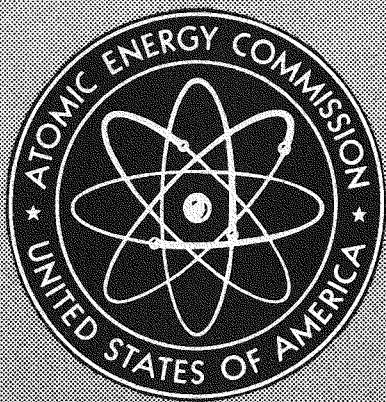




MASTER

MND-M-2935 (Vol. II)

PM-1 NUCLEAR POWER PLANT PROGRAM
PLANT PERFORMANCE STUDIES

Final Periodic Report, September 1, 1962 to December 31, 1962

April 1963

Aerospace Division
Martin — Marietta Corporation
Baltimore, Maryland

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

This report has been reproduced directly from the best available copy.

Printed in USA. Price \$4.00. Available from the Office of Technical Services, Department of Commerce, Washington 25, D. C.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

PM-1 Nuclear Power Plant Program
FINAL PERIODIC REPORT

Plant Performance Studies

Contract AT(30-1)-2345

April 1963



Approved by:

R. L. Baer
Assistant Program Manager

FOREWORD

This report has been prepared by the Nuclear Division of the Martin Marietta Corporation. It is submitted to the U. S. Atomic Energy Commission in compliance with Contract AT (30-1)-2345.

ACKNOWLEDGEMENTS

The following have made significant contributions to these studies:

R. Bagley
R. Hoffmeister
J. Holloway
P. Karnoski
G. Lainas
E. Marsh
J. McNeil
C. Murphy
W. Osmeyer
D. Owings
W. Rankin
P. Reichelsdorfer
H. Rosenthal
J. Sluss
C. Smith
I. Starr

ABSTRACT

Data obtained during the performance testing of the PM-1 plant was compiled and evaluated. The plant powers an Air Defense Command radar station located at Sundance, Wyoming, and is required to supply extremely high quality electrical power (minimum of frequency and voltage fluctuations) even during severe load transients.

The data obtained was compiled into the following format:

- (1) Operating Requirements
- (2) Startup Requirements
- (3) Plant as an Energy Source
- (4) Plant Radiation Levels and Health Physics
- (5) Plant Instrumentation and Control
- (6) Reactor Characteristics
- (7) Primary System Characteristics
- (8) Secondary System Characteristics
- (9) Malfunction Reports.

It was concluded from the data that the plant performance in general meets or exceeds specification. Transient and steady-state electrical fluctuations are well within specified limitations. Heat balance data for both the primary and secondary system agree reasonably well with design predictions. Radiation levels are below those anticipated. Coolant activity in the primary system is approximately at anticipated levels; secondary system coolant activity is negligible.

The core life was re-estimated based on as-built core characteristics. A lifetime of 16.6 Mw-yr is predicted.

CONTENTS

	Page
Foreword	iii
Acknowledgment	iv
Abstract	v
Introduction.	1
I. Operating Requirements	3
A. Effort Required to Operate the Plant	3
B. Effort Required to Maintain the Plant	3
C. Spare Part Consumption.	5
D. Consumption of Operating Supplies.	5
E. Liquid Wastes	5
F. Plant Gaseous Effluents	18
G. Plant Liquid Balance	20
II. Startup Requirements	27
A. Manpower Required for Startup	27
B. Auxiliary Power Required for Startup.	27
C. Time Required for Startup	27
D. Source Strength	31
III. Plant as an Energy Source	35
A. Plant Operating History	35

CONTENTS (continued)

	Page
B. Steady-State Heat Balances	37
C. Crud Buildup	40
D. Steady-State Electrical Characteristics	43
E. Transient Response to Dynamic Changes in Load	44
F. Response of the Reactor to Rod Motion	57
G. Plant Auxiliary Power Requirements	60
H. Scram Data	61
I. Estimated Refueling Time	63
J. Parallel Operation	63
IV. Plant Radiation Levels and Health Physics	65
A. Radiation Levels During Power Operation	65
B. Argon Activity	77
C. Radiation Levels After Shutdown	77
D. Personnel Radiation Exposure Histories	78
V. Plant Instrumentation and Control	81
A. Scram Response Times	81
B. Actuator Power	83
C. Final Instrumentation Scram and Set Points	83
D. Amount of Overlap Between Channels	83
E. Flux to Which Nuclear Instrumentation Is Exposed	86
F. Xenon Override Demonstration	87

CONTENTS (continued)

	Page
G. Instrumentation Recalibration Data	88
H. Radiation Monitor Detection Limits and Alarm Points	88
I. Instrumentation Electrical Power Require- ments	89
VI. Reactor Characteristics	91
A. Estimated Core Life	91
B. Variation in Control Rod Bank Position with Core Life	94
C. Control Rod Worth	94
D. Temperature Coefficient of Reactivity	97
E. Xenon and High Cross-Section Fission Products. . .	98
F. Shutdown Margin.	101
G. Control Rod Speeds	105
H. Pressure Coefficient of Reactivity	108
VII. Primary System Characteristics.	109
A. Primary Coolant Properties	109
B. Primary Coolant Chemical Consumption	114
C. Primary Coolant Pump Power.	115
D. Purification System Resin Life	115
E. Primary System Conditions During Cooldown.	116
F. Shield Water System Steady-State Data	118
G. Shield Water System Coolant.	119
H. Shield Water Demineralizer Life	121

CONTENTS (continued)

	Page
I. Primary System Heat Losses	121
VIII. Secondary System Characteristics.	123
A. Steam Generator Blowdown	123
B. Secondary System Chemical Consumption	123
C. Secondary System Coolant Properties.	125
D. Condensate and Feedwater Pump Cavitation.	125
E. Time Required to Get Turbine Generator on Line	125
IX. Malfunction Reports	131
A. Turbine Governor	131
B. Nuclear Instrumentation.	132
C. Control Rod Actuators	133
X. References	143
Appendix A--Plant Transient Data.	145
Appendix B--Experimental Determination of Gamma and Neutron Fluxes.	147
A. Thermal Flux Measurements in Reactor and Steam Generator Packages.	147
B. Interconnect Thermal Flux Measure- ments	151
C. Fast Flux	152
D. Gamma Dose Rate Measurements	155
Appendix C--Analytical Determination of Neutron Fluxes	157

CONTENTS (continued)

	Page
A. Thermal Flux	157
B. Fast Flux	158
C. Argon Activity Calculations	159
Appendix D--Nuclear Analytical Techniques	163
A. Initial Reactivity	164
B. Control Rod Worth	167
C. Core Lifetime	170
D. Xenon Reactivity Effects	171

INTRODUCTION

The PM-1 plant is a prepackaged air-transportable nuclear power plant, having an output of one megawatt (e) and seven million Btu/hr of heat. The plant electrical output is utilized by an Air Defense Command Radar Station located near Sundance, Wyoming. Extremely high quality electrical power is required, even under severe transient conditions.

Data obtained during the performance testing of the plant was compiled and evaluated. This data is presented in the following format:

- (1) Operating Requirements
- (2) Startup Requirements
- (3) Plant as an Energy Source
- (4) Plant Radiation Levels and Health Physics
- (5) Plant Instrumentation and Control
- (6) Reactor Characteristics
- (7) Primary System Characteristics
- (8) Secondary System Characteristics
- (9) Malfunction Reports.

I. OPERATING REQUIREMENTS

The PM-1 plant operating requirements, based on actual operating experience, are described in this chapter under these headings:

- (1) Effort Required to Operate the Plant
- (2) Effort Required to Maintain the Plant
- (3) Spare Part Consumption
- (4) Operating Supply Consumption
- (5) Liquid Radioactive Wastes
- (6) Gas Fluids Discharged from the Plant
- (7) Plant Liquid Balance

A. EFFORT REQUIRED TO OPERATE THE PLANT

The number of men on each shift during plant operation was tabulated; a summary is presented in Table 1. As this table shows, two to four men were utilized per shift, with a three-man crew the most common. The normal military crew consists of a control room operator, an equipment operator and a shift supervisor. The average operating crew effort was 24.1 man-hours per shift; for a 30-day month, a total of 2170 man-hours would be required.

Although the crew members each have a skill specialty, this was not a significant item in the crew composition.

The organization of the Air Force Crew, immediately subsequent to the transfer of the plant, is shown in Fig. 1. The intention is to train the operation and maintenance crews so that any crew can perform either effort. This will permit the Air Force to handle both plant operation and maintenance efforts with five three-man crews, each of which is trained to perform both jobs. The crews are to be rotated as to function on a regular schedule. The five crews will provide seven-day week, three-shift operating coverage and five-to-seven-day-week, one-shift maintenance coverage. This corresponds quite closely to the actual manpower utilized during the plant test program.

B. EFFORT REQUIRED TO MAINTAIN THE PLANT

The number of men engaged in plant maintenance during periods of plant operation and the type of maintenance effort conducted are presented in Table 2. The average daily effort was 26 hours, nearly 1/3 of which was in maintenance supervision and general plant cleaning and upkeep.

TABLE 1

Summary of Manpower Utilized to Operate Plant

Period Covered	April 10, 1962 to October 2, 1962
<u>Number of Men per Shift</u>	<u>Total Number of Shift</u>
2	13
3	232
4	19

TABLE 2

Effort to Maintain Plant
(Man-Hours)

Date	Mechanical	Instrument.	Electrical	Health Physics and Water Chemistry	Supervision and General Cleaning and Upkeep	Total
6/15/62	4	6		8	22.75	40.75
6/16/62				10	4	14
6/17/62				5		5
6/18/62	6	12	1	5.5	21.5	46
6/19/62	6	6		7.5	27.5	47
6/20/62	5	18	3	8	8	42
6/21/62	6	12		4.5	12.5	35
6/22/62		8	2	11	5	26
7/23/62	8			14.5	17	39.5
7/24/62		8		16	16	40
7/25/62		8		16		24
7/26/62		12	2	11	18	43
7/28/62				7	1	8
7/29/62				4	4	8
7/30/62		12	4		8	24
7/31/62		8		8	13	29
8/1/62		8	2	10	20	40
8/2/62		8		10	22	40
8/3/62		12		4	8	24
8/4/62		21	4			25
8/6/62		8		8	8	24
8/7/62		2		8	8	18
9/12/62				8		8
9/13/62		14		16		30
9/14/62		8		14		22
9/15/62		8		5.5		13.5
9/16/62		8		7.5		15.5
9/17/62		14		9		23
9/18/62						
9/19/62		16		10.5		26.5
Total Hours	35	237	18	246.5	244.25	780.75
Total Days	30					
Average Hours/ Day	1.167	7.9	0.6	8.22	8.15	26.0

Applying the average daily effort to a 30-day month yields to a total of 780 man-hours per month for maintenance efforts.

A design objective of the PM-1 plant is a maximum yearly down time of 21 days; that is, plant availability of 94.3%. This corresponds closely to 400 hours of continuous operation without exceeding 24 hours of down time, a condition which was achieved during the plant test program. Plant operation from the time of transfer to the Air Force indicates that the specified availability percentage is being approached during routine operation. It is probable that the design objective will be achieved with the present level of maintenance effort.

C. SPARE PART CONSUMPTION

The consumption of spare parts is indicated in Tables 3 through 6. Table 3 lists all spare parts consumed, defines the type of part and shows the date the part was actually utilized. The various parts are listed in the order in which it was noted repairs were required. Tables 4 through 6 are recompilations of Table 3 by type of spare part; Table 4 lists electrical, Table 5 lists mechanical, and Table 6 lists instrumentation spare parts. These items are also listed in the order in which it was noted they were required.

Items which required adjustment or cleaning, but which did not result in the consumption of spare parts are not included in the tabulations.

The data presented on spare consumption represents only the experience obtained during the test program. It is impossible, without further plant operating experience, to extrapolate this data to estimate future spare part consumption rates.

D. CONSUMPTION OF OPERATING SUPPLIES

Plant supplies, other than water treatment chemicals, which were consumed during the calendar quarter from July 1 to September 30, 1962 are itemized in Table 7. Data pertaining to the consumption of water treatment chemicals is presented in Sections VII-B and VIII-B of this report for the primary and secondary systems, respectively.

E. LIQUID WASTES

Data pertaining to plant liquid waste was compiled and is presented in Table 8. This data was utilized to determine the daily rate of waste accumulation, presented in graphical form in Fig. 2. The monthly rates of accumulation, in gallons, were determined from Fig. 2 as:

May	2450
June	3040
July	1330
August	1460
September	1915

Most of the liquid waste is slightly radioactive laundry water whose accumulation is essentially continuous. Water available at the plant site usually has a natural activity nearly at the MPC value of 10^{-7} $\mu\text{c}/\text{cm}^3$, and occasionally exceeding this value. Thus, only a few drops of moderately radioactive water are sufficient to make the activity of a number of washer loads exceed MPC activity levels.

The data presented in this section reflects waste accumulation after the installation of a 250 gallon-laundry holdup tank. This tank permits monitoring of the activity of the water discharged from the washing machine to determine whether or not it needs to be processed prior to dumping into the sewerage system. Before installation of this tank, all the laundry wash water and all the dryer condensing water was processed by the waste disposal system.

TABLE 3
Spare Parts Consumption

<u>Part Description</u>	<u>Type of Spare Part</u>	<u>Date Used</u>
2--Flexitalic gaskets--21-15F --21-15B	Mechanical	5/7/62
Valve internals--1/2-inch glove	Mechanical	5/7/62
Spindle for 4-inch gate valve plus packing and gaskets	Mechanical	5/9/62
Stem for 6-inch gate valve plus packing and gaskets	Mechanical	5/7/62
Circuit breaker for primary coolant pump	Electrical	5/4/62
Circuit breaker for feedwater pump	Electrical	5/4/62
Pipe insulation, 10 feet for 2-inch pipe	Mechanical	5/10/62
2--1/2-inch copper tubing coupling	Instrument.	5/7/62
4--1/2-inch close nipples		
2--400-6 Swagelok fitting		
1--400-7 Swagelok fitting		
22--400-2-1 Swagelok fitting		
22--403-1 Swagelok fitting		
22--404-1 Swagelok fitting		
8 feet--1/4-inch tubing		
Replaced broken pin on air filter paper drive monitor	Instrument.	5/7/62
New dial face on d-c current meter	Instrument.	6/26/62
New 6-inch gate, solenoid-operated valve	Mechanical	5/9/62
Solenoid 27-501-1 for drain valve DV02-VL9	Electrical	6/1/62
Heater for spent core circulating pump	Electrical	8/1/62
Pacific valve seal kit for hydrogen manifold	Mechanical	8/1/62
Thermoswitches for silver nitrate reactor	Electrical	5/25/62
1-810-1-3 Swagelok fitting on hotwell D/P cell	Instrument.	5/25/62
2-402-1 Swagelok fitting on hotwell D/P cell D/P cell		
2-403-1 Swagelok fitting on hotwell D/P cell		
2-404-1 Swagelok fitting on hotwell D/P cell		

TABLE 3 (continued)

<u>Part Description</u>	<u>Type of Spare Part</u>	<u>Date Used</u>
Condenser air cooler section controller	Instrument.	7/ 16/ 62
2 each micrometallic filters, cartridge No. 2NSMX-G	Mechanical	5/ 22/ 62
Bonnet and steam 1-inch glove valve (1500 psi)	Mechanical	5/ 22/ 62
Auxiliary boiler water pump oil seal PN 12594	Mechanical	5/ 23/ 62
Auxiliary boiler valve cap gaskets PN 12379	Mechanical	
Nipple 1/2-inch on boiler tube inlet	Mechanical	
6--10-ohm, 2-watt resistors	Instrument.	5/ 23/ 62
1--4.7-ohm, 2-watt resistors		
1--2N158A transistor		
1--2N335A transistor	Instrument.	6/ 2/ 62
1--1/4-inch Whitey valve	Mechanical	5/ 25/ 62
1--1/4-inch Swagelok tee		
Junction box cover	Electrical	5/ 28/ 62
Thermocouple	Instrument.	5/ 29/ 62
New gage on decontamination pump	Mechanical	5/ 30/ 62
1--1/2-inch gage valve	Mechanical	7/ 9/ 62
Disk and seat for 2-inch gate valve	Mechanical	6/ 25/ 62
1--2N414 transistor	Instrument.	7/ 25/ 62
Pressurizer datum column	Mechanical	6/ 6/ 62
Steam generator vent line pipe support	Mechanical	6/ 9/ 62
Pressurizer relief valve expansion joint	Mechanical	6/ 5/ 62
1--Anton GM Tube No. 112	Instrument.	6/ 7/ 62
1--1/2-inch globe valve	Mechanical	6/ 8/ 62
1--1/2-inch x 4-inch pipe nipple		
2--lengths gage glass	Instrument.	6/ 12/ 62
2--1-inch gaskets Flexitalic No. 24600	Mechanical	6/ 12/ 62
2--1-inch gaskets Flexitalic	Mechanical	6/ 7/ 62

TABLE 3 (continued)

<u>Part Description</u>	<u>Type of Spare Part</u>	<u>Date Used</u>
1--BF3, WL 6307 tube (compensated)	Instrument.	6/8/62
Cable connector	Electrical	6/8/62
18-foot appliance wire	Electrical	6/21/62
1--3/8-inch Hoke valve	Mechanical	6/11/62
1--1/2-inch to 3/8-inch reducing bushing	Mechanical	
1--1/2-inch, 90-deg ell	Mechanical	
1--1/2-inch nipple	Mechanical	
2 foot--1/8-inch packing	Mechanical	6/12/62
Spindle, packing, gaskets for 4-inch gate valve	Mechanical	6/22/62
D-battery	Electrical	6/25/62
3--2-ampere fuses	Instrument.	6/25/62
1--indicator light		
1--resistor	Instrument.	6/14/62
1--D/P cell amplifier	Instrument.	6/22/62
Bistable on steam generator level	Instrument.	6/26/62
1--1/2-inch 3-4-600 Flexitalic gasket	Mechanical	6/19/62
30-inches--5/16 valve packing	Mechanical	6/21/62
Loop balance resistor	Instrument.	6/21/62
Garlock gasket on turbine throttle valve drain line	Mechanical	6/20/62
1--oil seal (M-136)	Mechanical	6/26/62
20 feet--1/2-inch copper tube		
1--valve cover and gasket		
Rod lift power supply diodes	Instrument.	6/22/62
O-ring--CRDM vent valve	Mechanical	6/24/62
Pressure gage--RWDS steam reducing station	Mechanical	6/25/62
Conductivity recorder drive cable	Instrument.	6/26/62
Pilot valve, 3/4-inch, 1500-psi globe valve	Mechanical	6/25/62
Pilot valve, 3/4-inch, 1500-psi globe valve	Mechanical	6/26/62

TABLE 3 (continued)

<u>Part Description</u>	<u>Type of Spare Part</u>	<u>Date Used</u>
Capacitor C-4	Instrument.	6/27/62
Transistor 2N336	Instrument.	6/27/62
1--6UB4 tube		7/2/62
1--6AL5 tube		
3--NE51 bulbs	Instrument.	7/2/62
Gasket--diesel	Mechanical	7/6/62
Valve--portable demineralizer inlet	Mechanical	8/11/62
4--1N1184A transistor	Instrument.	7/9/62
3--4-microfarad capacitor	Instrument.	8/2/62
1--400-2-4 Swagelok	Instrument.	7/31/62
4--402-1 Swagelok		
4--403-1 Swagelok		
4--404-1 Swagelok		
Control console touch-up paint, 1/6 pt	Instrument.	8/1/62
1--gage glass	Instrument.	8/1/62
Hand drill repair parts	Mechanical	8/10/62
Ion chamber cables, connectors, can and chamber	Instrument.	8/4/62
1--2N336 transistor	Instrument.	8/15/62
6--1/4-inch NC bolts, 3/4-inch long	Mechanical	8/15/62
Jacuzzi pump packing	Mechanical	8/10/62
2--3AG-type, 2-ampere fuses	Electrical	8/12/62
1--SCR 2400 power supply	Instrument.	8/17/62
1--2N6104 transistor	Instrument.	8/17/62
1--Crosby gage glass	Instrument.	8/21/62
5 inches--1/2-inch packing	Mechanical	8/30/62
3 feet--3/16-inch packing	Mechanical	9/1/62
Fuel solenoid--diesel	Mechanical	9/19/62
1--amprex 100 c tube	Instrument.	9/15/62

TABLE 3 (continued)

<u>Part Description</u>	<u>Type of Spare Part</u>	<u>Date Used</u>
6-inches sight glass	Instrument.	9/19/62
5Y3 tube scaler	Instrument.	9/24/62
Condenser pressure regulator	Instrument.	9/25/62
1--6AL5 tube	Instrument.	9/27/62
1--6U8 tube		
1--transistor 1N54A		

TABLE 4

Electrical Spare Parts Consumption

<u>Part Description</u>	<u>Date Used</u>
Circuit breaker--primary coolant pump	5/4/62
Circuit breaker--feedwater pump	5/4/62
Solenoid 27-501-1	6/1/62
Heater for spent core circulating pump	8/1/62
Thermo switch for silver nitrate reactor	5/25/62
Junction box cover	5/28/62
Cable connector	6/8/62
18-foot appliance	6/21/62
D-battery	6/25/62
2 each 3AG-type, 2-ampere fuses	8/12/62

TABLE 5

Mechanical Spare Parts Consumption

<u>Part Description</u>	<u>Date Used</u>
Flexitalic gasket--21-15F	5/7/62
Flexitalic gasket--21-15B	
Valve internals, 1/2-inch globe	5/7/62

TABLE 5 (continued)

<u>Part Description</u>	<u>Date Used</u>
Spindle, packing gaskets, 4-inch gate valve	5/9/62
Steam, packing gaskets, 6-inch gate valve	5/7/62
10 feet--2-inch pipe insulation	5/10/62
6-inch solenoid-operated gate valve	5/9/62
Pacific valve seal kit for hydrogen manifold	8/1/62
2 each micrometallic filter cartridges No. 2NSMX-G	5/22/62
Bonnet and stem, 1-inch globe valve, 1500 psi	5/22/62
Auxiliary boiler water pump oil seal--PN 12594	5/23/62
Auxiliary boiler valve cap gaskets--PN 12379	
Nipple 1/2-inch boiler tube inlet	
1--1/4-inch Whitey valve	5/28/62
1--1/4-inch Swagelok tee	
Gage on decontamination pump	5/30/62
1-1/2-inch globe valve	7/9/62
Disk and seat for 2-inch gate valve	6/25/62
Pressurizer datum column	6/6/62
Steam generator vent line pipe support	6/9/62
Pressurizer relief valve expansion joint	6/5/62
1/2-inch globe valve	6/8/62
1/2-inch x 4-inch pipe nipple	
2--1-inch Flexitalic gaskets No. 24600	6/12/62
2-inch Flexitalic gaskets	6/7/62
3/8-inch Hoke valve	6/11/62
1/2-inch to 3/8-inch reducing bushing	
1/2-inch, 90-deg ell	
1/2-inch nipple	
2 feet of 1/8-inch packing	6/12/62
Spindle packing gaskets for 4-inch gate valve	6/22/62
1/2-inch Flexitalic gasket 3-4-600	6/19/62
30 inches of 5/16-inch valve packing	6/21/62

TABLE 5 (continued)

<u>Part Description</u>	<u>Date Used</u>
Garlock gasket on turbine throttle valve drain line	6/20/62
Oil seal M-136	6/26/62
20 feet of 1/2-inch copper tubing	
Valve cover and gasket	
O-ring--CRDM vent valve	6/26/62
Pressure gage RWDS steam reducing station	6/25/62
Pilot valve, 3/4-inch, 1500-psi globe valve	6/25/62
Pilot valve, 3/4-inch, 1500-psi globe valve	6/26/62
Gasket--diesel	7/6/62
Valve--portable demineralizer inlet	8/11/62
Hand drill repair parts	8/10/62
6--1/4-inch NC bolts, 3/4-inch long	8/15/62
Jacuzzi pump packing	8/10/62
5 inches of 1/2-inch packing	8/30/62
3 feet of 3/16-inch packing	9/1/62

TABLE 6

Instrument Spare Parts Consumption

<u>Part Description</u>	<u>Date Used</u>
2--1/2-inch copper tubing couplings	5/7/62
4--1/2-inch close nipples	
2--400-6 Swagelok fittings	
1--400-7 Swagelok fittings	
22--400-2-1 Swagelok fittings	
22--403-1 Swagelok fittings	
22--404-1 Swagelok fittings	
8 feet--1/4-inch tubing	
Pin on air filter monitor paper drive	5/7/62
Dial face for d-c current meter	6/26/62

TABLE 6 (continued)

<u>Part Description</u>	<u>Date Used</u>
1--810-1-3 Swagelok fitting on hotwell D/P cell	5/25/62
2--402-1 Swagelok fitting on hotwell D/P cell	
2--403-1 Swagelok fitting on hotwell D/P cell	
2--404-1 Swagelok fitting on hotwell D/P cell	
Condenser air cooler section controller	7/16/62
6--10-ohm, 2-watt resistors	5/23/62
1--4.7-ohm, 2-watt resistors	
1--2N158A transistor	
1--2N335A transistor	6/2/62
Thermocouple	5/29/62
2N414 transistor	7/25/62
Anton GM Tube No. 112	6/7/62
2 lengths gage glass	6/12/62
1--BF3 WL6307 tube (compensated)	6/8/62
3--2-ampere fuses	6/25/62
Indicator light	
Resistor	6/14/62
D/P cell amplifier	6/22/62
Steam generator level bistable	6/26/62
Loop balance resistor	6/21/62
Rod lift power supply diode	6/22/62
Conductivity recorder drive cable	6/26/62
Capacitor C-4	6/27/62
Transistor 2N336	6/27/62
Tubes 6UB8	7/2/62
Tubes 6AL5	
3--NE51 bulbs	
4--1N1184A transistor	7/9/62
3--4 microfarad capacitor	8/2/62
1--400-2-4 Swagelok fittings	7/31/62
4--402-1 Swagelok fittings	
4--403-1 Swagelok fittings	
4--404-1 Swagelok fittings	

TABLE 6 (continued)

<u>Part Description</u>	<u>Date Used</u>
Control console touch-up paint, 1/6 pt.	8/1/62
Gage glass	8/1/62
Ion chamber, cables, connectors, can chamber	8/4/62
2N336 transistor	8/15/62
SCR 2400 power supply	8/17/62
2N6104 transistor	8/17/62
Crosby gage glass	8/21/62
Amprex 100c tube	9/15/62
6-inch sight glass	9/19/62
5Y3 tube, scaler	9/24/62
Condenser pressure regulator	9/25/62
Tubes 6AL5	
Tubes 6U8	
Tubes 68H5	
Transistor 1N54A	

TABLE 7

Consumption of Operating Supplies
(July 1, 1962 to Sept 30, 1962)

Grease and Oils

5000 gal diesel fuel
50 lb Dex-2 grease
37 lb Solvo grease
10 gal No. 60 oil
55 gal Stoddard solvent
52 gal 2190 TEP lubricating oil

Gases

5 cylinders nitrogen
2 cylinders 90% argon--10% methane
3 cylinders oxygen
3 cylinders hydrogen

TABLE 7 (continued)Resins

- 15 ft³ XE150 resin (portable demineralizer)
- 2 ft³ ILCO-NR8 (shield water demineralizer)
- 1.18 ft³ ILCO-NR8 (purification demineralizer)

Health Physics

- 95 disposable lab coats and shoes
- 3--4-oz. bottles stannous chloride
- 20 beakers, 500 ml
- 12 cloth lab coats
- Replacement glassware (assorted pieces)

Miscellaneous

- 100 lb wiping rags
- 170 Foxboro recorder charts
- 50 filters for electronic equipment
- 5 roles brush recorder paper

In general, the rate of waste accumulation is related to primary system maintenance activities. When the plant is in operation, there is less maintenance activity in the primary system, less contaminated protective clothing and less radioactive laundry water. During periods when the plant is inoperative, the opposite is the case. Recent data is probably the most representative of that to be expected in the future. During plant power operation, liquid waste is accumulated at a rate of 55 to 60 gallons per day; when the plant is inoperative, accumulation is slightly over 80 gallons per day.

Table 8 also lists the activity of the waste disposal system effluent. The values quoted are probably high and should be treated with some suspicion, since:

- (1) Raw water at the site, counted in an identical manner, yielded almost identical activity measurements.
- (2) Some of the samples utilized had a volume of only 100 ml. For an activity of $10^{-7} \mu\text{c}/\text{cm}^3$, with this sample size, the uncertainty in the activity is greater than 100%.
- (3) There was no detectable activity in a sample returned to Baltimore for more complete analysis. Sensitivity of Baltimore equipment is $2 \times 10^{-8} \mu\text{c}/\text{cm}^3$.

TABLE 8
RWDS Operation

<u>Date</u>		<u>Tank Level (in.)</u>		<u>Quantity Processed (gal)</u>	<u>Hours in Operation</u>	<u>Activity After Processing ($\mu\text{c/cc}$)</u>
<u>Start</u>	<u>Stop</u>	<u>Begin</u>	<u>End</u>			
4/17/62	4/18/62	100	0	1224	28	2.84×10^{-7}
4/23/62	4/24/62	78	17	776	19	2.0×10^{-7}
5/18/62	5/19/62	100	0	1224	31	3×10^{-7}
5/29/62	5/30/62	77	0	813	10	3.25×10^{-7}
5/30/62	5/31/62	70	0	715	8-1/2	1.93×10^{-7}
6/4/62	6/4/62	78	0	827	18-1/2	1.01×10^{-7}
6/7/62	6/8/62	92	71	294	18-1/2	1.3×10^{-7}
6/9/62	6/10/62	66	0	759	20-1/2	1.9×10^{-8}
7/2/62	7/2/62	103	80	322	10-1/2	4.4×10^{-7}
7/3/62	7/3/62	80	77	42	4-1/2	0×10^{-7}
7/3/62	7/3/62	81	61	280	4	2.85×10^{-7}
7/4/62	7/4/62	61	12	587	7-1/2	8.15×10^{-8}
7/6/62	7/6/62	12	0	106	3	6×10^{-8}
7/8/62	7/8/62	11	0	94	4	0×10^{-7}
7/19/62	7/20/62	73	0	664	16-1/2	1.9×10^{-7}
8/10/62	8/10/62	66	0	748	24	1.45×10^{-7}
9/7/62	9/7/62	110	81	358	5-1/2	5.2×10^{-8}
9/9/62	9/7/62	105	83	308	6-3/4	9.3×10^{-8}
9/8/62	9/9/62	83	0	979	24	
9/17/62	9/17/62	60	16	520	12	2.9×10^{-8}
9/27/62	9/27/62	57	18	<u>434</u>	<u>11</u>	1.45×10^{-7}
Total				12,074	268-3/4	

Avg Processing Rate 44.9 gph

The samples analyzed in Baltimore only establish the activity from relatively long-life products. The short half-life products decay to undetectable magnitudes during the period which extends from the time the sample is obtained at Sundance to the time detailed analyses are performed at Baltimore. Data on liquid waste samples analyzed in Baltimore is presented in Table 9. The activities listed in that table were adjusted to show their correct value at the time the samples were obtained at the Sundance site.

F. PLANT GASEOUS EFFLUENTS

The volume and activity of plant gaseous effluents were monitored during the 400-hour run. The volume of discharge, in cubic feet, is tabulated below:

TABLE 9
Nuclides in PM-1 Water Samples
(microcuries per milliliter)

<u>Sample Source</u>	<u>Co-58</u>	<u>Mn-54</u>	<u>Ba-140</u>	<u>Fe-59</u>	<u>Sr-89</u>	<u>Co-60</u>	<u>Total Activity</u>	<u>Total Solids (ppm)</u>	<u>Chloride (ppm)</u>
Primary coolant before demineralizer	10.7×10^{-5}	9.1×10^{-6}	2.2×10^{-5}	5.5×10^{-6}	4.8×10^{-6}	9.1×10^{-7}	1.49×10^{-4}	13	--
Primary coolant after demineralizer	1.3×10^{-6}	1.9×10^{-7}	2.1×10^{-7}	$< 2 \times 10^{-8}$	6.7×10^{-8}	$< 2 \times 10^{-8}$	3.66×10^{-6}	3	
Shield water before demineralizer	8.9×10^{-7}	9.1×10^{-8}	1.1×10^{-6}	$< 2 \times 10^{-8}$	$< 2 \times 10^{-8}$	2.6×10^{-7}	2.34×10^{-6}	28	0.028
Shield water after demineralizer	9.6×10^{-8}	4.8×10^{-8}	$< 2 \times 10^{-8}$	$< 2 \times 10^{-8}$	$< 2 \times 10^{-8}$	$< 2 \times 10^{-8}$	1.5×10^{-7}	20	0.004
Main steam condensate	--	--	Not Radioactive	--	--	--	$< 2 \times 10^{-8}$	25	--
Feed water	--	--	Not Radioactive	--	--	--	$< 2 \times 10^{-8}$	21	--
RWDS sump	6.8×10^{-5}	3.7×10^{-6}	3.6×10^{-6}	2.5×10^{-6}	4.2×10^{-7}	9.5×10^{-7}	7.82×10^{-5}	443	--
RWDS condensate return	--	--	Not Radioactive	--	--	--	$< 2 \times 10^{-8}$	10	--

Startup	9
Power Operation	0
Shutdown	6

In no case was the activity of the gases sufficient to actuate the alarm. A sample of gas was returned to Baltimore for a detailed analysis. The measured gamma spectrum of the sample was indistinguishable from the background spectrum. Hence, it was concluded that the gas contained no measurable radioactive constituents.

G. PLANT LIQUID BALANCE

A liquid balance of the plant, developed from data obtained during the 400-hour test, is illustrated in Fig. 3. This shows that the average amount of water required by the plant and its personnel is 1363 gallons per day, including not only plant makeup but also laundry water and sanitary sewerage requirements. Of the 1363 gallons per day, makeup to the plant is approximately 245 gallons, virtually all attributable to secondary system vent losses.

Figure 3 shows that the liquid processed by the waste disposal system is dumped to the sewerage system. This is the usual situation. The processed liquid is not returned to the condensate storage tank because of its high pH and low resistivity. As was stated previously, most of the radioactive waste results from laundry water. The waste disposal system adequately reduces the liquid activity to MPC requirements; however, there is a carry-over of detergent, resulting in the high pH and low resistivity.

The total plant makeup was metered by an integrating-type flow meter. Data for the 400-hour run is shown in Table 10. The daily quantity varies from 690 to 1770 gallons, with 1363 as the daily average.

The primary system makeup was determined by measuring the length of time the charging pumps were in operation and multiplying this by the volumetric pumping rate of these positive displacement pumps. The data for the 400-hour run is shown in Table 11.

The primary system leak rate was determined from the change in pressurizer level during periods of operation when the primary coolant average temperature, hence density, was constant. Data for all periods of plant operation, presented in Table 12, shows a continuous trend toward a negligible leak rate. During April and May there were problems with weeping pressurizer relief valves and a few of the manually operated valves. Before these were repaired, the leak

rate went as high as 37.2 gallons per hour, and 20 gallons per hour was not unusual. When operation was resumed, during July and August, the leak rate ranged from 0.3 to 2.75 gallons per hour. During the 400-hour run, the leak rate had been further reduced to negligible amounts; the highest rate during that test period was 0.44 gallon per hour.

TABLE 10
Total Plant Makeup Water

<u>Date</u>	<u>Flowmeter Reading (gal)</u>	<u>Quantity (per 24-hr period) (gal)</u>
September 12, 1962	21400	700
September 13, 1962	22100	700
September 14, 1962	23080	980
September 15, 1962	24350	1270
September 16, 1962	26120	1770
September 17, 1962	28500	2400
September 18, 1962	30200	1700
September 19, 1962	30920	720
September 20, 1962	31610	690
September 21, 1962	32880	1270
September 22, 1962	33720	840
September 23, 1962	35100	1380
September 24, 1962	36320	1200
September 25, 1962	37840	1520
September 26, 1962	39600	1760
September 27, 1962	40920	1320
September 28, 1962	42700	1780
September 29, 1962	44780	2080
September 30, 1962	46100	1320
October 1, 1962	47300	1200

TABLE 11
Primary System Makeup Requirements

<u>Date</u>	<u>Operating Time (minutes)</u>	<u>Gallons</u>
September 12, 1962	1	1
	2	2
	1	1
	2	2
September 13, 1962	2	2
	17	17
September 15, 1962	1	1
	2	2
September 17, 1962	2	2
September 21, 1962	3	3
September 22, 1962	1	1
September 23, 1962	7	7
September 24, 1962	8	8
	3	3
September 27, 1962	4	4
September 29, 1962	3	3
	2	2
	2	2
October 1, 1962	1.5	1.5
	2	2
		66.5

Total

Average per day 3.325

TABLE 12
Primary System Leak Rate

<u>Date</u>	<u>Shift*</u>	<u>Leak Rate (gph)</u>
April 11, 1962	2	7.2
April 11, 1962	3	7.5
April 12, 1962	1	10.2
April 12, 1962	3	7.2
April 13, 1962	2	18.0
April 14, 1962	2	37.2
April 14, 1962	3	7.2
April 15, 1962	3	19.2
April 17, 1962	3	25.8
April 18, 1962	2	12.0
May 24, 1962	2	6.0
May 25, 1962	2	1.8
May 25, 1962	3	2.6
May 26, 1962	2	3.0
May 26, 1962	3	3.2
May 27, 1962	2	4.1
May 27, 1962	3	6.6
May 28, 1962	1	30.0
May 28, 1962	2	9.6
May 28, 1962	2	21.6
July 23, 1962	3	0.3
July 23, 1962	3	0.14
July 24, 1962	1	2.75
July 25, 1962	1	1.75
July 26, 1962	2	0.5
July 26, 1962	3	0.64
July 27, 1962	1	0.75
July 28, 1962	1	0.82

TABLE 12 (continued)

<u>Date</u>	<u>Shift*</u>	<u>Leak Rate (gph)</u>
July 28, 1962	3	0.8
July 29, 1962	2	0.8
July 30, 1962	2	1.0
July 30, 1962	2	0.88
July 31, 1962	3	1.5
July 31, 1962	2	1.4
August 1, 1962	3	1.6
August 2, 1962	1	1.8
August 2, 1962	1	1.65
August 3, 1962	2	1.94
August 3, 1962	1	1.75
August 4, 1962	3	1.81
August 5, 1962	1	1.74
August 5, 1962	2	2.19
August 5, 1962	3	2.04
August 7, 1962	1	2.18
August 7, 1962	2	2.4
August 8, 1962	1	2.68
September 15, 1962	3	0
September 16, 1962	3	0
September 17, 1962	2	0
September 18, 1962	3	0.165
September 19, 1962	3	0.44
September 20, 1962	1	0
September 21, 1962	2	0
September 22, 1962	2	0
September 23, 1962	3	0.044

TABLE 12 (continued)

<u>Date</u>	<u>Shift*</u>	<u>Leak Rate (gph)</u>
September 24, 1962	3	0
September 25, 1962	3	0.0685
September 26, 1962	2	0
September 27, 1962	3	0.22
September 28, 1962	2	0
September 29, 1962	3	0.11
September 30, 1962	3	0.165
October 1, 1962	3	0
October 2, 1962	1	0.125

* 1 (0000 to 0800)

2 (0800 to 1600)

3 (1600 to 2400)

II. STARTUP REQUIREMENTS

This section discusses PM-1 plant startup requirements. The following individual items are considered:

- (1) Manpower required
- (2) Auxiliary power required
- (3) Time required
- (4) Actual source strength available.

A. MANPOWER REQUIRED FOR STARTUP

Military manpower utilized during plant startups was compiled from the plant operating logs. In each case, three men were employed on each shift: the military shift supervisor, a control room operator and an equipment operator. The date of the startup, the number of shifts during which startup was being conducted and the specialties of the military crew are presented in Table 13. Regular crews are utilized for plant startup; no specialty skill is required for startup of the plant.

The plant can be started with a two-man crew, provided that the men are properly qualified. One man must have the qualifications of the military shift supervisor; that is, thorough understanding of all the plant operations and detail knowledge of the startup procedure. The other crew member need only be a qualified equipment operator.

B. AUXILIARY POWER REQUIRED FOR STARTUP

The auxiliary power required during startup was compiled from data obtained at the initiation of the 400-hour run. The power utilized for each step of the operation is listed in Table 14.

C. TIME REQUIRED FOR STARTUP

The major steps required to start up the PM-1 plant are shown in Table 15. Also shown is the actual time at the completion of each major step for the startup performed at the beginning of the 400-hour test run. A total of 13.5 hours was required. An additional 8.5 hours' delay was encountered because the radar station was on alert and did not wish to undergo any transients which might result if the plant paralleled the site diesel generators.

Total startup time for other dates can be obtained from Table 13 by noting the number of shifts required.

TABLE 13
Startup Manpower

<u>Date</u>	<u>Number of Shifts</u>	<u>Mechanical</u>	<u>Military Crew Specialties (man-shifts)</u>		<u>Health Physics</u>
			<u>Instrumentation</u>	<u>Electrical</u>	
<u>Cold Startups</u>					
June 10 June 11	4	5	3	3	1
July 16 July 17	4	8	2	1	1
July 22 July 23	4	5	3	3	1
Sept 12	1-1/4	1-1/2	1	1	1/4
<u>Hot Startups</u>					
July 25	1/2	3/4	1/4	1/4	1/4
Sept 14	1-1/4	1-1/4	1/2	2	0
Sept 17	3/4	1	0	1/4	1
Sept 22	3/4	1-1/4	1/4	3/4	0
Sept 29	3/4	1-1/2	0	1/2	1/4

TABLE 14
Startup Auxiliary Power Requirements

Normal house load	52.5 kw
Main coolant pump turned on	101
Pressurizer heaters A, B and C turned on	128.5
Rod control in service	133
Pressurizer heaters B and C turned off	106
Rod motion starts	109
Criticality	112
Normal conditions	101
Motor driven feedwater and condensate pumps in service	147
Turbine being rolled	147
Exciter energized	192

Operation since the plant was transferred to the Air Force has shown that a typical cold startup has required between 1-1/2 and 2 shifts. Thus, the figure of 13.5 hours, given on page 27, is a representative value. Typical hot startups have required approximately 2 hours.

Time required to perform the prestartup check list is not included in the startup times quoted. The performance of the cold startup check list takes approximately 8 hours. This complete check list is utilized if the plant has been shut down more than 24 hours or if major maintenance was performed during the shutdown. Otherwise, the short form check list is utilized and requires about 1-1/2 hours.

TABLE 15
Startup Steps and Time

<u>Time</u>	<u>Step</u>
0001	T _{ave} 200° F with main coolant pump operating
0128	Checked rod control system.
0230	Reactor critical with rod bank at 9.90 inches Primary pressure 285 psig, T _{ave} 225° F, heating primary loop.
0250	Closed steam generator vent valves.
0430	Blew down steam generator to 16 inches.
0555	T _{ave} 463° F, pressurizer pressure 1300 psi, rod bank 13.0 inches.
0608	Sampled primary loop and added 6.5 ft ³ of hydrogen.
0650	Started shield water cooler fans 1C and 1D.
0700	Reactor critical at temperature and pressure. Ready to roll turbine. Site would not give permission to parallel site genera- tors because of alert.
1400	Started shield water cooler fan 1A.
1505	Started warming main steam line.
1507	Secured warming main steam line.
1517	Started condenser fans slow forward.
1530	Received site permission to parallel.
1540	Scram--reactor at too low power to cut in secondary system rapidly.
1607	Completed check list 6. Started raising rods.
1639	Reactor critical with rods at 12.38 inches.
1723	Started heating main steam line.
1735	Closed main steam valve because of suspected steam leak.
1745	Found desuperheater bypass valve open.
1747	Reopened main steam valve and warmed secondary.
1810	Started feedwater pump 1A.
1830	Placed auxiliary steam system in service and started pulling vacuum with hogging jet.

TABLE 15 (continued)

<u>Time</u>	<u>Step</u>
1840	Started 1A condensate pump.
1842	Started rolling turbine; condensers A, B and C in operation.
2007	Started lube oil cooler fan 1B.
2018	Attempted to parallel with site, but generator breaker malfunctioned.
2152	Paralleled with site and assumed load.

D. SOURCE STRENGTH

The startup source strength, as determined by plant instrumentation, was compiled as a function of time, total core energy output and approximate coolant temperature. This information is presented in Table 16. In all cases, the required startup rate of 5 cps was exceeded by at least a factor of two.

The discriminator setting on the pulse amplifier is so established that an acceptable thermal neutron pulse count rate is obtained, while extraneous noise, neutron pulses and fast gamma pulses are minimized. Thus, the data in Table 16 is strongly influenced by discriminator setting as well as by the source strength. However, the data obtained toward the end of the test period indicates that the source of neutrons for startups was not the primary source since, by that time, the strength of this source had decayed by a factor of 7.

The PM-1 core contains a polonium-beryllium primary source located at the core centerline. The reactor contains two identical antimony-beryllium secondary neutron sources in the thermal shields. The primary source has an initial strength of 1.3×10^8 n/sec and a half-life of 138.4 days. The strength of each secondary source was calculated to be 2.3×10^8 n/sec after six months of operation at 80% power. Burnup of the secondary sources over a 20-year period is negligible. The secondary source decays with a half-life of 60 days.

The primary source was purchased during August 1961. Therefore, by September 17, 1962 (the last startup for which data is presented in Table 16), its strength had decreased by a factor of 7. Thus, startups toward the end of the test period were not dependent on neutrons from the primary source.

TABLE 16

Date	Integrated Core Output (Mw-days)	Startup Temperature or Condition	Instrumentation Reading (cps)	
			Channel 1	Channel 2
Mar 1	0	Cold startup*	70	25
Apr 9	0.05	Cold startup	35	45
Apr 23	15	Cold startup	50	275
May 14	25	Cold startup	35	25
June 3	48	Cold startup	80	70
July 15	82	Cold startup	23	11
July 21	92	Cold startup	27	12
July 25	98	440° F	120	30
Aug 4	126	380° F	80	30
Aug 8	132	345° F	50	20
Aug 9	132	Cold startup	42	20
Sept 14	137	425	40	60
Sept 17	146	446	20	60

*During "cold startup" the primary system is being heated by the primary circulating pump. Instrumentation readings listed are average values for a typical temperature range of 125° to 175° F.

The variation of nuclear instrumentation readings with control rod position for a hot startup (primary coolant average temperature of 426° F) is presented below. This is the data used to develop the 1/m plot.

