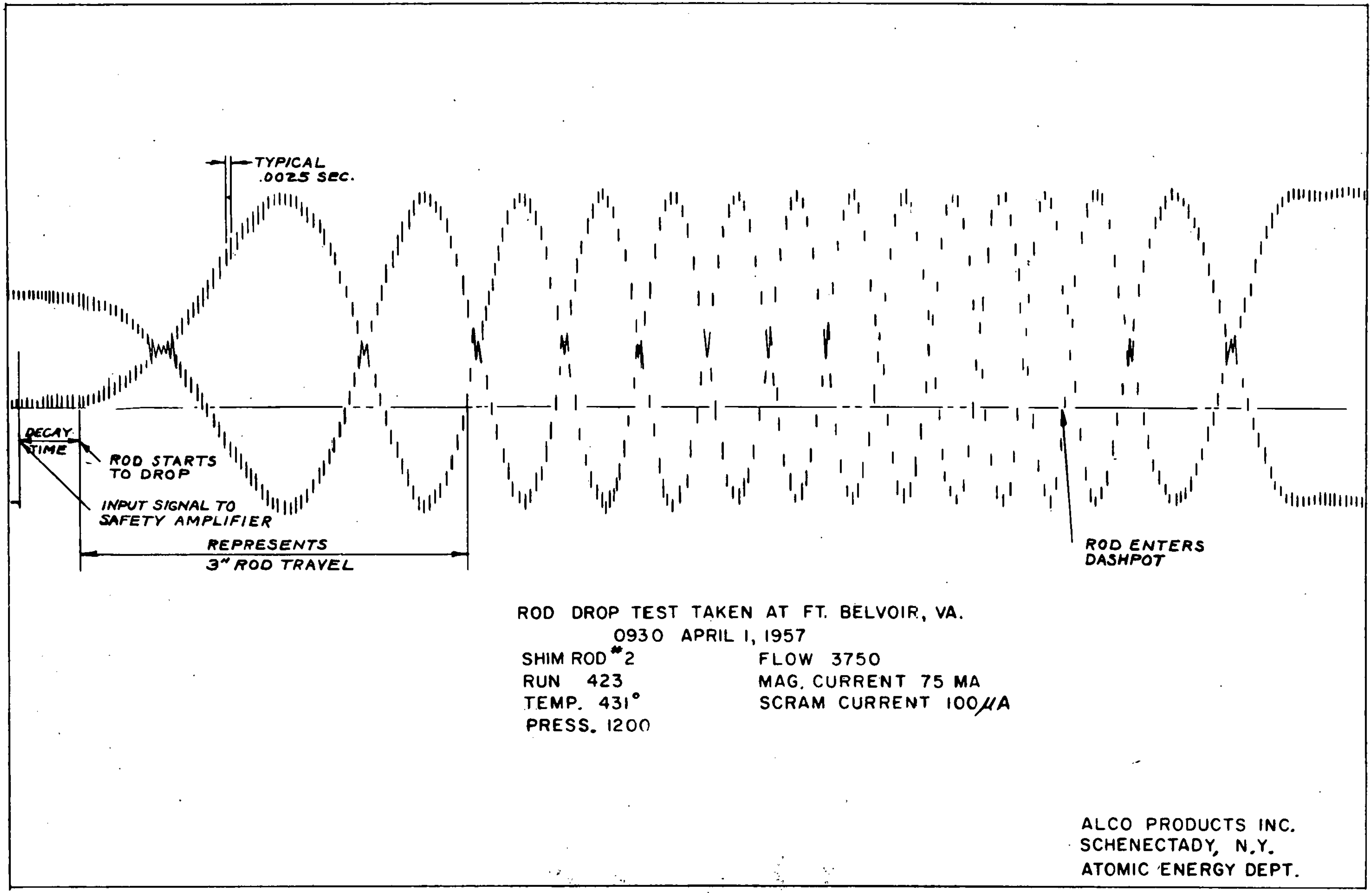


FIG. III - 2 TYPICAL TRACE OF ROD DROP

envelope, indicating rod insertion of 3 in., and the drop time including decay time, is 84 cycles or 0.210 sec.

It will be noted that precision of determining the instant at which rod motion starts is not very high. This is partly responsible for the variation in clutch decay time and accelerations noted during the test. Total drop times showed somewhat better consistency since the end points were more precisely defined.

In order to prepare a more comprehensive control rod drive performance report[9], Visicorder records were reduced to the form of curves of rod insertion versus time. A family of these curves for one representative series of drops is reproduced in Figure III-3 for reference.

6. Inspection of Dummy Fuel Loading Used During the Non-Critical Test Run (E. C. Edgar)

The dummy fuel loading used during the non-critical test was inspected before and after the test run in accordance with Appendix D, (Test Requirements), Contract No. AT(11-1)-318 and procedure set up in letter of October 16, 1956 from E. C. Edgar to J. K. Leslie.

The braze metal joints were very poor but appeared adequate for the few days exposure in the reactor. The control rod fuel elements had not been thoroughly cleaned and a stripe of braze metal stop-off remained along each braze joint. This stop-off is mainly composed of TiO_2, in lacquer carrier and during the brazing operation the lacquer sublimes leaving a coating of TiO_2 (Rutile), which is practically inert chemically so that no effect to the primary system water would be expected or was found.

Several (7) fuel elements were picked at random and inspected closely for dimensions. These same elements were re-measured after the test and it was

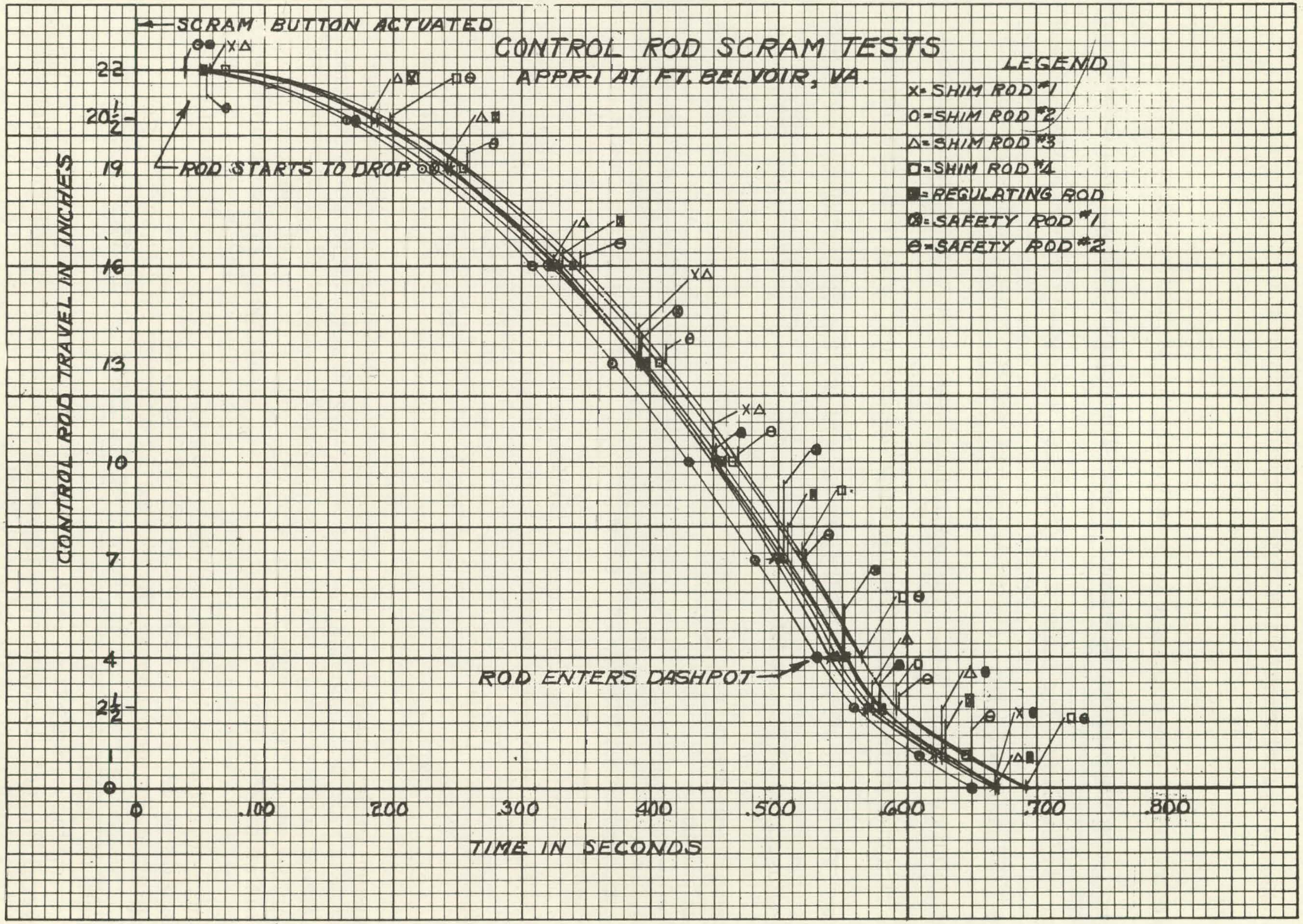


FIG. III - 3

CONTROL ROD SCRAM TESTS

determined that the elements were dimensionally stable at the conclusion of the test.

Specimens for the embrittlement test and microstructure study were removed from control rod fuel element No. 8. The embrittlement specimens of fuel plates before and after the non-critical test were given a 270^{o} cold bend test around a 1/8" mandrel. No embrittlement was noted. These bend test results are illustrated by Figure III-4. The before test specimen was prepared from sections of a fuel plate that had gone through the brazing cycle with one of the elements. The microstructure of the braze joint joining the fuel plate to the side plate is illustrated in Figure III-5. No corrosion of either side plate, fuel plate or braze joint can be seen.

No pitting or gross deposits were observed on the outside or inside fuel plates. A slight overall layer of a brownish crud was observed. This crud on analysis of an ignited specimen showed the following:

Fe_2O_3	63.9%	SiO_2	0.7%
Cr_2O_3	20.6%	Mn_3O_4	4.2%
NiO_2	8.5%		

This is a typical analysis of crud occurring in a stainless steel system. The stainless steel had changed from a silvery gray to a light brownish color, as is expected in a water environment at a temperature of $450^{o}F$.

Details of this test are shown in APAE Memo 99[10].

7. <u>Corrosion Analyses During Dummy Core Tests</u> (A. L. Medin)

During the dummy core tests, chemical analyses were performed to ascertain the corrosion rates of the primary system components. These results,

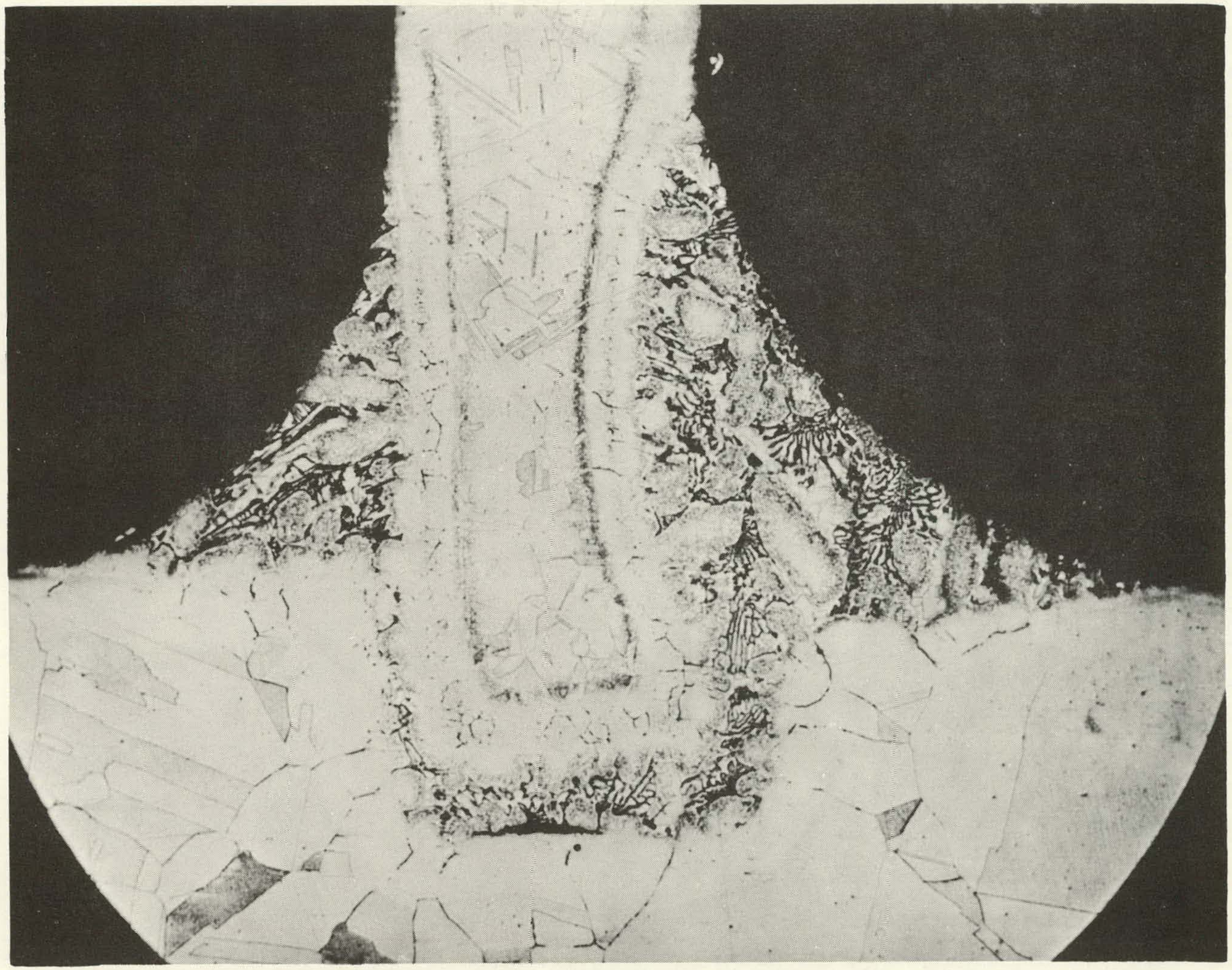


FIG. III - 4 COLD BEND TEST SPECIMENS OF DUMMY FUEL PLATES AFTER BENDING 270°. 3X. REDUCED 3/4 X FOR REPRODUCTION

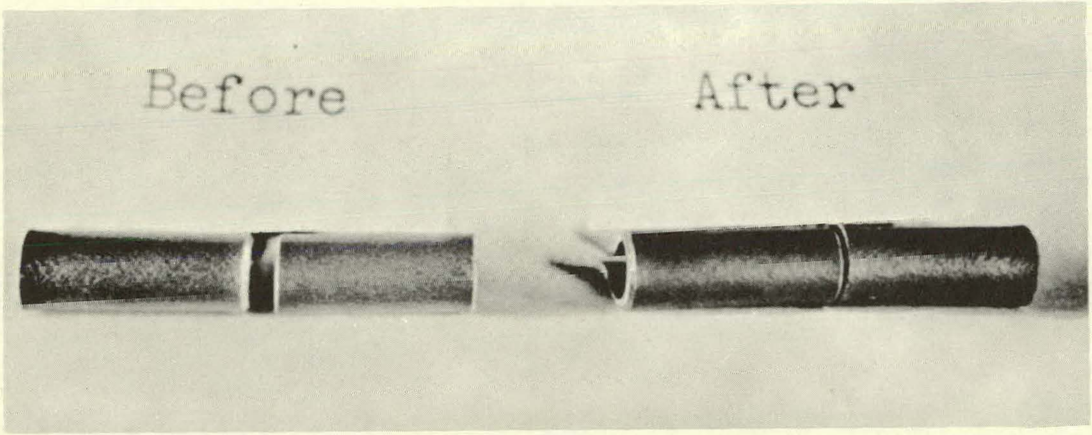


FIG. III-5 BRAZE JOINT - TOP FUEL PLATE TO SIDE PLATE AFTER NON-CRITICAL TEST, ETCHED ELECTROLYTICALLY IN OXALIC ACID. 75X. REDUCED 3/4 X FOR REPRODUCTION indicated that the amount of corrosion was not significant from the standpoint of corrosion products entering the primary system water. The quantity of corrosion products adhering to the metal surfaces could not be determined at the APPR-1. Table III-3 shows the results of the water analysis:

TABLE III-3 CORROSION PRODUCT ANALYSES

Date	Time	Iron ppm	Chromium ppm	Nickel ppm	Cobalt ppm
3/19/57	0030	0.00	0.00	0.0	0.0
3/19/57	1900	0.01	0.00	0.0	0.0
3/20/57	2015	0.00		-	-
3/21/57	0300	0.003		-	-
3/21/57	1340	0.03	0.0	-	-
3/29/57	1730	0.00	0.002	0.06	-
3/29/57	2345	0.008	0.00	0.006	-
3/30/57	1625	0.00	0.00	0.012	.0015
3/30/57	2345	0.004	0.00	-	-

CHAPTER IV - FUEL LOADING

(R. C. DeYoung)

1. General Discussion (J. L. Meem)

Upon completion of the dummy core test and removal of the dummy fuel elements, the reactor was ready for loading with uranium bearing fuel. A cross section of the reactor is shown in Figure IV-1. Note that the BF_3 and fission counters were moved from their normal position in the outer shield to within the reactor vessel for the fuel loading. The temporary position of these counters is shown in the figure.

The upper grid assembly is made up of 4 doors. When raising a door to insert fuel elements in that quadrant next to one of the counters, the counter had to be moved and then repositioned after the door was lowered. Accordingly, a loading procedure was planned with the minimum amount of opening and closing of the grid doors.

Four of the fuel elements were marked specially at ORNL before shipment to the site. These markings are for the purpose of aiding in the sectioning of the fuel elements for examination after longtime irradiation. These specially marked elements are numbered 79, 80, 81, and 82 and were inserted in core positions 43, 34, 45 and 47, respectively. The final loading of the reactor is shown in Figure IV-2. The fuel element number and the number of the position in which it was placed is shown, as well as the sequence in which the elements were added. As noted by the shaded area, the reactor went critical with 17 fuel elements.

2. Initial Criticality

The seven control rod fuel elements comprised the first fuel addition to the

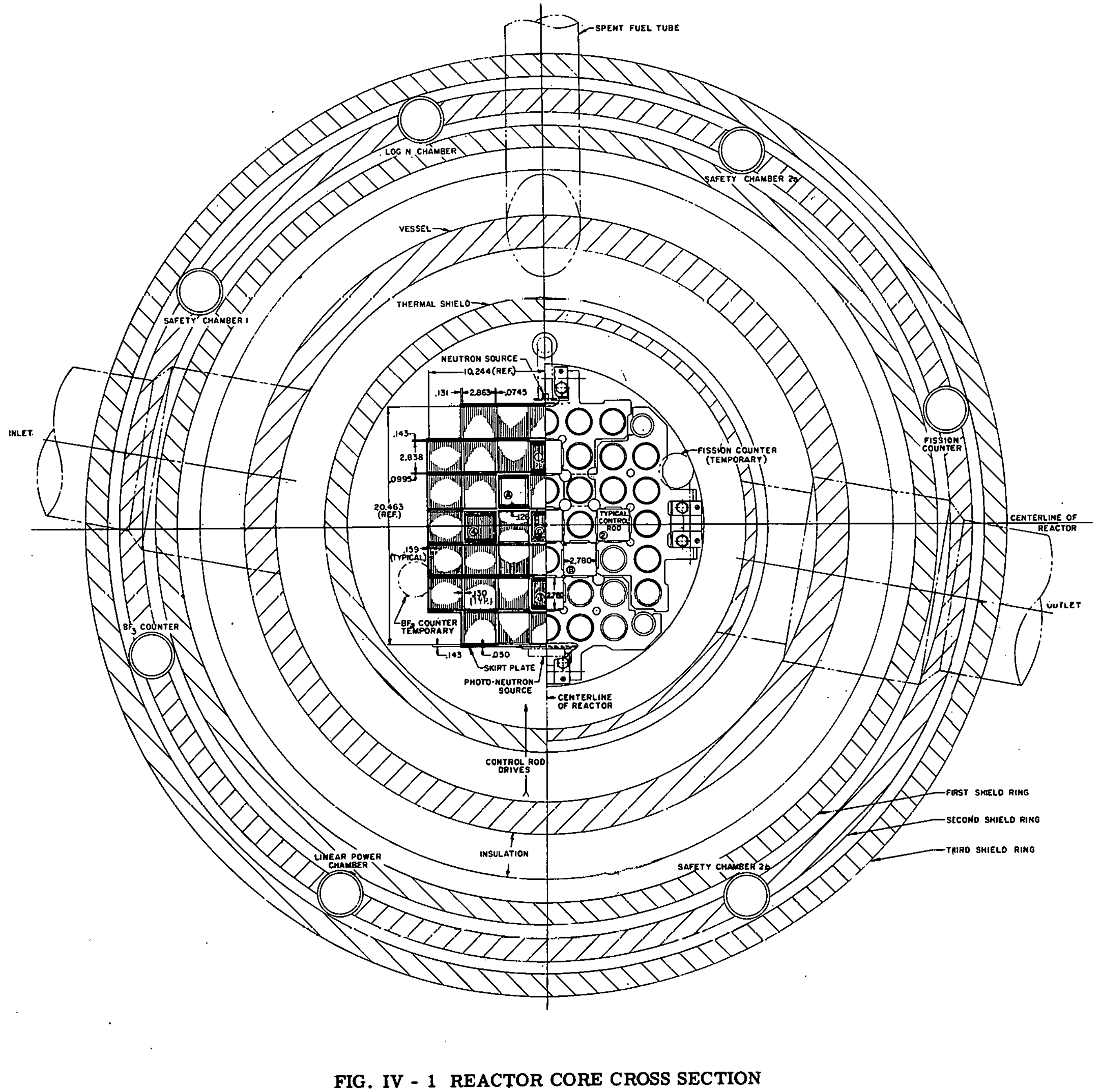
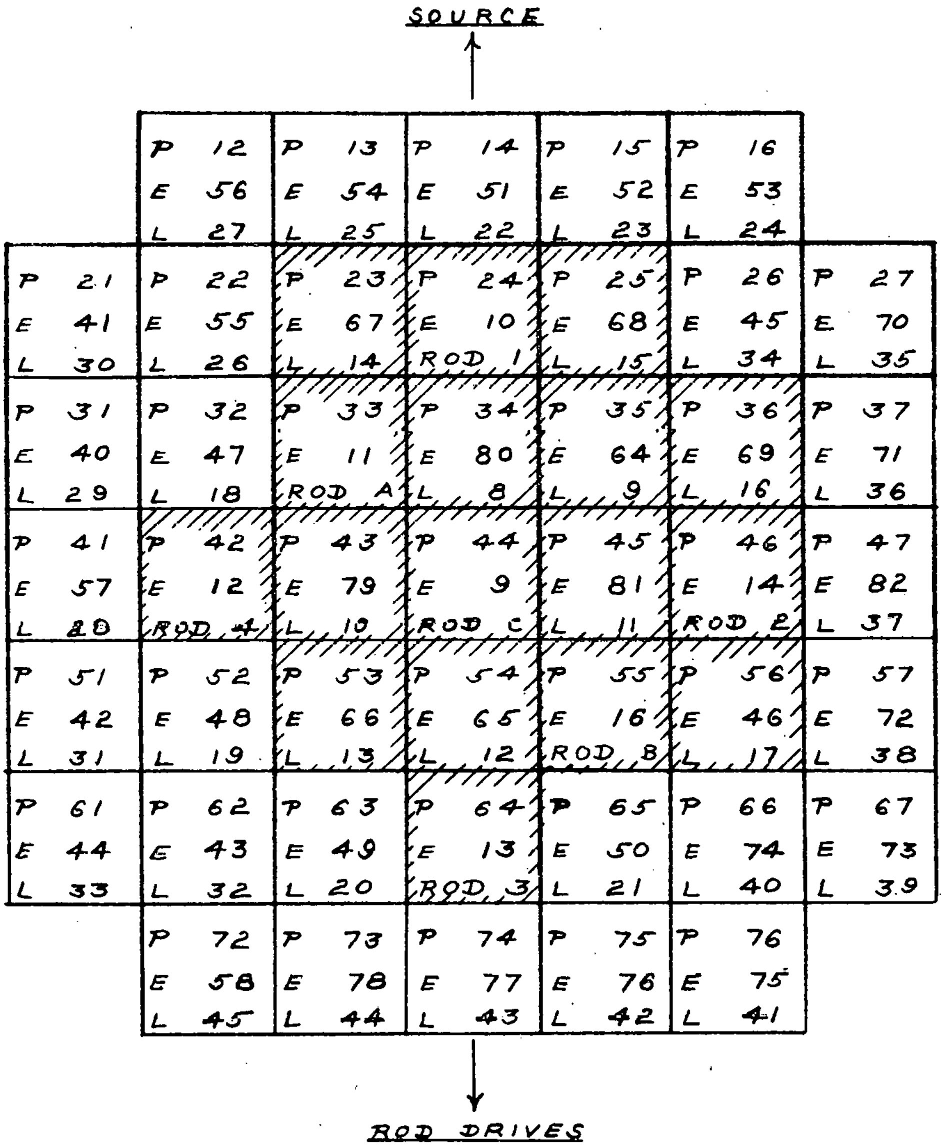


FIG. IV - 1 REACTOR CORE CROSS SECTION

SHADED AREA INDICATES INITIAL CRITICAL LOADING



EXPLANATION

P 12 E 56 L 27	ELEMENT 56, THE 27 th ELEMENT LOADED, WAS LOADED INTO POSITION 12

THE SEVEN CONTROL RODS WERE LOADED FIRST IN THE SEQUENCE : C, 1, 2, 3, 4, A, B

FIGURE IV - 2

LOADING SEQUENCE

system. The stationary fuel elements were then added to the system in groups of two or three.

Criticality was attained with seventeen elements distributed as shown in Figure IV-2. The control rod positions for this critical loading are presented in Table IV-1.

To minimize misinterpretation of terminology several terms are defined at this point:

Active Core - That region defined by the upper and lower average limits of the U^{235} distributions in the stationary fuel elements.

Bank Position - The average withdrawal measured from the bottom of the active core of five or seven control rods. Five rod bank position refers to the position of control rods 1, 2, C, 3 and 4 with rods A and B fully withdrawn. Seven rod bank position refers to the position of all seven control rods.

Rod Position - A rod's position is defined as the distance withdrawn from its deepest insertion, i.e., the nominal alignment of the bottom of the active core with the top limit of U^{235} distribution in the control rod fuel element.

Rod Withdrawal - Refers to the withdrawal of the boron section of the control rod from the active core and the consequent simultaneous insertion of fuel.

TABLE IV - 1

ROD WITHDRAWAL FOR INITIAL CRITICALITY

Case	Inches Withdrawn of Indicated Rods						
	A	B	C	1	2	3	4
1	22.00	22.00	16.50	22.00	22.00	22.00	22.00
2	18.50	18.50	17.92	17.83	18.00	17.95	17.88
3	22.00	22.00	17.83	17.78	17.84	17.89	17.83

To predict the number of elements with which criticality would be attained a plot of the ratio of a reference count rate to that for subsequent larger configurations versus the kilograms of U-235 in the system was made. The usual reference count rate is that given by the source with no fuel in the system. Due to the low count rates by the start-up chambers this source count rate was deemed unreliable. The reference count rate was taken as that obtained following the addition of the seven control rod elements and two stationary fuel elements to the system. This and subsequent source multiplication measurements were taken with all seven control rods completely withdrawn. The method correctly predicted the number of elements required for criticality.

Figure IV-3 presents the data obtained during the approach to criticality. The data is normalized to that obtained during the Zero Power Experiment performed at the Alco Criticality Facility[7].

3. <u>Complete Loading</u>

The position of the five rod and seven rod banks was noted as additional fuel elements were added to the system. Figure IV-2 presents the loading sequence. Figure IV-4 presents the configuration and critical control rod bank positions following each fuel addition. Included is comparative data as determined from the Zero Power Experiments [7].

Figure IV-5 graphically presents critical bank positions versus number of elements in core.

With the core fully loaded the critical position of the five rod bank was 3.70 inches withdrawn. The critical position of the seven rod bank was 5.50 inches withdrawn.

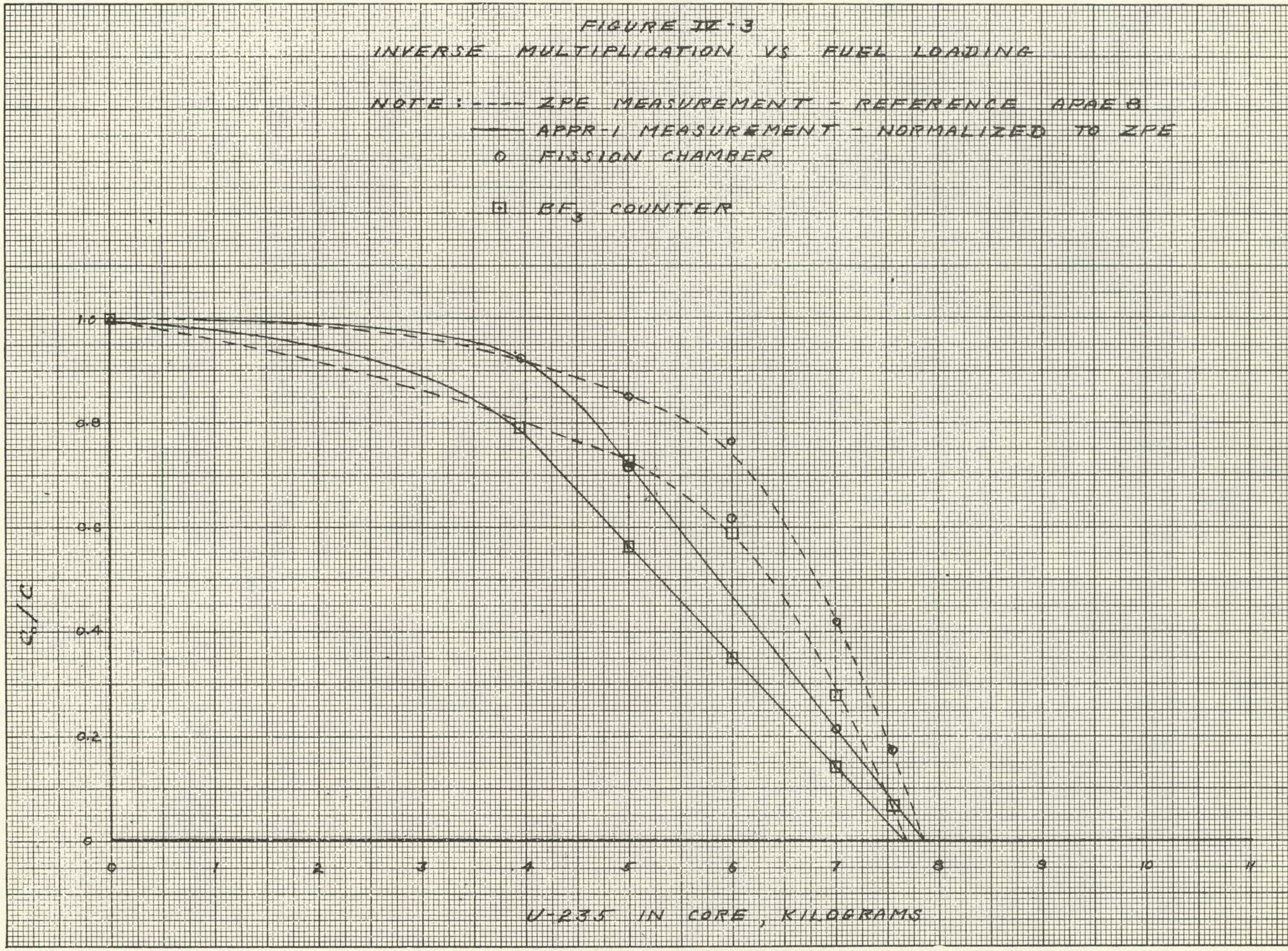


FIGURE IV - 3
INVERSE MULTIPLICATION VS FUEL LOADING

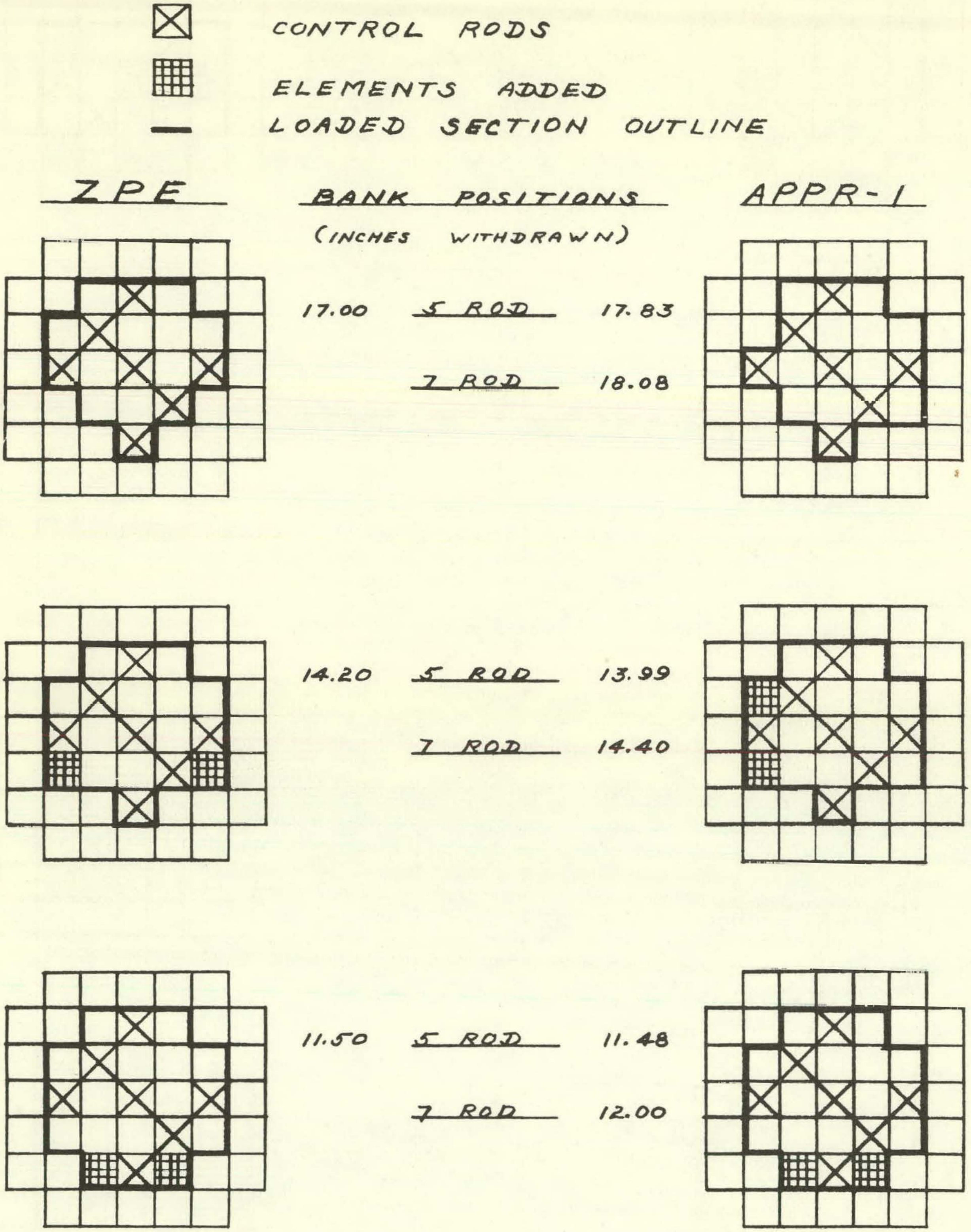


FIGURE IV - 4

CRITICAL CONFIGURATIONS

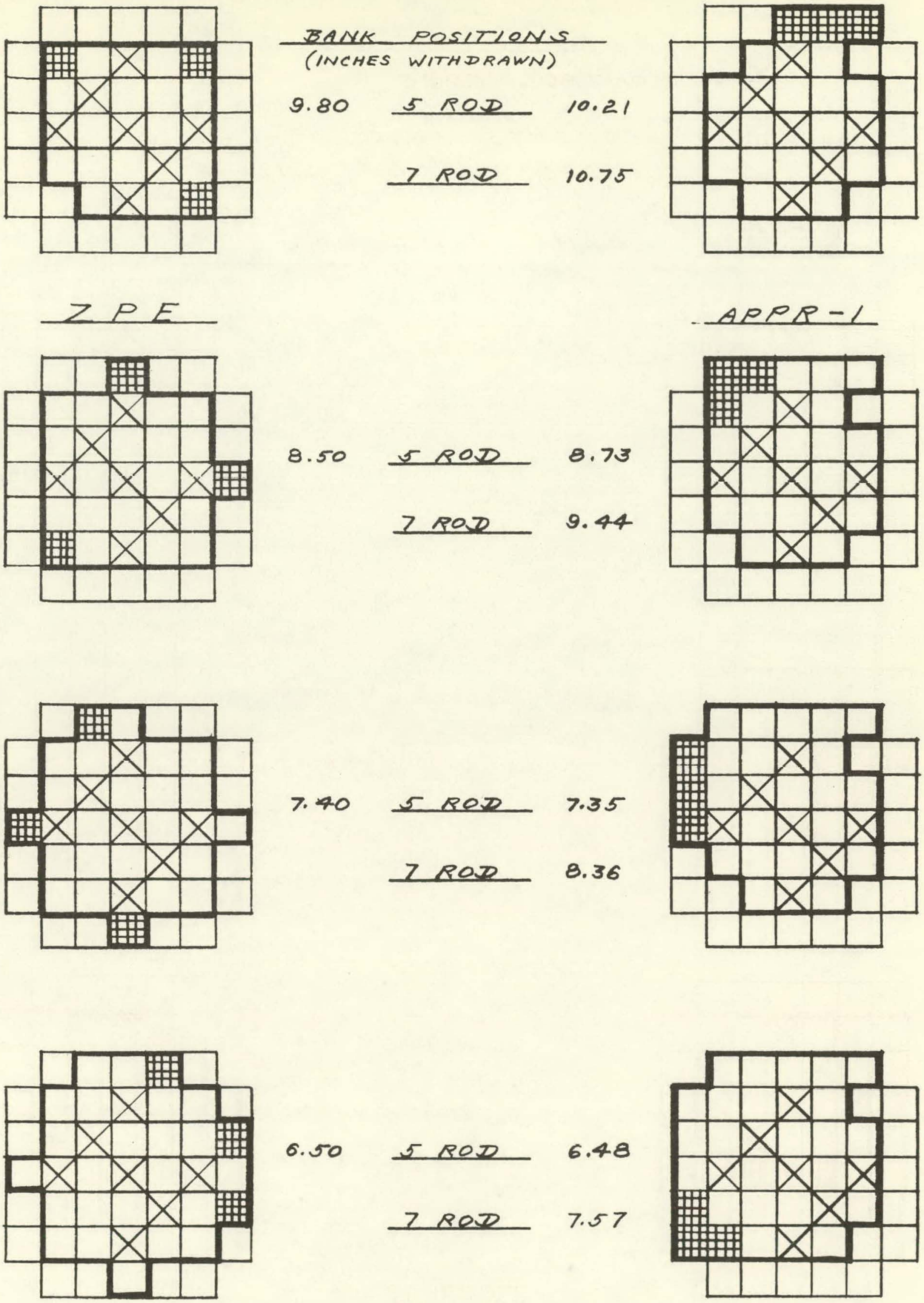


FIGURE IV - 4 (Continued)
CRITICAL CONFIGURATIONS

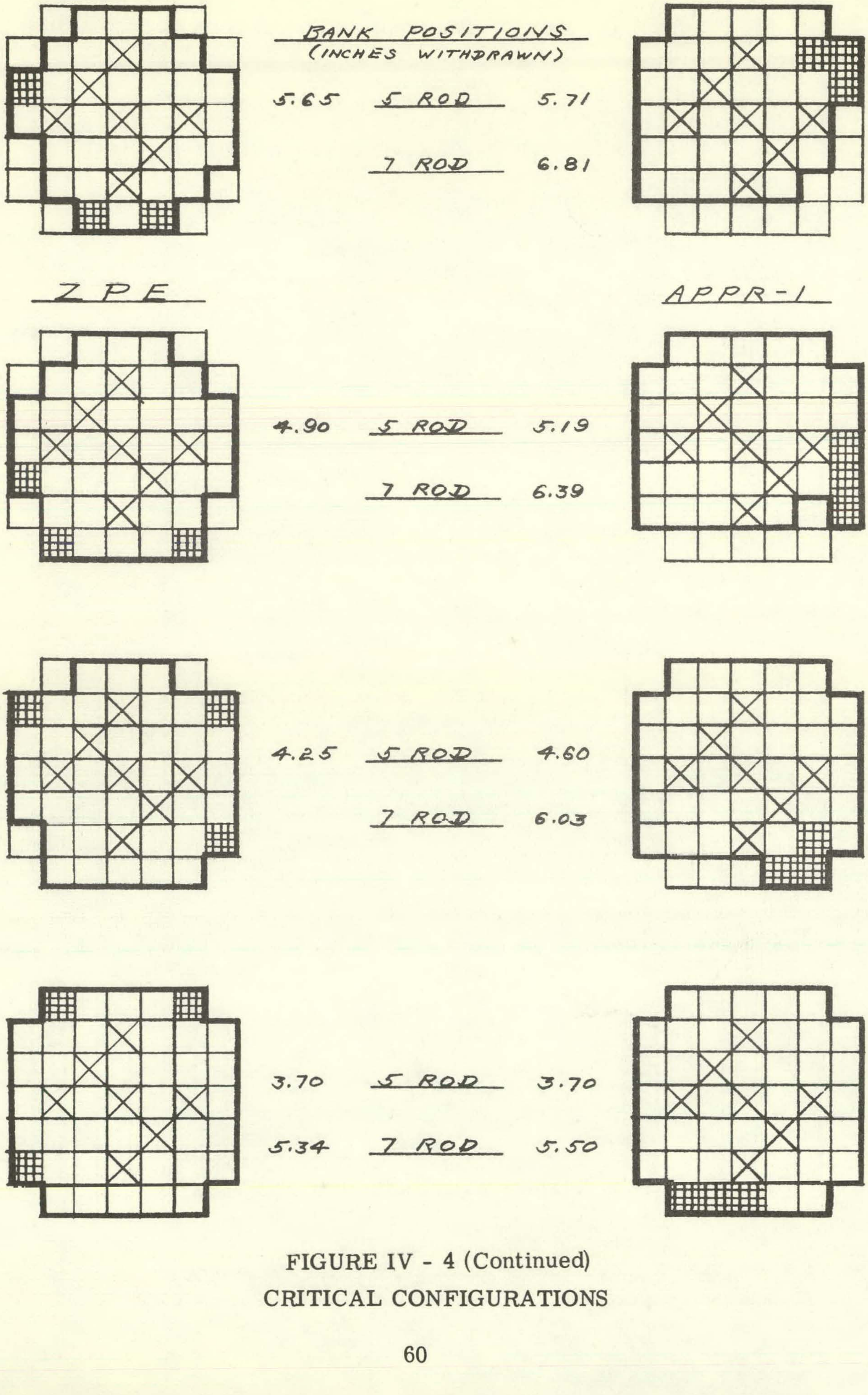


FIGURE IV - 4 (Continued)
CRITICAL CONFIGURATIONS

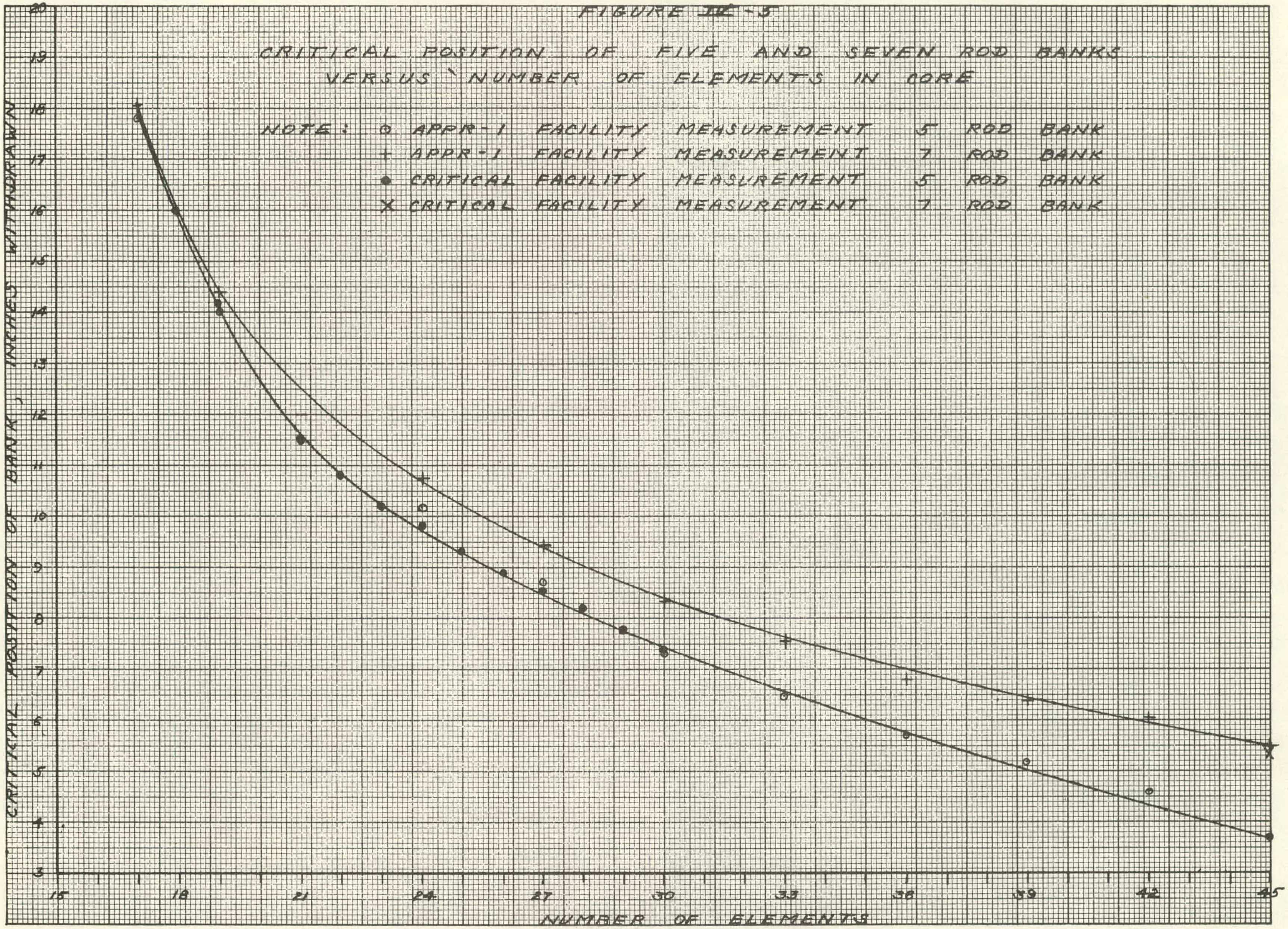


FIGURE IV - 5
CRITICAL POSITION OF FIVE AND SEVEN ROD BANKS
VERSUS NUMBER OF ELEMENTS IN CORE

CHAPTER V - TEMPERATURE AND PRESSURE COEFFICIENTS

(R. C. DeYoung)

1. General

The temperature and pressure coefficients of reactivity were calculated using data obtained during the initial rise to operating temperature. The primary system was heated by introducing steam into the secondary side of the steam generator as described in the chapter on Dummy Core Tests. The rate of temperature rise was limited to less than 30°F per hour. Prior to heating, the pressure coefficient was measured from atmospheric pressure to 600 psi. After the system had been heated to 450°F, the pressure coefficient was then measured from 600 psi to 1200 psi.

The essential data is tabulated in Tables V-1 through V-6. The delayed neutron fraction of 0.0073 used in relating period to worth is the same as used in interpretation of the Zero Power Experiments(7). Considerable scatter in the data exists. The greatest error probably occurred in establishing the rod positions, especially for bank measurements. Also, an error existed in determining the period from the log N recorder trace.

2. Calculations

Curves of critical position versus temperature are presented in Figures V-1, V-2, and V-3. A curve of critical position versus pressure at 450°F, is presented in Figure V-4.

Figures V-5 and V-6 present curves of rod worth versus position for Rod C. A similar curve for the five rod bank is not presented since the scatter in the data was too great to warrant an estimated curve of any significance. An estimate of the worth per inch for the five rod bank was obtained by averaging the values determined by period measurements and presented in Table V-3. This procedure gave a value of 1.51 % per inch. In Figures V-5 and V-6, no attempt was made to reproduce the fine structure as presented in Reference 7. However, the error due to curve plotting is believed to be consistent with the errors associated with the basic data.

The temperature coefficient of reactivity as measured with Rod C was calculated using Figures V-1, V-2, V-5, and V-6. Tables V-7, V-8, and V-9 present a summary of the calculations. The temperature coefficient of reactivity as measured with the five rod bank was calculated using Figure V-3 and data from Table V-3. The calculations are summarized in Table V-10.

Figures V-7a and V-7b, based on data contained in Table V-11 presents the critical position of the five rod bank as a function of temperature with Rods A and B withdrawn 20 inches. While this data has not been used in any of the calculations cited in this chapter it is pertinent to the operation of the APPR-1 and is included on this basis.

The pressure coefficient of reactivity was calculated by relating the pressure change to the measured change in the critical position of Rod C. This position change was related to a reactivity change as determined from period measurements. An average rod worth over the interval of interest was assumed to exist. The calculations are summarized in Table V-12.

3. Results

The temperature coefficient of reactivity as a function of temperature is presented in Figure V-8. The value of $-\Delta K / \Delta T$ at 450° is estimated to be 2.3×10^{-4} $(^{\circ}F)^{-1}$. A value of 2.94×10^{-4} $(^{\circ}F)^{-1}$ was predicted from the Zero Power Experiments[11].

The pressure coefficient of reactivity was measured to be in the range of 2 to 4×10^{-6} $(psi)^{-1}$ at 450°F and 1200 psi.