Control Rod Bank Position (inches withdrawn)	Instrumentation Reading (cps)	
	Channel 1	Channel 2
0	38	55
3	38	55
5	38	55
6	60	40
7	45	70
8	55	80

Control Rod Bank Position (inches withdrawn)	Instrumentation Reading (cps)	
	Channel 1	Channel 2
10	90	120
11	150	200
11.5	200	300

The same data for a typical cold startup is:

Control Rod Bank Position (inches withdrawn)	Instrumentation Reading (cps)	
	Channel 1	Channel 2
0	19	16
3	19	16
4	19	16
5	20	19
6	21	22
7	25	27
9	68	55
9.5	150	110

III. PLANT AS AN ENERGY SOURCE

Data presented in this chapter is concerned with:

- (1) Plant operating history.
- (2) Steady-state heat balances.
- (3) Heat exchanger crud buildup.
- (4) Steady-state electrical characteristics.
- (5) Transient response to dynamic changes in load.
- (6) Transient response to changes in control rod position.
- (7) Auxiliary power.
- (8) Scrams.
- (9) Estimated refueling time.
- (10) Parallel operation with other sources of power.

A. PLANT OPERATING HISTORY

Plant logs from the time initial criticality was achieved were reviewed to develop a profile of plant operating history. This profile, shown in Fig. 4, extends from February 25 until October 31. Some of this data was presented in the previous quarterly progress report, but is repeated here for completeness. Figure 4 includes the following information:

- (1) Periods during which the plant was critical.
- (2) Periods during which the plant was at operating temperature.
- (3) Periods during which the plant was operating at power.
- (4) Number of scrams.
- (5) Dates various tests were performed.
- (6) Efforts performed when plant was shut down.

This profile provides basic information as to the periods of time when various operating log data will be of prime interest. That is, periods during which the plant was operating at power are clearly isolated from periods when the plant was not operating and routine maintenance and/or crew training was being conducted.

As shown in Fig. 4, modifications were made to the plant during the period from August 8 to September 12. These modifications were generally minor. All efforts during this period are summarized below:

- (1) Installed a larger capacity electric-motor-driven condensate pump and increased the diameter of suction and discharge piping.
- (2) Installed larger air ejector jet nozzles on steam jet air ejector and increased the size of the suction and discharge piping.
- (3) Removed 6-inch hotwell vent line to turbine exhaust and installed a 4-inch hotwell vent line to the air cooled condensers to improve condensate removal.
- (4) Replaced solenoid valves with air-operated valves in the primary system. The following valves were changed:
 - (a) Pressurizer spray valve.
 - (b) Pressurizer vent valves.
 - (c) Pressurizer drain valves.
 - (d) Silver nitrate iodine removal unit isolation and bypass valves.
 - (e) Expansion tank drain valves.
- (5) Added air manifold for air-operated valves which replaced solenoid valves. Installed air supply to the manifolds.
- (6) Added check valves to primary charging pump discharge lines to prevent backflow.
- (7) Rearranged shield water piping (pump suction at top of tank, shield water discharge into bottom of tank) to provide better circulation of shield water.
- (8) Revised shield water air blast cooler air ducting to eliminate short circuiting of hot discharge air back into the suction side.

- (9) Replaced 250-kva station transformer with 500-kva transformer.
- (10) Installed poles and ran wire for plant lightning protection.
- (11) Installed additional lighting in secondary building.
- (12) Added Fisher Level-Trol to the waste disposal system evaporator. Increased the pipe size of the 500-gallon waste disposal tank vent line to improve operability of the system.
- (13) Added permanent sampling point in the radioactive waste disposal system condensate return line.
- (14) Installed a centrifugal vent fan in the roof of the addition to the secondary building to increase air circulation.
- (15) Installed permanent drain lines from the auxiliary boiler and air compressors.
- (16) Installed a pipe in the auxiliary boiler condensate tank to prevent overflow from backing up into the raw water line if float valve fails.
- (17) Added piping to feedwater heater and deaerator relief valves to exhaust steam blow-off outside of building.
- (18) Added drainage ditch adjacent to diesel generator building to prevent flooding of diesel floor.
- (19) Erected concrete slab and small building over the tank which contains the waste disposal system.
- (20) Poured concrete slab and erected shielding enclosure for the off gas monitor.
- (21) Added guard rails to the top of the air cooled condensers for safety.

The total integrated output of the PM-1 core is shown as a function of time in Fig. 5.

B. STEADY-STATE HEAT BALANCES

Plant data during steady-state operation is presented in this section and compared with design predictions. Discussed below, individually, are:

- (1) The secondary system heat balance.
- (2) The primary system heat balance.
- (3) Required core power and calibration of nuclear instrumentation.

1. Secondary System Heat Balance

A secondary system heat balance was prepared from design data for a power level corresponding to the maximum steady-state electrical output obtained during the 400-hour test. These conditions were:

- (1) Gross electrical output of 1270 kw.
- (2) Net electrical output of 985 kw.
- (3) Steam generator blowdown secured.
- (4) Secondary system evaporator inoperative.

A required reactor power of 7.24 Mw(t) was predicted for this set of conditions. The measured reactor power (computed on the basis of steam flow and feedwater flow) was 7.6 Mw(t), a difference of 5%.

The heat balances predicted from design data and those derived from experimental data are presented in Fig. 6. Since not all flow rates, temperatures and pressures are measured at the plant, only actual measured values are presented. The available information is sufficient to define the overall performance and efficiency of the secondary system.

The ambient temperature range during the period when this data was obtained was relatively constant. Hence, no effect of ambient temperature on the system efficiency was noted. Since the turbine exhaust pressure is held constant, the system performance is not expected to change with ambient temperature, except for a decrease in auxiliary power requirements during cooler weather.

The step variation of auxiliary power utilization with net power output shows the effect of removing a condenser unit from operation. The same sort of variation will occur in cold weather even at full load.

2. Primary System Heat Balance

A primary system heat balance was prepared from design data for a power level corresponding to a gross plant output of 1270 kw(e).

To make this heat balance correspond closely to actual operating conditions, the following conditions were assumed:

- (1) Secondary system evaporator inoperative.
- (2) No spent core in the spent core tank.
- (3) Waste disposal system inoperative.

The resultant heat balance is shown in Fig. 7. Along with the design predictions, temperatures and flow rates measured during the 400-hour run are shown for power levels corresponding to gross electrical output of 440, 857 and 1270 kw. In general, the measured parameters are in agreement with design predictions. Since the plant in general, (and the primary system in particular), was designed as an operational rather than a test plant, all temperatures and flow rates are not obtainable on a continual basis. For example, there is no direct measurement of reactor coolant system flow rate, and the purification system flow rate is preset by a flow control valve and a flow measuring orifice. This flow rate is checked from time to time but is not accessible during plant operation.

The data pertaining to the shield water system was all obtained with only three of the four shield water coolers in operation. This demonstrates that the plant can continue to operate with one shield water cooler out of service, as long as the ambient temperature does not exceed 70° F and provided the waste disposal system is not in operation. Further discussion is provided in Section VII-F of this report.

3. Required Core Power

The core heat production rate must be sufficient to overcome the primary system heat losses in addition to supplying the energy specified by the secondary system heat balance. To determine the required core output, data pertaining to net primary system heat losses was evaluated.

During precritical testing, the primary system was pressure tested at normal operating temperature and pressure, using an electrical heater instead of the core as the principal energy source. A heater power of approximately 70 kw was required to maintain a coolant temperature of 463° F with the primary circulating pump in operation and the purification system inoperative. Thus, the total primary system heat loss, not supplied by the net energy of the pump, is 70 kw.

Data from Fig. 7 shows that the energy loss associated with the operation of the purification system is approximately 40 kw. Hence, the core power production must exceed the energy requirements of the secondary system by approximately 0.11 Mw(t).

The information pertaining to primary system energy losses was used to determine the approximate accuracy of the calorimetric calculations performed, during operation, to set and check the nuclear instrumentation. The calorimetric calculation is based entirely on feedwater flow rate, steam temperature and feedwater temperature. That is, primary system energy losses are ignored and the loss associated with steam generator blowdown is only approximated. Actually, the energy loss resulting from blowdown of liquid from the shell side of the steam generator is overestimated in the calorimetric calculation. The calculation treats the enthalpy of the blowdown as saturated vapor rather than saturated liquid, a difference of 47 kw at normal operating conditions. Thus, the net result of the calorimetric calculation is an underestimate of core power of approximately 0.063 Mw(t). At full power, this is an error of approximately -0.7%, which is more than sufficiently accurate for the purpose of setting and monitoring the nuclear power instrumentation.

The accuracy of the instrumentation used in preparing the calorimetric calculation was briefly reviewed. The approximate uncertainties in the data are:

- (1) Steam temperature: $\pm 3^\circ$ F due to possible instrument calibration inaccuracy and reading errors.
- (2) Feedwater temperature: $\pm 3^\circ$ F due to possible instrumentation inaccuracy and reading errors.
- (3) Feedwater flow rate: $\pm 2.5\%$ at design flow rate. This total uncertainty is composed of 0.5% in the measuring circuit, 0.5% in the recorder, 0.5% in the orifice coefficient and 1% in reading the recorder.

The calculated reactor power is directly proportional to the feedwater flow rate; hence, a 2.5% uncertainty exists due to this factor. The temperature uncertainties contribute a possible error of 0.36% at full power. Thus, the total uncertainty, at full power, of the calorimetric calculation is 3.6%.

C. CRUD BUILDUP

The variation of heat exchanger performance with plant operating history was investigated. Data from periods when the plant was

operating at power was utilized to calculate the relative variation of the overall heat transfer coefficient of the:

- (1) Purification cooler.
- (2) Purification system economizer.
- (3) Main steam condensers.
- (4) Shield water coolers.
- (5) Steam generator.

No effect of crud buildup was discernible in any of the first four heat exchangers listed above. This is not surprising since, in the purification system cooler and economizer, both tube- and shell-side fluids are continually demineralized. The main steam condensers and the shield water heat exchangers are air cooled; the air-side film coefficient is the controlling thermal resistance. Therefore, the effect of any tube-side crud would be masked.

The effect of some variation in heat transfer coefficient was noted in the steam generator. However, it is not known with certainty whether the variation was the result of crud buildup or instrumentation drift. In any event, the resultant increase in thermal resistance was less than that anticipated in the steam generator design requirements.

PM-1 steam generator specifications require that the manufacturer include in his design considerations a fouling factor for the outside of the tubes, to provide for a thermal resistance of $0.0004 \text{ hr-ft}^2\text{-}^\circ\text{F/Btu}$. At full-power conditions this amounts to a 31% reduction in the steam generator overall heat transfer coefficient as compared to the clean condition. At a power level of 5.5 Mw(t), corresponding to a steam flow rate of 20,000 lb/hr, this fouling factor amounts to a 28% reduction in overall heat transfer coefficient. Plant data at this power level indicates that the actual coefficient decreased approximately 18% during June and July.

The shell side of the steam generator is in the boiling regime over most of the tube bundle length. The boiling coefficient of heat transfer is heat flux-dependent; hence, the overall heat transfer coefficient is also somewhat heat flux- or power-dependent. Therefore, in order to analyze crud buildup in this heat exchanger, it is necessary to compare the performance of the unit at various times but at constant load. There are two methods by which steam generator heat load can be determined:

- (1) Enthalpy change of the reactor coolant from inlet to outlet.
- (2) The difference in enthalpy between the feedwater entering the unit and the steam leaving.

Neither of the above methods is precise. The inaccuracy in the first method occurs because the temperature rise is small and any instrument error is a large fraction of the difference. In general, the second method is more accurate; however, the permanent plant records of steam and feedwater flow consist of narrow strip charts, which are difficult to read with precision. Despite this, the second method was considered superior and was employed. Data during May, June and July was evaluated at a steam flow rate of 20,000 lb/hr. Evaluation of this data showed that the overall heat transfer coefficient at this heat load decreased by approximately 18% during June and July. Design data at this steam flow rate shows that the fraction of the total thermal resistance corresponding to complete crud buildup would be 28%. Since the unit had not been operated appreciably prior to May, it appears that this 18% reduction in heat transfer coefficient could represent the effect of crud buildup.

An attempt was made to utilize data from the 400-hour run, recorded toward the end of September, as an additional data point. However, the main steam flow rate instrumentation was recalibrated just prior to the final testing and is not consistent with the previous data. The 400-hour run data indicates that the steam generator performance during September was essentially identical to that during May. Thus, it is not known with certainty whether the apparent reduction in heat transfer coefficient resulted from crud buildup or instrumentation drift.

The calculated values of overall heat transfer coefficient, based on the tube outside surface area, are presented below. All data is based on a steam flow rate of 20,000 lb/hr and a heat transfer rate of 5.5 Mw(t).

<u>Date</u>	<u>LMTD (°F)</u>	<u>U_0 (Btu/hr-ft²-°F)</u>
5/26	26.5	644
6/16	31.3	545
7/26	32.2	530
9/23	26.9	634

D. STEADY-STATE ELECTRICAL CHARACTERISTICS

To determine the steady-state voltage, frequency and harmonic characteristics of the PM-1 turbine-generator, 18 Brush recorder traces were examined. It was found that the actual electrical characteristics were equal to or better than design requirements. A comparison is presented in Table 17.

TABLE 17
Steady-State Electrical Characteristics

	Measured Value (%)	Design Requirements (%)
Voltage fluctuations	Less than 0.5	0.5
Frequency fluctuations	Less than 0.1	0.25
Harmonic content	1.5	2

The voltage and frequency measurements were obtained directly from the Brush recorder traces; the harmonic content was determined by a harmonic analyzer. The measured voltages were plotted and are shown in Fig. 8. A summary of the harmonics, through the 13th, is presented in Table 18. Also listed in Table 18, for the purposes of comparison, are the voltages of a perfect sine wave with a 117-volt fundamental.

Figure 8 was compared to a perfect sine wave with a maximum amplitude of 117 volts. Paragraph 1.190 of AIEE Publication No. 503, Test Code for Synchronous Machines, states that, "The deviation factor of a wave is the maximum difference between corresponding ordinates of the wave and of the equivalent sine wave to the maximum ordinate of the equivalent sine wave when the waves are superimposed in such a way as to make this maximum difference as small as possible." When Fig. 8 is superimposed on the perfect sine wave in this manner, the maximum deviation is observed to be less than 1.5%. This is well within the requirement of 2%, even with the added distortion of two transformers in the system.

The results of the harmonic content analysis are conservative, since the Brush recorder was connected to a point in the plant 120-volt distribution system. This was done to avoid overloading the 4160 potential transformer. Thus, the recorded data contains harmonics introduced by the plant station transformer and the 120-volt lighting transformer.

TABLE 18

Angle (degrees)	Theoretical Voltage (perfect sine wave)	Measured Harmonic Content (volts)
0.8	1.638	1.798
1.0	2.048	2.248
1.3	2.650	2.909
1.5	3.065	3.364
2.0	4.083	4.477
3	6.119	6.705
5	10.202	11.107
10	20.311	21.781
12	24.324	25.862
15	30.282	31.787
30	58.500	59.129
45	82.732	82.400
60	101.320	101.387
90	117.000	116.805

E. TRANSIENT RESPONSE TO DYNAMIC CHANGES IN LOAD

Data pertaining to dynamic changes in load, obtained by utilizing the Martin Mobile Data Acquisition System, is presented and discussed in this section. The following topics are treated individually:

- (1) General experimental technique and data van hookup.
- (2) Description and discussion of data.
- (3) Predicted and actual transient performance.

- (4) Pressurizer performance.
- (5) Transient electrical performance.

1. Experimental Technique and Van Hookup

Transient data was recorded, using the Martin Mobile Data Acquisition System. This is a trailer-mounted facility which records data in the form of punched IBM cards.

Installation at the PM-1 site consisted of making connections from the plant instrumentation to the trailer. These connections are illustrated schematically in Figs. 9 and 10. Approximately 75% of the connections were from instrumentation located in the plant control console; the other 25% were from miscellaneous thermocouple locations within the plant. The data van receives the same signal input as the plant instrumentation and records the information, in terms of percent of full scale reading, on a magnetic tape. This tape is subsequently utilized to reproduce data in the form of punched IBM cards. The cards were transported back to the Martin Computer Center where they were listed and also utilized to automatically plot the data.

The variation in each instrumented parameter was recorded four times per second by the data acquisition system; it requires 1/4 second to scan all circuits. Recording of data was initiated before changes in electrical output were imposed on the plant and continued until the steady-state conditions were again reached. As the transient effects became negligible, data was recorded intermittently (at 10- to 30-second intervals).

Changes in plant electrical output were imposed by use of an electric load bank. The unit was relay-operated; hence, virtually instantaneous changes in electrical load could be imposed. The load bank remote controls and the acquisition system remote controls were both located in the plant control room during testing. This allowed continual visual monitoring of plant performance data concurrent with recording by the van system.

2. General Description of Transient Data

a. Presentation of data

Appendix A contains graphical representations of the data collected during the plant transient tests. Table 19 correlates the figures or runs with the "millisadic run number." The latter is the number assigned to the raw test data for future reference or verification. Also listed in the table is the number of variables chosen to be plotted to properly reflect the plant response.

TABLE 19

Run No.	Page No. (Appendix A)	Milli- sadic Run Number	Number of Variables	Plant Transient (kwe net)	Load Change (kwe net)
I	A-3 to A-8	150	26	0 to 1000	1000
II	A-9 to A-14	160	26	1000 to 0	
III	A-15 to A-17	110	15	200 to 600	400
IV	A-18 to A-20	180	15	600 to 200	
V	A-21 to A-23	120	15	600 to 1000	400
VI	A-24 to A-26	170	15	1000 to 600	
VII	A-27 to A-29	10	15	0 to 200	200
VIII	A-30 to A-32	60	15	800 to 1000	200
IX	A-33	70	5	1000 to 800	
X	A-34	20	5	200 to 400	200
XI	A-35	100	5	400 to 200	
XII	A-36	30	5	400 to 600	200
XIII	A-37	90	5	600 to 400	
XIV	A-38	50	5	600 to 800	200
XV	A-39	80	5	800 to 600	
XXIII	A-58 to A-64	260	30	Scram from 90% power	

The largest number of variables was selected for the most severe plant transients, that is, plant scram from a high power level and 1000-kw load swing.

In all cases, the first figure is reactor power and the time interval is fifteen minutes to permit a complete indication of plant recovery.

Transient data during the 1000-kw load change (see Figs. I and II of Appendix A) was grouped as follows:

- (1) Reactor parameters--power, period, T_c and T_h .
- (2) Pressurizer parameters--level, pressure and temperature.
- (3) Purification system parameters--coolant temperature and shield water temperature.
- (4) Steam generator system parameters--level, pressure, feedwater and main steam flow.
- (5) Condensate system controls--deaerator and hotwell level, throttle steam flow, etc.
- (6) Condenser controls--exhaust pressure, condenser tube temperatures, etc.

For the scram from high power transient (Fig. XXIII), four additional parameters were chosen to be plotted (intermediate and source range instrumentation data, primary coolant pump power and primary coolant pump differential pressure) since a scram trips the coolant pump and reactor flux level trips the source range.

A smaller number of variables were plotted to represent the remaining, less severe transients. The variables chosen were those that would show the trend with which the plant responded. Purification system temperatures, pressurizer temperatures, feedwater temperatures, etc., were not included, as the severe transients showed almost no effect.

The traces depicting reactor outlet temperature, pressurizer pressure and feedwater flow show harmonics. These harmonics occur before, during and after the transient and are not a result of the transient. They are probably due to instrumentation disturbances. A review of the console recorder charts for these variables showed no such harmonics. Throttle steam flows at high loads are also not smooth plots. Turbine-generator governor difficulties did exist during these tests and affected throttle steam flow.

- b. General trends (Runs I and II, 1000-kwe load change--A-3 through A-14, Appendix A)

This transient was imposed on the plant with the use of the load bank. With the plant operating at a reactor power of 30% (Run I), supplying station service loads, a load of 1000 kwe was applied with the load bank. Reactor power was increased to 84%. Run II showed plant response for a decrease in plant net electrical load of 1000 kwe with a subsequent drop in reactor power from 84% to 30%.

Primary

No unexpected parameter changes were noted.

Pressurizer

Pressurizer level varied as coolant average temperature, with a subsequent similar pressurizer pressure variation. The pressure variation was not great enough to reflect a temperature change in the pressurizer.

Purification system

The temperature values for the system are in error, since the change in temperature on the shell side (reactor inlet temperature minus purification inlet temperature) of the economizer does not equal the change in temperature on the tube side (demineralizer inlet temperature minus the primary return temperature). However, the correct trend is shown. In Run II (a decrease in load), a rise in purification cooler inlet temperature and a rise in economizer return temperature are noted. This would be expected, since reactor inlet temperature rises as a result of the reduction in energy transferred to the secondary system. An increase in load (Run I) results in a lowering of reactor inlet temperature and a subsequent lowering of the economizer temperature. In either case, the lag of temperature change is due to the stored energy in the component. In both cases, shield water outlet and purification cooler outlet temperatures do not change. This is to be expected, since the shield water flow is over 13 times greater than the purification flow.

Steam generator

The steam generator level control functioned quite well during both transients, considering the small capacity of the steam generator. Comparing transients, it will be noted that a larger change in level occurs during the increases in load. This can be attributed to main steam flow demand. Run II shows steam demand as almost a step change to the new steady-state requirement, while Run I has steam

demand exceeding the steady-state requirement by 9000 lb/hr. Referring to throttle steam flow, the turbine-generator governor reaction accounts for the overshoot in steam demand. The remaining parameters are as expected.

Condensate system controls

Deaerator and hotwell level controls functioned well within the specified range of operation. Deaerator pressure during the down transient (Run II) holds constant. During the up transient (Run I), pressure decreases (with a similar decrease in feedwater inlet temperature), which would result upon an increase of condensate flow. However, the pressure seems to level out at 7.5 psig, which is below the alarm point. This appears to be a recording error, as no alarm was annunciated during the test and other steady-state runs at this load proved that the capacity of the deaerator steam supply valve from the auxiliary steam line was not exceeded.

At full load, before the transient (see Run II), deaerator pressure is adequately maintained at 11 psig.

Condenser controls

During the increase in load transient (Run I), turbine exhaust pressure rises as expected, but the tube temperatures fall. This is because the main bank louvers are controlled from the turbine exhaust pressure controls located in the turbine exhaust line. As the main bank louvers opened, tube temperatures fell before the added flow of steam could reach the condensers. The louvers then cycled until normal exhaust pressure was recovered and tube temperatures returned to normal. Note that air cooler differential pressure fluctuated; this can be attributed to the fluctuating cooling air flow as the main bank louvers cycled. The air cooler louvers were operated in the remote manual mode and were kept in one position.

During the decrease in load (Run II), turbine exhaust pressure fell at the instant of the transient, followed by closing of the louvers. Four condensers were in operation at the time of the transient with all condenser fans on full, far exceeding the required condensing capacity after the transient. This is reflected in the cycling of the tube temperature and air cooler differential pressure. Slight changes in the main bank louvers resulted in sharp changes in cooling air flow across the air cooler section and a completely unbalanced condensing system, as is readily seen from the traces.

400- and 200-kwe load swings--Runs III through XV, A-15 through A-39, Appendix A

Transients of 400 and 200 kwe (net) are shown on Runs III to XV. A comparison with Runs I and II shows that the swings in the plant parameters at the lesser transients were correspondingly less and that there were no unexpected deviations.

c. Scram from full power

The purpose of this test was to determine plant response to a scram from full power. Full power, however, could not be obtained, as the load bank was inoperative at the time of the test. The test van required power during the scram; therefore, the 480-volt bus had to be energized continuously. Due to an ADC alert, the radar site could not supply power to PM-1; and the tieline circuit breaker had to remain open during the test.

The test method was, therefore, as follows:

- (1) The desuperheater steam flow control valve was opened to dump steam to the condensers. Steam flow was limited by the amount of condensate flow required for desuperheating. Immediately before scram, approximately 80% power was indicated on the power range channels.
- (2) The PM-1 diesel was started and paralleled with the turbine-generator to assume load after scram.
- (3) Simultaneously with the manual scram, excess electrical loads such as the feedwater pump, condensate pump, condenser fans, etc., were manually tripped at the console to prevent overloading of the diesel. This was done successfully, as the diesel-generator did not trip and there was no evidence of voltage or frequency fluctuation.

Analysis of data (Run XXIII, A-58 through A-64, Appendix A)

Primary. Primary conditions were as expected following a scram and did not approach any critical values. Response was excellent.

The primary coolant pump was started two minutes after scram, whereupon T_h and T_c assumed average values. Purification cooler inlet temperature slowly rose as purification flow was resumed. Pressurizer level fell as flow was restored through the core.

Nuclear. Reactor power range channels fell to zero at the time of the scram, and the intermediate channels showed steady decay over a period of seven minutes.

Unusual fluctuation is shown in the reactor period during the four- to six-minute interval after scram. It is obvious that the source channel period was being monitored. (See figures of reactor period and Channel 1 or Channel 2 log count rate.) Fluctuations occurred from bistable chatter (Bistable 8, source range level disable). After the flux was well within the source range, period and count rate leveled out.

Secondary. Secondary system parameters reacted in a normal manner to a scram. Steam generator pressure rose slowly and then abruptly when the coolant pump was turned on, circulating higher temperature primary water. The hotwell level rose to a maximum from the water draining from the condensers over a period of two minutes. Three minutes were required for condenser pressure to reach atmospheric.

A part of the drop in steam generator level can be attributed to the water lost to blowdown before it was secured.

3. Predicted and Actual Transient Performance

Data from the PM-1 performance tests was evaluated and compared with design predictions. The evaluation is discussed below in three portions, namely:

- (1) Conclusions concerning overall system transient performance resulting from changes in electrical power.
- (2) Comparison of actual transient performance with analog simulation predictions.
- (3) Changes in steady-state primary coolant average temperatures before and after a transient.

a. Conclusions

The following conclusions, concerning actual plant transient performance resulting from abrupt changes in plant electrical demand, were obtained as a result of this evaluation:

- (1) The plant performance is well within design specifications.

- (2) Design predictions of transient performance were over-conservative. The inherent load-following characteristics of the PM-1 primary system and the feedwater controller tend to reduce the magnitude of plant transients resulting from abrupt changes in electrical demand.
- (3) The steady-state average coolant temperatures before and after transient differ slightly, the magnitude of the difference being a function of power. The cause is not completely explainable; however, it is apparent that this effect tends to reduce the magnitude of the transients.
- (4) The mathematical model used in the analog simulation is adequate, component-wise, in predicting basic plant performance. However, to more closely duplicate the actual plant performance, additional input must be incorporated. These include a more accurate simulation of the feedwater and steam flow relationship and a coefficient of reactivity to allow for the variation of average temperature with load.

b. Comparison of actual data with analog predictions

Design parameters, as predicted by the PM-1 analog computer simulation, were compared to actual test data. The conditions under which the plant data was taken differed somewhat from that simulated on the computer. Because of this, no precise comparison could be made. However, by reducing selected portions of the actual plant data to a graphical form similar to that produced by the analog computer, useful comparisons were made between simulated and actual performance. The most applicable test runs for this comparison were load swings between 200 to 600 kw, 0 to 1000 kw, 1000 to 0 kw and 600 to 200 kw. These were the actual transients nearest to those simulated on the computer.

The analog computer data had been calculated in terms of percentage of reactor thermal power. To obtain a comparison, the actual performance data was compiled and plotted in a similar manner, as follows:

- (1) Fig. 11--Load change from 600 to 200 kwe (net) represents a change in reactor power from 60.5 to 40.5%.
- (2) Fig. 12--Load change from 1000 to 0 kwe (net) represents a change in reactor power from 85.3 to 27.3%.

The nearest simulator transients available are shown in Figs. 13 and 14. Figure 13 is a typical transient of 30% (from 40 to 10% of rated reactor power). Figure 14 represents an 80% change (from 100 to 20% of rated reactor power).

Direct comparison of these figures indicates that the predicted temperature transients are much larger and of shorter duration than those actually experienced. For example, the plant test data indicates a maximum peak main steam temperature change of approximately 8° F from a transient leading to a 20% change in reactor power (Fig. 11), while the predicated change in peak steam temperature was +25° F during the transient corresponding to a 30% change in reactor power (see Fig. 13). Furthermore, the transient depicted by Fig. 12 resulted in an increase of steam temperature of 25.3° F, while the predicted peak for the case of an 80% change in reactor power was 48° F. These results show that the actual plant was experiencing the same increase of main steam temperature during a 58% change in reactor power that the simulator was predicting for a 30% power change.

To determine the reasons for this difference, all the steam-side and steam generator parameters used in the mathematical model were evaluated from the test data. No significant discrepancies could be detected in the total water storage volume, specific heats, heat transfer coefficient and heat of vaporization as they appear in the steam-side equation of the simulator model,

$$M_s c_p \dot{\bar{T}} = UA (\bar{T}_b - T_s) = W_g h_{fg} - W_{fw} (h_s - h_{fw})$$

where

T_s = saturation temperature of the water on steam side

$\bar{T}_b$ = primary side average water temperature

h_{fw} = feedwater inlet enthalpy

h_s = enthalpy of saturated fluid

W_g = mass steam flow rate

W_{fw} = mass feedwater flow rate

M_s = water mass in secondary side

c_p = specific heat of water

UA = overall heat transfer coefficient of the steam generator

h_{fg} = heat of vaporization.

One basic assumption made in the simulation was that the feedwater control system functioned to maintain mass steam flow and mass feedwater flow approximately equal. However, in evaluating the plant test data, it was discovered that the level compensating feedwater control system was actually causing a large mismatch between these two parameters during the early portion of the increased power transient. This caused a large excess of cold feedwater inflow over that assumed in the analog simulation (see Fig. 15). The effect on the magnitude of T_s can be seen by referring to the above equation. The higher value of feedwater flow above steam flow for a large portion of the total transient time increases the effect of feedwater cooling term in the equation. This, in turn, decreases the rate of change of steam temperature.

The reduction of the magnitude of the steam temperature transient reduces the magnitude of the transient imposed on the primary system, including the reactor. This serves to decrease the speed of response of the reactor by reducing reactivity effects and to minimize temperature transients in the primary loop. Furthermore, these smaller temperature changes result in smaller primary loop water volume changes, causing reduced pressure transients.

Load transients of the opposite direction exhibit the same characteristics noted above (see Figs. 16 through 19). In the plant test runs, the feedwater flow was again observed to be lagging behind steam flow during the transient, serving to reduce the effective amount of cooling during the transient time. Again, the thermal transients are smaller than those predicted from the simulation.

The overall effect of assuming matched steam and feedwater flow in the simulator was to build a high degree of conservatism into the plant analysis, with the result that all thermal transients are well within the predicted design values. In fact, it was concluded from analysis of this test data that the PM-1 plant is overdesigned for its transient performance specifications.

c. Average coolant temperature changes

Another area of evaluation of the PM-1 plant performance data is the primary loop calculated average temperature after system load transients. The figures exhibit a tendency for the steady-state average temperature to rest at a slightly different equilibrium value after the power transient. In the maximum power increase test case experienced

(Fig. 17), equilibrium errors amount to about -4.5°F . This includes the extremes of the scatter in the data; a better estimate is $3.5^{\circ} \pm 1.0^{\circ}\text{F}$.

The negative power transients (Fig. 12) indicate a negative temperature error of about $+6^{\circ}\text{F}$, including scatter; a better estimate is $+4.5^{\circ} \pm 1.5^{\circ}\text{F}$. These trends indicate that the arithmetic mean of the core inlet and outlet temperature is not an exact measure of core reactivity; and/or, during the course of the load transients, a negative reactivity effect other than the moderator coefficient is encountered. The $+4.5^{\circ}\text{F}$ moderator temperature error represents a core reactivity change of approximately $0.07\% \Delta\rho$; the change in moderator density corresponding to the 4.5°F temperature difference is less than 0.5% .

The plant transient data presented in Appendix A was investigated to determine trends. For precise conclusions, it would be necessary to have more data in which only a single independent variable is changed--for example, a complete series of load transients all initiated at one reactor power. Since this much testing was neither planned nor performed, only general trends can be deduced from the data. These are:

- (1) The magnitude of change of equilibrium reactor temperature is a function of power. Little or no effect is found for transients in which reactor power does not exceed 50% . For transients in which higher power levels are experienced, the magnitude increases as the power increases.
- (2) On transients during which reactor power increases, there is a power overshoot at low powers and not at high power. For example, Run VII of Appendix A, illustrating a load change from 0 to 200 kwe (net), shows that maximum reactor power is 7.5% compared to equilibrium power of 6.0% . In contrast, a more severe transient from 0 to 1000 kwe (net) (Run I of Appendix A) shows no power overshoot. A power swing from 200 to 600 kw shows a slight overshoot. This is exactly contrary to expectations.
- (3) On transients during which reactor power is decreased to below about 50% , a power undershoot is experienced, as was predicted by the analog simulation.

There appear to be two possible explanations: the variation of representative core moderator temperature with primary loop temperature and local boiling.

There is no doubt that the average of the reactor inlet and outlet temperatures does not exactly represent the true core average tem-

perature. In a load swing where reactor power is decreased appreciably, the inlet temperature increases and the outlet temperature decreases. If the average temperature, after the transient, is to be equal to the initial conditions, the temperature changes must be equal. However, there is no assurance that a 10° F temperature increase, for example, in the core inlet temperature has exactly the same reactivity effect as a 10° F decrease in outlet temperature. Thus, some change in the arithmetic mean temperature is not completely unexpected.

Another plausible explanation would be voids due to local (subcooled) boiling. Recent data (Ref. 1) shows that local boiling void fractions are larger than those anticipated during the PM-1 design phase. No complete analysis was attempted, but design data was briefly evaluated. It was found that local boiling in the PM-1 core begins at about 50% of design power. Predicted voids, even at full power, do not account for an average coolant density change of 0.5%. However, local boiling, of course, occurs in the high power production regions of the core, and the associated reactivity effects of density change would be greater than those based on average coolant conditions.

The net result on the plant performance of this characteristic is a reduction in magnitude of thermal transients. Again, this adds conservatism into the analog simulation in that the temperature and pressurizer level changes are less in the actual plant than those simulated on the analog.

4. Pressurizer Performance

Actual PM-1 pressurizer performance was evaluated. It was found that the actual plant transients were less severe than anticipated, as discussed above. At no time did the pressurizer pressure increase to the point where the pressurizer spray valve would operate. Volume surges into the pressurizer were about half the predicted values. This agrees with the comparison between the core coolant water temperature transients experienced and those predicted.

PM-1 pressurizer performance was analyzed, using data from the 1000- to 0- to 1000-kw and 1000- to 600- to 1000-kw load transients. The remaining runs resulted in surge volumes and pressure swings which were small compared to the recording error. The maximum volume surge encountered was 0.42 ft³ during the 1000- to 0-kw load transient. This resulted in a pressure rise of 30 psia. The pressurizer pressure ratio versus the expansion ratio was calculated and showed that the expansion process taking place in the pressurizer is slightly less severe than a saturation process and a good deal less severe than the isentropic

process for which the pressurizer was designed. With PV^K as a constant, the measured K is 0.5, compared to a value of 0.62 for a saturation process and 1.3 for an isentropic process. The following may be concluded from this analysis:

- (1) The PM-1 pressurizer can accommodate the maximum surge generated by the plant and maintain the primary pressure within narrow limits (± 30 to 40 psi).
- (2) In future PM plant pressurizer designs, consideration should be given to sizing on the basis of a saturation process, especially where surges of 60- to 70-sec duration are encountered.

An IBM program to size pressurizers, written as a portion of the PM R&D program, assumes the saturation process and is included in Ref. 11.

5. Transient Electrical Characteristics

Step changes in electrical load of 200, 400 and 1000 kw were utilized to obtain data to determine plant transient characteristics. Table 20 is a summary of the results obtained from the Brush recordings. The variation of voltage and frequency with time is shown in Fig. 20 for step changes of 0 to 1000 and 600 to 1000 kw.

The load bank did not permit step load changes of 300 kw, but the results indicate that, even with a 400-kw step load change, the design requirements were achieved. These same requirements were approached and in some cases achieved with a 1000-kw load change. It may be concluded that the turbine-generator meets all voltage and frequency response requirements under steady-state and transient conditions.

F. RESPONSE OF THE REACTOR TO ROD MOTION

1. General

The plant parameters were recorded during transients initiated by rod motion. Table 21 summarizes the test runs performed and lists the appropriate runs in Appendix A. Rods were inserted singly, in a group of three and in a group of six, at 91 and 46% power. Rods were extracted in a six-rod bank group at 42% power.

The most severe transient resulted from the insertion of a single rod for a distance of two inches, at the rate of two inches per minute. This particular case is discussed in Item 2 on the following page. The

other combinations gave the rough equivalent of a single-rod motion for one and one-half inches, though with a faster initial transient.

2. Transient Response

a. Reactor systems

As can be seen from Run XVI, the response of the system was as expected. The reactor went on a strong negative period as soon as rod motion started and the reactor output decreased. The reactor temperatures also decreased. With the decrease in temperature and the resultant volume contraction of the primary coolant, the pressurizer level and pressure decreased. After approximately six and one-half minutes, the control rod was slowly withdrawn to prepare for the next test run.

b. Secondary system

Since the demand on power had not changed, the main steam flow increased to satisfy the demand from the turbine. The transients on the remainder of the secondary systems were in line with these changes.

TABLE 20
Transient Electrical Characteristics

Load Swing (kwe, net)	Frequency Change (cycles, %)		Time to Return to:		Voltage Change (volts, %)		Time to Return to:	
			0.25% (sec)	0% (sec)			0.5% (sec)	0% (sec)
1000 to 0	0.25	0.416	2.8	9	2.47	2.06	2.2	2.8
0 to 1000	0.25	0.416	2.5	7.3	3.0	2.5	2.6	2.8
1000 to 600	0.12	0.2	0	4.7	0.84	0.7	1.2	2.0
600 to 200	0.11	0.183	0	3.5	0.986	0.8	0.9	1.1
200 to 400	0.1	0.167	0	3.2	1.62	1.35	1.3	4.0
0 to 200	0.08	0.133	0	2.5	0.92	0.77	0.6	2.7
800 to 1000	0.09	0.15	0	4.5	0.99	0.82	0	2.5
400 to 600	0.083	0.14	0	3.0	0.71	0.59	0	2.0
600 to 800	0.08	0.133	0	4.1	1.2	1.0	1.0	1.4
600 to 1000	0.12	0.2	0	5.5	0.92	0.76	0.4	1.0
400 to 200	0.09	0.15	0	2.3	0.60	0.5	0	1.2
1000 to 800	0.083	0.14	0	3.3	0.75	0.62	0.2	0.7
600 to 400	0.083	0.14	0	2.3	0.84	0.7	0.5	2.0
800 to 600	0.058	0.10	0	2.5	1.2	1.0	0.7	2.2

c. Conclusions

Recovery of both systems was as expected. The fluctuations in steam and feed flow are probably due to the turbine governor, which was giving trouble during this period.

The other rod motion transients exhibited the same characteristics as noted above, except that the initial transient was faster, as expected. The steam and feedwater flow fluctuations noted above were somewhat less severe.

The extract motion generated transients of the same order of magnitude as the insert transients, but in the opposite directions.

The test information available was also used to evaluate, in a rough manner, the apparent temperature coefficient of the PM-1 plant at normal operating temperature. Figure 21 shows test data taken for a 1-rod insertion at the slow rate from critical position of all rods at about 15.10 inches from bottom of the core. Total rod motion was 2.02 inches.

Single-rod worth varies from approximately 5% for five rods fully out and one rod fully inserted to about 2% for one rod fully withdrawn with the remainder of the rods fully inserted. Figure 21 shows that the equilibrium average temperature decreased by 14.5° F. Using the minimum and maximum rod worths, a range for the temperature coefficient can be established as 0.929×10^{-4} and $2.32 \times 10^{-4} \Delta \rho / ^\circ\text{F}$, respectively. The actual value used in the PM-1 power range simulations was $-1.7 \times 10^{-4} \delta K / ^\circ\text{F}$. Since the average value of rod worth will generally be somewhere between 2% and 5%, the value used in the simulations is realistic and valid.

TABLE 21

<u>Run No.</u>	<u>Page No. (Appendix A)</u>	<u>Milli-sadic Run No.</u>	<u>No. of Variables</u>	<u>Plant Transient</u>	<u>Load (kwe, net)</u>
XVI	A-40 to A-45	220	26	Rod insert, 1 rod 2 inches	1000
XVII	A-46 to A-47	230	8	Rod insert, 3 rods 1/2 inch	1000

TABLE 21 (continued)

<u>Run No.</u>	<u>Page No. (Appendix A)</u>	<u>Milli-sadic Run No.</u>	<u>No. of Variables</u>	<u>Plant Transient</u>	<u>Load (kwe, net)</u>
XVIII	A-48 to A-49	240	8	Rod insert, 6 rods 1/4 inch	1000
XIX	A-50 to A-51	190	8	Rod insert, 1 rod 2 inches	200
XX	A-52 to A-53	200	8	Rod insert, 3 rods 1/2 inch	200
XXI	A-54 to A-55	210	8	Rod insert, 6 rods 1/4 inch	200
XXII	A-56 to A-57	215	8	Rod extract, 6 rods 1/4 inch	200

G. PLANT AUXILIARY POWER REQUIREMENTS

The variation of auxiliary load requirements with plant output is illustrated in Fig. 22 for the situation where the electric motor-driven condensate and feedwater pumps are utilized. Figure 22 and Table 14 show that the hotel load--the power required for lighting, air conditioners and building ventilation--is 52.5 kw. To develop power up to the point where the generator can be synchronized with the source of startup power requires approximately 200 kw. The major variation in plant auxiliary power as a function of total generator output is caused by the power consumed by the condenser fans. The remainder of the auxiliary power load remains fairly constant with changes in both ambient temperature and plant load. The plot of auxiliary power versus plant load (Fig. 22) shows three distinct steps. These correspond to four, six and eight fans operating at high speed. The fans have 20-hp motors and thus require about 15 kw apiece. The amount of auxiliary power required will vary between 200 kw and 285 kw, depending on load and ambient temperature. The minimum value is expected in cold weather at low loads when one or more condensers may be secured and only one fan is necessary in the remaining units. The maximum amount will be required in warm weather at high loads when all condensers have both fans in operation.

The auxiliary power balance prepared during the final design phase of the PM-1 estimated that the average auxiliary power utilization, at full electrical power output, would be 210 kw. (A 250-kw auxiliary power transformer was specified for the plant.) Since this is 75 kw less than the actual auxiliary power consumption, the difference was investigated by using power measurements obtained at various times during the test program. The elements of these greater-than-predicted power consumptions are:

- (1) Hotel load is 39 kw greater than that contemplated during the final design phase, due to major revisions in the building and consequent changes in lighting and ventilating power consumption.
- (2) Condenser fan power is some 20 kw greater than predicted because the condenser fan blade angle was adjusted to enable full electrical power production with an ambient temperature of 85° F. (Design requirement is 70° F.)
- (3) Various individual items of power consumption were slightly underestimated. The combined average power consumption of the shield water cooler fans, the pressurizer heaters, feedwater pump, condensate pump and air compressor is approximately 16 kw greater than predicted.

H. SCRAM DATA

The plant operating history profile, Fig. 4, shows the occurrence of all unintentional scrams. Those scrams associated with malfunction reports are listed in Table 22. Corrective actions are described in Section IX of this report.

TABLE 22
Scrams

<u>Cause</u>	<u>Date</u>
Nuclear Instrumentation	
Channels 1 and 2	4/14/62
	5/18/62
	8/8/62
Channels 3 and 4	3/3/62
	3/30/62
	4/3/62
	6/16/62
	9/17/62

TABLE 22 (continued)

<u>Cause</u>	<u>Date</u>
Radiation monitoring	7/25/62
Turbine Governor Malfunction	6/19/62
	6/21/62
	8/4/62
	9/22/62
Electrical Malfunctions	
Station auxiliary power	4/10/62
	4/19/62
	7/18/62
Control rod actuator power supply	4/13/62
	6/13/62
Radar station malfunctions	4/3/62
	4/3/62
	9/22/62
Miscellaneous	3/30/62
	6/21/62
	8/7/62
Operator Error	5/16/62
	6/6/62
	7/19/62
	7/19/62
	7/23/62
	7/27/62
	9/13/62
	10/2/62
	10/2/62
	10/3/62
Undetermined Cause	6/7/62
	9/14/62
	9/17/62

Information concerning the time required to return to power operation is not particularly significant for the earlier data. In many cases, the plant was not at power prior to the scram and/or the plant was shut down for repairs, modifications, maintenance or crew training after the scram. The most representative data pertaining to the current ability to recover from unintentional scrams was obtained during the 400-hour run. This data is presented in Table 23.

TABLE 23

<u>Date</u>	<u>Time Required to Resume Power Operation</u>
September 12, 1962	10 hours, 5 minutes
September 14, 1962	9 hours, 42 minutes
September 17, 1962	5 hours, 47 minutes
September 22, 1962	5 hours, 30 minutes
September 29, 1962	5 hours, 17 minutes

I. ESTIMATED REFUELING TIME

The log data during the period between February 20 and February 26 was reviewed in detail. During this period, the dummy core was removed from the reactor vessel, and the actual core was inserted. The various steps conducted during each shift are illustrated in Fig. 23. It required a total of 128 hours to perform the removal and insertion of the core; however, a relatively large fraction of this time can be attributed to problems arising during this first simulation of refueling. The time lost due to problems which probably will not recur is also defined in Fig. 23.

Figure 24 presents the estimated refueling time span after elimination of time lost in correcting problems. Actual times required to perform the various steps during February were utilized in preparing the estimate. The estimate assumes that transfer of the spent core from the reactor tank to the spent core tank will require essentially the same time that the dummy core requires. This assumption is realistic, since much of the time involved in transferring the spent bundles will be concurrent with bundle removal activities. The estimate illustrated in Fig. 24 shows that a total of 86 hours is required from the time the plant is shut down until the venting and hydro testing of the primary system after insertion of the new core is accomplished.

J. PARALLEL OPERATION

The ability of the nuclear power plant to parallel other sources of power has been demonstrated by the many times the plant has operated in parallel with the site diesel generators.

In order to operate two or more generators in parallel, it is necessary that all, or all but one, of the units have a drooping characteristic. That is, the generator rotational speed must decrease somewhat

with increasing load. The stringent frequency requirements of the PM-1 plant preclude such a characteristic in the plant turbine-generator unit. Therefore, to make parallel operation possible, it was necessary to adjust the governors on the radar site diesel generators to obtain this drooping characteristic.

When operating in parallel with the site diesel generators, it is the PM-1 unit, with its nondrooping characteristic, which establishes the system voltage and frequency. If the total system load were to increase, the entire increase would be picked up by the PM-1 unit. This would occur because every other diesel generator would already be carrying a load uniquely defined by its frequency-load characteristic and the system frequency. Therefore, a load limiter was installed on the PM-1 generator unit to protect it from exceeding its design rating.

IV. PLANT RADIATION LEVELS AND HEALTH PHYSICS

Data presented in this chapter is concerned with:

- (1) Radiation levels during power operation.
- (2) Argon activity.
- (3) Radiation levels following power operation.
- (4) Personnel radiation exposure history.

A. RADIATION LEVELS DURING POWER OPERATION

A thorough determination of radiation levels throughout the plant was made. This included:

- (1) Experimental measurements and analytical predictions of fluxes within the reactor tank.
- (2) Experimental measurements of fluxes within the steam generator tank.
- (3) Measurements to determine possible radiation streaming from the reactor and steam generator tanks.
- (4) Measurements of dose rates at all plant and area monitors.

The fast and thermal neutron flux and the gamma dose rate were measured in a radial plane extending outward from the lead shield surrounding the PM-1 reactor pressure vessel. The thermal neutron flux and the gamma dose rate were measured in a vertical traverse along the length of the steam generator tank in the vicinity of the ladder which is fastened to one side of the tank. The thermal neutron flux was measured in the steam generator tank along a vertical traverse across the face of the shielding at the interconnect pipe leading to the reactor tank. The gamma dose rate was measured along several traverses through the primary building itself and at several locations on the shelf at the top of the double wall of the reactor tank.

The techniques used for making these measurements are described in Appendix B. The following individual results obtained are reported for the four areas listed above. Where applicable, analytical predictions are included and discussed.

1. Reactor Tank

The neutron and gamma flux within the reactor tank was determined experimentally and analytically. Results were normalized to correspond to a power level of 10.0 Mw(t). Experimental measurements were made using the foils and foil holders described in Ref. 2.

In addition to experimental measurements, predictions of flux levels were made to aid in establishing the experimental program and techniques and also to provide a comparison of analytical and experimental results. Detailed discussions of the analytical methods are provided in Appendix C.

The flux measurements made in the reactor tank were concerned with thermal flux, fast flux and gamma flux. The experimental results and analytical predictions are discussed below.

a. Thermal flux

In the reactor tank, calculations were made of the radial and axial distributions of thermal flux levels in the shield water. The resulting data was interpolated to obtain flux distributions at the same locations as those at which the flux levels were determined experimentally, thus permitting a direct comparison between analytical and experimental data. The thermal neutron flux calculations were based on the P1MG method (Ref. 3).

Figure 25 presents the axial thermal neutron flux distributions calculated at radial distances of 4.1, 24.4 and 50.0 centimeters from the lead shield. These correspond to 72.7, 93.0 and 118.6 centimeters, respectively, from the center of the core. The axial orientation of these curves has been normalized to the axial peaks of the experimentally determined curves by shifting all analytical data up 5 centimeters axially. The experimental points measured at the same radial locations have also been plotted in Fig. 25. The uncertainty band for each point represents the spread of experimental data and does not include the 12% experimental uncertainty. This 12% represented the relative uncertainty. The absolute error of the experimental data would be somewhat higher.

The calculated thermal flux values vary from 43% higher than the experimental data at 4.1 centimeters radially from the lead shield to 53% lower at 50.0 centimeters from the lead.

The radial thermal neutron flux distribution calculated at 31.7 centimeters above the bottom of the active core is shown in Fig. 26.

This axial location corresponds to the experimental radial traverse measured near the peak of the axial flux. As in Fig. 25, Fig. 26 includes the experimental data points with the data spread which correspond to the analytical curve. Additional experimental points are presented in Fig. 27, showing the radial variation of thermal flux at an axial location 74.9 centimeters above the bottom of the core.

It is concluded that the analytical technique, utilizing the P1MG approximation, calculated thermal neutron fluxes with a reasonable degree of reliability in the peak flux regions. In general, there is good agreement between the shapes of the analytical and experimental distributions. As would be expected, though, the method tends to overestimate the thermal flux near the radial flux peak and to underestimate the flux at deep penetration. Recognizing that the absolute experimental error associated with these measurements is somewhat greater than $\pm 12\%$ of the measured value, the agreement between analysis and experiment is quite good, thus validating the analytical technique.

b. Fast flux

Calculations of the radial and axial distributions of fast flux in the reactor tank were performed, using the P1MG method for determination of thermal flux distributions. The fast flux was considered to be the first 7 energy groups, extending from 1.74 to 10 Mev.

The experimental fast neutron data was measured and reported in terms of the activation of Sulfur-32. To permit comparison of calculated and measured fast neutron data, the calculated flux was converted to Sulfur-32 activations and is presented in this manner in Figs. 28 through 30.

The correlation between activation and flux is not constant, because the activation cross section varies with neutron energy and the flux spectrum varies spatially. The approximate fast flux (greater than 1.74 Mev) can be determined within the axial confines of the core from the equation:

$$\text{Flux} \approx \text{Activation} (11.6 - 0.124 x)$$

where x is the distance in centimeters from the lead shield.

Figure 28 shows the fast neutron Sulfur-32 activations calculated at radial distances of 6.0, 17.5, 28.9 and 40.3 centimeters from the lead shield. These correspond to 74.6, 86.1, 97.5 and 108.9 centimeters, respectively, from the center of the core. As was done previously, Fig. 28 includes the experimental data points and their data spread, measured at the same radial locations. The axial peaks of the analytical curves have been matched to the experimentally

determined peak axial activations by shifting the axial data up 5 centimeters, as for the thermal flux curve.

Agreement between the analytical and experimental data at the peak of the axial flux varies radially. The ratio of experimental to calculated values is 1.8 at 6.0 centimeters from the lead shield and 2.8 at 40.3 centimeters from the lead. This radial variation is not unusual; the accuracy of flux calculations usually decreases as one moves farther from the core.

Figure 29 presents the analytical curve and experimental data for the radial fast activation distribution at 30.5 centimeters above the bottom of the active core, which is the axial location of the experimental measurement nearest to the peak of the axial flux. Figure 30 shows the experimentally determined radial distribution at a distance of 64.8 centimeters above the core.

The analytical predictions were utilized to estimate the integrated flux at the pressure vessel. Considering all energy groups above 1.0 Mev, an integrated flux of 5.2×10^{20} nvt is predicted for a 20-year period. As pointed out above, the calculational accuracy improves as one approaches the core; thus, the predicted number is probably accurate. However, based on the experimental data, the prediction might be low by as much as a factor of 1.8.

c. Gamma flux

Gamma fluxes were determined experimentally and also calculated from the thermal neutron flux distributions described above. The source distributions were integrated over 5 gamma energy groups and spatially integrated, using the SPEND code. In each radial region, the axial source term varies throughout the region; however, the SPEND code can only consider a single value. To determine and illustrate the effect of this calculational uncertainty (there are many others), the gamma fluxes were calculated from the axial thermal flux distributions at both the beginning and the end of each region. The results of these calculations are shown as the "upper and lower limits" in Figs. 31 and 32 and in Table 24.

Table 24 lists the contributions of the various sources to the gamma flux at different radial locations. Figure 31 shows the analytical and experimental axial gamma dose rates at 17.5, 28.9 and 40.3 centimeters from the lead shield, corresponding to 86.1, 97.5 and 108.9 centimeters, respectively, from the center of the core. The same five-centimeter upward axial normalization was applied to the analytical curves that had been used for the thermal and fast neutron flux curves. The gamma fluxes were calculated in $\text{Mev/cm}^2\text{-sec}$. For

comparison with experimental data, the calculated fluxes in each energy group were converted to dose rates in rads per hour and summed over all energies. At each radial location, two analytical curves have been plotted to show the limits established by the variation in axial source term. The curve for the upper limit was established using the source term at the outer boundary of the region. The lower limit curve was based on the source term at the inner boundary.

TABLE 24
Contribution of Various Sources of Gamma
Radiation to the Total Dose Rate

A. A point 7.4 centimeters from lead surface and 30.4 centimeters from bottom of core.

<u>Source</u>	<u>Dose Rate (rads/hr)</u>	
1. Core gammas	2.7×10^3	
2. Steel capture gammas	1.23×10^6	
3. Lead capture gammas	5.6×10^4	
4. Hydrogen capture gammas in shield water (upper and lower limits)	$(2.4 \times 10^6)_u$	$(1.95 \times 10^6)_L$
Total	$(3.7 \times 10^6)_u$	$(3.15 \times 10^6)_L$

B. A point 37.4 centimeters from lead surface and 30.4 centimeters from bottom of core.

<u>Source</u>	<u>Dose Rate (rads/hr)</u>	
1. Core gammas	5.6×10^1	
2. Steel capture gammas	9.0×10^4	
3. Lead capture gammas	5.1×10^3	
4. Hydrogen capture gammas in shield water (upper and lower limits)	$(1.23 \times 10^5)_u$	$(0.96 \times 10^5)_L$
Total	$(2.2 \times 10^5)_u$	$(1.95 \times 10^5)_L$

Analytical and experimental gamma dose rate data could be compared at the axial peak at only two radial locations. Based on the lower limit curves, the calculated values are a factor of 1.83 and 1.61 higher than the experimental data at 28.9 and 40.3 centimeters, respectively, from the lead shield. Considering the upper limit curves, calculations are a factor of 2.15 and 1.88 higher than experiment at the same 28.9 and 40.3 centimeters from the lead.

In Fig. 32, the radial analytical and experimental distributions at 7.6 and 64.8 centimeters above the bottom of the active core are presented. As in the previous figures, both the upper and lower limit curves for the analytical distributions are shown.

In conclusion, it is found that the agreement between analytical and experimental gamma flux distributions is as reasonable as should be expected. The analytical values are conservative, being higher in all cases than the experimental values. The shapes of the analytical and experimental curves agree quite well. The magnitude of the calculated values is higher by a factor of 2.2 to 1.6 than the experimental values, which have an associated experimental error of somewhat more than $\pm 12\%$ of the measured value. For this type of calculation, this represents reasonable agreement.

The SPEND code, which was used to compute gamma dose rates, utilizes a three-dimensional Simpson's rule technique to numerically integrate the gamma point kernel over source regions to the detector point. Within any source region, the source distribution is assumed separable in the space coordinates with sources represented by third degree polynomials. Obviously, with an increasing number of source points, results may be expected to approach the correct solution asymptotically. Practically, however, calculations must necessarily be limited to a reasonable number of points to prevent excessive computing costs.

The major portion of the discrepancy between theory and experiment can be attributed directly to the above discussed inherent uncertainties in SPEND calculations. For reactor design, the calculations are considered sufficiently accurate, since the discrepancies are relatively small and theoretical results are conservative.

2. Flux Within the Steam Generator Tank

Gamma fluxes in the steam generator package were determined experimentally. A detailed discussion of experimental techniques is provided in Appendix B. Experimental data, normalized to a power level of 10.0 Mw(t), concerning thermal neutron flux and gamma flux is presented below.

a. Thermal neutron flux

The thermal flux in the steam generator tank is shown in Fig. 33 as a vertical traverse through the tank in the region of the ladder which extends the length of the tank. The thermal flux at the interconnect in the steam generator tank is shown in Fig. 34. Precise detector positioning was difficult, so the difference between the two separate measurements in the interconnect region probably reflects the existence of fairly steep gradients in the neutron flux due to some inhomogeneities in the shielding. However, this does not represent any gross streaming, since the measurements indicate far less than an order of magnitude variation and levels are lower than predicted by design calculations.

It is significant to note that the measured thermal neutron flux in the air of the steam generator tank (with a maximum value of approximately 3×10^4 n/cm²-sec) indicates that the design objective of limiting the flux to less than 10^6 n/cm²-sec at any point within this component has been met. Although measurements in air do not indicate flux maximums in water or steel, the flux at any point does not exceed 10 times the maximum in air. Thus, thermal neutron flux in the package probably does not exceed 3×10^5 . This means that long term buildup of neutron-induced activity of the steel within this package will not significantly increase the after-shutdown gamma dose rates in the package.

b. Gamma flux

The gamma dose rate measurements in the steam generator tank are shown in Fig. 35. The relatively large scatter of the data reflects both the inaccuracies of the film dosimetry technique and the relatively poor positioning accuracy of the films.

The experimentally determined vertical gamma dose rate distribution in the steam generator package shows an approximately exponential decrease from the bottom of the tank with no increase in the region of the interconnect. This indicates that there is little or no gamma streaming in the interconnect areas. The exponential decrease in the dose rate near the bottom of the tank may be predicted by considering the "effective" increase in shielding due to increasing slant paths through shield materials.

3. Measurements of Possible Streaming

Gamma dose measurements were performed at critical areas in the primary building to determine if significant radiation streaming occurs. The measurements showed that there was no detectable streaming, even with the steam generator tank cover removed.

Of particular interest were:

- (1) The reactor package double wall (air space between the water and earth shield).
- (2) Streaming from the steam generator tank.

During plant design, it was determined that these areas represented the most likely regions for streaming to occur. Particular emphasis was placed on evolving a design which minimized any such streaming. For the reactor package double wall, a portion of this air space near the shield water surface was filled with polyethylene shot, thus eliminating a "straight path" for radiation. Blocks of polyethylene and Sigma-K shielding in the interconnect were "staggered" to eliminate streaming paths. However, due to piping penetrations and material tolerances, it was anticipated that streaming could occur. Thus, a steam generator shield tank lid was provided primarily to attenuate neutrons scattered in the lower region of the steam generator package in the event that there was appreciable streaming of neutrons through the interconnect. As previously indicated, no streaming is present even with this lid removed.

The gamma survey in the primary building was performed at five locations on the reactor tank ledge and along five horizontal traverses through the building as shown in Fig. 36. The measurements on Traverses I, II and III were taken at a level of approximately 3 inches above the deck plates and at increments of 1 foot. Traverses IV and V were taken at a level of approximately 40 inches above the deck plates and at increments of approximately 10 inches. The measured dose rates are shown in Tables 25 through 27.

4. Dose Rates at Plant and Area Monitors

As part of the test procedure for the PM-1 plant, radiation surveys were performed to survey normal access areas of the plant during operation so as to establish radiation levels. Generally, results of measurements were lower than those predicted during the design of the system. In most cases, measured values at full power are a factor of 2 to 10 lower than the predicted maximum levels. Results of the survey are presented in Table 28.

In many cases, the geometry utilized in the calculations represented a "worst case" with respect to radiation levels and self-shielding. This was particularly true in the case of estimating gamma dose rates from decay of N^{16} in the primary coolant within the steam generator. Final design of this component was completed after shield design. Calculational uncertainties led to an underestimate of self-shielding. In general, such instances tend to indicate that the factors of 2 to 10 mentioned above are reasonable.

TABLE 25
Gamma Dose Rate in Primary Building at
Full Reactor Power

Distance from Building Wall (ft)	Gamma Dose Rate (mr/hr)*		
	Traverse I	Traverse II	Traverse III
1	38	--	1.7
2	94	--	2.8
3	123	28	5.3
4	113	30	7.5
5	66	19	13
6	38	7.5	36
7	53	8.5	47
8	43	9.5	45
9	34	9.5	53
10	30	5.7	66
11	38	6.5	51
12	32	6.5	47
13	30	7.5	38
14	32	6.5	28
15	25	4.5	4.7
16	7.5	2.3	--
17	7.5	2.3	--
18	2.6	1.3	--
19	1.7	1.1	--

*Instrument utilized was Jordan Radgun.

TABLE 26
Gamma Dose Rate In Primary Building at
Full Reactor Power

Distance from Steam Generator Tank Railing (in.)	Gamma Dose Rate (mr/hr)*	
	Traverse IV	Traverse V
0	100	57
10	120	57
20	140	100
30	150	130
40	140	140
50	150	160
60	180	170
70	190	230
80	130	230
90	60	250
100	--	250

*Instrument utilized was Jordan Radgun.

TABLE 27
Gamma Dose Rate on Reactor Tank Ledge at
Full Reactor Power

Location	Dose Rate* (mr/hr)
A	53
B	3.2
C	1.8
D	4.9
E	38

*Instrument utilized was Jordan Radgun.

TABLE 28

PM-1-T46 for Plant Performance Tests (all numbers in mr/hr unless indicated)*

Location of Measurement	Power Level 66 to 70%; Time 0930; Date 9/17/62	Power Level 100%; Time 0845; Date 9/19/62	Power Level 45%; Time 1440; Date 9/25/62
1. Main door to secondary building between office and building	0.052	0.048	0.055
2. Entrance to control room	0.054	0.052	0.051
3. Center of control console	0.055	0.055	0.055
4. Shift supervisor's desk (out- side control room)	0.055	0.050	0.050
5. Center aisle at junction of maintenance and switchgear packages	0.055	0.052	0.051
6. Entrance to radar station walkway	0.052	0.055	0.055
7. Center of walkway between secondary building and decontami- nation package	0.055	0.065	0.057
8. Entrance to "cold" side of decon- tamination package	**0.175	**0.22	**0.22
9. Entrance to "hot" side of decon- tamination package	0.15	0.15	0.15
10. Center of "hot" side of decon- tamination package	0.15	0.15	0.15
11. Entrance to primary building	0.40	0.43	0.30
12. Entrance to primary building storage area	0.48	0.33	0.27
13. Top of spent fuel tank (waist height)	0.70	0.70	0.50
14. Top of shield water tank (waist height)	6.5	15	5.5
15. Top of steam generator package (ladder at floor)	110	200	110
16. Shield water cooler package (inside)	35	80	25
17. Top of waste disposal tank	0.13	0.125	0.095
18. Outside secondary building (front door)	0.065	0.065	0.050
19. Walkway to radar station (tech- nical supply door)	0.060	0.060	0.047
20. Technical supply building (inside)	0.055	0.055	0.048
21. Entrance to tower (FPS6)	0.052	0.050	0.050
22. South entrance to operations building (outside)	0.055	0.055	0.054
23. AP control and ID building	0.054	0.053	0.052
24. Outside primary building (double doors)	0.22	0.24	*1.0
25. Between off gas monitor and primary building	2	4	1.7
26. Outside shield water cooler package (south, east, north)	1.2	2.6 3.0 5.5	1.25 1.4 3.5
27. Outside north wall primary building (at piping braces)	2.4	6 to 10	4
28. Ten feet east of shield water cooler package	0.25	0.40	0.30
29. Ten feet north of primary building north wall	0.18	0.30	0.25

*Instrument utilized was Jordan Radgun.

**Additional source of gamma radiation present due to source storage and/or testing.

TABLE 28 (continued)

Location of Measurements	Power Level 66 to 70%; Time 0930; Date 9/17/62	Power Level 100%; Time 0845; Date 9/19/62	Power Level 45%; Time 1440; Date 9/25/62
30. Center steam generator tank top	200	250	150
31. Side of pressurizer at 1 foot out	350	600	260
32. Grating in steam generator tank	4 r/hr	6 r/hr	2.6 r/hr
33. Top of primary coolant pump	25 r/hr	30 r/hr	15 r/hr
34. Bottom of primary coolant pump	35 r/hr	50 r/hr	30 r/hr
35. Main coolant piping below coolant pump	70 r/hr	100 r/hr	60 r/hr
36. Side of steam generator (4 feet down)	1.5 r/hr	1.8 r/hr	1 r/hr
T46-A-1 Readings taken on deck at tank top			
1. North side steam generator tank (waist height)	20	35	20
2. East side steam generator tank (waist height)	100	150	55
3. West side steam generator tank (waist height)	35	38	31
4. South side steam generator tank (waist height)	7	12	7

B. ARGON ACTIVITY

Airborne radioactivity may be produced as a result of neutron interaction with the basic constituents of air and airborne contamination in regions where relatively high neutron fluxes exist. In the PM-1 plant, significant activity is most likely to be produced within the steam generator package, wherein the highest neutron flux levels in air exist. However, levels were below detectable limits; hence, the argon activity level was calculated, for a conservative case, using measured values of neutron flux. It was found that for this "worst case," levels were only one-fifth of the mpc value of 4.0×10^{-8} microcuries/milliliter for unrestricted areas.

Forced ventilation required for cooling this package during power operation continually passes air through the unit and could conceivably form a closed loop circuit which has as its components the ventilation ducts, steam generator package and the primary building above the packages. Activity produced as a result of this closed, circulating air system represents the "worst case" in regard to normal plant operation.

Of particular importance is the A^{40} (n, γ) thermal neutron reaction which produces radioactive A^{41} with a 1.83-hour half-life and decays by negative beta emission with an associated 1.29-Mev gamma. Argon activity produced in air in a flux field will normally be much greater in magnitude than other activity produced in the same air sample. Utilizing experimental results for the thermal neutron flux in the steam generator package and the conservative closed-loop model described above, the maximum A^{41} activity was computed to be 7.6×10^{-9} microcuries per milliliter of air. This value is 19% of that prescribed by CFR-10-20 for unrestricted areas (CFR value is 4×10^{-8} microcuries/milliliter of air).

Details of the computations are presented in Appendix C.

C. RADIATION LEVELS AFTER SHUTDOWN

Data concerning radiation levels in the steam generator and waste disposal packages after shutdown was compiled. Dose rates in the waste disposal system were measured just prior to initiation of waste processing on September 28. After processing was complete, levels were again measured. This data is presented in Table 29.

TABLE 29
Radiation Levels in Radioactive Waste Disposal Package

<u>Location</u>	<u>Before Operation (mr/hr)</u>	<u>After Operation (mr/hr)</u>
Hatchway opening	0.125	0.15
Top of sump tank	0.4	4.2
Bottom of sump tank	4.7	4.2
Top of evaporator	2.0	3.5
Side of condenser	1.0	0.9
Bottom of evaporator	1.1	1.25
Condensate pump	0.9	1.6
Transfer pump	0.5	0.9
Deck (above storage tank)	0.5	1.0

It was intended to measure the radiation levels in the steam generator package as a function of time after shutdown. However, due to the tight schedule, many of the plant performance tests were conducted as part of the 400-hour test. At the completion of this test, the plant was shut down but kept at temperature in an attempt to measure reactivity changes due to xenon buildup. As discussed in Section V, G, it was necessary to operate the reactor at low power during this test. Thus, no after-shutdown radiation levels in the steam generator package could be obtained. The only data available was obtained toward the end of May, and this represents radiation levels 2 hours after shutdown from the power level of 700 kw. This data is presented in Table 30.

D. PERSONNEL RADIATION EXPOSURE HISTORIES

Radiation exposure histories of the military crew were tabulated for the first three calendar quarters of 1962. This information is presented in Table 31.

TABLE 30
Radiation Levels in Steam Generator Package
(2 hours after shutdown)

<u>Location</u>	<u>Radiation Level (mr/hr)</u>
Steam generator package (floor level, primary building)	0.4
Pressurizer	1.5
Expansion tank	7.0
Demineralizer	7.0
Primary pump	15 north side; 25 west side
Purification cooler	5.0
Steam generator	10 at grating
General area average	
5 feet down	1.0
10 feet down	2.0
15 feet down	10.0
20 feet down	10.0
25 feet down	20.0
Interconnect to shield water tank upper portion	
North side	30.0
Lower portion	50.0

TABLE 31
Radiation Exposure Histories

<u>Individual</u>	<u>Jan 1 to Mar 31, 1962</u> (millirem)		<u>Apr 1 to Jun 30, 1962</u> (millirem)		<u>Jul 1 to Sep 30, 1962</u> (millirem)		<u>Integrated Exposure</u> <u>Prior to PM-1</u> <u>Assignment</u> (rem)
	<u>Dosim- eter</u>	<u>Film Badge</u>	<u>Dosim- eter</u>	<u>Film Badge</u>	<u>Dosim- eter</u>	<u>Film Badge</u>	
Bockstedt	10	< 50	101	*	110	< 20	2.58
Hobson	38	< 50	170	*	38	< 20	25.865
Singleton	48	< 50	217	*	102	< 210	0.440
Allen	22	< 50	39	*	76	< 20	1.09
Fierabend	10	< 50	123	*	28	< 20	1.551
Feil	80	< 50	137	*	20	< 20	0.555
Brown	7	< 50	69	*	62	< 20	1.36
Gay	66	< 50	192	*	59	< 20	3.801
Ondek	54	< 50	171	*	126	41	0.252
Sprey	21	< 50	5	*	5	< 20	3.386
Brownlee	32	< 50	268	*	81	< 20	0.343
Ellis	191	< 50	166	*	200	< 20	0.314
Leckell	151	< 50	317	*	131	< 20	0.218
Spurlin	184	< 50	385	*	184	< 20	1.548
Ulrich	42	< 50	145	*	99	143	0.324
Hofer	170	< 50	80	*	18	< 20	0.548
Garrison	129	< 50	110	*	144	< 20	1.262
Raskin	67	< 50	46	*	309	< 20	0.177
Quick	10	< 50	137	*	124	< 20	None
Kligman	72	< 50	118	*	134	179	1.387

*Film badge record not available, since badge was fogged.

V. PLANT INSTRUMENTATION AND CONTROL

Data pertaining to the PM-1 plant instrumentation and control is presented in this section. The following items are discussed individually:

- (1) Scram response times.
- (2) Actuator electrical power requirements.
- (3) Final instrumentation set and trip point.
- (4) Amount of overlap between channels.
- (5) Flux to which nuclear instrumentation is exposed.
- (6) Xenon override demonstration.
- (7) Instrumentation recalibration data.
- (8) Radiation monitor detection limits and alarm points.
- (9) Instrumentation electric power requirements.

A. SCRAM RESPONSE TIMES

The scram response times of the nuclear and process instrumentation were determined from Brush recorder data obtained during a test where simulated parameters were introduced into the circuits to provide the initial scram conditions. The results are presented in Table 32 and show that the majority of time delay is in the process instrumentation amplifiers and bistables.

The process instrumentation chain (instrumentation associated with temperature, pressure or pump power scrams) is listed below with average delay time or time constants.

- (1) Process amplifier--time constant of 320 milliseconds.
- (2) Process bistable--delay time of 420 milliseconds.
- (3) Process bistable relay, safety system and scram amplifier--delay time of 9 milliseconds.

TABLE 32
Scram Response Times

Elapsed Time from Simulated Signal Input (milliseconds)											
	High Temperature		Low Pressure		Low Flow	Low Pump Power	High Power			Short Period	
	Expanded Range	Full Range	Expanded Range	Full Range			Channel 5	Channel 6	Channel 7	Channel 3	Channel 4
Time to trip scram bistable	659	912	600	738	650	878	~0	~0	~0	48	62
Time to initiate scram amp trip	676	923	608	748	660	884	4	5	5	50	65
Time to drop out scram relay	682	930	617	753	668	889	9	10	9	54	73
Time to initiate rod motion	768	1016	697	820	743	973	89	87	88	136	154
Time for rod movement of 12 inches	977	1225	906	1029	952	1182	298	296	297	345	363

- (4) Control rod scram relay--delay time of 4 milliseconds.
- (5) Initiation of control rod motion--delay time of 82 milliseconds.

Similar data for the components of the nuclear instrumentation chain (instrumentation associated with reactor power or short period scrams) is presented below:

- (1) Nuclear bistable--delay time of ~ 0 millisecond.
- (2) Control rod scram relay--delay time of 4 milliseconds.
- (3) Initiation of control rod motion--delay time of 82 milliseconds.

The response is dependent on the magnitude of the change and the set point of the bistable. This fact is especially noticeable in the high temperature scram response, where the time delays until bistable trip for the expanded range and full scale range were 659 and 912 milliseconds, respectively. These delays were a result of a simulated step change in temperature from approximately 465° to 600° F. If the simulated temperature change had been less, the time delay would have been even greater. The response times of the basic temperature sensing elements are not considered, but these would probably add considerably to the delay shown by the electronic instrumentation.

B. ACTUATOR POWER

The 440-volt actuator power utilization was measured at 4.4 kva when the 6-rod bank was being withdrawn and 1.2 kva with the rods stationary. The 120-v input power was not obtained during this test; the PM-3A value was approximately 0.8 kva.

C. FINAL INSTRUMENTATION SCRAM AND SET POINTS

Data concerning the final instrumentation set points associated with the safety circuits was compiled. The other instrumentation set points are listed in the final revision to the operating manual.

Table 33 contains data on the scram and rod hold instrumentation settings. Table 34 presents information concerning alarm and interlock bypass set points.

D. AMOUNT OF OVERLAP BETWEEN CHANNELS

Data related to the overlap between the source, intermediate and

TABLE 33

Final Scram and Rod Hold Set Points

<u>Item</u>	<u>Instrumentation Category</u>	<u>Nuclear Instrumentation Channel No.</u>	<u>Nuclear Instrumentation Bistable No.</u>	<u>Set Point</u>
<u>Scram Points</u>				
Short period	Source range	1	2	15 sec
Short period	Source range	2	4	15 sec
Short period	Intermediate range	3	6	15 sec
Short period	Intermediate range	4	9	15 sec
High power	Power range	5	11	120% of full power
High power	Power range	6	12	120% of full power
High Power	Power range	7	13	120% of full power
Reactor outlet temperature	Process	NA	NA	500° F
Primary coolant low pressure				
(1) Expanded scale	Process	NA	NA	1210 psig
(2) Full range	Process	NA	NA	1210 psig
Primary coolant pump power	Process	NA	NA	90% or less at rated power
Pressure drop across primary coolant pump	Process	NA	NA	30 psi
<u>Rod Hold Points</u>				
Low count rate	Source range	1 or 2	1	5 cps
Short period	Source range	1 or 2	3	30 sec
Short period	Intermediate range	3 or 4	7	30 sec

TABLE 34

Final Alarm and Interlock Bypass Set Points

<u>Item</u>	<u>Instrumentation Category</u>	<u>Set Point</u>
<u>Alarm Points</u>		
High power	Power range	110% of full power
Channel discrepancy	Nuclear instrumentation	10% of full range
Short period	Source range	30 sec
Short period	Intermediate range	30 sec
<u>Interlock Bypass Points</u>		
Startup period bypass	Intermediate range	10^4 nv
High voltage removal	Intermediate range	10^5 nv
Intermediate period bypass	Power range	10% of full power

power range channels was compiled. It was found that the intermediate range channels overlap the source range by approximately 1.8 decades and overlap the entire power range instrumentation.

The minimum intermediate range meter reading is 2×10^3 nv. The intermediate range meter reading corresponding to the maximum practical source range reading is 1×10^5 nv. Thus, the intermediate range overlaps the source range by approximately 1.8 decades.

The intermediate range is operational to 2.5×10^{10} nv, and fully overlaps the power range instrumentation to its maximum of 150% full reactor power.

E. FLUX TO WHICH NUCLEAR INSTRUMENTATION IS EXPOSED

During the PM-1 plant performance studies, the radial and axial thermal neutron flux was measured in the shield water region outside the reactor pressure vessel. These measured fluxes were utilized to determine the fluxes to which the nuclear instrumentation was exposed. It was found that, at 100% power, the average flux experienced by the power, intermediate and source range counters are 3.7×10^9 nv, 4.5×10^9 nv and 3.2×10^9 nv, respectively.

The tops of the counters are located approximately at the plane of the top of the active core. The lengths are shown in Table 35. Since these lengths are appreciable, it was necessary to axially average the flux over the length and position of the counter. This axially averaged flux is shown in Fig. 37, and, along with the radial location of the counters, determines the average flux at the counters. This data is presented in Table 35.

TABLE 35

Flux at Nuclear Instrumentation

<u>Counter</u>	<u>Length (cm)</u>	<u>Radial Distance from Lead Shield (cm)</u>	<u>Axially Averaged Flux (nv)</u>
Power range	35	30	3.7×10^9
Intermediate range	69	30	4.5×10^9
Source range	23	33	3.2×10^9

The actual flux at the intermediate counters was compared to the counter readings. To determine the intermediate channel readings at 100% full power, the operational logs of the PM-1 were used to plot the intermediate channel readings as a function of the power channel readings. This is shown in Fig. 38. (The power channel readings were calibrated to reactor power as discussed below in Section G.) Approximately 220 readings, taken over a period of eleven days, were plotted for Channels 3 and 4. A line which equates the rate of change of the counter readings to the rate of change of the power readings has been superimposed on each set of data and is used to determine the counter reading for any given percent power value between 40 and 110%. It was found that the counter utilized for Channel 4 agrees closely with the measured flux and that the counter associated with Channel 3 reads low by a factor of almost 2.

F. XENON OVERRIDE DEMONSTRATION

An attempt was made to demonstrate that the PM-1 reactor is capable of overriding the maximum rate and total magnitude of xenon buildup. There has been no difficulty due to xenon effects in restarting the reactor throughout the entire PM-1 performance program; however, operational difficulties and the basic reactor characteristics early in core life made it impossible to follow the test procedures precisely. The major problem stems from the fact that, at the beginning of life, the heat input due to fission product decay and the primary coolant pump power do not equal the heat loss in the reactor coolant system. Thus, it is impossible to keep the system at operating temperature without the reactor operating at some power.

To determine the rate of buildup of xenon after shutdown, control rod position must be adjusted so as to keep the reactor just critical, and at temperature, without operating at power. The difference in rod position can then be related to xenon reactivity effects. It is necessary to keep the reactor at temperature so that the xenon effects are not masked by the temperature coefficient and to operate at virtually zero power so that the xenon is not burned up. As stated above, it was impossible to meet these conditions at the PM-1 site. The reactor was kept at temperature but was operating at a power level of approximately 2%. Thus, much of the xenon was burned up and virtually no rod movement was required to keep the reactor critical.

An analytical analysis of xenon effects is discussed in detail in Section VI D of this report. It shows that the anticipated maximum rate of xenon buildup results in a reactivity decrease of approximately 0.1% an hour. Even with the rods inserted only one inch into the core, the six-rod bank has a nuclear worth of 0.1% per inch. Since normal rod speed is two inches per minute, there would be no difficulty in with-

drawing the rods at a rate fast enough to overcome maximum xenon buildup.

G. INSTRUMENTATION RECALIBRATION DATA

During the plant performance studies, calorimetric calibrations of the nuclear power instruments were performed on September 15, 18 and 29, 1962. Agreement between the calorimetric calibration and the nuclear power instrument readings was within 2% for all three dates. Since the power range chambers have been relocated on various occasions (due to operational or experimental requirements), no comparison with initial calibration can be made. The last nuclear instrument calibration was performed on July 14, 1962, when the instruments were matched to a calorimetric power of 48.7% full power. As noted above, agreement between power and instrument readings has remained constant since July 14, 1962.

H. RADIATION MONITOR DETECTION LIMITS AND ALARM POINTS

The detection limits and alarm settings of the radiation monitors are presented below:

1. Detection Limits (identical to meter range)

- (1) Area monitors (direct radiation)--0.10 mr/hr to 10 r/hr
- (2) Fluid monitors--10 to 10^6 cpm (each 10 cpm above background = 2×10^{-7} microcuries/cm³).
- (3) Off gas monitor--10 to 10^6 cpm (each 10 cpm above background = 4.3×10^{-8} microcuries/cm³).

2. Alarm Points

- (1) Area monitors--10 mr/hr (except for area monitor No. 1 over steam generator, which is set at 30 mr/hr).
- (2) Fluid monitors-- 5×10^3 cps (background approximately 1×10^3 cpm).
- (3) Off gas monitor-- 4×10^4 cpm (background approximately 1×10^4 cpm).

I. INSTRUMENTATION ELECTRICAL POWER REQUIREMENTS

Instrumentation power requirement data is tabulated below. The instrumentation is served by three main lines which carried the following measured loads during reactor operation:

Phase A	0.48 kva
Phase B	0.49 kva
Phase C	0.63 kva
Total	1.60 kva

Seven individual channels are fed from these three main sources and their measured full load requirements are:

<u>Vital Bus Circuit No.</u>	<u>Channel</u>	<u>kva</u>
1	Radiation monitoring	0.14
2	Process instrumentation	0.46
3	Nuclear instrumentation	0.39
4	Process instrumentation	0.33
5	Not used	--
6	Annunciator and scan system	0.20*
7	Communications system	0.08
	Total	1.60

* An additional 0.1 kva is drawn when the circuit is being tested.

VI. REACTOR CHARACTERISTICS

Data pertaining to the core and control rod characteristics is presented in this section. The following topics are discussed:

- (1) Prediction of core life based on actual operating data.
- (2) Variation of control rod position with life.
- (3) Control rod worth.
- (4) Temperature coefficient of reactivity.
- (5) Xenon and high cross section fission product reactivity effects.
- (6) Reactor shutdown margin.
- (7) Control rod speeds.
- (8) Pressure coefficient of reactivity.

A. ESTIMATED CORE LIFE

The estimated life of the PM-1 core was recalculated, using the following experimental data from the as-built core:

- (1) Initial core cold reactivity.
- (2) Cold control rod worth.
- (3) Six-rod bank position at beginning of life--cold.
- (4) Six-rod bank position at beginning of life--hot.

Previous estimates of core life were based on nuclear measurements of an experimental mockup of the core. Manufacturing tolerances resulted in a slightly less reactive core (approximately 0.5%) and, hence, a decrease in estimated core life. The current calculations predict a core life of 16.6 megawatt-years. Using the same correction applied previously and reported in Ref. 9, a lifetime of 17.4 megawatt-years is predicted, in comparison to the previous estimate of 18.2 megawatt-years. The correction factor is based on a comparison of the actual life of the SM-1 core with predictions using essentially identical analytical techniques. However, there are uncertainties as to the exact composition of the SM-1 core stainless steel; therefore, this correction factor might not be completely applicable.

A comparison of final design and "as-built" fuel element and lumped poison rod concentration is presented in Table 36.

TABLE 36
Comparison of the PM-1 Final Design and
As-Built Core Composition

	<u>Final Design</u>	<u>As Built</u>
1. Fuel Element Data		
Type	Tubular, cermet	
Number	741	
Length, overall (in.)	33.25	33.254
Pitch, triangular (in.)	0.665	
Diameter, outside (in.)	0.506	0.507
Diameter, inside (in.)	0.416	0.417
Matrix thickness (in.)	0.030	0.0284
Matrix composition		
UO ₂ (wt %)	28.17	29.63
304 L SS (wt %)	71.83	70.37
Clad thickness (in.)	0.0075	~0.0083
Loading, U-235/tube, nominal (gm)	40.13	40.43
Total U-235 (kg)	29.737	29.959
2. Burnable Poison Element Data		
Type	Rod, solid	
Number		
Full length	72	
2/3 length	18	
Pitch, triangular (in.)	0.665	
Diameter, (in.)	0.496	
Composition		
Natural boron (wt %)	0.275	0.2794
304 SS (wt %)	0.7206	0.7206
Loading, B-10/full LPR nominal (gm)	0.3829	0.3764
Total B-10 (gm)	32.161	31.572

The axial dead-end region compositions were obtained from "as-built" drawings. These concentrations are given in Table 37.

TABLE 37
PM-1 Dead-End Material Concentrations

	<u>Upper Dead End</u>	<u>Lower Dead End</u>
Thickness (in.)	1.75	1.625
Volume fraction of SS	0.2816	0.4077
Volume fraction of water	0.7184	0.5923

The re-estimate of PM-1 core life required three individual calculations, namely:

- (1) The determination of control rod worth as a function of six-rod bank position, so that control rod effects could be included in the burnup calculations.
- (2) Calculation of initial reactivity and comparison with the measured value to determine a bias to be applied to the non-uniform burnup calculations.
- (3) Performance of nonuniform burnup calculations to determine core life.

These three individual calculations are discussed briefly below; more detail is provided in Appendix D.

1. Control Rod Worth

To consider the effect of control rod position throughout life in the core life estimate, it was first necessary to calculate control rod worth as a function of bank position. Two-dimensional CURE calculations were performed to obtain three-group, homogenized cross sections by flux weighting techniques with Blackness Theory absorber constants. The three-group constants were then utilized in a one-dimensional diffusion code (window-shade model) to determine the hot reactivity worth of the six-rod bank as function of position. The procedure was repeated to determine cold rod worth. In both cases, actual beginning-of-life six-rod bank positions were utilized in the window-shade model. The results are presented in Fig. 39 and show rod worth as a function of the six-rod bank position.

2. Initial Core Reactivity

The initial reactivity of the PM-1 core was calculated by a one-dimensional diffusion code. An S_6 approximation to the transport equation was used to obtain cell corrections of the core components. The actual stainless steel composition utilized in the PM-1 core and the latest data pertaining to boron cross sections were utilized in the calculations. The results, which were normalized to agree with the experimental value of core reactivity, showed the hot and cold reactivities to be 8.58% and 12.5%, respectively.

3. Lifetime Calculations

The re-estimate of the PM-1 lifetime was made, utilizing nonuniform burnup calculations and considering the effect of six-rod bank position throughout life. The results, showing the predicted control rod bank position as a function of days of full-power operation, are presented in Fig. 40. The actual beginning-of-life rod positions without xenon and the bank position as of November 30, 1962 are shown.

B. VARIATION IN CONTROL ROD BANK POSITION WITH CORE LIFE

The predicted variation of control rod position as a function of core life is illustrated in Fig. 40. Also shown are actual hot (463° F) bank positions as follows:

Beginning of life	12.41 inches withdrawn
November 30, 1962	15.68 inches withdrawn

C. CONTROL ROD WORTH

The nuclear data obtainable at the PM-1 site was limited to those configurations of six control rods which resulted in a k_{eff} of approximately unity in a relatively fresh core. In addition, the inability to maintain reactor operating temperature at zero reactor power (see Section V.F) precluded the possibility of getting meaningful data at that temperature. Despite these limitations, sufficient data was obtained at the site to verify design predictions. For the sake of completeness, some data is also presented from critical experiments performed on a nuclear mockup of the PM-1 during the core design phase. This mockup was designated PMZ-Core 27 in Ref. 10 and will be referred to by that designation in this report. All data from PMZ-Core 27 reported herein has been corrected to correspond to the effective delay fraction of the PM-1 core.

These topics are considered individually:

- (1) Worth of individual rods.
- (2) Worth of 2-, 3-, 4-, 5- and 6-rod banks.

1. Worth of Individual Rods

The critical, 5-rod bank height, with Rod 6 fully inserted and fully withdrawn, was measured at the PM-1 reactor. It was found that with Rod 6 at 30.74 inches withdrawn, the critical bank height was 5.31 inches; and, with Rod 6 inserted, the critical bank height was 9.55 inches. Utilizing the data on 5-rod worth from Fig. 41, the approximate worth of Rod 6 was determined to be 2.69%.

The critical five-rod bank position with each rod fully withdrawn was also determined at the PM-1 site. By comparing the critical bank height with that which occurred with Rod 6 withdrawn and using the five-rod bank worth from Fig. 41, the approximate worth of the other five rods was determined. This data is presented in Table 38.

TABLE 38
Individual Rod Worth

<u>Rod Number</u>	<u>Critical Bank Position of Other Five Rods (inches withdrawn)</u>	<u>Correction Worth of Rod No. 6 (% reactivity)</u>	<u>Approximate Rod Worth (% reactivity)</u>
6	5.31	0.0	2.69
5	5.35	~0.0	2.69
4	5.12	+0.06	2.75
3	4.96	+0.12	2.81
2	4.53	+0.28	2.97
1	5.04	+0.09	2.78

The worth of the five-rod bank, presented in Fig. 41, was determined in PMZ-Core 27. Rod worth is, of course, dependent on the distribution of other poison material in the core. The data shown in Fig. 41 was obtained by poisoning out the excess core reactivity with a relatively uniform distribution of boron-polyethylene strips. Therefore, its value for other configurations is only approximate.

During the PM-1 final design phase, the worth of a typical Y-rod was determined in PMZ-Core 27 with the other rods withdrawn and the core poisoned out uniformly. This data yielded an approximate rod worth of 2.4%. The difference between this value and those shown in Table 38 is probably representative of the effect of different methods utilized to poison out the core reactivity.

2. Worth of Rod Banks

Data concerning the worth of 2-, 3-, 4- and 5-rod banks was obtained at a temperature of 68° F for PMZ-Core 27. This data is presented in Fig. 41. It should be noted that the two-rod bank data shown in Fig. 41 pertains to the situation where the other four rods are fully inserted. Thus, the value of 8.5% $\Delta \rho$, with two rods fully withdrawn, represents the worth of the four fully inserted rods.

The three-rod bank data, shown in Fig. 41, represents a configuration where alternate rods are being moved as a unit. That is, the bank does not contain any two adjacent rods. On the other hand, the four-rod bank data pertains to the case where two adjacent rods are fully withdrawn. For some of these configurations, the total worth of the three-rod bank exceeds that of the four-rod bank. Actually, the data in Fig. 41 represents the reactivity of the unpoisoned core regions. The configurations of the unpoisoned regions of the core are a function of both bank configuration and location.

A summary of the total worth of various rod configurations is presented in Table 39.

TABLE 39
PM-1 Total Bank Worth

<u>Number of Rods in Bank</u>	<u>Location of Bank</u>	<u>Configuration of Other Rods</u>	<u>Worth (% $\Delta \rho$)</u>
2	14.75 inches withdrawn	All in	4.3
3	Fully inserted	All out	8.8
4	Fully inserted	All out	8.46
5	Fully inserted	All out	13.06
6	8.41 inches withdrawn	--	12.55

The worth of the 6-rod bank was measured at the PM-1 site at moderate temperatures and agrees quite well with the design prediction. The predicted control rod worth data shown in Fig. 39 was replotted in terms of rod worth per inch as shown in Fig. 42. Along with the predicted worth, Fig. 42 shows the PM-1 and the PMZ-Core 27 data.

The control rod poison material is europium, whose daughter products are also rare earths. Nuclear calculations have shown that the worth of the control rods will change only negligibly over the life of the core. Therefore, core lifetime calculations assumed no change in rod worth due to control material burnup. The data from the early stages of core life (the only data presently available) shows that actual bank position is in good agreement with that predicted (see Fig. 40). This is an indication that control rod worth has not changed with control material burnup.

D. TEMPERATURE COEFFICIENT OF REACTIVITY

Temperature coefficient of reactivity at low and moderate temperatures was measured extensively on PMZ-Core 27. It was planned to measure the temperature coefficient of reactivity at operating temperature at the PM-1 site. However, the same difficulties described above for xenon buildup measurements hindered temperature coefficient and worth measurements. That is, the primary loop temperature could not be maintained at 460° F without reactor heat. To use reactor heat to maintain loop temperature requires the reactor to be at some power level, which violates the zero-power prerequisite of the test.

It was also observed that, immediately after the reactor power level was lowered from the power range (at 463° F), the reactor temperature dropped very rapidly to below the 400° F mark. This temperature decrease was too rapid to allow temperature coefficient measurements to be made. Inability to maintain even a near constant temperature at 463° F made it impossible to make accurate measurements at that temperature. At 366° F, a valid temperature coefficient of reactivity was measured and found to be $-1.96 \times 10^{-4} \Delta \rho / ^\circ\text{F}$. At 160° F, the temperature was measured at $-0.92 \times 10^{-4} \Delta \rho / ^\circ\text{F}$. The beginning-of-life predictions at 68° and 463° F were -0.675×10^{-4} and $2.83 \times 10^{-4} \Delta \rho / ^\circ\text{F}$, respectively. Thus, the limited data obtained at the site tends to confirm the design predictions.

Experimental data, at cold and moderate temperatures, obtained at the Martin Company Critical Facility for the prototype PM-1 core is included in this report for completeness. Reactivity as a function

of temperature and six-rod bank position was obtained for the PMZ-Core 27 over a range of 20° to 70° C. This data is presented in Fig. 43.

The curves in Fig. 43 were differentiated and temperature coefficients were obtained. This is summarized in Table 40.