TABLE V - 1

TEMPERATURE COEFFICIENT DATA

(Rods 1, 2, 3 and 4 at 3.72 Inches)

Temperature °F	Pressure psi	Critical Pos. of Rod C. Inches	Final Pos. of Rod C. Inches	Period Seconds	Worth of Rod C %/Inch
93	560	3.72	4.12	48.90	0.30
96	560	3.79	4.20	34.74	0.36
116	650	4.04	4.46	40.16	0.32
121	700	4.12	4.43	52.10	0.37
127	650	4.24	4.63	41.20	0.34
137	600	4.39	4.76	41.20	0.36
145	660	4.56	4.89	38.00	0.42
155	600	4.78	5.20	34.74	0.35
170	690	5.05	5.44	43.42	0.33
182	695	5.30	5.70	36.90	0.36
193	640	5.59	5.98	39.08	0.35
210	700	6.01	6.38	39.06	0.37
216	620	6.11	6.48	43.67	0.35
229	630	6.53	6.85	44.40	0.40
244	600	6.78	7.12	45.66	0.37
245	620	7.01	7.25	74.35	0.37
260	590	7.42	7.70	47.06	0.44
302	600	8.88	7.92	48.90	0.35
341	600	10.98	11.48	39.08	0.28
356	620	12.01	12.50	47.76	0.25
370	610	12.97	13.52	48.85	0.216
377	600	13.61	14.35	38.00	0.189
386	600	14.51	15.31	39.08	0.172
403	620	16.49	17.34	46.68	0.146

TABLE V-2

TEMPERATURE COEFFICIENT DATA

(Rod 2 at 0.23 and Rod 3 at 0.17 Inches)
(Rod 1 and 4 at 0.26 Inches)

Temperature °F	Pressure Psi	Critical Pos. of Rod C. Inches	Final Pos. of Rod C. Inches	Period Seconds	Worth of Rod C. %/Inch
94	580	9.39	9.70	39.10	0.45
97	565	9.42	9.74	40.16	0.42
122	585	9.74	10.08	45.60	0.37
122	580	9.75	10.00	64.00	0.37
132	700	9.90	10.36	29.30	0.36
139	650	10.10	10.43	32.60	0.47
149	600	10.20	10.60	39.10	0.35
160	600	10.50	10.90	35.82	0.36
175	600	10.81	11.28	36.90	0.25
184	620	11.07	11.59	26.05	0.34
196	680	11.41	11.80	43.42	0.33
208	650	11.78	12.18	39.08	0.35
219	680	12.08	12.52	51.02	0.26
248	600	12.88	13.24	45.25	0.35
246	620	12.90	13.29	73.96	0.23
256	620	13.61	14.04	48.10	0.28
284	600	14.81	15.32	51.10	0.23

TABLE V-3

TEMPERATURE COEFFICIENT DATA

(5 Rod Bank Measurements)

Temperature °F	Pressure Psi	Critical Pos. Inches	Final Pos. Inches	Period Seconds	Bank Worth %/Inch
92	610	3.72	3.81	43.40	1.44
94	520	3.73	3.83	40.16	1.33
123	600	3.85	3.94	35.82	1.62
122	625	3.84	3.90	56.50	1.80
135	670	3.90	3.98	48.85	1.48
142	665	3.95	4.00	69.50	1.87
151	620	3.98	4.06	41.25	1.67
167	630	4.08	4.18	33.65	1.51
178	670	4.12	4.25	31.50	1.21
188	640	4.20	4.31	32.56	1.40
203	650	4.31	4.38	52.10	1.64
213	600	4.37	4.46	54.70	1.23
225	600	4.47	4.55	43.44	1.61
238	600	4.57	4.65	40.82	1.64
252	620	4.63	4.72	37.70	1.59
266	580	4.75	4.83	41.80	1.66
307	600	5.10	5.20	47.20	1.22
337	600	5.48	5.56	38.00	1.75
353	620	5.64	5.72	36.90	1.82
373	600	5.85	5.96	32.57	1.40
382	600	5.92	6.05	38.00	1.08

TABLE V-3 (Continued)

Temperature °F	Pressure Psi	Critical Pos. Inches	Final Pos. Inches	Period Seconds	Bank Worth %/Inch
396	620	6.13	6.25	26.05	1.47
408	615	6.28	6.38	41.25	1.34
416	600	6.41	6.50	38.00	1.55
422	600	6.49	6.59	34.74	1.48
444	600	6.84	6.96	31.48	1.31

TABLE V-4

TEMPERATURE COEFFICIENT DATA

(Rods 1, 2, 3 and 4 at 4.50 Inches)

Temperature °F	Pressure Psi	Critical Pos. of Rod C. Inches	Final Pos. of Rod C. Inches	Period Seconds	Worth of Rod C. %/Inch
400	620	12.10	12.70	41.25	0.223
412	600	13.50	14.30	35.82	0.182
420	600	14.30	15.17	43.42	0.148
424	600	14.92	15.90	38.00	0.143

TABLE V-5

PRESSURE COEFFICIENT DATA

(Rod 2 at 0.23 and Rod 4 at 0.26 Inches)
(Rods 1 and 3 at 15.5 Inches)

Temperature °F	Pressure Psi	Critical Pos. of Rod C. Inches	Final Pos. of Rod C. Inches	Period Seconds	Worth of Rod C. %/Inch
450	600	4.36	4.62	64.00	0.381
450	800	4.26	4.60	56.40	0.318
450	1000	4.13	4.47	67.30	0.282
452	1200	3.88	4.35	38.00	0.298
448	600	4.32	4.75	36.90	0.333

TABLE V-6

PRESSURE COEFFICIENT DATA

(Rod 1 at 3.70, Rod 2 at 3.76 and Rods
3 and 4 at 3.74 Inches)

Temperature °F	Pressure Psi	Critical Pos. of Rod C. Inches	Final Pos. of Rod C. Inches	Period Seconds	Worth of Rod C. %/Inch
84	15	3.71	-	-	-
84	210	3.68	3.71		0.325
84	400	3.63	3.68		0.324
84	600	3.58	3.63		0.323

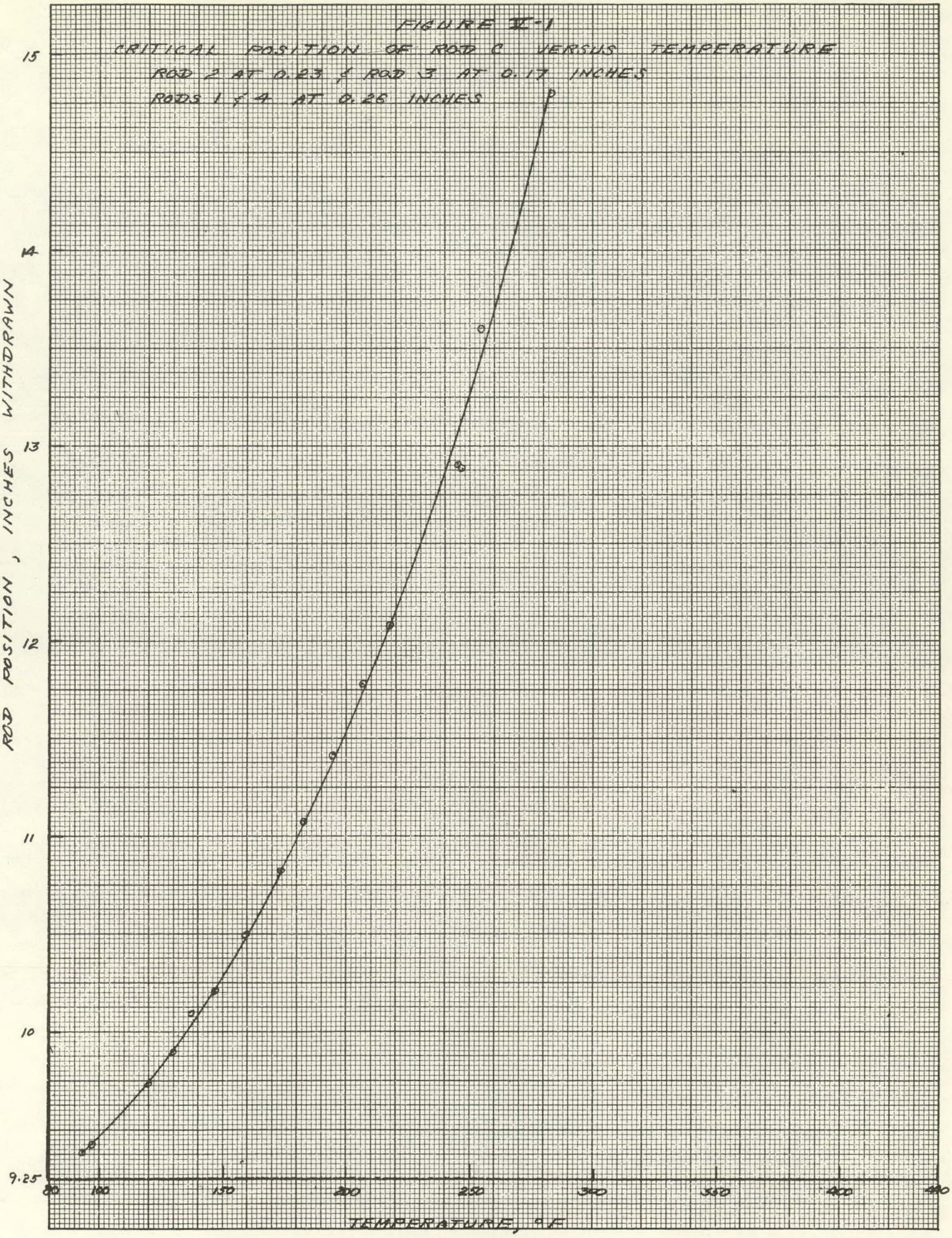


FIGURE V - 1
CRITICAL POSITION OF ROD C VERSUS TEMPERATURE

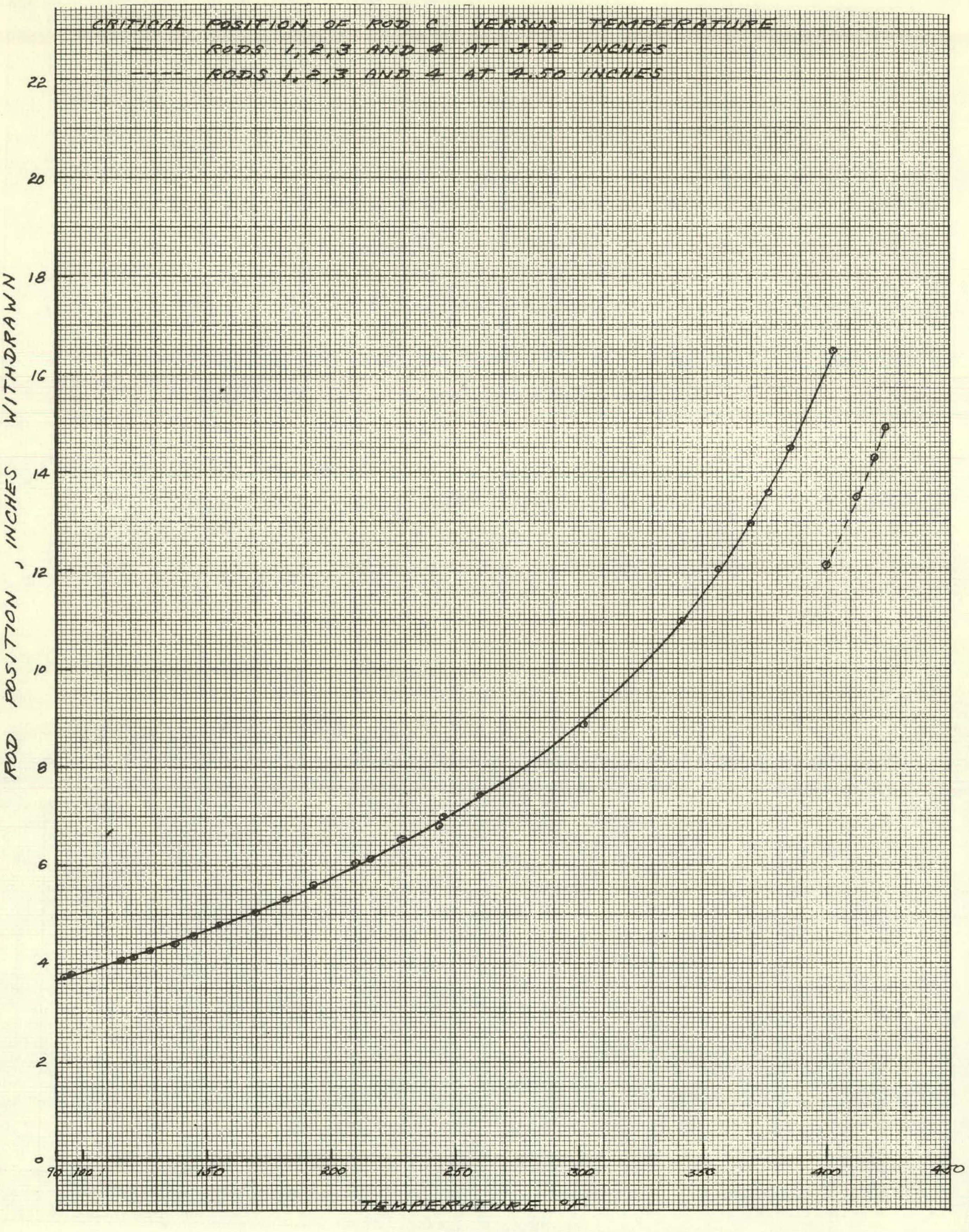


FIGURE V - 2
CRITICAL POSITION OF ROD C VERSUS TEMPERATURE

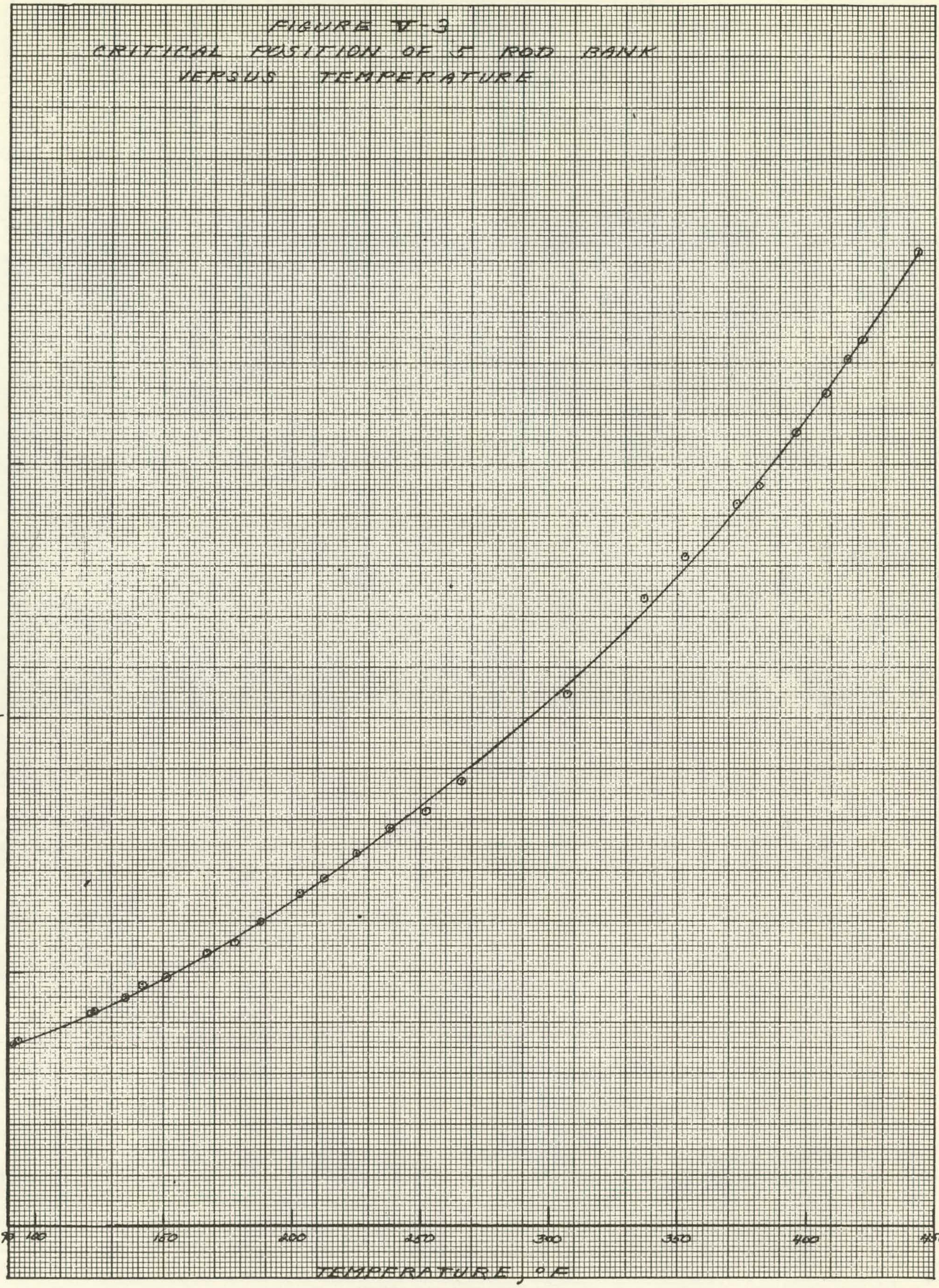


FIGURE V - 3
CRITICAL POSITION OF 5 ROD BANK VERSUS TEMPERATURE

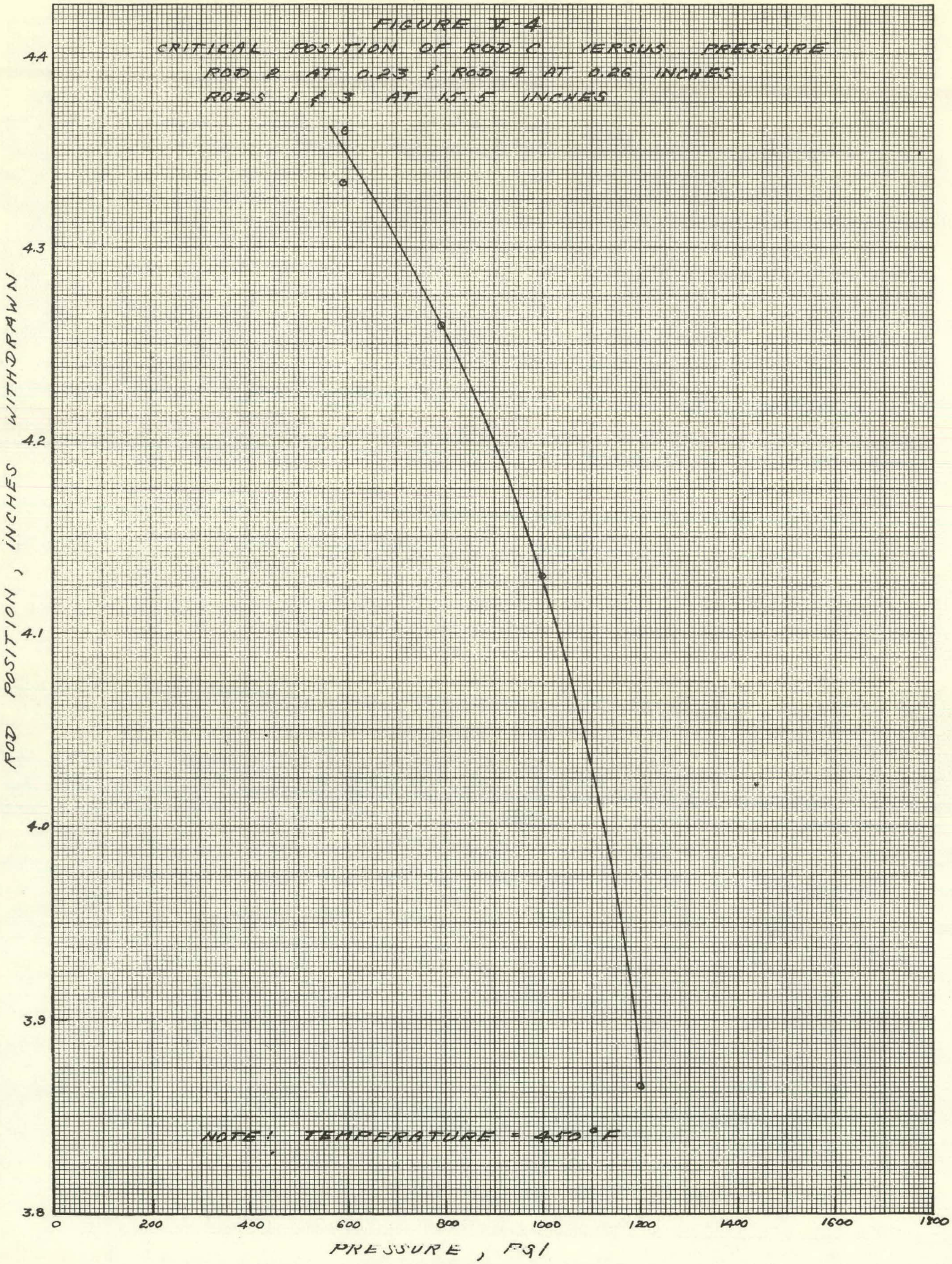


FIGURE V - 4
CRITICAL POSITION OF ROD C VERSUS PRESSURE

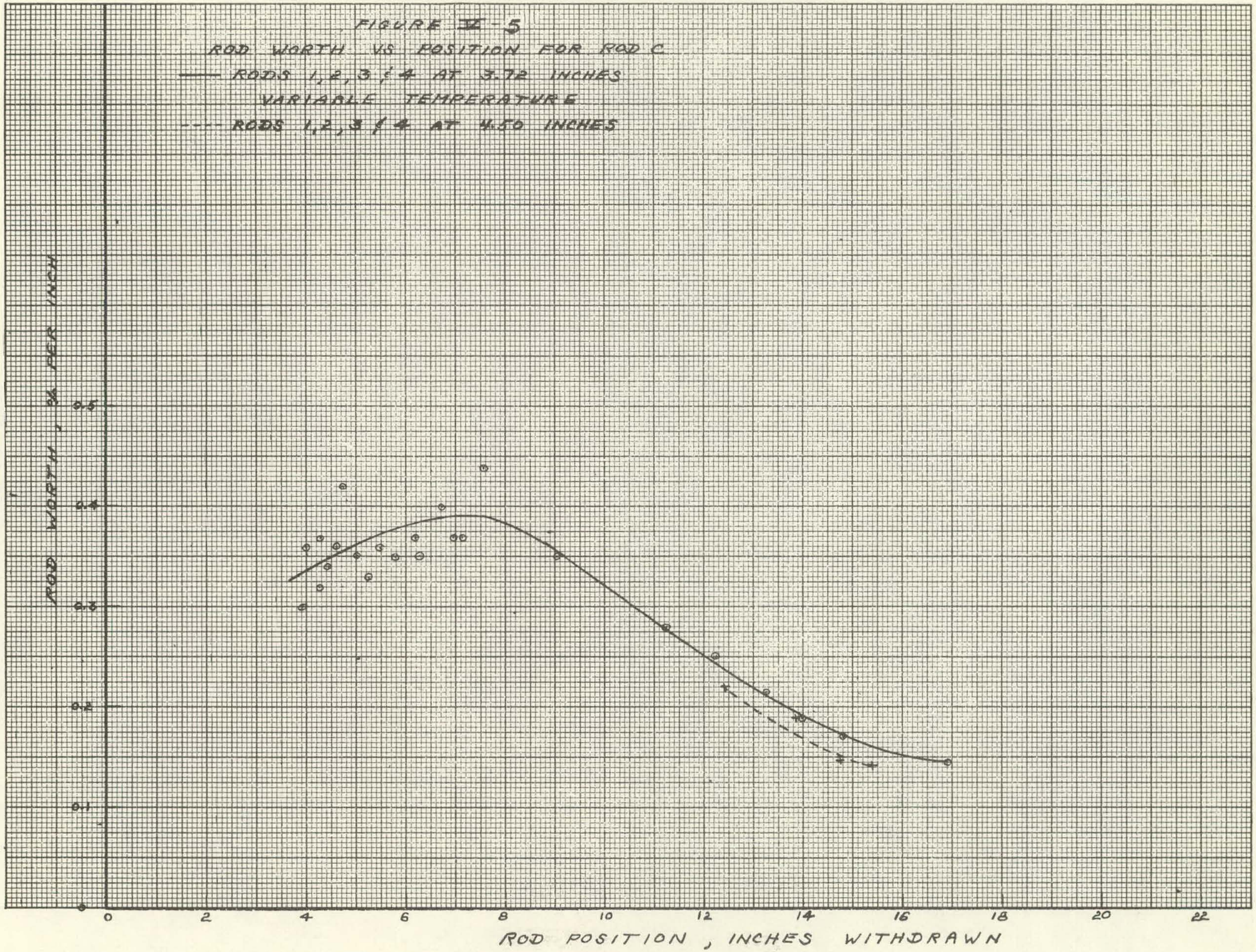


FIGURE V - 5
ROD WORTH VS POSITION FOR ROD C

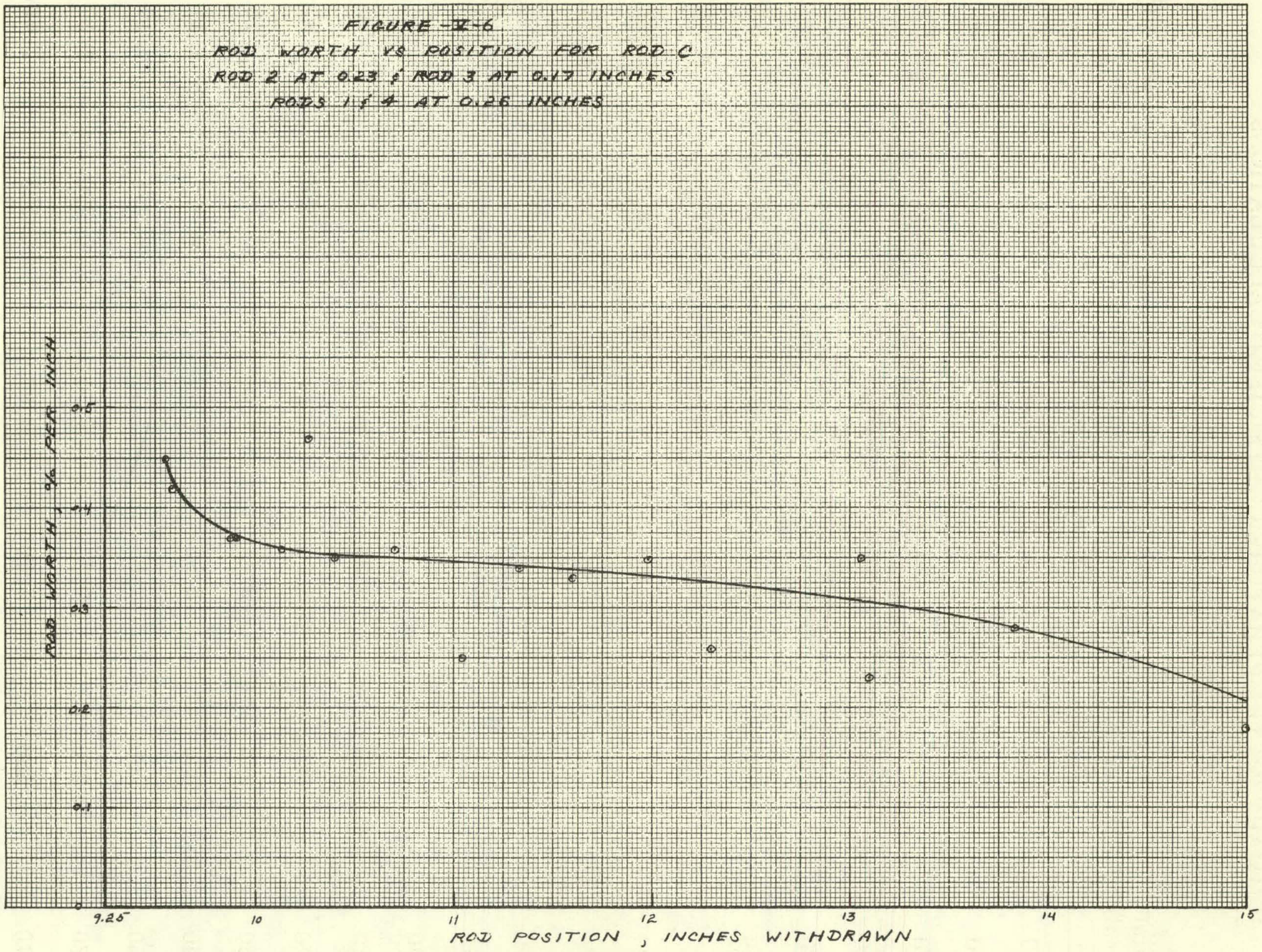


FIGURE V - 6
ROD WORTH VS. POSITION FOR ROD C

TABLE V-7

SUMMARY OF TEMPERATURE COEFFICIENT CALCULATIONS

(Rods 1, 2, 3 and 4 at 3.72 Inches)

Initial Temperature °F	Final Temperature °F	Initial Position Inches	Final Position Inches	Avg. Worth Per Unit Lgth. %/Inch	$-\frac{\Delta K}{\Delta T} \times 10^4$ $(^{o}F)^{-1}$
90	100	3.70	3.80	0.330	0.33
100	110	3.80	3.95	0.332	0.50
110	120	3.95	4.12	0.337	0.57
120	130	4.12	4.30	0.342	0.62
130	140	4.30	4.50	0.347	0.69
140	150	4.50	4.65	0.352	0.53
150	160	4.65	4.85	0.356	0.71
160	170	4.85	5.05	0.362	0.72
170	180	5.05	5.25	0.366	0.73
180	190	5.25	5.48	0.370	0.85
190	200	5.48	5.70	0.375	0.83
200	210	5.70	5.95	0.379	0.95
210	220	5.95	6.20	0.383	0.96
220	230	6.20	6.45	0.386	0.97
230	240	6.45	6.75	0.387	1.16
240	250	6.75	7.05	0.390	1.17
250	260	7.05	7.35	0.391	1.17
260	270	7.35	7.70	0.390	1.37
270	280	7.70	8.08	0.385	1.46

TABLE V-7 (Continued)

Initial Temperature °F	Final Temperature °F	Initial Position Inches	Final Position Inches	Avg. Worth Per Unit Lgth. %/Inch	$-\frac{\Delta K}{\Delta T} \times 10^4$ $(^{\circ}F)^{-1}$
280	290	8.08	8.48	0.377	1.51
290	300	8.48	8.88	0.367	1.47
300	310	8.88	9.32	0.353	1.55
310	320	9.32	9.85	0.336	1.78
320	330	9.85	10.40	0.317	1.74
330	340	10.40	10.95	0.297	1.63
340	350	10.95	11.55	0.277	1.66
350	360	11.55	12.20	0.255	1.66
360	370	12.20	13.00	0.232	1.86
370	380	13.00	13.90	0.205	1.85
380	390	13.90	14.90	0.182	1.82
390	400	14.90	16.05	0.158	1.82
400	410	16.05	17.40	0.146	2.12

TABLE V-8

SUMMARY OF TEMPERATURE COEFFICIENT CALCULATIONS
(Rod 2 at 0.23 and Rod 3 at 0.17 Inches)
(Rods 1 and 4 at 0.26 Inches)

Initial Temperature ^{o}F	Final Temperature ^{o}F	Initial Position Inches	Final Position Inches	Avg. Worth Per Unit Lgth. %/Inch	$-\frac{\Delta K}{\Delta T} \times 10^4$ $(^{o}F)^{-1}$
100	110	9.47	9.60	0.465	0.60
110	120	9.60	9.74	0.400	0.56
120	130	9.74	9.90	0.380	0.61
130	140	9.90	10.08	0.367	0.66
140	150	10.08	10.28	0.358	0.72
150	160	10.28	10.50	0.353	0.78
160	170	10.50	10.75	0.350	0.88
170	180	10.75	11.00	0.348	0.87
180	190	11.00	11.25	0.345	0.86
190	200	11.25	11.53	0.341	0.95
200	210	11.53	11.80	0.338	0.91
210	220	11.80	12.10	0.333	1.00
220	230	12.10	12.44	0.326	1.11
230	240	12.44	12.79	0.318	1.11
240	250	12.79	13.18	0.310	1.21
250	260	13.18	13.60	0.297	1.25
260	270	13.60	14.05	0.282	1.27
270	280	14.05	14.58	0.261	1.38

TABLE V-9

SUMMARY OF TEMPERATURE COEFFICIENT CALCULATIONS

(Rods 1, 2, 3 and 4 at 4.50 Inches)

Initial Temperature °F	Final Temperature °F	Initial Position Inches	Final Position Inches	Avg. Worth Per Unit Lgth %/Inch	$-\frac{\Delta K}{\Delta T} \times 10^4$ $(^{\circ}F)^{-1}$
400	410	12.10	13.20	0.210	2.31
410	420	13.20	14.40	0.175	2.10

TABLE V-10

SUMMARY OF TEMPERATURE COEFFICIENT CALCULATIONS
(5 Rod Bank Measurements)

Initial Position Inches	Final Position Inches	Reactivity from Period Measurement %	ΔT From Fig. V-3 $^{\circ}F$	Avg. Temp. $^{\circ}F$	$-\frac{\Delta K}{\Delta T} \times 10^4$ $(^{\circ}F)^{-1}$
3.72	3.81	0.129	20	92	0.65
3.73	3.83	0.133	21	94	0.63
3.85	3.94	0.146	19	123	0.77
3.84	3.90	0.108	12	122	0.90
3.90	3.98	0.119	16	135	0.74
3.95	4.00	0.093	10	142	0.93
3.98	4.06	0.134	16	151	0.84
4.08	4.18	0.151	16	167	0.94
4.12	4.25	0.157	22	178	0.71
4.20	4.31	0.154	16	188	0.96
4.31	4.38	0.114	10	203	1.14
4.37	4.46	0.110	12	213	0.92
4.47	4.55	0.129	11	225	1.17
4.57	4.65	0.134	10	238	1.34
4.63	4.72	0.143	11	252	1.30
4.75	4.83	0.132	9	266	1.49
5.10	5.20	0.130	11	307	1.18
5.48	5.56	0.140	7	337	2.00
5.64	5.72	0.145	7	353	2.07
5.85	5.96	0.154	9	373	1.71
5.92	6.05	0.140	11	382	1.27
6.13	6.25	0.176	10	396	1.76
6.28	6.38	0.134	8	408	1.68
6.41	6.50	0.140	7	416	2.00
6.49	6.59	0.148	7	422	2.11

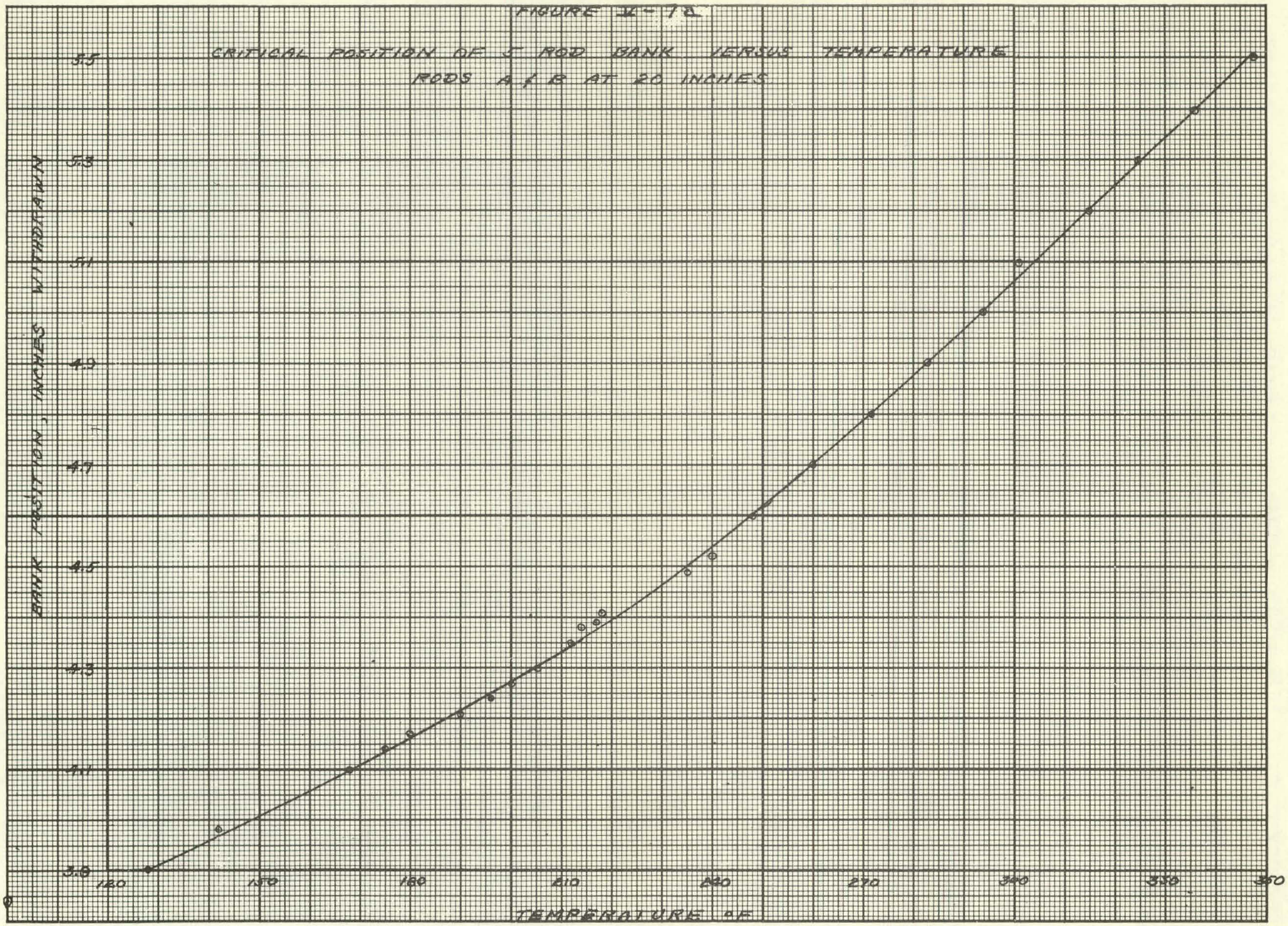


FIG. V - 7a
CRITICAL POSITION OF 5 ROD BANK VERSUS TEMPERATURE

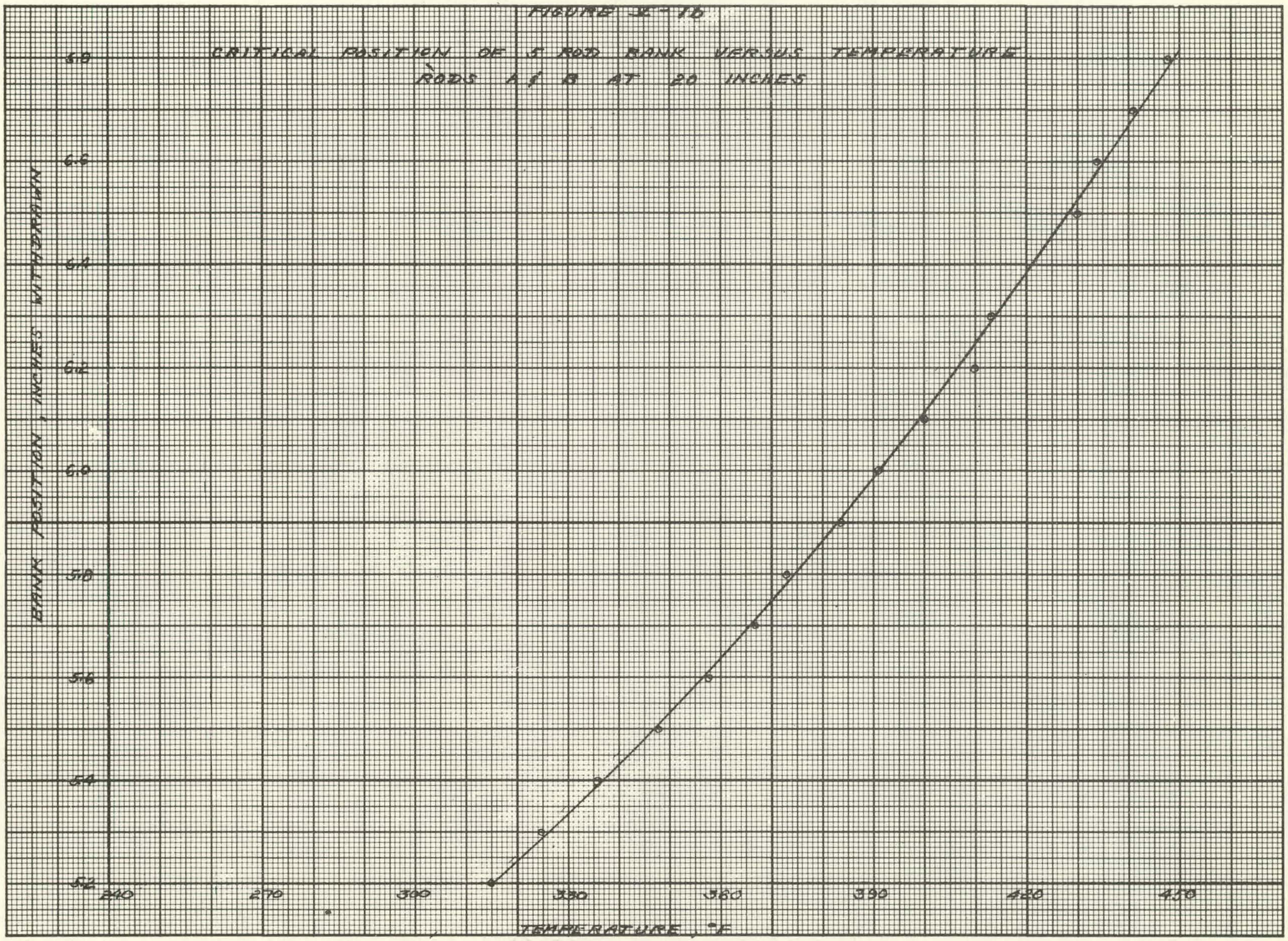


FIG. V - 7b
CRITICAL POSITION OF 5 ROD BANK VERSUS TEMPERATURE

TABLE V-11

FIVE ROD BANK POSITION VERSUS TEMPERATURE
(Rods A and B at 20 Inches)

Reactor Temp. °F	Reactor Pressure Psi	Rod 4 Position Inches	Rod 3 Position Inches	Rod 2 Position Inches	Rod 1 Position Inches	Rod C Position Inches
128	200	3.90	3.90	3.90	3.90	3.90
142	190	3.97	3.98	3.98	3.98	3.99
168	188	4.10	4.10	4.10	4.10	4.11
175	250	4.15	4.16	4.12	4.15	4.14
180	280	4.18	4.18	4.16	4.18	4.17
190	330	4.21	4.21	4.21	4.21	4.21
196	400	4.24	4.24	4.24	4.25	4.25
200	480	4.26	4.26	4.26	4.28	4.28
205	620	4.30	4.31	4.30	4.30	4.31
212	600	4.34	4.34	4.34	4.36	4.36
214	590	4.40	4.37	4.38	4.39	4.38
217	620	4.40	4.40	4.40	4.38	4.40
218	590	4.40	4.42	4.41	4.40	4.40
235	615	4.48	4.49	4.48	4.49	4.49
240	615	4.54	4.54	4.52	4.51	4.51
248	640	4.60	4.60	4.60	4.60	4.60
260	600	4.70	4.70	4.70	4.70	4.70
272	585	4.80	4.80	4.80	4.80	4.80
283	610	4.90	4.90	4.90	4.90	4.90
294	605	5.00	5.00	5.00	5.00	5.00

TABLE V-11 (Continued)

Reactor Temp. °F	Reactor Pressure Psi	Rod 4 Position Inches	Rod 3 Position Inches	Rod 2 Position Inches	Rod 1 Position Inches	Rod C Position Inches
301	610	5.10	5.10	5.10	5.10	5.10
315	605	5.20	5.20	5.20	5.20	5.20
325	595	5.30	5.30	5.30	5.30	5.30
336	650	5.40	5.40	5.40	5.40	5.40
348	1040	5.50	5.50	5.50	5.50	5.50
358	1145	5.60	5.60	5.60	5.60	5.60
367	1210	5.70	5.70	5.70	5.70	5.70
373	1210	5.80	5.80	5.80	5.80	5.80
384	1210	5.90	5.90	5.90	5.90	5.90
391	1210	6.00	6.00	6.00	6.00	6.00
400	1200	6.10	6.10	6.10	6.10	6.10
410	1200	6.20	6.20	6.20	6.20	6.20
413	1215	6.30	6.30	6.30	6.30	6.30
430	1250	6.50	6.50	6.50	6.50	6.50
434	1250	6.60	6.60	6.60	6.60	6.60
441	1240	6.70	6.70	6.70	6.70	6.70
448	1210	6.80	6.80	6.80	6.80	6.80

TABLE V-12

SUMMARY OF PRESSURE COEFFICIENT CALCULATIONS

(Temperature - 450^{o}F, With Flow)

Critical Position Inches	Reactivity From Period Measurement % Inch	Pressure Psi	Average Pressure Psi	$-\frac{\Delta K}{\Delta P} \times 10^{-6}$ (Psi)$^{-1}$
4.36	0.381	600		
			700	1.50
4.26	0.318	800		
			900	1.95
4.13	0.282	1000		
			1100	3.88
3.865	0.298	1200		

Assumed Average = 0.300

(Temperature - 84^{o}F, No Flow)

Critical Position Inches	Reactivity From Period Measurement % Inch	Pressure Psi	Average Pressure Psi	$-\frac{\Delta K}{\Delta P} \times 10^{-6}$ (Psi)$^{-1}$
3.71	0.326	15		
			112	0.50
3.68	0.325	210		
			305	0.85
3.63	0.324	400		
			500	0.81
3.58	0.323	600		

Assumed Average = 0.324

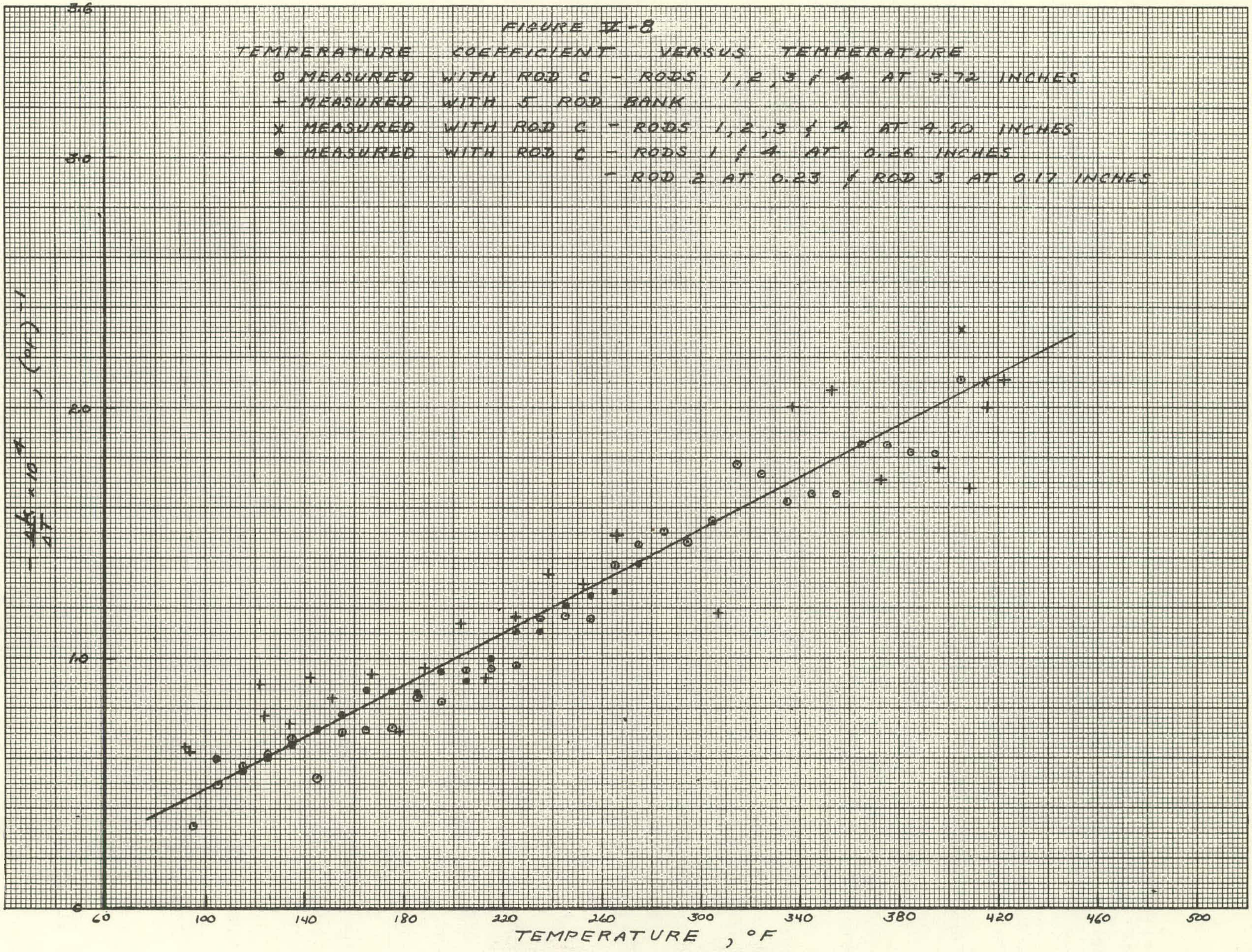


FIGURE V-8
TEMPERATURE COEFFICIENT VERSUS TEMPERATURE

CHAPTER VI - POWER LEVEL CALIBRATION

(F. B. Fairbanks)

1. Purpose

Usually when starting up a power reactor the determination of the exact power level in the low power range is unimportant. When the reactor starts generating appreciable quantities of heat, the exact power can be determined easily from the flow rate of the coolant and its temperature rise, ΔT, across the reactor. However, when shielding measurements are involved, the exact power level at the very low and intermediate power levels becomes important, since this is the range where the bulk of the shielding measurements are made.

Furthermore, considerable zero power data[7] had been collected on the APPR-1 core at the Alco Criticality Facility. To correlate the data from that Facility with the data obtained in the APPR-1 during startup, power level calibrations of the instruments were necessary.

2. Determination from Foil Irradiations

Gold foils have been irradiated in the center slot of the element in position 25, designated in Figure IV-2.

The flux patterns obtained are shown in Figure VI-1, compared to the distribution measured in the Zero Power Experiments[7]. As far as can be determined from the data available, the control rods were in essentially identical configurations in all measurements. The variance in shape probably resulted from difficulty in accurately positioning the foil holder in the APPR-1. Because of the desire to complete low power tests as fast as possible, no flux measurements were made in other core channels.

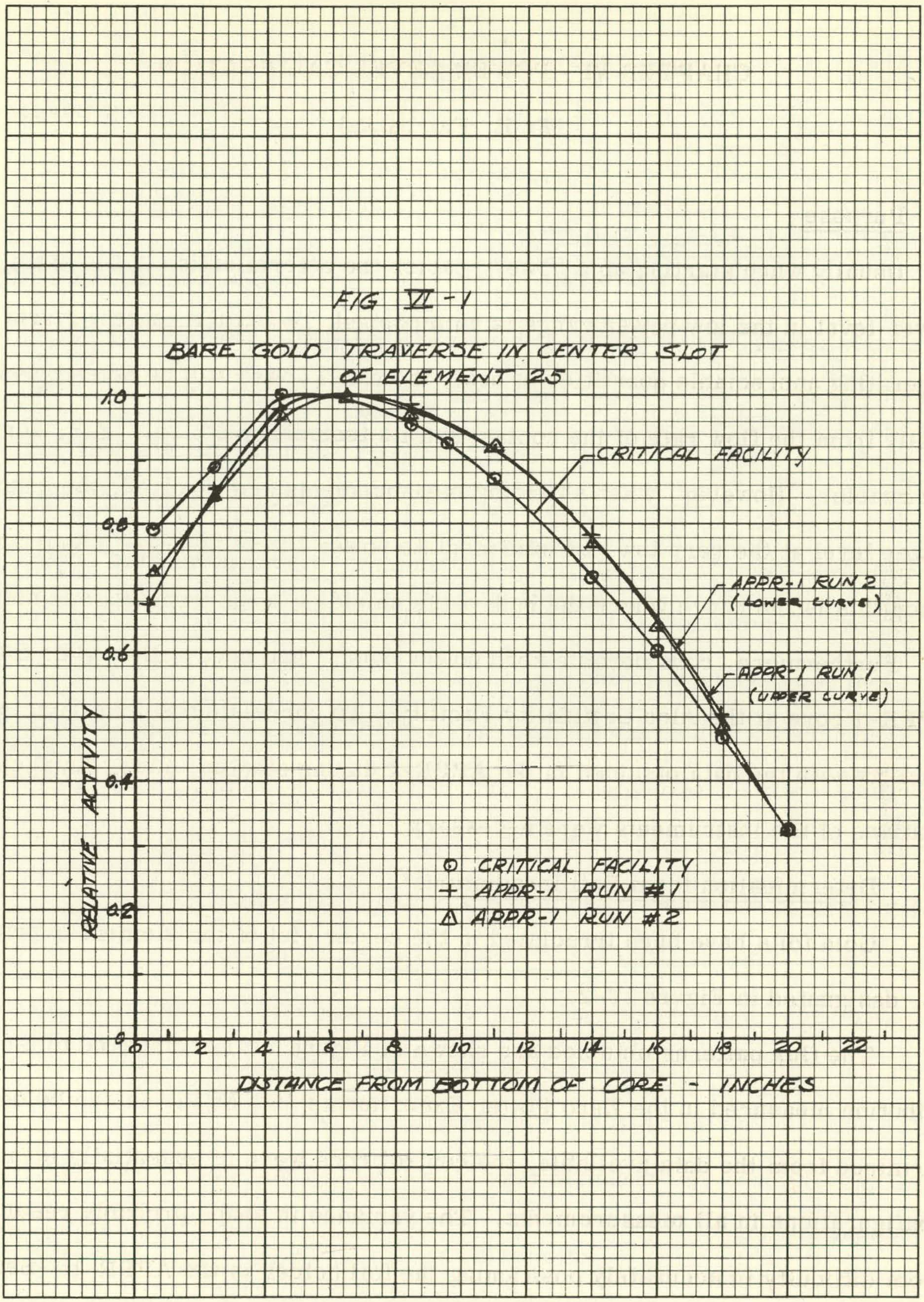


FIG. VI - 1
BARE GOLD TRAVERSE IN CENTER SLOT
OF ELEMENT 25

The foils were counted on the Alco critical facility counter. Based on comparison with previous experiments on power level calibration[7], the power levels were determined to be 6.77 watts on run 1 and 72.4 watts on run 2.

3. Comparison of Instrument Readings

During the low and high power tests, numerous readings have been made of various instruments at a given condition. Of particular interest in relating low and operating power levels are the linear power and Log N recorders. Corresponding Log N and linear power readings have been plotted in Figure VI-2.

During startup, the chamber positions were changed occasionally. Table VI-1 indicates the changes which were made in the linear power and Log N chambers. The magnitude of the change has been determined from Figure VI-2 and from comparison with measured reactor power described in Section 5.

The agreement between the two instruments appears excellent, considering the range of 6 decades involved. Much of the scatter in the data probably results from errors in reading the Log N recorder. At times, some drift in the Log N measurements has apparently occurred in the power range.

4. Effect of Coolant Temperature on Instrument Readings

Two-group multi-region calculations were made to determine the relative boron capture at the chamber positions with the core at 68^{o} and $440^{o}F$. The water surrounding the chamber was assumed to vary from 68^{o} to $150^{o}F$. It was found that the instrument should read 2.95 times as much hot than cold for a given power level.

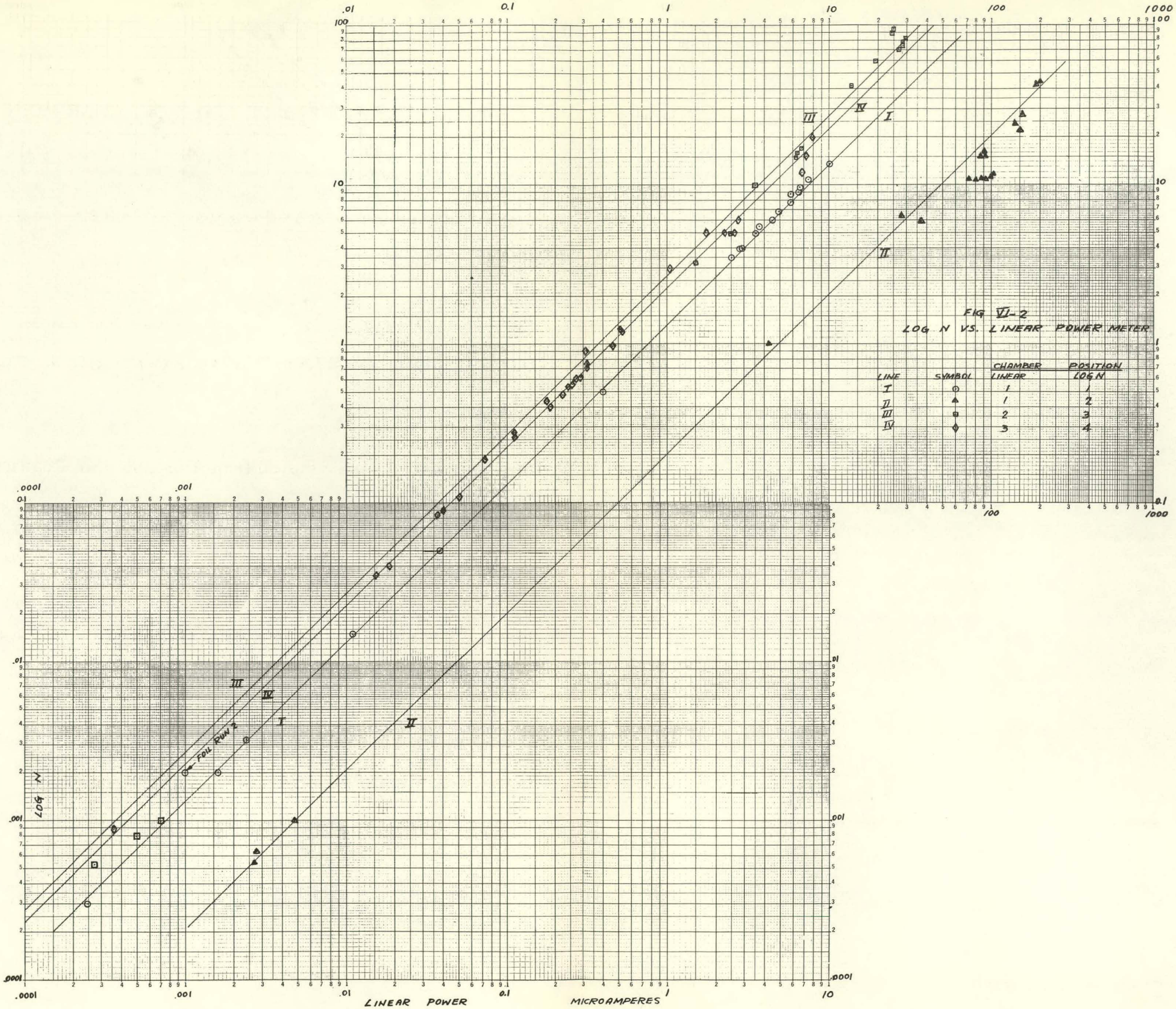


FIG. VI - 2

LOG N VS. LINEAR POWER METER

TABLE VI-1

LINEAR POWER AND LOG N CHAMBER MOVEMENT

April 15 (1:50 am)

Log N chamber moved from Position 1 to Position 2. Readings changed by a factor of 0.155.

April 18 (10:00 pm)

Log N chamber moved from Position 2 to Position 3. Readings changed by a factor of 1.090.

Linear power chamber moved from Position 1 to Position 2. Readings changed by a factor of 0.0841.

May 1 (5:15 am)

Log N chamber moved from Position 3 to Position 4. Readings changed by a factor of 0.617.

Linear power chamber moved from Position 2 to Position 3. Readings changed by a factor of 0.744.

A curve to obtain the correction factor to be applied to instrument readings when the inlet temperature is other than 440°F is given in Figure VI-3. The curve is obtained from the equation

$$\frac{P_{440}}{P_T} = e^{6.19\,(\rho_T - \rho_{440})},$$

where

P_T = instrument reading at Temperature T.

ρ_T = water density in gm/cm^3 at Temperature T.

The constant 6.19 was chosen to fit the calculated ratio from 68° to 440°F.