TABLE 40
Temperature Coefficient of Reactivity

<u>Six-Rod Bank Position (in. withdrawn)</u>	<u>Temperature (°C)</u>	<u>Temperature Coefficient (% $\Delta \rho$ / °C)</u>
8.95	25	0.0075
	70	0.0205
20.25	25	0.0054
	70	0.0119
31.00	25	0.0022
	60	0.0081

E. XENON AND HIGH CROSS-SECTION FISSION PRODUCTS

As discussed in Section III.G, it was not possible to obtain definitive data as to xenon effects after shutdown and upon restarting. In order to keep the reactor at temperature, it was found necessary to operate at approximately 2% power. This resulted in enough xenon burnup that no buildup was noted. This is not too surprising, since the predicted change in core reactivity due to xenon buildup after shutdown is only 0.3%.

Analytical predictions of xenon and high cross-section fission product effects are presented below. The following topics are discussed individually:

- (1) Initial buildup of xenon-135.
- (2) Xe-135 buildup after shutdown at equilibrium xenon.

- (3) Xe-135 effects after startup at time of maximum xenon after shutdown.
- (4) Equilibrium xenon and high cross-section fission products effect as a function of core lifetime.

Details as to the calculational methods used are provided in Appendix D.

1. Initial Buildup of Xenon

Buildup of xenon in the clean PM-1 core was calculated at several times during life. The xenon concentration was determined for each core radial region as a function of time. A one-dimensional diffusion calculation was then performed to determine the effect of xenon on core reactivity. Axial effects were treated by using an axial windowshade model.

The effect of xenon buildup on core reactivity, in a clean core, is given in Table 41 and Fig. 44.

TABLE 41
Effect of Xe-135 Buildup After Initial Startup

<u>Time (hr)</u>	<u>% $\Delta \rho$</u>
0	-0.0
5	-0.28
10	-0.71
20	-1.38
40	-1.86
Equilibrium xenon	-1.94

2. Xenon Buildup After Shutdown

Buildup of Xe-135 from equilibrium xenon conditions, after a shutdown, was determined analytically. The results are shown in Table 42 and Fig. 44.

TABLE 42
Effect of Xenon Buildup After Shutdown

<u>Time (hr)</u>	<u>% $\Delta \rho$</u>
0	0.0
1	-0.12
3	-0.26
5	-0.30
10	-0.14
20	+0.54

3. Xenon Reactivity Effects Upon Restarting

The effect on core reactivity of restarting the reactor at time of peak xenon was determined analytically. The results are shown in Table 43 and Fig. 45.

TABLE 43
Xenon Reactivity Effects Upon Restarting

<u>Time (hr)</u>	<u>% $\Delta \rho$</u>
0	0.0
5	-0.43
20	-0.45
40	-0.33
Equilibrium xenon	-0.30

4. Equilibrium Xenon and High Cross-Section Fission Product Effects

The effects of equilibrium xenon and high cross-section fission products on core reactivity were calculated. The fission products considered were:

- (1) Cadmium-133.
- (2) Samarium-149.
- (3) Samarium-151.
- (4) Europium-155.
- (5) Gadolinium-155.
- (6) Gadolinium-157.

The material concentrations at several time steps were obtained directly from the burnup calculations and evaluated in a one-dimensional diffusion calculation. Axial nonuniform effects were determined from a windowshade model. The results obtained are presented in Table 44.

TABLE 44
Xenon and High Cross-Section
Fission Product Worth

<u>Time</u> <u>(days at 9.37 Mw(t))</u>	<u>Equilibrium</u> <u>Xenon Worth</u> <u>(% $\Delta\rho$)</u>	<u>Fission</u> <u>Product Worth</u> <u>(% $\Delta\rho$)</u>
0	-1.94	--
200	-2.05	-0.85
400	-2.15	-0.85
600	-2.25	-0.86

F. SHUTDOWN MARGIN

The PM-1 core can be shut down cold throughout life with no difficulty, when using six control rods. In fact, the beginning-of-life shutdown margin is approximately 7%. Since the core reactivity decreases monotonically with life, the shutdown margin increases continuously.

The PM-1 core has two additional shutdown requirements. Throughout life, the control rods are to be adequate to accomplish a cold shutdown, even with one rod fully withdrawn or any two adjacent rods held

in their operating positions. The ability of the control system to meet these requirements was evaluated from experimental data whenever possible.

1. One Stuck Rod Condition

At the PM-1 site, each rod, in turn, was withdrawn to its full-out height and the remaining five rods were brought to the required bank position for criticality to prove that the core would not go critical with one rod stuck fully withdrawn. These tests were, however, performed at a temperature of approximately 105° F. To prove that the one-stuck-rod criteria was still met at 70° F, the highest worth rod (No. 2) test was repeated at 70° F. The five-rod bank had to be withdrawn to 3.82 inches to bring the reactor critical, therefore proving the one stuck rod requirement was met. This data is summarized in Table 45.

Figure 41 shows that the additional worth of a five-rod bank inserted from a position 3.82 inches withdrawn is 0.5% $\Delta\rho$. Extrapolating data for Fig. 43 from 21° C (70° F) to 4° C shows that core reactivity will increase approximately 0.17% $\Delta\rho$. Hence, the shutdown margin at 4° C, with the rod of greatest worth fully withdrawn, is estimated to be 0.33% $\Delta\rho$.

Since the core reactivity decreases monotonically throughout life, the minimum shutdown margin, with one rod fully withdrawn, occurs at the beginning of life.

TABLE 45
Cold Critical Bank Heights

<u>Configuration</u>	<u>Critical Height (in.)</u>	<u>Temperature (° F)</u>	<u>Pressure (psig)</u>
Rod 6 at 30.74 in. Rods 1,2,3,4 and 5 banked	5.31	104.6	95
Rod 5 at 30.74 in. Rods 1,2,3,4 and 6 banked	5.35	104.4	85
Rod 4 at 30.77 in. Rods 1,2,3,5 and 6 banked	5.12	105.7	85
Rod 3 at 30.64 in. Rods 1,2,4,5 and 6 banked	4.96	105.6	100

TABLE 45 (continued)

<u>Configuration</u>	<u>Critical Height (in.)</u>	<u>Temperature (°F)</u>	<u>Pressure (psig)</u>
Rod 2 at 30.79 in. Rods 1,3,4,5 and 6 banked	4.53	105.6	90
Rod 1 at 30.79 in. Rods 2,3,4,5 and 6 banked	5.04	104.9	~120
Rod 2 at 30.76 in. Rods 1,3,4,5 and 6 banked	3.82	70	92

2. Two Stuck Rod Conditions

The evaluation of the two rods stuck at operating position is more complex, since cold clean core reactivity and rod bank position under various conditions must be obtained. Experimental data obtained at the site shows that the core could be shut down at the beginning of life with two rods stuck in the operating condition. A combination of analytical studies and experimental data shows that the core will meet the two-rod shutdown criterion throughout life.

a. Analytical studies

Cold clean reactivities were determined by recalculating the non-uniform depletion case with cold contents and no xenon or equilibrium fission products. The results are shown in Fig. 46.

The six-rod bank positions under conditions of equilibrium xenon, equilibrium fission products and burnup were obtained from the xenon study (Fig. 44) and the lifetime studies (Fig. 40). Cold worth versus insertion for the two-rod bank is presented in Fig. 41. From these, the shutdown margin throughout life can be determined. The results are summarized in Table 46 and show that the two stuck rod shutdown criterion is met throughout life. The smallest margin occurs at the time of initial equilibrium xenon and other fission product buildup. Rods would have been withdrawn somewhat to overcome xenon effects. If, at that time, two rods got stuck and the reactor was shut down, the xenon would eventually decay and the core reactivity would be essentially that of the fresh core.

TABLE 46
Shutdown Margin for Two Rods Stuck
at Operating Position

<u>Condition</u>	<u>Time (days)</u>	<u>Six-Rod Bank (in. out)</u>	<u>Control Rod Worth at 68° F--Two Rods Fully Withdrawn (% $\Delta\rho$)</u>	<u>Cold Clean Reactivity (% ρ)</u>	<u>Shutdown Margin (% $\Delta\rho$)</u>
Clean	0	12.41	14.08	12.55	-1.53
Equilibrium xenon	~ 2	14.37	13.02	12.53	-0.49
Equilibrium	5 to 10	15.3	12.57	12.45	-0.12
xenon and	100	15.7	12.40	11.19	-1.21
high cross-	200	16.5	12.02	10.66	-1.36
section fis-	300	17.9	11.45	10.13	-1.32
sion products					

b. Experimental data

The critical bank position of Rods 1 and 2, with the other rods fully inserted, was determined at the PM-1 site. This corresponds to the situation where two rods are stuck and shutdown is accomplished with the other four rods. It was found that the two-rod critical bank position was 15.15 inches withdrawn, with the core temperature at 103.5° F. The original hot rod bank position was 12.41 inches withdrawn.

Rods 1 and 2 were selected for the experiment, since the preshipment wet critical tests indicated that this configuration would yield the minimum shutdown margin. Two adjacent rods stuck is by far the most severe condition, since this yields the largest unpoisoned core region. Rod 2 has the greatest rod worth, so it was selected for test. The worth of Rod 3 is slightly greater than that of Rod 1; however, the reactivity of the bundle in which Rod 1 is inserted is greater. Therefore, the most stringent condition occurs when Rods 1 and 2 are partially withdrawn.

The reactor did not start up and go immediately to full power, building up equilibrium xenon as assumed in the analytical model. Hence, there is no clearly defined rod position corresponding to initial equilibrium xenon buildup which could be compared to the bank height of 15.15 inches and corrected for temperature effects.

It should be pointed out that this time of minimum shutdown margin has passed without incident and that shutdown margin, for both the one and two stuck rod criteria, will continue to increase throughout life.

G. CONTROL ROD SPEEDS

Control rod speeds, during both insertion and extraction, were checked at the PM-1 site. The live core was utilized to determine both slow and fast insertions between approximately 8-1/2 inches and 3 inches out. The results are shown in Table 47. Fast withdrawal speeds were measured with the dummy core in order to evaluate for extended travel. Later observations with the live core indicated no significant difference in speed. Withdrawal speeds are presented in Table 48. The speeds shown in Tables 47 and 48 are applicable whether the control rods are moved individually or in a banked configuration.

The control system drive speeds measured in the plant are less than the original design specification. The rate of travel for each rod is determined by the length of step taken by the rod for each movement cycle--the number of movement cycles per minute being constant for all rods and fixed by a central timing oscillator. However, the length of step taken can, and does, vary between rods as a result of the interplay between slightly different coefficients of friction, magnetic field buildup and decay factors, and magnetic coil current settings.

As a result of these conditions, it would be possible to adjust the timing oscillator so that the fastest rod would travel at the design speeds, or that the average-speed rod was at the design speeds. There will, of course, still be a variation between the individual rods. The only reason to increase the present rod speeds to the design values would be to make faster adjustments of reactivity during such maneuvers as xenon burnout. The present rod drive speeds are more than adequate for these maneuvers and no justification exists for increasing the drive speeds.

Control rod scram speeds, measured on three different occasions, are shown in Table 49. The data is for a rod travel of 12 inches and includes the 82-millisecond delay from the time the scram relay is tripped until rod motion is initiated.

The data in Table 49 indicates that there was an average decrease in scram speed, between the first and second test, of 37 milliseconds. It is believed that this increase represents differences in experimental techniques rather than an actual variation in control rod scram speed.

In determining the scram speeds, three possible inaccuracies occur, each of which could account for all, or part, of the 37-millisecond variation between the first and second run. These possible uncertainties are:

- (1) The design of the control rod actuator mechanism is such that, during testing, the position indicator is moving at the rate of approximately 0.5 inch per second at the time the rod is scrammed. Therefore, the distance the rod travels is not known exactly.
- (2) The scram time can be determined either by measuring the length of the recorder trace and dividing by the nominal value of recorder speed or by counting the number of cycles of the standard 60-cps frequency also recorded on the trace. The former method was used for the first test; the latter method, for the two subsequent tests.
- (3) The passage of the control rod position slug through the position indicator is determined by a null point on the recorder trace. The exact location of this null point on the trace cannot be determined within a distance corresponding to two cycles or 33 milliseconds.

TABLE 47
Rod Insertion Speeds

<u>Rod Number</u>	<u>Slow Insertion Rate (in./min)</u>	<u>Fast Insertion Rate (in./min)</u>
1	1.93	4.87
2	2.00	4.93
3	1.92	4.87
4	2.27	5.44
5	2.17	5.47
6	2.50	5.97

TABLE 48
Rod Extraction Speeds

<u>Rod Number</u>	<u>Fast Extraction Rate (in./min)</u>
1	3.84
2	4.46
3	3.79
4	4.61
5	4.21
6	4.98

TABLE 49
Control Rod Scram Speeds

<u>Rod Number</u>	<u>Time (millisecond)</u>		
	<u>Test of April 8</u>	<u>Test of July 22</u>	<u>Test of September 9</u>
1	270	298	300
2	250	281	300
3	200	298	300
4	260	298	294
5	250	269	276
6	260	265	267

H. PRESSURE COEFFICIENT OF REACTIVITY

Measurements of the pressure coefficient were made at the PM-1 site. However, the magnitude of the experimental error exceeded the magnitude of the coefficient. Various measurements indicated that the pressure coefficient was about $\pm 10^{-6} \Delta \rho$ /psi. For all practical purposes, this coefficient is negligible. Analytical calculations also predicted a negligible pressure coefficient.

VII. PRIMARY SYSTEM CHARACTERISTICS

This section is concerned with:

- (1) Primary coolant water chemistry.
- (2) Primary coolant chemical consumption.
- (3) Primary coolant pump power.
- (4) Purification system resin life.
- (5) Primary system conditions following a scram from full power.
- (6) Shield water system performance.
- (7) Shield water chemistry.
- (8) Shield water demineralizer resin life.
- (9) Steam generator package heat load.

A. PRIMARY COOLANT PROPERTIES

Data pertaining to the primary coolant activity, total dissolved solids, radioactive constituents, conductivity and pH was compiled.

The coolant activity during periods of plant operation is shown in Table 50. Both demineralizer influent and effluent were monitored. The fluid entering the demineralizer is representative of the loop activity and varies over a considerable range. As expected, the effluent activity is not only lower but, for the most part, falls in a narrow range around 10^{-4} microcuries/cm³. This constant effluent characteristic is typical of demineralizer performance.

The radioactive constituents and total dissolved solids of the primary coolant were analyzed from samples taken during the 400-hr run and sent from the site to Baltimore. This data is presented in Table 9, above.

The activity of the relatively long half-life constituents shown in Table 9 was adjusted to yield the correct values at the time the sample was drawn. However, the short-life radioactive constituents decay to undetectable levels between the time the sample is obtained at the PM-1

TABLE 50
Primary Coolant Activity ($\mu\text{c}/\text{cc}$)

<u>Purification Demineralizer</u>		
<u>Date</u>	<u>Inlet</u>	<u>Outlet</u>
5-23	$(1.017 \pm 0.035) \times 10^{-3}$	$(1.263 \pm 0.141) \times 10^{-4}$
5-24	$(1.125 \pm 0.0125) \times 10^{-2}$	$(2.94 \pm 0.196) \times 10^{-4}$
5-25	$(2.5 \pm 0.018) \times 10^{-3}$	$(3.9 \pm 0.694) \times 10^{-4}$
5-27	$(4.86 \pm 0.067) \times 10^{-2}$	$(1.05 \pm 0.03) \times 10^{-2}$
5-28	$(8.79 \pm 0.00033) \times 10^{-1}$	$(9.94 \pm 1.66) \times 10^{-4}$
7-24	$(2.15 \pm 0.04) \times 10^{-2}$	$(2.1 \pm 0.4) \times 10^{-4}$
7-25	$(1.01 \pm 0.037) \times 10^{-2}$	$(2.16 \pm 0.52) \times 10^{-4}$
7-26	$(2.20 \pm 0.03) \times 10^{-2}$	$(1.94 \pm 0.31) \times 10^{-4}$
7-27	$(3.30 \pm 0.145) \times 10^{-3}$	$(9.66 \pm 2.5) \times 10^{-5}$
7-28	$(1.15 \pm 0.03) \times 10^{-2}$	$(1.6 \pm 0.14) \times 10^{-4}$
7-29	$(1.84 \pm 0.123) \times 10^{-3}$	$(7.69 \pm 1.015) \times 10^{-5}$
7-30	$(1.31 \pm 0.02) \times 10^{-2}$	$(1.16 \pm 0.26) \times 10^{-4}$
7-31	$(1.47 \pm 0.02) \times 10^{-2}$	$(7.66 \pm 2.3) \times 10^{-5}$
8-1	$(1.13 \pm 0.036) \times 10^{-2}$	$(1.48 \pm 0.458) \times 10^{-4}$
8-2	$(6.73 \pm 0.16) \times 10^{-3}$	$(4.62 \pm 0.47) \times 10^{-4}$
8-3	$(3.12 \pm 0.199) \times 10^{-3}$	$(2.79 \pm 5.81) \times 10^{-4}$
8-4	$(2.35 \pm 0.095) \times 10^{-5}$	$(5.08 \pm 2.2) \times 10^{-5}$
8-6	$(3.8 \pm 0.178) \times 10^{-3}$	$(5.39 \pm 3.71) \times 10^{-5}$
8-7	$(5.26 \pm 0.266) \times 10^{-2}$	$(1.76 \pm 0.526) \times 10^{-4}$
9-16	$(9.76 \pm 0.06) \times 10^{-2}$	Not recorded

TABLE 50 (continued)

<u>Purification Demineralizer</u>		
<u>Date</u>	<u>Inlet</u>	<u>Outlet</u>
9-19	Not recorded	$(5.77 \pm 0.97) \times 10^{-4}$
9-20	$(1.44 \pm 0.02) \times 10^{-2}$	$(2.26 \pm 0.32) \times 10^{-4}$
9-21	$(3.72 \pm 0.04) \times 10^{-2}$	$(7.02 \pm 0.53) \times 10^{-4}$
9-22	$(3.5 \pm 0.11) \times 10^{-3}$	Not recorded
9-24	$(3.08 \pm 0.03) \times 10^{-2}$	$(9.20 \pm 0.06) \times 10^{-4}$
9-25	Not recorded	$(2.29 \pm 0.35) \times 10^{-4}$
9-26	$(1.92 \pm 0.03) \times 10^{-2}$	$(4.02 \pm 0.42) \times 10^{-4}$
9-27	$(3.62 \pm 0.06) \times 10^{-2}$	$(6.6 \pm 0.87) \times 10^{-4}$
9-28	$(3.63 \pm 0.068) \times 10^{-2}$	$(7.99 \pm 1.02) \times 10^{-4}$
9-29	$(3.72 \pm 0.036) \times 10^{-2}$	$(1.18 \pm 0.07) \times 10^{-3}$
9-30	$(4.41 \pm 0.04) \times 10^{-2}$	$(9.97 \pm 0.66) \times 10^{-4}$
10-1	$(3.44 \pm 0.04) \times 10^{-2}$	$(7.63 \pm 0.55) \times 10^{-4}$

site and the time at which detailed analysis is completed in Baltimore. The samples analyzed represent PM-1 conditions during the period from September 24 to September 27. The difference in activity of a factor about 10^2 between that observed at the site and that found in Baltimore represents this relatively short-lived activity.

The following conclusions were made as a result of the analysis performed at Baltimore:

- (1) The total radioactivity level in the primary coolant sample is of the order of magnitude that might be expected from a reactor of this type, operated for this period of time and allowed about one month's decay.

- (2) Over 90% of the radioactivity is contributed by corrosion products from the structural material. The predominant nuclide, about 80% of the radioactivity, is Cobalt-58 which is formed by an n, p reaction of fast neutrons with Nickel-58. Manganese-54 is also formed by n, p reaction on Iron-54. The Iron-59 is formed by thermal neutron activation, $\text{Fe-58} (n, \gamma) \text{Fe-59}$. Chromium-51 is undoubtedly present, but in too low a concentration to detect. These data are in conformity with the findings of Ref. 15.
- (3) Less than 10% of the radioactivity is contributed by fission products, the predominant ones being Barium-140 and Strontium-89. Shorter-lived fission products have decayed, and longer-lived ones have not yet built up in sufficient concentration to be observed during this relatively short period of reactor operation. This data agrees with Ref. 16.
- (4) The concentration of the fission product activity may be accounted for by the surface uranium contamination on the fuel elements. There is no indication of any fuel defect of sufficient magnitude to contribute fission products to the coolant.
- (5) The absence of any detectable radioactivity in samples of main steam condensate, feedwater and radioactive waste disposal system returns indicates the facility is being kept clean of any cross contamination. The relative levels of contamination in the various samples are also generally in the proper order, indicating control of cross contamination.

The pH and resistivity of the coolant entering and leaving the purification system demineralizer are presented in Figs. 47 through 49. Three periods are covered: Fig. 47 presents data on operation for the end of May, Fig. 48 presents data for July and August, and Fig. 49 presents data for September.

In general, the primary coolant pH was held within the desired range of 9.5 to 10.5. However, the conductance of the primary coolant water exceeded the specified maximum value of 2×10^{-6} /ohm-cm; this value was based on the conductance resulting from dissolving small quantities of corrosion products in the water. The factors which increased the conductance of the primary coolant above the specified value were investigated. It was found that the increase was due almost entirely to the ammonia utilized to control pH.

Experiments were conducted with primary coolant water, shield water and high purity water (resistivity greater than 10^6 ohm-cm) to

determine the effect of added ammonia ions upon the pH and conductivity of the water. All the samples indicated that the resistivity of the water decreased rapidly as ammonia was added. In the case of the high resistivity water, the variation of conductance with pH was:

<u>pH</u>	<u>Conductance</u> <u>(10^{-6}/ohm-cm)</u>
7.5	1.5
8.0	2.0
8.5	3.3
9.0	5.5
9.1	6.1
9.2	6.8
9.3	7.8
9.4	9.0
9.5	10.7
9.6	13.0
9.7	15.7
9.8	18.3
9.9	21.9
10.0	27.5

To determine the net resistivity of the coolant due to dissolved constituents other than ammonia, the following formula is to be applied:

$$R_{\text{net}} = \frac{1}{\left(\frac{1}{R_{\text{meas}}} - C \right)}$$

where

R_{meas} = measured resistance of the water from Figs. 47 through 49 (ohm-cm)

C = conductance due to the addition of ammonia, tabulated above (1/ohm-cm)

R_{net} = resistivity of the water if no ammonia had been added (ohm-cm).

The use of the above formula is restricted to cases where $1/R_{\text{meas}} > C$. Values of $C > 1/R_{\text{meas}}$ indicate that R_{net} equals or exceeds 10^6 ohm-cm, the resistivity of the control sample.

B. PRIMARY COOLANT CHEMICAL CONSUMPTION

Information pertaining to primary coolant chemical utilization was recorded and compiled for the periods of operation between July 23 through August 8 and September 12 through October 2. This data is presented in Table 51.

TABLE 51
Primary System Water Treatment

Date	Hydrogen (H ₂) (ft ³ at STP)	Hydrazine (ml)	Chemical Treatment
			Ammonium Hydroxide (NH ₄ OH) (ml)
7-23	--	--	40
7-24	1.5	--	--
7-25	--	--	17.7
7-26	--	--	40
7-27	--	--	20
7-28	--	--	40
7-29	--	--	40
8-1	2.0	--	--
8-2	1.5	--	--
8-4	--	--	20
9-12	--	--	80
9-13	2.5	200	80
9-15	--	--	40

TABLE 51 (continued)

Date	Hydrogen (H ₂) (ft ³ at STP)	Chemical Treatment	
		Hydrazine (ml)	Ammonium Hydroxide (NH ₄ OH) (ml)
9-21	1.5	--	--
Total (37 days of operation)	9.0	200	417.7

Since the data presented in Table 51 was obtained toward the end of the test period, it is probably representative of future chemical requirements. On a monthly basis, the following quantities were consumed:

Hydrogen	7.3 ft ³ /month
Hydrazine	162 ml/month
Ammonium hydroxide	338 ml/month

C. PRIMARY COOLANT PUMP POWER

The power consumed by the primary coolant pump was recorded as a function of coolant water temperature on three occasions, namely:

July 23, 1962

August 8, 1962

September 12, 1962.

This data is presented in Fig. 50.

D. PURIFICATION SYSTEM RESIN LIFE

The primary coolant purification system demineralizer was originally charged with resin on February 24, 1962 and was recharged on October 3,

1962. This represents a resin life of slightly over 7 months, compared to a predicted life of 12 months. It is believed that this shorter life resulted from the large amounts of makeup water utilized during the plant test period and that future resin charges will exhibit a longer life. However, no re-estimate of resin life can be made from the limited data currently available.

It should be noted that replacement of the resin in the PM-1 is relatively simple. The spent resin is sluiced to the waste storage tank, the lines are flushed, and fresh resin is sluiced into the demineralizer. The entire operation for both the primary coolant and shield water demineralizers required approximately 10 hours. No handling of the spent (radioactive) resin is involved and there are no spent resin shipping casks required.

E. PRIMARY SYSTEM CONDITIONS DURING COOLDOWN

Primary system conditions during a cooldown from full power were compiled and evaluated. The conditions immediately after a scram from full power were discussed in Section III, D and illustrated in Run XXIII of Appendix A.

The data relating to cooldown to low temperature on October 2 is presented in this section. The test schedule was such that the cooldown followed a normal shutdown rather than a scram; however, this should not appreciably affect the cooldown rate.

Figure 51 shows the rate of reactor coolant system temperature and steam generator pressure decrease as a function of time. The individual steps in the cooldown procedure are indicated in Fig. 51 and are listed below in more detail:

- (1) The secondary plant was secured in accordance with normal shutdown procedures.
- (2) During the first hour and 45 minutes, cooldown was adequately maintained by the purification system.
- (3) Steam was then admitted to the main steam line and cooldown was maintained by steam leakage through the hogging jet.
- (4) The hogging jet was put into operation at a steam pressure of 150 psig and a vacuum was pulled on Condenser 1A air cooler section. Again, steam escaping through the hogging jet was adequate to maintain plant cooldown.

- (5) At a steam pressure of 40 psig, initial dumping of steam to Condenser 1A air cooler section was initiated. The de-superheater steam flow control valve was opened until tube temperatures reached 150° to 160° F. One fan was put on high speed and the louvers were opened. Steam flow was increased to maintain tube temperatures. Valve position at this time was approximately 1/16 open. As cooldown proceeded, the valve was opened further to maintain tube temperatures.
- (6) At a primary temperature of 250° F, the primary coolant pump was secured, and the decay heat system was put in operation.

NOTE: From 1330 to 1700 hours, the decay heat system was intermittently operated only to maintain loop temperature for other plant tests.

The operating procedure limits the maximum cooldown rate to 100 degrees per hour. This limitation is imposed by thermal stress considerations. Thermal predictions show that there would be no difficulty in maintaining this cooldown rate when steam was bled to the condensers. Actual cooldown to 260 degrees took slightly less than 3 hours; consequently, the average rate was approximately 70 degrees per hour. The difference between the 100-degree-per-hour rate and the 70-degree-per-hour rate represents the steps taken by the operating crew to make sure that it would not exceed the maximum rate. Discussions with the operating crew indicate that cooldown rates much greater than 100 degrees per hour could be accomplished by bleeding steam to the air condensers.

At 250 degrees, the decay heat removal system was placed in operation, in accordance with the operating procedures. In 2 hours, the loop had cooled to 165 degrees. The design predictions were that the loop could be cooled from 260 to 204 degrees in 2 hours. During the next 6 hours, the loop was cooled from 165 to 118 degrees. Design predictions indicated that during this period the loop would be cooled from 205 to 157 degrees. Hence, it is evident that actual cooldown rate with the decay heat system exceeded design predictions.

Since this test of plant cooldown rate occurred relatively early in core life, the question arises as to the effect of end-of-life fission product decay heat. At the time the test was conducted, the total integrated core energy output was approximately 230 megawatt days. The plant had been operating at approximately 75% power for the 10 days immediately preceding the test. From design predictions, it

was concluded that the fission product decay heat under this condition would be about 60% of the end-of-life, full-power condition. During the initial stages of cooldown, afterheat amounts to approximately 50% of the total heat to be rejected. During the later stages, this increases to about 70%. Thus, the data presented is reasonably representative of conditions to be expected in the future. However, cooldown rate with the decay heat removal system will probably decrease somewhat but still be equal to or greater than the design predictions.

F. SHIELD WATER SYSTEM STEADY-STATE DATA

Steady-state performance data of the PM-1 shield water system is presented as a function of reactor power in the primary system heat balance in Section III, A of this report (see Fig. 7). The data reflects operation at an ambient temperature of 70° F, or less, with 3 of the 4 shield water coolers in operation. It was found that the system heat load was easily handled under this condition, provided the waste disposal system was inoperative. This agrees with the design predictions.

The variation in shield water coolant outlet temperature as a function of power, shown in Fig. 7, is not indicative of the heat transfer capacity of the system; rather, it is a measure of the difference in the individual louver controls of each cooler. During the course of the test, various units were placed in operation. Each controller has a slightly different dead band and/or temperature sensing device. Thus, a slight variation in actual temperature results.

The shield water system is not normally operated with the four coolers in service. However, if the plant is operating at power and the waste disposal system is in operation, four coolers are required to handle the heat load. It has been found that the shield water coolers can maintain the shield water at the design temperature of 125° F under these conditions, provided the ambient temperature does not exceed 60° F. Thus, the waste disposal system can be operated during the daytime in the winter, spring, fall and cool summer days. On warm summer days, the operation of the waste disposal system will be limited to the nighttime. This presents no problems, since the waste disposal sump tank provides storage capacity for approximately one month's waste accumulation, based on an average accumulation rate, during operation, of 55 to 60 gallons per day (see Section I, E).

G. SHIELD WATER SYSTEM COOLANT PROPERTIES

Data pertaining to the shield water activity, pH, conductivity, total dissolved solids and constituents was compiled. The coolant activity during periods of plant operation is shown in Table 52. Both demineralizer influent and effluent were monitored. The fluid entering the demineralizer is representative of the shield water tank activity and, for the most part, varies from 6×10^{-5} to $10^{-3} \mu\text{c}/\text{cm}^3$. The activity of the demineralizer effluent, like that of the primary coolant, varies over a much narrower range and has an activity of an order of magnitude less than that of the primary coolant.

TABLE 52
Shield Water Activity ($\mu\text{c}/\text{cm}^3$)

<u>Shield Water Demineralizer</u>		
<u>Date</u>	<u>Inlet</u>	<u>Outlet</u>
5-23	$(2.85 \pm 0.736) \times 10^{-5}$	$(1.4 \pm 0.56) \times 10^{-5}$
5-24	$(5.74 \pm 0.845) \times 10^{-5}$	$(7.76 \pm 4.9) \times 10^{-6}$
5-24	$(2.0 \pm 0.195) \times 10^{-4}$	$(2.52 \pm 1.22) \times 10^{-5}$
5-28	$(1.39 \pm 0.139) \times 10^{-4}$	$(2.32 \pm 0.205) \times 10^{-5}$
7-24	$(2.01 \pm 0.31) \times 10^{-4}$	Not recorded
7-25	$(2.4 \pm 0.61) \times 10^{-4}$	Not recorded
7-26	$(3.48 \pm 0.37) \times 10^{-4}$	Not recorded
7-27	$(3.12 \pm 0.40) \times 10^{-4}$	Not recorded
7-30	$(3.15 \pm 0.36) \times 10^{-4}$	Not recorded
7-31	$(3.18 \pm 0.37) \times 10^{-4}$	Not recorded
8-1	$(2.98 \pm 1.77) \times 10^{-4}$	Not recorded
8-3	$(2.92 \pm 0.644) \times 10^{-4}$	Not recorded

TABLE 52 (continued)

<u>Shield Water Demineralizer</u>		
<u>Date</u>	<u>Inlet</u>	<u>Outlet</u>
8-4	$(1.93 \pm 0.32) \times 10^{-4}$	Not recorded
8-6	$(2.11 \pm 0.55) \times 10^{-4}$	Not recorded
8-7	$(2.54 \pm 0.619) \times 10^{-4}$	Not recorded
9-11	Not recorded	$(1.23 \pm 0.4) \times 10^{-5}$
9-14	$(3.8 \pm 1.6) \times 10^{-5}$	$(0 \pm 1.3) \times 10^{-5}$
9-15	$(6.73 \pm 2.21) \times 10^{-5}$	$(1.59 \pm 1.77) \times 10^{-5}$
9-16	$(5.89 \pm 0.5) \times 10^{-4}$	$(2.39 \pm 1.86) \times 10^{-5}$
9-19	$(5.51 \pm 0.47) \times 10^{-4}$	$(1.9 \pm 0.91) \times 10^{-5}$
9-20	$(4.43 \pm 0.44) \times 10^{-4}$	$(3.2 \pm 1.9) \times 10^{-5}$
9-21	$(5.96 \pm 0.5) \times 10^{-4}$	$(6.14 \pm 2.07) \times 10^{-5}$
9-22	$(3.42 \pm 0.3) \times 10^{-4}$	$(1.42 \pm 1.07) \times 10^{-5}$
9-24	$(5.5 \pm 0.5) \times 10^{-4}$	$(5.49 \pm 2.1) \times 10^{-5}$
9-25	$(4.77 \pm 0.43) \times 10^{-4}$	$(3.70 \pm 1.92) \times 10^{-5}$
9-26	$(4.62 \pm 0.45) \times 10^{-4}$	$(3.80 \pm 1.8) \times 10^{-5}$
9-28	$(7.45 \pm 0.956) \times 10^{-4}$	$(1.14 \pm 0.49) \times 10^{-5}$
9-29	$(8.54 \pm 0.6) \times 10^{-4}$	$(2.19 \pm 0.21) \times 10^{-5}$
9-30	$(8.66 \pm 0.6) \times 10^{-4}$	$(1.51 \pm 0.29) \times 10^{-4}$
10-1	$(1.02 \pm 0.06) \times 10^{-3}$	$(1.13 \pm 0.26) \times 10^{-5}$

Data pertaining to the pH and resistivity of the coolant entering and leaving the shield water demineralizer is presented in Figs. 52 through 54. The same three periods of power operation for which primary coolant data was presented in Section A are again covered. Also, the correction to water resistivity discussed in that section applies.

The radioactive constituents and total dissolved solids of the shield water were analyzed from samples taken during the 400-hour run and sent from the site to Baltimore. This data is presented in Table 9. Again, this represents only the long-life radioactive constituents.

H. SHIELD WATER DEMINERALIZER LIFE

The shield water demineralizer was originally charged with resin on February 24, 1962 and was recharged on October 3, 1962. This represents a resin life of slightly over 7 months, compared to a predicted life of 12 months. It is believed that this shorter life resulted from the large amounts of makeup water utilized during the plant test period and that future resin charges will exhibit a longer life. However, no re-estimate of resin life can be made based on the currently available data.

I. PRIMARY SYSTEM HEAT LOSSES

Data was compiled and evaluated to determine:

- (1) The total primary system heat loss.
- (2) The steam generator package heat loss.

1. Total Primary System Heat Loss

During precritical testing at the PM-1 site, the primary system was tested at operating temperature and pressure. The system was brought up to operating conditions with an electric heater (in place of the core) and the primary coolant pump as energy sources. The shell side of the steam generator contained fluid; therefore, the total system heat losses were representative of those experienced under normal operating conditions.

It was found that a heater input of approximately 70 kw was required to maintain the primary system at 463° F with the primary coolant pump in operation and the purification system inoperative.

The net energy input supplied by the primary coolant pump at operating temperature was determined, from the data presented in Figs. 7 and 50, to be 25 kw. Thus, the total primary system heat loss is approximately 324,000 Btu/hr.

2. Steam Generator Package Heat Load

The heat load in the steam generator package was determined by measuring the air velocity in the duct to the steam generator tank and the temperature rise of this air. The air velocity was measured with a portable hot wire anemometer probe in an extension of the duct. The ambient temperature and the temperature in the top of the tank were measured. The heat load was found to be 66,400 Btu/hr with 2000 ft/min duct air velocity, ambient temperature of 48° F and tank temperature of 120° F.

VIII. SECONDARY SYSTEM CHARACTERISTICS

This section presents data concerning these items:

- (1) Steam generator blowdown.
- (2) Secondary system chemical consumption.
- (3) Secondary system water chemistry.
- (4) Condensate and feedwater pump cavitation.
- (5) Time required to get turbine generator on line.
- (6) Time required to get diesel generator on line.
- (7) Noise level in the secondary building.

A. STEAM GENERATOR BLOWDOWN

The desired concentration of chemicals and desired pH in the steam generator is defined by the PM-1 Operating Manual as:

Phosphates--125 to 175 ppm

Sulphates--greater than 10 ppm

pH--10.1 to 10.5

Data was compiled for three periods of plant operation during May, July-August and September. This is presented in Figs. 55 through 57 and shows that, in general, the desired concentrations and pH were maintained.

In all cases, the blowdown rate was fixed at 200 lb/hr.

B. SECONDARY SYSTEM CHEMICAL CONSUMPTION

Data pertaining to the secondary system chemical consumption is presented in Table 53 for operating periods during July, August and September. Since this period was the latter portion of the PM-1 test program and includes the 400-hour run, this data is probably representative of future chemical utilization.

TABLE 53
Secondary Water Treatment

<u>Date</u>	<u>Sulphite (NaSO₃) (grams)</u>	<u>Trisodium Phosphate (grams)</u>	<u>Disodium Phosphate (grams)</u>
7-23-62	100	1, 150	2, 020
7-24-62	150		
7-25-62	175	1, 150	2, 020
7-27-62	175		
7-29-62	175		
7-30-62	172		
8-1-62	129		
8-2-62	129	1, 150	2, 020
8-3-62	129	1, 332	2, 023
8-4-62	175		
8-6-62	150		
8-7-62	175		
8-8-62	175		
9-14-62	164	1, 437	2, 525
9-15-62	167		
9-16-62	187		
9-17-62	187		
9-18-62	187		
9-19-62	187	1, 150	2, 020
9-21-62	250	1, 130	1, 696
9-23-62	250	1, 130	1, 696
9-24-62	250	1, 130	1, 696
9-25-62	200		
9-26-62	250	1, 130	1, 696
9-29-62	300	1, 130	1, 696
9-30-62	500		
10-1-62	500		
Totals (34-day period)	5570	11, 889	19, 412

Table 53 shows the total quantities of chemical consumption. Based on this, the predicted monthly consumption would be:

Sulphite	10.8 lb/month
Trisodium phosphate	22.1 lb/month
Disodium phosphate	37.8 lb/month

C. SECONDARY SYSTEM COOLANT PROPERTIES

Data pertaining to the activity, pH, conductivity, total dissolved solids and radioactive constituents of the secondary system water was compiled.

The activity of the condensate storage tank, steam generator blow-down and deaerator liquid, and main steam is presented in Table 54. It can be seen that the levels are quite low.

Data on the pH and resistivity of the fluids in the main condenser, the main steam line and feedwater line is presented in Figs. 58 through 60 for periods of reactor operation from May to September.

Samples of the main steam flow and feedwater flow were sent to Baltimore for more detailed analysis. No long-term radioactive constituents were found in the fluids. The sensitivity of the instrumentation and techniques utilized at Baltimore is $2 \times 10^{-8} \mu\text{c}/\text{cm}^3$. No chlorides were detectable in either sample. Total solids in the main steam condensate and the feedwater samples were 25 and 21 ppm, respectively.

D. CONDENSATE AND FEEDWATER PUMP CAVITATION

Conditions at which cavitation in the condensate and feedwater pumps occurs are presented in Table 55. Cavitation does not occur at normal operating conditions, including transient conditions more severe than the design criteria.

E. TIME REQUIRED TO GET TURBINE GENERATOR ON LINE

Typical data of the time required to get the turbine generator on the line was tabulated from the operating log of September 18. One hour and 15 minutes elapsed from the time rolling of the turbine started to the time the turbine generator was paralleled with the site.

TABLE 54
Secondary System Activity ($\mu\text{c}/\text{cm}^3$)

<u>Date</u>	<u>Condensate Storage Tank</u>	<u>Steam Generator Blowdown</u>	<u>Deaerator</u>	<u>Main Steam</u>
5-25-62	$1.42 \pm 0.756 \times 10^{-7}$	$(7.19 \pm 7.19) \times 10^{-8}$	$(3.14 \pm 7.02) \times 10^{-8}$	$(6.51 \pm 7.19) \times 10^{-8}$
5-26-62		$(1.028 \pm 0.656) \times 10^{-7}$		
5-27-62		Less than background		
5-28-62		$(199 \pm 4.63) \times 10^{-8}$		
7-24-62		$(1.17 \pm 0.11) \times 10^{-7}$	$(2.14 \pm 5.1) \times 10^{-8}$	
7-25-62		$(6.16 \pm 7.19) \times 10^{-8}$		$(8.25 \pm 32.9) \times 10^{-9}$
7-26-62		$(1.03 \pm 2.05) \times 10^{-8}$		
7-27-62	$2.87 \pm 6.2 \times 10^{-8}$	$(4.72 \pm 6.4) \times 10^{-8}$	$< 5.75 \times 10^{-8}$	$< 5.54 \times 10^{-8}$
7-29-62		$(2.06 \pm 2.46) \times 10^{-8}$		
7-30-62		$(6.36 \pm 2.26) \times 10^{-8}$		$(2.81 \pm 0.47) \times 10^{-7}$
7-31-62		$(5.74 \pm 3.08) \times 10^{-8}$	$(1.23 \pm 2.06) \times 10^{-8}$	
8-2-62		$(1.33 \pm 9.55) \times 10^{-8}$		
8-7-62		$(1.03 \pm 0.93) \times 10^{-7}$		$(8.26 \pm 8.76) \times 10^{-8}$
9-15-62		$(9.49 \pm 6.5) \times 10^{-8}$	$< 6.1 \times 10^{-8}$	
9-16-62		$(1.38 \pm 5.9) \times 10^{-8}$		
9-18-62		$< 6.0 \times 10^{-8}$	$(0.97 \pm 6.38) \times 10^{-8}$	$< 6.2 \times 10^{-8}$
9-21-62		$(6.81 \pm 6.44) \times 10^{-8}$		
9-22-62		$(1.2 \pm 1.4) \times 10^{-8}$		

TABLE 54 (continued)

<u>Date</u>	<u>Condensate Storage Tank</u>	<u>Steam Generator Blowdown</u>	<u>Deaerator</u>	<u>Main Steam</u>
9-24-62		$(4.49 \pm 7.32) \times 10^{-8}$		
9-25-62		$(2.36 \pm 0.72) \times 10^{-7}$		
9-26-62		$(1.56 \pm 6.15) \times 10^{-8}$		
9-27-62		$(9.14 \pm 8.25) \times 10^{-8}$		
9-29-62		$(8.00 \pm 6.8) \times 10^{-8}$		
9-30-62		$< 0.8 \times 10^{-8}$		
10-1-62	$(2.00 \pm 0.67) \times 10^{-7}$			

TABLE 55
Condensate and Feedwater Pump Cavitation

<u>Condensate Pump</u>	<u>Conditions for Cavitation</u>
Hotwell level	4 inches* started cavitating 7 inches* violent cavitation 9 inches* lost flow
Hotwell temperature	151° F
Hotwell pressure	8-1/2 inches Hg Abs
Condensate flow	10,000 lb/hr
<u>Feedwater Pump</u>	
Deaerator level	15-3/4 inches* started cavitating
Deaerator temperature	236° F
Deaerator pressure	10.9 psig
Feedwater flow	10,000 lb/hr

*Height below normal operating level

F. TIME REQUIRED TO GET DIESEL GENERATOR ON LINE

Typical time required to place the diesel generator in operation was tabulated from the operating log of September 22nd. It required six minutes from the time the diesel generator was started until it was carrying the PM-1 auxiliary load. Two additional minutes were required for the diesel generator to be paralleled with the radar site. Other operations were being performed concurrently with the diesel generator startup.

G. NOISE LEVEL IN THE SECONDARY BUILDING

The noise level in the secondary building was recorded for two conditions, namely:

- (1) Diesel generator off, auxiliary boiler off, plant load 430 kw.
- (2) Diesel generator off, auxiliary boiler on, plant load 1250 kw.

The maximum noise level for the first condition was 100 decibels, which occurred one foot away from the generator. Maximum noise level for the second set of conditions was 104 decibels, and this occurred one foot away from the turbine generator reduction gear.

All readings were taken with a General Radio, Model 1555-A, Sound Meter, which was calibrated on July 31, 1962. All recorded data is presented in Table 56.

TABLE 56
Noise Level in Secondary Building

1. Test Conditions

Diesel generator off
Auxiliary boiler off
Plant load, 430 kw

<u>Locations of High Noise Levels</u>	<u>Decibels</u>
Control room console	72 to 75
Maintenance area	85
Counting room	78
Office area	80
Turbine generator--reduction gear 1-1/2 feet away	94
Turbine generator--generator 1 foot away	100
Turbine generator--generator 2 feet away	95

Maximum noise level--100 db one foot away from generator.
Source of maximum noise level--generator

2. Test Conditions

Diesel generator off
Auxiliary boiler on
Plant load, 1250 kw

<u>Locations of High Noise Levels</u>	<u>Decibels</u>
Control room console	77
Maintenance area	88
Counting room	82
Office area	80
Heat transfer area	90
Switchgear area	97
Turbine generator --reduction gear 1 foot away	104
Turbine generator-- reduction gear 2 feet away	94
Turbine generator-- generator 1 foot away	98
Turbine generator-- generator 2 feet away	96

Maximum noise level T-G reduction gear 1 foot away--104 db

IX. MALFUNCTION REPORTS

The plant malfunction reports are summarized in Table 57. The various reports were first categorized as to general area, namely:

- (1) Primary system
- (2) Secondary system
- (3) Nuclear instrumentation
- (4) Reactor component
- (5) Electrical equipment
- (6) Operator error.

Summary information was then compiled for each malfunction as follows:

- (1) Specific malfunction
- (2) Date
- (3) Malfunction report number
- (4) Corrective action taken.

Table 57 provides information as to the immediate corrective action taken as a result of each malfunction. Additional efforts were required to obtain acceptable operation in three areas, namely:

- (1) Turbine governor
- (2) Nuclear instrumentation
- (3) Control rod actuators.

A. TURBINE GOVERNOR

The manufacturer's representative observed the operation of the turbine and inspected the turbine governor on a number of occasions. Finally it was concluded that the main cause for governor malfunction was excessive air entrained in the turbine oil. Air bubbles passing over the lip of the speed governor cup valve will cause a false signal to go to the pilot relay of the servomotor. Also, excess air will tend to make the oil spongy.

Several steps were taken to minimize the effect of this air.

- (1) An orifice vent was installed in the high pressure line at the inlet to the speed governor.
- (2) The pilot relay of the servomotor was vented to the drain.
- (3) The cavity above the servomotor operating piston was also vented.
- (4) Baffle screens were added at the bottom of the reduction gear housing to prevent the windage of the gear from whipping up the oil and to break up the drain oil into small particles for better liberation of air before entering the pump.
- (5) The main oil pump suction was changed by adding a drilled section of 2-inch pipe. This will tend to prevent any vortexing which may have existed and also get a better sampling of oil closer to the bottom of the reservoir.
- (6) An accumulator was installed in the control oil line.

B. NUCLEAR INSTRUMENTATION

A number of relatively minor changes to the nuclear instrumentation system were accomplished over a period of time. These changes, which are summarized below, significantly improved the operation of the system.

- (1) Nuclear instrumentation signal cables for the source range were replaced with triax (double shielded) cable.
- (2) The ground bus in the nuclear instrumentation cabinet was modified to provide a lower resistance to ground.
- (3) The wiring of the primary coolant pump bypass coupler was revised to provide proper contactor arc suppression.
- (4) Arc suppression devices (condensers) were installed on the temperature scan stepping switch.
- (5) Ventilation in the nuclear instrumentation cabinet was improved to lower the temperature of the electronic equipment.

C. CONTROL ROD ACTUATORS

The operation of the control rod actuators was improved by lowering the operating temperature of the associated electronic gear. This was accomplished by improving the air circulation in the control rod drive mechanism cabinet.

TABLE 57
Malfunction Reports

General Area	Specific Malfunction	Date	Malfunction Report No.	Corrective Action Taken
Primary system equipment	Leakage on charging pump valves prevents pump operation	3/3/62	4	Maintenance done on pumps
	Radioactive iodine in primary coolant	4/3/62	15	False indication of iodine
	Purification demineralizer inlet sample valve leakage	4/13/62	18	Welded leak
	Pressurizer relief valve leakage	4/20/62	28	Replaced outlet piping with flexible pipe
	Pressurizer drain valve leakage	4/20/62	29	Replaced solenoid valve with air-operated valve
	Radioactive waste disposal system condensate pump cable short	6/20/62	49	Replaced cable
	Radioactive waste disposal system sump pump failure	7/30/62	73	Replaced pump
Secondary system equipment	Motor-driven condensate pump not pumping at capacity	3/17/62	5	Replaced with a larger capacity pump
	Condenser Valve 1B leakage from cracked body	3/19/62	6	Welded leak
	Steam generator blowdown line frozen	3/30/62	8	Reran line avoiding interconnect trough
	Turbine throttle valve drain line flange leakage	4/15/62	21	Repaired
	Main steam trip valve failure to close	4/15/62	22	Performed maintenance on valve

TABLE 57 (continued)

General Area	Specific Malfunction	Date	Malfunction Report No.	Corrective Action Taken
Secondary system equipment (continued)	Turbine throttle valve lifting rod binding caused frequency oscillation	6/16/62	41	Enlarged clearances
	Turbine-generator governor malfunction	6/19/62	44	Readjusted governor and servomotor needle and relay springs, respectively. Replaced servo relay valve and governor cup valve assemblies
		7/24/62	61	
		7/28/62	71	
		8/4/62	76	
		9/22/62	84	
	Turbine exhaust pressure control malfunction while placing ID condenser on line--vacuum lost	6/21/62	45	Performed inst maintenance
	Scram--turbine generator could not handle frequency response on step load 200 to 600 kw, low power primary pump scram	6/21/62	47	See malfunction No. 44
	Turbine-generator oil supply line leakage	7/19/62	52	Repaired
	Steam driven feedwater pump bearing failure	7/29/62	72	Dismantled pump and repaired bearings
	Evaporator blowdown backup condensate tank overflow line decreasing water quality	7/31/62	74	Check valve installed in condensate storage tank overflow
	Steam leakage in trap drain in inlet to desuper-heater	8/3/62	75	Welded leak and replaced pipe

TABLE 57 (continued)

General Area	Specific Malfunction	Date	Malfunction Report No.	Corrective Action taken
Nuclear instrumentation system equipment	Scram--turned to test on Channel 1, a false short period indication	4/3/62 5/18/62	14 31	Revised circuit
	Channel 7--loose pin connector on detector	5/23/62	33	Repaired
	Scram--erratic readings on Channels 1 and 2	4/14/62	20	Recalibrated
	Scram--no overlap between Channels 1, 2, 3 and 4, caused short period scram	6/16/62	42	Recalibrated source and intermediate range channels
	Scram--short period scram Channels 1 and 2 on shutdown	8/8/62	78	Recalibrated bistables
	Scram--short period, noise on Channel 3	3/3/62	3	See malfunction No. 31
	Scram--when changing Channel 3 from calibrate to run, a false short period was indicated	3/30/62	7	See malfunction No. 31
	Scram--Channel 3 gave false short period scram	7/23/62	56	Repaired leaking can on Channel 3 detector
	Channel 3 would not read in operator position	7/25/62	67	Operating and compensating voltages re-adjusted
	Radiation monitoring system power supply, detector and cable damaged by lightning striking plant	7/2/62	50	Protection against lightning added
	Scram--radiation monitoring system blown fuse	7/19/62	55	Replaced shorted component
		7/25/62	64	

TABLE 57 (continued)

General Area	Specific Malfunction	Date	Malfunction Report No.	Corrective Action Taken
Nuclear instrumentation system equipment (continued)	High voltage to radiation monitoring of steam generator blowdown failure	7/23/62	59	Replaced shorted component
	Low reading--Area 1 monitor, due to power supply	7/25/62	63	Replaced shorted component
	Scram--bypass coupler coolant pump trip	6/21/62 3/30/62	46 9	Additional ventilation added to NIS cabinets and some circuitry revised
	Scram--false short period, Channels 3 and 4, during startup	9/17/62	82	Realignment of NIS
Reactor component failure	No. 1 control rod dropped in, K-1 relay failed	2/25/62	1	Replaced K-1 relay
	No. 1 control rod dropped, No. 4 stuck K-1 relay	4/3/62	11	Replaced K-1 relay
	No. 4 control rod stuck, on way in, at a position 1.84 in. withdrawn	2/27/62	2	Removed rod bundle and repaired buffer piston of No. 4
	No. 4 control rod would not pull out (low current)	6/7/62	38	Increased hold, lift and grip currents to No. 4
	No. 4 control rod stuck on way out and bound	6/22/62	48	Replaced lift power supply of No. 4
	No. 4 control rod will not insert unless scrammed	7/25/62	66	Increased grip current to No. 4
	No. 6 control rod indicator out of calibration	7/25/62	65	Re-zeroed No. 6 position indicator
	Rod position indicators not reading correctly	6/7/62 6/14/62	39 43	Re-zeroed position indicators

TABLE 57 (continued)

General Area	Specific Malfunction	Date	Malfunction Report No.	Corrective Action Taken
Electrical equipment failures	Scram--station transformer primary breaker tripped when putting third condenser in service	4/10/62	16	Installed larger capacity transformer
	Scram--station transformer secondary tripped on over-current	7/18/62	51	Installed larger capacity transformer
	Scram--control rod power supply fuse blew	4/13/62	17	Cleaned control rod cabinet filters
		4/14/62	19	
	Automatic voltage regulator not operating	4/19/62	23	Westinghouse representative made repairs
	Manual voltage regulator not operating at 460 kw	4/19/62	25	Westinghouse representative made repairs
	Main generators kw and current meters not functioning	4/19/62	24	Instrument maintenance
	Diesel generator governor malfunction	4/19/62	26	Serviced governor
	Scram--starting diesel caused low voltage on vital a-c bus	4/19/62	27	Changed phase load distribution of vital a-c and d-c buses
	Scram--control rod hold power fuse became unsoldered due to high temperature	6/13/62	40	Cleaned air filter of rod control cabinet
	Scram--overheated switchgear	8/7/62	77	Additional ventilation added to switchgear area
	Load bank inoperative	5/25/62	34	Relay malfunction due to moisture

TABLE 57 (continued)

General Area	Specific Malfunction	Date	Malfunction Report No.	Corrective Action Taken
Electrical equipment failures (continued)	Low line voltage from diesel plant caused motor-generator set to trip out	3/30/62	10	Operate vital a-c system in isolated mode while diesels of radar site are supplying PM-1 load
	Scram--quality of radar or loss of site power	9/22/62	85	No action required
		4/3/62	12	
		4/3/62	13	
	Tieline breaker trip--short in switching handle	9/21/62	83	Repaired switching handle
	Load bank time delay would not pick up properly	9/30/62	87	Load bank removed before repairs could be made
	Turbine-generator exciter and main breaker did not open upon manual scram	10/2/62	88	Improper position of trip shaft--repaired
Operator error	Scram--auxiliary oil pump on turbine generator not secured when turbine started to roll	6/6/62	36	No action required
	Scram--improper removal of secondary equipment from operation on shutdown caused short period scrams	7/19/62	53	No action required
	Scram--driving rods in on shutdown caused false short period scram Channels 1 and 2	7/19/62	54	No action required

TABLE 57 (continued)

General Area	Special Malfunction	Date	Malfunction Report No.	Corrective Action Taken
Operator error (continued)	Scram--false short period indicated on Channels 1 and 2, wrong adjustment	5/16/62	30	No action required
	Channel 3 bistable not properly adjusted	7/23/62	57	No action required
	Channel 3 bistable not properly adjusted	7/24/62	58	No action required
	Scram--improper maintenance on Channel 3 chambers, short period scram	7/23/62	60	No action required
	Steam generator level too low on startup, operating procedures changed	7/25/62	68	No action required
	Scram--short on lighting circuit, resulting in tripping of main steam valve	7/27/62	69	Maintenance man inadvertently shorted circuit while performing maintenance
	Scram--false period on Channels 1 and 2	10/2/62	89	Channel 4 was switched from test to operate before the power level was well above the source scram disable level
	Scram--short period admitting steam to main steam line rapidly	9/13/62	79	No action required

TABLE 57 (continued)

General Area	Special Malfunction	Date	Malfunction Report No.	Corrective Action Taken
Operating error (continued)	Scram--admitting feed-water to steam generator too rapidly	10/2/62	90	No action required
	Scram--flux fell rapidly in intermediate range causing scram of source range instrumentation bistable No. 8 and No. 5	10/3/62	91	No action required
Unaccounted for scrams		6/7/62	37	
		9/29/62	86	
		9/17/62	81	
		9/14/62	80	



X. REFERENCES

1. Maurer, G. W., "A Method of Predicting Steady-State Boiling Vapor Fractions in Reactor Coolant Channels," WAPD-BT-19 (June 1960).
2. Zindler, G. F., et al., "14th PM-1 Quarterly Progress Report," MND-M-1825, September 1962.
3. Anderson, D. C. and Shure, K., "Thermal Neutron Flux Distributions in Metal-Hydrogenous Shields," Nucl Sci and Eng 8 260, 1960.
4. Tittle, C. W., "Slow-Neutron Detection by Foils-II," Nucleonics, 9, 60, July 1951.
5. Stewart, L., "Neutron Spectrum and Absolute Yield of a Plutonium-Beryllium Source," Phys Rev 98, 740, (1955).
6. Neutron Cross Sections, BNL 325, 2nd edition.
7. Swope, M. G., "Dosimetry in the Argonne High-Level Gamma-Irradiation Facility," ANL-5819.
8. Journal of Chemistry and Physics, 24, 56-9, 1956.
9. Bagley, R., Cox, F., et al., "PM-1 Reactor Core Final Design Report," MND-M-1866, January 1962.
10. Rosenthal, H., Bagley, R., et al., "PM-1 Critical Experiments and Zero Power Testing," MND-M-1858, September 1961.
11. Baer, R. L., "PM Development--Plant Design Studies," MND-MD-2541, December 1961.
12. Carlson, B., Lee, C., and Worlton, J., "The DSN and TDC Neutron Transport Codes," LAMS-2346, 1959.
13. Keppler, J. G. and Orr, W. L., "A Three-Group Diffusion Calculation (Program F3)," XDC 58-7-18.
14. Keppler, J. G. and Orr, W. L., "Three-Group Reactor Program Using Epithermal Cell Corrections (704 Program C-3)," XDC 58-3-178, February 1958.
15. Small, W. J., Zegger, J. H., and Medin, A. L., "Long-Lived Circulating Activity During APPR-1 700-Hour Test," Preprint 82, Nuclear Engineering and Science Conference, March 1958.

16. Bolles, R. C. and Ballou, N. E., "Calculated Activities and Abundances of U-235 Fission Products," USNRDL-456, August 30, 1956.

APPENDIX A

PLANT TRANSIENT DATA

The PM-1 transient performance was recorded by an automated Data Acquisition System. The data is presented in graphical form in the Illustrations portion of this report. The data is generally described in Sections E and F of Chapter III. The specific transient runs and the corresponding pages of the illustrations are defined in Tables 19 and 21.

APPENDIX B

EXPERIMENTAL DETERMINATION OF GAMMA AND NEUTRON FLUXES

Details of the experimental method are provided in this appendix. Four areas are discussed, namely:

- (1) Thermal flux measurements in the reactor and steam generator tanks.
- (2) Thermal flux measurements at the interconnect.
- (3) Fast flux measurements.
- (4) Gamma dose rate measurements.

In general, the discussion for each type of measurement includes information on:

- (1) Detector materials
- (2) Positioning
- (3) Procedures
- (4) Experimental analysis.

A. THERMAL FLUX MEASUREMENTS IN REACTOR AND STEAM GENERATOR PACKAGES

1. Detector Materials

Thermal flux was measured by means of the activation of metal foils by (n, γ) reactions. For the measurements in the reactor tank, the sensitive material was dysprosium in the form of a 5% dispersion of dysprosium in aluminum. Two sizes of the foil were used: one, 3/16 inch in diameter and 0.005 inch thick and the other, 1/32 inch in diameter and 0.001 inch thick. For the measurements in the steam generator tank, the sensitive material was pure indium in the form of a foil 0.010 inch thick and 3/4 inch in diameter.

2. Positioning

The measurements in the reactor tank were carried out using a 1/8-inch thick plexiglass plate containing an array of 1/4-inch diameter holes (Ref. 2). The foils were held in place in the center of these holes between plugs of polyethylene, which, in turn, were held in place with

mylar tape. Unused holes were left empty and allowed to fill with the shield water. The plexiglass plate was held in place against the lead shielding surrounding the pressure vessel and in a vertical and radial plane of the reactor core by means of an aluminum framework hung from the beams of the refueling work platform.

For the measurements in the steam generator tank and at the top of the reactor tank, the foils were positioned by enclosing them in polyethylene bags and lowering them into place by strings.

It is estimated that positioning in the reactor tank was reproducible to approximately $\pm 1/16$ inch axially and $\pm 1/4$ inch radially from the core. Positioning reproducibility in the steam generator tank for the vertical traverse along its length was approximately $\pm 1/4$ inch vertically and up to ± 2 inches horizontally in any direction for measurements at the bottom of the tank. For the measurement at the interface in the steam generator tank, the positioning accuracy was approximately $\pm 1/8$ inch vertically and $\pm 1/2$ inch horizontally.

3. Procedures

The foils in their holders were inserted into position and removed while the reactor was operating at power. The insertion and removal were accomplished as quickly as possible and the times for these operations were recorded. While the foils were in position, the reactor power level was observed on the nuclear power level instruments in the reactor control room, and all significant fluctuations were recorded. The calibration of the power level instruments was checked several times during the course of the program against a calorimetric heat balance computation, but no significant variations were observed.

4. Analysis

Following the activation, the foils were counted in a windowless flow proportional counter, using preset time techniques. The data was then analyzed. The correction factors which were used are discussed below in the order in which they were usually applied to the data.

a. Resolving time

The resolving time of the flow proportional counter instrumentation was determined by counting repeatedly the apparent activity of a highly activated dysprosium foil as it decayed. (This also verified that there was no significant extraneous activity of other half-lives in the foils.) With the data thus obtained, the resolving time was determined by using the formula for counting losses in nonparalyzable-type counting circuits, namely:

$$N = \frac{n}{1 - n\rho} \approx n(1 + n\rho)$$

where

n = observed counting rate

N = true counting rate

ρ = resolving time.

This measurement of the resolving time was carried out each time the thermal flux was measured in the reactor tank. The thermal flux measurements in the steam generator tank and the fast flux measurements in the reactor tank resulted in activities too low for the resolving time to produce a significant effect. The data for the counting rates in dysprosium was corrected for counting loss due to this measured resolving time.

b. Background

The background counting rate for the counter with no foil or with an unactivated foil was measured at the beginning and end of the counting of each set of foils from a reactor run, and more often when conditions warranted. The measured background rate was subtracted from the observed counting rates after they were corrected for resolving time.

c. Foil decay

The count rates were next corrected for the decay of the foils between the time they were irradiated and the time they were counted. The decay constants used were $0.012803 \text{ minute}^{-1}$ for indium and $0.004980 \text{ minute}^{-1}$ for dysprosium.

d. Reactor power and exposure time

Using the above decay constants, the duration of the irradiation time and the reactor power level during irradiation, the observed activities were next corrected to the saturation activity at full reactor power.

e. Intercalibration

Since the foils differed slightly from each other in the amount of indium or dysprosium each contained, the activities were next corrected by intercalibration factors. These intercalibration factors were obtained,

in the case of the indium foils, by weighing each one to the nearest 0.0001 gram (approximately 0.02%). In the case of the dysprosium foils, this method could not be used because of their small size and because of the uncertainty as to whether the percent concentration of the dysprosium was constant. The dysprosium foils were therefore intercalibrated by irradiating them simultaneously on a rotating wheel in the PMZ reactor so that each received the same average neutron flux. The measurement of the relative activation of these foils was then used to compute the intercalibration factors.

f. Epithermal corrections

Since indium has a large epithermal activation cross section, the thermal activation of the indium foils was determined by activating both bare foils and foils covered with 0.030-inch thick cadmium sheet. The thermal activation was then computed, using the formula

$$A_{th} = A_T - F_{cd} A_{cd}$$

where

A_T = total activity measured with bare foils

A_{cd} = activity measured with cadmium-covered foils

F_{cd} = correction factor to account for the attenuation of epithermal neutrons in cadmium, taken to be equal to 1.073 (Ref. 4).

Dysprosium has a relatively low epithermal activation; hence, the data was not corrected for it. Spot checks of the cadmium-covered activity were made at several locations, however, and these indicated that neglect of this correction might cause an overestimation of the flux by about 1% near the reactor core and by as much as 6% at the farthest measured location from the core.

g. Absolute calibration

The final correction made to the data was the absolute calibration of the counting rates in terms of thermal fluxes. This calibration was performed by irradiating a few selected foils in the known thermal flux of the critical facility sigma pile and then counting the foils in the same type of counter as was used previously. The observed counting rates, when corrected for background and corrected to saturation activity, were then used to obtain the absolute calibration factor in the form of

(neutrons/cm² second)/(counts/minute). The counter used was not the same one as had been used in the experiment; however, an intercalibration of the two counters, using standard sources, showed their sensitivities to be the same within approximately $\pm 1\%$.

The sigma pile itself was calibrated in terms of thermal neutron flux by using a technique involving the activation and absolute counting of gold foils with a $\beta - \gamma$ coincidence detection instrument. Due primarily to the uncertainty in the gold activation cross section, the accuracy of the absolute calibration is approximately $\pm 10.6\%$.

B. INTERCONNECT THERMAL FLUX MEASUREMENTS

The procedures and analyses for the thermal measurements at the steam generator interconnect were somewhat different because the foils could not be inserted into place while the reactor was running and because there was only one opportunity for doing so with the reactor shut down. Accordingly, three sets of foils were placed on the interconnect before the reactor startup. Each set formed a vertical traverse across the face of the interconnect. The three sets were placed as close to one another as practicable; however, they were separated from each other by approximately two-inch spacings in the direction horizontally across the face of the interconnect.

The first set of foils was removed after the reactor had been brought up to power and the second set, some time later. The measured activity in the first set of foils was corrected for background, decay before counting and the intercalibration factors; then it was further corrected to the activity that was remaining at removal of the second set. The activity of the second set of foils, which contained both bare and cadmium-covered foils, was also corrected for background, intercalibration and decay since removal. The ratio of the corrected activities of the bare foils in the first and second sets of foils then gave the fraction of the observed activity in the second set which was due to their irradiation during the time after the first set had been removed. The fraction was then used to correct the bare and cadmium-covered activities. The activities were then corrected for the power level and irradiation time for the period after removal of the first set of foils. The thermal activity was then computed, and the absolute flux was found. Unfortunately, wide fluctuations in the reactor power level during this irradiation limited the accuracy of the results.

The third set of foils was left in place until such time as the reactor had been operating at one stable power for a long enough interval (approximately four hours) to establish the saturation activity in the foils of that power level. The data was then analyzed in the normal way, ex-

cept that the epithermal correction was obtained from the observed cadmium ratios of the second set of foils, mentioned previously.

C. FAST FLUX

1. Detector Material

The fast neutron flux was measured by the activation of the S^{32} content of sulfur by means of the (n, p) threshold reaction. The detector "foils" were formed by pressing flowers of sulfur into polystyrene cups approximately 13/16 inch in diameter and 3/8 inch deep. The cups were then covered over with mylar tape to seal the sulfur in place. In the initial reactor exposure, some sulfur foils of a larger size were also used. These foils were approximately 2 inches in diameter and 1/4 inch thick. They were also contained in polystyrene cups, but did not have the mylar tape covering them. Instead, they were sealed with a polystyrene cover, which was removed during the counting operation.

2. Positioning

Positioning in the reactor tank was accomplished with the same metal positioning tool as for the thermal flux measurement, but with a different plexiglass plate (Ref. 2) to hold the foils. The plate, in this case, was one inch thick, drilled with blind holes 2-1/4 inches in diameter and 1/2 inch deep into which the foils were inserted. In the first series of fast flux measurements, this plate was covered over with a solid one-inch thick plexiglass cover for purposes of waterproofing. In subsequent runs, the second plate was discarded, and mylar tape was used to seal the foils in place. The reproducibility of positioning of the foils in the tank was approximately $\pm 1/4$ inch vertically and $\pm 3/8$ inch radially from the core.

3. Procedures

The irradiation procedure followed the same techniques described in the thermal flux measurements.

4. Analysis

Following the irradiation, the foils were sent back to the critical facility in Baltimore where they were counted on a thin window proportional flow counter, using preset time techniques. The counting rates were found to be sufficiently low that resolving time corrections were not necessary. The decay of two of the foils was followed over a sufficiently long time to establish that there was no appreciable amount of

extraneous activity of other half-lives. In particular, the S^{35} activity was negligible. The factors which were used in correcting the data are described below.

a. Background

The background counting rate was measured in the manner described previously and subtracted from the observed counting rates.

b. Foil decay

The count rates were then corrected for decay of the foil in the interval between irradiation and counting. The half-life used for this correction was 343.2 hours.

c. Reactor power and exposure time

The activities were then corrected to the saturated activities at full reactor power.

d. Intercalibration

The foils had purposely been made sufficiently thick that self-absorption of the beta activity of the P^{32} rendered the foil essentially independent of the exact amount of sulfur present and dependent only on the foil area. Since variations in the area of the foils were insignificant, intercalibration of the foils was neither necessary nor attempted.

e. Absolute calibration

The absolute calibration of the foils was carried out by exposing a few foils to the fast neutron flux from a plutonium-beryllium neutron source. The observed counting rate was corrected for background and corrected to the saturation activity. This activity can be expressed by the formula

$$AK = \phi_T \sigma$$

where

A = activity

ϕ_T = total neutron flux from the source

σ = S^{32} (n,p) cross section weighted over the Pu-Be neutron spectrum

K = unknown calibration factor, which depends on counter efficiency and similar factors.

Using this formula and substituting in the observed activity, the neutron flux (as computed from the source manufacturer's quoted neutron emission, emission growth rate and emission anisotropy, plus the geometrical factors of the irradiation) and the weighted cross section (whose computation will be explained later), the value of K is found to be 5.442 barn neutrons/cm² second/counts/minute. This factor was then used to correct the data. Multiplying this by the observed foil activities in counts per minute gives the product of cross section and flux in units of barn neutrons/cm² second. The quantity ($\sigma \phi$) equals

$$\int_{E_0} \sigma(E) \phi(E) dE \text{ of the total flux (above the energy } E_0) \text{ times the}$$

activation cross section (weighted against this flux). It is this quantity which is reported here, rather than the absolute flux, since the flux spectrum must be known before the flux magnitude can be obtained from these results.

The flux-weighted cross section for the plutonium source was calculated from both a theoretical and an experimental spectrum for the Pu-Be source (Ref. 5) and an experimental sulfur activation cross-section curve (Ref. 6). The calculation used a technique which involved breaking each curve up into increments of energy and graphically interpreting each curve in each energy increment to find the average over that increment. Two sets of energy increments were used: one set of equal increments 1 Mev wide and the second based on equal lethargy increments. The results obtained for the Pu-Be weighted sulfur cross section are shown in Table B-1 for the experimental and theoretical curves and the equal energy or lethargy increments. The value used in the calculation of the data was the average of the four values shown.

TABLE B-1
Pu-Be Flux-Weighted Cross Sections
for Sulfur in Millibarns

<u>Increments</u>	<u>Experimental Flux</u>	<u>Theoretical Flux</u>
Energy	213.4	199.4
Lethargy	213.4	201.5

D. GAMMA DOSE RATE MEASUREMENTS

1. Techniques

Three different techniques were used for measuring the gamma dose rate. In the reactor tank, chemical dosimetry was employed, using the standard $\text{Fe}^{++} - \text{Fe}^{+++}$ dosimetry technique (Ref. 7), which has an energy-independent response in the range from 40 kev to 2 Mev. The chemical solution, which consisted of 0.001 normal FeSO_4 , 0.001 normal NaCl and 0.8 normal H_2SO_4 , was used in glass vials approximately 1/2 inch in diameter and 1-3/4 inches long. In the steam generator tank, film dosimetry techniques were employed, using standard Dupont Type 558 film packets approximately 1-1/4 by 1-3/4 inches in size. In the primary building itself, a Victoreen "Radgun" model AGB-10KG-SR was used to make the measurements. This has an energy-independent response in the range from 80 kev to 1.2 Mev.

2. Positioning

The vials were positioned in the reactor tank by means of the fast flux foil holder described earlier. The vials were placed in a vertical position in the holder. Positioning reproducibility was comparable to that of the fast flux foils. The dosimetry film packets were placed in the steam generator tank by using the same techniques employed with the thermal flux foils and with equal reproducibility.

3. Procedure and Analysis

The same general irradiation procedure was used with the chemical and film dosimeters as was described for the neutron flux irradiations. Meter readings from the Radgun were obtained in the primary package while the reactor was operating at stable power. The data was then corrected for the reactor power level so as to give results in the form of mr/hr at full power.

The film packets were developed by a commercial health physics laboratory. The data was reported in units of rep. This was then corrected for reactor power and duration of exposure so as to give results in terms of rep/hr at full reactor power.

The chemical dosimeters were measured on a Beckman spectrophotometer at a wavelength of 305 millimicrons and a slit width of 0.4 millimeter. The raw data was obtained in the form of an optical density. This was corrected first by subtraction of the optical density of the unirradiated solution. The data was converted to rads by use of the formula:

$$R = \frac{KADE}{PG}$$

where

R = radiation dose in rads

K = instrument calibration factor in micromoles Fe^{+++} /liter/
optical density

A = constant 6.025×10^{17} molecules/micromole

D = measured optical density change in the solution

E = constant 1.6×10^{-12} rad/100 ev absorbed/gram

P = solution density of 1078.7 grams/liter

G = 15.45 ± 0.11 molecules of Fe^{+++} created/100 ev absorbed
(Ref. 8).

Substituting in the above values: $R = KD (57.9 \pm 0.4)$.

The calibration factor K, relating optical density to ferric concentration, was determined in two ways. The first method involved irradiating a batch of the solution to saturation so that all the ferrous ions were oxidized to ferric. The optical density of this known concentration of ferric was then measured, as was the optical density of various dilutions of this solution. This served not only to establish the value of K but to verify the linearity of the relationship between optical density and ferric concentration. The second method of determining K was to irradiate several vials of the solution for different lengths of time in the known gamma flux of the Martin gamma irradiation facility. The calibration of the gamma facility itself was checked with a Victoreen roentgen ratemeter. The calibration factor K which was used in the analysis of the data was 424 micromoles Fe^{+++} /liter/optical density. It is estimated from the agreement of the above two methods that the accuracy is approximately $\pm 10\%$.

Substituting this value into the previous equation gave $R = 2.46 \times 10^4 D \pm 10\%$.

APPENDIX C

ANALYTICAL DETERMINATION OF NEUTRON FLUXES

This appendix discusses the analytical methods utilized to predict flux levels in the PM-1 plant, namely:

- (1) Thermal flux computations.
- (2) Fast flux computations.
- (3) Calculation of argon activity.

A. THERMAL FLUX

Thermal flux was calculated as a function of radial distance from the PM-1 reactor core at various axial heights, using Eq (1) as presented in Ref. 3.

$$\phi_{th}(r) = \frac{\phi_w^{PK}(r) \phi^{P1MG}(r)}{\phi_w^{P1MG}(r)} \quad (1)$$

where

$\phi_{th}(r)$ = 2200-m/sec thermal neutron flux at a distance r from the core in the actual metal-water configuration

$\phi^{P1MG}(r)$ = P1 multigroup estimates of the Maxwellian thermal neutron flux in the actual metal-water configuration

$\phi_w^{PK}(r)$ = 2200-m/sec thermal neutron flux at a distance r in an all-water medium surrounding the core

$\phi_w^{P1MG}(r)$ = P1 multigroup estimate of the Maxwellian thermal flux in an all-water medium surrounding the core.

The point kernel used in the calculation was determined from Eq (2), which was considered to be valid within 15% in the range from 15 to 160 cm of water.

$$\frac{4 \pi r^2 \phi(r)}{S} = 17.2 e^{-0.1557 r} + 0.357 e^{-0.0951 r} \quad (2)$$

where

$\phi(r)$ = thermal neutron flux at a distance r from the point source ($\text{n/cm}^2\text{-sec}$)

r = distance (cm) from source in water at STP

S = source strength in fission neutrons per second.

To determine the flux distribution in the all-water medium, the point kernel was integrated over the reactor core, using the SPEND code.

The P1 multigroup diffusion flux distribution was determined through the P1MG code, which utilizes 54 energy groups to cover the range from 10 Mev to 0.625 ev and a single group to describe thermal energies.

B. FAST FLUX

As described above, the modified P1MG approximation, used to calculate thermal neutron flux, calculates flux in 55 energy groups. It has been demonstrated (Ref. 3) that this method produces a reliable computation of flux distribution for fast neutrons. Calculation of fast neutron flux distributions for the PM-1 Plant Performance Studies programs was accomplished by considering the first seven energy groups (from 10 to 1.74 Mev). In applying the P1MG approximation to fast neutron flux, Eq (1) must be modified. Equation (1) may be rewritten as:

$$\phi_{\text{fast}}(r) = R \phi^{\text{P1MG}}(r) \quad (3)$$

where

R = ratio of fluxes in an all-water medium surrounding the core, i.e.,

$$R = \frac{\phi_w^{\text{PK}}(r)}{\phi_w^{\text{P1MG}}(r)} \quad (4)$$

Since $\phi_w^{PK}(r)$ represents a 2200-m/sec thermal neutron flux while the thermal flux represented by $\phi_w^{P1MG}(r)$ is based on a Maxwellian distribution, these fluxes must be corrected using Eq (5) when computing this fast flux to remove the Maxwellian dependence.

$$\sigma_{2200} \phi_{2200} = \sigma_{\max} \phi_{\max} \quad (5)$$

The normalization factor, the ratio of the 2200 m/sec to Maxwellian total macroscopic cross sections of water, was calculated to be 1.24.

The experimental fast neutron data were measured and reported in terms of the activation of Sulfur-32. To permit comparison of calculated and experimental fast neutron data, the calculated fast flux data was converted to Sulfur-32 activations by applying Eq (6), integrated over the first seven energy groups.

$$A(r, Z) = 1.24 R(r, Z) \sum_{i=1} \phi_i^{P1MG}(r) \sigma_i \Delta_i \quad (6)$$

C. ARGON ACTIVITY CALCULATIONS

A volume weighted average thermal neutron flux of 1.4×10^4 n/cm²-sec for the steam generator package during full power operation was determined from the experimental data. This is shown in Fig. 33. The saturation (maximum) activity in a closed loop is given by

$$A = \sigma \phi N \frac{1 - e^{-\lambda t}}{1 - e^{-\lambda T}}$$

where

A = saturation activity (disintegration/sec/milliliter)

σ = activation cross section (cm²)

ϕ = thermal flux (neutrons/cm²-sec)

N = number of atoms of the parent nuclide (A^{40}) per unit volume

t = time that sample spends in the flux field (sec)

T = total loop circulating time (sec)

λ = decay constant of A^{41} (sec^{-1})

For the case under consideration, $\sigma \phi \ll \lambda$ and $\lambda T \ll 1$ and the above expression may be adequately approximated as

$$A = \sigma \phi N \frac{t}{T}$$

Assuming the times t and T to be directly proportional to the volume of the steam generator package ($V \approx 1700 \text{ ft}^3$) and the total volume in the package and maintenance area ($V \approx 11,100 \text{ ft}^3$), respectively, then for a flow rate of $644 \text{ ft}^3/\text{min}$ to the steam generator package:

$$t = \frac{V}{F} = \frac{1700 \text{ ft}^3}{644 \text{ ft}^3/\text{min}} = 2.6 \text{ min}$$

$$T = \frac{V}{F} = \frac{11,100}{644 \text{ ft}^3/\text{min}} = 17.2 \text{ min}$$

For Argon-40, assuming air at a density of $1.29 \times 10^{-3} \text{ gm/ml}$

$$\sigma = 0.53 \times 10^{-24} \text{ cm}^2$$

$$N = 2.56 \times 10^{17} \frac{\text{atoms } A^{40}}{\text{ml of air}}$$

(isotopic abundance of A^{40} in argon is 0.996; weight fraction of argon in air is 0.013).

Thus the argon activity is

$$A = 2.8 \times 10^{-4} \text{ dis/ml-sec}$$

or

$$A = 7.6 \times 10^{-9} / \frac{\mu \text{ curies}}{\text{ml of air}}$$

It is significant to note that the above calculation should represent a very conservative estimate of the argon actually present at the plant during full-power operation since, in the present plant design, cooling air for the steam generator is taken directly from outside the maintenance building. Thus, the present setup closely approximates a single-pass system for which the activity per unit volume of air produced would be much lower.



4

4

4

4



APPENDIX D

NUCLEAR ANALYTICAL TECHNIQUES

This appendix discusses the details of the nuclear analytical techniques utilized in this program. In many instances, methods are identical to those described in Ref. 9. However, some changes were made in the geometrical models used in the calculations.

During the PM-1 final design, a six-radial region breakup of the core was used. It was felt that better results could be obtained if more radial regions were used; therefore, the radial core configuration was increased to seven regions, as shown in Fig. D-1. This configuration will be referred to as the seven-radial region core.

An addition was also made in the radial reflector region. During the final design studies, preliminary diffusion calculations indicated that the stainless steel pressure vessel would have little effect on core reactivity. However, comparison of PM-1 rod bank positions obtained during the PM-1 preshipment criticals and at initial startup at Sundance showed a pressure vessel worth of $\sim 0.23\%$ in reactivity. Therefore, the pressure vessel has been included in all radial calculations.

Cell correction calculations performed during this study used an S_6 approximation to the transport equations, as did the calculations in the final design. However, a change was made in the IBM machine code used. Previously, one-group calculations, with corrections for source and slowing down, were made for each level. The latest calculations were performed with a multigroup code which automatically handles source and slowing down.

A change was also made in the geometrical model used in the cell correction. During the final design studies, it was found that consistently low cell corrections were obtained when transport theory was used with a zero gradient at the outer boundary of the lumped poison cell. The condition was corrected by surrounding the cell with a large homogenized core region and by forcing the flux to zero at the outer boundary. Subsequent studies have shown that the same results are obtained when the homogenized core region is reduced and a zero flux gradient is used. (This reduction in size is advantageous in that it reduces the running time of the problem.)

The following topics are discussed individually in this appendix:

- (1) Initial reactivity
- (2) Control rod worth

- (3) Core lifetime
- (4) Xenon reactivity effects.

A. INITIAL REACTIVITY

The calculation of initial core reactivity is discussed. Methods of obtaining cell corrections (to compensate for the heterogeneity of the core) and core buckling (to account for neutron leakage) are treated individually. Basic cell corrections, radial and axial bucklings, and reflector group constants are reported.

1. Cell Corrections

Epithermal and thermal cell corrections, defined as the ratio of the average flux in the material to the average flux in the cell, were calculated for the PM-1 fuel elements and lumped poison rods. GAM 22-group, fast cross sections and spectrum hardened (i.e., T_{eff} corresponding to the average hardened temperature in the core) thermal cross sections were used in a multigroup, multiregion S_6 transport calculation (Ref. 12).

The geometric models used are shown in Figs. D-2 and D-3. The two models, necessitated by the PM-1 core configuration were:

- (1) Fuel element cell--one fuel element surrounded by its associated water.*
- (2) Lumped poison rod cell--one lumped poison element surrounded by five homogenized fuel elements.

The 22-group cell corrections obtained were reduced by averaging over lethargy to the desired 19 groups required by the diffusion theory code used to calculate core reactivity. The resulting cell corrections are given in Tables D-1 and D-2.

*Previous studies have shown that, in cylindrical geometry transport theory, the use of a zero gradient at the outer radius of the cell yields cell corrections which are consistently low, especially if the last region is water. Therefore, a region of homogenized fuel elements was placed around the fuel element cell to minimize this effect.

TABLE D-1
PM-1 As-Built Cold Cell Corrections

<u>Cell Corrections</u>						
<u>Lumped Poison Rod Model</u>				<u>Fuel Element Model</u>		
<u>Lethargy</u>	<u>Lumped Poison</u>	<u>Water</u>	<u>Homogenized Core</u>	<u>Fuel Matrix</u>	<u>Fuel Element Steel</u>	<u>Water</u>
10	0.988	1.002	1.001	0.993	0.994	1.001
12	1.000	1.003	0.999	0.984	0.986	1.003
14	0.975	1.007	1.002	0.973	0.977	1.005
15.5	0.938	1.010	1.010	0.987	0.988	1.002
16.5	0.882	1.014	1.014	0.976	0.979	1.004
Thermal	0.561	1.066	1.053	0.838	0.857	1.030

It should be noted that, when the cell corrections were used in the reactivity calculations, they were normalized to the cell geometry of the problem. For example, in the seven-regional radial calculations, the fuel element cell correction was defined as

$$g = \frac{\Phi_{\text{fuel}}}{\Phi_{\text{fuel element cell}}}$$

in the regions without lumped poison rods and

$$g = \frac{\Phi_{\text{fuel}}}{\Phi_{\text{fuel element cell}}} \times \frac{\Phi_{\text{homogenized core region}}}{\Phi_{\text{lumped poison rod cell}}}$$

in the regions with lumped poison rods.

2. Buckling and Reflector Constants

Radial and axial core buckling, for use in the diffusion theory reactivity calculations, were obtained from diffusion theory flux shapes.

The average B^2 was calculated by integrating the Laplacian over the core.

TABLE D-2
PM-1 As-Built Hot Cell Corrections

<u>Lethargy</u>	<u>Lumped Poison Rod Model</u>				<u>Fuel Element Model</u>							
	<u>Lumped Poison</u>		<u>Water</u>		<u>Homogenized Core Region</u>		<u>Fuel Matrix Fuel Element Steel</u>				<u>Water</u>	
	<u>Bol</u>	<u>Eol</u>	<u>Bol</u>	<u>Eol</u>	<u>Bol</u>	<u>Eol</u>	<u>Bol</u>	<u>Eol</u>	<u>Bol</u>	<u>Eol</u>	<u>Bol</u>	<u>Eol</u>
10	0.982	0.980	0.998	0.994	1.002	1.003	0.992	0.994	0.994	0.996	1.001	1.001
12	0.996	1.001	1.007	1.003	1.000	1.000	0.984	0.988	0.986	0.990	1.003	1.002
14	0.972	0.998	1.008	1.008	1.002	1.000	0.973	0.980	0.977	0.983	1.005	1.004
15.5	0.914	0.982	0.984	0.999	1.010	1.002	0.987	0.990	0.989	0.992	1.002	1.002
16.5	0.882	0.977	0.979	1.000	1.014	1.002	0.976	0.983	0.980	0.985	1.004	1.003
Thermal	0.653	0.946	0.938	1.033	1.041	1.002	0.876	0.908	0.893	0.919	1.022	1.017

NOTE:

Bol--Beginning of life

Eol--End of life.

$$\overline{B_r^2} = \frac{- \int_{\text{core}} \vec{\nabla} \cdot \vec{\nabla} \phi(r) dr}{\int_{\text{core}} \phi(r) dr} \quad (1)$$

The resulting three-group bucklings are presented in Table D-3. The negative values in the thermal group show leakage into the core from the reflector.

Reflector constants were obtained by using the neutron leakage (i. e., $DB^2 \phi$) out of the core as the source spectrum in modified age theory slowing-down calculations. The resulting three-group constants are given in Table D-4.

3. Core Reactivity

Initial core reactivities were calculated by using a seven-region radial core configuration (see Fig. D-1) in a one-dimensional, three-group, multiregion diffusion code (Refs. 13 and 14). Modified age theory was used to obtain three-group core and reflector constants. The thermal cross sections were evaluated at an effective neutron temperature to account for spectrum hardening in the core. The effective temperature was calculated by:

$$T_{\text{eff}} = T_o \left(1 + 0.75 \frac{\sum a_{\text{thermal}}}{\xi \sum s_{0.683 \text{ ev}}} \right)^2 \quad (2)$$

where T_o = core temperature in °K.

B. CONTROL ROD WORTH

This section discusses the calculation of worth versus insertion of the PM-1 six-rod bank by means of a window shade model technique. Both hot and cold conditions were investigated. Critical rod bank positions were also calculated and normalized to experimental data.

A window shade model technique was used for calculating worth versus insertion of the PM-1 six-rod bank. The window shade technique consists of simulating the rod bank insertion by smearing a uniform poison over the rodded region of the core in an axial calculation.

TABLE D-3
 PM-1 As-Built Radial and Axial
 Three-Group Buckling (68° F and 463° F)

<u>Temperature</u>	<u>Radial Buckling</u>			<u>Axial Buckling</u>		
	<u>B₁²</u>	<u>B₂²</u>	<u>B₃²</u>	<u>B₁²</u>	<u>B₂²</u>	<u>B₃²</u>
68° F	5.750 x 10 ⁻³	4.161 x 10 ⁻³	-8.766 x 10 ⁻³	1.421 x 10 ⁻³	1.114 x 10 ⁻³	-1.562 x 10 ⁻³
463° F	5.694 x 10 ⁻³	3.555 x 10 ⁻³	-7.877 x 10 ⁻³	1.430 x 10 ⁻³	1.046 x 10 ⁻³	-2.143 x 10 ⁻³
Energy breakpoints	Fast	2.470 x 10 ⁴ ev				
	Epithermal	0.683 ev				

TABLE D-4
Three Group Reflector Constants
for the PM-1 As-Built Core
(68° F and 463° F)

	<u>Water</u>		<u>Stainless Steel</u>		<u>Upper Dead Ends</u>		<u>Lower Dead Ends</u>	
	<u>68° F</u>	<u>463° F</u>	<u>68° F</u>	<u>463° F</u>	<u>68° F</u>	<u>463° F</u>	<u>68° F</u>	<u>463° F</u>
D_1	1.8277	2.3135	1.0574	1.0825	1.5120	1.7502	1.4165	1.5936
D_2	0.5899	0.7245	0.3292	0.3300	0.4815	0.5411	0.4457	0.4870
D_3	0.1468	0.2316	0.3062	0.3240	0.1721	0.2518	0.1864	0.2620
Σ_{a_1}	0.0003	0.0003	0.0019	0.0019	0.0006	0.0006	0.0008	0.0008
Σ_{a_2}	0.0009	0.0007	0.0087	0.0087	0.0043	0.0042	0.0059	0.0057
Σ_{a_3}	0.0195	0.0120	0.2370	0.1773	0.0808	0.0586	0.1082	0.0794
Σ_{sl_1}	0.0861	0.0707	0.0038	0.0038	0.0710	0.0591	0.0623	0.0521
Σ_{sl_2}	0.1349	0.1099	0.0011	0.0011	0.0959	0.0780	0.0784	0.0636

The uniform poison was calculated by weighting PM-1 Y-rod absorber constants, obtained from Blackness Theory, with the ratio of the flux in a Y-rod blade to the flux in the core. This flux ratio was obtained from a two-dimensional, few-group calculation of the PM-1 core with the Y-rods explicitly represented.

The three-group uniform absorber constants are given in Table D-5.

TABLE D-5
Uniform Absorber Constants
for Window Shade Calculation

	<u>Cold</u>	<u>Hot</u>
Σ_{a_1}	2.46×10^{-4}	2.49×10^{-4}
Σ_{a_2}	1.70×10^{-2}	1.70×10^{-2}
Σ_{a_3}	1.69×10^{-2}	1.82×10^{-2}

The axial calculation employed a three-group, multiregion diffusion code in slab geometry. The control rod worth as a function of insertion was calculated from full-out to the critical position for both the hot and cold conditions.

C. CORE LIFETIME

This section discusses the PM-1 core burnup calculations. Non-uniform burnup lifetimes were determined with and without control rods inserted.

1. Lifetime with Rods Fully Withdrawn

The radial nonuniform burnup calculation treated a seven-region core configuration in a one-dimensional, three-group, multiregion diffusion code (Refs. 13 and 14). The program calculated regionwise burnup and fission product buildup in 100-day time steps. The reflector group constants and the perpendicular buckling were calculated as described above. Epithermal and thermal cell corrections were varied as a function of burnup by the expression:

$$g_i = \frac{1}{\alpha_i + \beta_i F} \quad (3)$$

where

g_i = cell correction at level i

F = fraction of material remaining

$\alpha_i = \frac{1}{g_i}$ (at beginning of life)

$\beta_i = \frac{1}{g_i} - \frac{1}{\alpha_i}$ (at end of life).

The axial nonuniform burnup was performed in a manner similar to the radial, except that the core was divided into 12 equal axial regions.

The uniform burnup in both the radial and axial directions (the results are identical) were likewise performed as described above, except that a one-region homogenized core region was used.

Applying the radial nonuniform burnup effect (i. e., ρ uniform - ρ nonuniform) to the axial nonuniform burnup resulted in a calculated lifetime of 16.5 Mw-yr.

2. Lifetime with Rods Partially Inserted

Core burnup, with rods inserted to their critical bank positions at each time step, was calculated with the use of the window shade model described above. The calculation was performed on a 12-region axial core. The diffusion code used (Ref. 13) varies the control rod bank position on a pointwise basis at each time step to obtain a desired reactivity. The desired reactivity may be varied over lifetime to include the effects of radial nonuniform burnup or to override any bias present in the calculation.