5. Calibration of Instruments at Power

With the reactor operating at power levels of 1 MW or greater, the Log N and linear power indications were compared with the measured reactor heat power. Data comparing linear power records with measured power is presented in Figure VI-4. Similar data for the Log N recorder is in Figure VI-5. All instrument readings are corrected to an inlet temperature of 440°F.

Table VI-2 compares the relations of the readings of linear power, Log N, and measured heat power. The data in this table is based on Figures VI-2 and VI-4, and consequently may not agree exactly with Figure VI-5.

In Figure VI-6 the calibration curve of the linear power as established in Figure VI-4 is extrapolated to include the range from 10 watts to 10 megawatts. The foil run at 72.4 watts is indicated. The agreement is remarkable, considering the wide range involved, the uncertainty of the gold foil power measurement, and the required temperature correction.

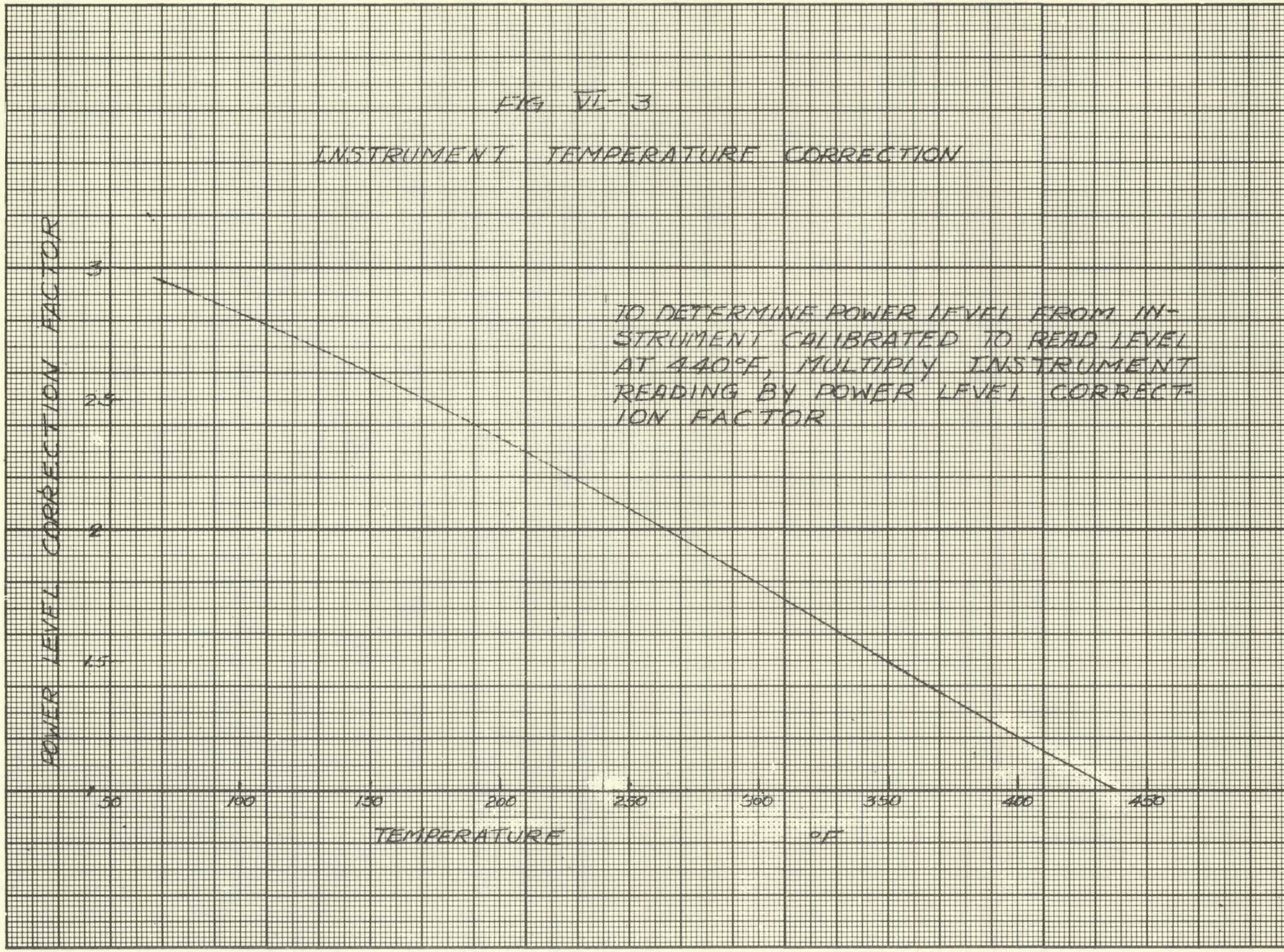


FIG. VI - 3
INSTRUMENT TEMPERATURE CORRECTION

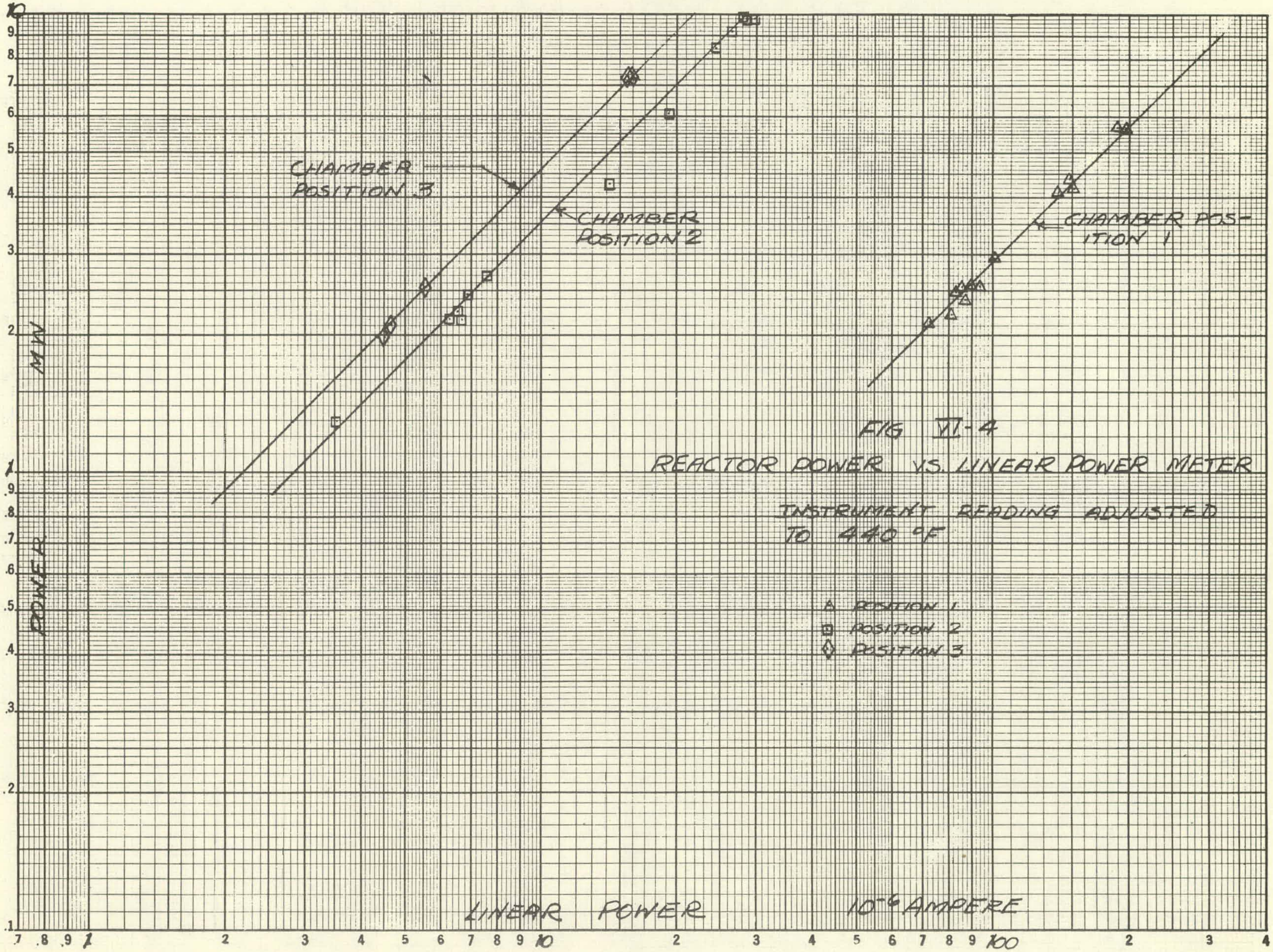


FIG. VI - 4
REACTOR POWER VS. LINEAR POWER METER

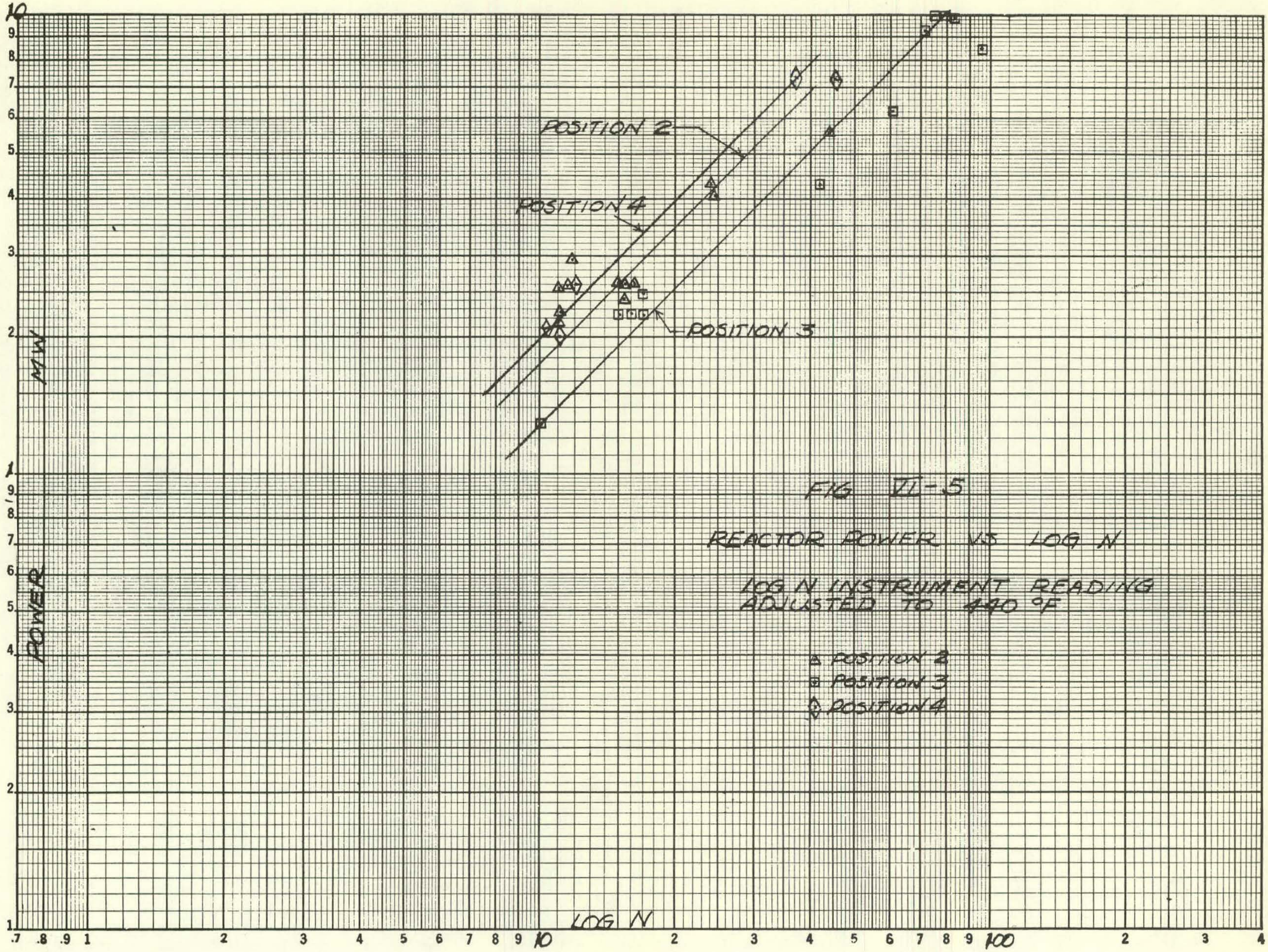


FIG. VI -5
REACTOR POWER VS. LOG N - POWER RANGE

TABLE VI-2

COMPARATIVE INSTRUMENT READINGS

All values are normalized to a power level of 1 megawatt.

Scale	Position	Reading
Linear Power (microamperes)	1	34.4
	2	2.89
	3	2.15
Log N (arbitrary units)	1	45.9
	2	7.11
	3	7.75
	4	4.78
Power (megawatts)	-	1.00

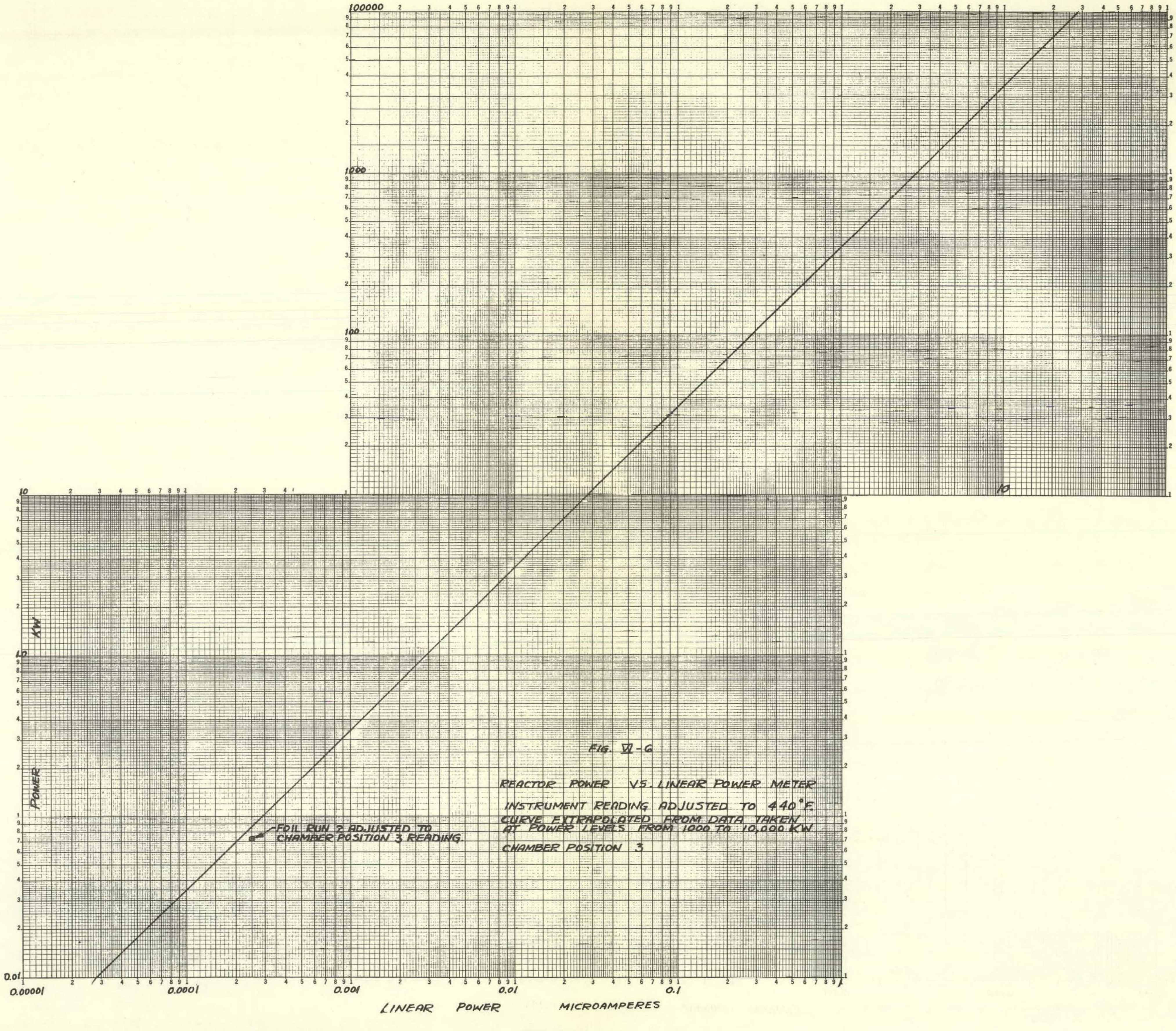


FIG. VI - 6

REACTOR POWER VS. LINEAR POWER METER - EXTENDED RANGE

CHAPTER VII - EMERGENCY COOLING

(J. L. Meem)

1. System Design (W. M. S. Richards)

The APPR-1 primary system was designed so that on loss of flow the reactor would scram and the afterheat would be removed by thermal convection. Attainment of this thermal convection was accomplished by using a steam generator with vertical U-tubes at an elevation higher than the reactor vessel. The installation is described in detail in previous reports[5] but the significant dimensions are as follows:

Height of U-tubes in steam generator, 7.08 ft. Length of reactor passages, 1.83 ft. Height differential from top of reactor passages to bottom of steam generator tubes, 2.42 ft.

Because the density varies throughout the first two of these lengths, their pumping effect is based on half the overall density difference between "hot" and "cold" legs. In the third length the full density difference applies. Based on the half-difference as the common multiplier, the effective column height is thus 7.08 + 1.83 + 2 x 2.42 = 13.75 ft. This generates a pressure differential of 13.75 $\frac{\rho_1 - \rho_2}{2}$ lb/ft^2, where ρ is in lb/ft^3. The thermal coefficient of expansion of water in the range of APPR-1 primary operating temperatures is 0.045 $\frac{\text{lb}}{\text{ft}^3\ {}^\circ\text{F}}$. Thus the pumping effect, in feet of water, is 13.75 x $\frac{0.045}{2}$ x $\frac{1}{\rho_{avg.}}$ per $^\circ$F or 0.309 x $\frac{\Delta T}{\rho_{avg.}}$.

Flow resistance, in feet of fluid, of the primary loop of the APPR-1 is extremely low, amounting to 1.8 x 10^{-6}(gpm)^2. The amount of circulation obtained by natural convection is the result of the equilibrium of these forces.

2. Test Procedure

With this design, the APPR-1 can suffer a complete loss of power without boiling in the core due to afterheat. In case of such a complete loss of power, the reactor will scram, the primary coolant pump (and all other pumps) will stop and the pressurizer heaters will go off. A series of tests at succeedingly higher power levels were run simulating this shutdown from complete loss of power. The procedure was to turn off the primary coolant pump after stabilizing for a given length of time at a particular power level. As the flow dropped the reactor shut down on the low flow scram within a fraction of a second. The pressurizer heaters were turned off to further simulate complete loss of power, but all instrumentation was left on so the results could be recorded.

Calculations indicated that upon loss of flow and scramming of the reactor from full power the temperature in the primary system would rise for 15 to 30 minutes because of afterheat and then the overall system temperature would begin to drop from heat losses through the insulation. To detect if boiling occurred during this period the reactivity of the critical reactor was used. The reactor was brought critical at a very low power as soon as possible after the scram. The neutron level recorders were observed to see if any detectable void formation was occurring in the core.

3. Results and Discussion

This procedure was followed at successively higher power levels each time increasing the power by about 2 megawatts and holding constant power level for 2 hours before shutting off the primary coolant pump. After running the test at 10 megawatts for 2 hours the final run was made at 10 megawatts for 10 hours before shutting off the pump. The records for this run are shown in Figures VII-1, VII-2, and VII-3.

In Figure VII-1 is shown the chart from the Log N recorder. The primary pump was shut off at time 0 and the reactor scrammed from loss of flow immediately. Within a minute the operator started to withdraw the rods and after 10 minutes the reactor was approaching criticality at about 100 watts. After 14 minutes the reactor was definitely critical as determined from the rod position and the behavior of the instruments in comparison with their behavior on previous runs.

The recording of reactor outlet temperature and steam generator outlet temperature is shown in Fig. VII-2, the difference of about 20°F prior to reactor scram agreeing with the reactor Δ T. At time zero the primary coolant pump was shut off and the reactor scrammed. The steam generator outlet temperature is to be disregarded after the pump was shut off as it is automatically disconnected, as the steam generator outlet thermometer is in the pipe leading to the primary pump. Since there are two pumps, there are two thermometers, and a thermometer is connected only when the corresponding pump is turned on. It will be observed from the trace of the reactor outlet temperature that the primary system temperature began to rise after about 20 minutes and continued to rise for approximately one-half hour due to afterheat from the reactor. The system heat losses then overrode the increase in temperature from afterheat and the temperature began to drop. If boiling due to afterheat has not been observed up until the time the temperature in the primary system has passed this peak, it may be concluded that boiling will not occur thereafter.

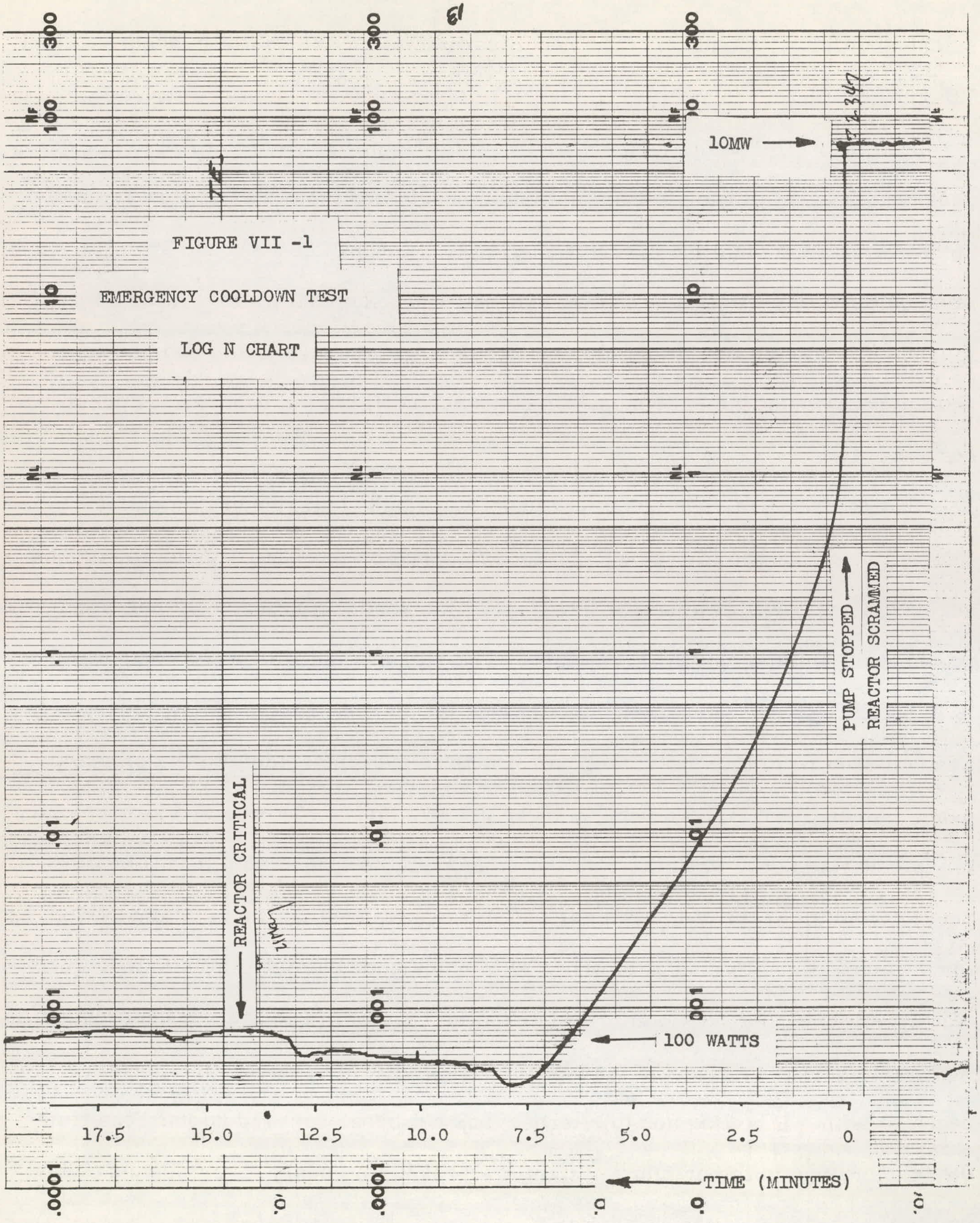


FIG. VII - 1 EMERGENCY COOLING - LOG N CHART

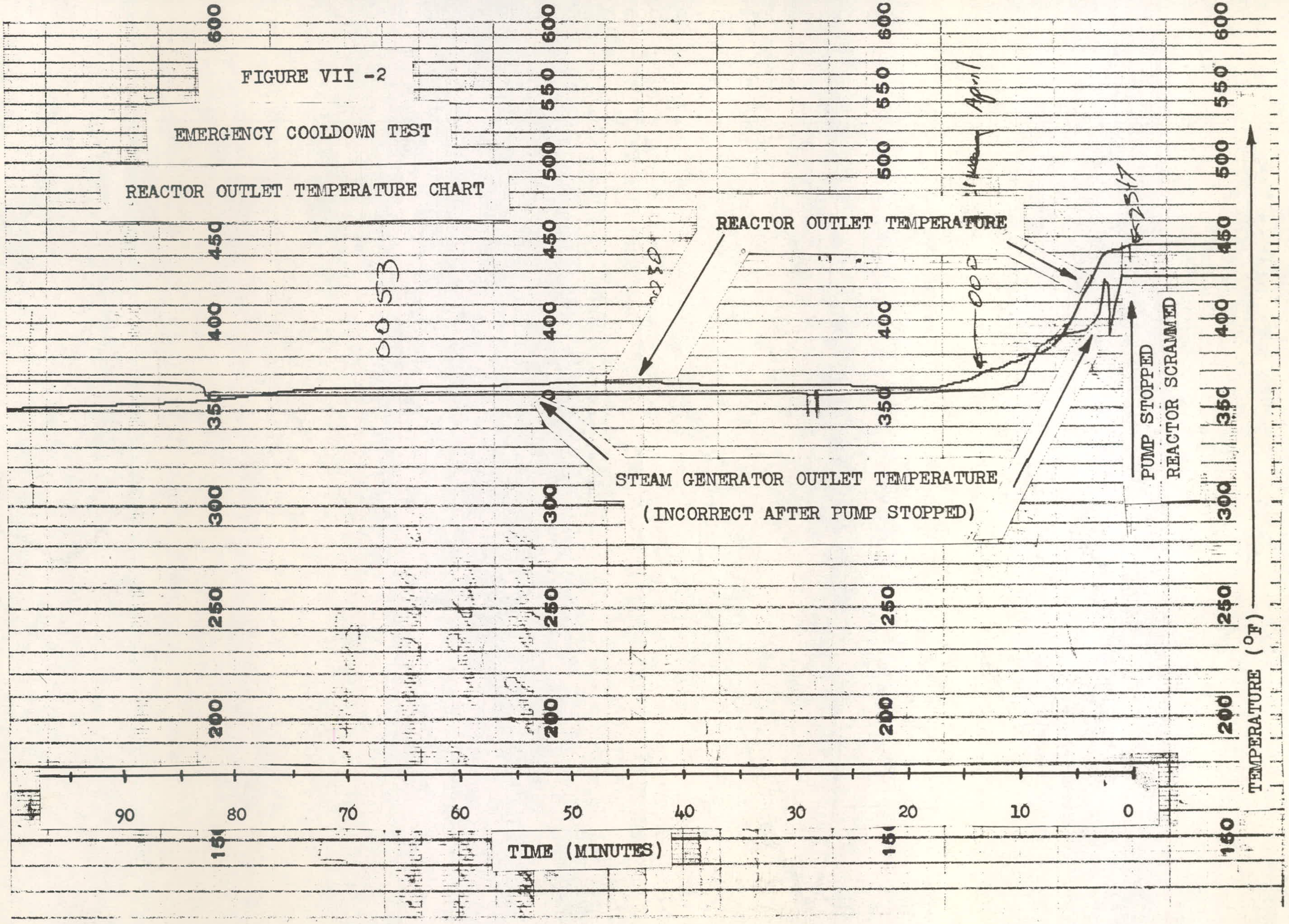


FIG. VII - 2 EMERGENCY COOLING - REACTOR OUTLET TEMPERATURE

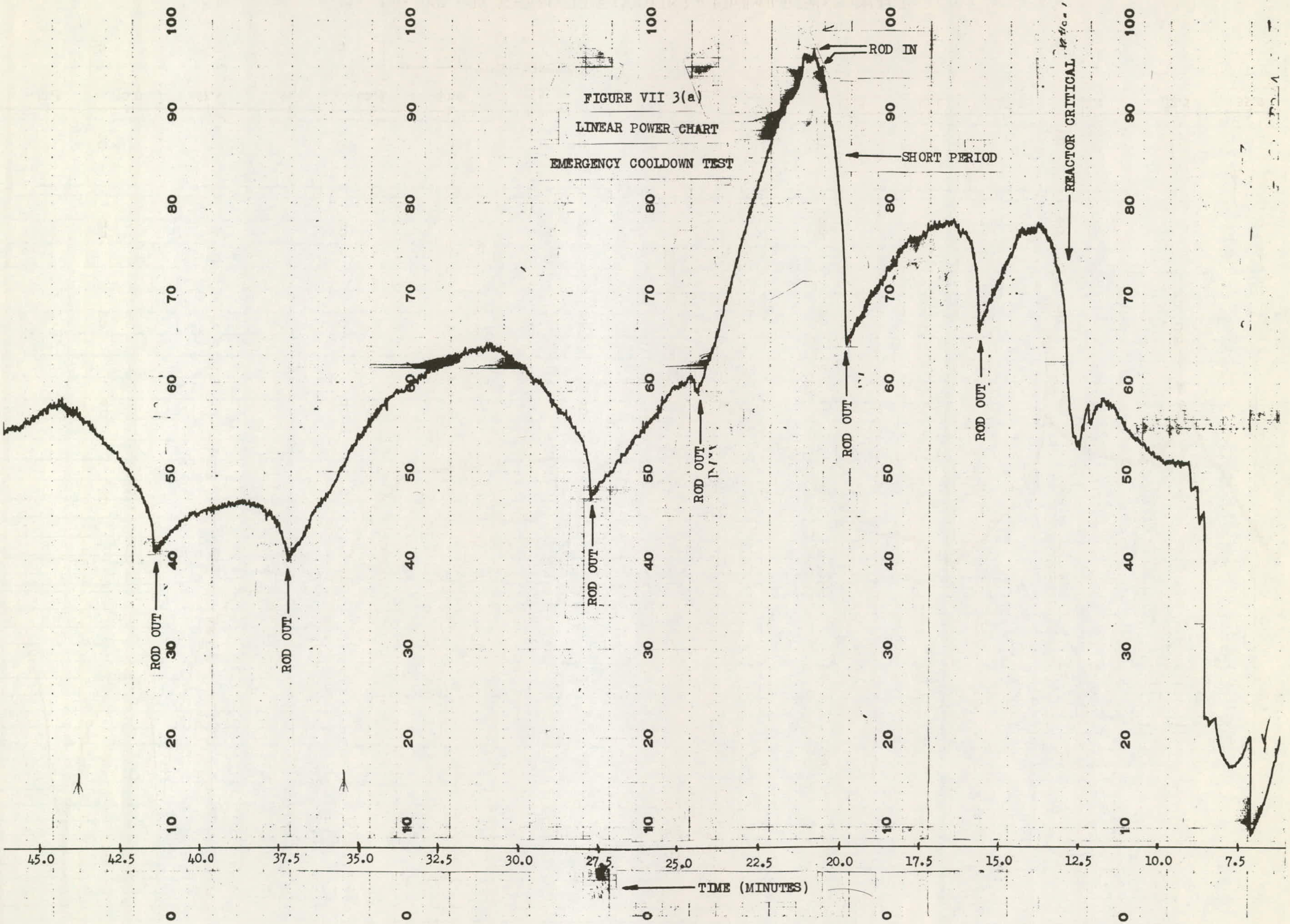


FIG. VII - 3a EMERGENCY COOLING - LINEAR POWER CHART

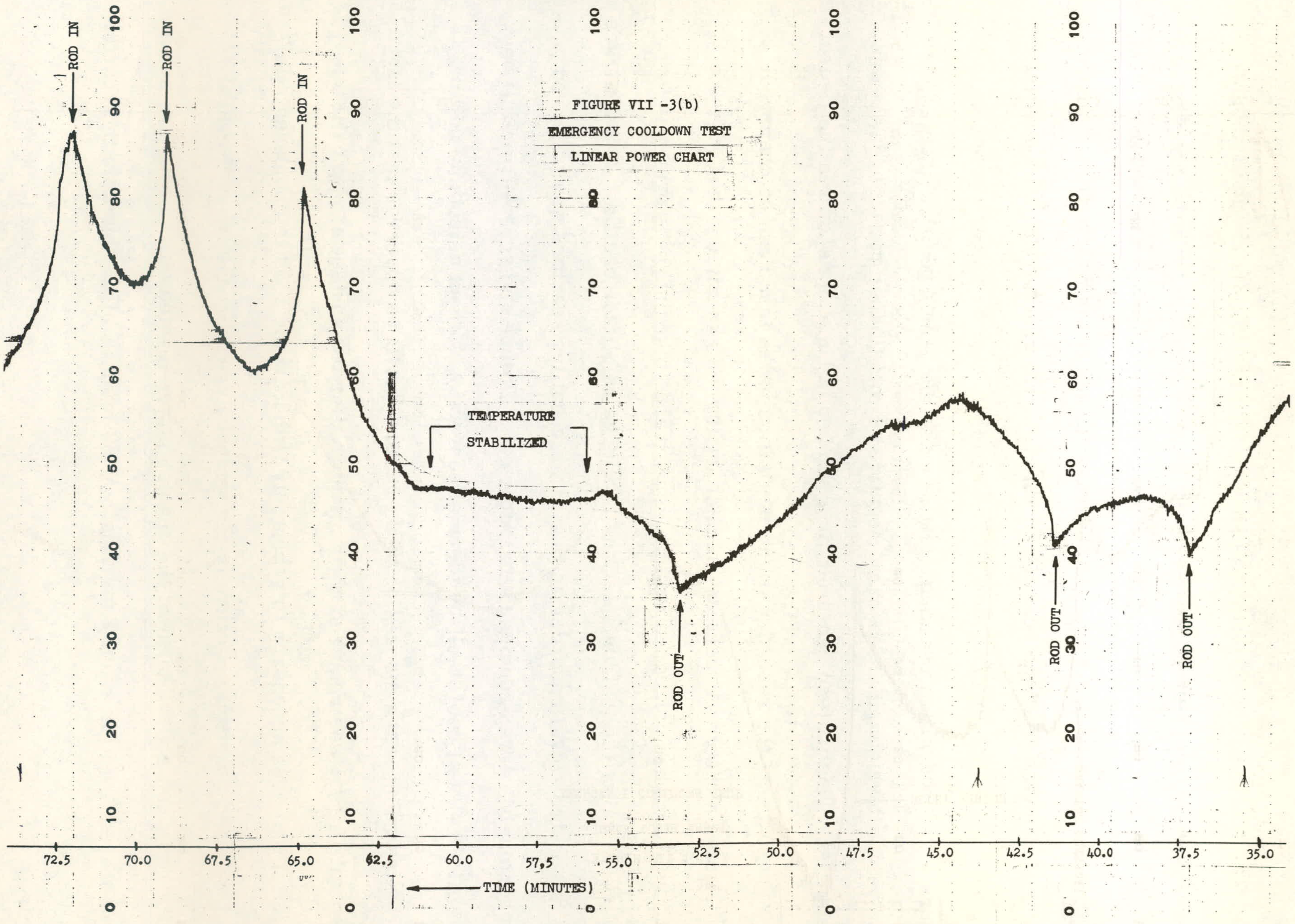


FIG. VII - 3b EMERGENCY COOLING - LINEAR POWER CHART

The linear power recording is shown in Figure VII-3 starting at about 6 minutes after the pump shutoff and scram. Prior to this time the record is of no significance because of scale changes. The record shows the neutron level as the reactor was being brought critical with numerous rod movements up until 13 minutes after the scram. At this time the reactor was on a positive period but leveled off in about a minute and then went on a negative period until there was a rod withdrawal sufficient to make the period positive again. Again the period leveled out and became negative so that further rod withdrawal was required. This pattern repeated for a number of times as shown in the figure.

Throughout the record a slight oscillation in the trace can be observed. This is no more pronounced at one time during the experiment than during another and when compared with previous traces taken at low power was found to be typical of the instrumentation. No large amplitude oscillation in the trace of the neutron level recorder as is typical of boiling reactors was observed. Accordingly, it was concluded that after pump failure and scram from full power the APPR-1 will cool itself by thermal convection without detectable boiling.

Returning to the discussion of the record it will be observed that the last rod withdrawal was required at 53 minutes after the shutdown. Up until this time it had been necessary to repeatedly withdraw the rods to compensate for a decreased reactivity because of the temperature increase from afterheat. After this last rod withdrawal the reactor power leveled off in about 3 minutes and stayed level for about 7 minutes. At 63 minutes after shutdown the reactor started to indicate increased reactivity. This obviously meant that the peak in temperature had been passed and the primary system was beginning to cool down. From that time on rod insertions were required to keep the reactor from going on too short a period as shown. The experiment was then terminated.

CHAPTER VIII - HEAT BALANCE

(G. W. Knighton and H. L. Hoover)

1. Summary

Three heat balance runs were made during the 700-hour performance test. All three runs are included to show the effect of operating variables. The predicted heat balance for the APPR-1 is shown in Figure VIII-1, and all data is tabulated in Table VIII-1.

The first heat balance run, Figure VIII-2, was made with the evaporator secured and no blowdown from the steam generator. Gunston Cove water temperature was 79^{o}F which resulted in a condenser back pressure of 1.82 inches of mercury. 10.59 MW of heat energy was delivered to the steam generator and 2000 KW of electricity was supplied to the ERDL bus. This resulted in a net station heat rate of 18,080 BTU/KW-hr.

The heat balance calculations for this and subsequent runs indicated that the measured steam and feedwater flow rates were too low. Flow rates required to transport the primary system energy to the turbine were calculated and are presented in the heat balance diagrams.

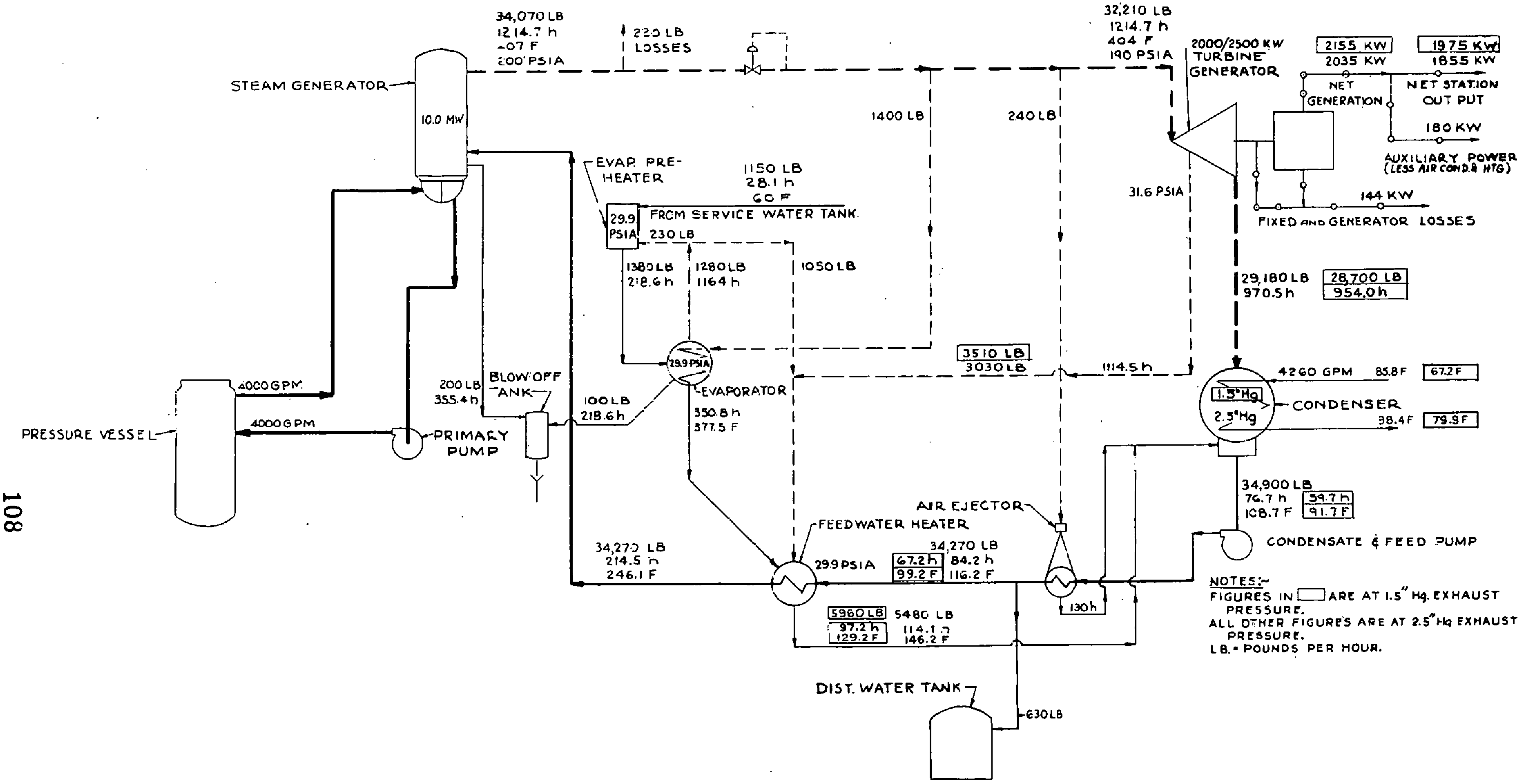
The second heat balance run, Figure VIII-3, was made with the evaporator in service. Due to high solids in the evaporator, the evaporator blowdown rate was high (965 lb/hr) which created an excessive heat loss. The steam generator blowdown rate was 44 lb/hr which was below the predicted rate of 200 lb/hr. Cove water temperature was 85.6^{o}F and gave a condenser back pressure of 2.43 inches of mercury. 10.74 MW of heat energy was delivered to the steam generator and 1925 KW of electricity was supplied to the ERDL bus which gave a net station heat rate of 19,039 BTU/KW-hr.

The third heat balance run, Figure VIII-4, was made with the evaporator in service also. In this run, evaporator blowdown was closer to the predicted value (223 lb/hr vs. 100 lb/hr) but still high. The rate of distilling, however, was well above the predicted rate (1557 lb/hr vs. 1050 lb/hr). The steam generator blowdown rate was still below the predicted rate (135 lb/hr vs. 200 lb/hr). Cove water temperature was 86.6^{O}F and resulted in a condenser back pressure of 2.40 inches of mercury. 10.64 MW of heat energy was delivered to the steam generator and 1900 KW of electricity was delivered to the ERDL bus which gave a net station heat rate of 19,107 BTU/KW-hr.

The predicted heat balance, Figure VIII-1, neglected the air conditioning and heating load. During all three runs, the air conditioning units were in operation. There are four units in the APPR-1 plant each with a 5 Hp motor or 26.8 KW of load. Assuming the motors were using 75% of their rated KW, the air conditioning load would be 20 KW. The net station heat rate for the three heat balance runs were calculated assuming the 20 KW as part of the net station output. The corrected net station heat rates are tabulated below:

Run No.	Net Station Heat Rate
1	17,900 BTU/KW-hr
2	18,843 BTU/KW-hr
3	18,908 BTU/KW-hr

Run No. 2 most nearly meets the guaranteed load conditions as stated in the APPR-1 contract in Appendix B and its attachment 1. This requirement states that, with 10 MW of heat, the net electrical output of the station should be a minimum of 1825 KW when the Gunston Cove water is at 85^{O}F. This produces a net



NET STATION HEAT RATE @ 2.5" Hg = $\frac{34.1 \times 10^6}{1855}$ = 18,385 BTU/NET KWH

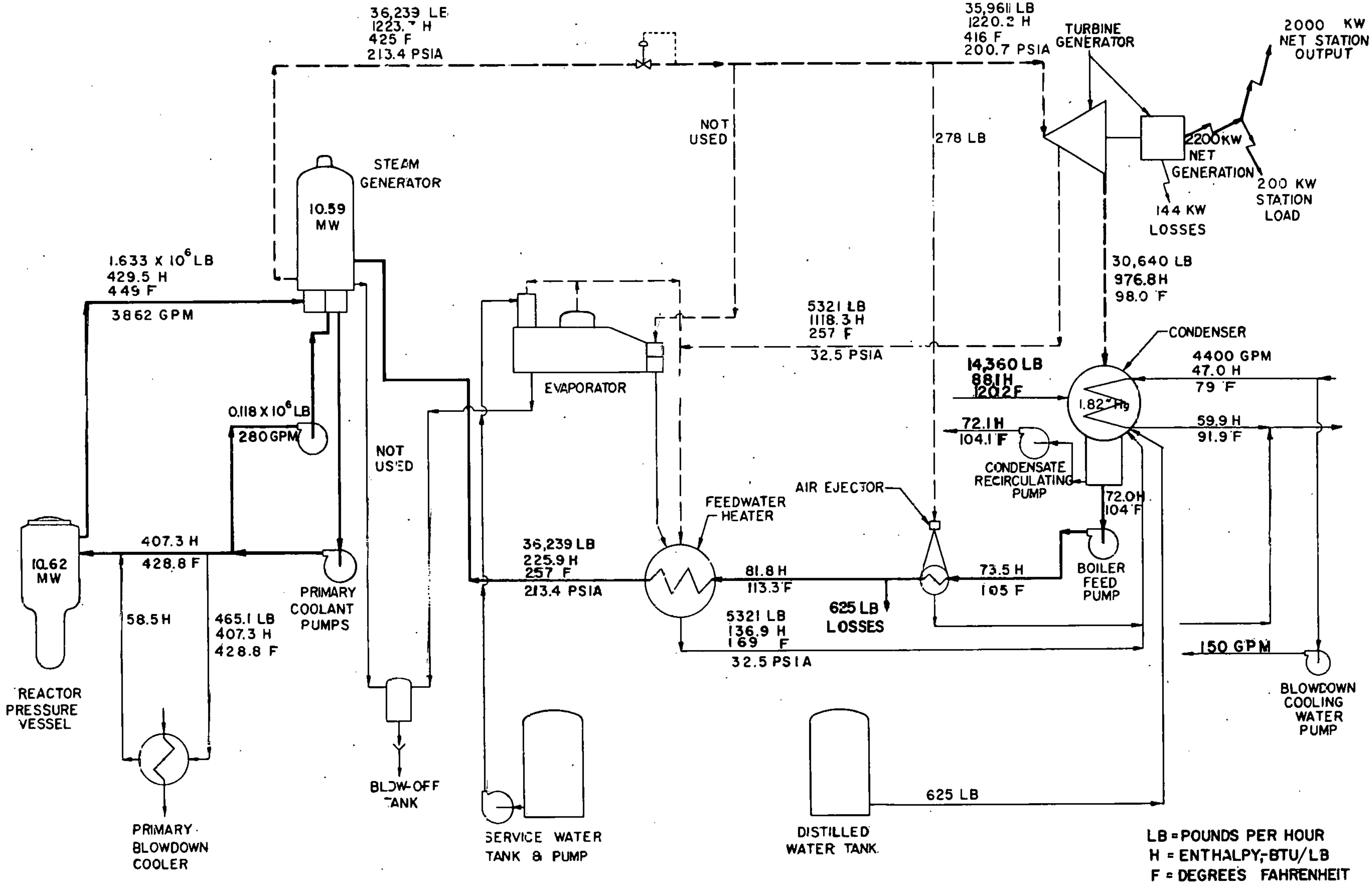
NET STATION HEAT RATE @ 1.5" Hg = $\frac{34. \times 10^6}{\boxed{1575}}$ = $\boxed{17,265}$ BTU/NET KWH

FIG. VIII - 1 PREDICTED HEAT BALANCE, APPR-1

TABLE VIII-1

HEAT BALANCE DATA

Run No.	Predicted 1.5" Hg.	Predicted 2.5" Hg.	1	2	3
PC flow, gpm	4000	4000	3862	3862	3862
Reactor ΔT, ^{o}F	19.7	19.7	20.2	20.3	20.1
Reactor Outlet Temp., ^{o}F	450	450	449	448	450
Reactor Power, MW	10 +	10 +	10.62	10.77	10.66
Steam Generator Heat, MW	10.0	10.0	10.59	10.74	10.64
Steam Flow, lb/hr	34,070	34,070	36,239	36,585	36,472
Steam Press, psia	200	200	213.4	213.4	214.7
Steam Temp., ^{o}F	407	407	425.0	425.0	420.8
Steam Conditions Downstream of Pressure Reducing Valve					
Pressure, psia	190	190	200.7	209.9	214.4
Temperature, ^{o}F	404	404	416	417	415.2
Steam to Evaporator, lb/hr	1400	1400	---	1700	2060
Steam to Ejectors, lb/hr	240	240	278	314	342
Steam to Turbine, lb/hr	32,210	32,210	35,961	34,571	34,070
Extract. Steam to FW Heater, lb/hr	3510	3030	5321	3594	3060
Steam to Condenser, lb/hr	28,700	29,180	30,640	30,977	31,010
Condenser Back Pressure, "Hg	1.5	2.5	1.82	2.43	2.40
River Water Temperature, ^{o}F	67.2	85.8	79.0	85.6	86.6
Cooling Water Flow, gpm	4260	4260	4400	4375	4080
Distilled Water Make-Up, lb/hr	630	630	-625	486	1050

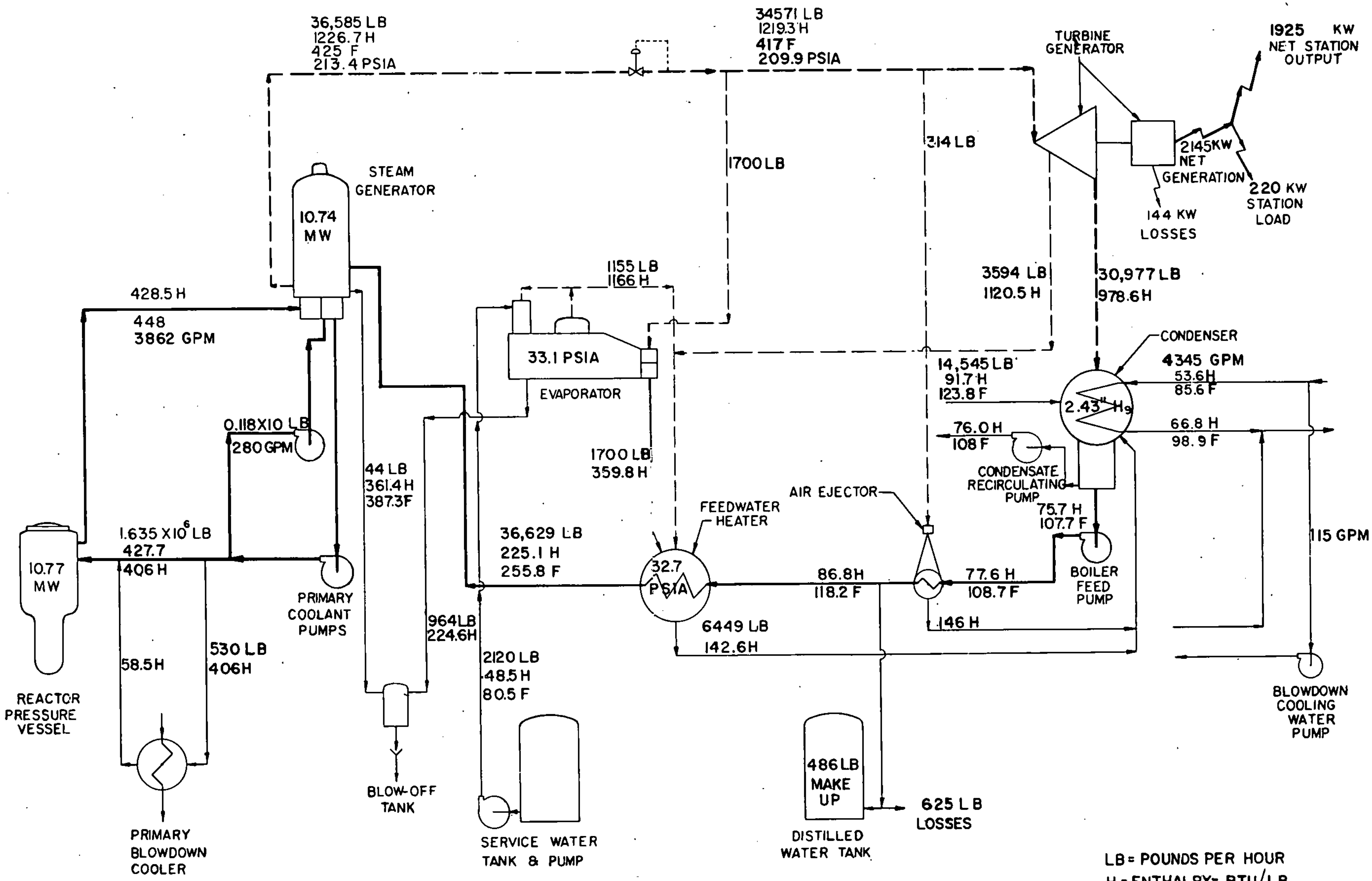


NET STATION HEAT RATE:

EXCLUDING AIR COND. = $\frac{36,159,000}{2000}$ = 18,080 BTU PER KW HR

CORRECTED FOR 20 KW AIR COND. LOAD = $\frac{36,159,000}{2020}$ = 17,900 BTU PER KW HR

FIG. VIII - 2 HEAT BALANCE RUN 1

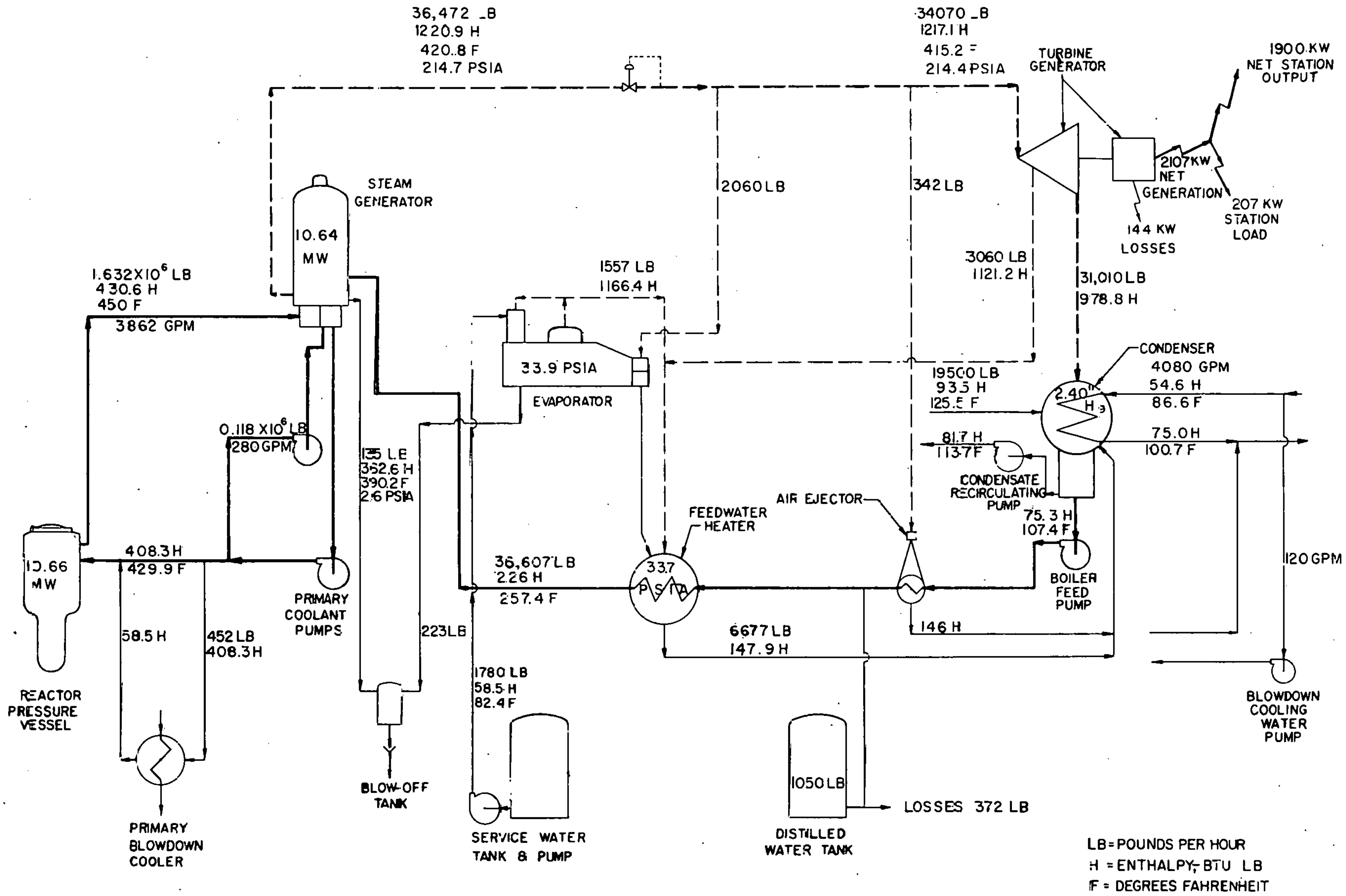


LB = POUNDS PER HOUR
H = ENTHALPY; BTU/LB
F = DEGREES FAHRENHEIT

NET STATION HEAT RATE:
EXCLUDING AIR COND. $= \frac{36,649,000}{1925} =$ 19,039 BTU/KW HR

CORRECTED FOR 20 KW AIR COND. LOAD $= \frac{36,649 \times 10^6}{1945} =$ 18,843 BTU/KW HR

FIG. VIII - 3 HEAT BALANCE RUN 2



NET STATION HEAT RATE: EXCLUDING AIR COND. $= \dfrac{36.304 \times 10^6 \text{ BTU/HR}}{1900 \text{ KW}} = 19{,}107$ BTU/KW-HR

CORRECTED FOR 20 KW AIR COND. LOAD $= \dfrac{36.304 \times 10^6}{1920} = 18{,}908$ BTU/KW-HR

FIG. VIII - 4 HEAT BALANCE RUN 3

station heat rate of 18,700 BTU/KW-hour. The contract also states that this output shall be obtained excluding air conditioning or heating requirements to the station. As previously stated and tabulated above, the actual net station output was corrected to neglect the air conditioning requirements.