The result of this calculation, after normalizing the initial hot bank position to the experimental value, is 16.6 Mw-yr.

D. XENON REACTIVITY EFFECTS

This section discusses the effect on core reactivity and rod bank position of the following conditions:

- (1) Initial buildup of Xenon-135.

(2) Xenon buildup after shutdown at equilibrium xenon.

(3) Xenon buildup after startup at time of maximum.

1. Initial Buildup of Xenon 135

Buildup of xenon in the clean PM-1 core was calculated at several times after initial startup by the following equation:

$$X(t) = X_o (1 - e^{-(\lambda_{XE} + \sigma_{XE} \phi) t} + \frac{\lambda_I I_o}{\lambda_{XE} - \lambda_I + \sigma_{XE} \phi} (e^{-(\lambda_{XE} + \sigma_{XE} \phi) t} - e^{-\lambda_I t})) \quad (4)$$

where

$$X_o = \text{Eq Xe} = \frac{(\gamma_I + \gamma_{XE}) \Sigma_f \phi}{\lambda_{XE} + \sigma_{XE} \phi}$$

$$I_o = \text{Eq I} = \frac{\gamma_I \Sigma_f \phi}{\lambda_I}$$

λ = decay constant

γ = fission fraction

σ_{XE} = Xe absorption cross section

ϕ = absolute flux

Σ_f = core fission cross section

t = time

Using Eq (4), xenon concentrations versus time were calculated for each core region in the seven-region radial core configuration. These concentrations were then included with the clean core material concentrations in a one-dimensional diffusion calculation to find their effect on core reactivity.

Since a one-dimensional calculation assumes a uniform flux and material concentration in the perpendicular direction, a method was derived by which the axial effect of rod bank insertion could be evaluated. This method consisted of calculating the equilibrium xenon concentration in each of the 12 axial regions when the Y-rods were inserted to their critical position. These concentrations were then included in an axial window shade model calculation and their effect on reactivity evaluated. The second step consisted of evaluating the effect of uniform

xenon concentration in a window shade calculation. The axial non-uniform effect was then defined as the difference in the two calculations. This effect was calculated to be ~0.48% in reactivity. This effect was then applied proportionally to the radial results.

It was interesting to note that, when a nonuniform radial xenon effect was obtained in the seven-radial region core configuration, it was found to have no effect (i. e., < 0.05%) on core reactivity.

2. Buildup of Xenon After Shutdown

Buildup of Xenon-135 after shutdown at equilibrium xenon was determined as a function of time from the following equation:

$$X(t) = \frac{\lambda_I}{\lambda_{XE} - \lambda_I} I_0 \left(e^{-\lambda_I t} - e^{-\lambda_{XE} t} \right) + X_0 e^{-\lambda_{XE} t} \quad (5)$$

The nomenclature is the same as in Eq (4). The effect on core reactivity was obtained as described above and the axial xenon effect applied proportionally.

3. Xenon Buildup Upon Restarting

Xenon concentration was calculated as a function of time after re-starting. It was assumed that the startup occurred at the time of maximum xenon buildup after shutdown. The following equation was used:

$$X(t) = X_{\max} e^{-At} + \frac{\lambda_I I_{\max}}{A - \lambda_I} \left(e^{-\lambda_I t} - e^{-At} \right) + Xe(t) \quad (6)$$

where

$$A = \lambda_{XE} + \sigma_{XE} \phi$$

$$X_{\max} = \text{Xe concentration at time of maximum Xe}$$

$$I_{\max} = \text{I concentration at time of maximum Xe}$$

$$Xe(t) = \text{initial buildup of Xe after startup.}$$



ILLUSTRATIONS

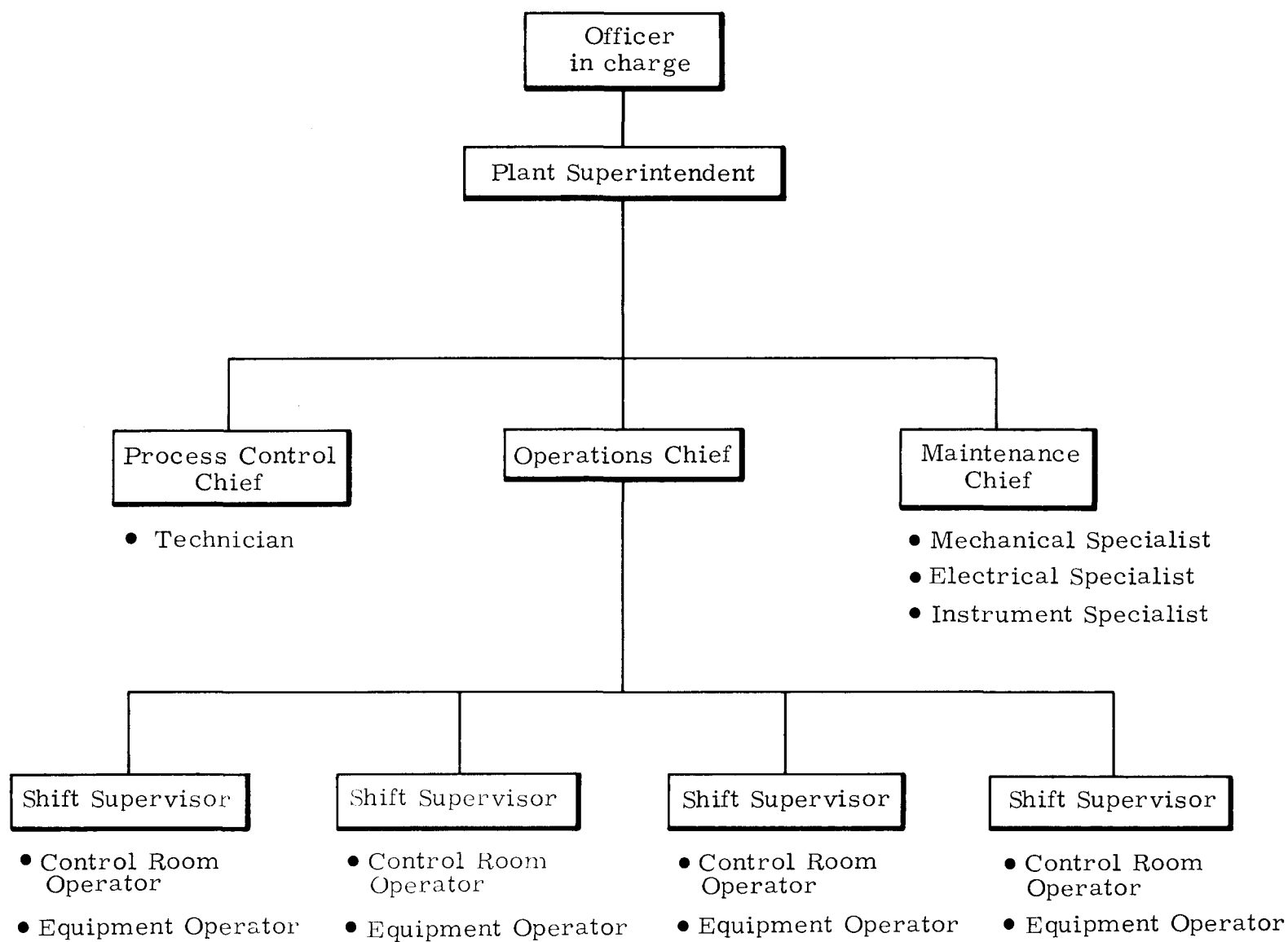


Fig. 1. Air Force Crew Organization (as of Nov 1, 1962)

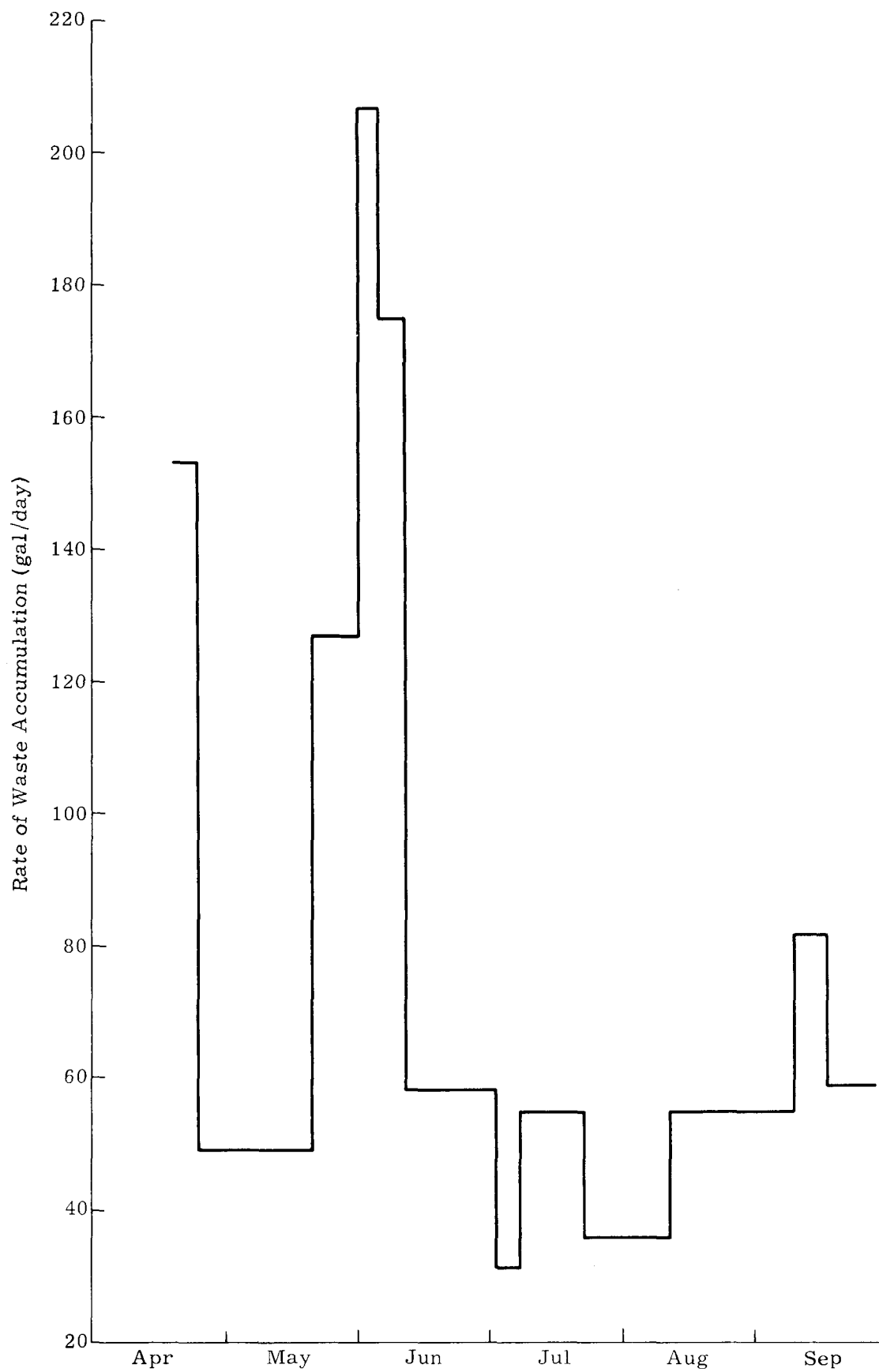


Fig. 2. Liquid Radioactive Waste Accumulation

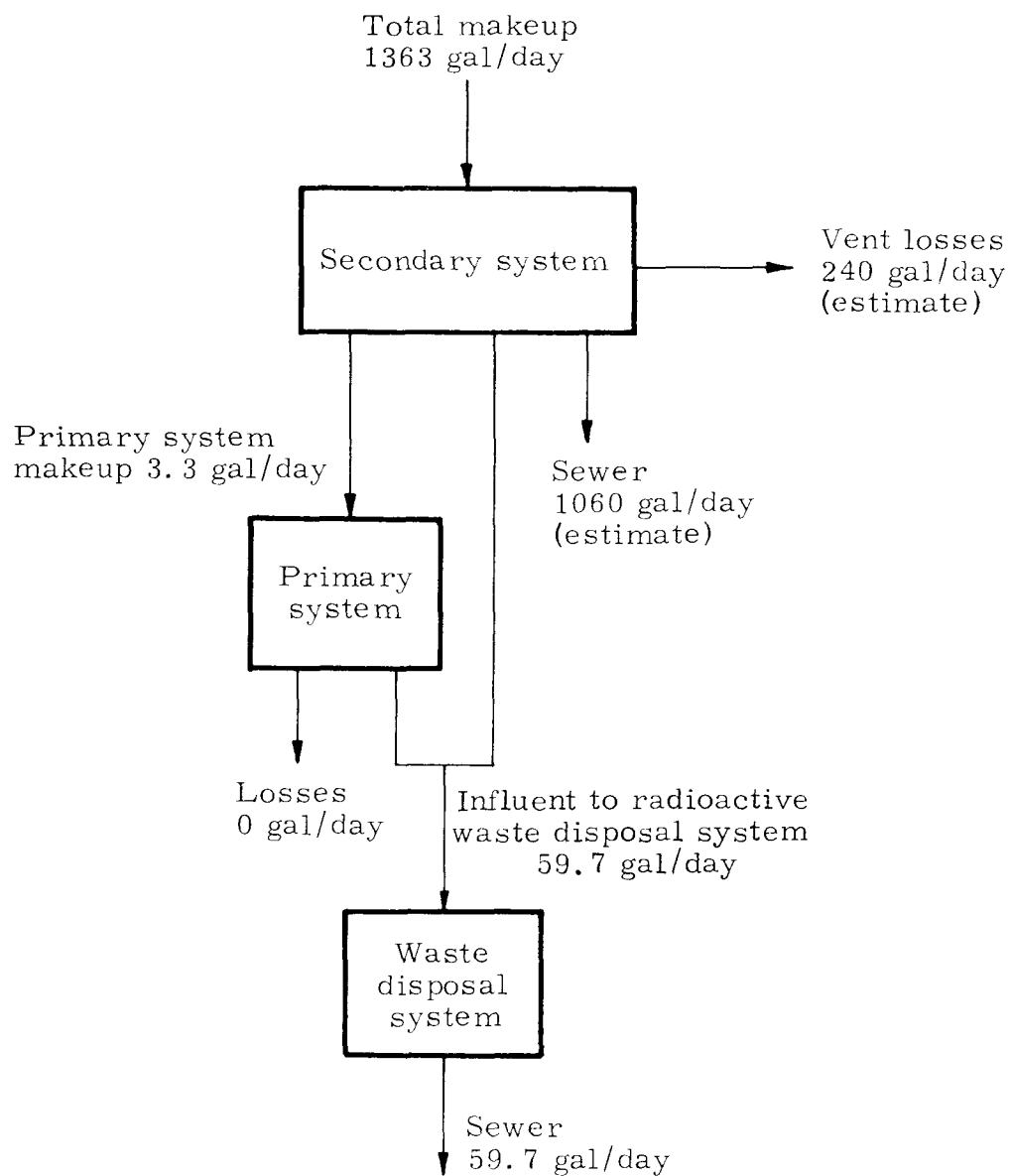


Fig. 3. Plant Liquid Balance--Based on Data Collected During 400-Hour Run

Fig. 4. Plant Operating History

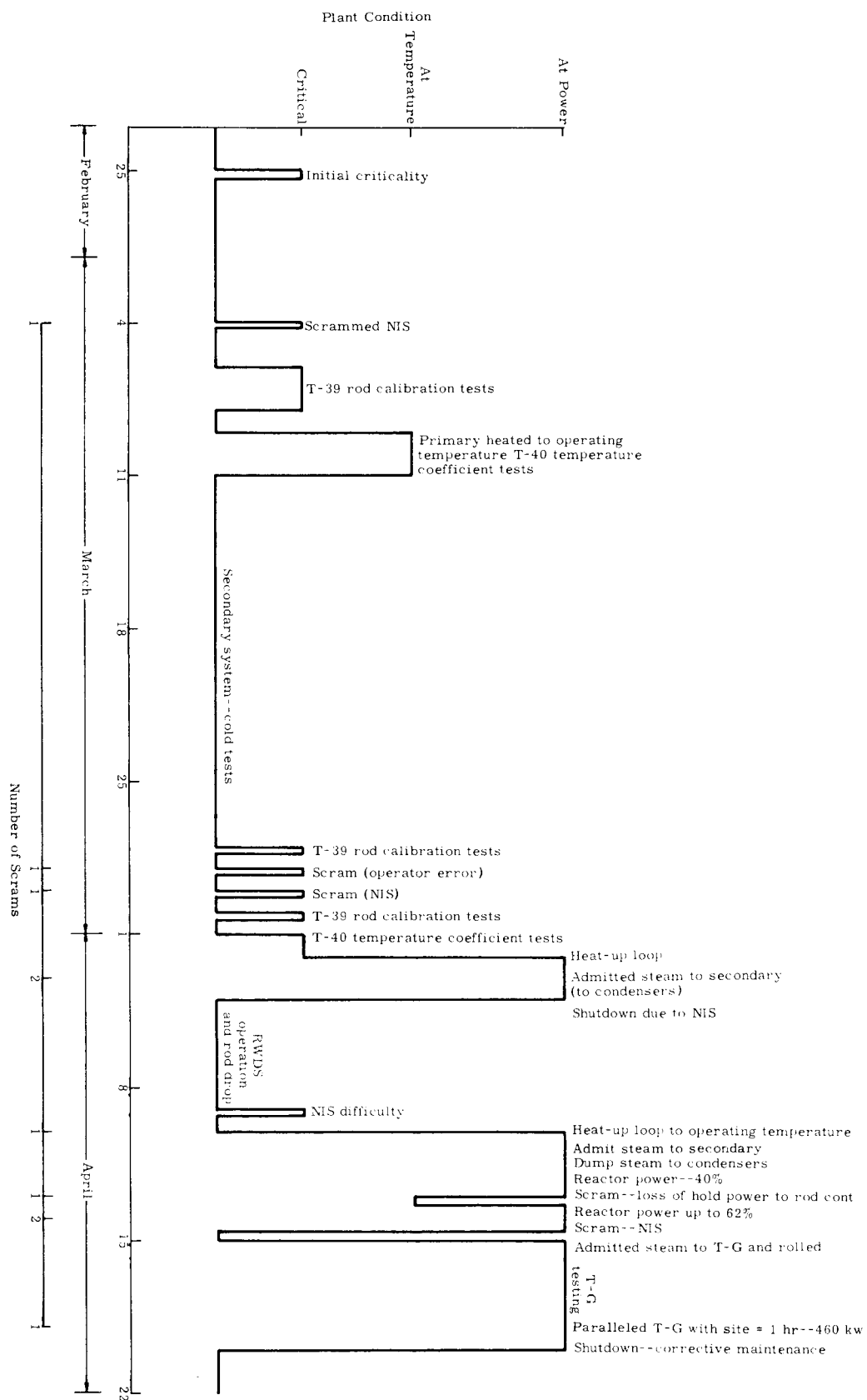
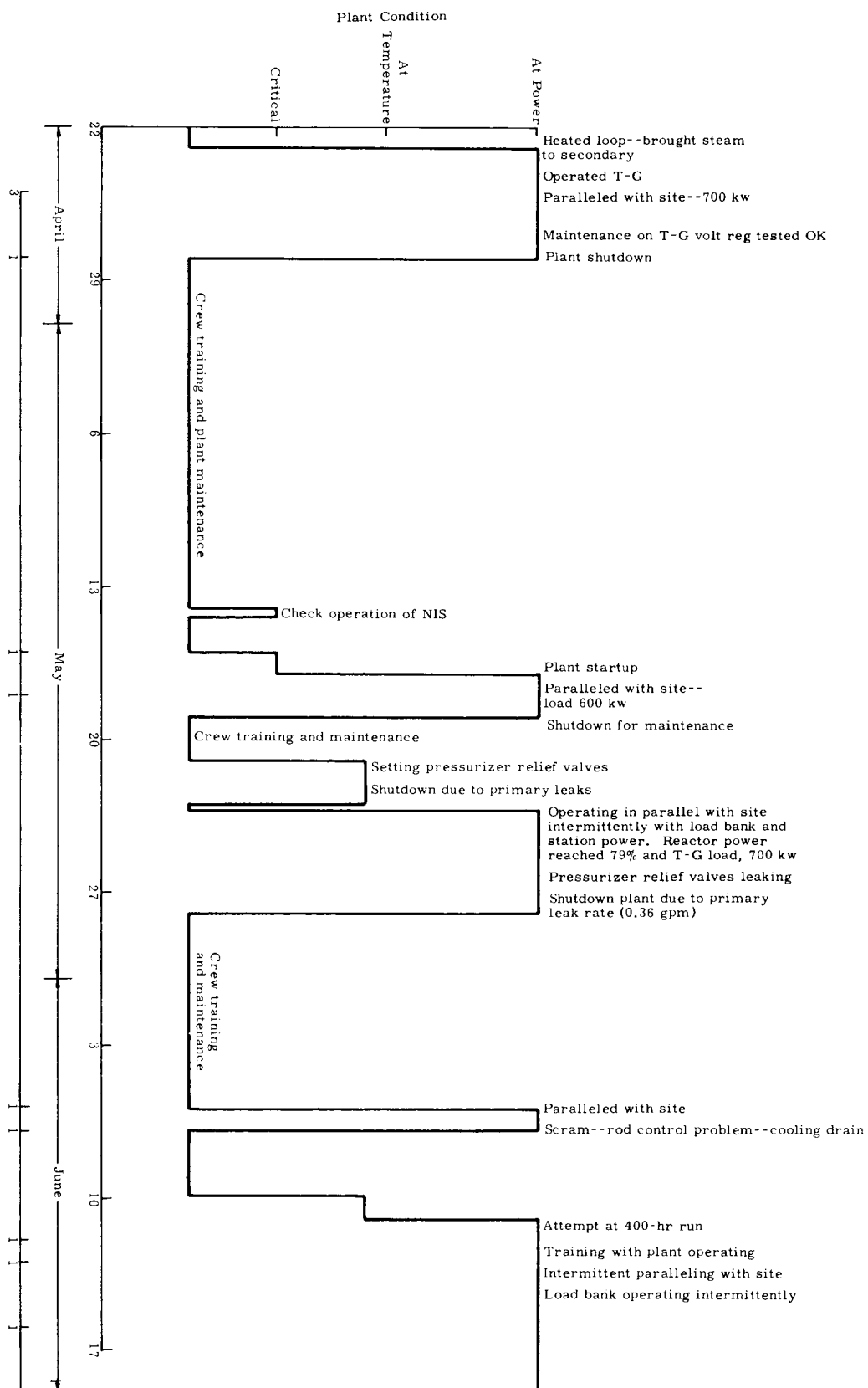


Fig. 4. (continued)



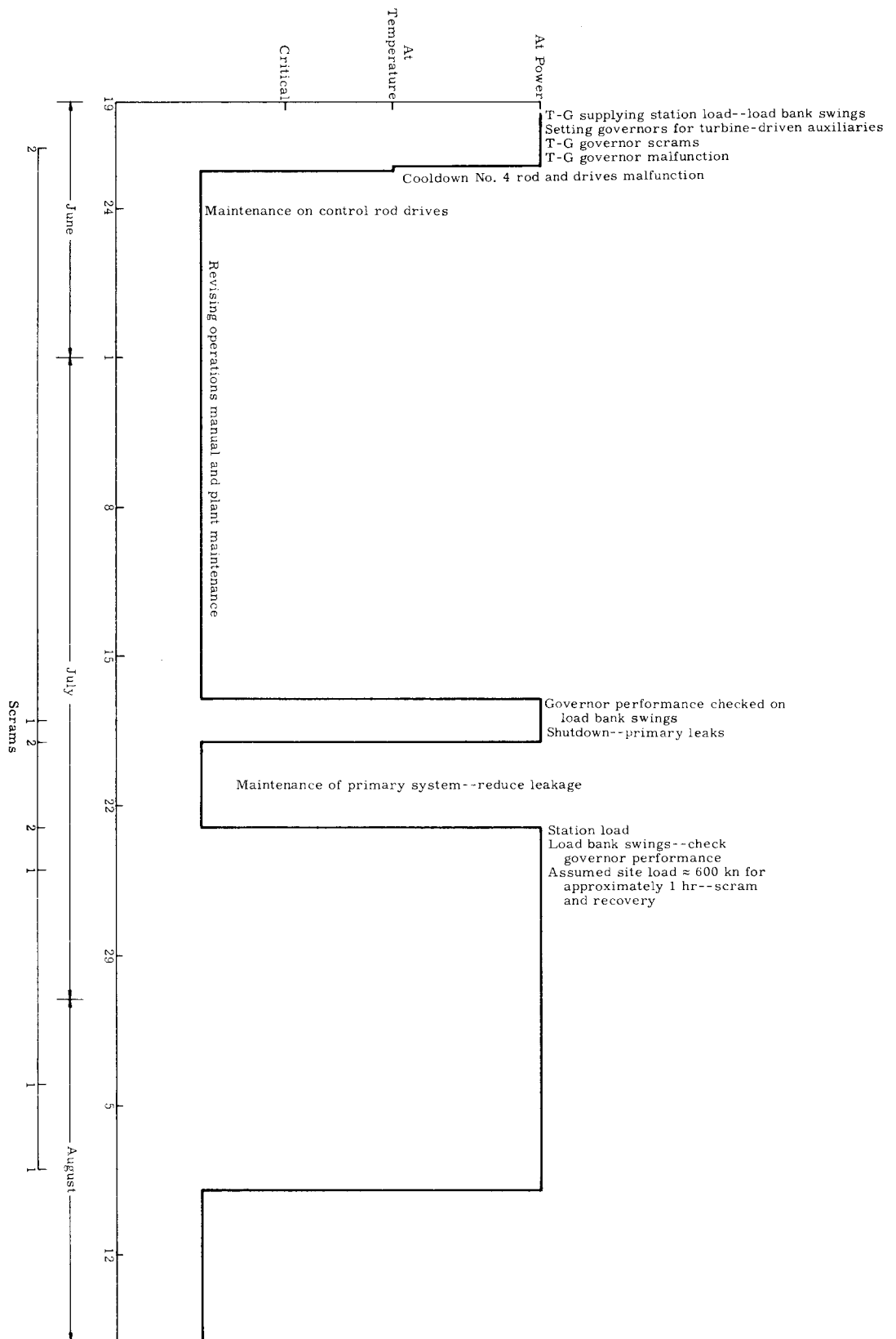


Fig. 4. (continued)

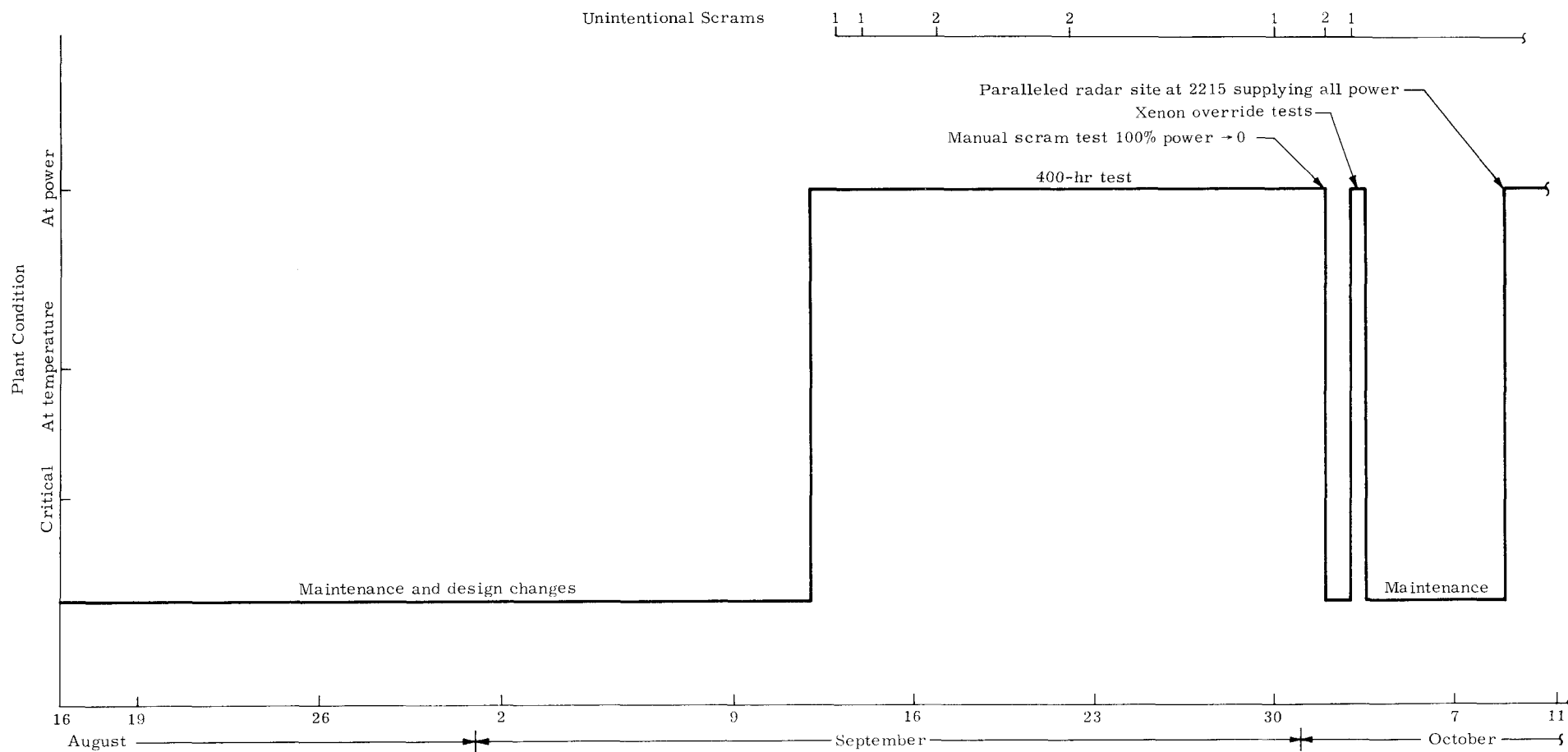


Fig. 4. (continued)

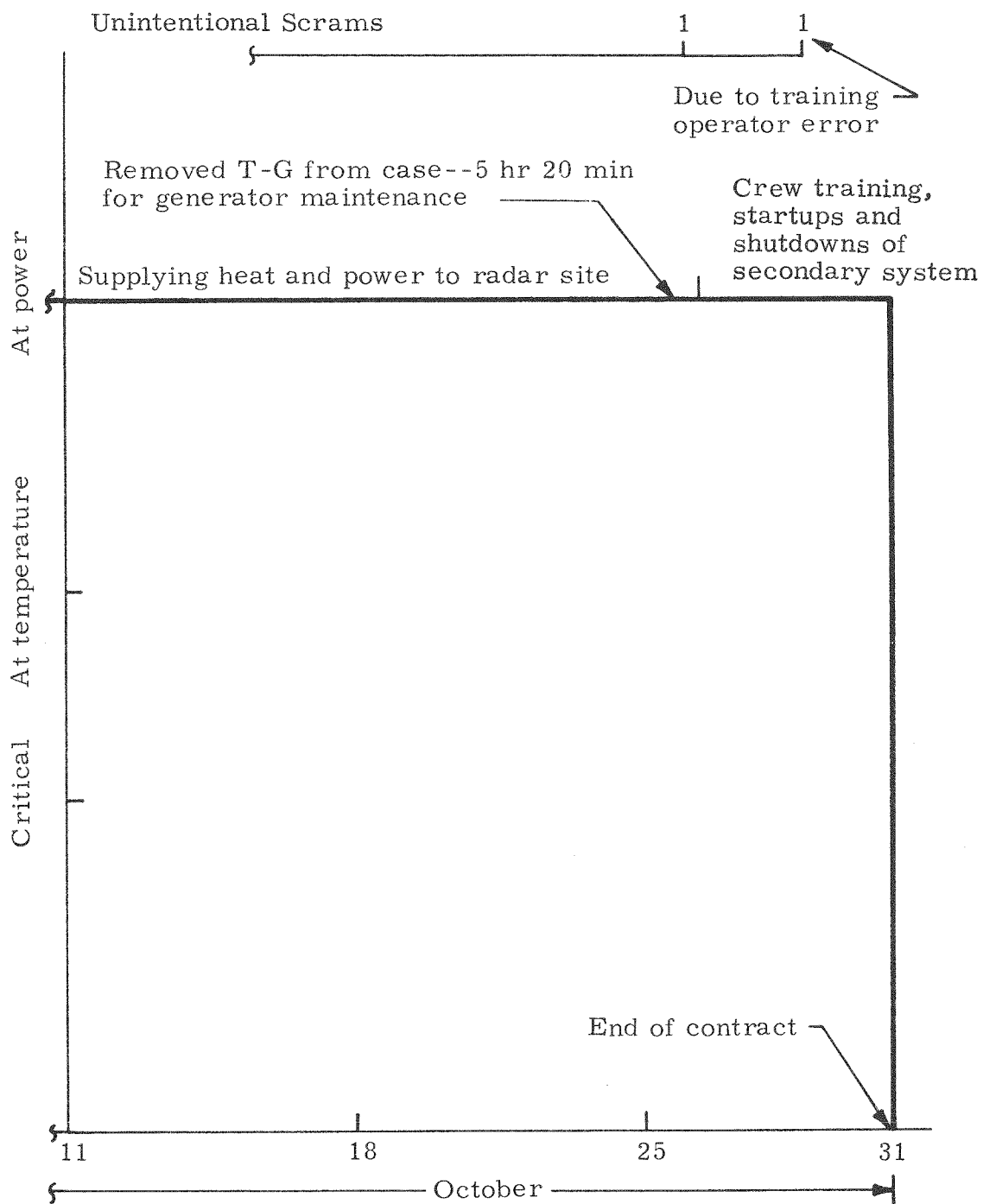


Fig. 4. (continued)

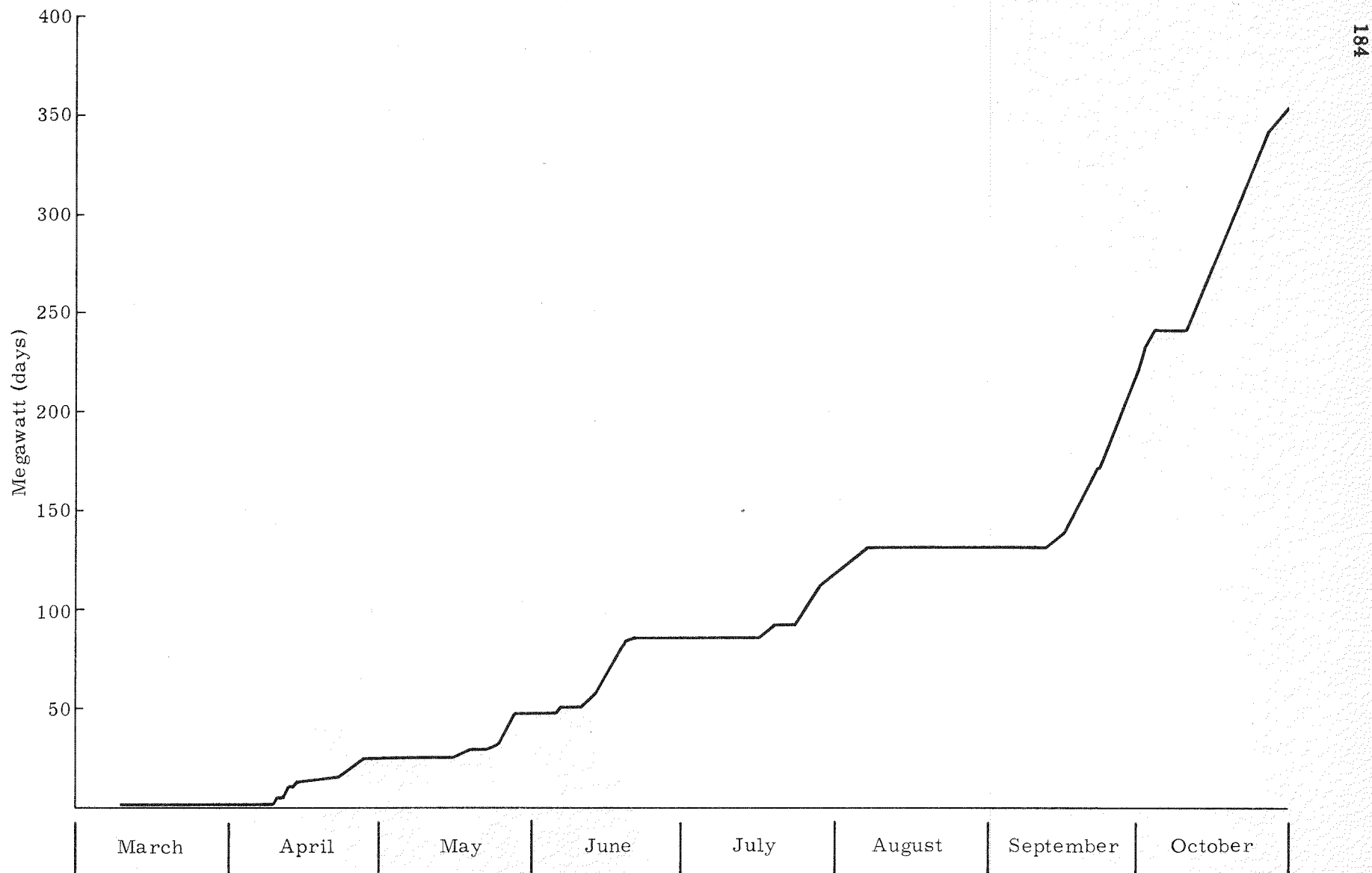


Fig. 5. Integrated Core Energy Output

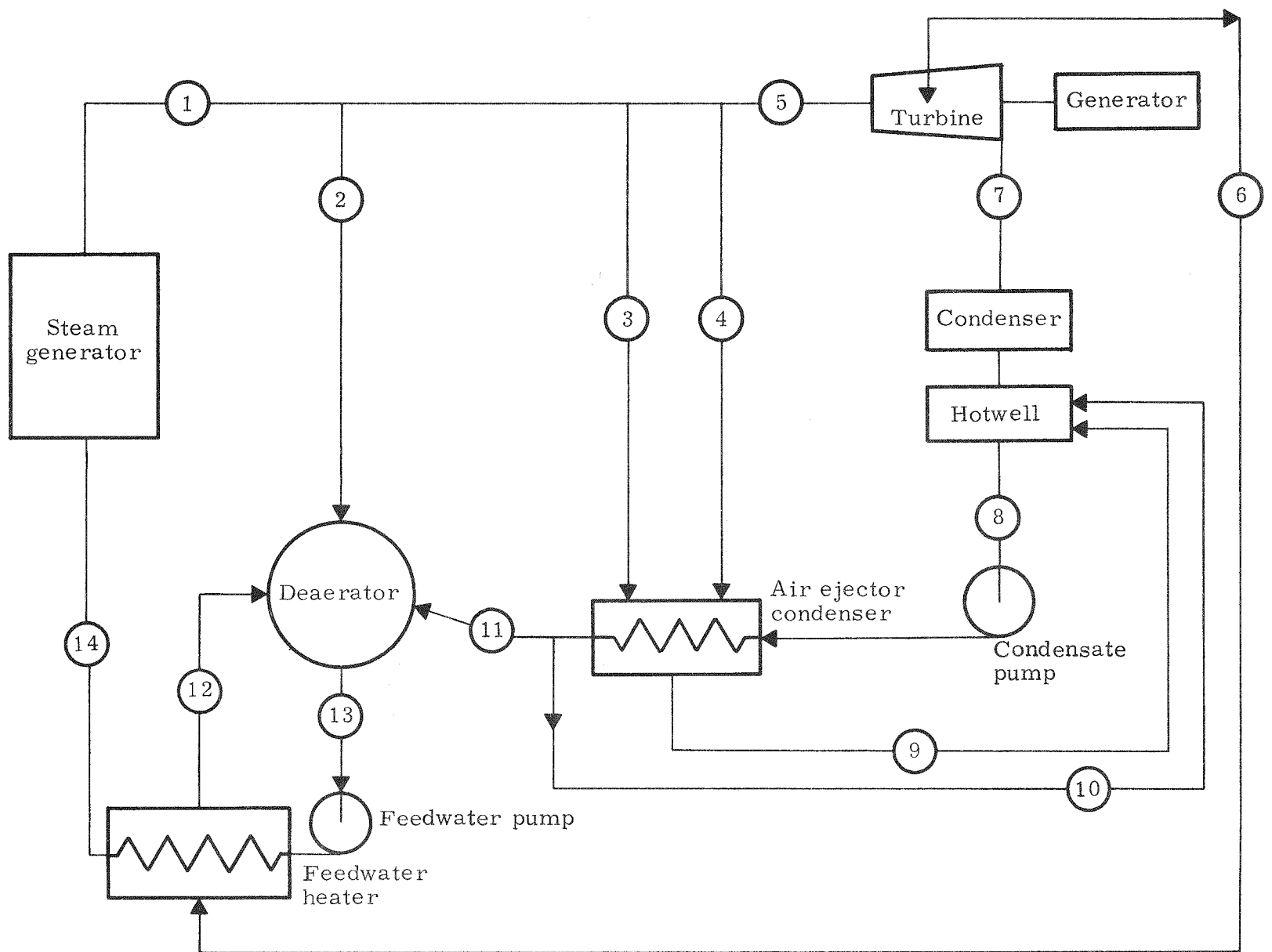


Fig. 6. Secondary System Heat Balance

Item No.	Description	Design Predictions				Actual Measurements																			
		1270 kw				1270 kw				1080 kw				857 kw				675 kw				440 kw			
		psia	°F	lb/hr	Btu/lb	psia	°F	lb/hr	Btu/lb	psia	°F	lb/hr	Btu/lb	psia	°F	lb/hr	Btu/lb	psia	°F	lb/hr	Btu/lb	psia	°F	lb/hr	Btu/lb
1.	Steam generator output	337	428	26,800	1203.7	332	425	28,500	1203.6	347	430	24,250	1203.8	364	433	20,250	1,204.1	372	430	16,200	1204.2	392	455	12,000	1204.4
2.	Steam to deaerator	22.5	325	1,212	1203.7																				
3.	Steam to air ejector	112	360	450	1203.7																				
4.	Gland steam to air ejector	12	320	223	1203.7																				
5.	Turbine throttle steam	325	425	24,915	1203.7																				
6.	Extraction steam	50	281	2,470	1142.0			2,540	1150			2,000				1,550				1,200		No extraction			
7.	Turbine exhaust steam	11 in. Hg abs	165	22,445	1001.5	10 in. Hg abs	157			10.1 in. Hg abs	157			10.2 in. Hg abs	156			10.8 in. Hg abs	155			10.1 in. Hg abs	151		
8.	Condensate from hotwell	9 in. Hg abs	157	27,400	125.0	8 in. Hg abs	148		115.9	8.2 in. Hg abs	150		117.9	8.5 in. Hg abs	152		119.9	8.6 in. Hg abs	154		121.9	9 in. Hg abs	154		121.9
9.	SJAE drains	11.5	200.1	673	168.0																				
10.	Condensate pump re-circulation		182.8	4,282	150.4																				
11.	Condensate to deaerator		182.8	23,118	150.4		166		134		168		135.9		169		136.9		172		139.9		175		142.9
12.	Feed heater drains	85.0	316.2	2,470	286.4	80.0	312	2,540	282	68.0	301	2,000	270.6	56.0	288.2	1,550	257.5	46.0	275.8	1,200	244.7	No extraction			
13.	Feedwater from deaerator	22.5	233.7	26,800	202.5	21.0	230	28,100	198.8	22.5	232	24,000	200.25	22.4	232	19,700	200.25	22.8	233	16,100	201.2	23.0	232	11,700	200.2
14.	Feedwater to steam generator		311.5	26,800	281.4		310	28,100	279.9		300	24,000	269.6		285	19,700	254.0		277	16,100	243		235	11,700	203.3
	Required core power	7.24 Mw				7.6 Mw				6.58 Mw				5.50 Mw				4.53 Mw				3.43 Mw			
	Net cycle efficiency	13.60%				13.0%				12.0%				10.9%				9.24%				6.6%			
	Gross cycle efficiency	17.58%				16.75%				16.19%				15.60%				14.9%				12.81%			
	Net power	985 kw				985 kw				793 kw				601 kw				418 kw				227 kw			
	Auxiliary power	285 kw				285 kw				287 kw				256 kw				257 kw				213 kw			

Fig. 6. Secondary Plant Heat Balance Summary

Description	Actual Measurements						Design Prediction	
	440 kwe gross		857 kwe gross		1270 kwe gross		1270 kwe gross	
	Flow (lb/hr)	Temperature (°F)	Flow (lb/hr)	Temperature (°F)	Flow (lb/hr)	Temperature (°F)	Flow (lb/hr)	Temperature (°F)
1. Reactor inlet		452		449.1		443	870,975	449.8
2. Reactor outlet		472		473.6		476	870,975	476.2
3. Supply to economizer shell		453		449.1		443	1,000	446.7
4. Economizer shell outlet		245		241		235	1,000	203
5. Cooler outlet (economizer tube inlet)		110		103		100	1,000	120
6. Purification return (economizer tube outlet)							1,000	370.2
7. Air blast cooler inlet	36,000	110	36,000	107	36,000	104	37,170	120
8. Air blast cooler outlet	36,000	97	36,000	92.5	36,000	91	37,170	106
9. Shield water demineralizer supply	920	97	920	92.5	920	91	1,000	106
10. Shield water to spent fuel tank	5,740	97	5,740	92.5	5,740	91	7,410	106
11. Shield water from spent fuel tank	5,740	94	5,740	92	5,740	88	7,410	106
12. Shield water to waste disposal evaporator	6,040	97	6,040	92.5	6,040	91	4,930	106
13. Shield water from waste disposal evaporator	6,040	92	6,040	88	6,040	86	4,930	106
14. Shield water to purification system	13,000	97	13,000	92.5	13,000	91	13,830	106
15. Shield water from purification system	13,000	100	13,000	97	13,000	94	13,830	112
16. Shield water to coolant pump	10,300	97	10,300	92.5	10,300	91	10,000	106
17. Shield water from coolant pump	10,300	102	10,300	100	10,300	97	10,000	112
18. Shield water return	36,000	106	36,000	105	36,000	103	37,170	110.1