During run 2 at 1945 KW net electrical load, the Gunston Cove water temperature was 85.6^{o}F and the condenser produced a back pressure of 2.43 inches of mercury rather than 2.5 as designed. The resulting net station heat rate based on steam generator output was 18,843 BTU/KW-hr. and based on the reactor output 18,897 BTU/KW-hr. These values are respectively within 0.8% and 1.0% of the guarantee rate of 18,700 BTU/KW-hr. at 1825 KW net electrical output. Thus the plant fully meets the contractual guarantee.

2. Determination of Steam Flow for 700-Hour Test (J. K. Leslie and W. R. Johnson)

In accordance with supplemental AEC requirements, data was taken with an artificial increased back pressure in the condenser. Steam flow was regulated to produce a constant 1825 KW net output to the line which did not include the air conditioning correction as previously used. The condenser pressure was regulated, using two methods:

1. Bleeding air into the condenser.
2. Throttling the coolant flow through the condenser cooling tubes.

Due to the insensitivity of control of condenser pressure, using the prescribed air bleed method, it was impossible to run the test holding a constant back pressure of 2.5 inches of mercury. However, the 1825 KW level was maintained for over two hours at back pressures varying from 3.44 to 2.34 inches of mercury. A curve of steam flow versus condenser vacuum, Figure VIII-5, for an 1825 KW line output, was plotted to determine the steam flow required to maintain this power level at 2.5 inches of mercury in the condenser.

As an additional check, flow throttling of condenser coolant was tried, in order to reduce the vacuum. This method gave much finer control of condenser back pressure and data taken agreed well with the "Air Bleed" data. It was necessary to curtail the flow throttling test before 2.5 inches of mercury pressure was reached, however, since the coolant discharge temperature was greater than the range of the thermometers in these lines (100^{O}F).

Steam flow and KW output data were taken from control room instrumentation. Condenser vacuum was determined as the average of three instruments in the plant (at the condenser, at the turbine gage board, and on the graphic panel). Barometric pressure was obtained from Davidson Field.

Figure VIII-6 shows a plot of steam flow, condenser pressure and line output versus time for the trace of the artificial condenser back pressure test.

Although heat balance data was not taken, Table VIII-2 lists pertinent data which was recorded during the test. The results of this test established 34,000 lb/hr recorded steam flow as the maximum output to be maintained, consistent with available load, during the 700-hour test.

3. Procedure used for Heat Balance Calculations

A primary system heat balance was calculated using the recorded values of primary coolant flow, reactor Δ T, and primary blowdown rate. The energy added by the primary coolant pump was included. This balance established the energy transfer rate in the steam generator.

Using the energy transfer rate above, a steam flow rate was calculated. Secondary system water losses were established by the first heat balance run when

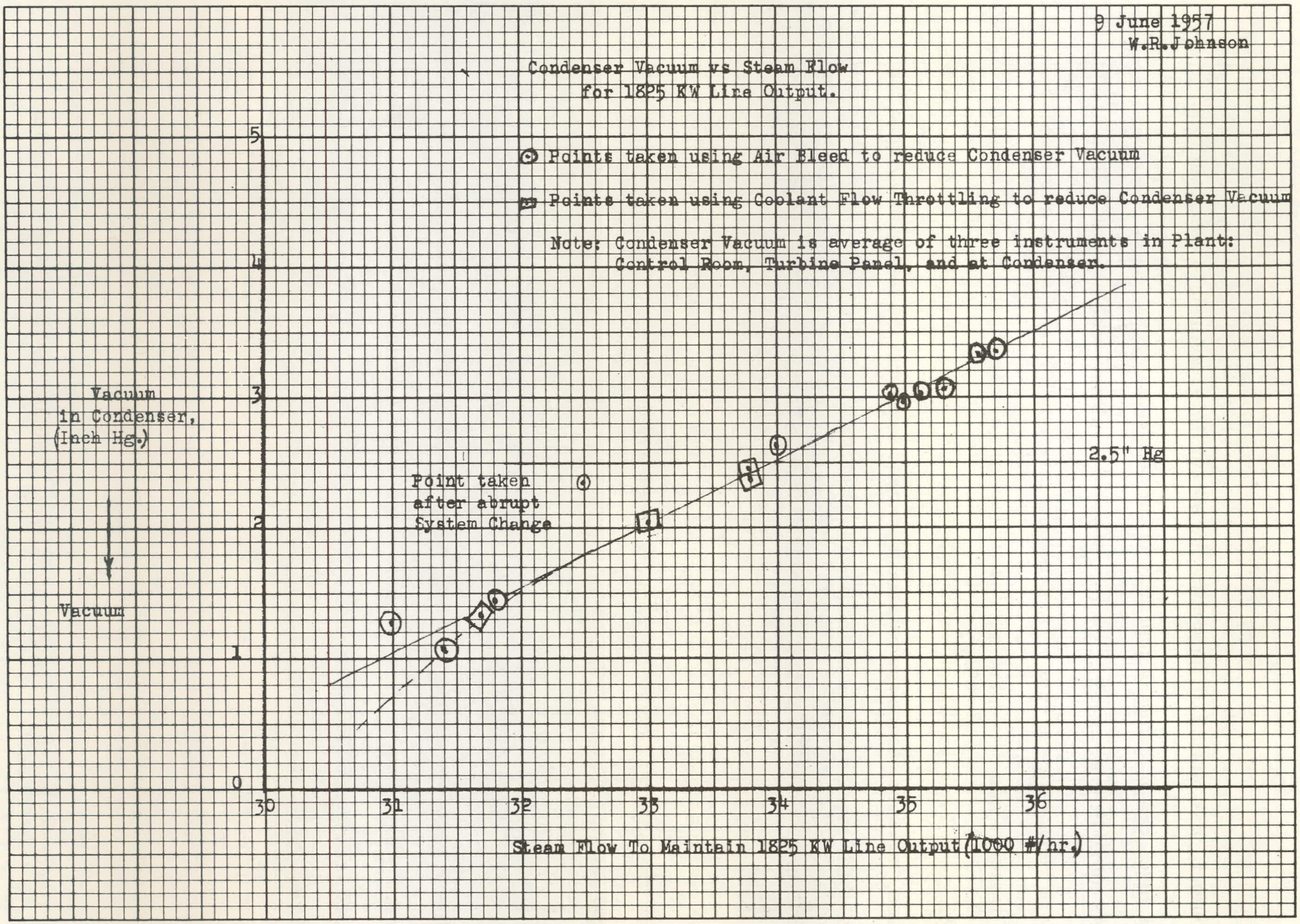


FIG. VIII - 5 CONDENSER VACUUM VS. STEAM FLOW FOR 1825 KW LINE LOAD

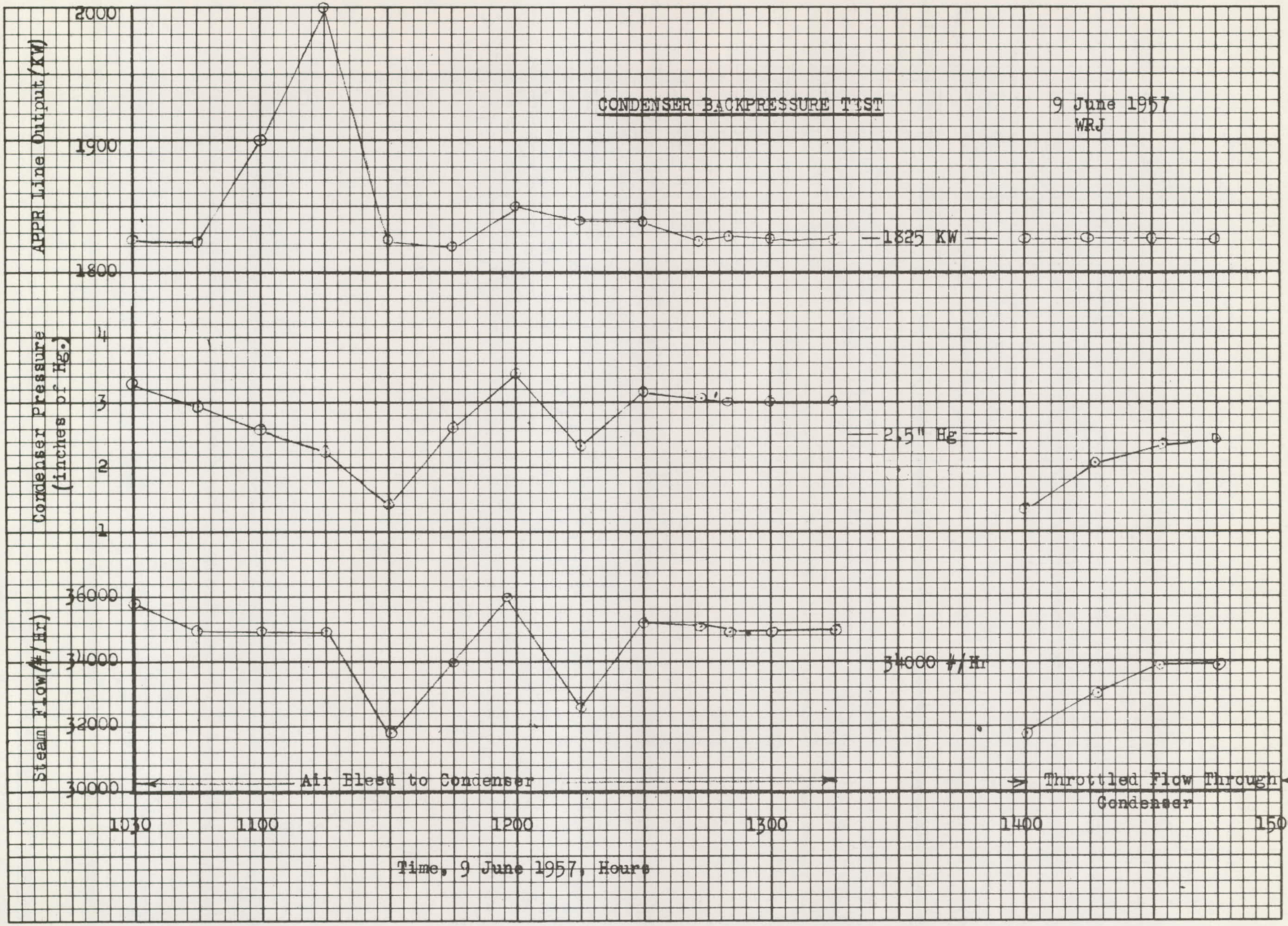


FIG. VIII - 6 CONDENSER BACK PRESSURE TEST CURVES

Beginning June 9, 1957
Ending June 10, 1957

TABLE VIII-2

DATA: STEAM FLOW VERSUS CONDENSER BACKPRESSURE

Note: Line KW 1825 +15/-5 During Test

			Condenser Pressure, "Hg				
Time	Steam Flow	Reactor ΔT	Control Room	Cond.	Turbine Panel	Boron Press	Avg. Condenser Backpressure
0900	31,400	18.25	29.5	29.0	28.7	30.14	1.07
1030	35,700	20.7	27.2	26.9	26.4		3.31
1045	35,000	20.2	27.4	27.3	26.9		2.97
1130	31,800	18.8	29.0	29.0	28.2		1.44
1145	34,000	19.9	28.2	27.5	27.0		2.64
1215	32,500	18.5	28.1	27.4	28.0		2.34
1225	35,300	20.4	27.7	27.0	26.5		3.08
1243	35,150	20.3	27.5	26.9	26.8		3.06
1250	34,900	20.1	27.6	27.0	26.8		3.01
1300	34,900	20.1	27.6	27.0	26.8	30.13	3.01
1400	31,700	18.9	29.2	28.9	28.3	30.12	1.34
1417	33,000	19.4	28.6	28.0	27.7		2.02
1432	33,800	19.8	28.2	27.75	27.3		2.37
1445	33,800	19.7	28.1	27.75	27.2		2.42
0823	31,000	18.5	29.1	28.9	28.5	30.13	1.29

the steam generator blowdown and the evaporator were secured. This loss rate of 625 lb/hr was used for run 1 and 2. Prior to run 3, a second loss rate was determined. This rate of 372 lb/hr was used for run 3.

A material balance around the secondary system determined the evaporator feed rate for runs 2 and 3. A heat balance around the evaporator then determined the steam flow rate to the evaporator. A heat balance around the ejector determined the steam flow rate to the ejector.

A heat balance around the feedwater heater determined the energy requirement of the extraction steam from the turbine. By using the turbinc performance curves and a trial and error method of solution, the extraction steam flow rate and enthalpy were determined.

An electrical energy balance around the generator determined the energy transferred from the turbine to the generator. The enthalpy of the exhaust steam from the turbine was then calculated.

The condensate recirculation rate was determined by the total dynamic head (TDH) of the condensate recirculating pump and the performance curve of TDH vs flow. A heat balance was then made around the condensate recirculation loop.

With all heat input to the condenser known, a heat balance around the condenser determined the condenser duty and the required cove cooling water flow rate. This flow rate was checked against the pumping rate determined by the TDH and performance curve for the cooling water pump.

4. Sample Calculations

Run 2 was selected for the sample calculations because it was the first run made utilizing the evaporator and conditions of load, cooling water temperature, etc., were nearest design conditions.

The nomenclature for enthalpy is that used in Keenan and Keyes text, "Thermodynamic Properties of Steam".

Reactor outlet Temp.	$= 448^{o}F$
h_f at $448^{o}F$	$= 427.9$ BTU/lb
$h-h_f$ at 1200 psia and $448^{o}F$	$= 0.6$ BTU/lb
h	$= 428.5$ BTU/lb
Reactor ΔT	$= 20.3^{o}F$
Reactor inlet Temp.	$= 427.7^{o}F$
h_f at $427.7^{o}F$	$= 405.3$ BTU/lb
$h-h_f$ at 1200 psia and $427.7^{o}F$	$= 0.7$ BTU/lb
h	$= 406.0$ BTU
PC flow	$= 3862$ gpm
	$= 1.634 \times 10^6$ lb/hr
Q (react) $= 1.634 \times 10^6 (428.5-406.0)$	$= 36.765 \times 10^6$ BTU/hr
Q (PC blowdown) $= 530 (406.0-58.5)$	$= 0.184 \times 10^6$ BTU/hr
Q (Pump work) $= 1.285 \times 10^{-3}$ BTU/ft-lb W_H	

From VIII-7,

$H = 30$ ft at 4142 gpm $(1.752 \times 10^6$ lb/hr)

Q (Pump work) $= 1.285 \times 10^{-3} \times 1.752 \times 10^6 \times 30 = 0.068 \times 10^6$ BTU/hr

Q (Steam Gen) = Q (React) + Q (Pump work) - Q (PC Blowdown)

$= (36.765 + 0.068 - 0.184) \times 10^6 = 36.649 \times 10^6$ BTU/hr

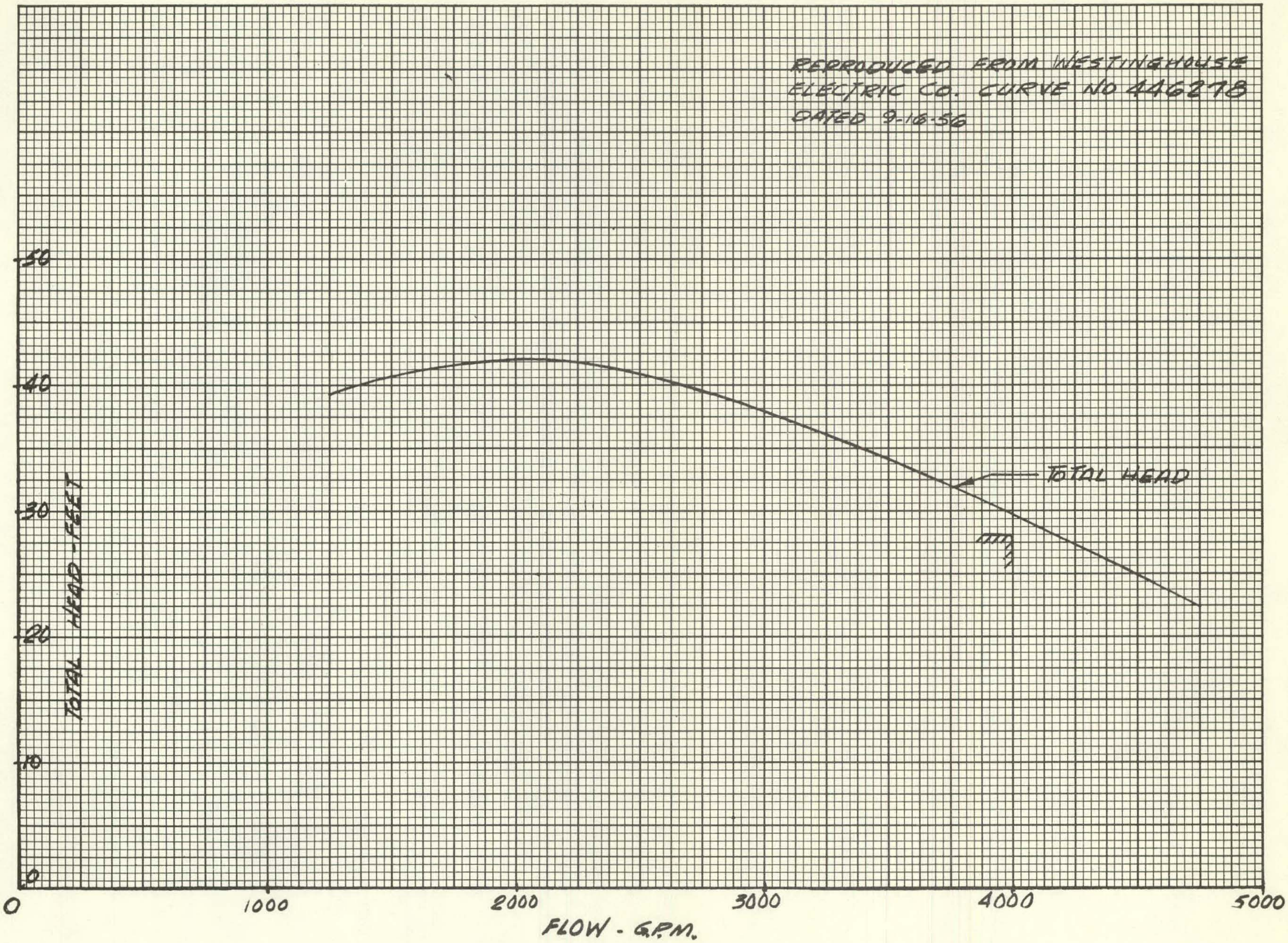


FIG. VIII - 7 PRIMARY COOLANT PUMP PERFORMANCE CURVE

Steam Conditions upstream of pressure reducing valve:

Temperature	425 ^{o}F
Gage Pressure	198.5 psi
Barometric Pressure	14.9 psi
Absolute Pressure	213.4 psi
Enthalpy	1226.7 BTU/lb

Steam generator blowdown temperature	387.3 ^{o}F
Enthalpy	361.4 BTU/lb
Flow rate	44 lb/hr
Feedwater Temperature	255.8 ^{o}F
Enthalpy	225.1 BTU/lb

Q (Steam Gen.) $= 36.649 \times 10^6$ BTU/hr

$= X$ lb/hr (1226.7) - 44 (361.4) - (X + 44) 225.1

Steam flow rate = X	=	36,585 lb/hr
Feedwater flow rate = (X + 44)	=	36,629 lb/hr

Steam conditions downstream of pressure reducing station:

Temperature	417 ^{o}F
Absolute pressure	209.9 psi
Enthalpy	1219.3 BTU/lb

Material balance for secondary system

Steam generator blowdown	=	44 lb/hr
Evaporator blowdown	=	965 lb/hr
Plant losses, previous run	=	625 lb/hr
Evaporator feed = 254 gal/hr	=	2120 lb/hr
Distilled water make-up = 2120 - 1634	=	486 lb/hr

Evaporator heat balance

$$2120\ (48.5) + X(1219.3) = (486 + 44 + 625)\ 1166 + 965\ (224.6) + X(359.8)$$

$$102{,}800 + X(1219.3 - 359.8) = 1{,}346{,}700 + 216{,}700$$

$$\text{Steam to evap.} = X = \frac{1{,}460{,}600}{859.5} = 1700 \text{ lb/hr}$$

Ejector heat balance

$$36{,}629\ (86.8 - 77.6) = (1219.3 - 146)X$$

$$\text{Steam to ejector} = X = \frac{337{,}000}{1073.3} = 314 \text{ lb/hr}$$

$$\text{Steam to turbine} = 36{,}585 - 1700 - 314 = 34{,}571 \text{ lb/hr}$$

Feedwater heater heat balance

$$Q\ (\text{feedwater}) = 36{,}629\ (225.1 - 86.8) = 5{,}065{,}800 \text{ BTU/hr}$$

$$Q\ (\text{from evaporator}) = 1700\ (359.8 - 142.6) + 1155\ (1166 - 142.6)$$

$$= 369{,}200 + 1{,}182{,}000 = 1{,}551{,}200 \text{ BTU/hr}$$

$$Q\ (\text{extraction steam}) = 5{,}065{,}800 - 1{,}551{,}200 = 3{,}514{,}600 \text{ BTU/hr}$$

$$3{,}514{,}600 = X(h - 142.6)$$

For $X = 3594$ lb/hr

From Fig. VIII-8, extraction pressure = 36.3 psia

From Fig. VIII-9, extraction enthalpy = 1120.5 BTU/lb

$$3{,}514{,}600 = 3594\ (1120.5 - 142.6) = 3{,}514{,}600$$

Generator energy balance

Net Station output	=	1925 KW
Station load	=	220 KW
Turbine - generator losses	=	144 KW
		2289 KW
2289 x 3413	=	7,812,400 BTU/hr

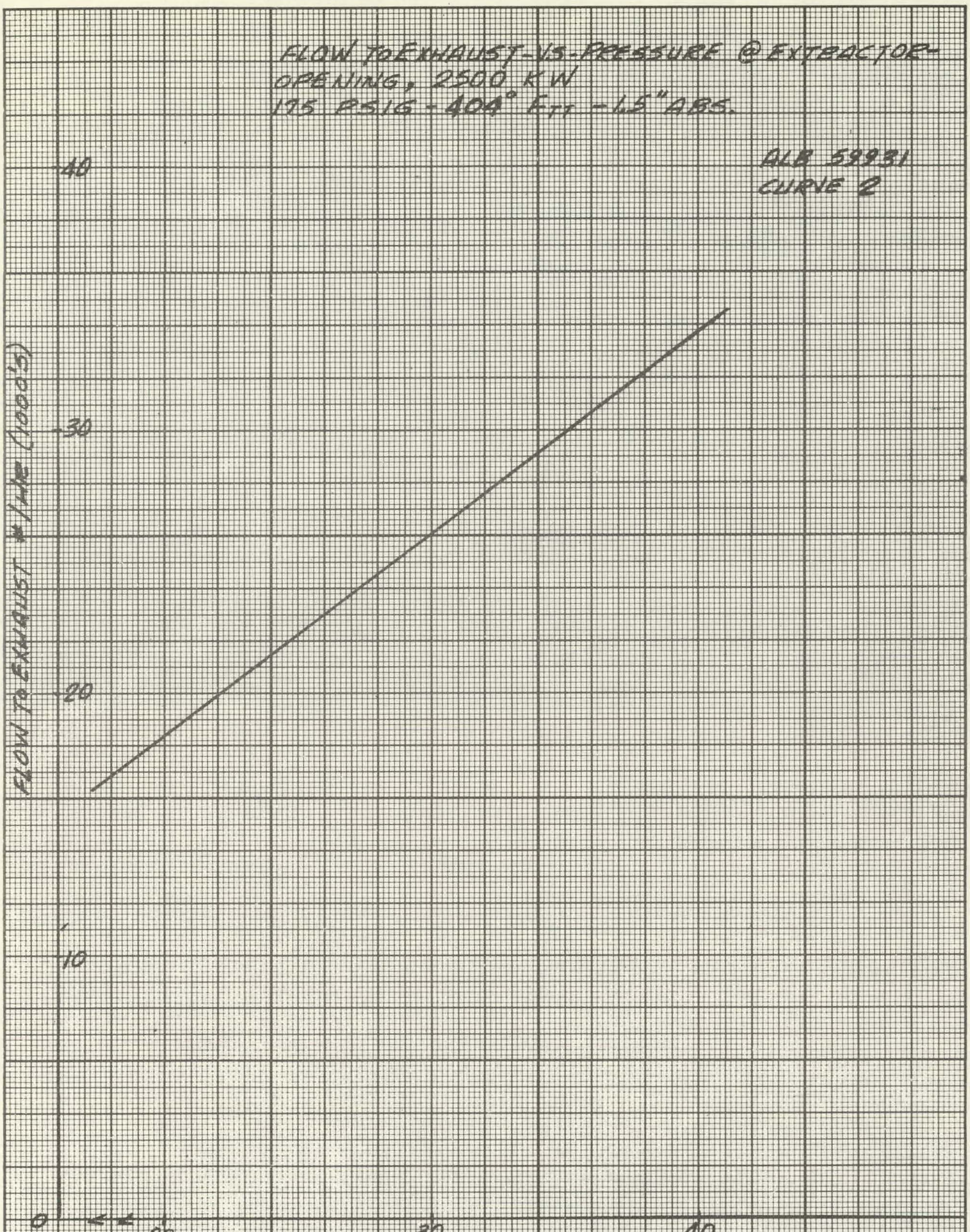


FIG. VIII - 8 EXTRACTION STEAM PRESSURE VS. FLOW TO TURBINE EXHAUST

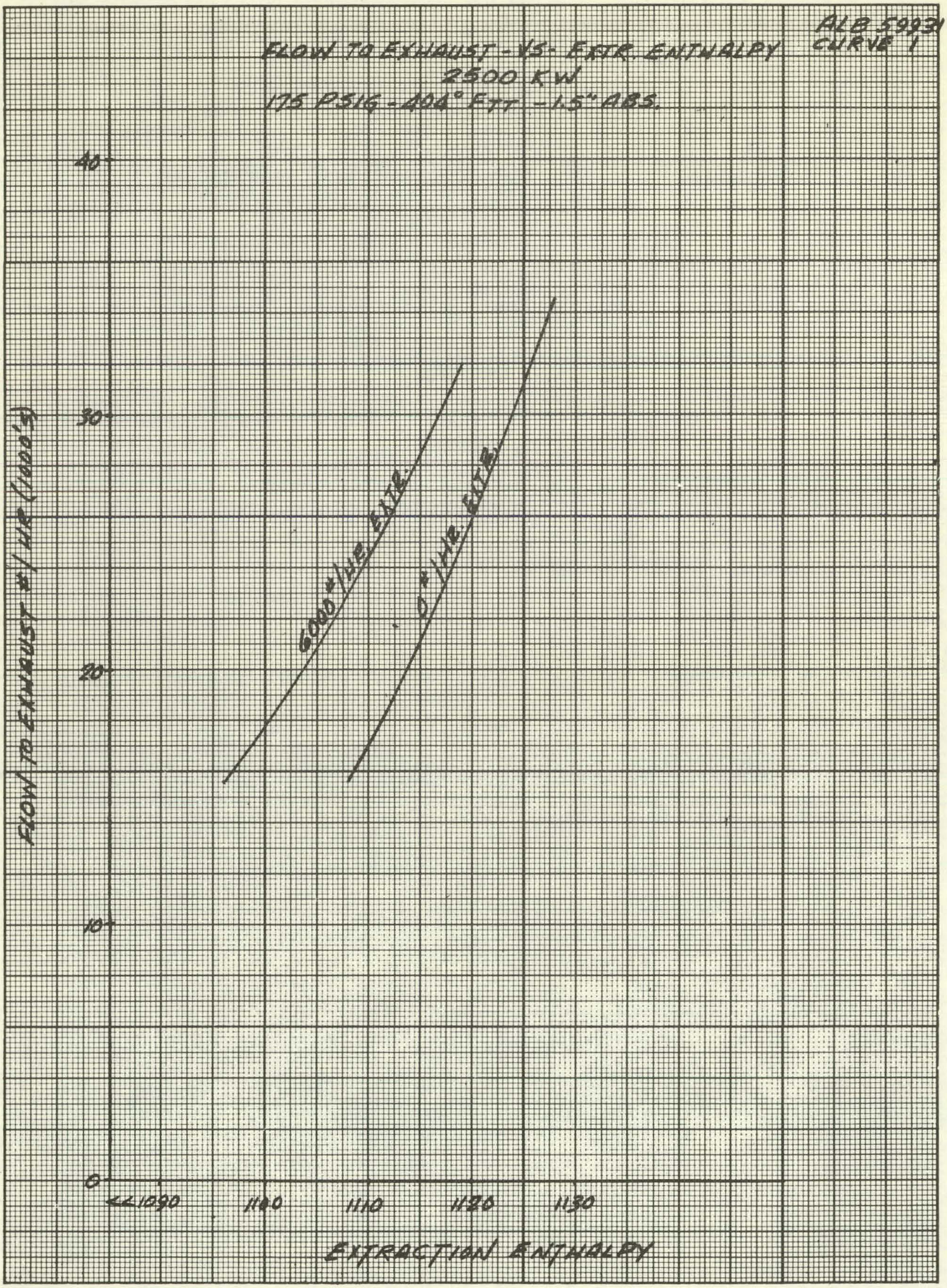


FIG. VIII - 9 EXTRACTION STEAM ENTHALPY VS. FLOW TO TURBINE EXHAUST

Turbine heat balance

$$7,812,400 = 34,571\,(1219.x - 1120.5) + (34,571 - 3,594)\,(1120.5 - h)$$

$$30,977\,h = 3,415,600 + 34,709,700 - 7,812,400$$

$$h = \frac{30,312,900}{30,977} \qquad = 978.6 \text{ BTU/lb}$$

Condensate Recirculation

Head on recirculation pump

Discharge pressure = 74.1 psig x 2.33 = 172.7 ft

Hotwell level above floor = 22 in. = 1.8 ft

Pressure gage elevation above floor = 60 in. = 5.0 ft.

Condenser vacuum = 32.3 ft.

Total head = 172.7 - 1.8 + 5.0 + 32.3 = 208.1 ft.

From Fig. VIII-10, flow = 31 gpm

= 15,376 lb/hr

Recirculating pump work = $1,285 \times 10^{-3}$ WH

= 1.285×10^{-3} (15,376) (208.1) = 4112 BTU/hr

Condensate recirculation loop heat balance

Q = 15,376 (91.7 - 76.0) + 4,100

= 241,400 + 4,100 = 245,500 BTU/hr

Condenser Duty

30,977 (978.6 - 75.7) + (1700 + 1155 + 3594) (142.6-75.7)

+ 314 (146 - 75.7) + 245,500

= 27,969,100 + 431,400 + 22,100 + 245,500 = 28,668,100 BTU/hr

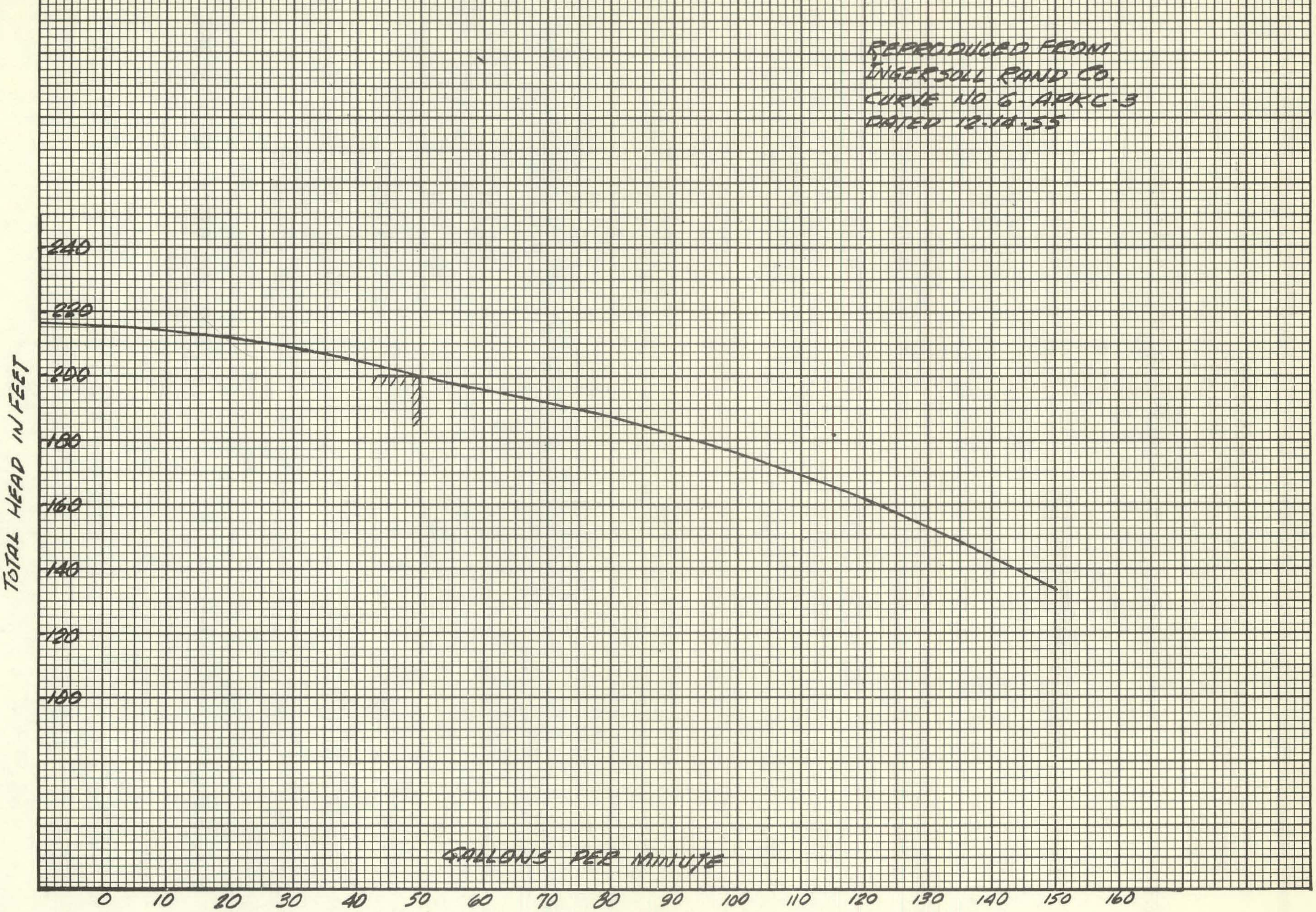


FIG. VIII - 10 CONDENSATE RECIRCULATING PUMP PERFORMANCE CURVE

Condenser cooling water flow $= \dfrac{28,668,100}{66.8-53.6}$

$= 2,171,800$ lb/hr $= 4345$ gpm

Calculated flow from TD_H

Discharge pressure	= 19 psig x 2.31	=	43.9 ft
Water level to gage		=	8.5 ft
			52.4 ft
From Fig. VIII-11, flow		=	4460 gpm
			-4345
Flow through booster pump		=	115 gpm

Boiler Feed Pump

Head on Pump

Discharge pressure	= 283 psi	=	659.4 ft
Hotwell level above floor		=	-1.8 ft
Pressure gage above floor		=	5.0 ft
Condenser vacuum		=	32.2 ft
TDH			694.8 ft
From Fig. VIII-12, flow		=	76 gpm
76 x 60 x 61.8/7.48		=	37,675 lb/hr

Feedwater flow rate from heat balance = 36,629 lb/hr

$$\frac{37,675 - 36,629}{36,629} \times 100 = 2.86\%$$

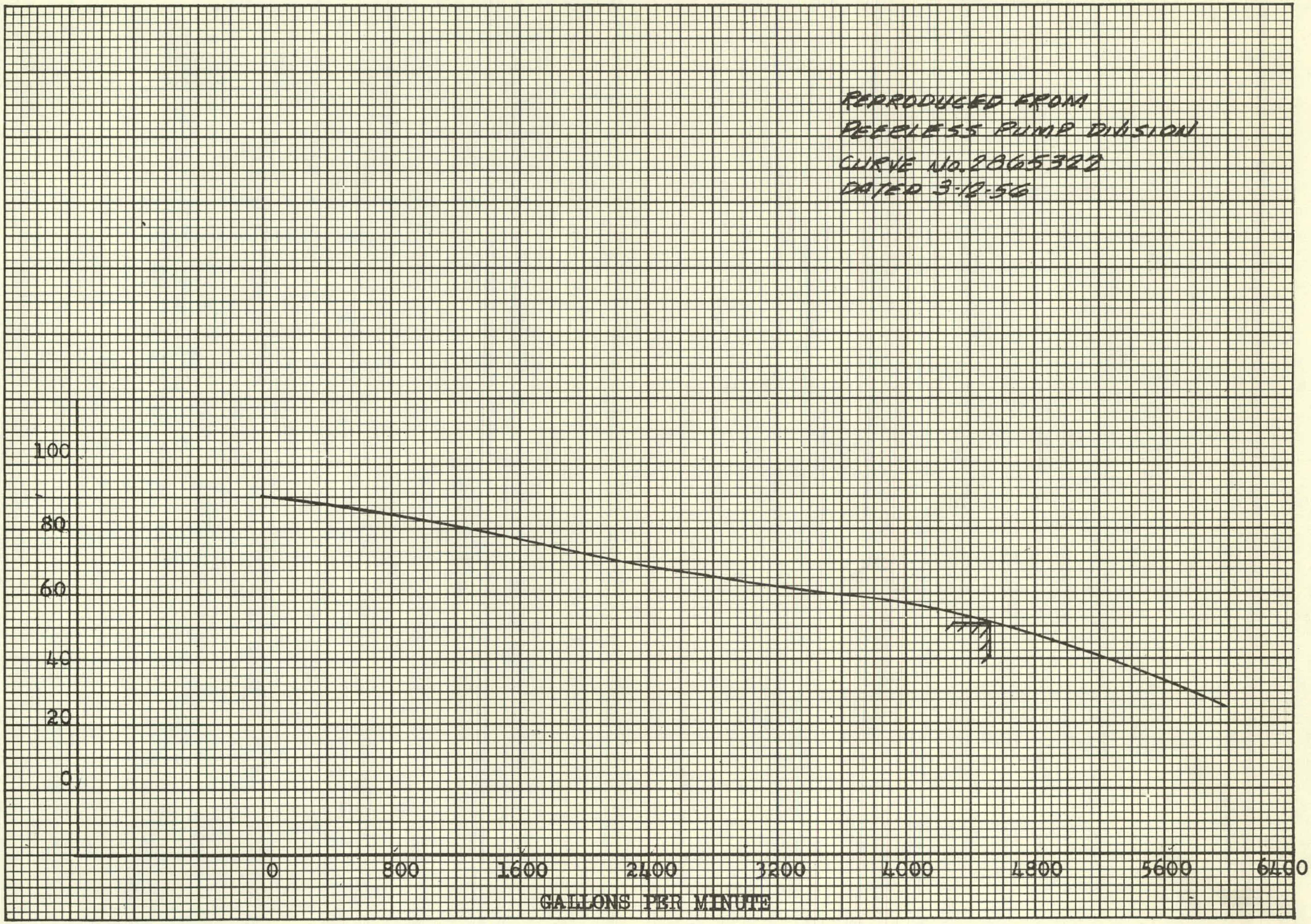


FIG. VIII - 11 CONDENSER COOLING WATER PUMP PERFORMANCE CURVE

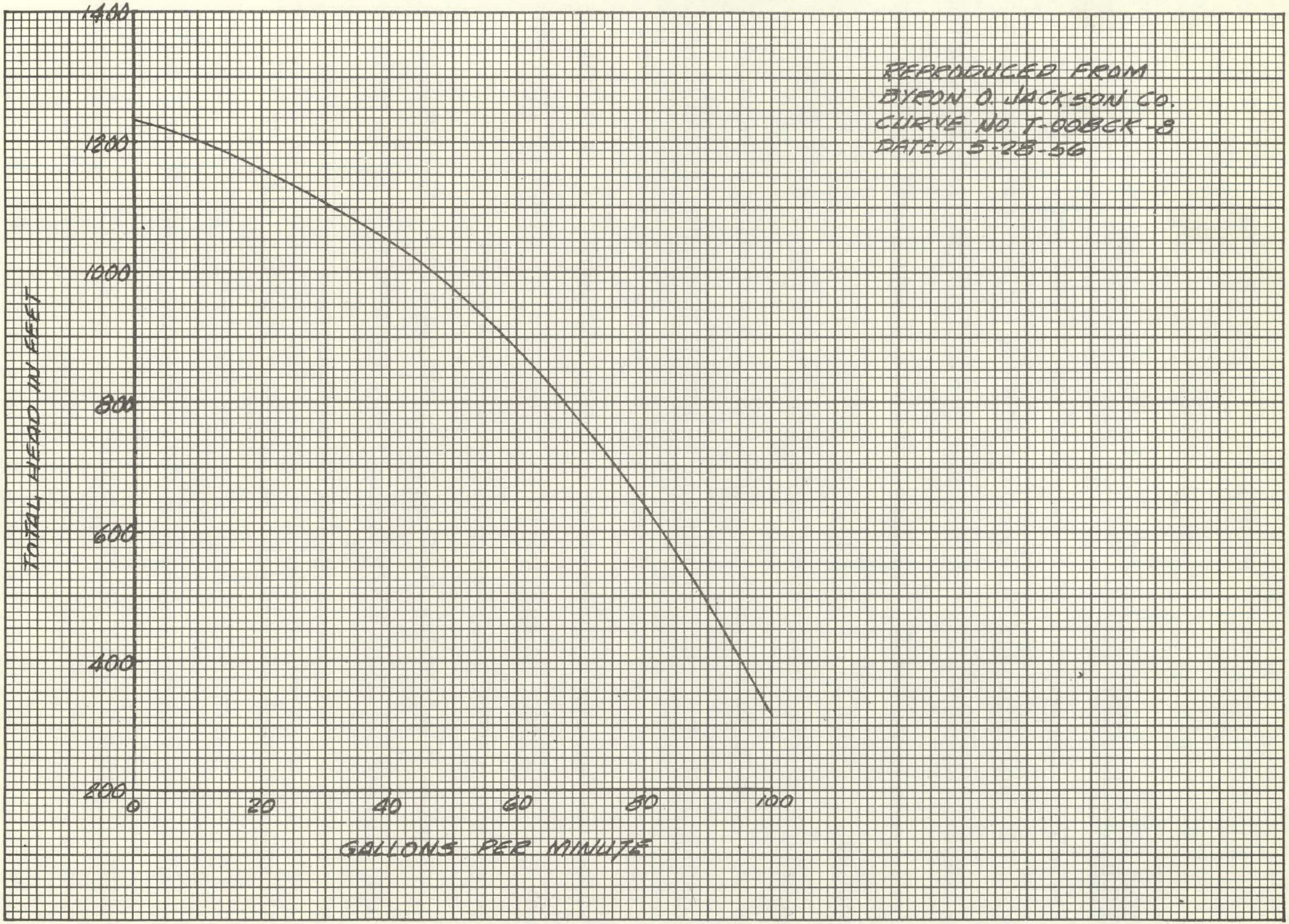


FIG. VIII - 12 BOILER FEED PUMP PERFORMANCE CURVE

CHAPTER IX - RESPONSE TO LOAD CHANGES

(T. M. Silks)

1. Stability of the APPR-1 (J. L. Meem)

The APPR-1 has proven to be an extremely stable power plant. So far as is known, it has demonstrated that the pressurized water nuclear power plant has better response to load demand than any other type of electrical generating plant now in operation in the world. This important characteristic cannot be over-emphasized.

It had been planned to subject the APPR-1 to a number of severe load transients. On April 20 the reactor first reached its full power rating of 10 megawatts of heat, and shortly thereafter a faulty circuit breaker connecting the APPR-1 to VEPCO opened. The plant output dropped from full load to station load smoothly and without any adjustment of the reactor controls. The plant performance was so smooth that it was about a minute before the operators realized that the plant was disconnected from VEPCO and was operating on its own power.

This sudden drop to station load was performed many times with the APPR-1 and it was decided to subject it to a complete loss of load by tripping the turbine off the line. This transient is described in this chapter.

On load increase it was found that the reactor would follow a load as fast as the load limiter on the turbine generator would allow. Data is presented where the generator output was increased from 200 kw to 2000 kw in 70 seconds. No adjustment of any of the controls was necessary.

This chapter includes data on steady state conditions at various power levels followed by descriptions of the transients following loss of load and increase in load mentioned above.

2. Steady State Conditions at Various Power Levels

Data were obtained by increasing the generator gross electrical output in approximately 100 KW increments from 200 KW to 2040 KW. Tables IX-1 through IX-4 present data taken during this series of power changes. The time between readings was approximately 20 minutes which insured that nearly steady state had been achieved after each transient. The data are summarized in the Figures which are described below:

2.1 Figure IX-1 Electrical Output versus Reactor Heat Output

The gross electrical generator output was increased in approximately 100 KW increments as measured by the generator wattmeter. The reactor power was calculated from the primary coolant flow rate and the primary coolant temperature increase across the reactor. At 1000 KW, 1600 KW and 2000 KW the evaporators were put in operation, readings taken and the evaporators shut down.

The points taken with the evaporators not in operation fall reasonably well on a straight line and indicate that the no electrical load plant losses are about 1 MW heat.

The points taken when the evaporators were in operation indicate that evaporator operation requires about 0.5 MW heat.

2.2 Figure IX-2 Secondary Steam Pressure versus Secondary Steam Temperature at Various Steady State Power Levels

As the reactor power level was raised, the measured secondary steam temperature and pressure was recorded. These quantities are plotted for the different power levels. It is seen that the steam in the steam generator remains

TABLE IX - 1 - STEADY STATE OPERATING DATA

REACTOR POWER	TURBINE		GRAPHIC PANEL MISCELLANEOUS DATA					CONDENSER CIRCULATING WATER			
	MAIN STEAM PRESSURE	MAIN STEAM TEMPERATURE	SECONDARY STEAM FLOW	FEEDWATER FLOW	SECONDARY STEAM TEMPERATURE	SECONDARY STEAM PRESSURE	CONDENSER PRESSURE	OUTLET TEMPERATURE RIGHT SIDE	OUTLET TEMPERATURE LEFT SIDE	INLET TEMPERATURE RIGHT SIDE	INLET TEMPERATURE
MW	PSIG	°F	LB/HR	LB/HR	°F	PSIG	In. Hg	°F	°F	°F	°F
1.82	193	387	5,000	3,000	445	340	28.5	75	73	71	71
2.40	193	389	7,000	5,000	440	322	29.0	76	73	71	71
2.66	193	389	7,800	7,000	438	310	29.2	76	74	71	71
3.23	193	390	10,300	9,000	438	305	29.0	74	74	71	71
3.59	193	390	11,600	10,200	438	302	29.0	77	75	71	71
4.06	193	393	13,000	11,500	438	300	29.0	77	75	71	71
4.42	193	397	14,400	12,500	438	290	29.0	77	76	72	72
4.79	193	399	15,500	14,000	438	288	29.0	78	77	72	72
5.31	193	402	17,200	15,000	435	280	29.0	79	78	72	72
5.77	193	402	19,000	17,500	435	270	29.0	79	78	72	72
5.72	193	403	18,200	17,000	435	270	29.0	78	78	72	72
6.24	193	406	20,300	18,000	435	265	29.0	79	79	72	72
6.55	190	408	21,300	19,700	434	260	29.0	80	79	72	72
6.97	190	409	23,000	21,000	434	255	29.0	81	79	72	72
7.28	190	410	24,000	22,000	433	250	29.0	81	80	72	72
8.23	190	412	27,000	25,000	430	230	29.0	82	81	71	71
7.81	190	412	25,500	24,500	430	240	29.0	81	80	71	71
8.22	190	412	27,000	25,000	430	232	29.0	81	79	71	70
8.64	190	412	28,200	27,000	430	230	29.0	81	81	71	70
8.96	190	412	29,700	27,400	430	220	29.0	81	81	71	70
9.53	190	415	31,100	29,500	429	208	29.0	82	82	71	71
10.10	190	415	33,200	31,200	429	208	29.0	82	82	71	71

TABLE IX - 2 - STEADY STATE OPERATING DATA

REACTOR POWER	FEEDWATER HEATER					EVAPORATOR			AIR EJECTOR		
	FEEDWATER OUTLET TEMPERATURE	FEEDWATER INLET TEMPERATURE	CONDENSATE OUTLET TEMPERATURE	STEAM INLET TEMPERATURE	STEAM INLET PRESSURE	VAPOR OUTLET TEMPERATURE	VAPOR OUTLET PRESSURE	STEAM INLET PRESSURE	FEEDWATER OUTLET TEMPERATURE	FEEDWATER INLET TEMPERATURE	STEAM INLET PRESSURE
MW	°F	°F	°F	°F	PSIG	°F	PSIG	PSIG	°F	°F	PSIG
1.82	165	128	135	175	0				128	82	150
2.40	175	135	128	175	0				135	80	150
2.66	182	125	125	182	0				125	82	150
3.23	187	114	125	189	0				114	82	150
3.59	195	114	122	195	0				114	84	150
4.06	200	109	122	200	0				109	84	150
4.42	205	109	125	206	0				109	85	150
4.79	210	107	125	210	0				107	87	150
5.31	215	107	128	215	0				107	87	150
5.77	237	104	165	237	0	256	17.5	193	104	87	150
5.72	218	105	135	220	0				105	87	150
6.24	224	106	135	224	0				106	87	150
6.55	226	105	140	227	0				105	87	150
6.97	230	104	140	231	0				104	87	150
7.28	235	105	145	235	0				105	87	150
8.23	240	104	163	240	10	256	17.5	190	104	92	150
7.81	238	104	148	238	10				104	92	150
8.22	240	104	150	240	10				104	92	150
8.64	243	104	152	243	10.5				104	92	150
8.96	245	105	155	245	12				105	94	150
9.53	247	105	157	247	14				105	95	150
10.10	247	104	165	247	14	257	17.5	190	104	95	150

TABLE IX - 3 - STEADY STATE OPERATING DATA

REACTOR POWER	FEED PUMP	ROD POSITIONS							GRAPHIC PANEL		
	CONDENSATE OUTLET PRESSURE	SHIM ROD #4	SHIM ROD #3	SHIM ROD #2	SHIM ROD #1	REGULATING ROD C	SAFETY ROD B	SAFETY ROD A	POWER INTEGRATOR	PRIMARY COOLANT BLOWDOWN	PRIMARY COOLANT WATER FLOW
MW	PSIG	In.	In.	In.	In.	In.	In.	In.	^{o}F DAYS	GPM	GPM
1.82	470	7.92	7.92	7.92	7.92	7.92	20.00	20.00	170.60	1.12	3850
2.40	470	7.95	7.95	7.95	7.95	7.95	20.00	20.00	170.60	1.12	3850
2.66	470	8.00	8.00	8.00	8.00	8.00	20.00	20.00	170.78	1.09	3850
3.23	460	8.02	8.02	8.02	8.02	8.03	20.00	20.00	170.84	1.09	3850
3.59	455	8.07	8.07	8.07	8.07	8.06	20.00	20.00	170.98	1.08	3850
4.06	450	8.10	8.10	8.10	8.10	8.10	20.00	20.00	171.09	1.20	3850
4.42	450	8.14	8.14	8.14	8.14	8.14	20.00	20.00	171.22	1.18	3850
4.79	440	8.17	8.17	8.17	8.17	8.17	20.00	20.00	171.54	1.18	3850
5.31	430	8.17	8.17	8.17	8.17	8.17	20.00	20.00	171.67	1.15	3850
5.77	405	8.17	8.17	8.17	8.17	8.17	20.00	20.00	171.99	1.08	3850
5.72	430	8.17	8.17	8.17	8.17	8.17	20.00	20.00	172.15	1.05	3850
6.24	415	8.17	8.17	8.17	8.17	8.17	20.00	20.00	172.32	1.10	3850
6.55	400	8.17	8.17	8.17	8.17	8.17	20.00	20.00	172.51	1.10	3850
6.97	400	8.17	8.17	8.17	8.17	8.17	20.00	20.00	172.71	1.10	3850
7.28	390	8.17	8.17	8.17	8.17	8.17	20.00	20.00	172.89	1.10	3850
8.23	350	8.12	8.12	8.12	8.11	8.11	20.00	20.00	173.18	1.10	3850
7.81	370	8.12	8.12	8.12	8.11	8.11	20.00	20.00	173.42	1.10	3850
8.22	375	8.12	8.12	8.12	8.11	8.11	20.00	20.00	173.72	1.10	3850
8.64	350	8.12	8.12	8.12	8.11	8.11	20.00	20.00	173.86	1.10	3850
8.96	350	8.10	8.09	8.10	8.10	8.09	20.00	20.00	174.14	1.10	3850
9.53	330	8.10	8.09	8.10	8.10	8.09	20.00	20.00	174.44	1.10	3850
10.10	300	8.10	8.09	8.10	8.10	8.09	20.00	20.00	174.70	1.10	3850

TABLE IX - 4 - STEADY STATE OPERATING DATA

REACTOR POWER	REACTOR OUTLET TEMPERATURE (NUCLEAR PANEL)	REACTOR T (GRAPHIC PANEL)	PRIMARY PRESSURE (PRESSURIZER)	GROSS ELECTRICAL POWER	NET ELECTRICAL POWER
MW	°F	°F	PSIG	KW	KW
1.82	441	3.5	1210	200	25
2.40	439	4.6	1210	300	125
2.66	441	5.1	1210	400	200
3.23	439	6.2	1210	500	325
3.59	442	6.9	1210	600	400
4.06	442	7.8	1210	700	490
4.42	443	8.5	1210	800	610
4.79	444	9.2	1210	900	680
5.31	443	10.2	1210	1000	780
5.77	443	11.1	1210	1000	780
5.72	444	11.0	1210	1100	900
6.24	444	12.0	1210	1200	1025
6.55	444	12.6	1210	1300	1080
6.97	446	13.4	1210	1400	1200
7.28	447	14.0	1210	1500	1300
8.23	445	15.8	1210	1600	1380
7.81	446	15.0	1210	1600	1400
8.22	447	15.8	1210	1720	1500
8.64	447	16.6	1200	1830	1600
8.96	447	17.2	1210	1920	1680
9.53	448	18.3	1200	2040	1800
10.10	449	19.4	1200	2040	1800

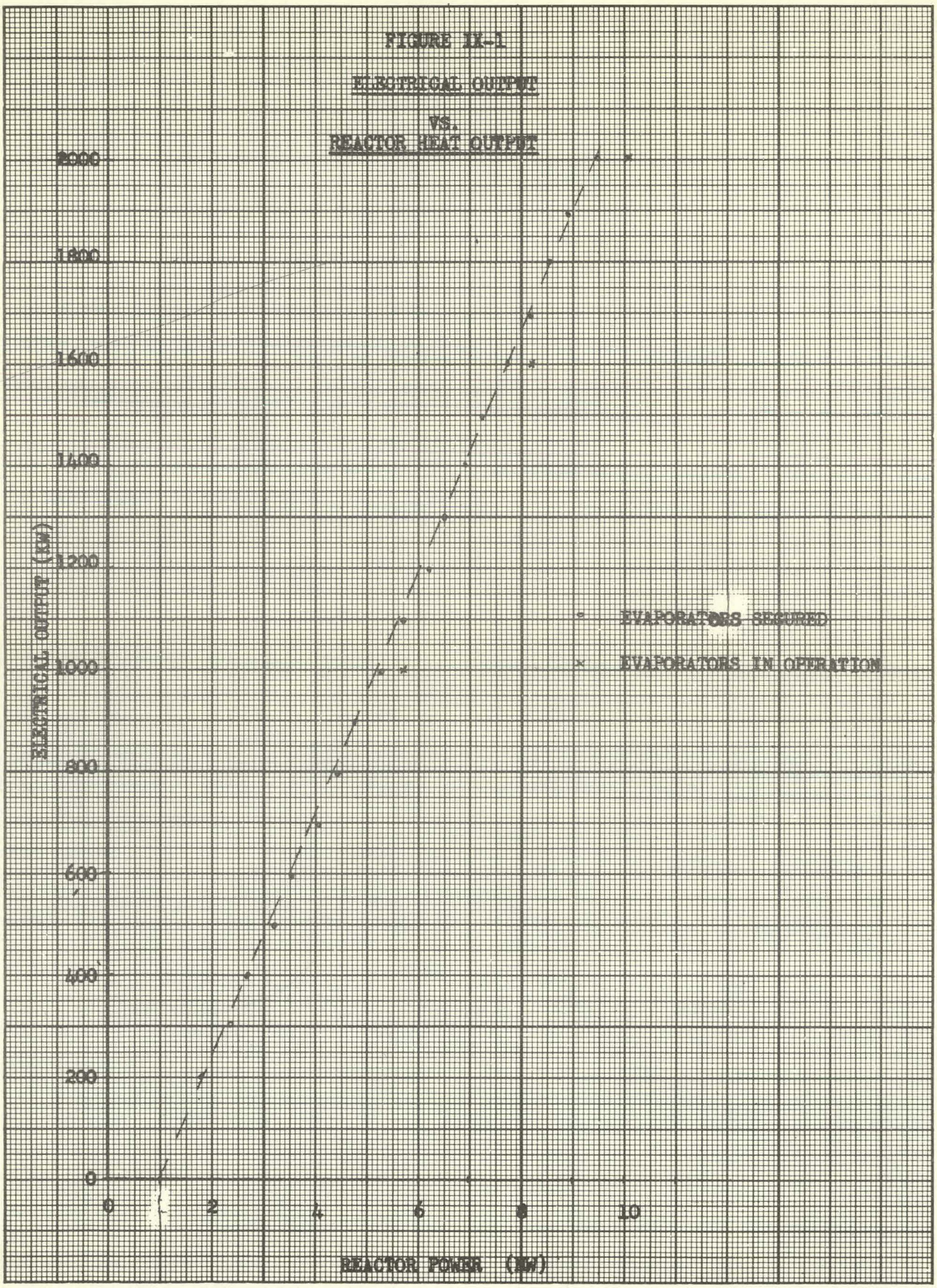


FIG. IX - 1 ELECTRICAL OUTPUT VS. REACTOR HEAT OUTPUT

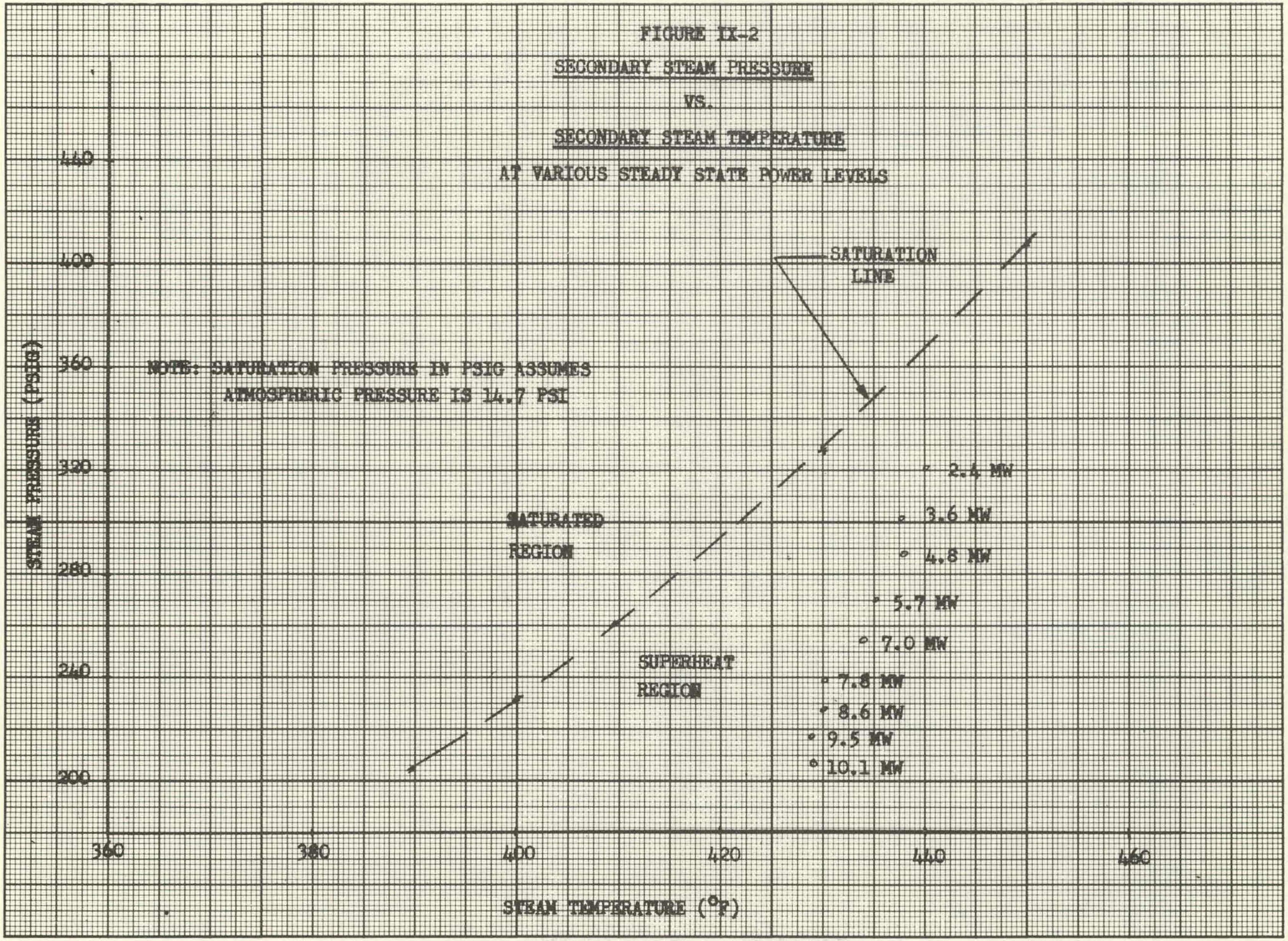


FIG. IX - 2 SECONDARY STEAM PRESSURE VERSUS SECONDARY STEAM TEMPERATURE AT VARIOUS STEADY STATE POWER LEVELS

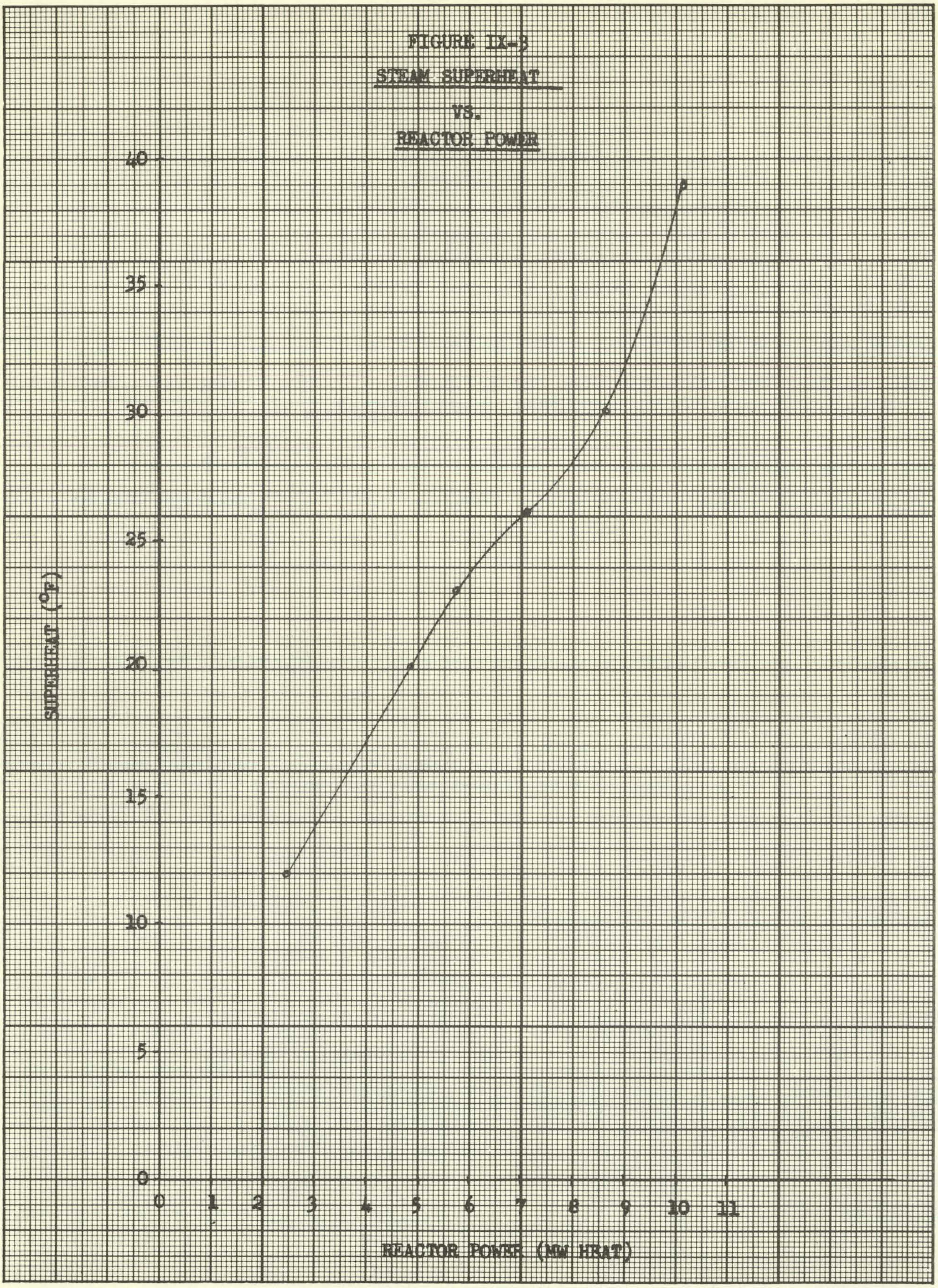


FIG. IX - 3 STEAM SUPERHEAT VS. REACTOR POWER

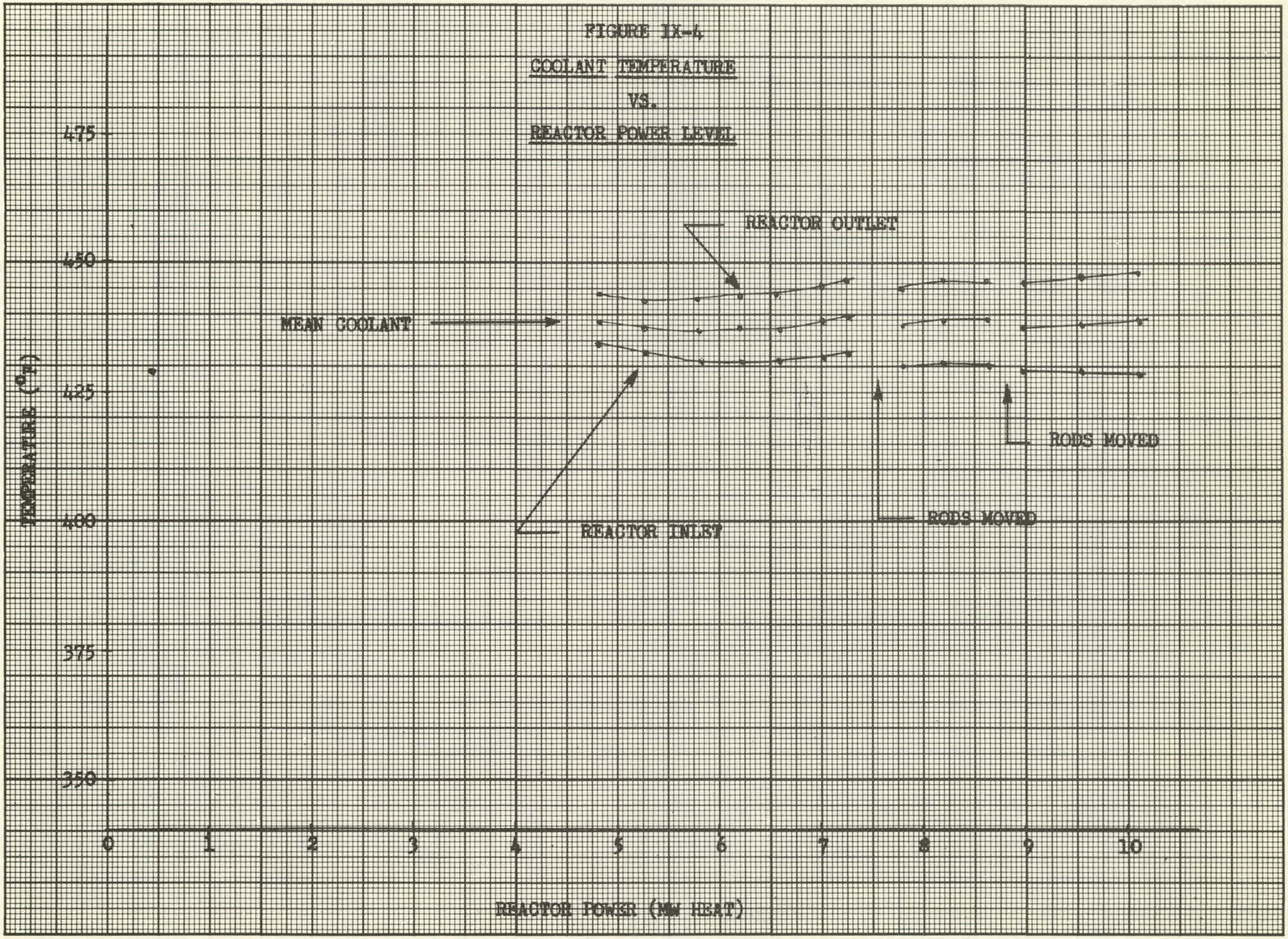


FIG. IX - 4 COOLANT TEMPERATURES VS. REACTOR POWER

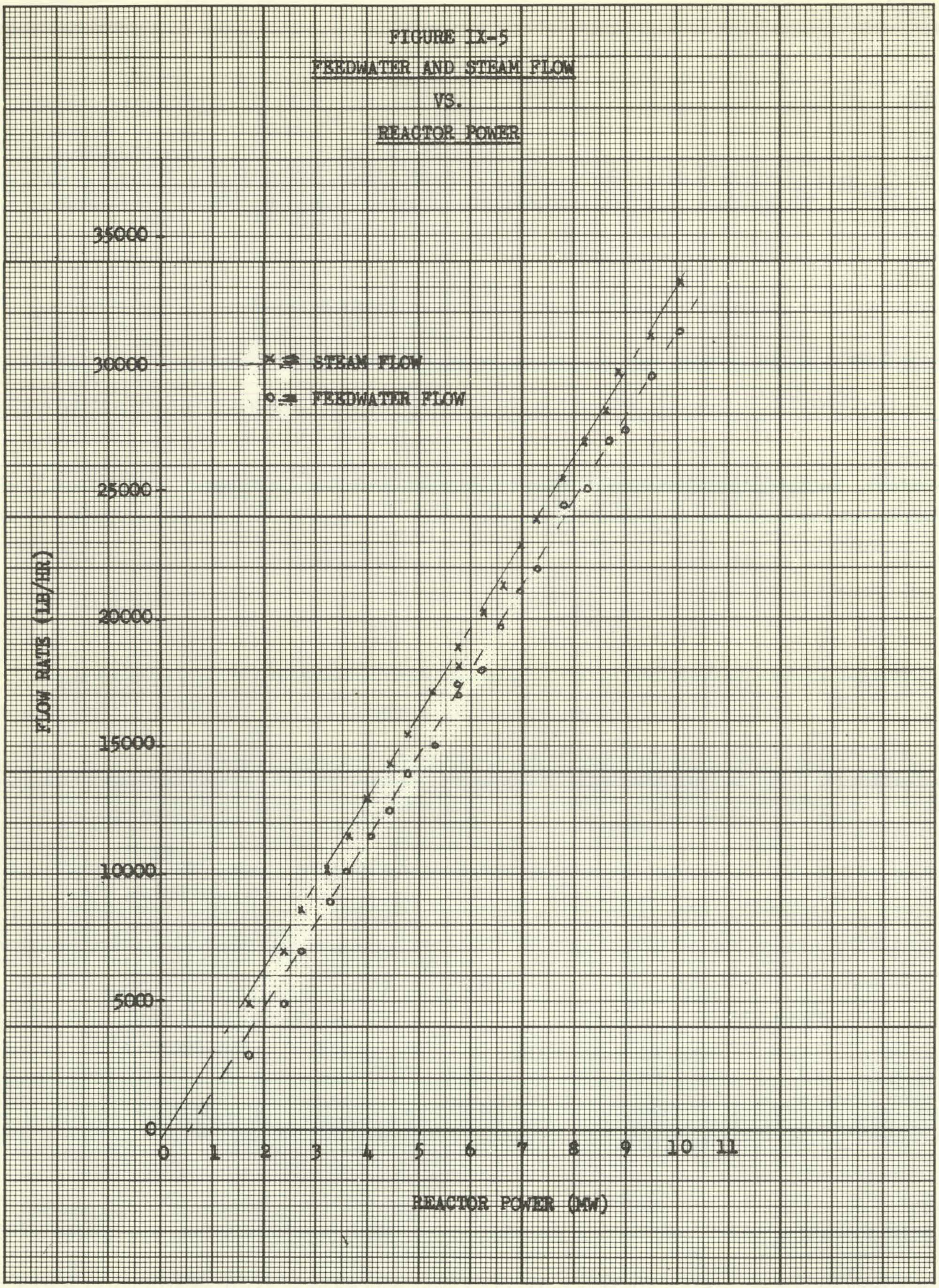


FIG. IX - 5 FEEDWATER AND STEAM FLOW VS. REACTOR POWER everywhere in the superheat region.

2.3 Figure IX-3 Steam Superheat versus Reactor Power

This is a plot from the same data as Figure IX-2. Here, the increase in steam superheat with increasing reactor power is shown more clearly.

2.4 Figure IX-4 Coolant Temperatures versus Reactor Power

The reactor outlet temperature and the temperature difference across the reactor were measured quantities. From these two temperatures the reactor inlet and mean coolant temperatures were calculated. The reactor power was determined as above.

Only values for power levels greater than about 5 MW are presented. Points on the plot joined by a curve are data obtained with rods not moved. At power levels less than 5 MW control rods were moved slightly after nearly every reading.

At about 7.25 MW rods 1 and C were inserted 0.06" and rods 2, 3 and 4 were inserted 0.05".

At about 8.6 MW rods C, 2 and 4 were inserted 0.02", rod 1 inserted 0.01" and rod 3 inserted 0.03".

These data indicate that very little operator adjustment to the plant was required by the changes in power demand. It is shown in later sections that if the control rods are positioned at full load, the plant power level can be changed to any other level and returned to full power with no rod adjustment.

2.5 Figure IX-5 Feedwater and Steam Flow versus Reactor Power

This is a plot of the measured steam and feedwater flows against calculated power level. It is apparent that at least one of the readings is in error since the feedwater and steam flow nowhere are equal. It appears that both are wrong since a straight line through either group of points does not go through the origin.

These conclusions are in agreement with those of Section 8 which were reached from heat balance considerations.

3. Transient Response to a Sudden Loss of Load

The loss of load transient was accomplished by tripping the turbine throttle valve when the reactor was at 10.6 MW heat steady state output. Earlier transients which were initiated by opening the electrical breakers are not reported since they are less severe than the reported transient.

Tripping the throttle valve removed all load from the steam generator with the exception of the turbine driven lube oil pump. This pump goes into operation whenever the turbine lube oil pressure drops to a specified value and will be present whenever this transient occurs. The loss of load transient reported here is the most severe that can be imposed on the plant without damage to secondary equipment.

3.1 Figure IX-6 Log N Chart Loss of Load Transient Test

Closing the throttle valve resulted in the loss of a heat sink for the primary circuit. Since the reactor continued to produce power for a period of time, the primary circuit temperatures rose and the reactor power output was increased by the effect of the negative temperature coefficient of reactivity.

Heat losses ultimately caused the primary circuit to drop in temperature. As the primary circuit temperature dropped, the reactivity of the core increased due to the effect of the temperature coefficient of reactivity. The reactivity increase resulted in an increase in power and hence primary system temperature.

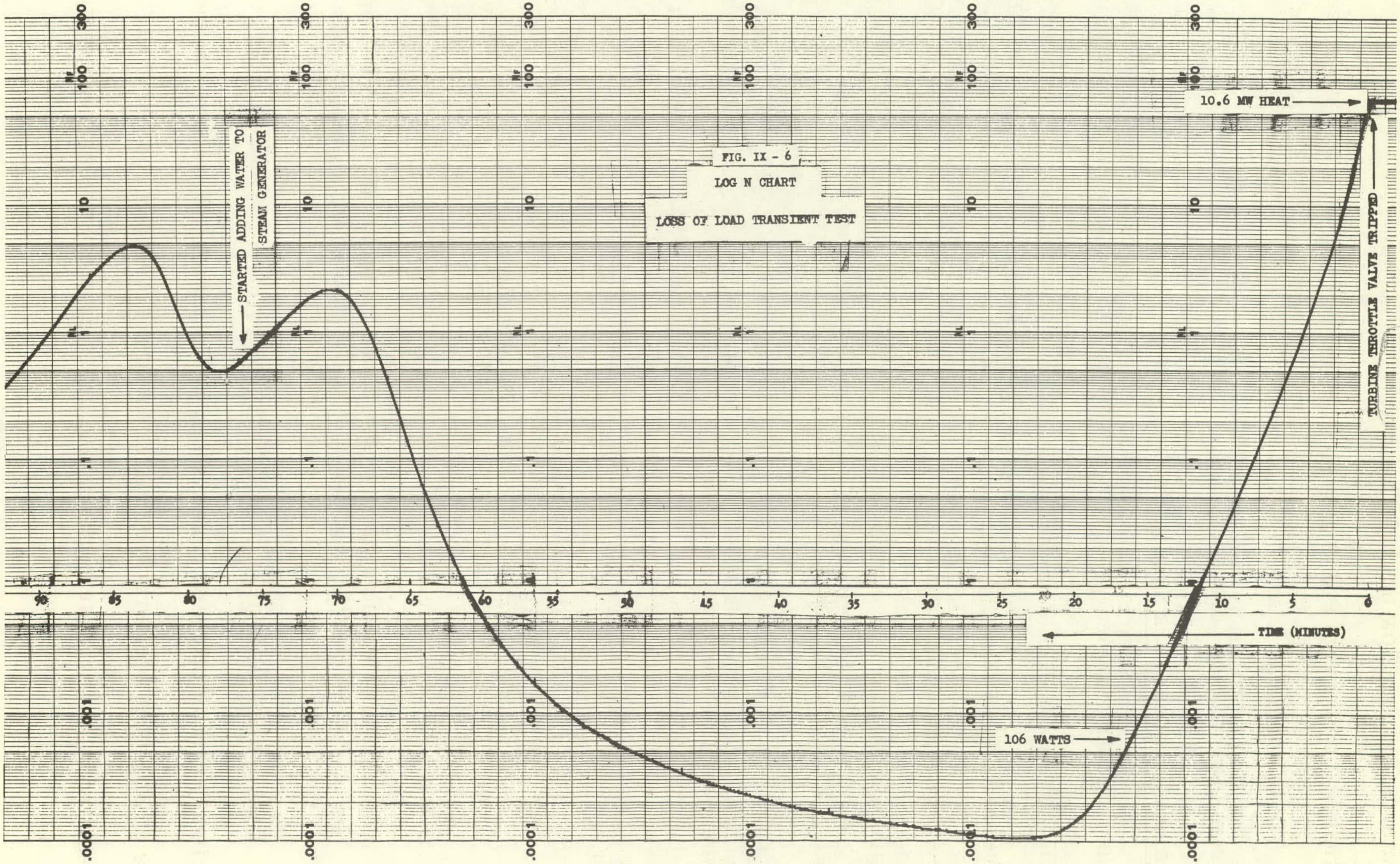


FIG. IX - 6 LOG N CHART, LOSS OF LOAD TRANSIENT TEST

The increasing primary temperature gain decreased the reactivity and diminished the power output.

At approximately 76 minutes, water was added to the steam generator to calibrate the steam generator level indicator. The added water was relatively cold and lowered the primary system temperature causing the reactor power increase. This increase in power output raised coolant temperature, again causing the power output to drop.

During this transient the reactor did not scram and the control rods were not moved. The plant went through the transient with no operator adjustment, being completely self-regulating.

3.2 Figure IX-7 Reactor Period Chart, Loss of Load Transient Test

This chart shows the reactor period during the sequence of events described above. There is considerable noise shown in this trace but the average value of the period during the steady state operation prior to the initiation of the transient is about - 200 seconds. The pen is therefore assumed to have been displaced about 1/8" from its correct position on the chart, since the reactor is on an infinite period at steady state operation. During the time 23 to 70 minutes the period is shown as negative while Figure IX-6 shows increasing power. This is attributed to the assumed displacement of the pen.

3.3 Figure IX-8 Reactor and Steam Generator Outlet Temperatures, Loss of Load Transient Test

This chart shows the reactor and steam generator outlet temperatures during the sequence of events described under Figure IX-6.

The temperatures are measured with resistance bulbs which have some time lag and were not designed for measuring transient temperatures. Since the

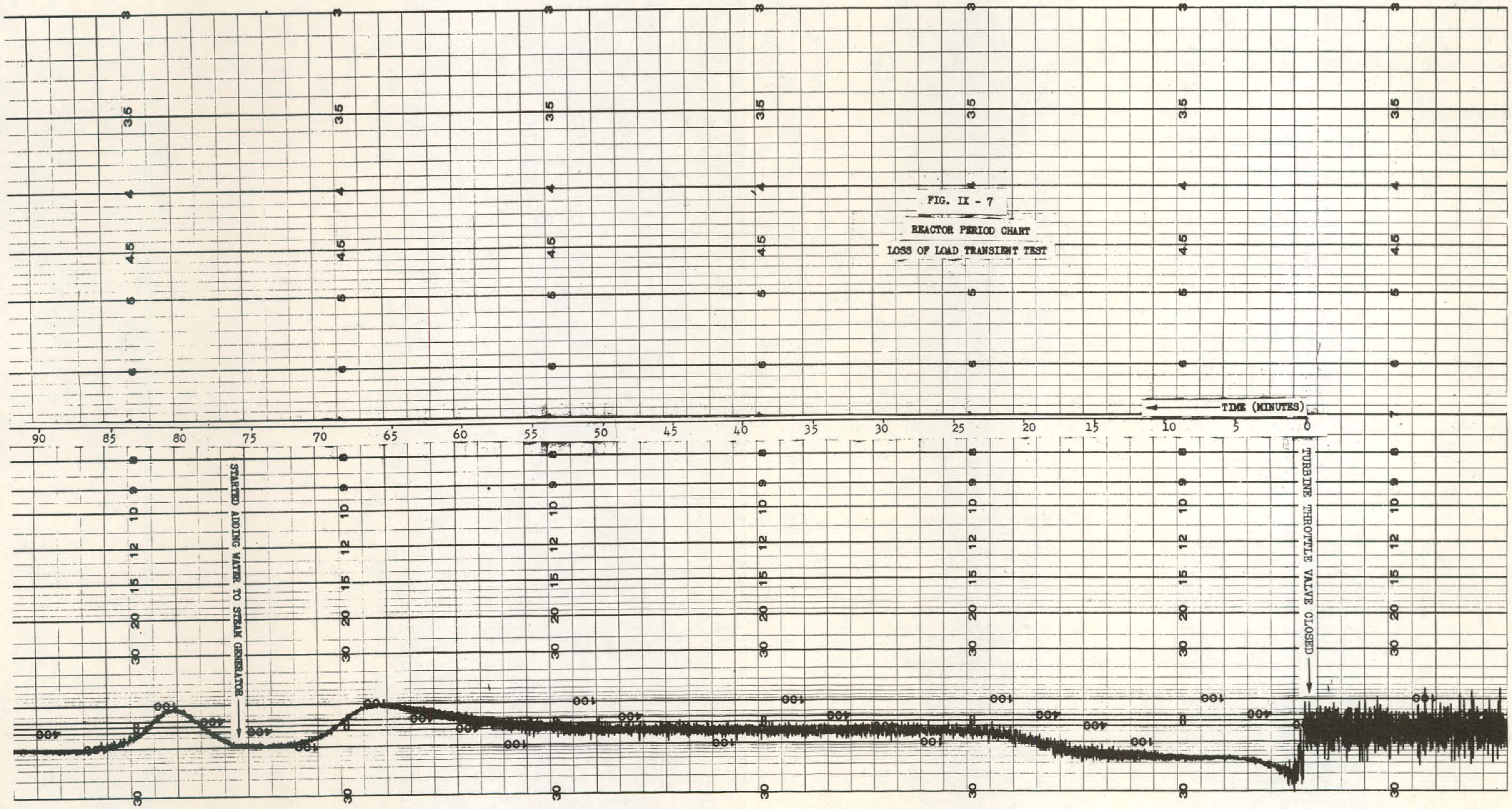


FIG. IX - 7 REACTOR PERIOD CHART, LOSS OF LOAD TRANSIENT TEST

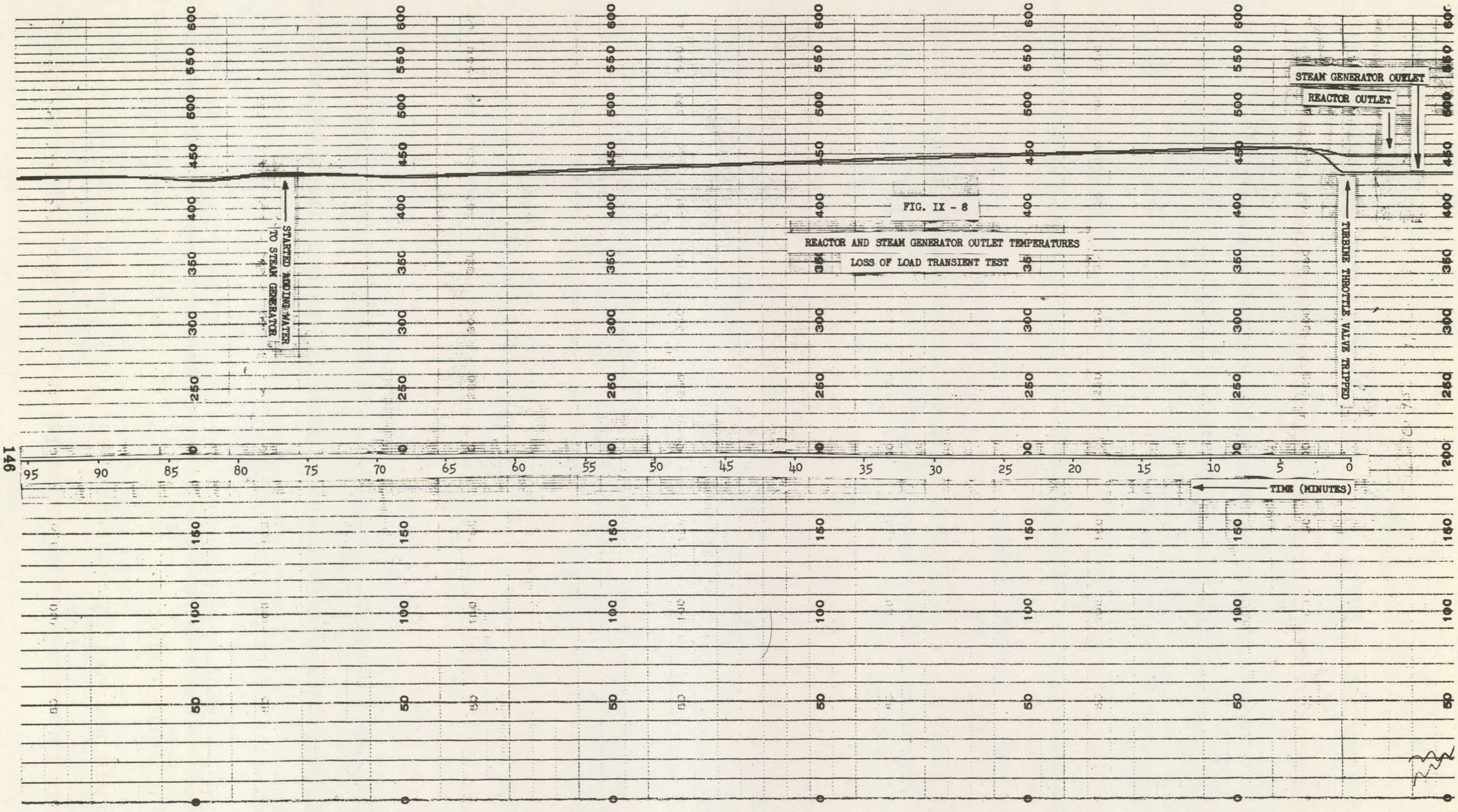


FIG. IX - 8 REACTOR AND STEAM GENERATOR OUTLET TEMPERATURE, LOSS OF LOAD TRANSIENT TEST chart extends from zero to 600°F, the readability is not good for high accuracy. This chart should therefore be regarded as furnishing qualitative information.

The temperature behavior is seen to conform well to that required to explain the Log N chart curve.

3.4 Figure IX-9 Reactor Δ T and Primary System Pressure versus Time

The chart speed on these recorders was 1 inch/hour and hence both the loss of load transient response and the increase of load transient response are presented on one curve.

Since the chart speed was slow, resolution of the curves along the time axis is poor. Maximum values of variables after a transient can be seen, however. The portion of the curves of interest for the loss of load transient is the first 60 minutes marked on the graph.

The reactor Δ T dropped to essentially zero at loss of load and remained at this value until water was added to the steam generator.

The primary system is pressurized by steam generated by electric heaters in the pressurizer. When the steam pressure drops to 1200 psig, the heaters go on and when the pressure rises to about 1225 psig, they are shut off. This accounts for the pressure cycling previous to load loss transient. After the loss of load the pressure rose from about 1210 psig to about 1300 psig or about 90 psig. Thereafter the pressure dropped until the heaters again went into operation at 1200 psig. The pressure reached after this severe loss of load transient was well below the design pressure of 1600 psia.

3.5 Figure IX-10 Steam Pressure and Steam Temperature versus Time

Immediately after the loss of load, the pressure in the steam generator rose to 435 psig and steadily decreased thereafter. The pressure relief valve in

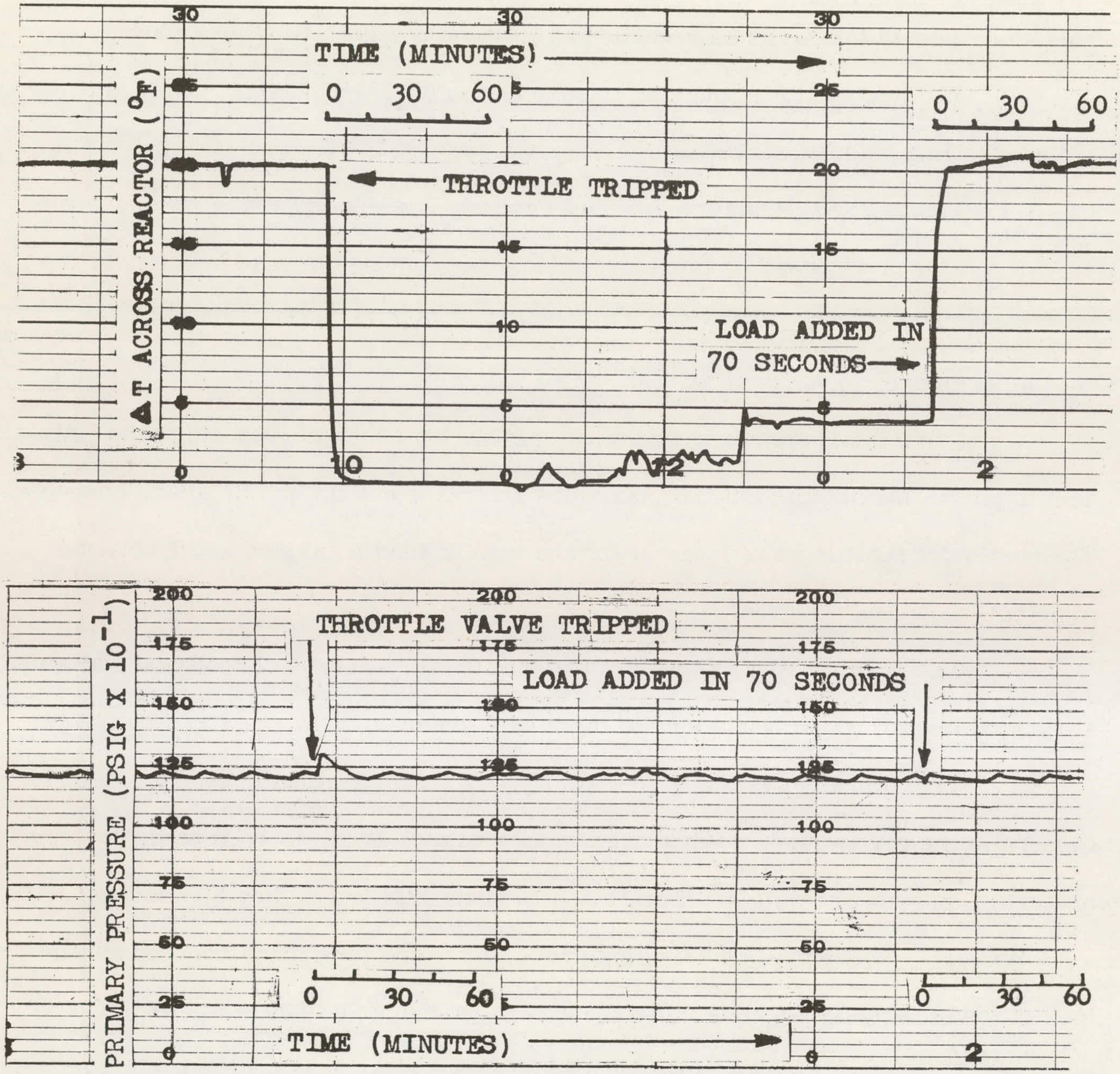


FIG. IX - 9 REACTOR ΔT AND PRIMARY SYSTEM PRESSURE VS. TIME

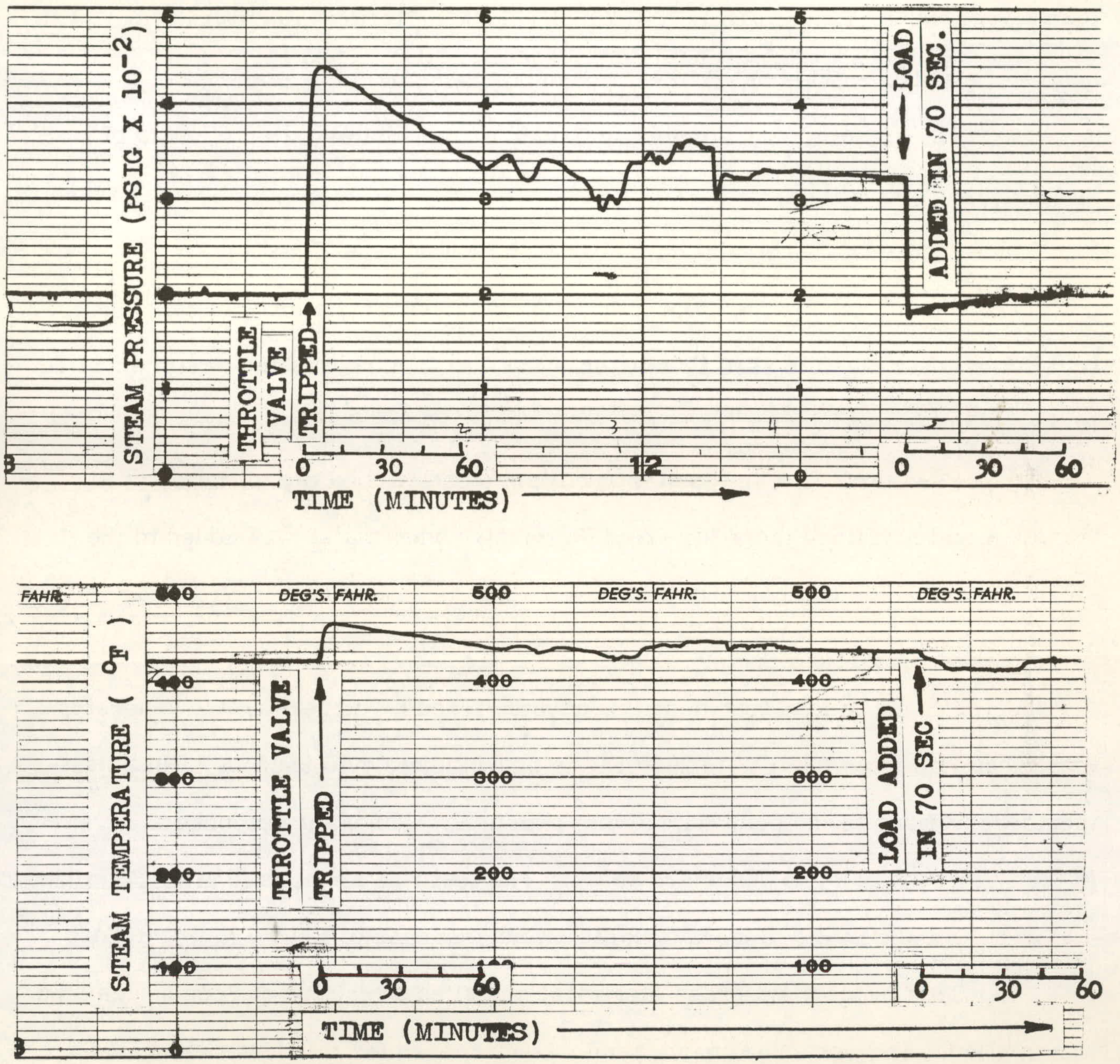


FIG. IX - 10 STEAM PRESSURE AND STEAM TEMPERATURE VS. TIME the secondary lifted briefly at the peak pressure shortly after the loss of load as had been expected. This, of course, does not appear on the trace.

The steam temperature rose to $460^{\circ}F$ immediately after the loss of load and decreased steadily thereafter.

3.6 Figure IX-11 Steam and Feedwater Flow versus Time

Steam flow dropped to a low value quickly after the throttle valve was tripped and decreased to zero thereafter. The low flow represents steam supplied to the steam driven lube oil turbine.

Feedwater flow dropped to zero immediately upon the initiation of the transient and remained there for about 75 minutes when water was added to the steam generator by manual controls.

4 Transient Response to a Sudden Increase in Load

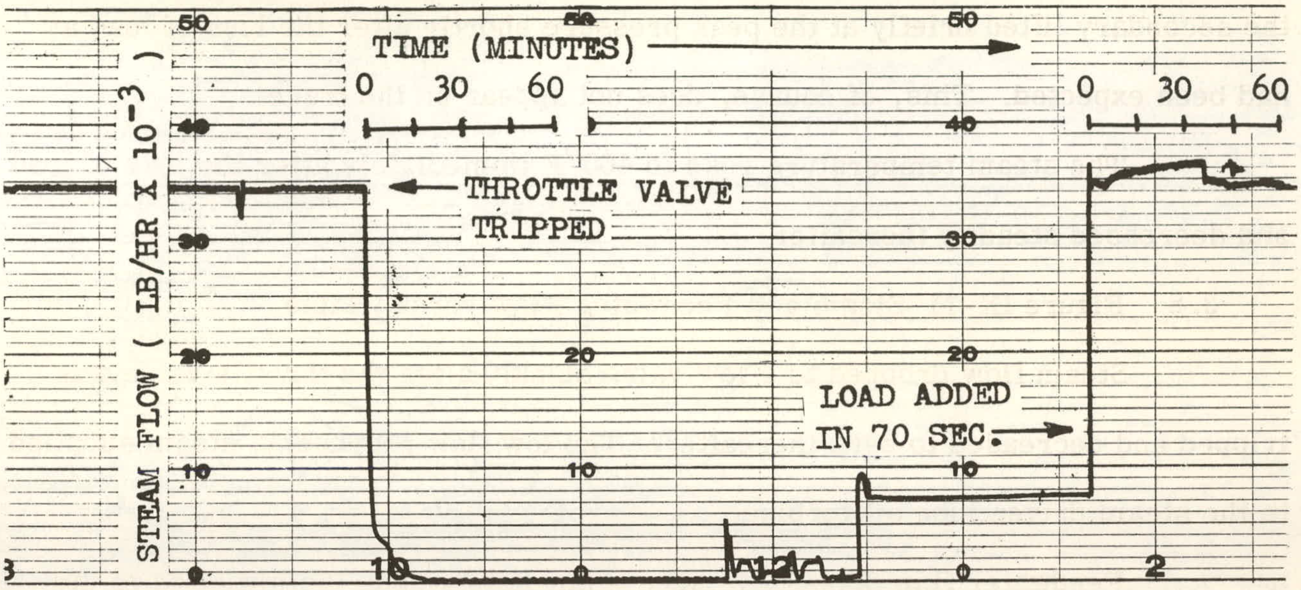
The increase in load transient was accomplished by increasing the generator electrical load from 225 KW to 2050 KW in approximately 75 seconds. No adjustments were made to the plant during this transient. The plant response is given in the Figures which are described below. The test was terminated after 30 minutes.

4.1 Figure IX-12 Generator Wattmeter, Increase of Load Transient Test

This is the plant gross electrical output as a function of time as the load was applied. The recorder chart speed was increased for this transient to show more clearly the initial application of load and hence only the first 120 seconds are presented.

4.2 Figure IX-13 Log N Chart, Increase of Load Transient Test

At the beginning of the transient the reactor heat output was about 2 MW, corresponding to the station electrical load of 225 KW. The power output of the



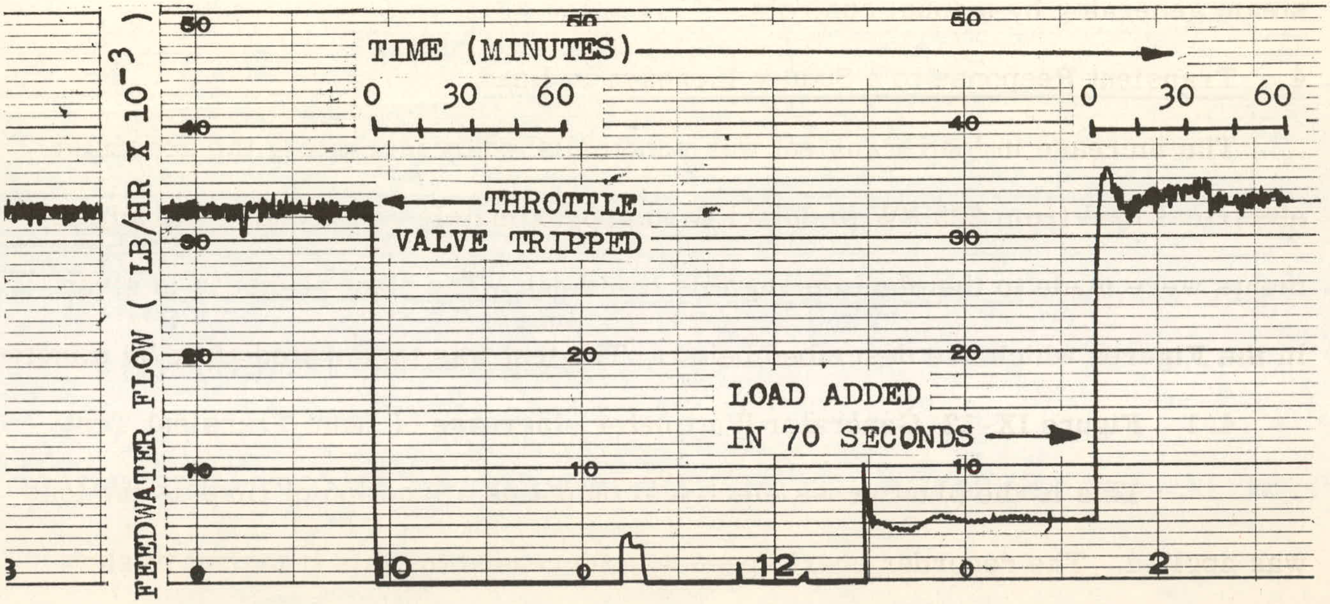


FIG. IX - 11 STEAM AND FEEDWATER FLOW VS. TIME

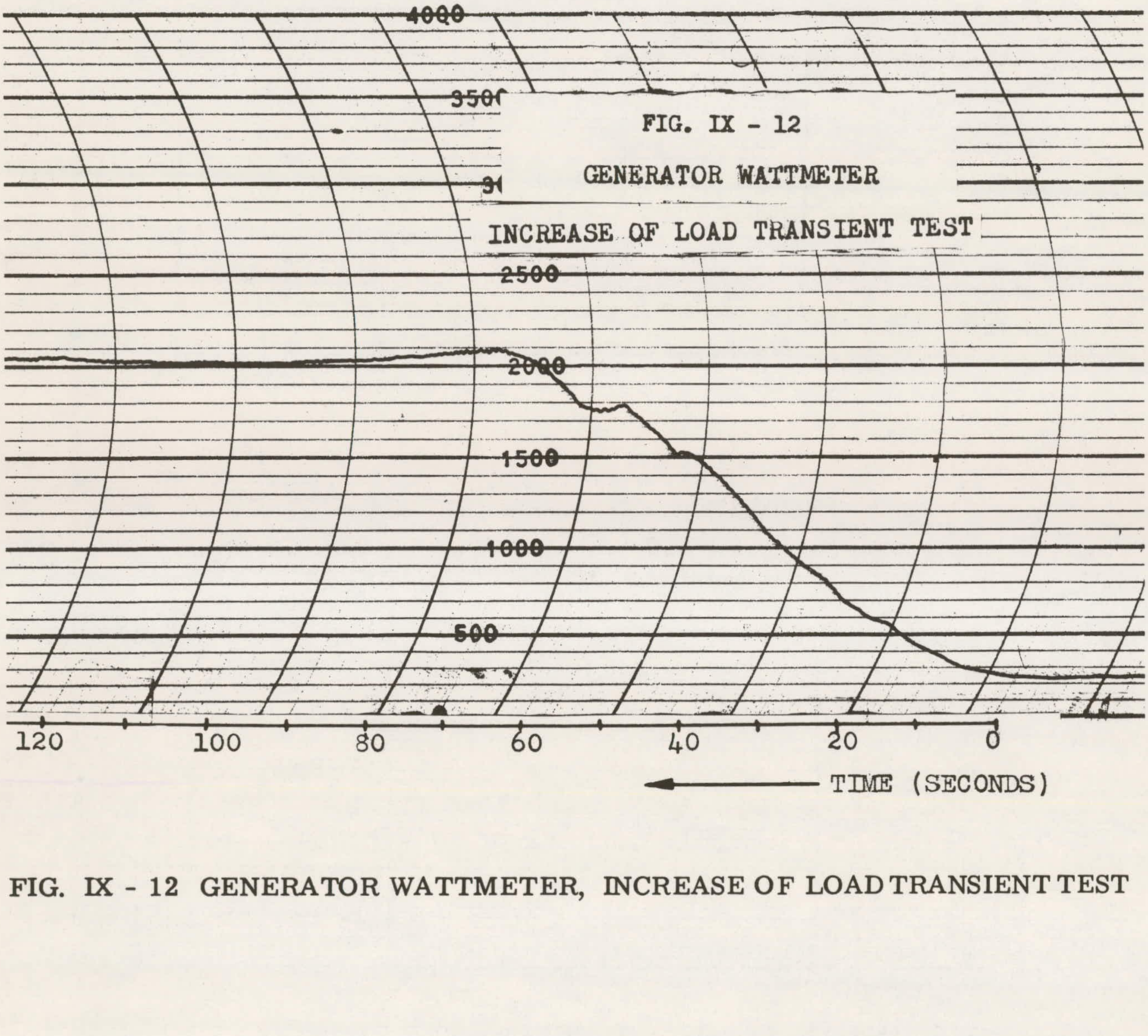


FIG. IX - 12 GENERATOR WATTMETER, INCREASE OF LOAD TRANSIENT TEST

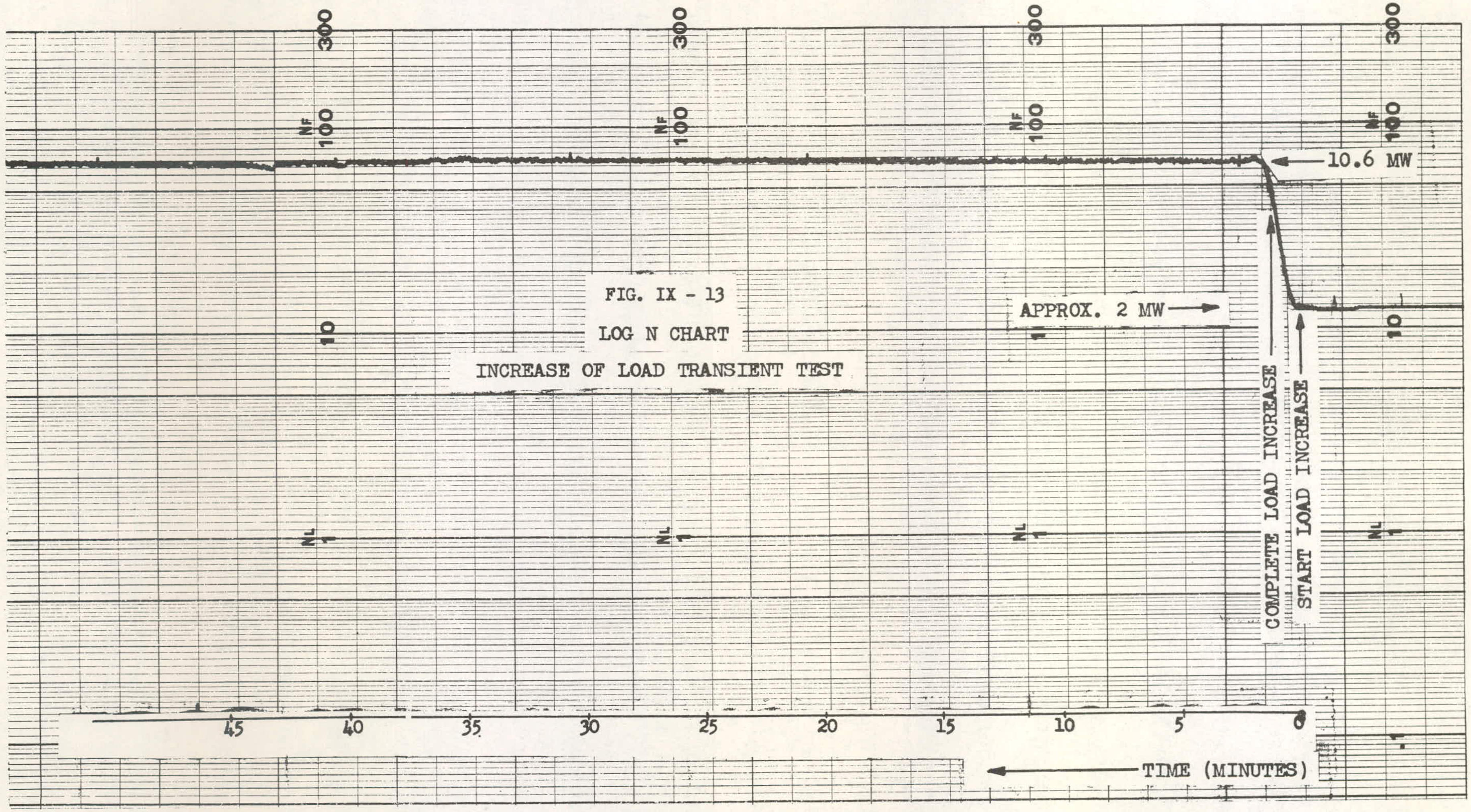


FIG. IX - 13 LOG N CHART, INCREASE OF LOAD TRANSIENT TEST reactor rose very nearly as fast as the plant demand and this curve indicates negligible power overshoot. Since this is logarithmic scale, the sensitivity in the higher decades is decreased. The reactor Δ T is a more sensitive detector of power overshoot.