Fig. 7. Primary System Heat Balance Summary

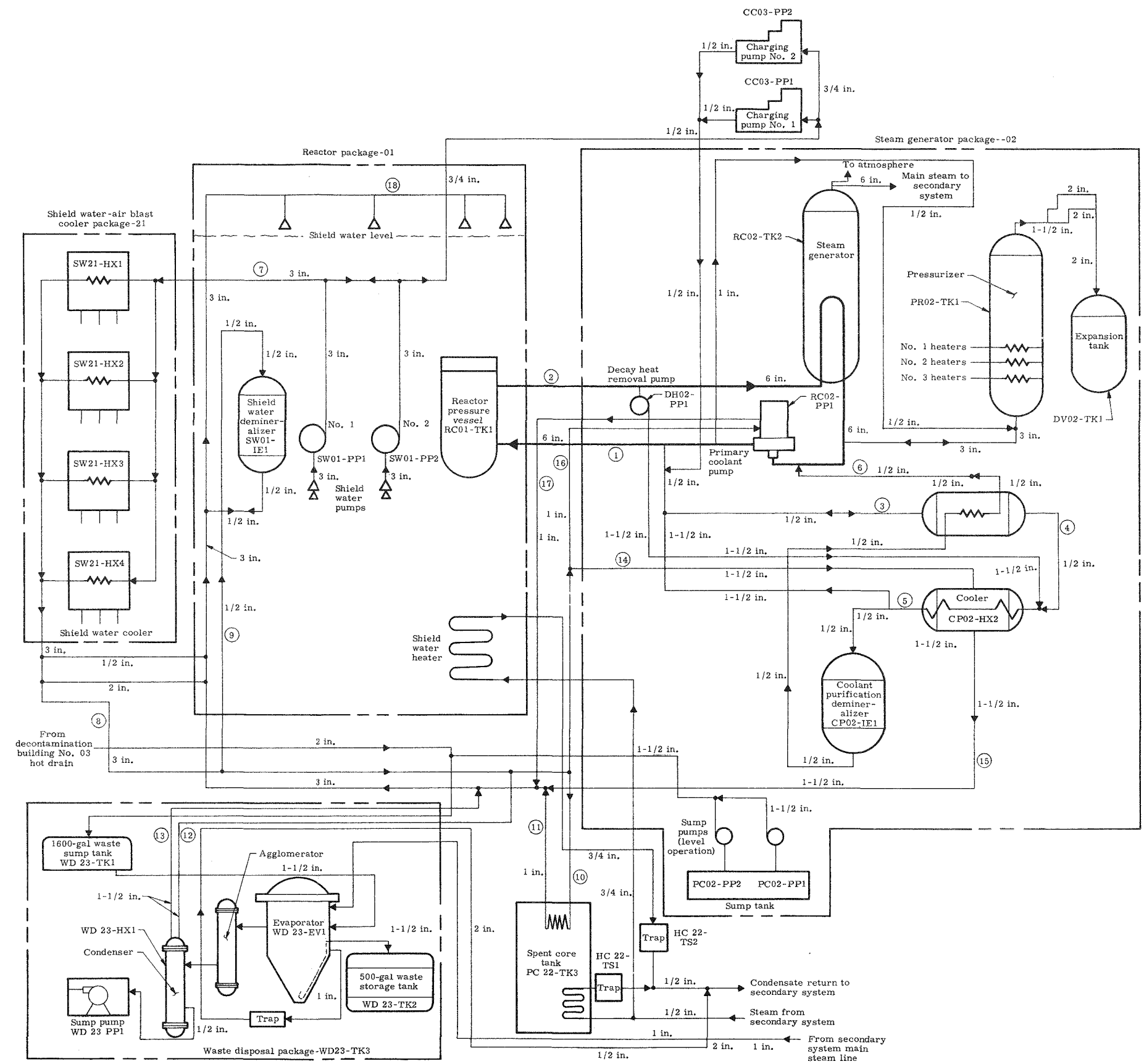


Fig. 7. Primary System Heat Balance

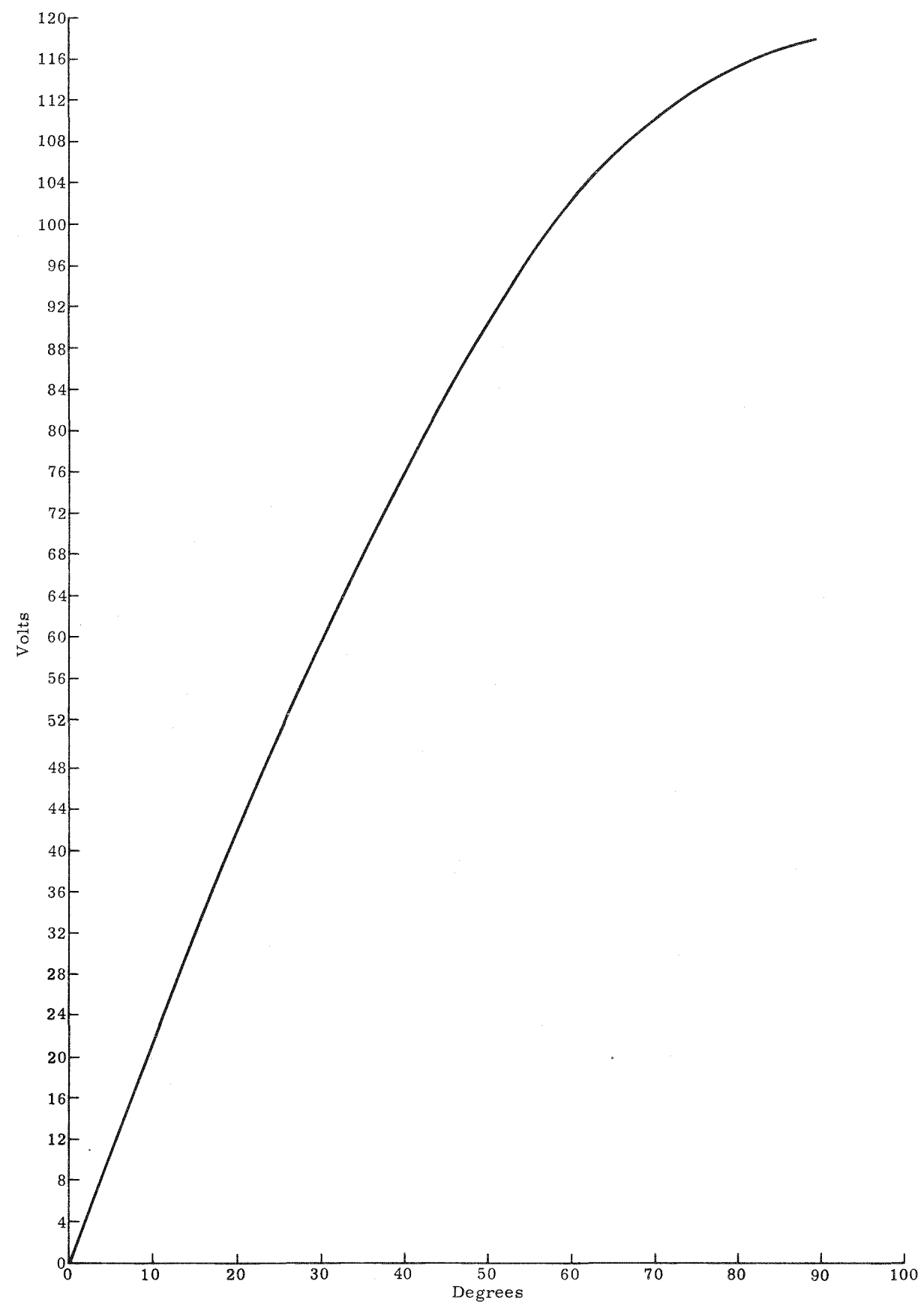


Fig. 8. PM-1 Output Voltage Fundamental Plus Harmonics
Through the 13th

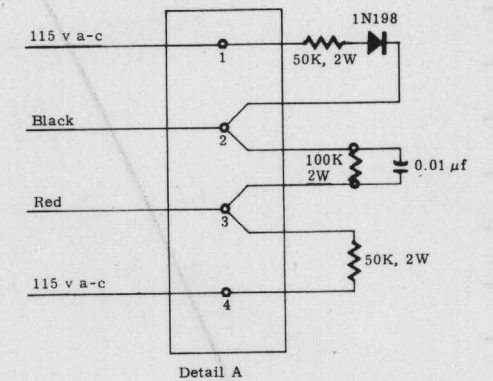
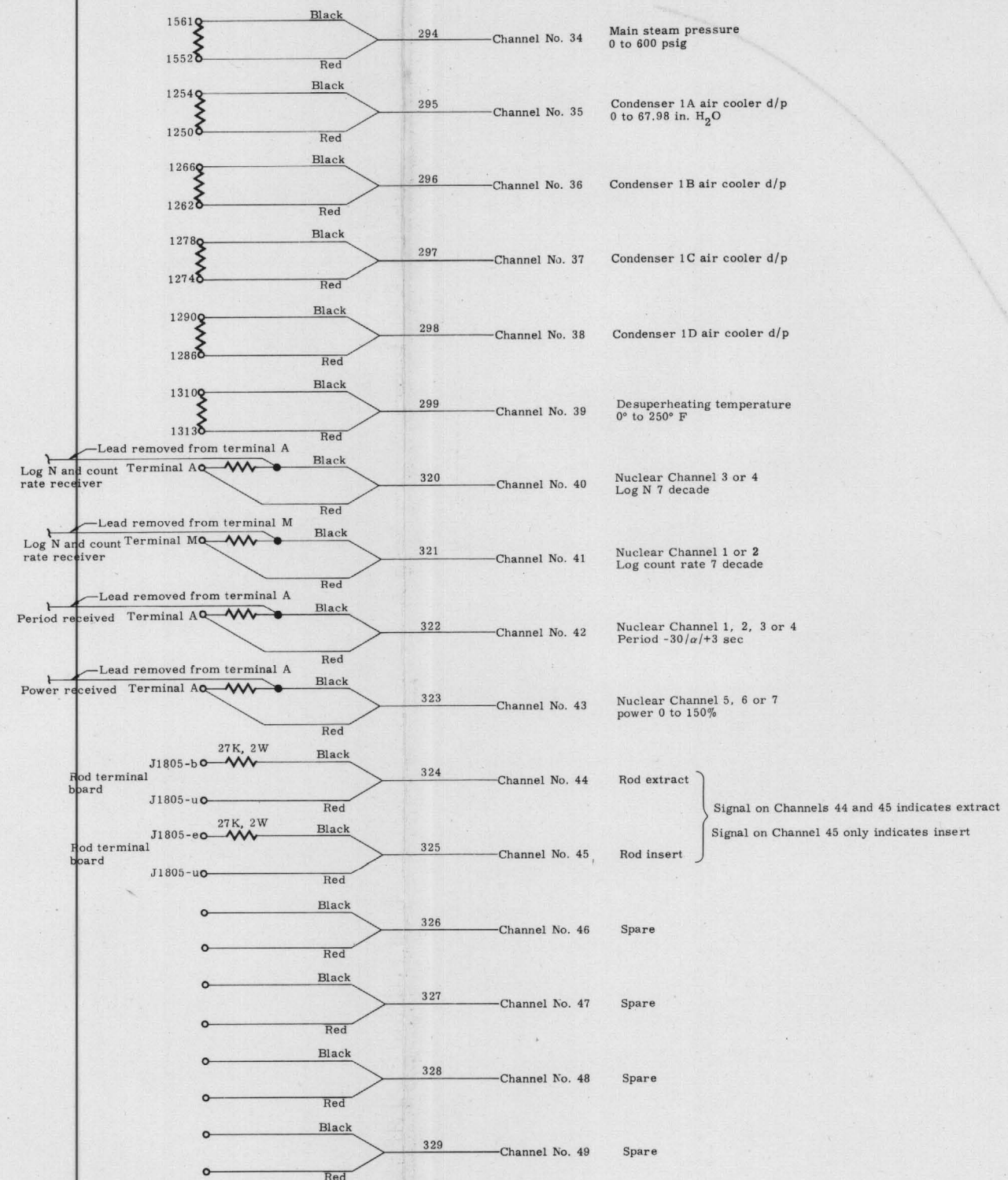
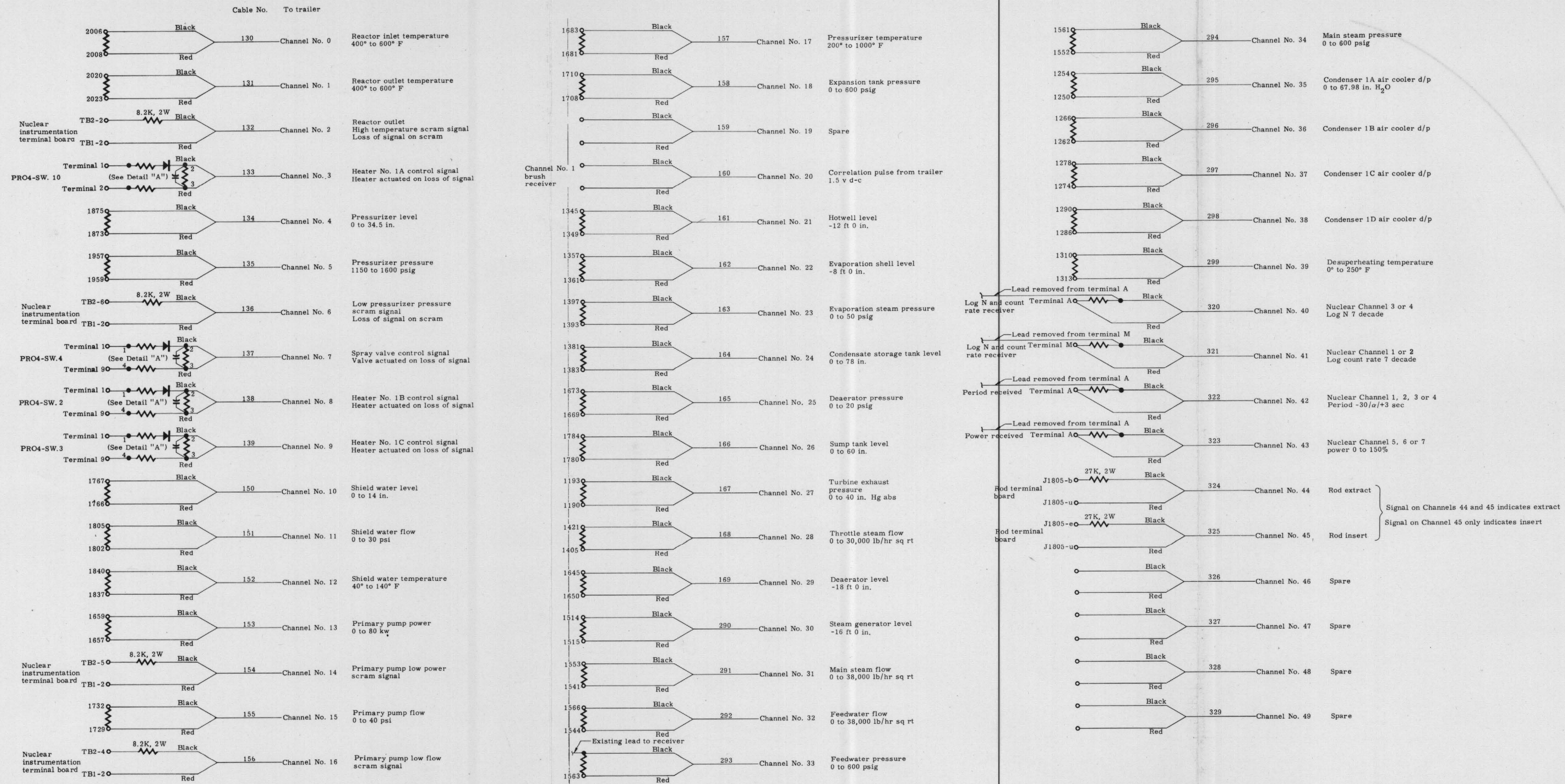


Fig. 9. PM-1 Plant Performance Test Instrumentation Hookup

Fig. 10. PM-1 Plant Performance Test Instrumentation Hookup

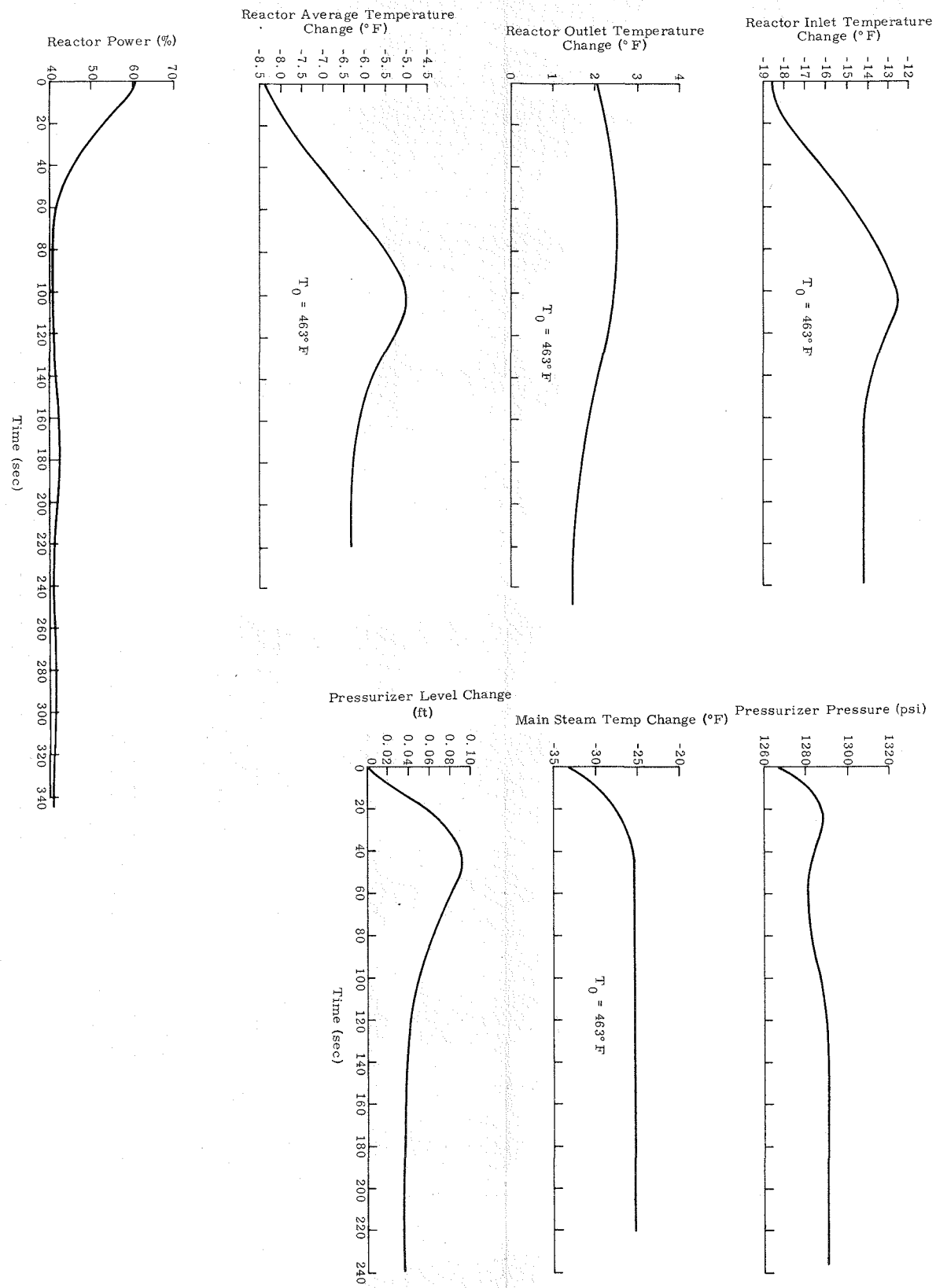
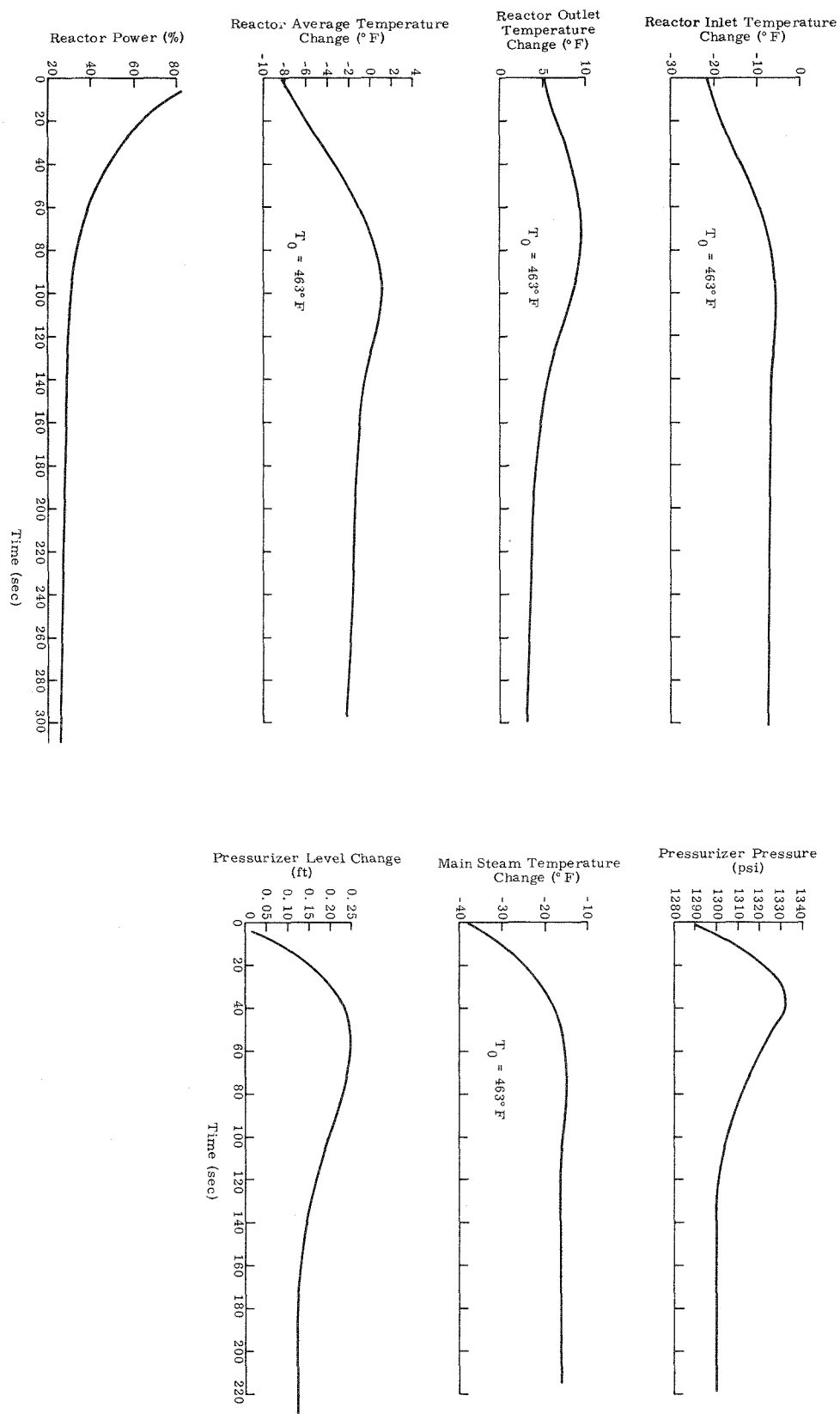
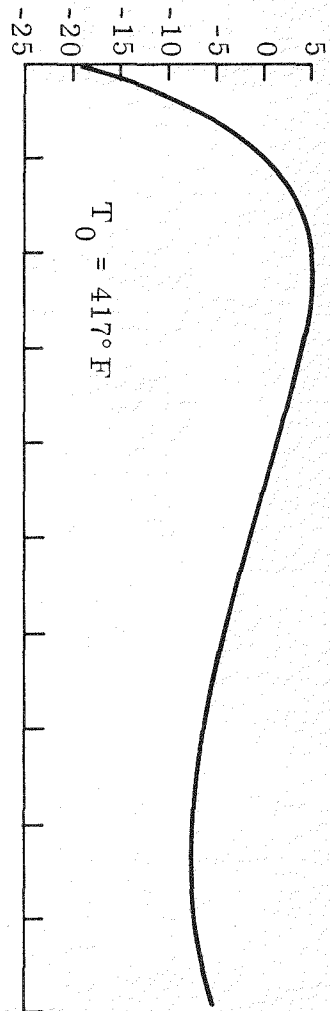
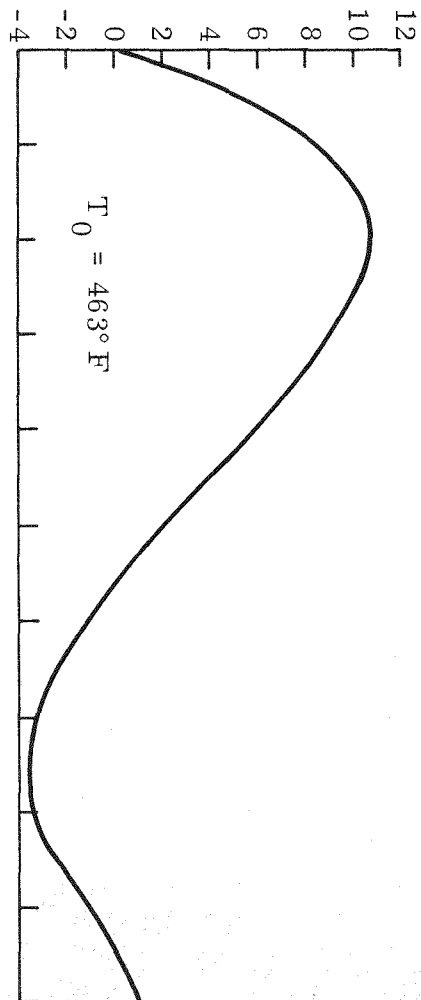


Fig. 11. 60% to 40% Load Transient--Plant

Fig. 12. 85.3 to 27.3 Load Transient--Plant



Main Steam Temperature
Change ($^{\circ}\text{F}$)Reactor Average Temperature
Change ($^{\circ}\text{F}$)

Reactor Power (%)

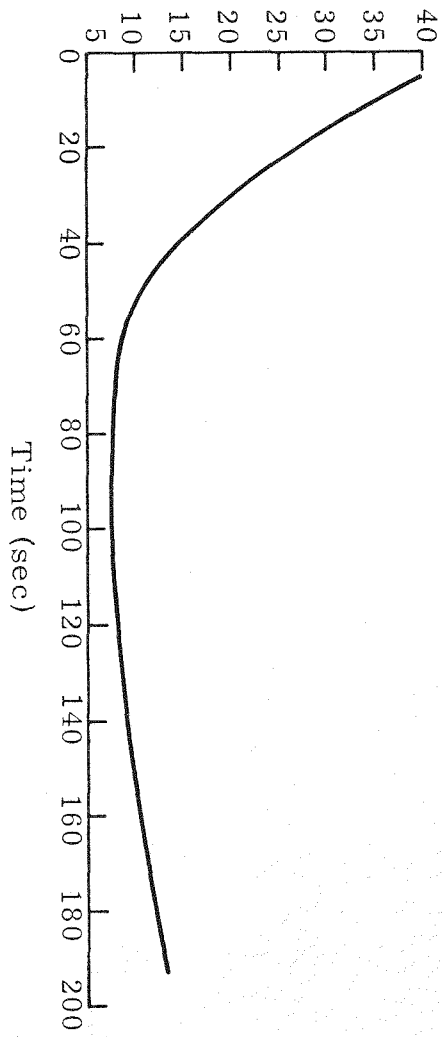


Fig. 13. 40% to 10% Load Transient (predicted)

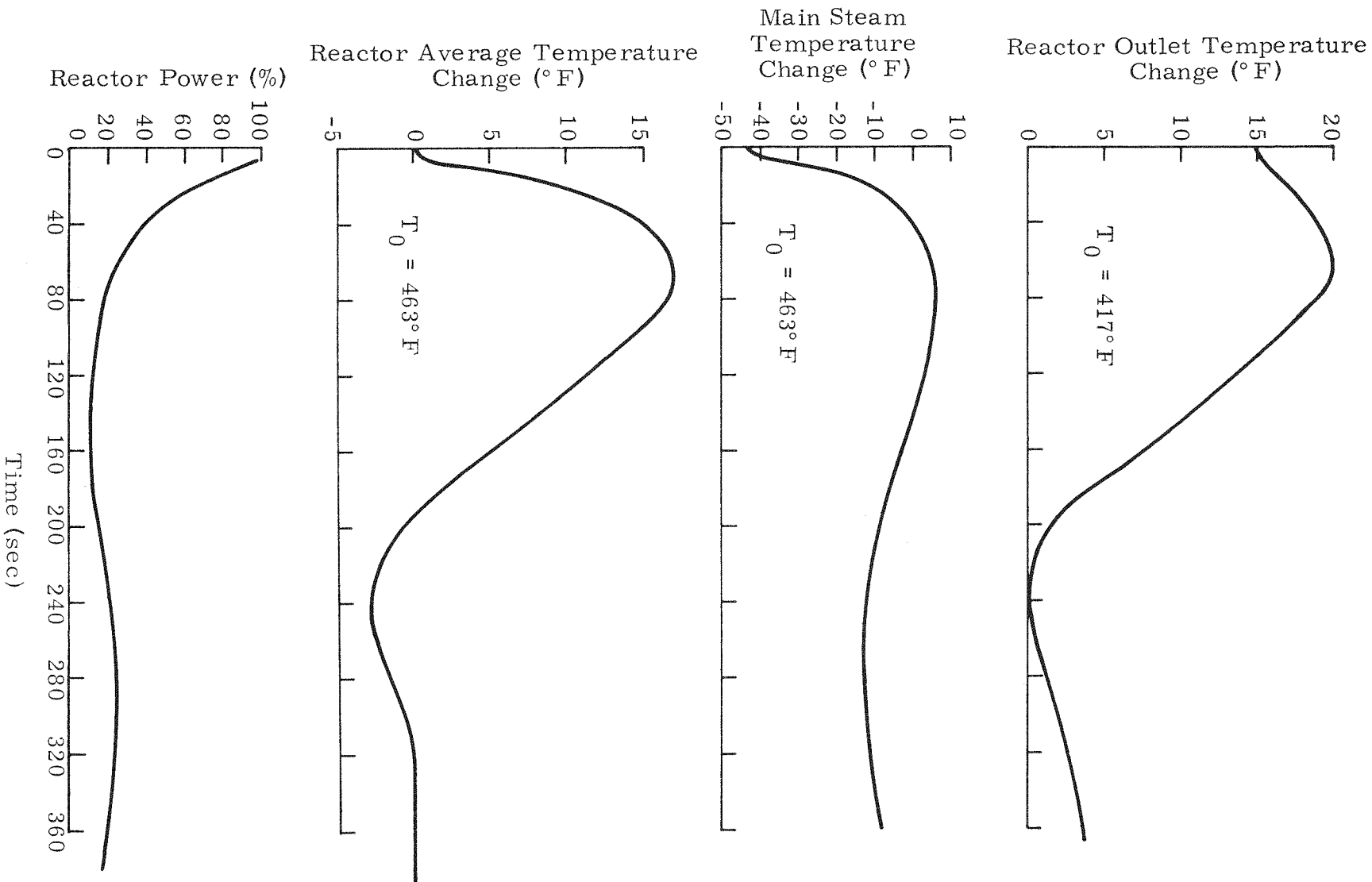


Fig. 14. 100% to 20% Load Transient (predicted)

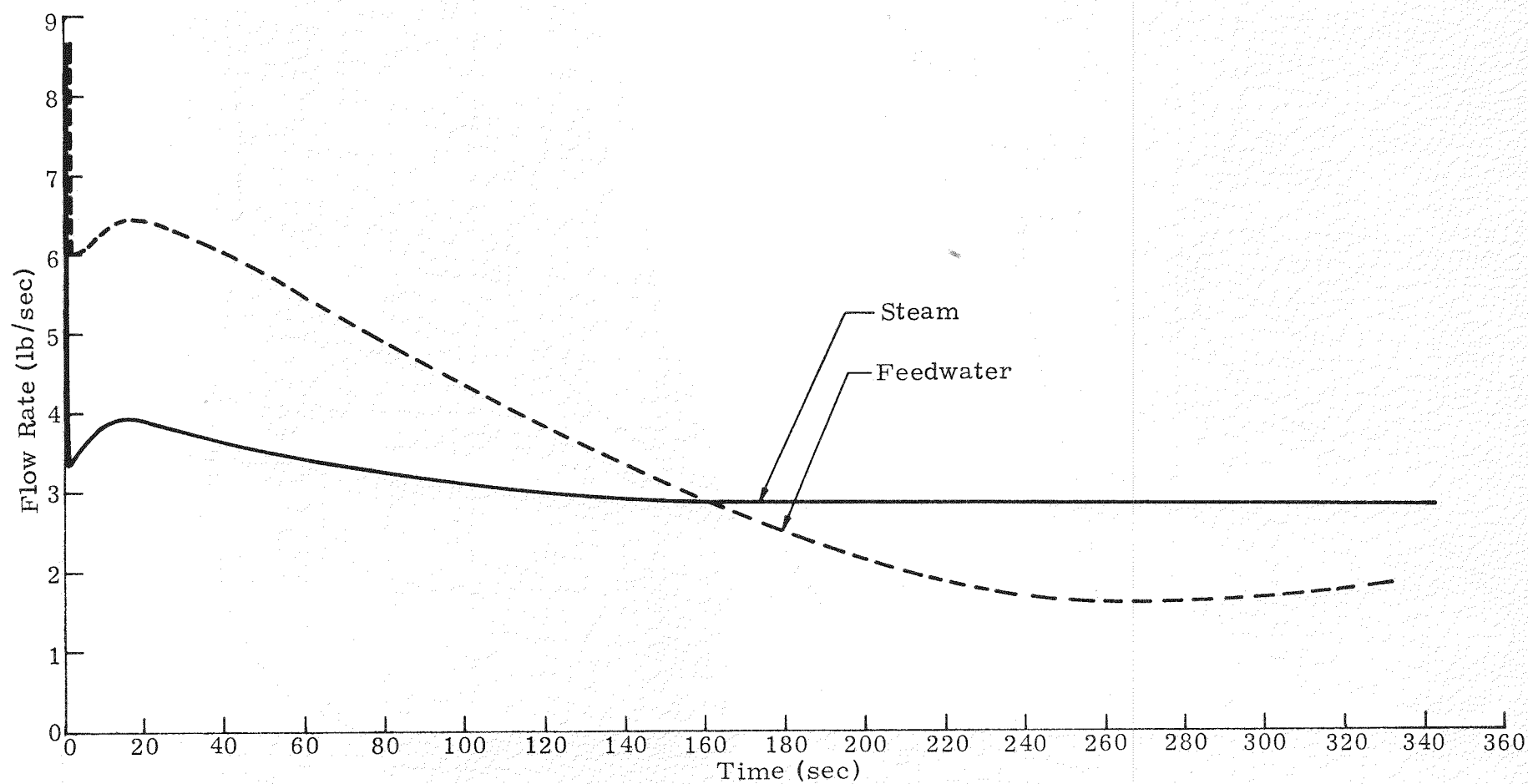


Fig. 15. Feedwater Flow and Steam Flow During 85.3% to 27.3% Load Transient--Plant

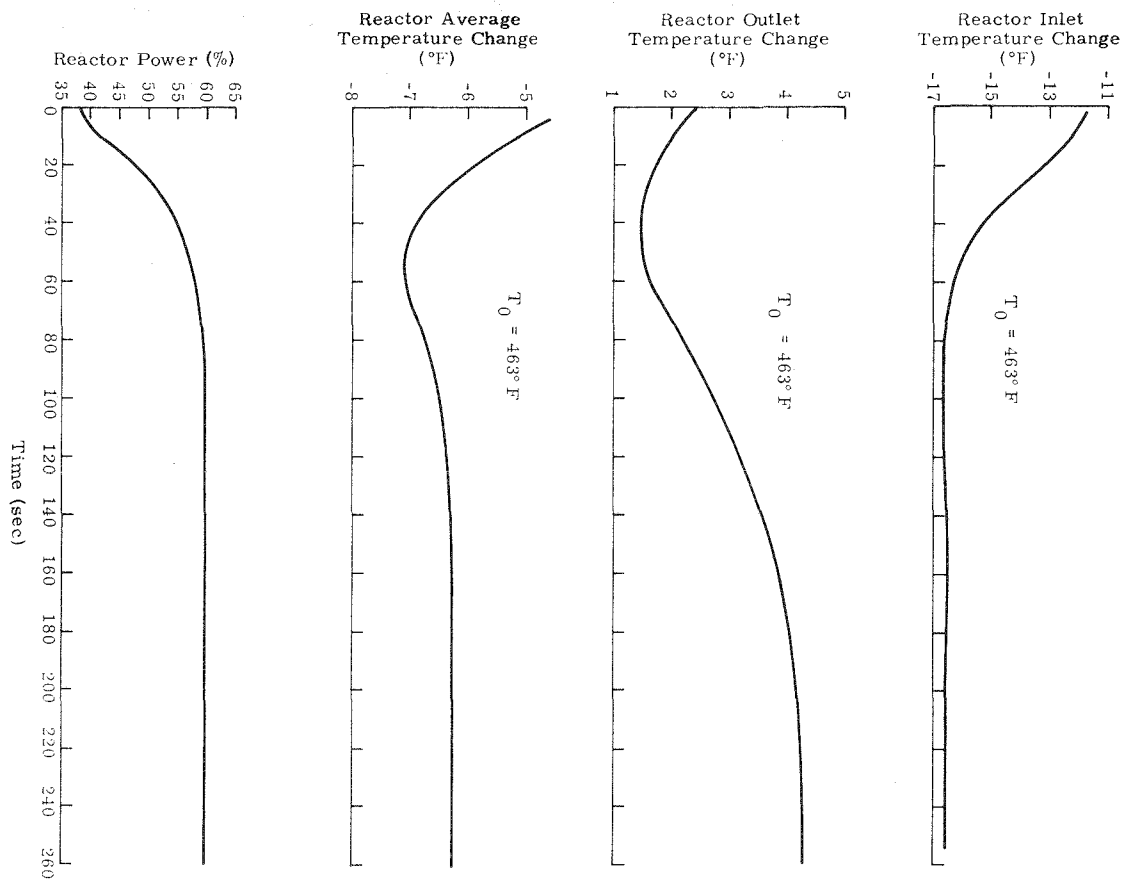
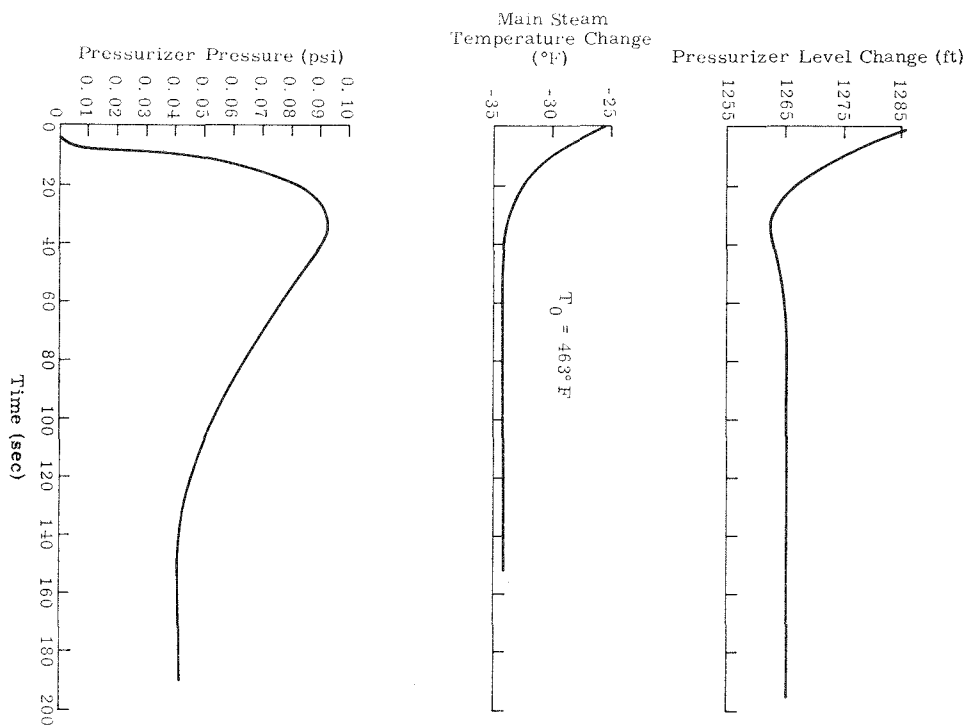


Fig. 16. 38.5 to 58.5 Load Transient--Plant



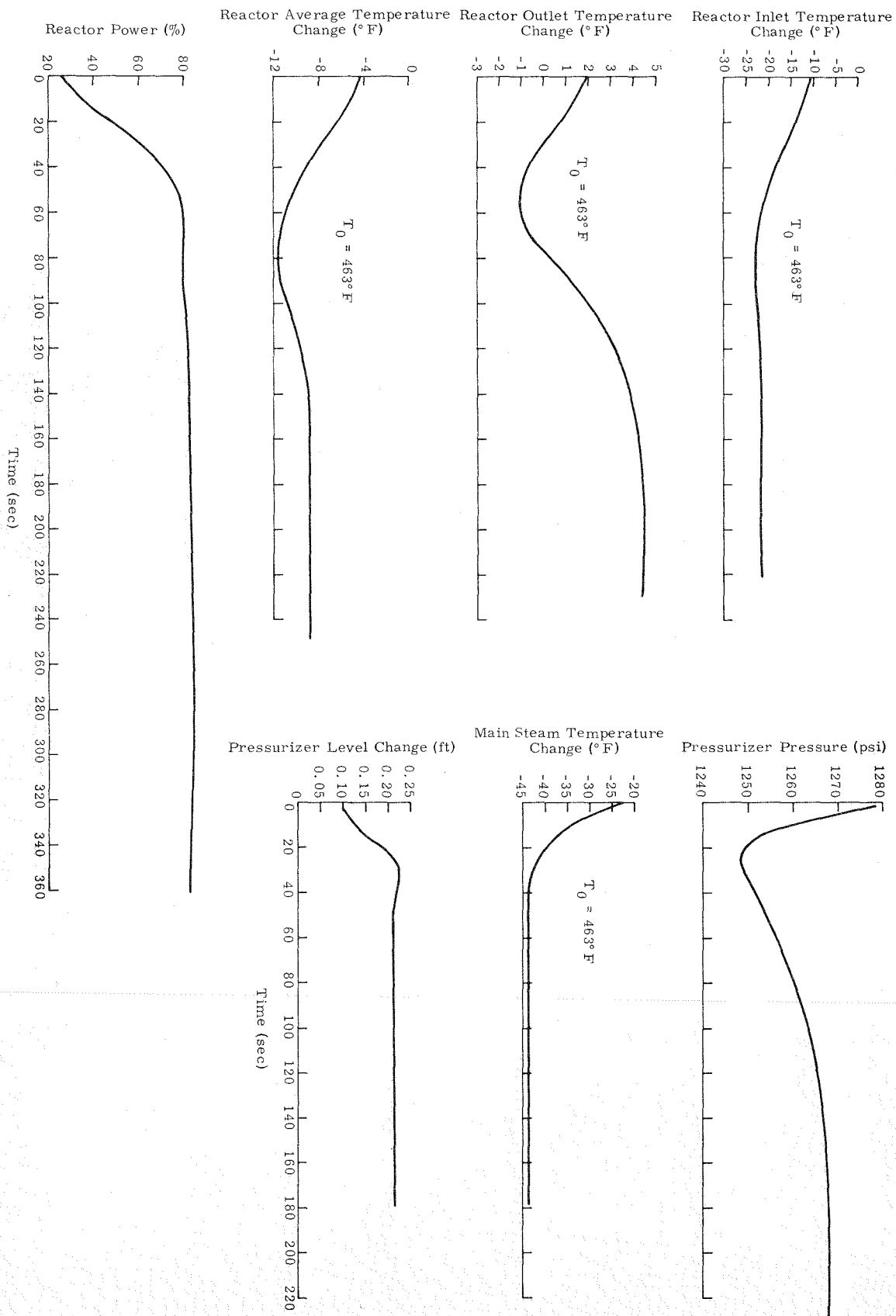


Fig. 17. 22% to 82% Load Transient--Plant

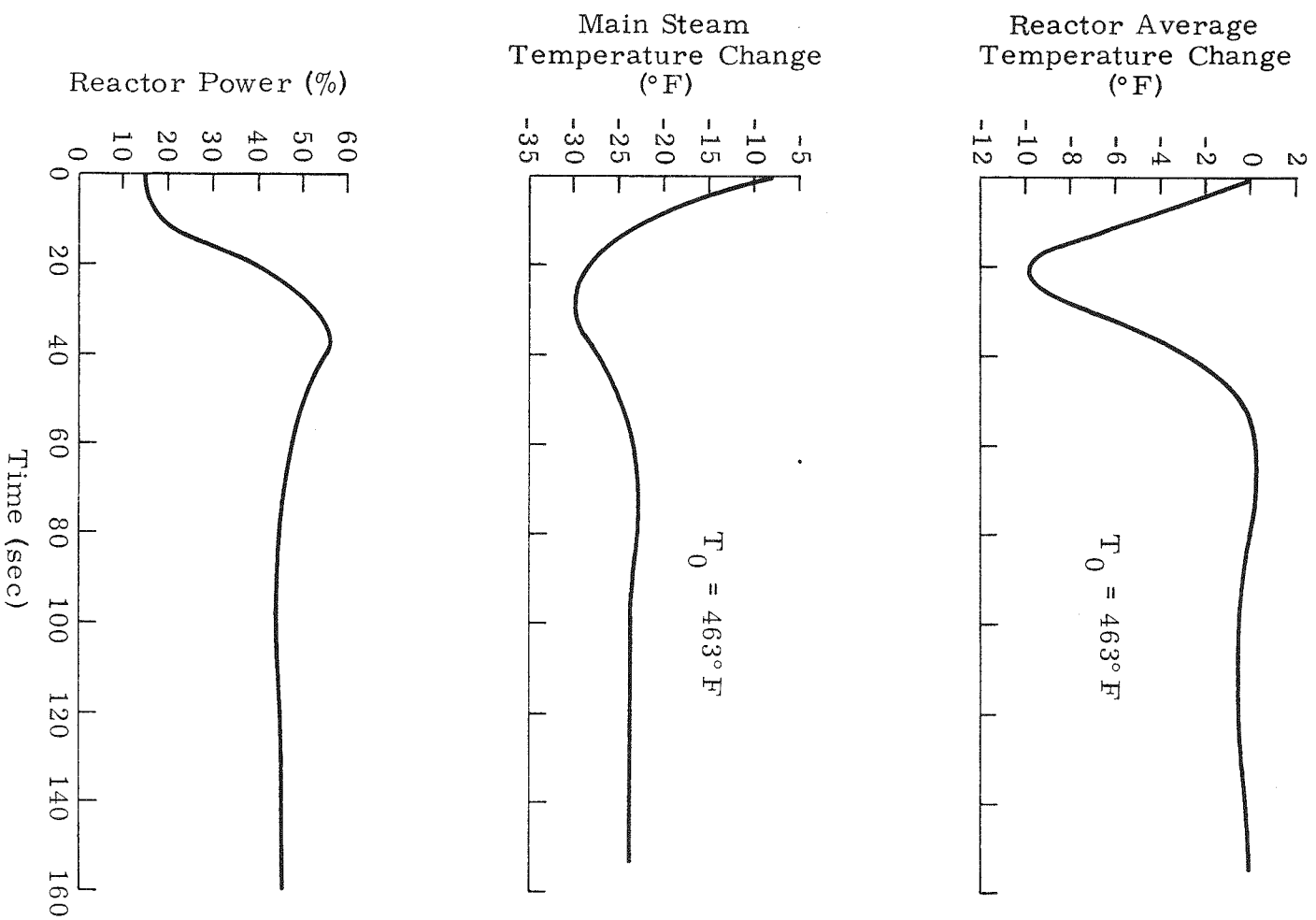


Fig. 18. 15% to 45% Load Transient (predicted)