4.3 Figure IX-14 Reactor Period Chart, Increase of Load Transient Test

As indicated earlier, it is felt that the pen on this recorder was displaced approximately 1/8" from its correct position. The chart indicates that the shortest period during this transient was 30 seconds but correcting the chart by the assumed pen displacement the shortest period was about 25 seconds. This is well above the scram setting of 3 seconds.

4.4 Figure IX-15 Reactor and Steam Generator Outlet Temperatures, Increase of Load Transient Test

Earlier comments regarding steady state and transient accuracy of this chart apply here. The temperature behaviors are in general as expected, however.

Initially the reactor inlet temperature dropped sharply, causing an increase in reactor power output. Thereafter both reactor inlet and outlet temperatures rose until a new equilibrium average temperature was established and the temperatures then held substantially constant. The 10.6 MW equilibrium average temperature is indicated on this chart as about 3^{O}F higher than the 2 MW equilibrium average temperature.

4.5 Figure IX-9 Reactor ΔT and Primary System Pressure versus Time

The portion of the reactor ΔT and primary pressure curves pertaining to the increase in load transient is the 30 minutes following the load addition as marked.

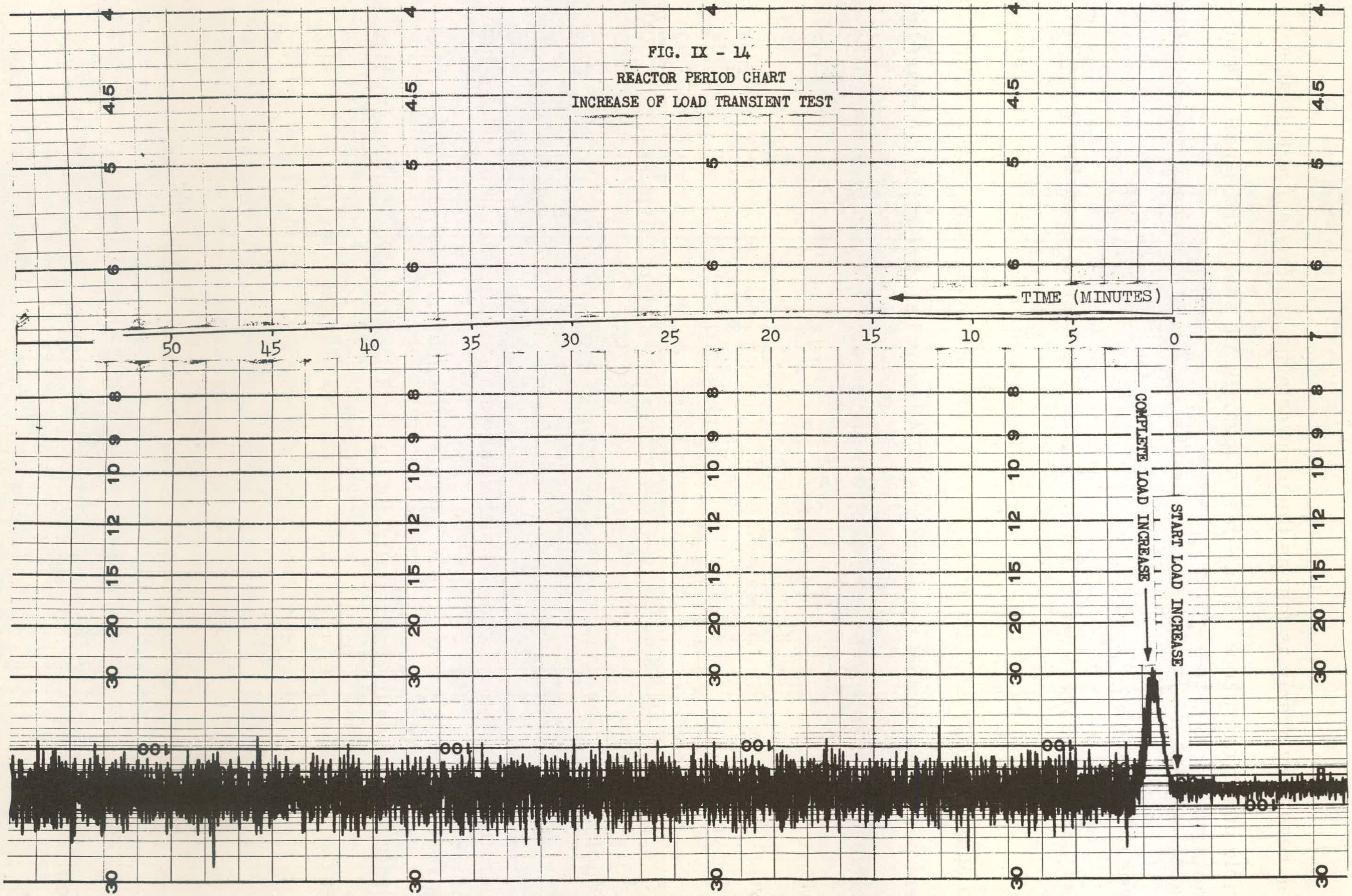


FIG. IX - 14 REACTOR PERIOD CHART, INCREASE OF LOAD TRANSIENT TEST

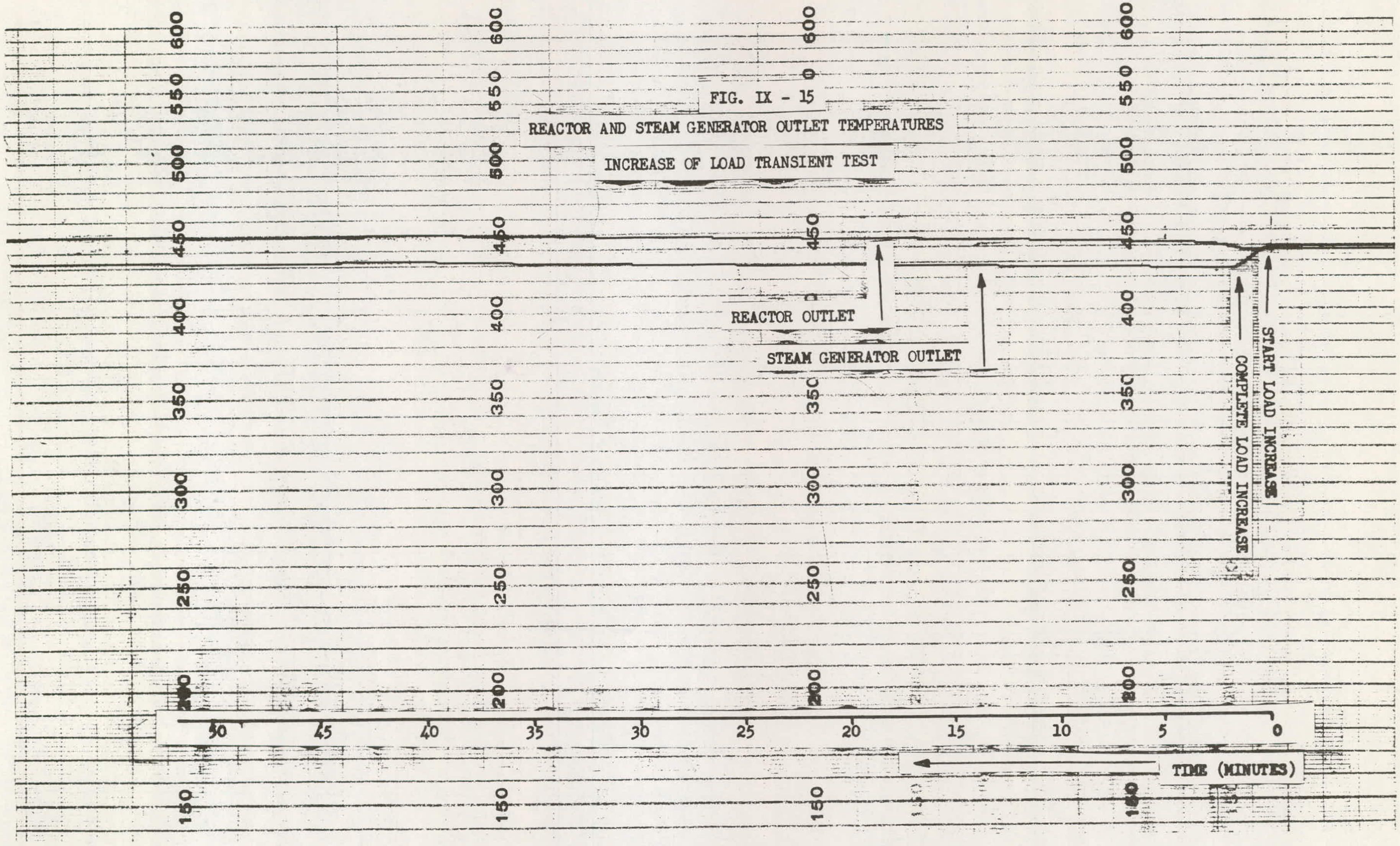


FIG. IX - 15 REACTOR AND STEAM GENERATOR OUTLET TEMPERATURES
INCREASE OF LOAD TRANSIENT TEST

Following the load addition the reactor Δ T increased rapidly to 20^{o}F and thereafter slowly increased to 21^{o}F. This corresponds to the gradually increasing steam flow evident in Figure IX-11.

Primary pressure dropped approximately 20 psi immediately after the initiation of the transient.

Since the curve dropped to 1200 psig upon load addition and remained at that value until it started to rise again, it appears that the pressurizer heaters added steam to the pressurizer at about the same rate as water was being withdrawn from the pressurizer during this short interval of time.

Since the pressure did not drop below 1200 psig (about 1215 psia) there appears to be no danger of boiling in the reactor during such a transient.

4.6 Figure IX-10 Steam Pressure and Steam Temperature versus Time

The portion of the steam pressure and steam temperature curves pertaining to this transient is the 30 minutes following the load addition.

The steam pressure dropped from 322 psig to about 175 psig immediately after the transient and then rose to a new equilibrium value of about 200 psig.

The steam temperature dropped from 430^{o}F to 410^{o}F in about 10 minutes and maintained this value thereafter.

The steam remained superheated throughout this transient.

4.7 Figure IX-11 Steam and Feedwater Flow versus Time

The portion of the steam flow and feedwater flow curves pertaining to the increase in load transient is the 30 minutes following the load addition as marked.

The steam flow initially overshot slightly the required flow, dropped to a minimum and then steadily increased thereafter to meet the secondary system demand.

The feedwater flow is seen to follow rather closely the steam flow as required to maintain a constant water level in the steam generator.

CHAPTER X - XENON REACTIVITY EFFECTS

(F. B. Fairbanks and W. R. Johnson)

1. Procedure

During the 700-hour test, there was opportunity to make relatively clean measurements on xenon transients. At the start of the 700-hour test, the plant had been shut down for several weeks for revision of the steam generator, and all xenon fission product poisoning had decayed. The total core burnup was low enough (approximately 0.14 megawatt-years) that poisoning from other fission products was small.

As the plant was started up for the 700-hour test, it was necessary to check out the steam generator, and accordingly it was not possible to hold steady power for the first several hours of operation. After this period, however, the reactor power was held steady at 7.7 megawatts for 48 hours to allow the xenon to come to equilibrium. At the end of this time the power was reduced to 2.1 megawatts and held constant for 48 hours so as to observe the peak xenon transient. The power level during this period of time is shown in Figure X-1, and the corresponding shim bank position in Figure X-2. Another continuous power run at 10.4 megawatts was made later in the 700-hour test.

2. Results

The equilibrium bank positions obtained at the various power levels are summarized in Table X-1. The bank position shown is the average position of the four shim and the center rods, with the safety rods A and B at 20.00 inches.

The reactivity corresponding to these xenon levels was obtained by determining the difference between the total five rod bank worth at the xenon level

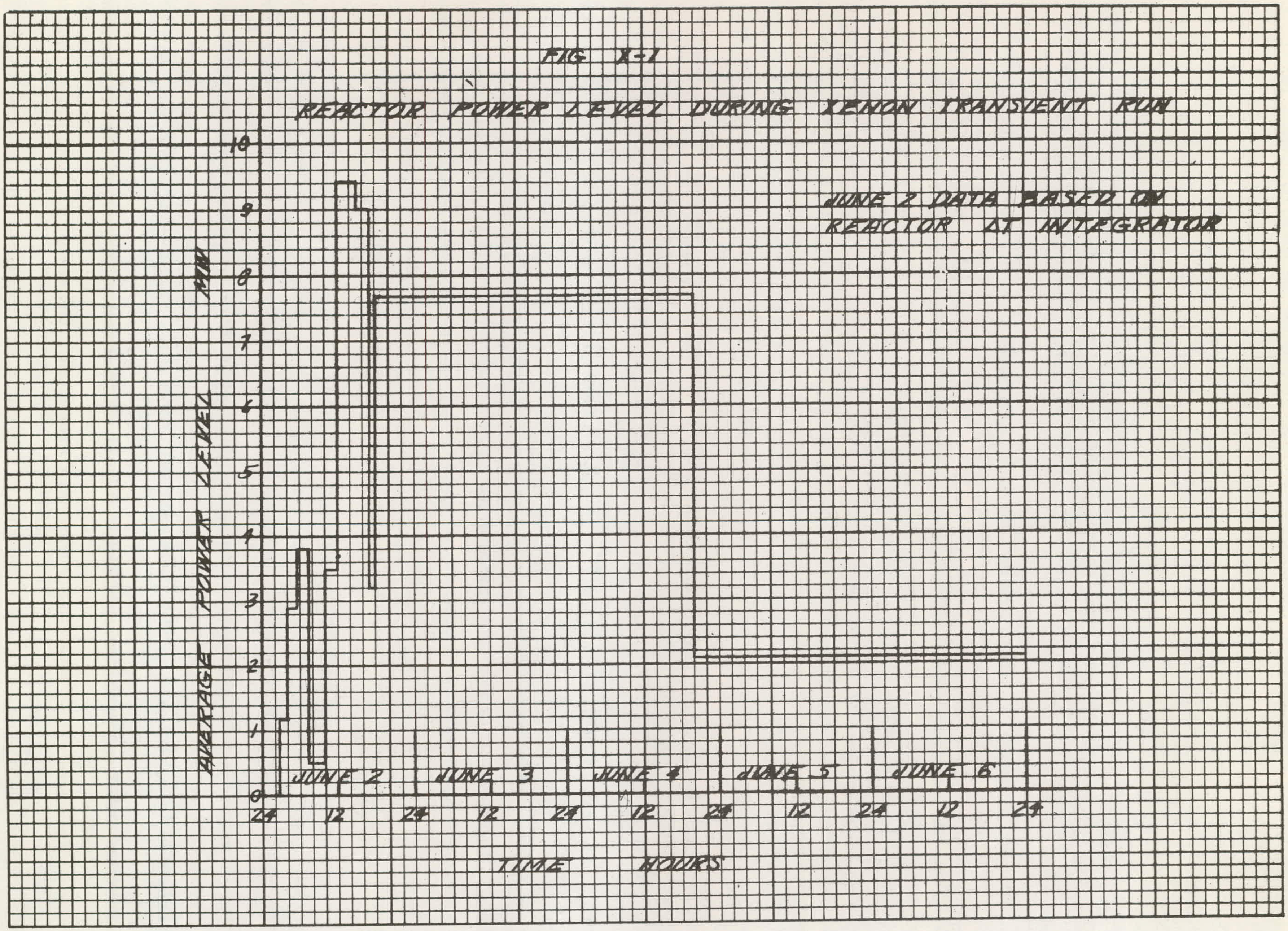


FIG. X - 1 REACTOR POWER LEVEL DURING XENON TRANSIENT RUN

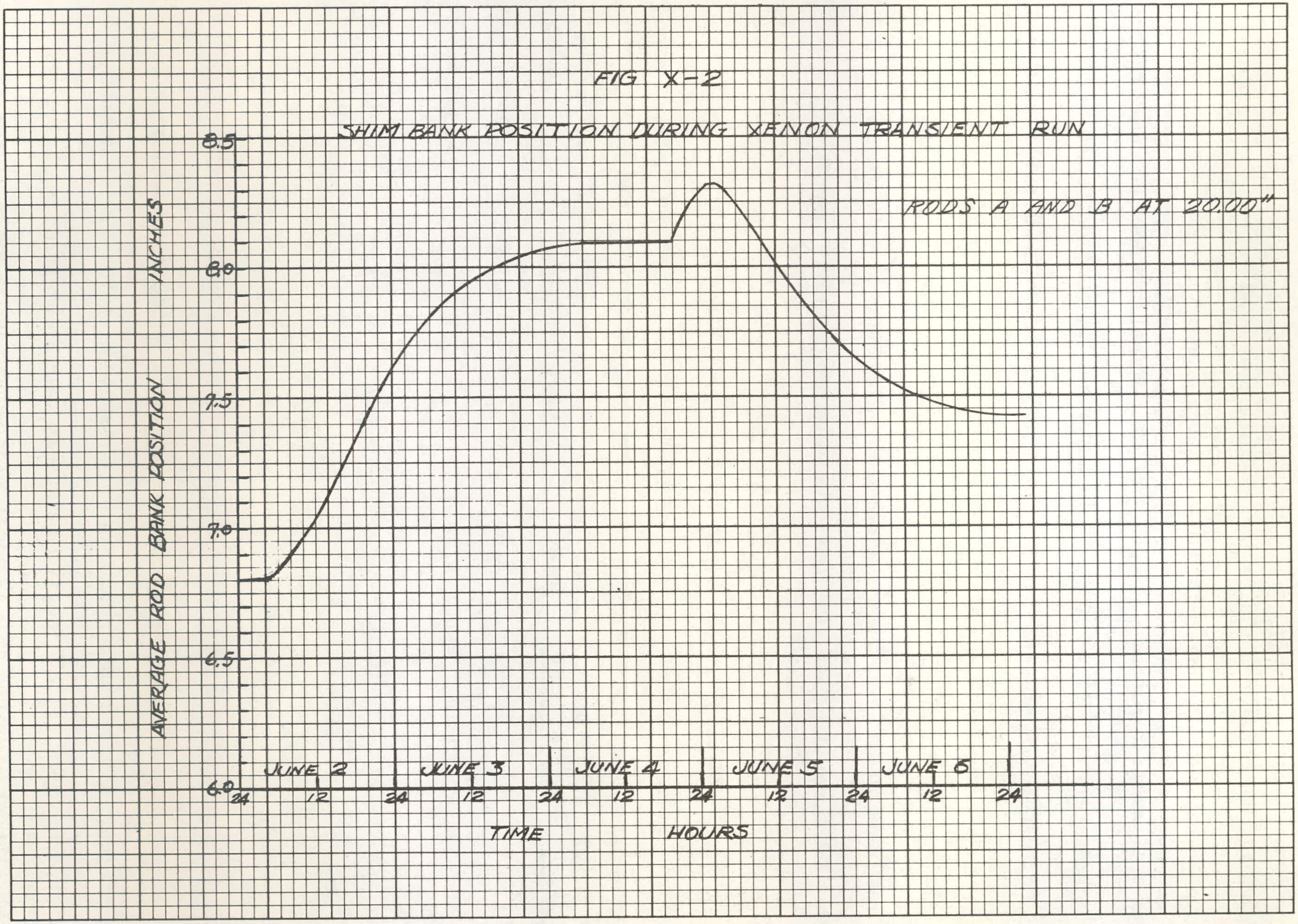


FIG. X - 2 SHIM BANK POSITION DURING XENON TRANSIENT RUN and at zero power. Figure 28 of reference 7 has been used to obtain a curve of total worth vs position by the method discussed on pages 5 and 6 of reference 12. The curve is shown in Figure X-3. It is recognized that this does not quite represent the total reactivity of the core, as the experimental work described in reference 7 was performed with rods A and B fully withdrawn to 22.00 inches. However, despite the slight variation in position of rods A and B the bank worth determined in the Zero Power Experiments is the best available evaluation of worth. The reactivities listed in Table X-1 were obtained by differences in bank position, and therefore the error due to the variation in safety rod position is small.

TABLE X-1

EQUILIBRIUM XENON MEASUREMENTS AT 440°F

Power Level (MW)	Core Burnup (MW-Yrs)	Bank Position Inches	Reactivity Controlled by Equilibrium Xenon (Percent): Based on Bank Position Data	Reactivity Controlled by Equilibrium Xenon (Percent): Corrected for Other Fission Prod.
0.0	0.14	6.80	0.0	0.0
7.7	0.19	8.10	1.9	1.8
2.1	0.20	7.44	0.9	0.8
10.4	0.35	8.43	2.4	2.2

The equilibrium xenon measurements listed in Table X-1 were taken at slightly different core burnups, and accordingly the bank positions were slightly affected. The run at zero power but at the operating temperature was made prior to the 700-hour test at a core burnup of 0.14 megawatt years. At the beginning of the 700-hour test, the 7.7 megawatt run was made followed by the 2.1 megawatt run with core burnups of 0.19 and 0.20 megawatt years respectively. The

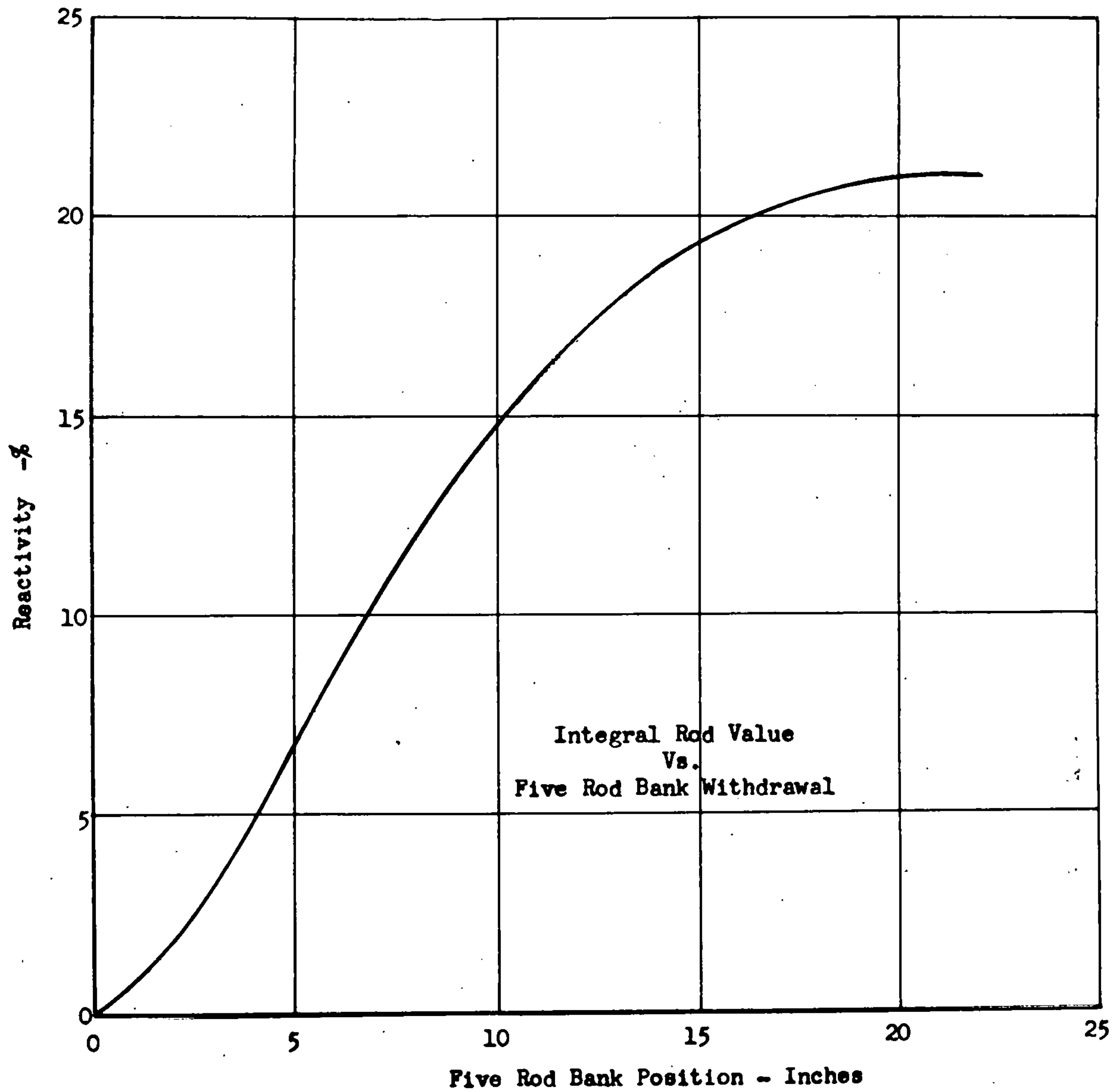


FIG. X - 3 TOTAL WORTH OF 5 ROD BANK VERSUS POSITION last equilibrium xenon run at 10.4 megawatts was made later in the 700-hour test at a core burnup of 0.35 megawatt years.

The lower curve of Figure 31 of reference 6 was used to determine the reactivity correction for core burnup. The reactivity controlled by fission products other than xenon varies from 0.2% at 0.14 megawatt years to 0.4% at 0.35 megawatt years. In the last column of Table X-1, all reactivities are corrected relative to the zero power run with 0.14 megawatt years burnup.

During the peak xenon transient shown in Figure X-2 in which the steady power of 7.7 megawatts was decreased to 2.1 megawatts, the maximum reactivity controlled by xenon was 2.1%, an increase of 0.3% above the steady state value at 7.7 megawatts.

CHAPTER XI - SHIELDING

(F. B. Fairbanks)

1. Instrumentation and Measurements

1.1 Locations Measured

During the low and high power runs, neutron and gamma measurements were made at numerous locations. The locations are described below and indicated in Figures XI-1 through XI-4.

Measurements were made immediately outside the primary shield, as indicated on Figures XI-1 and XI-2. Locations 1 through 8 are at locations where the primary shield is unbroken by penetrations, and the radiation from the primary coolant activity is negligible. Positions 9 through 16 are near the intersection of the projected center lines of the reactor inlet and outlet pipes with the outside of the primary shield tank. Locations 9, 10, 11, 12, 14 and 15 were selected prior to taking measurements. Points 13 and 16 were selected as places of maximum radiation from the shield penetration. Near the intersection of the primary outlet and inlet pipes with the primary shield tank are positions 17 through 19 and 20 through 24, respectively. Locations 17, 18, 20, 21, 22, and 23 were established prior to measurements. Points 19 and 24 were selected as the maximum radiation areas resulting from the penetrations.

Measurements on the top deck of the primary shield were made at locations 25 through 28. Position 28 is directly above an instrument tube.

Measurements at points 29 through 38 were taken at the vapor container wall to survey the variation of radiation intensity with radial direction. Positions 39 and 40 were at the inner wall of the vapor container.

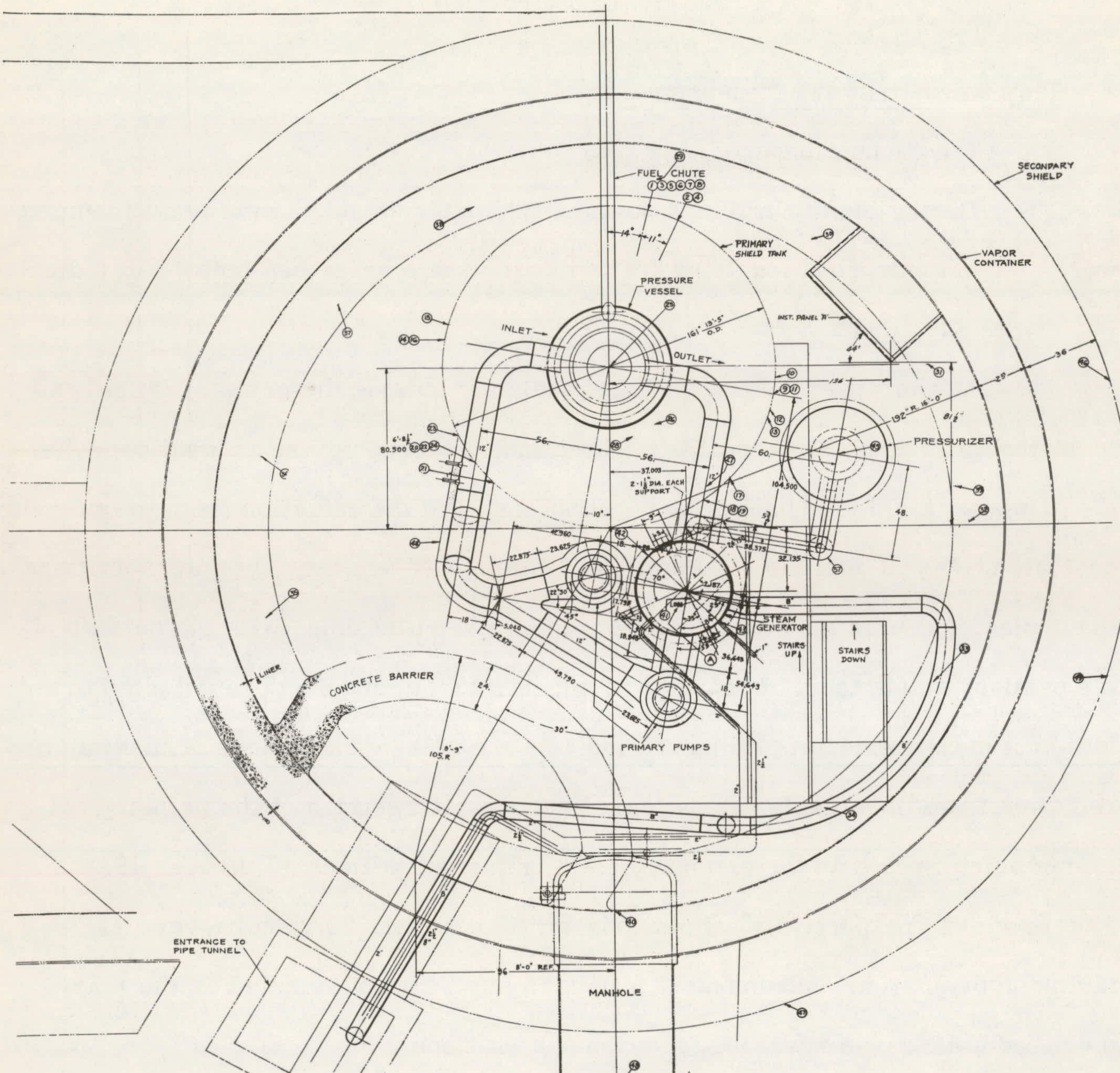


FIG. XI - 1 VAPOR CONTAINER PLAN SHOWING SHIELD MEASUREMENT LOCATIONS

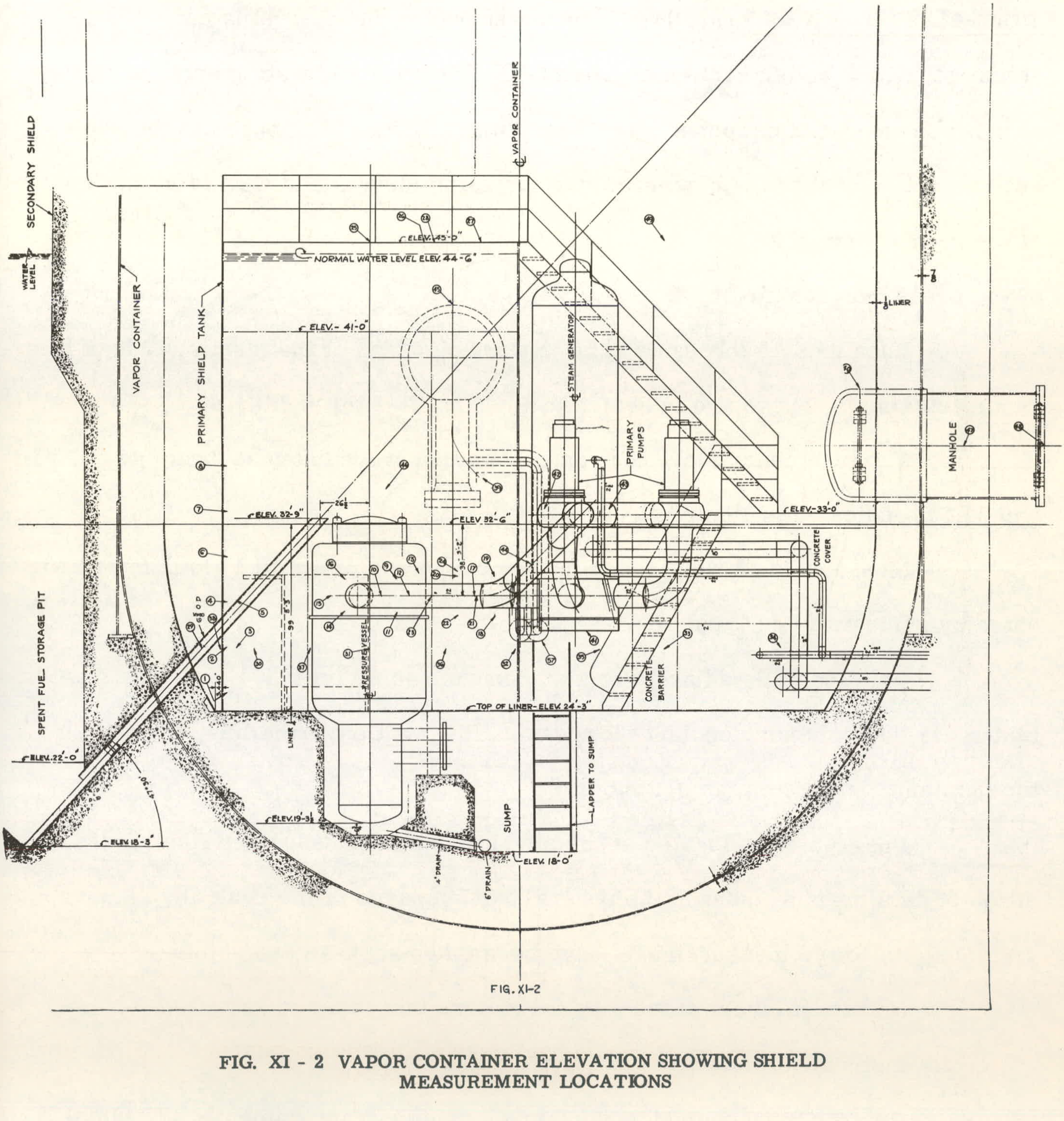


FIG. XI - 2 VAPOR CONTAINER ELEVATION SHOWING SHIELD MEASUREMENT LOCATIONS

To measure the importance of primary coolant sources, instruments were placed at points numbered 41 to 45, shown on Figures XI-1 and XI-2. Position 41 is directly underneath the steam generator channel, 42 is opposite the steam generator boiler section approximately midway in elevation between the top of the tube sheet and the normal secondary water level, 43 is opposite the superheater section of the steam generator of the same elevation as 42, 44 is on the side of the primary coolant inlet pipe, (outside of pipe insulation), and 45 is at the top of the pressurizer dome.

Locations 46 through 49 are in working areas of the plant outside of the vapor container. These are indicated in Figures XI-1 through XI-4.

Measurements above the vapor container were taken at locations 50, 51, and 52, as indicated in Figure XI-3.

Measurements were taken in the control rod drive pit at locations designated by numbers 53, 54 and 55 on Figures XI-3 and XI-4.

Position 56 is adjacent to a 1" Schedule 40 S, type-304 stainless steel tubing carrying primary coolant blowdown. The position of the line, located in the demineralizer room, is not shown.

Position 57 is located at the elbow of the 4" pressurizer line. Position 58 is a measurement in the general working area of the control rod drive pit. Only shutdown measurements have been taken at these two points.

Radiation levels in the pipe trench are discussed in section 2.1.

1.2 Instrumentation

Neutron measurements were made using two instruments. Thermal neutrons were measured with a BF_3 probe on a Nuclear Chicago Model 2112P

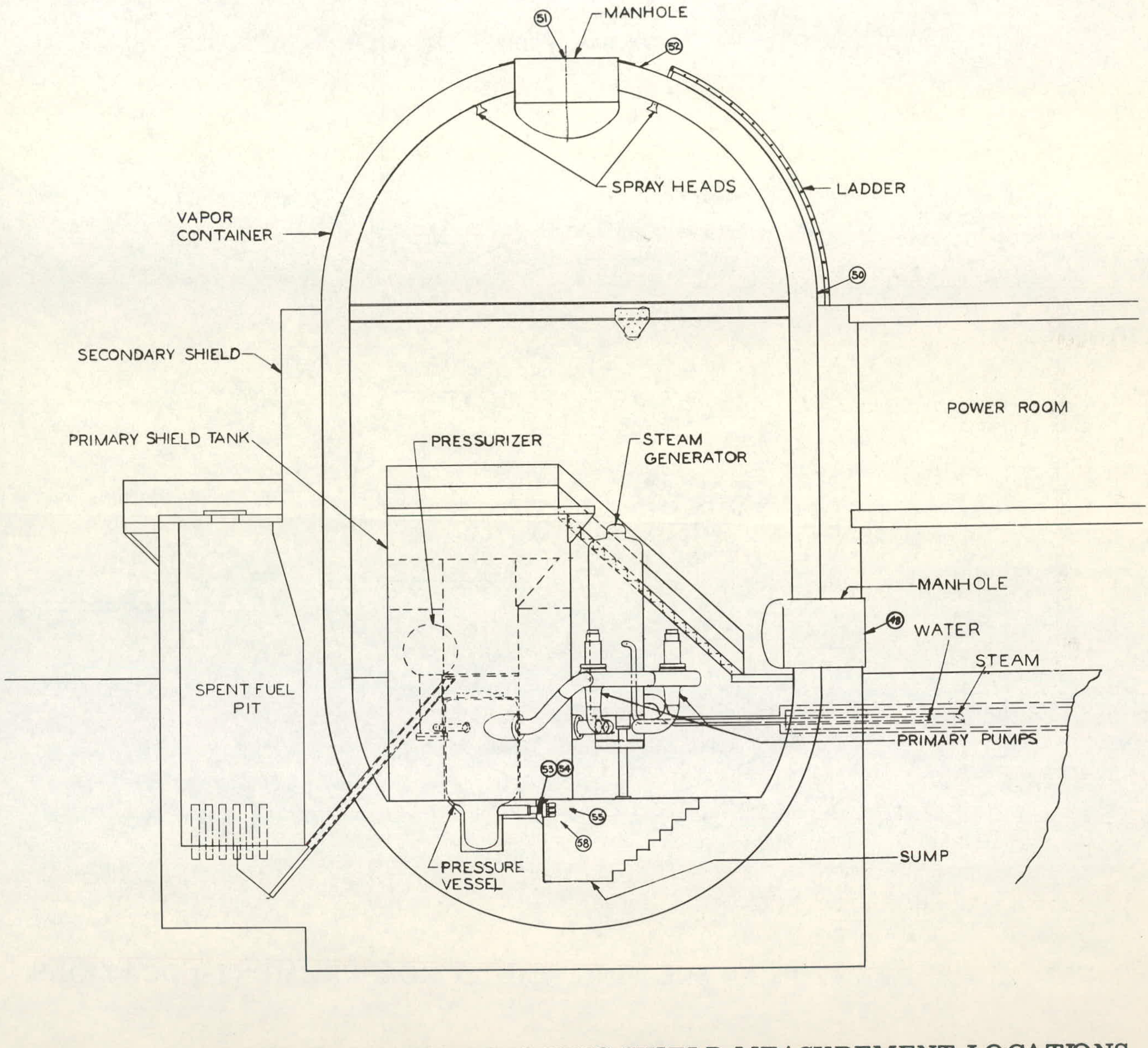


FIG. XI - 3 PLANT ELEVATION SHOWING SHIELD MEASUREMENT LOCATIONS

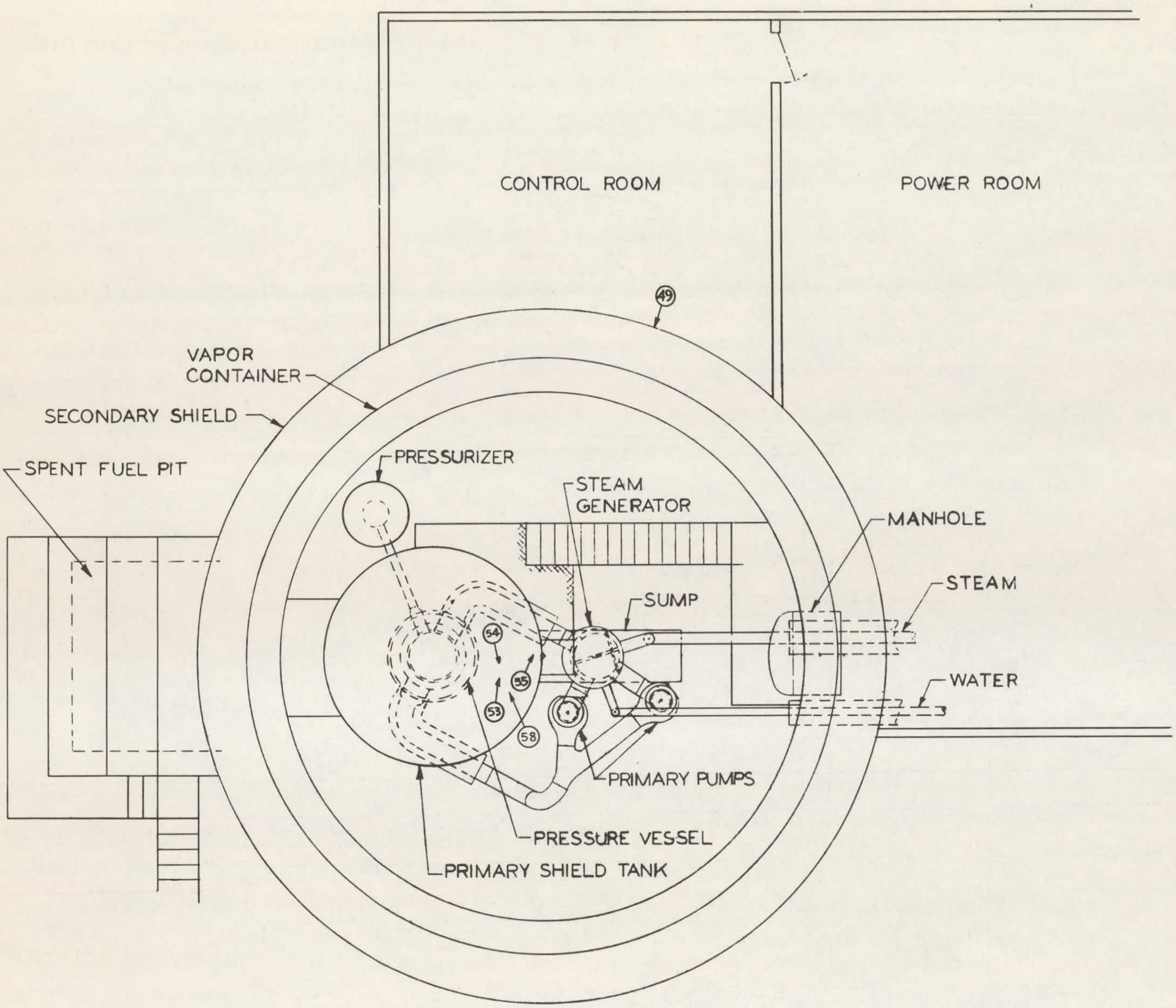


FIG. XI - 4 PLANT PLAN SHOWING SHIELD MEASUREMENT LOCATIONS alpha proportional counter modified for a BF_3 probe. A cadmium shield was available for the probe. The instrument was calibrated using a polonium-beryllium source with paraffin moderator. The calibration curve for obtaining thermal flux is presented in Fig. XI-5.

An attempt to measure fast neutrons was made using a Universal Atomics Ratemeter Model 522B with fast neutron scintillation crystal. The instrument, adjusted to bias against gammas of 1.3 Mev or lower in energy, was calibrated with a polyonium-beryllium source. It appears that the majority of counts recorded was due to gammas with energy greater than 1.3 Mev. Consequently, the data is not presented in this report.

Gamma dose rate was measured in the range 0.01 to 20 mr/hr using a El-Tronics Model PR-6 beta-gamma meter. In the higher range, gamma measurements were taken with a Technical Associates "Juno" survey meter, Model SRJ-3. Both instruments were calibrated with a cobalt-60 source.

1.3 Gold Foil Measurements

Bare and cadmium-covered gold foils were activated in a few reactor locations from April 19 to 21. Table XI-1 indicates the estimated operating power levels and the fraction each period contributed to the total saturated activity which would have existed if the reactor had operated continuously at 10 MW to 12 P.M. April 21. The foils reached 20.7% of this saturated activity.

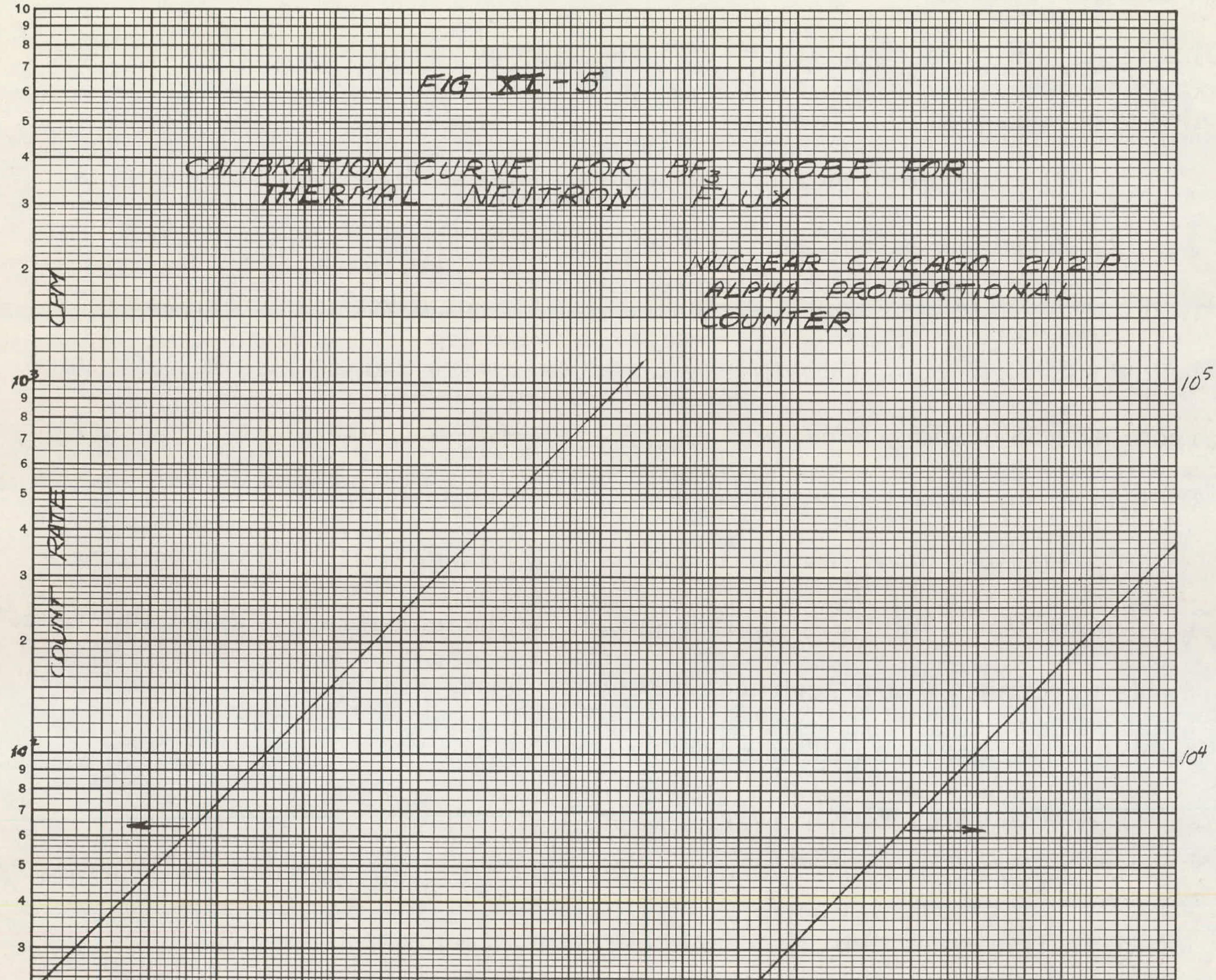


FIG. XI - 5 CALIBRATION CURVE FOR BF_3 PROBE FOR THERMAL NEUTRON FLUX

TABLE XI - 1

Reactor Operation During Foil Irradiation

Day April	Start of Period Hours	End of Period Hours	Estimated Mean Power Level MW	Fraction of 10 MW Saturated Activity at 12 P. M. April 21
19	1020	1408	0.2	.0004
	1408	1553	0.8	.0008
	1935	2207	1.1	.0018
	2207	2311	4.2	.0028
	2311	0021	6.0	.0045
20	0021	0120	8.2	.0052
	0120	0130	8.4	.0009
	0320	0735	1.3	.0039
	0735	0804	2.8	.0009
	0804	0835	9.2	.0034
	0835	0929	1.8	.0011
	0929	1316	2.2	.0061
	1316	1345	6.5	.0023
	1345	1409	10.0	.0030
	1409	2347	9.8	.0741
21	0615	0735	0.9	.0011
	0735	0837	1.5	.0018
	0857	1500	4.2	.0239
	1500	2300	8.5	.0690
			Total	.2070

The foil activity from thermal flux can be expressed by the equation

$$\text{Thermal activation} = (CR)_B - (CR)_{Cd} = K\, M\, \phi_{th}\; f_s$$

where

$(CR)_B$	Net count rate of bare foil	cts/min
$(CR)_{Cd}$	Net count rate of cadmium-covered foil	cts/min
K	Conversion factor	cts/min per n/cm^2-sec-gm
f_s	Saturation fraction	-
M	Mass of foil	gms
ϕ_{th}	Thermal flux	n/cm^2-sec

A value for K of 0.788 for the Alco Criticality Facility counter (in Schenectady) was determined from foil irradiations in the TTR[7]. By comparison of count rates for foils counted on the Schenectady and Ft. Belvoir counters, it was determined that the ratio of Schenectady to Belvoir counting rates was 1.07. Consequently the K used for the Belvoir data was 0.736.

The values of thermal flux indicated in Table XI-7 are calculated using $K = 0.736$, $M = 0.0314$ gms, and $f_s = 0.207$ from Table XI-1.

2. General Shield Performance

2.1 Intensities at Full Power Operation

Several runs at various power levels were taken, as shown in Table XI-2. The data taken during these runs is given in Tables XI-3 through XI-5. These data have been normalized to give fluxes and dose rates at full power operation. The values obtained are indicated in Table XI-6.

A survey was made at full power operation in the pipe trench where the steam line and other process piping leave the vapor container. The gamma dose rate in the area near the bottom of the ladder was 0.4 mr/hr. Near the penetrations of the pipes, the maximum dose rate was 6 mr/hr. Neutron levels were negligible.

Bare and cadmium-covered foils were irradiated in locations 2, 4, 13, 44, 53, 54, and 55. The net count rate from each of these foils, based on a background of 15.4 counts/minute, is indicated in Table XI-7. The values have been corrected to 12 P.M. April 21. The thermal flux was calculated using the equation and values from section 1.3.

It is recognized that the results of activation at locations 2, 4, and 13 are not accurate, as the net count rate was less than the background count rate. However, the figures do serve to indicate the order of magnitude of the thermal flux, and are probably better than the corresponding measurements made with the BF_3 counter which is sensitive to gamma rays.

The measurements indicate that the general primary and secondary shield design has been adequately conservative.

TABLE XI-2

SUMMARY OF SHIELDING RUNS

Run No.	Power Level (KW)	Date
1	2.15	April 12
2	24.1	April 13
3	200	April 13
4	2,690	April 15
5	10,100	April 20

TABLE XI-3

GAMMA INSTRUMENT MEASUREMENTS

Location Number	Run No. 1	2	3	4	5
	Power (MW)				
	2.15×10^{-3}	0.0241	0.200	2.69	10.10
	Instrument Reading (mr/hr)				
1	0.2	3.5	25.	-	-
2	0.3	4.5	33.0	-	-
3	0.3	5.0	34.0	-	-
4	0.15	1.5	12.0	-	-
5	0.15	1.2	12.0	-	-
6	0.03	0.2	2.0	-	-
7	0.01	0.07	1.0	-	-
8	0.005	0.05	1.0	-	-
9	1.9	28.5	200.0	-	-
10	0.8	12.0	80.	-	-
11	0.7	9.0	80.	-	-
12	1.2	16.0	140	-	-
13	4	33.0	220	-	-
14	0.65	11.0	60	-	-
15	0.70	14.0	80	-	-
16	1.9	18.0	140	-	-
17	4.5	33.0	230	-	-
18	5.1	39.0	280	-	-
19	6.0	37.0	280	-	-

TABLE XI-3 (Continued)

GAMMA INSTRUMENT MEASUREMENTS

Location Number	Run No. 1	2	3	4	5
	Power (MW)				
	2.15×10^{-3}	0.0241	0.200	2.69	10.10
	Instrument Reading (mr/hr)				
20	4.0	20.0	180	-	-
21	-	27.0	220	-	-
22	3.5	24.0	200	-	-
23	2.8	18.0	150	-	-
24	4.0	27.0	220	-	-
25	0.03	0.2	2	-	-
26	-	0.05	-	-	-
27	-	0.05	-	-	-
28	0.10	0.6	4	-	-
29	0.13	1.5	17.0	-	-
30	0.15	1.0	14.0	-	-
31	0.15	1.0	15.0	-	-
32	0.15	0.8	14.0	-	-
33	0.18	1.0	10.0	-	-
34	0.18	1.2	12.0	-	-
35	0.21	0.6	23.0	-	-
36	0.18	1.7	17.0	-	-
37	0.20	1.5	17.0	-	-
38	0.21	1.6	19.0	-	-

TABLE XI-3 (Continued)

GAMMA INSTRUMENT MEASUREMENTS

Location Number	Run No. 1	2	3	4	5
	Power (MW)				
	2.15×10^{-3}	0.0241	0.200	2.69	10.10
	Instrument Reading (mr/hr)				
39	0.1	1.2	12	-	-
40		0.7	7	-	-
41	0.35	6.0	60	-	-
42	0.5	4.5	38	-	-
43	0.8	11.0	70	-	-
44	4.0	29.0	220	-	-
45	0.02	0.3	-	-	-
50	-	-	-	.05	.05
51	-	-	-	3.5	13.5
52	-	-	-	0.2	0.7
53	120	1500	-	-	-
54	140	1700	-	-	-
55	27	250	-	-	-
56	-	0.02	-	-	-

TABLE XI-4

BF$_3$ - CHAMBER MEASUREMENTS

Location Number	Run No. 1	2	3	4	5
	Power (MW)				
	2.15×10^{-3}	0.0241	0.200	2.69	10.10
	Instrument Reading (cts/min)				
1	200	120	11000	-	-
2	250	400	4000	-	-
3	100	500	-	-	-
4	100	400	3000	-	-
5	75	450	-	-	-
6	200	350	3000	-	-
7	250	300	3000	-	-
8	150	300	3000	-	-
9	700	1100	8000	-	-
10	620	900	7000	-	-
11	700	1150	8000	-	-
12	450	1100	9000	-	-
13	-	-	9000	-	-
14	300	1700	12000	-	-
15	300	1700	13000	-	-
16	300	1700	12000	-	-
17	400	2500	off scale	-	-
18	800	6000	off scale	-	-

TABLE XI-4 (Continued)

BF_3 - CHAMBER MEASUREMENTS

Location Number	Run No. 1	2	3	4	5
	Power (MW)				
	2.15×10^{-3}	0.0241	0.200	2.69	10.10
	Instrument Reading (cts/min)				
19	300	1100	off scale	-	-
20	400	-	off scale	-	-
21	-	-	off scale	-	-
22	2000	-	off scale	-	-
23	7000	-	off scale	-	-
24	750	-	off scale	-	-
25	20	350	3000	-	-
26	-	400	3000	-	-
27	-	400	3000	-	-
28	20	600	4500	-	-
29	60	-	-	-	-
30	60	-	-	-	-
31	80	-	-	-	-
32	120	-	-	-	-
33	300	-	-	-	-
34	400	-	-	-	-
35	500	-	-	-	-
36	500	-	-	-	-
37	300	-	-	-	-

TABLE XI-4 (Continued)

BF$_3$ - CHAMBER MEASUREMENTS

Location Number	Run No. 1	2	3	4	5
	Power (MW)				
	2.15×10^{-3}	0.0241	0.200	2.69	10.10
	Instrument Reading (cts/min)				
38	350	-	-	-	-
39	60	-	5000	-	-
40	-	-	8000	-	-
41	4000	-	off scale	-	-
42	30	-	-	-	-
43	115	-	5500	-	-
44	-	-	off scale	-	-
50	-	-	-	30	15
51	-	-	-	1000	1050
52	-	-	-	60	60
53	-	off scale	-	-	-
54	-	off scale	-	-	-
55	-	off scale	-	-	-

TABLE XI-5

CADMIUM-COVERED BF_3-CHAMBER MEASUREMENTS

Location Number	Run No. 1	Run No. 2	Run No. 3	Run No. 5
	Power (MW)			
	2.15×10^{-3}	0.0241	0.200	10.10
	Instrument Reading (cts/min)			
1	70	120	2000	-
2	150	110	700	-
3	75	45	600	-
4	100	-	500	-
5	35	50	450	-
6	90	60	450	-
7	30	30	450	-
8	25	30	400	-
9	-	115	3000	-
10	-	120	2000	-
11	-	120	2000	-
12	-	120	2500	-
13	-	-	4500	-
14	-	130	2000	-
15		125	1500	-
16	-	-	2000	-
17	-	550	5000	-
18	-	1000	7000	-
19		140	-	-

TABLE XI-5 (Continued)

CADMIUM-COVERED BF_3-CHAMBER MEASUREMENTS

Location Number	Run No. 1	2	3	5
	Power (MW)			
	2.15×10^{-3}	0.0241	0.200	10.10
	Instrument Reading (cts/min)			
20	-	-	4000	-
21	-	-	15000	-
22	-	-	15000	-
23	-	-	7000	-
25	15	30	350	-
26	-	35	400	-
27	-	45	450	-
28	-	100	800	-
39	29	-	800	-
40	-	-	1500	-
41	-	-	off scale	-
42	30	-	-	-
43	55	-	900	-
44	-	800	8000	-
51	-	-	-	450
52	-	-	-	35

TABLE XI-6

FULL POWER GAMMA AND NEUTRON LEVELS AT SELECTED LOCATIONS

Location Number	Gamma Flux r/hr	Thermal Flux 10^3 n/cm^2- sec
1	1.3	17.
2	1.7	4.8
3	1.7	6.0
4	0.6	4.6
5	0.6	5.7
6	0.10	4.2
7	0.05	3.7
8	0.03	3.8
9	11.	8.4
10	4.0	7.8
11	3.7	11.
12	6.6	9.6
13	15.	7.2
14	4.0	17.
15	5.5	18.
16	7.5	17.
17	14.	22.
18	16.	63.
19	15.	13.
20	9.0	24.

TABLE XI-6 (Continued)

FULL POWER GAMMA AND NEUTRON LEVELS

AT SELECTED LOCATIONS

Location Number	Gamma Flux r/hr	Thermal Flux 10^3 n/cm^2-sec
21	11.	-
22	10.	130
23	8.0	40
24	12.	-
25	0.10	4.4
26	0.02	4.3
27	0.02	4.2
28	0.25	6.0
29	0.8	-
30	0.7	-
31	0.7	-
32	0.7	-
33	0.5	-
34	0.6	-
35	1.1	-
36	0.8	-
37	0.8	-
38	1.0	-
39	0.6	7.
40	0.3	10.

TABLE XI-6 (Continued)

FULL POWER GAMMA AND NEUTRON LEVELS

AT SELECTED LOCATIONS

Location Number	Gamma Flux r/hr	Thermal Flux 10^3 n/cm^2-sec
41	3.0	-
42	1.9	-
43	4.0	-
44	12.0	-
45	0.10	-
46	Undetectable	Undetectable
47	Undetectable	Undetectable
48	Undetectable	Undetectable
49	Undetectable	Undetectable
50	0.0001	Undetectable
51	0.013	0.035
52	0.0007	Undetectable
53	600.	-
54	700.	-
55	120.	-
56	0.008	-

TABLE XI - 7

FOIL MEASUREMENTS OF THERMAL NEUTRON FLUX

Location	Foil	Net Count Rate Corrected to 12 pm Apr. 21 Counts/minute	Thermal Flux at Full Power n/cm^2 - sec.	Cadmium Ratio
2	Bare	5.1	1070	-
4	Bare	0.7	150	-
13	Bare	3.3	170	1.3
	Cadmium-covered	2.5		
14	Bare	21.2	2400	2.2
	Cadmium-covered	9.5		
53	Bare	24,200	2.4×10^6	1.94
	Cadmium-covered	12,500		
54	Bare	21,200	2.3×10^6	2.11
	Cadmium-covered	10,040		
55	Bare	9,190	7.8×10^5	1.68
	Cadmium-covered	5,470		

2.2 Radiation after Shutdown

Gamma measurements were taken after shutdown from the 700-hour test. The reactor had been operating near full power until 12:40 p.m. July 1, at which time the reactor power was dropped to 2 MW for 4 hours. From 4:40 pm July 1 to 12:48 a.m. July 2, the reactor was again operated at 10 MW, at which time the reactor was shut down for measurements.

The radiation level at various points is shown in Tables XI-8 and XI-9 for various times after shutdown.

In general, radiation levels were quite reasonable. The dose rate at the pressurizer line elbow (location 57) was apparently the result of accumulation of activity in this stagnant pipe.

3. Control Rod Drive Pit Shielding

As a result of initial tests, the control rod drive pit area required slight modification of the shielding layout. The basic objective of the added shielding was to block streaming paths existing near the water box surrounding the control rod drive shafts.

Above the water box, blocks of lead approximately 1/2" thick were used to fill the space between the water box and the base plate of the shield tank. The lead extended 8" back from the outer face of the box. At the top in front of the water box, a 2"-high, 4"-deep, section of lead was added. In front of this was placed a 6"-high, 12"-deep section of high density wood (green heart), which essentially filled the available volume above the rod drive seals. The wood, applied in sections to permit removal when necessary to maintain rod drive equipment, was clad with cadmium sheet on the outside.

TABLE XI-8

GAMMA RADIATION LEVELS AFTER SHUTDOWN FROM 700-HOUR TEST

Time after Shutdown, Hours	Location No. 2	19	40	43	44	57	58
	Gamma Dose Rate, mr/hr						
7.9	2	54	2	39	-	300	140
8.7	2	-	2	38	60	300	140
10.0	2	42	2	37	56	300	140
12.2	1.5	34	1	33	44	300	140
13.9	1.5	28	1	27	40	300	140
16.1	-	28	-	24	-	300	-
18.0	-	28	-	23	-	320	-
20.2	-	27	-	20	-	320	-
24.0	-	25	-	21	-	320	-
26.0	-	25	-	21	-	320	-
28.4	-	26	-	22	-	300	-
30.5	-	31	-	20	26	300	73
37.0	-	19	-	17	22	280	60
80.0	-	16	-	14	19	240	52

TABLE XI-9
GAMMA RADIATION LEVELS 8.7 HOURS AFTER SHUTDOWN FROM 700-HOUR TEST

Location No.	Gamma Dose Rate Mr/Hr
2	2
4	1
5	1
6	1
12	4
15	16
17	50
18	45
20	45
21	38
22	39
23	46
25	0.1
29	6
40	2
41	3
43	38
44	60
57	300
58	140

To cover the small gaps on each side of the water box between the box and the adjacent concrete, blocks of hard wood (maple) were used. These blocks were 6 inches deep and 4 inches wide. In the area near the fill and return lines to the water box, additional concrete, encased in rubber tubes, was used to fill voids.

The neutron and gamma levels during operation and after shutdown have been reported in sections 2.1 and 2.2, respectively.

4. Pressure Vessel Wall Heating

Data has been recorded on three thermocouples attached to the outside of the reactor vessel wall at the midplane of the core. The thermocouples were located 120° apart.

Data for steady runs at three power levels are given in Table XI-10. For each position and power level, the average difference between the thermocouple and inlet water temperature readings have been calculated. The values are shown in Table XI-11. Assuming that the actual temperature difference is zero at zero reactor power, the corrected values shown in the lower half of Table XI-11 are obtained for each point by subtracting the average difference values at zero power from the average difference at the various powers. The average for each power of the corrected average temperature differences is listed in the last column of the table and is plotted in Figure XI-6.

TABLE XI-10

PRESSURE VESSEL WALL HEATING DATA

Reactor Power	Date	Time	Reactor Inlet Temperature °F	Thermocouple Readings Position 14 °F	Position 15 °F	Position 16 °F
0	April 20	5:59 a.m.	436	424	433	429
		7:32 a.m.	438	426	435	430
3.2	April 29	12:06 p.m.	408	399	410	407
		1:00 p.m.	409	399	410	407
		1:34 p.m.	408	399	410	407
		3:50 p.m.	409	398	409	406
		5:05 p.m.	408	-	408	406
9.7	April 20	4:14 p.m.	427	-	435	430
		7:38 p.m.	426	425	435	430
		8:20 p.m.	426	426	434.5	430
		9:18 p.m.	425	428	434	430
		10:22 p.m.	426	427	435	430
		11:35 p.m.	425	427	435	430

TABLE XI-11

ANALYSIS OF PRESSURE VESSEL WALL HEATING

Reactor Power MW	Average Difference Between Thermocouple Reading and Reactor Inlet Temperature			
	Position 14 °F	Position 15 °F	Position 16 °F	Average °F
Uncorrected				
0	-12.0	-3.0	-7.5	-
3.2	- 9.8	1.0	-1.8	-
9.7	1.0	8.9	4.2	-
Corrected				
0	0.0	0.0	0.0	0.0
3.2	2.2	4.0	5.7	4.0
9.7	13.0	11.9	11.7	12.2

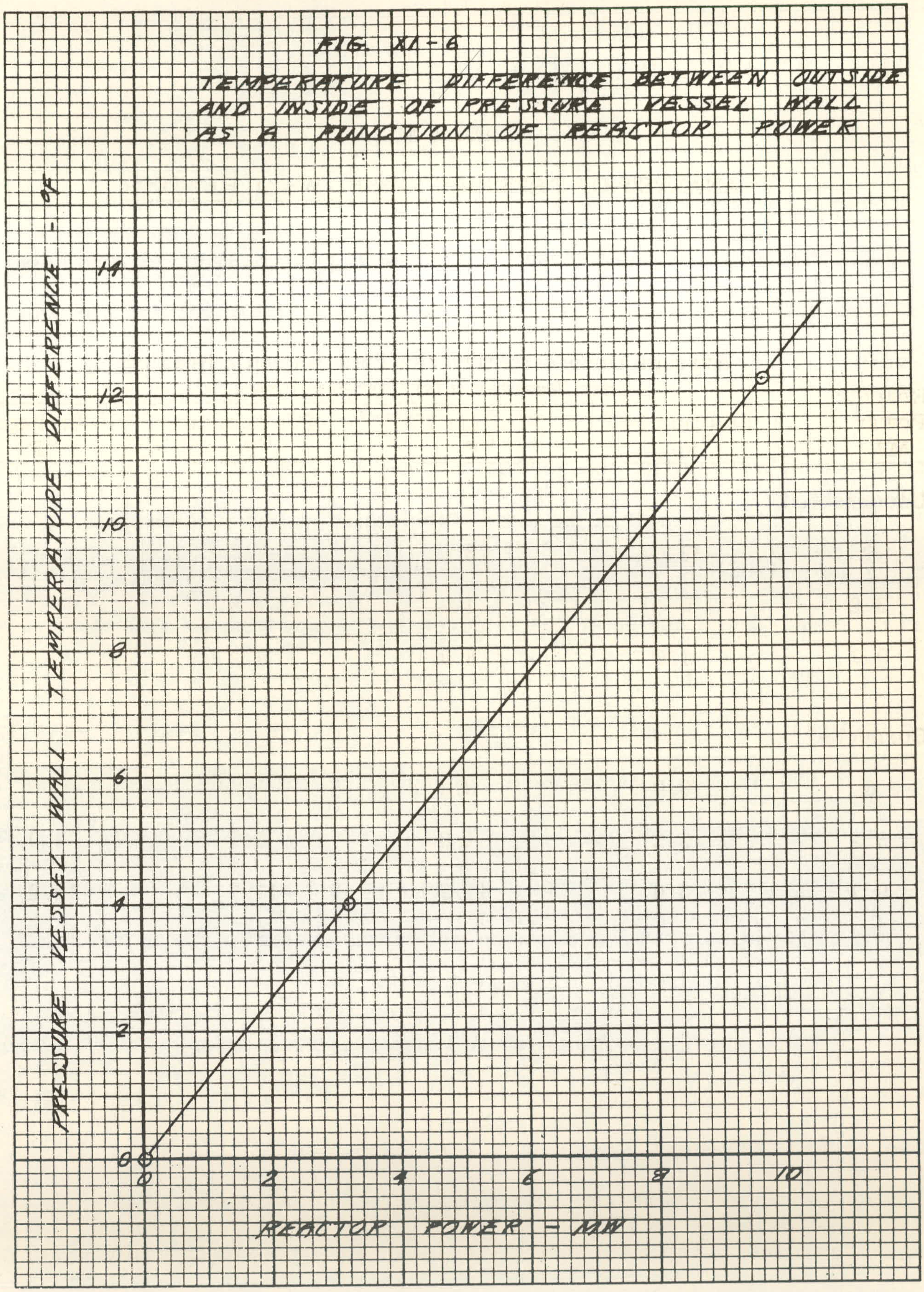


FIG. XI - 6 TEMPERATURE DIFFERENCE BETWEEN OUTSIDE AND INSIDE OF PRESSURE VESSEL WALL AS A FUNCTION OF REACTOR POWER

CHAPTER XII - WATER TREATMENT

(A. L. Medin)

1. Primary System Cleaning

To assure absolute cleanliness in the primary system, all equipment had been cleaned either in the vendor's shop or during the actual construction. Nevertheless, even with the close control a certain amount of dirt and oil was present on certain of the vessels. It should be emphasized, however, that because of the close control, this dirt and oil was at a minimum.

For the cleaning of the inner shield which had become dirty and greasy during the installation, a simple and successful procedure was used. All rust and imbedded foreign material was removed by grinding with a stainless steel brush. Except for deeply imbedded marks which appeared to be primarily as a result of heat treatment all foreign material was removed.

To remove oil, alcohol-soaked rags were employed. Again this method was successful and exceedingly fast. The wipe procedure was continued until the wipe cloths were substantially free of any foreign dirt or oil. During subsequent periods of operation whenever the inner shield was drained of water this shield was wiped with methyl alcohol-soaked rags.

Final cleaning of the pressure vessel was made prior to the installation of the fuel element grid. In practically all instances either methyl alcohol or other solvents were used to wipe any oily or foreign material from the vessels. Although these methods require care by the persons doing the cleaning, they are fast and reliable. Of great advantage is the fact that no noticeable film is left by these solvents which could subsequently cause possible interference in maintaining the primary water purity specifications.

Prior to the loading of the core, in-place cleaning of the primary system components was performed. A temporary line was installed from the lower control rod penetration at the bottom of the pressure vessel to the top of the pressurizer. A 20 gpm pump was inserted into this line to recirculate the water from the pressure vessel through the pressurizer and back to the pressure vessel. A cartridge type filter was placed in the temporary line between the pressurizer and the reactor vessel. The filter contained 24 cartridges. Water was circulated for approximately 32 hours. During the first 8 hours, a 25 micron filter was employed and removed dirt and grit from the primary system. During the final 24 hours a 10 micron filter was inserted until no additional foreign material was removed.

By this in-place cleaning which was closely checked by various water tests, it was deemed unnecessary to add any chemicals to clean the primary system. A substantial reduction in the necessary time to do in-place cleaning was obtained by this method.

2. Secondary System Cleaning

The secondary system components were cleaned according to standard practices. The condenser hotwell, evaporator, feedwater heater and other auxiliary items were cleaned by an alkaline boil-out procedure. These procedures produced vessels free of large foreign particles and free of oil and grease.

The steam generator was cleaned by circulating a mild detergent through it and the auxiliary piping. No oil, grease or foreign material were found after the cleaning.

3. Primary System Water Treatment During Precritical Test (L. Heider)

3.1 Chemical Feed System

Since during the dummy core test, no gamma radiation is present to catalyze the reaction, $H_2 + 1/2\,O_2 \rightarrow H_2O$ in the presence of an excess of hydrogen, hydrazine was used as the oxygen scavenger instead of hydrogen. As a result of the intermittent operations and frequent plant shut-downs, oxygen scavenging was extremely difficult. Nevertheless, the overall oxygen scavenging with hydrazine was successful. The chemical feed rates varied from 0 to 14 cc/min. at hydrazine concentrations of 0.04 to 1.0%.

No other chemical was added to the primary system.

3.2 Oxygen Control

Although the up and down plant operation made it difficult to keep the hydrazine feed at any predetermined rate, the overall oxygen scavenging was successful. Table XII-1 summarizes typical oxygen residuals present at various times. Since the reactor cover was opened frequently, the rapidity with which hydrazine scavenged oxygen was even more pronounced. Of particular interest is this rapidity of reaction between the hydrazine and the oxygen.

TABLE XII-1, OXYGEN RESIDUALS (Precritical)

Date	Time	Sample Number	Oxygen Residual, (ppm)
3/16/57	2230	1	5.4
3/17/57	0310	2	0.28
3/17/57	0700	4	0.19
3/17/57	1100	7	0.006
4/1/57	0810	103	0.13
4/1/57	2040	107	0.075
4/1/57	2330	113	0.017
4/2/57	0800	121	0.016
4/13/57	0720	249	0.0
4/13/57	1830	257	0.06

3.3 Chloride Control

The chloride control in the primary system was excellent at all times. During preliminary operations with distilled water, the chloride content was approximately 0.5 ppm. After the primary demineralizer was put into normal service, the chlorides dropped essentially to 0.0 when sufficient blow-down was established. This result is clearly evidenced around the beginning of April.

Table XII-2 summarizes the chlorides results. At the lower chloride level, the limit of detection is within 0.1 ppm. Thus, results of 0.1 ppm or lower could be essentially zero.

TABLE XII-2 CHLORIDE CONCENTRATION (Precritical)

Date	Time	Sample No.	Chloride Concentration (ppm)
3/16	1600	1	0.55
3/17	0800	2	0.47
3/19	0030	11	0.45
3/20	0230	22	0.41
3/21	0300	32	0.65
3/29	1025	56	0.57
3/30	0820	73	0.31
3/31	1700	93	0.38
4/1	0810	109	0.29
4/2	0800	121	0.31
4/3	0500	134	0.31
4/11	2350	222	0
4/12	0940	229	0
4/13	0200	245	0

3.4 Total Solids

The total solid concentration in the primary system was well within specifications. The limit of 2 ppm was seldom, if at all, exceeded. Table XII-3 summarizes the results. Since in the determination of the total suspended solids, the error in determination is oftentime greater than the actual quantity found, care should be used in interpreting this result.

TABLE XII-3 TOTAL SUSPENDED AND TOTAL DISSOLVED SOLIDS (Precritical)

Date	Time	Sample Number	Total Suspended Solids (ppm)	Total Dissolved Solids (ppm)
3/19	0930	14	0.30	2.1
3/20	2015	28	1.10	1.4
3/30	1615	80	-	1.74
3/31	0100	84	-	1.00
4/1	0130	99	-	0.63
4/12	1545	236	-	0.65

3.5 Conductivity and pH

Results of the conductivity and pH give an indication of the operating efficiency of the demineralizer. These results will be discussed in more detail in a later section. As would be expected, however, the pH downstream of the demineralizer was generally lower than that upstream. Naturally, the conductivity downstream was less than the upstream conductivity. Figures XII-1 and XII-2 illustrate the relationships between the pH and conductivity across the demineralizer.

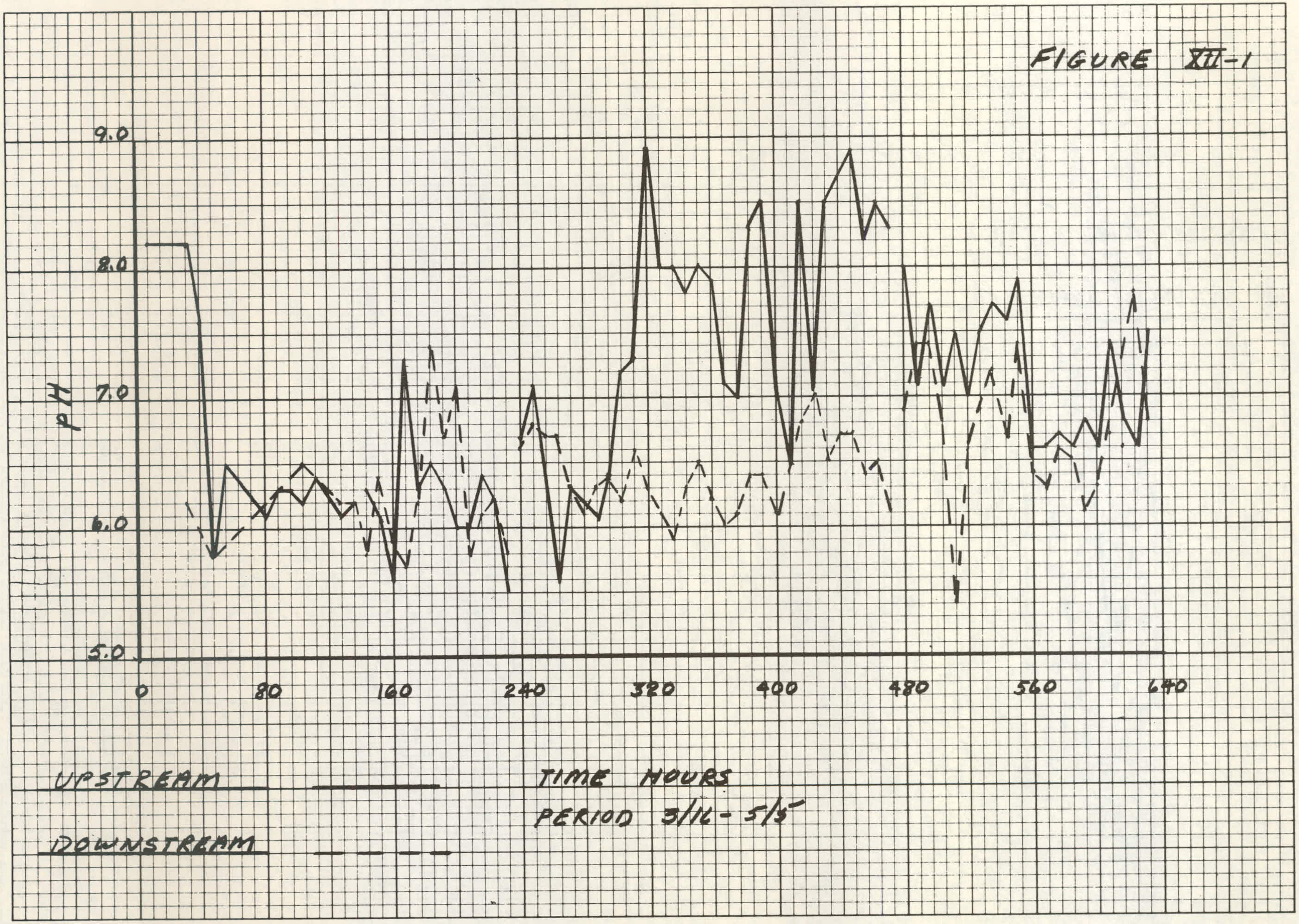


FIG. XII - 1 EFFECT OF DEMINERALIZER ON pH OF PRIMARY WATER

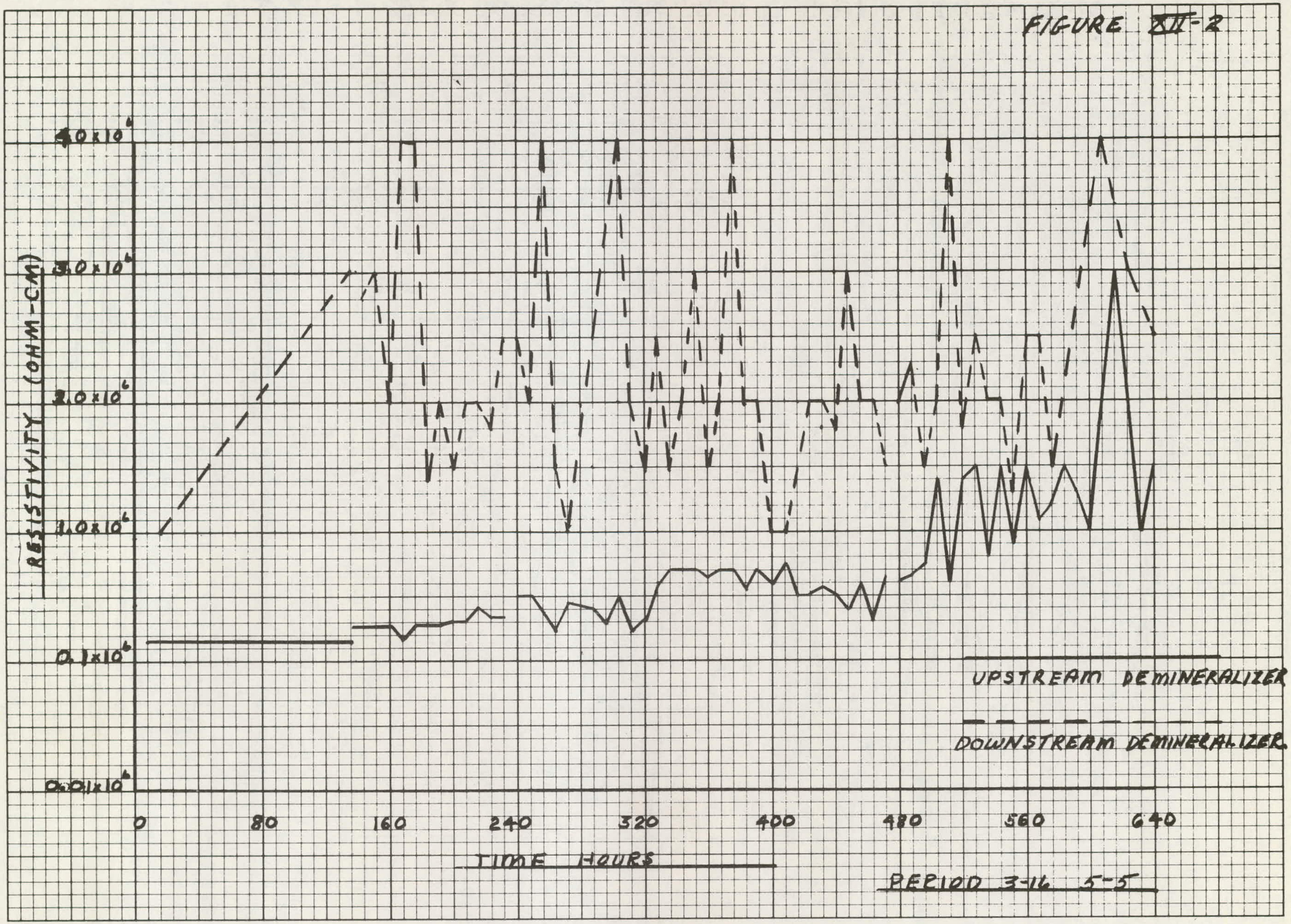


FIG. XII - 2 EFFECT OF DEMINERALIZER ON RESISTIVITY OF PRIMARY WATER

4. Secondary System Water Treatment During Precritical Test

During precritical operations, the secondary system was used to supply heat to the primary loop as described in Chapter III. As such, it was not possible to conduct the water treatment techniques in a directly applicable manner. Although every attempt was made to minimize oxygen, the many frequent shut-downs precluded the success of this program.

4.1 Condensate Corrosion Control

To minimize corrosion in the hotwell and condensate return, the pH was to be maintained above 8.0. During most of the operations, the pH was above this value as indicated in Table XII-4.

TABLE XII-4 pH IN CONDENSER HOTWELL (Precritical)

Date	Time	Sample Number	pH
3/17	0800	3	10.2
3/19	0445	13	7.7
3/20	0130	20	8.4
3/21	0415	34	9.1
3/29	1100	59	9.6
3/30	0930	75	9.8
3/31	0100	87	10.5
4/1	0130	102	9.8
4/2	0130	118	9.6
4/6	0130	169	7.5
4/7	1330	188	5.9
4/8	0115	196	8.1
4/11	1036	216	8.6
4/12	0955	231	6.9

4.2 Chloride Control in Steam Generator

To minimize the possibility of chloride stress corrosion, the feed to the steam generator was maintained at a low chloride level. It was believed that since the APPR-1 steam generator was acting only as a heat exchanger, the concentration of chlorides would either be very small or neglible. Typical chloride analyses in the steam generator feed is indicated in Table XII-5.

TABLE XII-5 - CHLORIDES IN STEAM GENERATOR FEED FROM PORTABLE UNIT

Date	Time	Sample No.	Chloride Concentration (ppm)
3/19	1700	18	0.56
3/20	0855	24	0.19
3/21	0100	30	0.24
3/29	0800	55	0.31
3/30	0940	76	0.31
3/31	0100	86	0.40
4/1	0130	101	0.18

4.3 Chloride Control in Hot-Well

The chlorides in the hot-well were higher than predicted. This fact led to some speculation regarding the possibility of a leak in the condenser tubes. On further checking however, the large chloride residual was traced to the sodium sulfite oxygen scavenger. Because of the large amounts of scavenger being used during plant shut-down, the chloride residual increased. As indicated in Table XII-6, on discontinuance of the sulfite addition, the chloride residual dropped.

TABLE XII-6 - CHLORIDE IN CONDENSER HOTWELL (PRECRITICAL)

Date	Time	Sample No.	Chloride (ppm)
3/18	1530	10	0.40
3/19	0445	13	0.60
3/20	0130	20	0.72
3/21	0100	31	0.58
3/29	1755	65	1.06
3/30	0930	75	1.67
3/31	0100	87	1.80
4/1	0130	102	0.94
4/2	0800	123	1.81 *
4/6	0130	169	0.53
4/7	1330	188	0.16
4/11	1036	216	0.26
4/12	0955	231	0.18
4/13	0030	242	0.17

* Sodium Sulfite discontinued.

5. Primary System Water Treatement During Plant Start-Up

5.1 Chemicals Added

After the reactor had gone critical the hydrazine addition was suspended. It its place hydrogen was added to the primary system. Since in the presence of gamma radiation hydrogen will react with oxygen to form water, they use of any other chemical scavenger was not necessary. Maintaining an approximate gauge reading of 15 to 20 psig from the hydrogen feed tank was adequate to maintain the necessary hydrogen concentration in the primary system.

5.2 Hydrogen

The hydrogen control on the primary system was excellent. Following the addition of hydrogen, no oxygen of any consequential nature was detected during this phase of the operation. The closeness of the hydrogen control is recognized by Table XII-7. The presence of other dissolved gases prevented the complete instantaneous solubility of hydrogen. On exhaustion of these gases by the various radiation reactions, the hydrogen residual in the primary make-up tank and the primary system approached equilibrium as indicated in Figure XII-3.

TABLE XII-7 - HYDROGEN RESIDUAL (STARTUP)

Date	Time	Sample Number	Hydrogen Residual (cc/l)
4/14	1050	273	10.0
4/15	0740	281	17.3
4/16	1750	318	18.9
4/17	0015	325	22.6
4/19	1800	341	24.6
4/20	0800	361	35.0
4/21	0820	387	34.8
4/27	1625	416	34.0
4/28	1505	428	25.4
4/30	2000	484	43.0
5/3	1945	533	36.0
5/4	2000	564	30.0

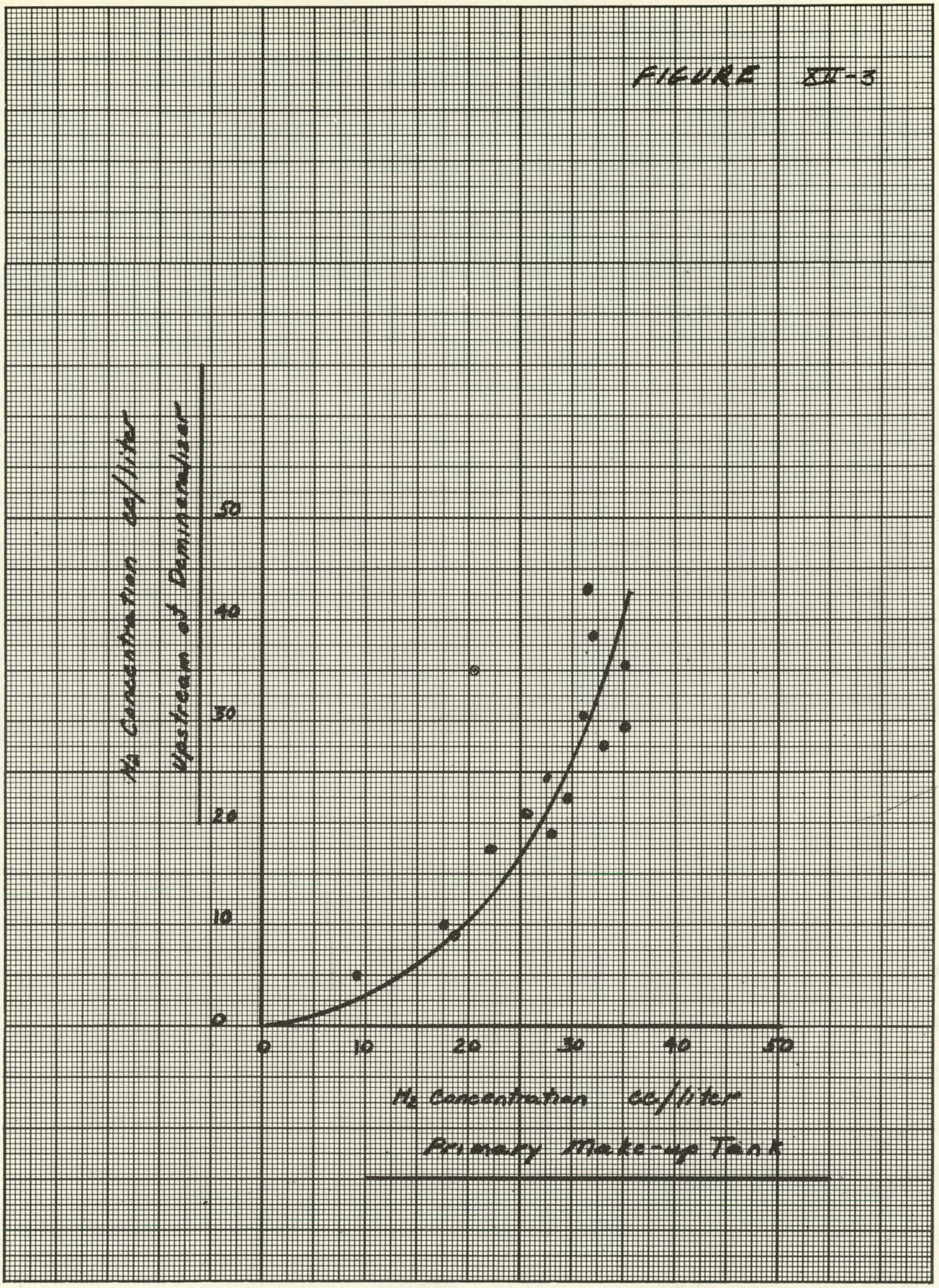


FIG. XII - 3 HYDROGEN CONCENTRATION IN PRIMARY WATER

5.3 Oxygen Control

As mentioned above, on the addition of hydrogen to the primary system, no oxygen was detected in the primary system. Table XII-8 shows the determined oxygen contents in the primary system. Initially, tests were performed for oxygen during every shift, but, since no oxygen was appearing in the primary system the frequency of analyses was reduced.

TABLE XII-8 - OXYGEN RESIDUAL (Startup)

Date	Time	Sample No.	Oxygen Residual (ppm)
4/14	2405	261	0
4/15	0740	281	0
4/16	0020	303	0.03
4/17	1845	335	0.03
4/19	0810	343	0
4/20	0010	355	0.15
4/21	0820	387	0
4/22	0500	408	0
4/27	1625	416	0
4/28	2335	436	0
4/29	2300	460	0
4/30	0730	473	0
5/1	0800	494	0.04
5/2	0820	514	0
5/3	1945	533	0
5/4	1230	553	0

5.4 Chloride

After the system became stabilized, the chloride residual dropped to essentially zero as indicated in Table XII-9.

TABLE XII-9 - CHLORIDE CONCENTRATION (Startup)

Date	Time	Sample No.	Chloride Concentration (ppm)
4/14	2405	261	0
4/15	0740	281	0
4/16	0020	303	0
4/17	0015	325	0
4/19	0810	343	0
4/20	1525	372	0.20
4/21	1545	400	0.10
4/28	0800	422	0
5/1	0800	494	0

5.5 Corrosion Products

The analyses for corrosion products indicated the presence of insignificant amounts of either iron, nickel or manganese. Table XII-10 indicates the results

TABLE XII-10 - CORROSION PRODUCTS (Startup)

Date	Time	Sample No.	Iron (ppm)	Nickel (ppm)	Manganese (ppm)
4/14	1615	275	0.025	-	-
4/16	0020	303	0.040	-	-
4/20	2330	381	0.012	-	-
4/30	0730	473	0.050	0.0	-
5/1	2340	508	0.004	-	-
5/3	2330	536	0.033	0.0	0.0

5.6 Total Solids

The determination of total solids indicated that the solid build-up in the primary system was extermely small and well within specifications. Table XII-11 shows the results.

TABLE XII-11 - TOTAL SUSPENDED AND TOTAL DISSOLVED SOLIDS (Startup)

Date	Time	Sample No.	T. S. S. (ppm)	T. D. S. (ppm)
4/14	1615	275	0.40	0.81
4/16	1750	318	-	0.85
4/22	0500	408	-	1.55
4/30	2000	484	-	0.91
5/3	2330	536	-	0.14

5.7 Resin Performance

The overall performance of the mixed bed resin in the demineralizer was very satisfactory. Since the unit in the APPR-1 must remove both suspended and dissolved solids from the blow down stream an efficiency reduction was expected. The reduction in efficiency was not deleterious to the operation of the primary system.

Both conductivity and pH measurements were made as a measure of the performance of the mixed bed resin. This data is listed in Table XII-12.

TABLE XII-12 - RESIN PERFORMANCE DATA (Startup)

Date	Upstream Resistivity (million ohm-cm)	Downstream Resistivity (million ohm-cm)	Ratio (down/up)	Upstream pH	Downstream pH
4/11	0.5	2.5	5.0	6.7	7.1
4/12	0.45	1.5	3.3	6.4	6.4
4/13	0.3	2.0	6.7	6.4	6.4
4/14	0.3	4.0	7.5	7.3	6.6
7/15	0.7	2.5	3.6	8.0	6.5
4/16	0.7	3.0	4.3	7.9	6.2
4/17	0.7	4.0	5.7	8.5	6.4
4/19	0.75	2.0	2.7	8.5	6.8
4/20	0.4	2.0	5.0	8.5	7.0
4/21	0.3	3.0	10	8.5	6.5
4/22	0.65	2.0	3.1	8.3	6.1
4/27	0.6	2.0	3.3	8.0	6.9
4/28	0.75	2.0	2.7	7.7	7.4
4/29	1.5	4.0	2.7	7.5	6.9
4/30	0.9	2.0	2.2	7.6	6.7
5/1	1.5	2.5	1.7	6.7	6.6
5/2	1.8	3.0	1.7	6.8	6.1
5/3	1.8	3.0	1.7	6.8	6.6
5/4	1.5	2.5	1.7	7.5	6.8

In examining the conductivity data cognizance should be given to the following:

(1) This was very high purity water, even before purification (upstream). Consequently meticulous attention must be given to sources of impurities which could normally be ignored but make a significant contribution in this case. Glassware especially must be scrupulously clean and care should be taken to exclude ionizable gases, such as CO_2, after collection of sample.

(2) Because of impurities leached from resin containers etc., a practical upper limit on downstream conductivity is about 5×10^6.

Some limitations of conductivity measurements as an assay for impurity concentration are listed below:

(1) Conductivity measures only dissolved ionizable solids. Thus any neutral or insoluble solid would not be measured.

(2) The relationship between conductivity and concentration depends upon the nature of the dissolved material. Conductivity as a method for measuring concentration is therefore strictly applicable only for solutions of a known compound.

(3) In the low concentration ranges the conductivity vs. concentration relationships are linear functions. At higher concentrations activity coefficients, dissassociation constant etc. change the shape of the curve.

6. Secondary System Water Treatment During Plant Start-up (L. Heider)

6.1 Chemical Feed

Two chemicals were added to this system, hydrazine and morpholine. The hydrazine was used to scavenge the oxygen whereas the morpholine was added to scavenge both the CO_2 and also raise the pH in the secondary system. Both of these operations were successful. In general, a 1% solution of both chemicals was prepared and fed into the suction of the boiler feed pump so as to allow more reaction time prior to the time the feed water entered the steam generator.

6.2 Oxygen Scavenging

When the reactor reached sufficient power such that the air ejectors became operative, the oxygen scavenging reached the desired specifications for the APPR-1. The air ejectors themselves did not reduce the oxygen concentration to the desired limits so that the chemical treatment was found to be absolutely necessary to reduce the oxygen concentration. Hydrazine was used in various concentrations to scavenge the oxygen. The results are shown in Table XII-13.

TABLE XII-13 - OXYGEN CONCENTRATION IN HOTWELL (Startup)

Date	Time	Sample No.	Oxygen Concentration (ppm)
4/15	1330	286	0.0
4/16	2340	298	0.1
4/20	0030	357	0
4/21	1125	394	0
4/28	1515	430	0
4/29	1615	454	0
4/30	1830	480	0.1
5/1	0820	496	0.67
5/2	0915	516	0.65
5/3	1500	524	0.10
5/4	1515	557	0
5/5	0930	595	0

6.3 Chloride Control

During normal operations the chloride control was very successful. The carry-over of chlorides from the steam generator to the hot-well was for all practical purposes not detectable. Table XII-14 illustrates the results.

TABLE XII-14 - HOTWELL pH AND CHLORIDE CONCENTRATION (Startup)

Date	Time	Sample No.	pH	Chloride Concentration (ppm)
4/15	1330	286	6.8	0.0
4/16	2340	298	8.5	0.1
4/17	0325	329	8.2	0
4/19	2030	350	8.6	0
4/20	0030	357	9.1	0
4/21	0330	385	9.2	0
4/28	0830	424	7.1	0
4/29	0750	446	8.7	0
4/30	0745	475	9.0	0
5/1	0410	496	9.0	0
5/2	0915	516	8.8	0
5/3	1500	524	7.3	0

The maintenance of chlorides below 0.5 ppm was accomplished during practically all of the plant operation. This is illustrated in Table XII-15.

TABLE XII-15 - pH AND CHLORIDE CONCENTRACTION IN STEAM BLOWDOWN (Startup)

Date	Time	Sample No.	pH	Chloride Concentration
4/15	2145	295	8.2	0.58
4/16	0430	305	7.4	0
4/19	2030	351	8.7	0.38
4/20	0200	360	8.6	0.1
4/21	0630	386	9.2	0.1
4/29	0930	449	8.3	0
4/30	1830	483	8.9	-
5/1	0405	492	8.7	0.44
5/2	1010	517	8.9	0.28
5/3	1630	529	8.9	0.51

6.4 Morpholine Control

To maintain a sufficiently high pH in the secondary system, morpholine was added. The pH in the hot-well increased and was maintained at 8.5-9.0 except during periods of sporatic operations. Table XII-14 and XII-16 illustrate the results in the hot-well and in the main steam.

TABLE XII-16 - MAIN STEAM RESISTIVITY AND pH (Startup)

Date	Time	Sample No.	pH	Resistivity (ohm-cm)
4/15	1330	287	6.8	430,000
4/16	2345	299	8.4	300,000
4/20	0035	359	9.0	200,000
4/21	1135	396	9.5	220,000
4/28	1600	433	8.3	390,000
4/29	0810	448	8.5	190,000
4/30	0800	477	8.9	170,000
5/1	0840	498	9.0	250,000
5/3	1600	526	8.0	160,000
5/4	0805	549	8.0	200,000
5/5	0915	597	9.2	160,000

6.5 Secondary System Blowdown Rates.

The maintenance of the chloride limit of 0.5 ppm in the steam generator was generally met at all times. However, the relationship between the blowdown rates and the chloride residual was quite pronounced. Should blowdown be suspended and the chloride in the makeup water either remain the same or be increased due to faulty operation of some piece of equipment, the chloride residual in the steam generator will rise very rapidly. An example of such an instance was during one period when the chlorides did rise to a very large level due to the suspension of all blowdown because of certain difficulties. Fortunately during this period the oxygen residual was at a minimum. Figure XII-4 illustrates the interdependence between chloride concentration in the steam generator and the blowdown rates.

7. Primary System Water Treatment During 700 Hour Test (R. J. Clark)

7.1 Conductivity and pH

Although both pH and conductivity remained relatively constant (Table XII-17) throughout the 700 hour rest, a close surveillance was maintained. These measurements give a reliable estimate of corrosion rates, hydrogen concentration, air in-leakage, and resin exhaustion. Specific resistance of the coolant was usually about 600,000 ohm-cm, as shown in Figure XII-5. Resistance of the demineralizer effluent was about 2 to $3x10^6$ ohm-cm, demonstrating that the resin bed was effectively removing the small amount of impurities present in the water.

Primary coolant pH varied between 7.2 and 8.9 as shown in Figure XII-6 during the 700 hour test, paralleling operating experience in the STR. The alkaline pH was due primarily to the presence of ammonia in the coolant. The ammonia is probably due to the in-leakage of air into the make-up system, since tests at the STR have shown that ammonia is synthesized from nitrogen and hydrogen under the

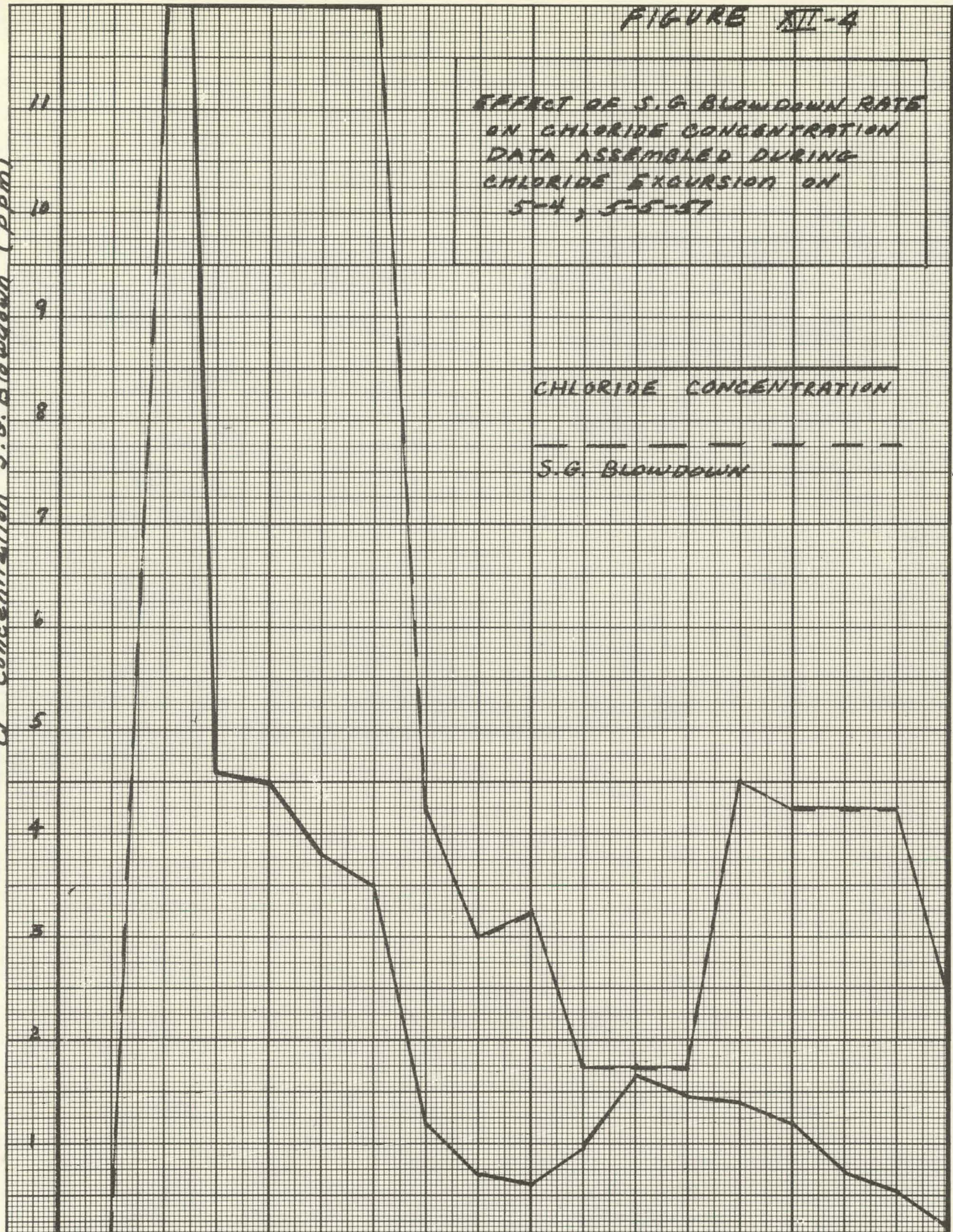


FIG. XII - 4 EFFECT OF STEAM GENERATOR BLOWDOWN RATE ON CHLORIDE CONCENTRATION

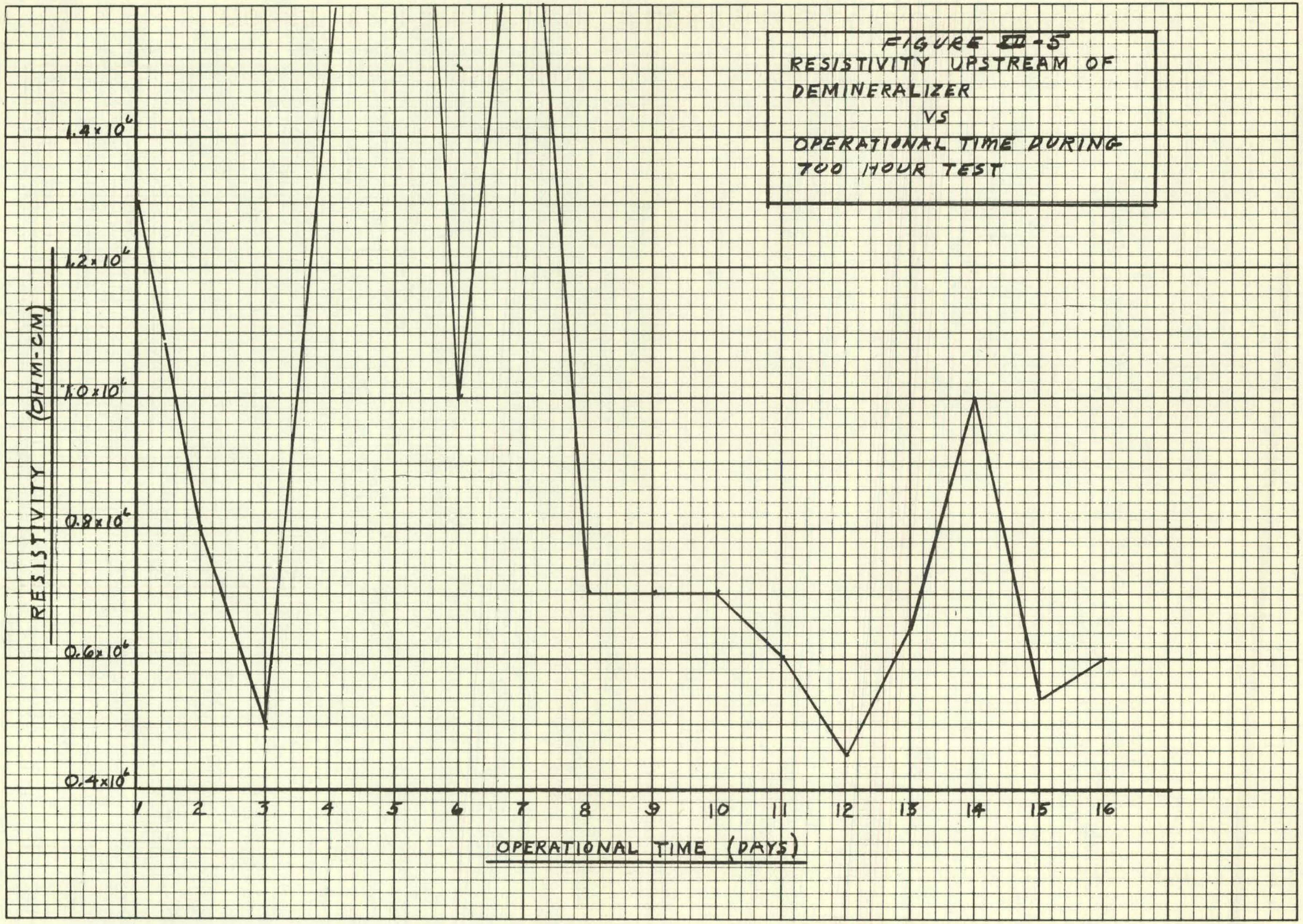


FIG. XII - 5 RESISTIVITY UPSTREAM OF DEMINERALIZER DURING 700 HOUR TEST

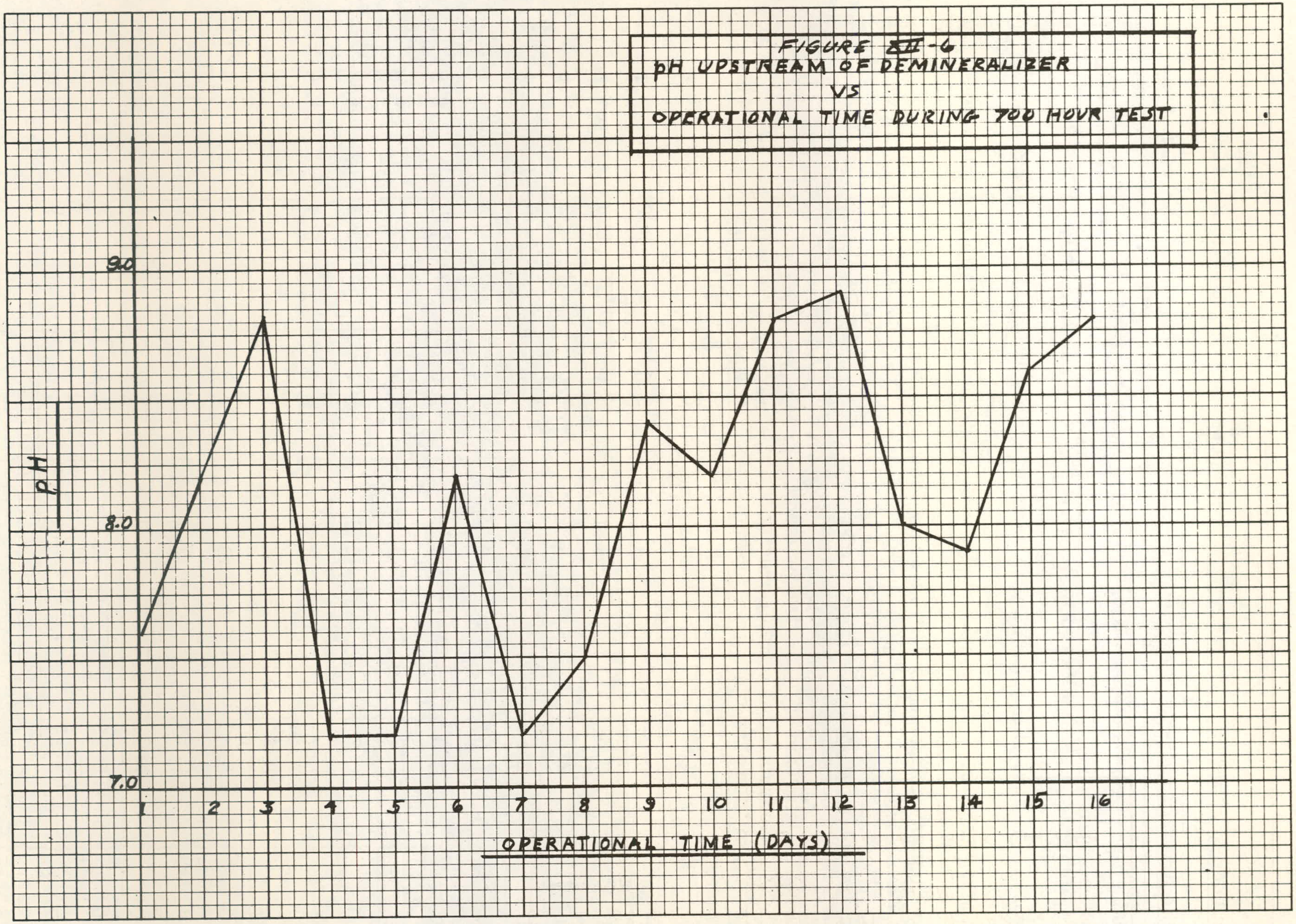


FIG. XII - 6 pH UPSTREAM OF DEMINERALIZER DURING 700 HOUR TEST influence of radiation. As the hydroxyl ion does not contribute to the radioactivity of the primary coolant, and experimental evidence indicates that there are beneficial effects from a higher pH on corrosion rates and prevention of crud deposition, the system is considered to be operating very satisfactorily.

The pH of the demineralizer effluent varied consistently between about 6.6 and 7.2, averaging approximately 6.8. This indicates the resin was effectively removing any ammonia or hydroxyl ions from solution.

TABLE XII-17 - RESIN PERFORMANCE DATA (700 Hour Test)

Date	Upstream Conductivity (million ohm-cm)	Downstream Conductivity (million ohm-cm)	Ratio down/up	Upstream pH	Downstream pH
6/2	1.3	4	3.1	7.6	7.1
6/3	0.8	4	5.0	8.2	7.1
6/4	0.5	4.5	9.0	8.8	6.9
6/5	1.5	4.5	3.0	7.2	7.0
6/6	2.5	4.0	1.6	7.2	6.8
6/7	1.0	2.5	2.5	8.2	5.4
6/8	2.0	4.0	2.0	7.2	6.8
6/9	0.7	4.0	5.7	7.5	6.6
6/10	0.7	4.0	5.7	8.4	6.9
6/11	0.7	3.5	5.0	8.2	6.8
6/12	0.6	4.0	6.7	8.8	6.3
6/13	0.45	4.0	8.9	8.9	7.0
6/14	0.65	2.0	3.1	8.0	6.4
6/15	1.0	2.0	2.0	7.9	6.8
6/16	0.55	2.0	3.6	8.6	7.1
6/17	0.6	2.0	3.3	8.8	7.2

7.2 Oxygen

During the 700 hour test, oxygen was present at times in the primary system, usually to the extent of 0.01 to 0.02 ppm as evident in Table XII-18, but never above 0.2 ppm. Because of the hydrogen release during any oxygen analysis and the corresponding opportunity for in-leakage of air to the sample, all oxygen readings are subject to some error. Without correcting for the error, however, the results indicate that water dissociation in the primary coolant is not appreciable. Although sufficient time was not available to obtain conclusive data, a relationship seemed to exist between the oxygen levels in the primary coolant and the primary make-up tank, the latter usually containing about 0.1 to 0.3 ppm of oxygen.

TABLE XII-18 - OXYGEN RESIDUAL (700 Hour Test)

Date	Time	Sample Number	Oxygen Residual (ppm)
6/1	0830	824	0.0
6/2	0930	849	0.0
6/3	1355	891	0.0
6/4	1450	917	0.0
6/6	0815	968	0.0
6/9	1100	1047	0.01
6/10	0745	1070	0.04
6/11	1120	1095	0.03
6/14	1710	1213	0.02
6/15	1330	1237	0.01
6/16	1230	1261	0.01
6/29	0215	1640	0.04
7/1	0115	1729	0.03

7.3 Hydrogen

Hydrogen concentrations in the primary coolant remained almost constant-28 to 40 cc/kg-throughout the run as is shown in Table XII-19 and Figure XII-7. At no time were any adjustments necessary on the part of the operators to maintain this concentration.

7.4 Chlorides

Chlorides were never detectable in the primary coolant. The high resistivity would discount the possibility of detectable chlorides being present.

7.5 Impurities

Attempts to measure the very low impurity levels in the coolant were only successful with iron and even for this element, only 0.03 to 0.04 ppm were detectable. Nickel, manganese, chromium, or cobalt were never in evidence. The data corrosion product analysis is shown in Table XII-20. The inconsistency of the analysis of ionic impurities shown in Table XII-21 would indicate that levels below 1 ppm are actually below the limit of detection.

7.6 Demineralizer Performance

The data shown in the Table XII-17 on pH and resistivity up and downstream of the demineralizer indicate that it was fulfilling its function more than satisfactorily. The data does show that besides maintaining low ionic impurity levels in the coolant, the resin effectively removed any suspended material in the water.

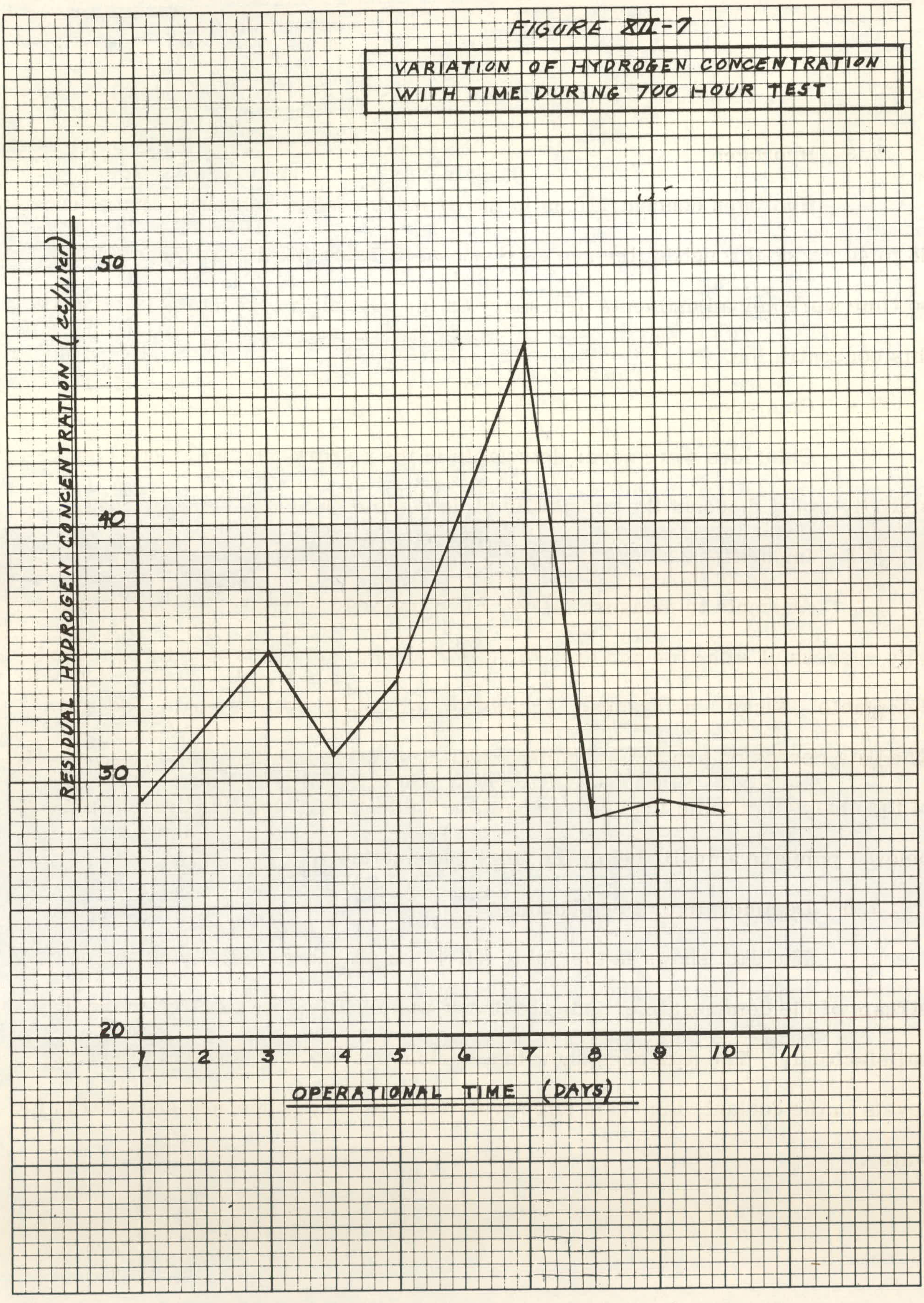


FIG. XII - 7 VARIATION OF HYDROGEN CONCENTRATION DURING 700 HOUR TEST

TABLE XII-19 - HYDROGEN RESIDUAL (700 Hour Test)

Date	Time	Sample Number	Hydrogen Residual (cc/1)
6/1	0830	824	29.4
6/3	1355	891	35.0
6/4	2030	932	31.0
6/5	2025	958	34.0
6/7	2045	993	47.0
6/8	2100	1024	28.4
6/9	1615	1049	29.0
6/10	1412	1075	28.7
6/18	2350	1361	31.7
6/21	0150	1417	27.2

TABLE XII-20 - CORROSION PRODUCTS (700 Hour Test)

Date	Time	Sample No.	Iron (ppm)	Nickel (ppm)	Manganese (ppm)
6/1	0830	824	0.06		
6/2	2100	861	0.0	0.0	0.0
6/3	2005	897	0.04		
6/4	2030	932	0.03		
6/5	2025	958	0.06		
6/8	0415	1003	0.04		
6/9	1615	1049	0.02		

TABLE XII-21 - TOTAL DISSOLVED SOLIDS (700 Hour Test)

Date	Time	Sample Number	T. D. S. (ppm)
6/2	2100	861	0.56
6/3	2005	897	0.57
6/4	2030	932	0.03
6/5	2025	958	0.54
6/7	2045	993	1.40

8. Secondary System Water Treatment During 700 Hour Test (R. J. Clark)

8.1 Oxygen Control

Two means were used during the 700 hour test to scavenge oxygen from the secondary system, feeding hydrazine to the system and use of air ejectors in the hotwell. As shown in Table XII-22, oxygen concentrations in the hotwell were normally about 0.01 ppm. Previous experience had shown that with this low level of oxygen in the hotwell, introduction of hydrazine to the boiler feedwater would reduce oxygen concentrations in the steam generator below detectable limits. Although trouble with the chemical feed system made it difficult to maintain a constant hydrazine residual in the feedwater, oxygen concentrations were always within acceptable limits.

8.2 Chloride Control

A very close surveillance was maintained on chlorides throughout the system, and particularly on make-up water to the system. By monitoring this make-up water almost continuously and adjusting the steam generator blowdown, chloride levels in the steam generator were almost consistently maintained at 0.1 to 0.2 ppm - well below the acceptable limits. Data is summarized in Tables XII-23 and XII-24. Figure XII-8 shows the chloride residual in the steam generator blowdown.

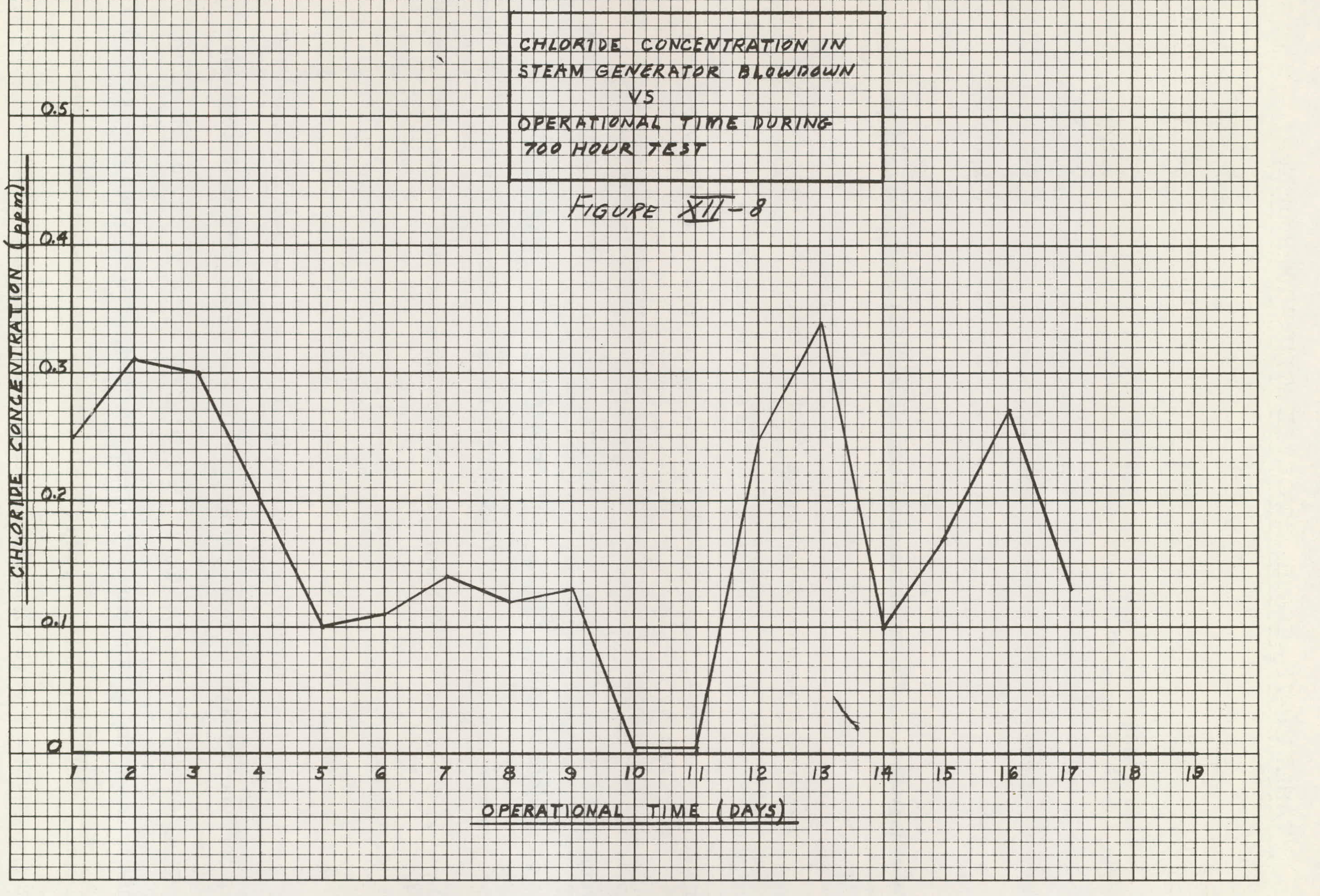


FIG. XII - 8 CHLORIDE CONCENTRATION IN STEAM GENERATOR BLOWDOWN
DURING 700 HOUR TEST

TABLE XII-22 - OXYGEN CONCENTRATION IN HOTWELL (700 Hour Test)

Date	Time	Sample Number	Oxygen Concentration (ppm)
6/3	1845	896	0.0
6/4	0730	911	0.0
6/5	1530	958	0.0
6/6	0910	970	0.02
6/7	1545	988	0.03
6/8	0750	1009	0.0
6/9	0825	1042	0.11
6/10	0755	1071	0.02
6/13	0745	1176	0.04
6/15	0020	1217	0.05
6/16	1000	1258	0.02
6/22	1915	1488	0.009
6/23	1122	1516	0.013
6/26	0025	1581	0.013

TABLE XII-23 - HOTWELL pH AND CHLORIDE CONCENTRATION (700 Hour Test)

Date	Time	Sample No.	pH	Chloride Concentration (ppm)
6/3	0055	868	8.9	0
6/4	0730	911	9.0	0
6/5	0010	934	8.5	0.1
6/6	0910	970	8.9	0
6/7	0305	977	9.2	0
6/8	0300	1001	8.5	0.1
6/9	0830	1042	8.9	0
6/10	0010	1056	9.3	0
6/11	1340	1097	8.8	0.1
6/12	1740	1153	9.0	0
6/13	1750	1183	8.9	0.1
6/18	0822	1341	8.8	0.1
6/19	0800	1374	8.3	0
6/20	1358	1401	9.0	0
6/21	1400	1437	9.1	0.1
6/22	1311	1479	9.0	0
6/23	0750	1514	7.8	0.1
6/24	1430	1545	8.9	0.0
6/25	0150	1554	9.1	0.0
6/26	2115	1594	9.1	0.0
6/27	1005	1606	8.3	0.0
6/28	0027	1622	8.6	0.0
6/29	0815	1657	9.5	0.0
6/30	0130	1683	9.5	0.0

TABLE XII-24 - pH AND CHLORIDE CONCENTRATION IN STEAM GENERATOR BLOWDOWN (700 Hour Test)

Date	Time	Sample No.	pH	Chloride Concentration (ppm)
6/3	0715	882	8.9	0.25
6/4	0510	910	8.5	0.31
6/5	0000	935	7.5	0.30
6/6	0130	964	7.9	-
6/7	0305	976	8.3	0.10
6/8	0300	1000	8.4	0.11
6/9	2340	1030	8.7	0.14
6/10	0000	1057	8.8	0.12
6/11	0600	1094	8.0	0.13
6/12	1845	1158	7.5	0
6/13	0330	1171	8.0	0
6/14	0000	1193	8.6	0.25
6/15	0915	1230	8.2	0.34
6/16	1530	1274	6.9	0.10
6/17	1410	1301	8.0	0.17
6/18	0740	1338	7.3	0.27
6/19	0015	1366	7.7	0.13
6/20	1401	1404	7.5	0
6/21	2345	1415	7.3	0.13
6/22	2330	1466	7.3	0.10
6/23	2330	1495	6.9	0
6/24	2400	1529	8.5	0.10

TABLE XII-24 (Continued)

Date	Time	Sample No.	pH	Chloride Concentration (ppm)
6/25	0150	1557	6.8	0.10
6/26	0027	1585	8.3	0.0
6/27	0020	1602	7.1	0.1
6/28	0026	1621	6.6	0.1
6/29	0115	1638	8.9	0.1
6/30	0130	1686	8.9	0.0
7/1	1810	1742	8.5	0.1

8.3 pH Control

To maintain a pH of about 8.5 in the main steam and condensate systems, morpholine was fed to the boiler feedwater line as required. As shown in Tables XII-23 and XII-25, this was accomplished without any difficulty. Since the secondary is essentially a closed system, only small, periodic additions of morpholine were required to give the desired results.

TABLE XII-25 - MAIN STEAM RESISTIVITY AND pH (700 Hour Test)

Date	Time	Sample No.	pH	Resistivity ohm-cm
6/3	0715	881	8.4	450,000
6/4	0730	913	9.0	300,000
6/5	0730	945	9.1	210,000
6/6	0920	972	9.2	275,000
6/7	1150	986	8.5	400,000

TABLE XII-25 (Continued)

Date	Time	Sample No.	pH	Resistivity ohm-cm
6/8	0800	1011	8.7	280,000
6/9	0830	1044	8.4	300,000
6/10	0805	1073	9.2	200,000
6/11	1350	1099	8.6	300,000
6/12	2345	1112	9.2	190,000
6/13	2330	1166	9.3	210,000
6/14	0105	1192	9.5	130,000
6/15	0030	1219	9.1	180,000
6/16	1005	1260	9.0	190,000
6/17	0805	1293	9.3	195,000
6/18	1645	1350	8.9	250,000
6/19	0010	1365	9.2	240,000
6/20	0000	1390	9.0	210,000
6/21	1620	1441	8.9	250,000
6/22	2330	1465	9.1	295,000
6/23	2330	1494	9.1	250,000
6/24	1610	1550	9.1	280,000
6/25	0740	1568	8.8	325,000
6/26	2120	1596	9.0	275,000
6/27	1545	1614	9.1	280,000
6/29	1530	1666	9.3	150,000
6/30	0130	1685	9.5	130,000

CHAPTER XIII- 700 HOUR ENDURANCE TEST

(J.K. Leslie and R. V. Lichtenberger)

1. Operating Criteria

The final test on the APPR-1 was an endurance run of 700 hours during which period no more than 40 hours down time was allowable. Down time was construed as that time which the plant could not meet the output demand within its design limits due to failure of its equipment, or when it was shut down to make adjustments or repairs.

The directives and agreements governing the operating conditions during the 700-hour test are too voluminous to include in this report, however, the test conditions may be summarized as follows:

a. Initial operation was to consist of a 48 hour xenon run at a power level of approximately 1500 KW electrical (reactor Δ T of 14.6°F).

b. The high power xenon run was to be followed by a low power xenon run with an electrical load of approximately 200 KW.

c. The full load operating condition was established as 34,000 lb/hr. by operating the plant at rated load and maintaining a condenser pressure of 2-1/2 inches of mercury by bleeding air into the condenser.

d. The plant was to be operated for a period of twenty-four (24) hours under varying load conditions. June 29, 1957 was designated for this run.

e. The plant was to be operated for a period of four (4) hours at station load. This run was authorized for July 1, 1957.

f. Load reduction from the full load operating condition was to be permitted as often as necessary in order that the Virginia Electric Power Company (VEPCO) electrical input to the Fort Belvoir System be maintained at a minimum range of 500 to 600 KW.

g. The special tests run during the 700-hour performance test were authorized as part of the 700-hour test.

2. Description of Operation

A chronology of the 700-hour test is given in Section 3 and the operation is shown graphically in Figures XIII-1 through XIII-5. The APPR-1 Gross Electrical output and the Fort Belvoir Net Load are plotted versus time in the Figures. The Fort Belvoir Net Load is the gross load carried by the fort minus the power furnished the fort by the APPR-1. Its significance is explained in the paragraphs following.

Since the temperature of the water from Gunston Cove which was used to cool the condenser had a direct effect on plant performance, this parameter is also plotted in the Figures. Other plant parameters may be obtained from the heat balance calculations given in Chapter VIII.

The 700-hour performance test was started at 2030 June 2, 1957 following performance tests on the steam generator modifications. The load was set at 1500 K. W. electrical generation and maintained at approximately this load for a period of 48 hours. By the end of this time the xenon was approaching equilibrium conditions.

The plant electrical generation was then reduced to 200 KW and maintained at approximately this level for an additional 50 hours. This again allowed the xenon to approach equilibrium conditions. These xenon runs yielded information that will be of considerable value in the design of future reactors.

On June 6, the AEC ordered the steam generator water level raised to assure three inches of water over the top of the tubes. Since superheat cannot be maintained with a high water level in the steam generator it was necessary to restrict the load to 20,000 lb/hour steam flow. Permission was also granted at this time to conduct tests as necessary to establish that the steam generator tubes were not being damaged as a result of low water level. A series of tests were conducted which established that the steam generator level indication was in error and that the water level was satisfactory throughout the operation. This testing and instrument recalibration covered the period from 2230 June 6 through 1900 June 8, with a total down time of 7 hours and 24 minutes.

These tests were among the most severe of all the tests performed throughout the test period and were the "bottling tests" referred to in Figure XIII-1. The plant was operated at 26,000 lb/hr steam flow, the turbine tripped off the line and all trip valves simultaneously closed, thereby completely isolating the steam generator. The primary mean temperature rose to only 454°F and the reactor was driven subcritical on a negative period of 50 seconds. This demonstrated the stability and fast response of the system. A total of four tests of this nature were performed with similar results. For further information on these tests, see references 13 and 14.

During this test period (1550 June 7) a surge, due to an electrical storm on the VEPCO system, caused low voltage which scrammed the reactor. A total of 46 minutes elapsed before the plant was returned to the normal operating condition.

At the conclusion of the steam generator water level tests, a series of superheat tests were run to determine water level and load conditions at which steam superheat could no longer be maintained. This was accomplished by gradually raising the system load while maintaining a constant water level. A throttling calorimeter was used to determine the superheat ahead of the turbine throttle valve. These tests established that superheat could be maintained at rated load with the steam generator water level well above the normal operating level.

After the superheat tests were completed the plant was returned to full load conditions. At this time the Belvoir Net Load became significant. One of the operating criteria for the APPR-1 was that VEPCO must supply part of the load for Fort Belvoir. The portion of the load supplied by VEPCO was known as Belvoir Net Load. The normal minimum Belvoir Net Load has been established as 500 to 600 KW. During periods of low Fort Belvoir load (nights and week ends) it was necessary that the APPR-1 load be reduced to maintain the Belvoir Net Load.

The AEC had specified that the full load steam flow for the APPR-1 was to be established by operating the plant with a net load of 1825 KW with a condenser pressure of 2-1/2 inches of mercury. On June 9, 1957 between 0900 and 1459 tests were run to establish this steam flow. Air was bled into the condenser to raise the pressure to 2-1/2" Hg. as specified by AEC. This was necessary since the condenser circulating water temperature was lower than design, resulting in a lower condenser pressure. As a result of these tests the full load steam flow was established as 34,000 lb/hr as read by the steam flow recorder. This was established as the normal operating load at 1459 on June 9, 1957.

A series of performance tests were authorized and were undertaken at 1515 June 9, 1957. The plant load was increased from station load to full load in increments and data taken at each power level. This data is tabulated in Chapter IX. These tests were concluded at 0820 June 10, 1957 and the plant returned to normal full level conditions. The data taken during these tests were used to check the performance of secondary equipment.

During normal operation at 0228 June 11, ERDL breaker #13 opened on a three phase fault. Breaker "A" (the breaker ties the APPR-1 to the VEPCO network) was tripped by the interlock with breaker #13. This scrammed the reactor and completely shut down the plant. All trip valves automatically closed on loss of plant air.

Since all cooling water is lost on loss of electric power the primary and secondary blowdown flows must be stopped to prevent over heating and damage to other equipment. In this instance the secondary blowdown was not shut off in time, and the radiation monitor on this system ruptured. A new emergency shut down procedure was issued to prevent incidents of this nature in the future.

When breaker #13 was manually reclosed, it opened again, but remained closed when reclosed a second time. This indicated a fault that burned clear on the first manual reclosure. It is suspected that a tree limb in ERDL was blown across a line. The plant was returned to the line at 0456 after a total down time of 2 hours 28 minutes.

During adjustment of a limit switch on 12 June, trip valve TV-1-1 was accidently closed thereby limiting the steam flow to the turbine. This resulted in low load from 0445 to 0515 or 30 minutes.

The APPR-1 was designed for a maximum condenser circulating water temperature of 85^{o}F when operating at rated load. During the 700-hour test the circulating water exceeded this value several times and the plant load was reduced at times to keep operating conditions as close to design as possible. Since water temperature was a limiting factor during the test, it has been plotted on the accompanying figures. On one occasion, the circulating water temperature rose to 94.5^{o}F. During this time the plant carried design load with a condenser pressure of approximately 3in. Hg.

The APPR-1 generator voltage regulator is very sensitive to rate of change of movement of the adjusting rheostat. At 0830 on June 17, 1957, an operator adjusted the generator voltage and over corrected, allowing the excitation to drop to a low value. Since the ERDL electrical system was heavily loaded at the time, the additional reactive load imposed on the system was enough to overload the system. ERDL breaker #13 was opened by phase A and B relays. This in turn tripped breaker "A" by interlock and scrammed the reactor, thereby completely shutting down the plant. At 0840 breaker #13 was reclosed but opened again at 0855 due to malfunction of the relay circuit at substation 234. The reactor control rods had been raised four inches when the second outage occurred. This of course scrammed the reactor again. At 0900 electrical power was restored, but by this time the reactor power had decayed to a level too low to be detected by the operating level nuclear channels. This caused additional delay due to the necessity of starting, warming, and adjusting fission chamber and BF_3 startup channels. The plant was back on the line at 1100 after a total down time of 2 hours 30 minutes.

Permission to run additional load transient tests and a steam generator level indicator recheck was granted by the AEC. The period from 0955 to 1340 on June 19 was used to run these tests, and the transient tests are described in Chapter IX.

June 29 was designated for the varying load demand run. The AEC provided a load schedule that approximated the ERDL load. This run demonstrated plant flexibility. Another run to demonstrate plant flexibility was made on July 1 from 1230 to 1630. The plant was operated at station load with breaker "A" open, thereby isolating the APPR-1 plant. Under these conditions, the APPR-1 must regulate the plant voltage and frequency.

The 700-hour test was concluded at 0030 July 2, 1957 with a total down time of 13 hours and 38 minutes. Of this down time only 7 hours and 24 minutes was due to malfunctioning of APPR-1 equipment. This time was a result of the error in calibration of the steam generator level indicator, and recalibration of the instrument corrected the trouble.

3. Chronology of 700-hour Test

Date	Time		Down Time
2 June	2030	Started 700-Hr. test and high power Xenon run	
4 June	2030	End high power Xenon run Start low power Xenon run	
6 June	1800	S. G. water level raised as per AEC instructions	
6 June	2230	End low power Xenon Run Start S. G. water level tests	
7 June	1550	Outage due to VEPCO	46 Min.
	1636	Back on line after above outage. Continued S. G. tests	
8 June	0204	Tripped turbine for bottling test	44 Min.
	0248	Back on line after above outage	
	0358	Tripped turbine for bottling test	17 Min.
	0415	Back on line after above test	
	0502	Tripped turbine for bottling test	13 Min.
	0515	Back on line after above test	
	1250	Tripped turbine for bottling test. Lowered S. G. level for level indicator check and calibration	6 Hrs. 10 Min.

Date	Time		Down Time
8 June	1900	Back on line after above test. Started check of super heat and resumed operation as permitted by VEPCO input limit.	
9 June	0900	Started condenser back pressure test. 2-1/2 in. Hg.	
	1459	Started 34,000 #/Hr. steam flow as result of above test.	
	1515	Started performance runs from station load to full power in 200 Kw. increments	
10 June	0820	Concluded performance runs. Resumed normal power.	
11 June	0228	Down due to ERDL breaker 13 opening	2 hrs. 28 min.
	1456	Back on line after above outage	
12 June	0445	Low load due to inadvertent closing of trip valve during limit switch adjustment. Stayed on line. Low output only.	30 min.
	0515	Return to full power	
17 June	0830	Down due to ERDL breaker 13 opening caused by operator error in adjusting voltage.	2 hrs. 30 min.
	0840	Breaker closed, started return to critical	
	0855	ERDL breaker 13 opened due to malfunction of relay at Sub Station 234.	
	0900	Breaker 13 reclosed and started return to critical	
	1100	Resumed normal operation	
19 June	0955	Start of transient test and S. G. level indicator recheck	
	1227	Conclusion of above test. Start of low load and load increase transient test.	
	1340	Conclusion of above test and resumption of normal operation	

Date	Time		Down Time
29 June	0001	Start of simulated varying load demand (24 hr. run). Load specified by AEC.	
30 June	0001	Resume normal operation End of varying load.	
1 July	1230	Start station load run Breaker A open.	
	1630	End station load run. Breaker A closed Resume normal operation	
2 July	0030	End 700 hr. test	
		Total Down Time 13 hrs. 38 min.	
		Down Time due to equipment failure 7 hrs. 24 min.	

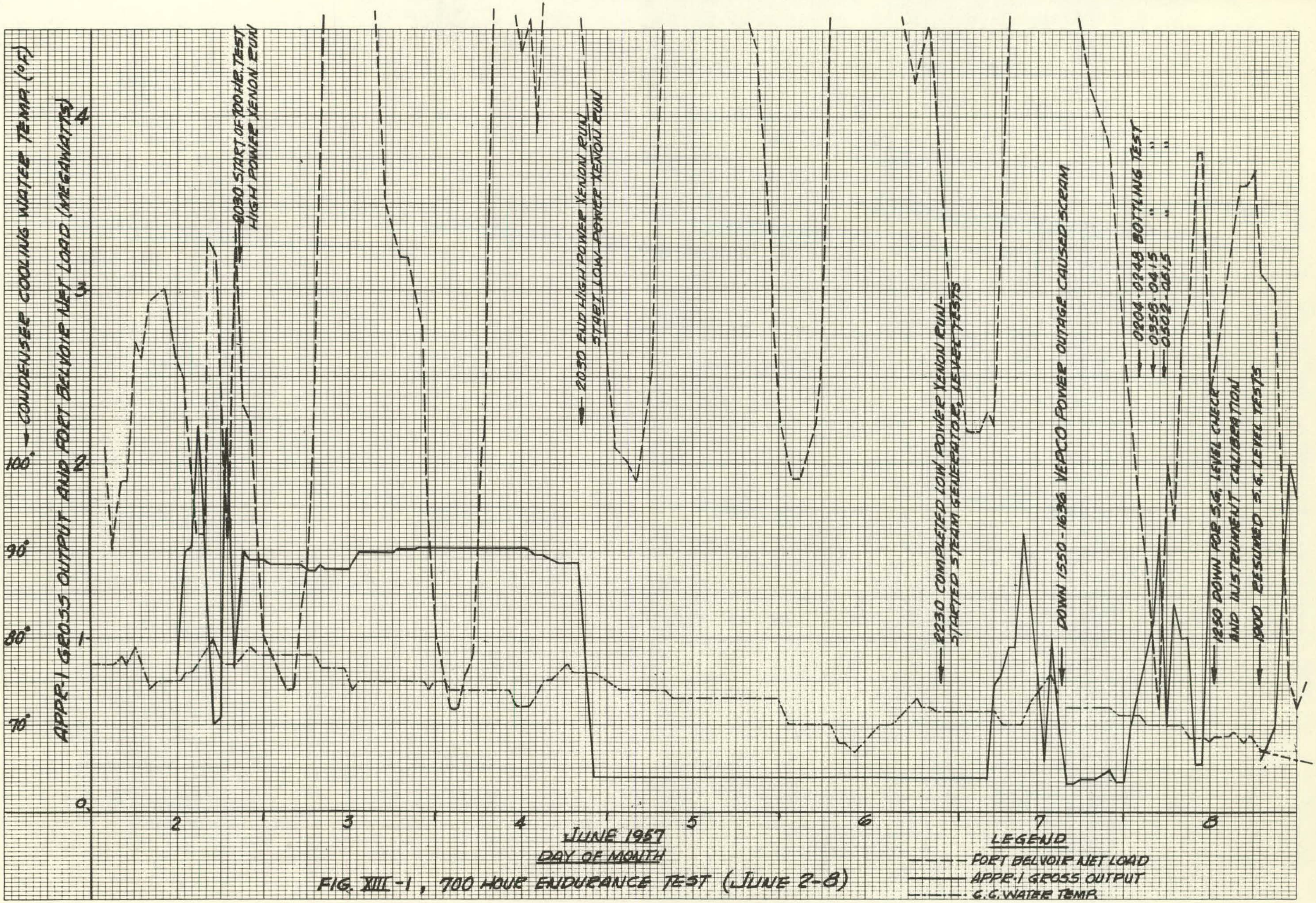


FIG. XIII - 1 700 HOUR ENDURANCE TEST (JUNE 2-8)

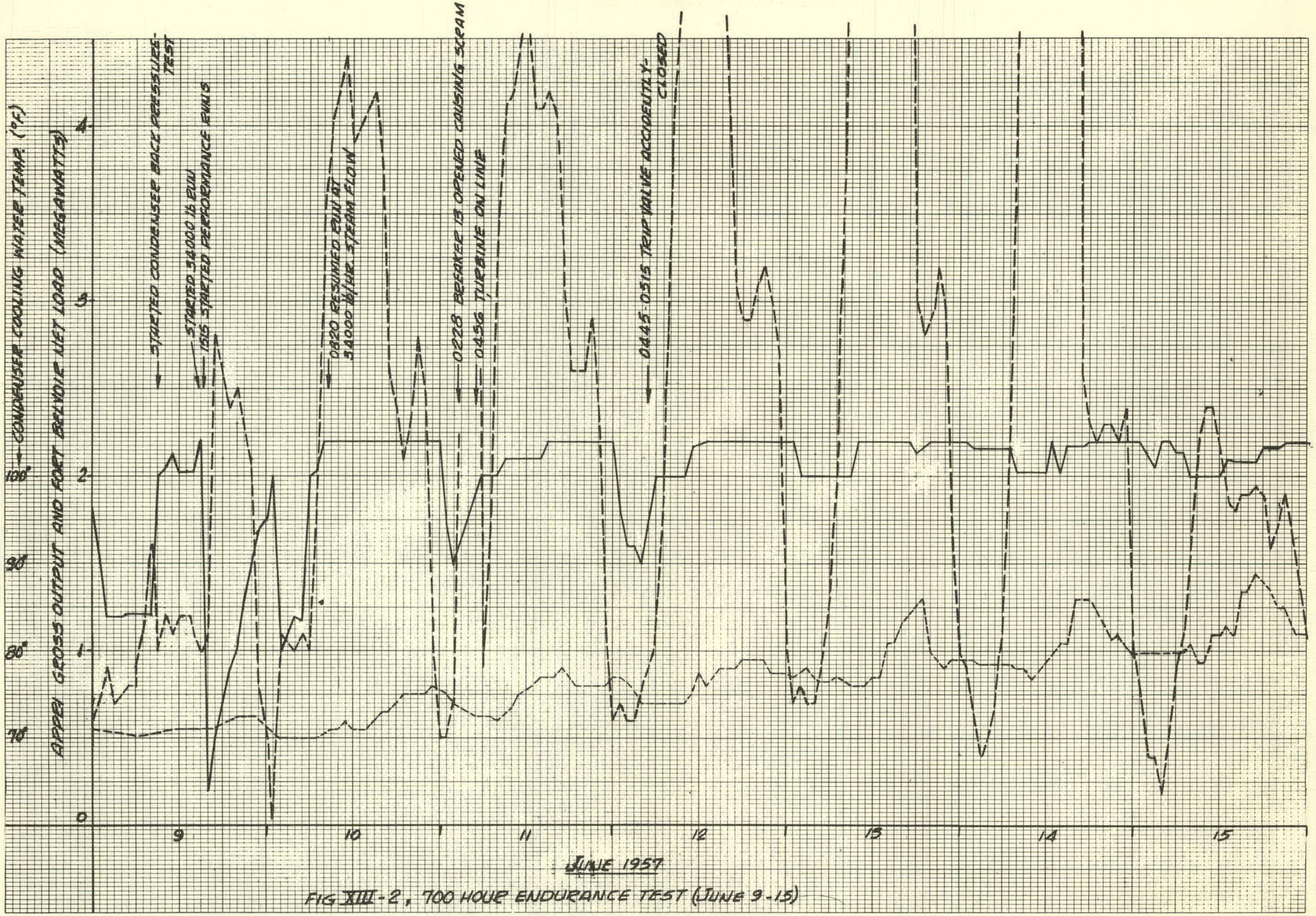


FIG. XIII - 2 700 HOUR ENDURANCE TEST (JUNE 9-15)

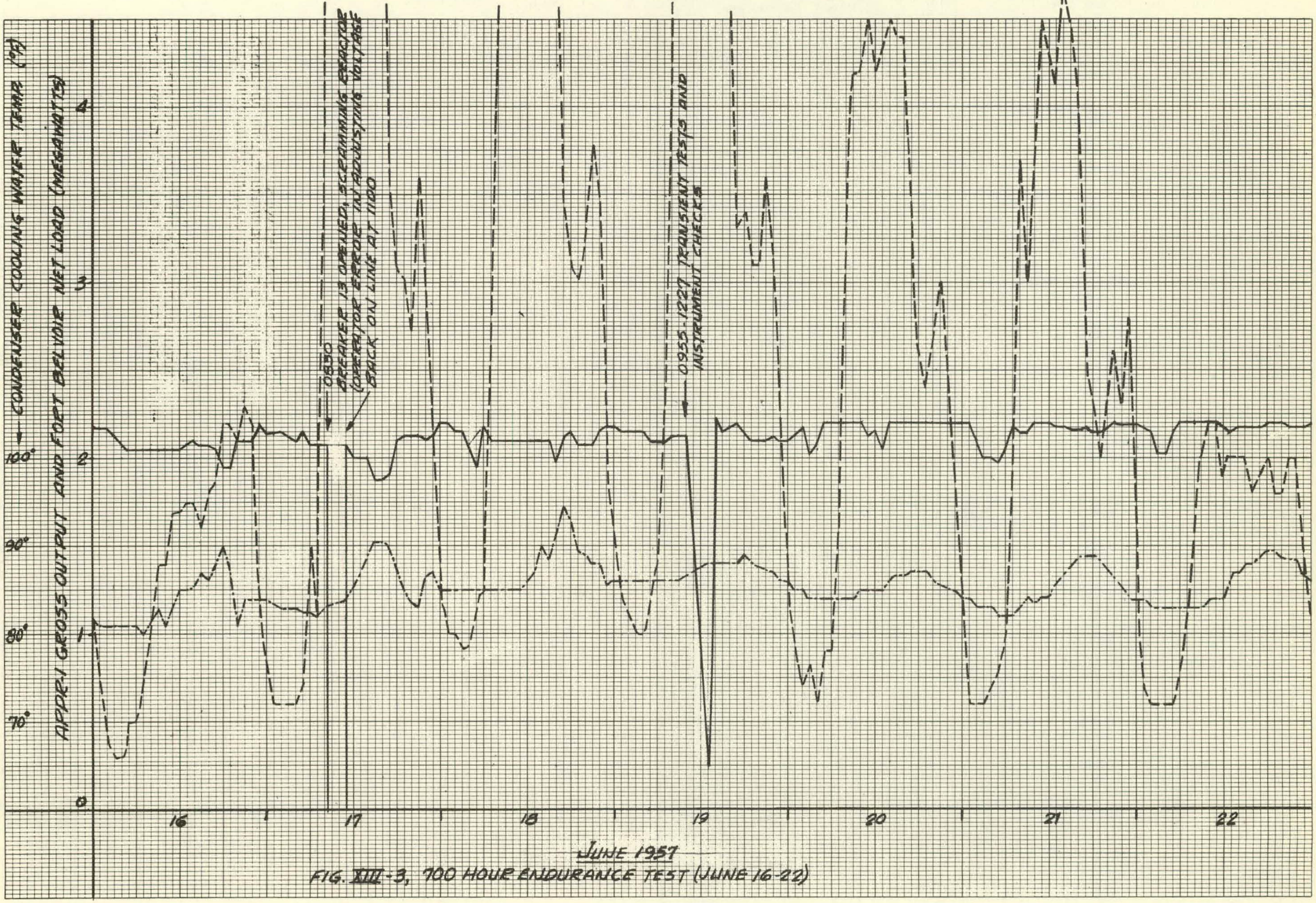


FIG. XIII - 3 700 HOUR ENDURANCE TEST (JUNE 16-22)

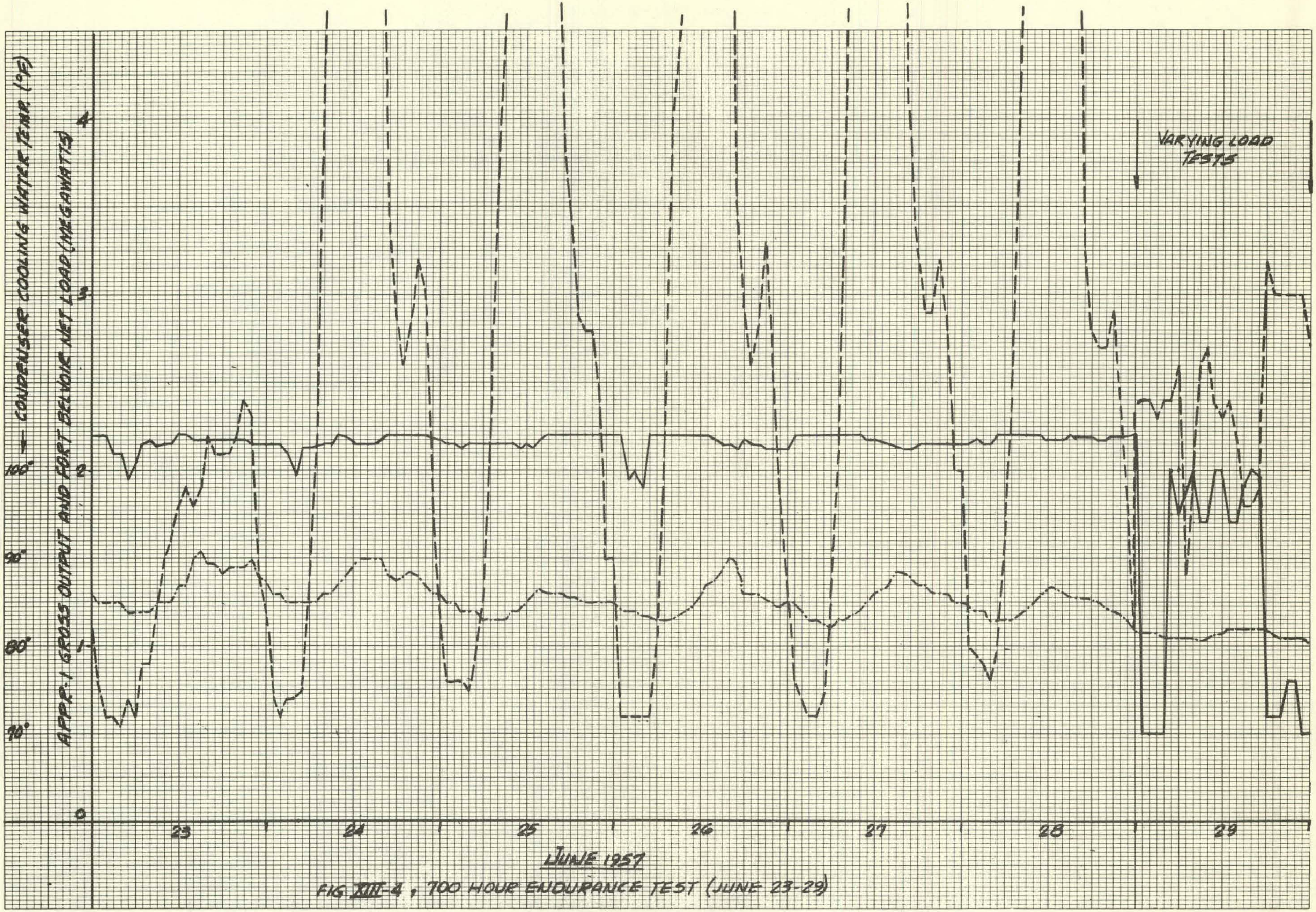


FIG. XIII - 4 700 HOUR ENDURANCE TEST (JUNE 23-29)

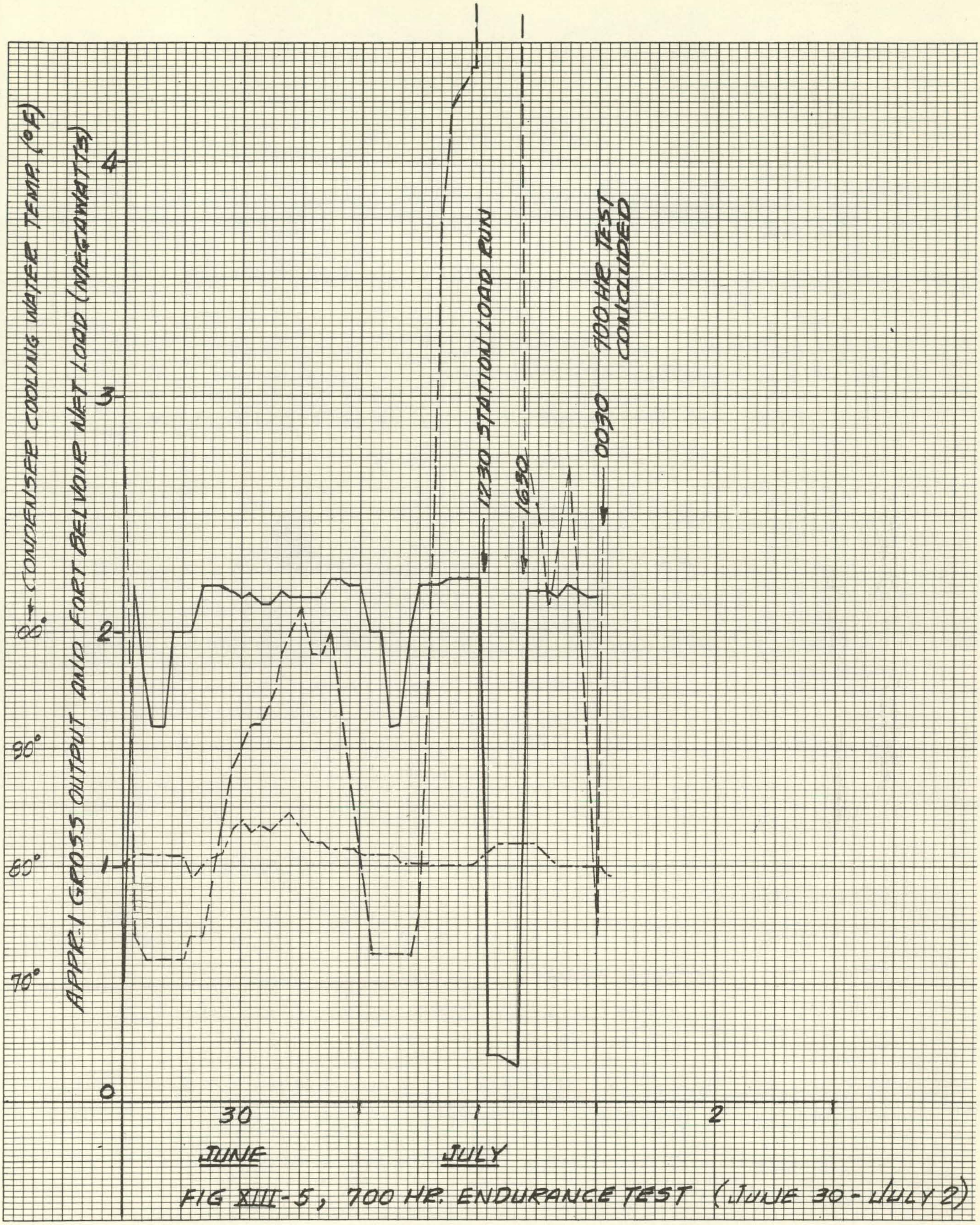


FIG. XIII - 5 700 HOUR ENDURANCE TEST (JUNE 30 - JULY 2)

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