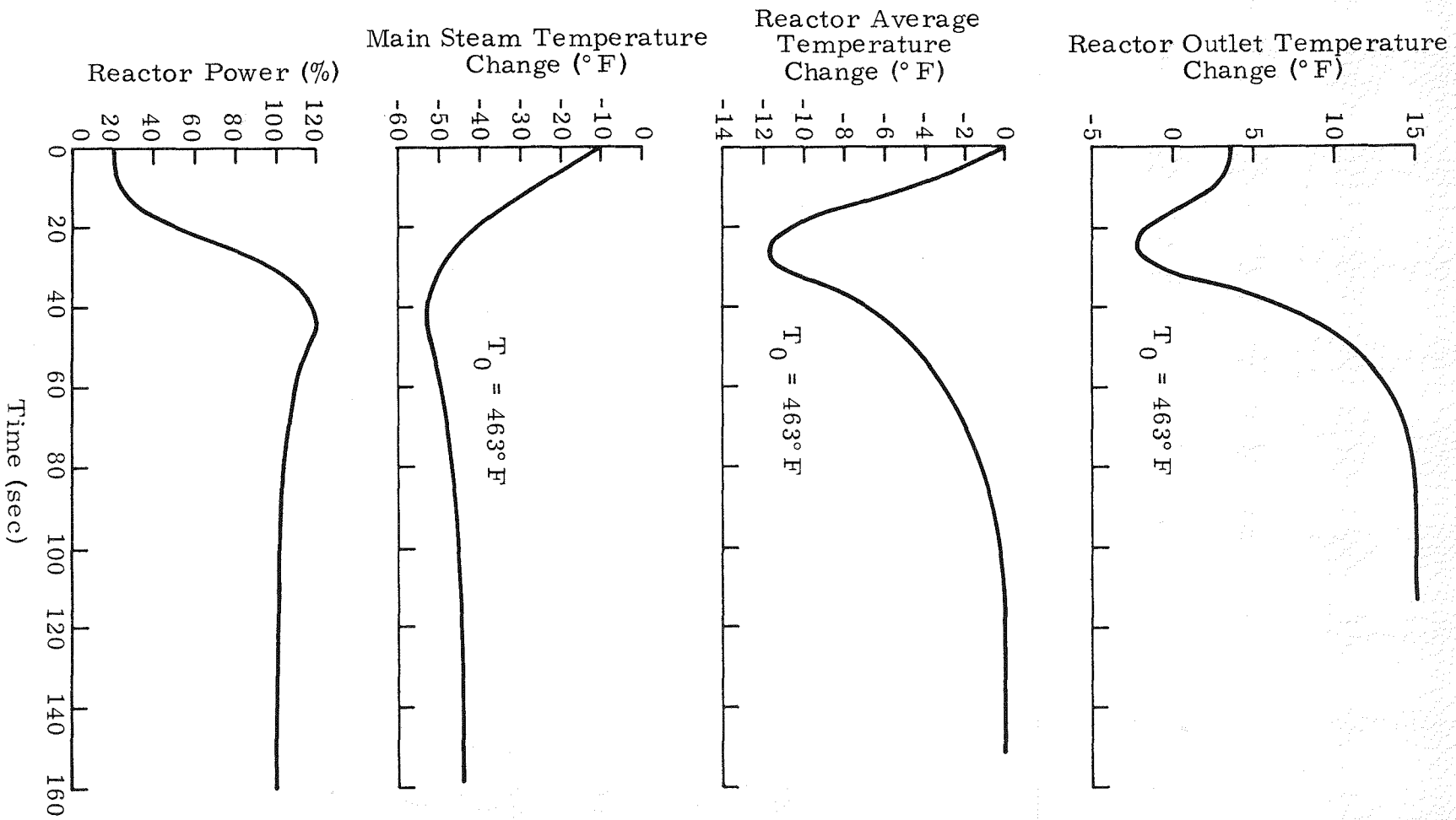


Fig. 19. 20% to 100% Load Transient (predicted)

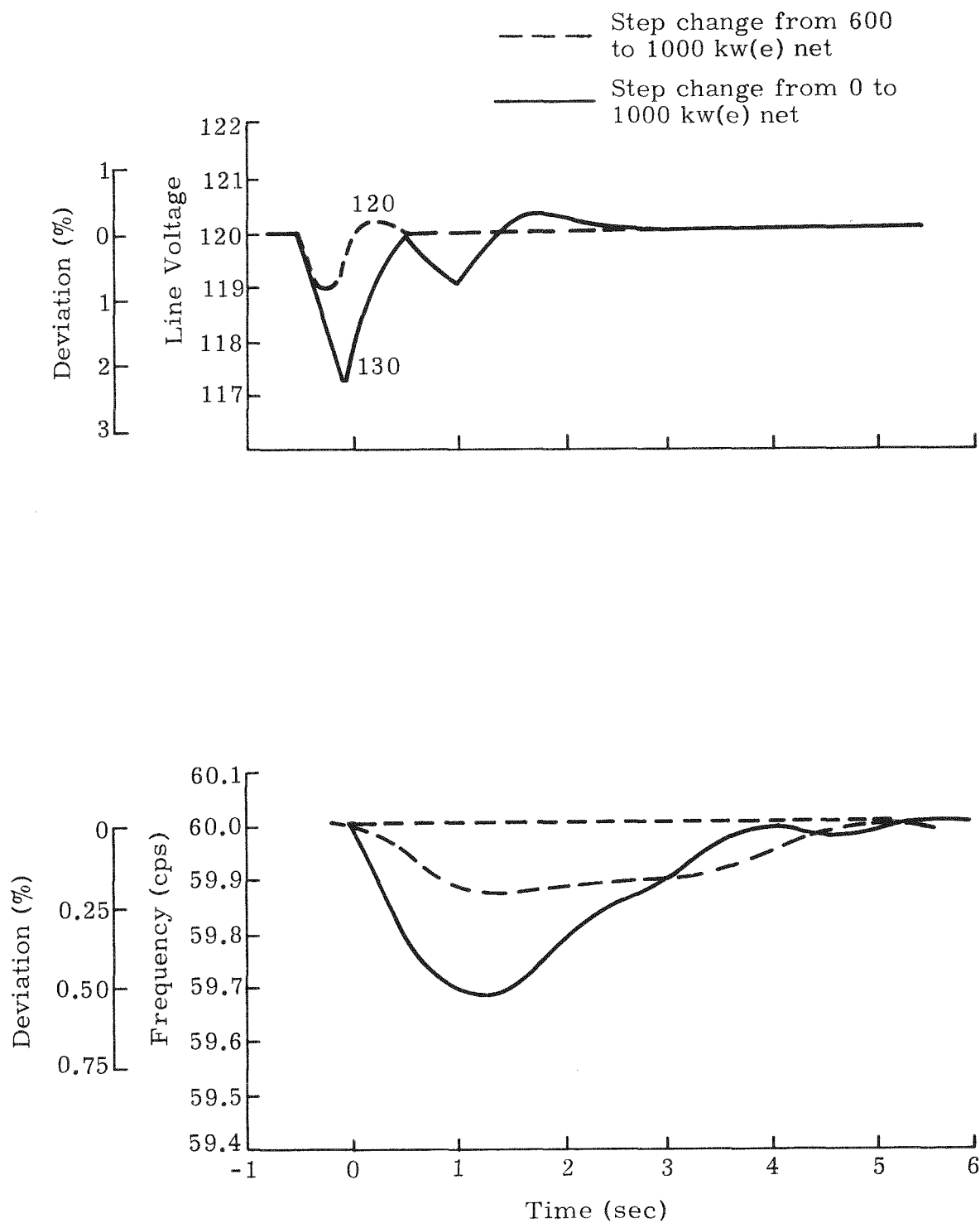


Fig. 20. PM-1 Transient Electrical Characteristics

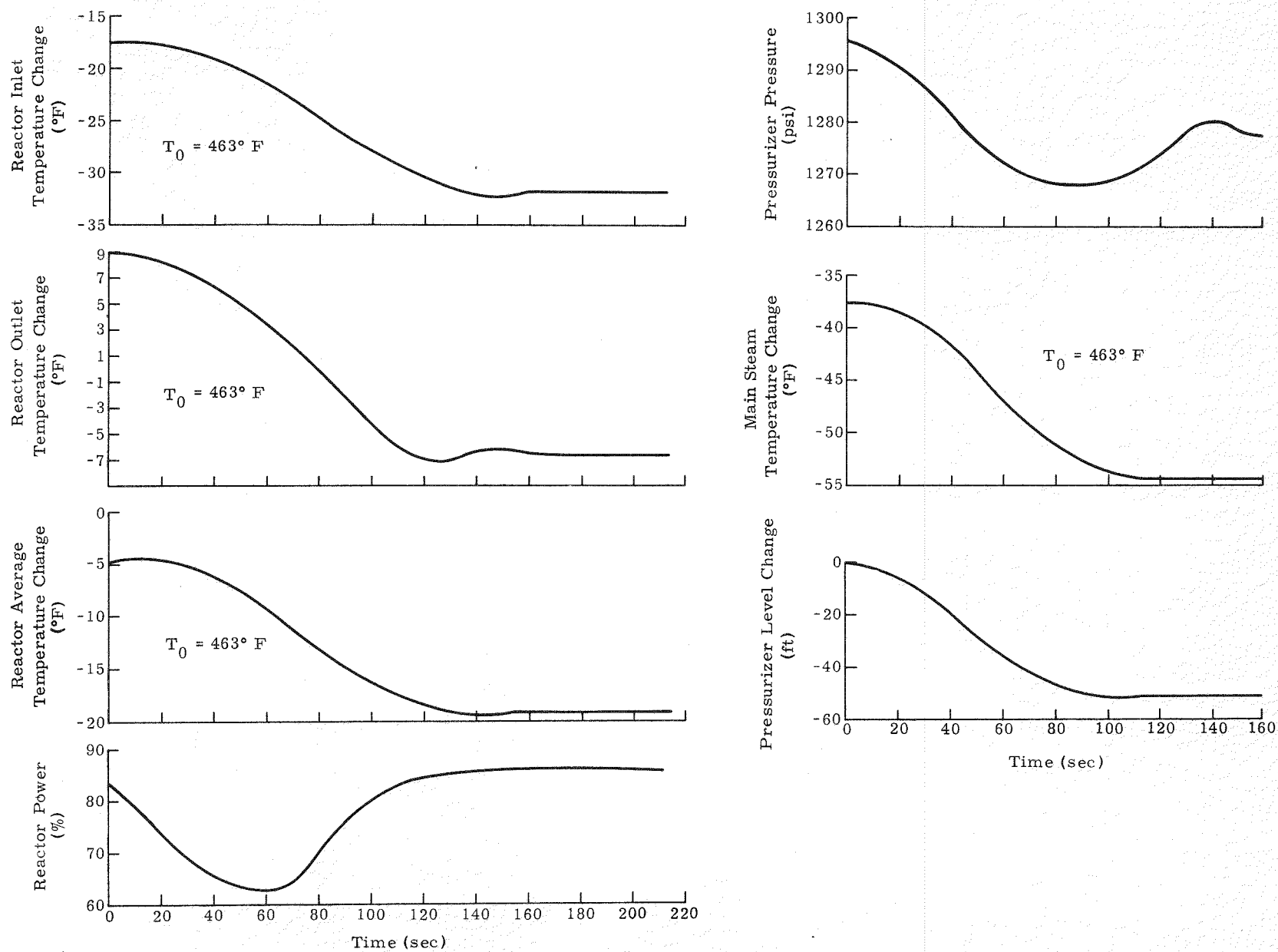


Fig. 21. Rod Insertion Transient--Plant (1 rod at 2 in./min)

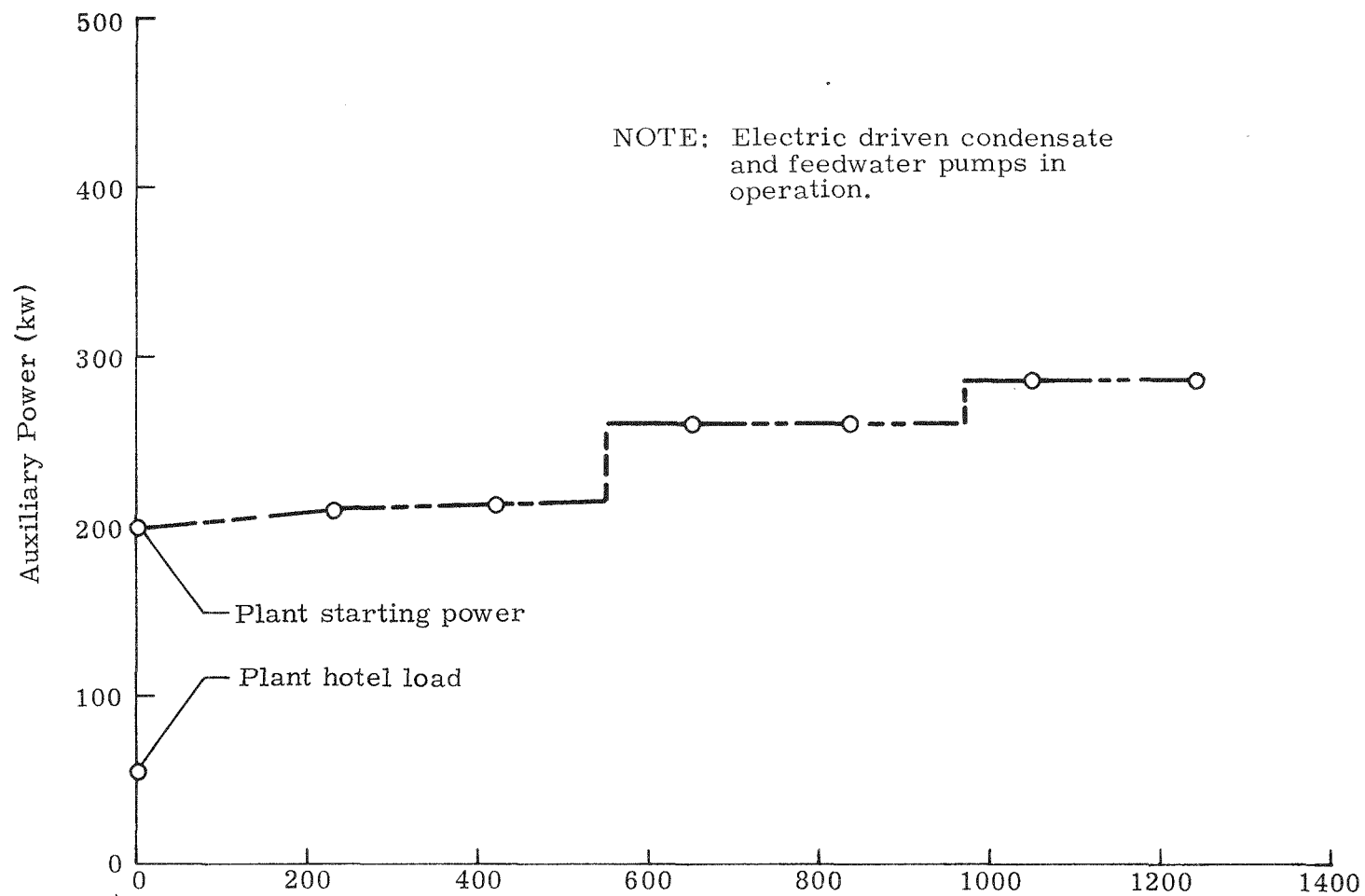


Fig. 22. Plant Gross Electrical Power (kw)

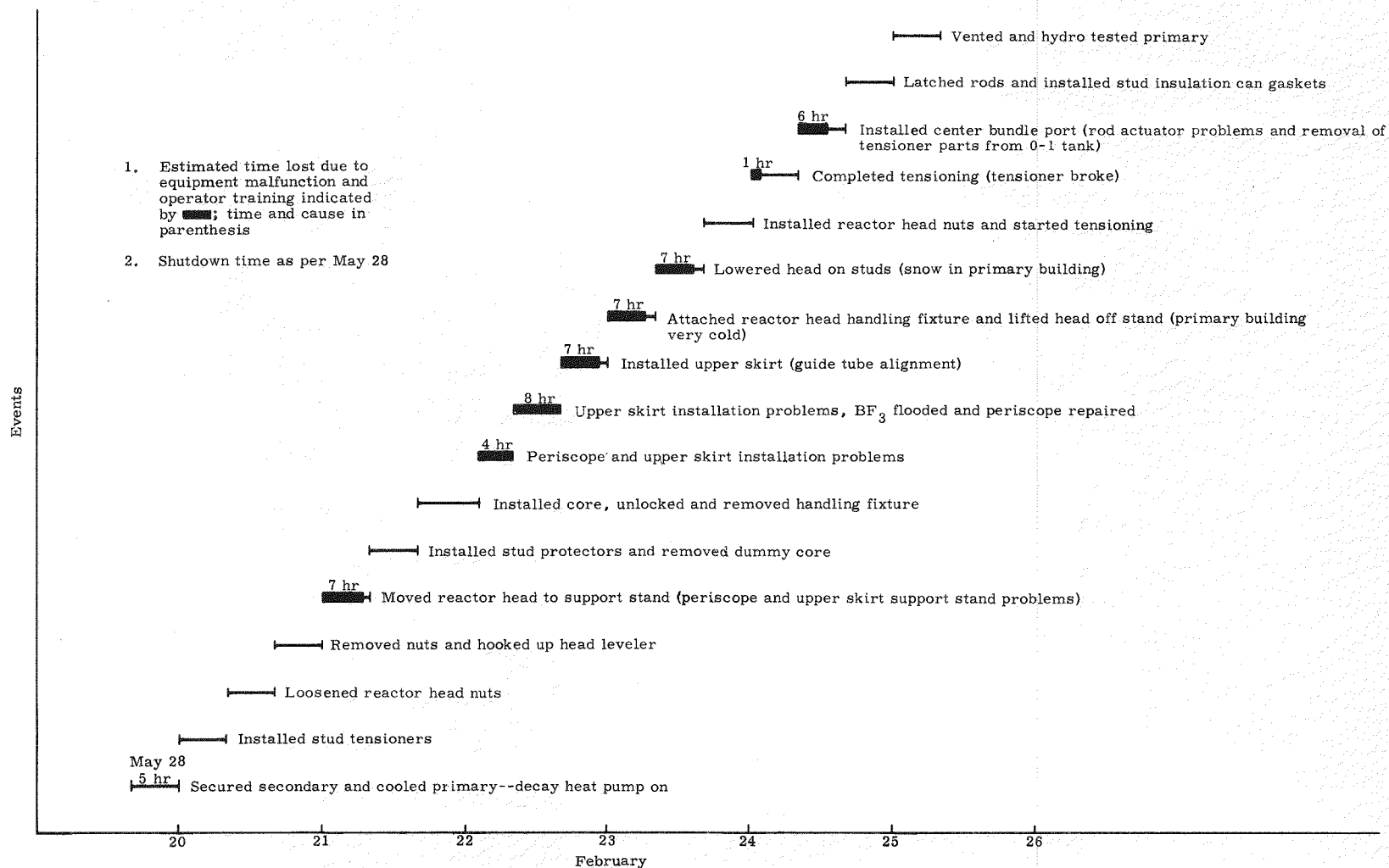


Fig. 23. Dummy Core Refueling--Actual Time

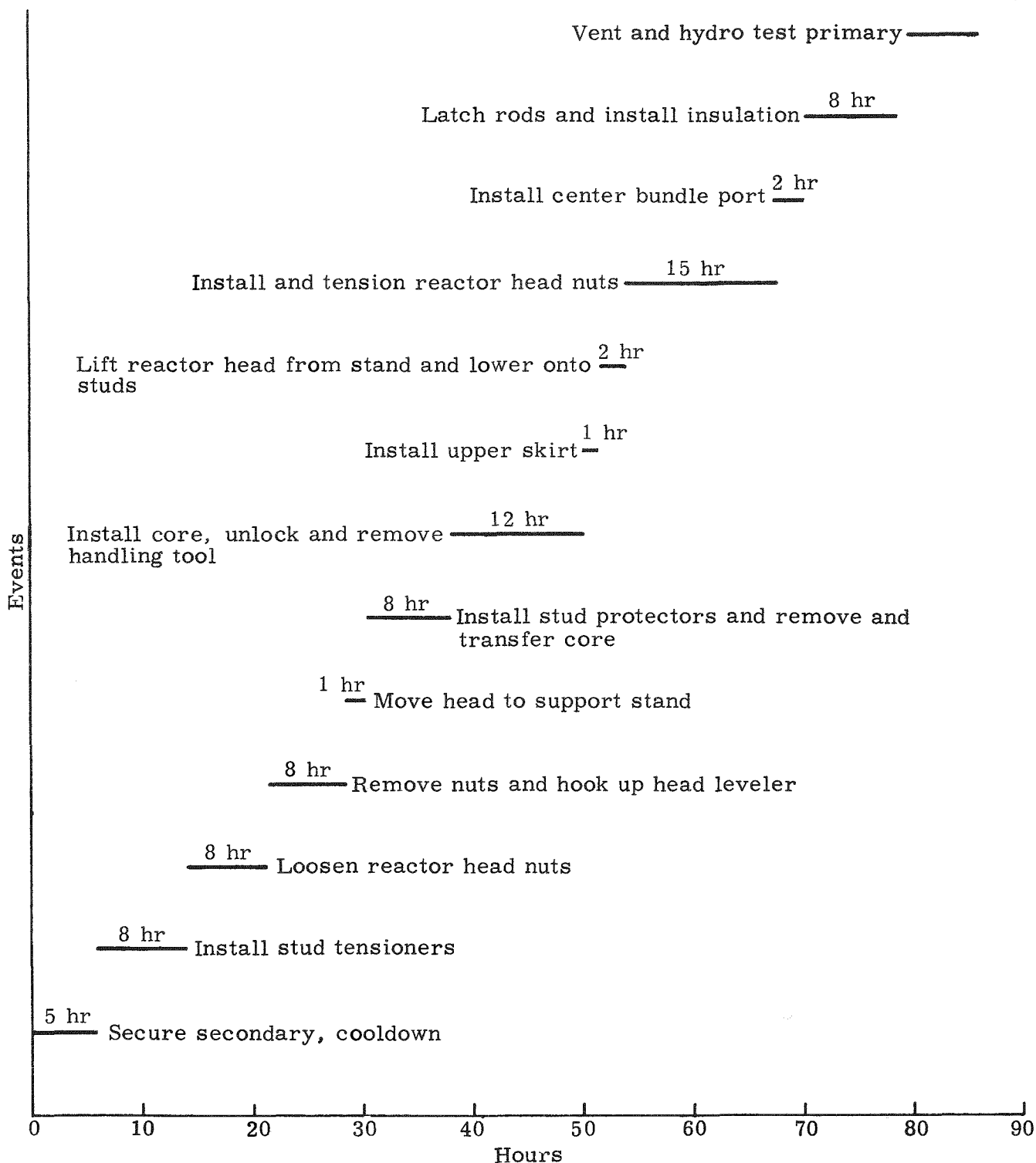


Fig. 24. Estimate of Refueling Time

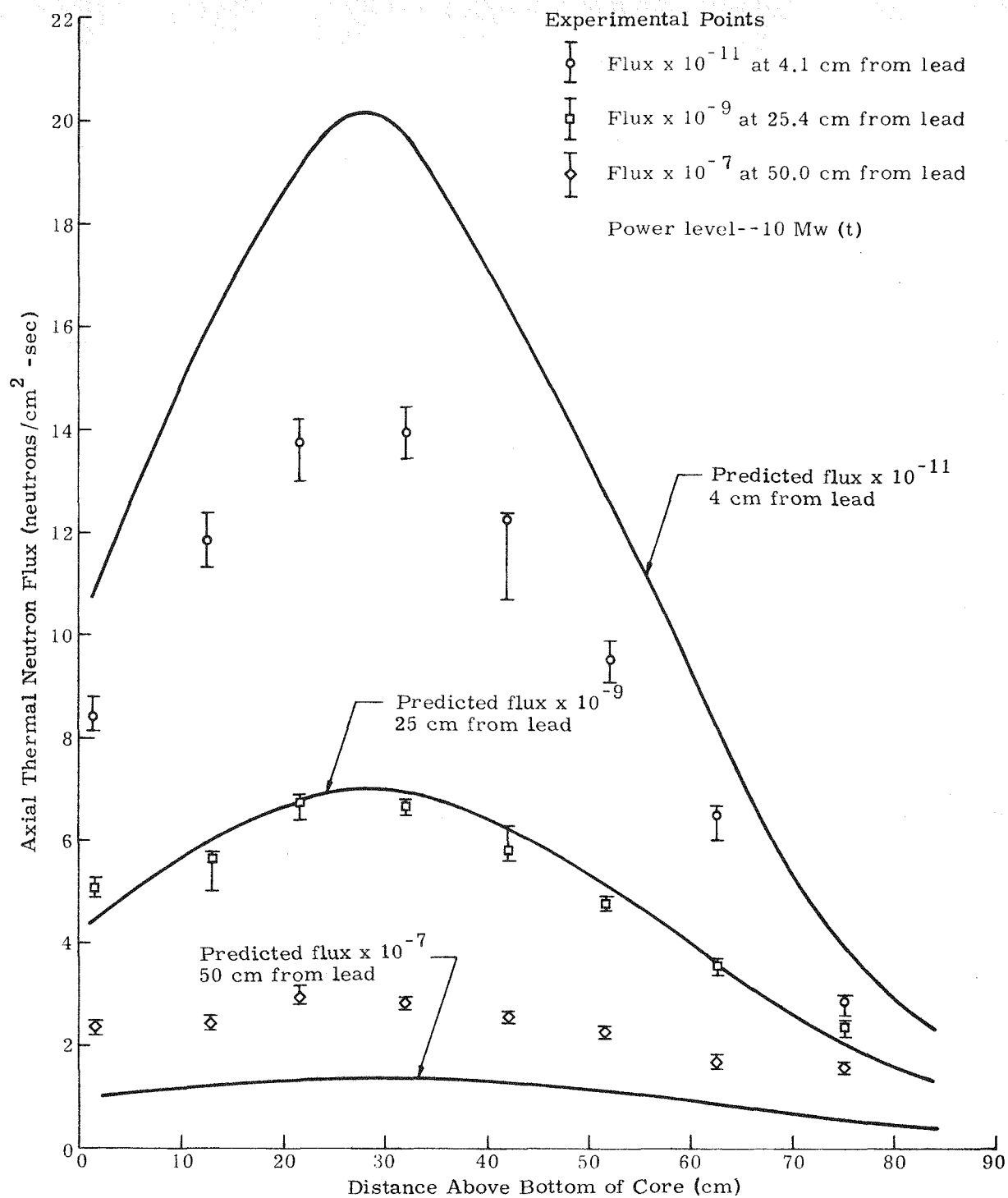


Fig. 25. Axial Distribution of Thermal Neutron Flux in the Reactor Tank

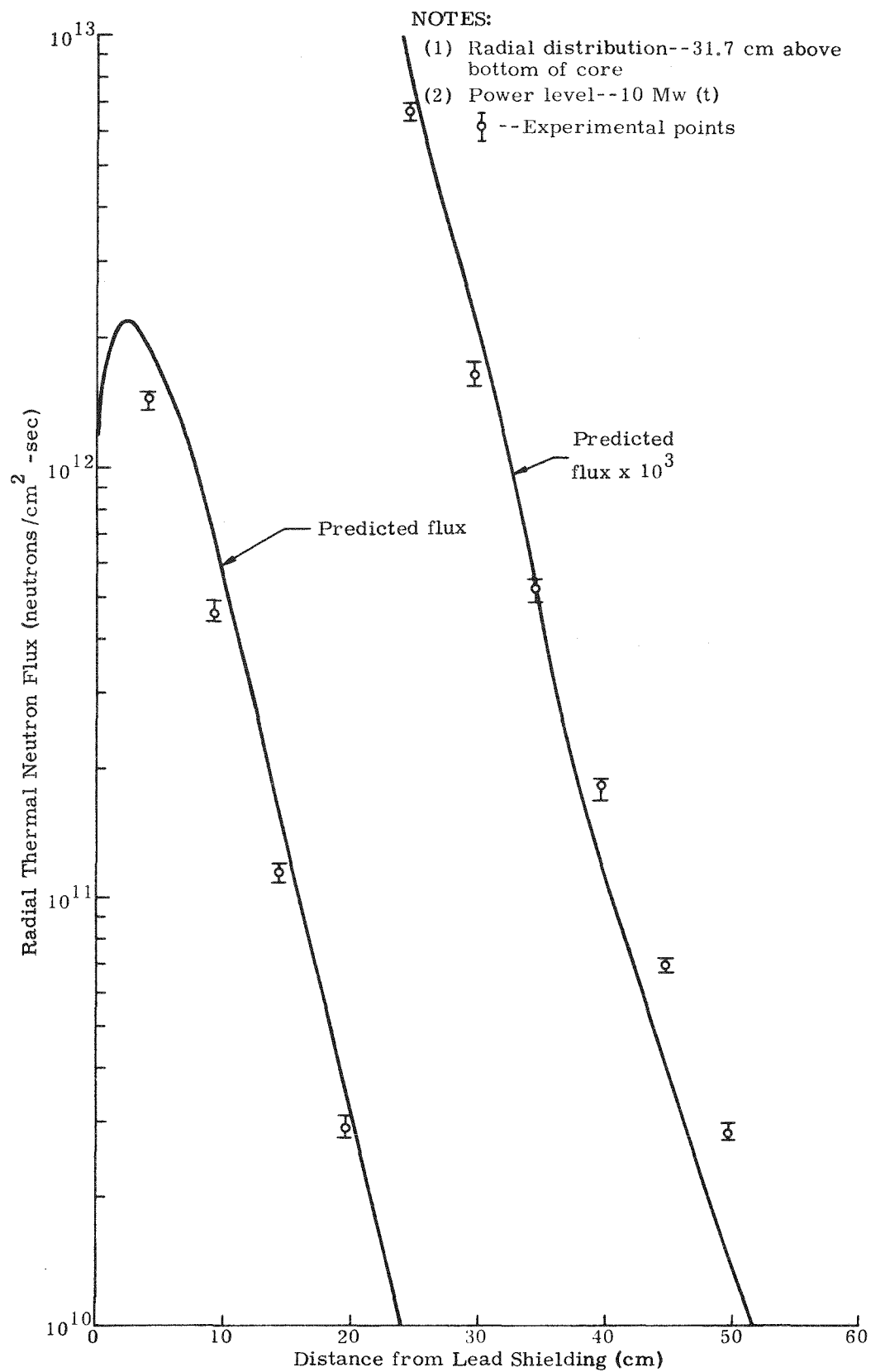


Fig. 26. Radial Distribution of Thermal Neutron Flux in the Reactor Tank

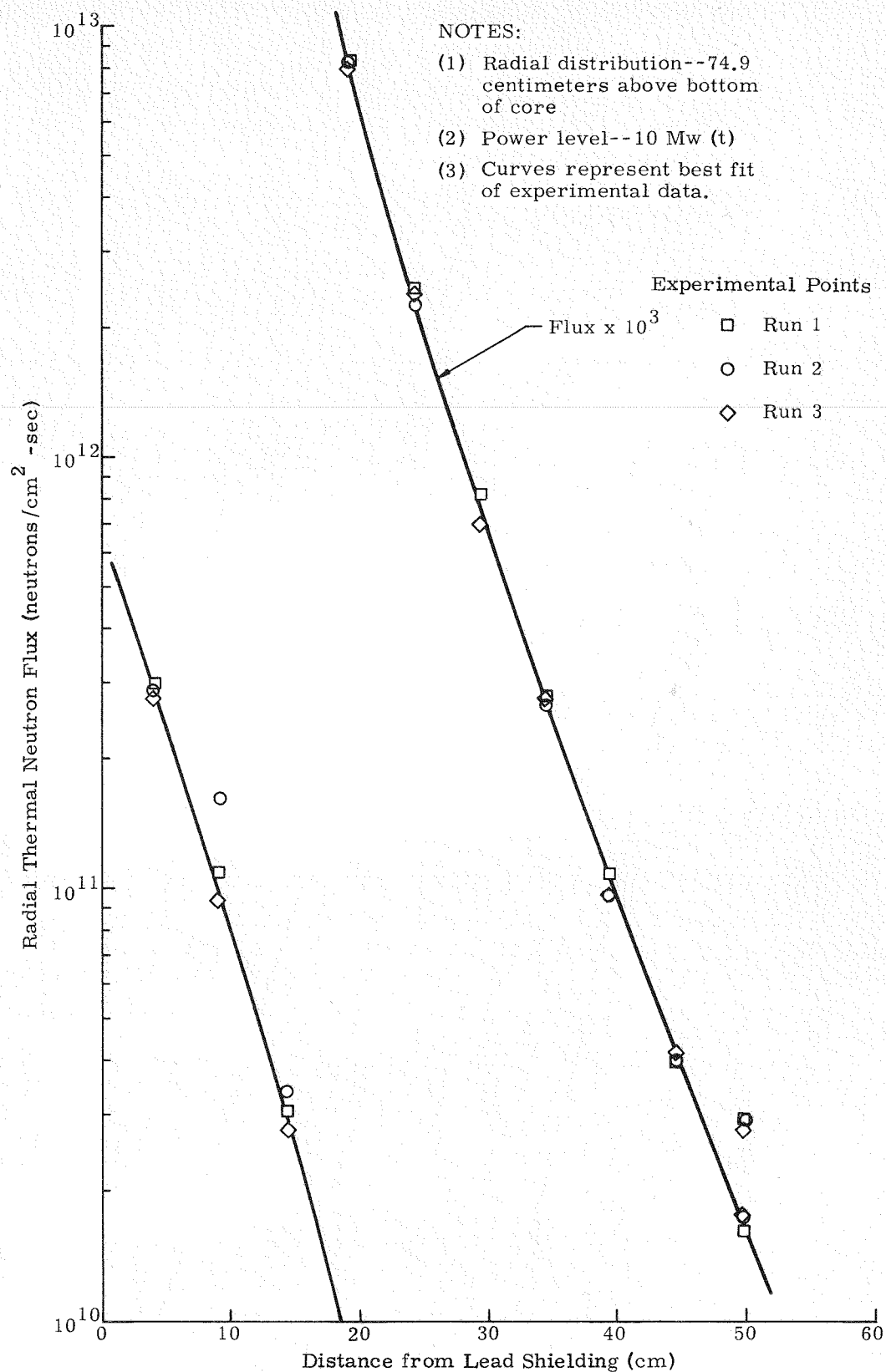


Fig. 27. Radial Distribution of Thermal Flux in the Reactor Tank

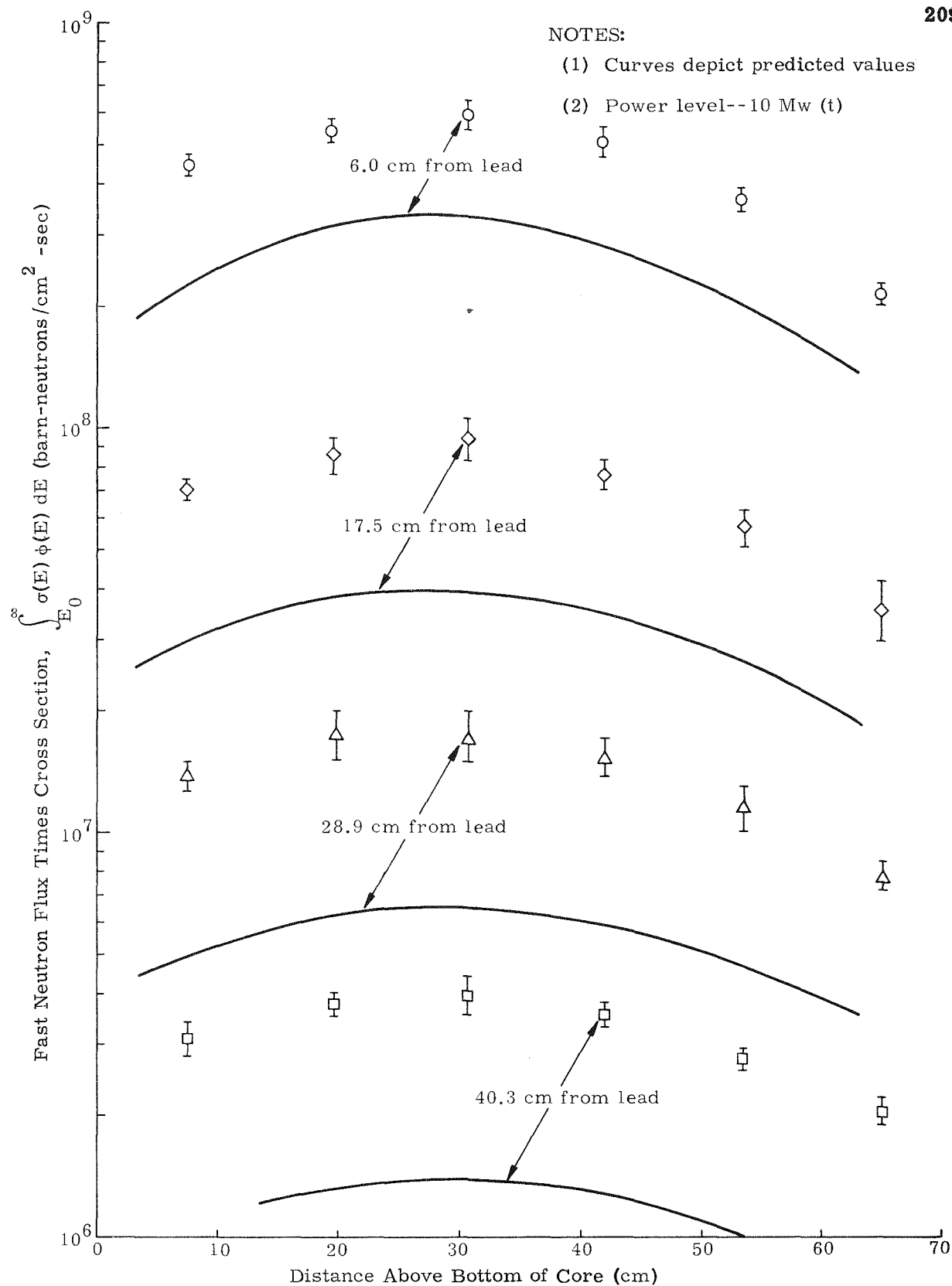


Fig. 28. Axial Distribution of Fast Flux in the Reactor Tank

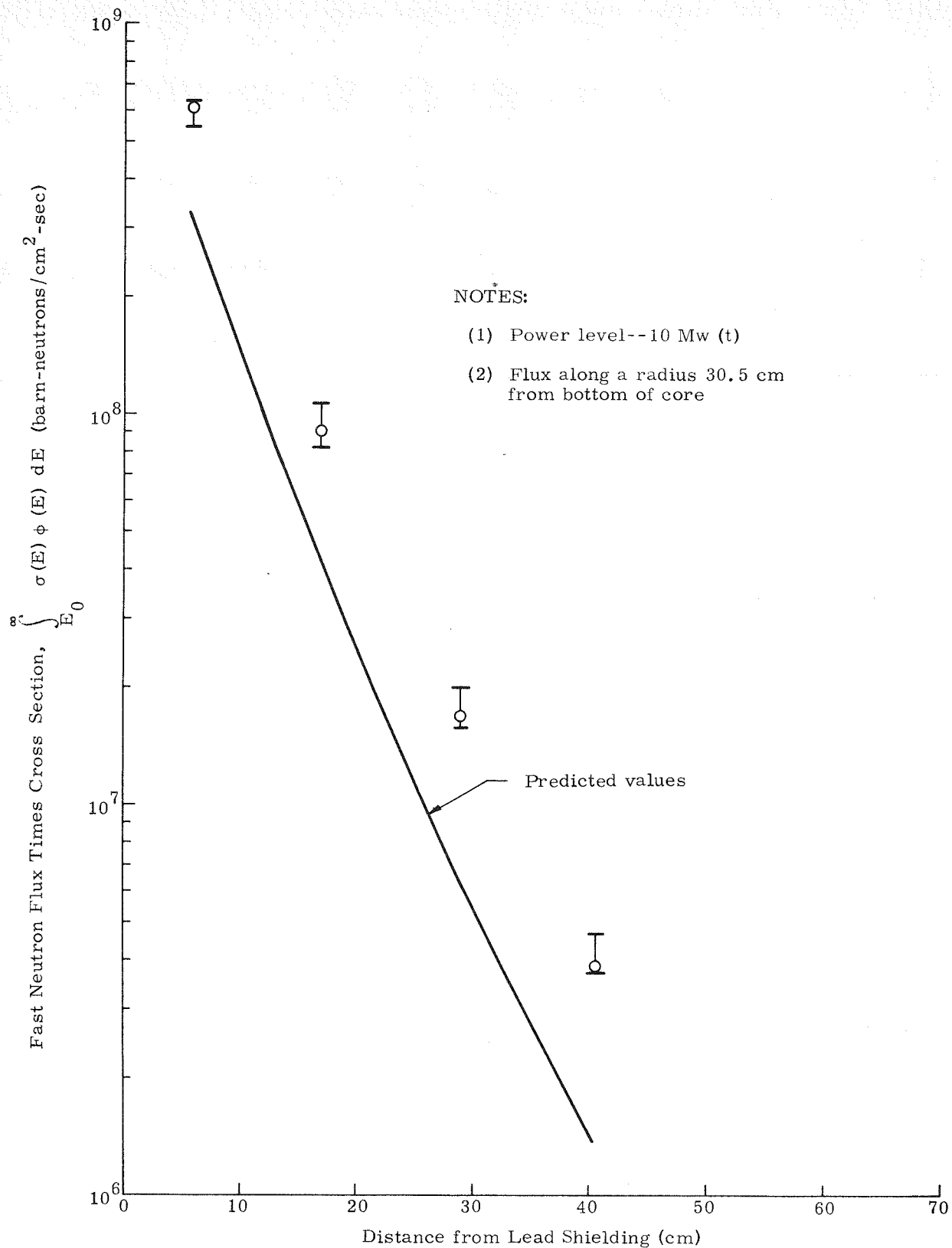


Fig. 29. Radial Distribution of Fast Flux in the Reactor Tank

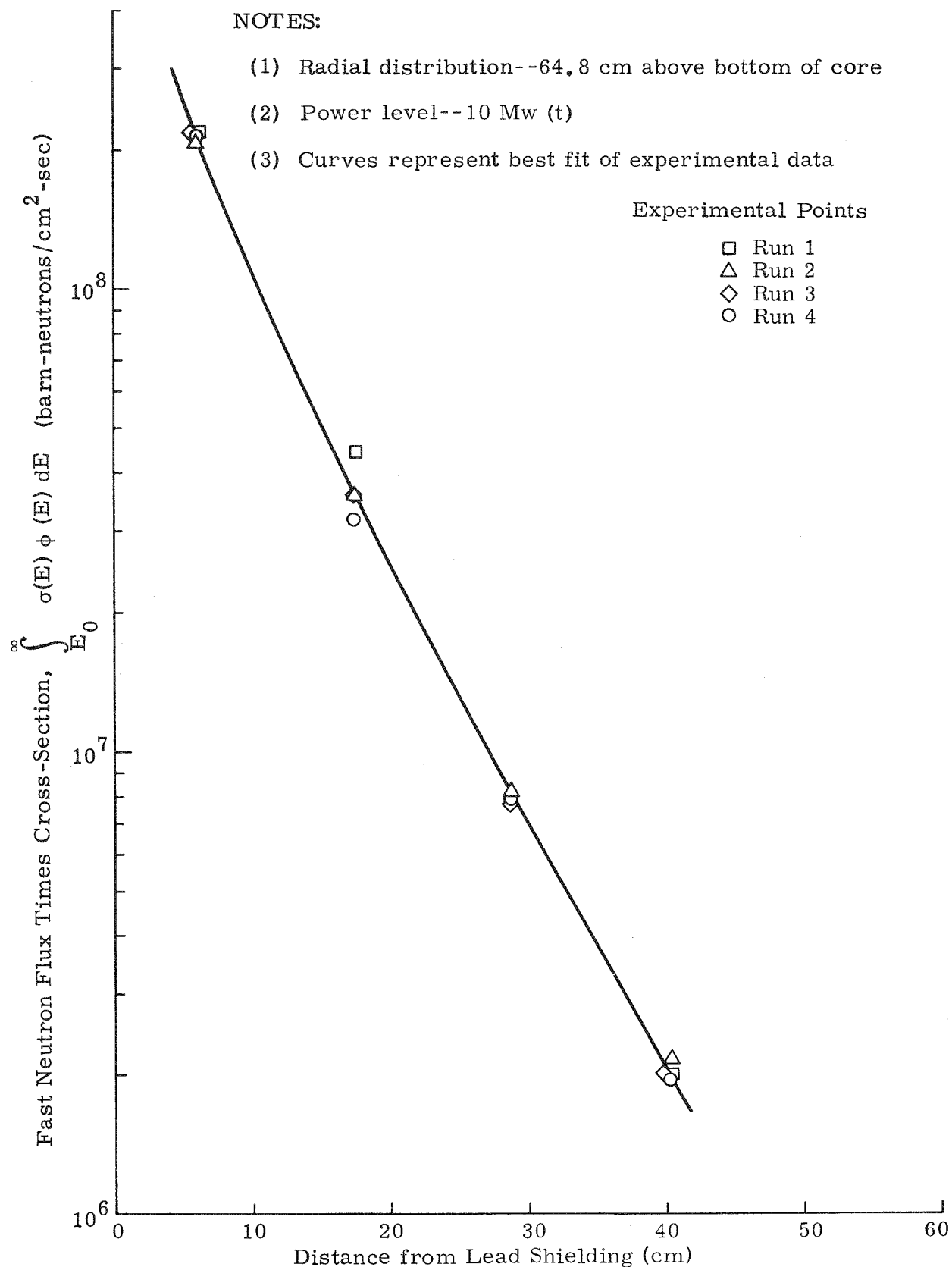


Fig. 30. Radial Distribution of Fast Flux in the Reactor Tank

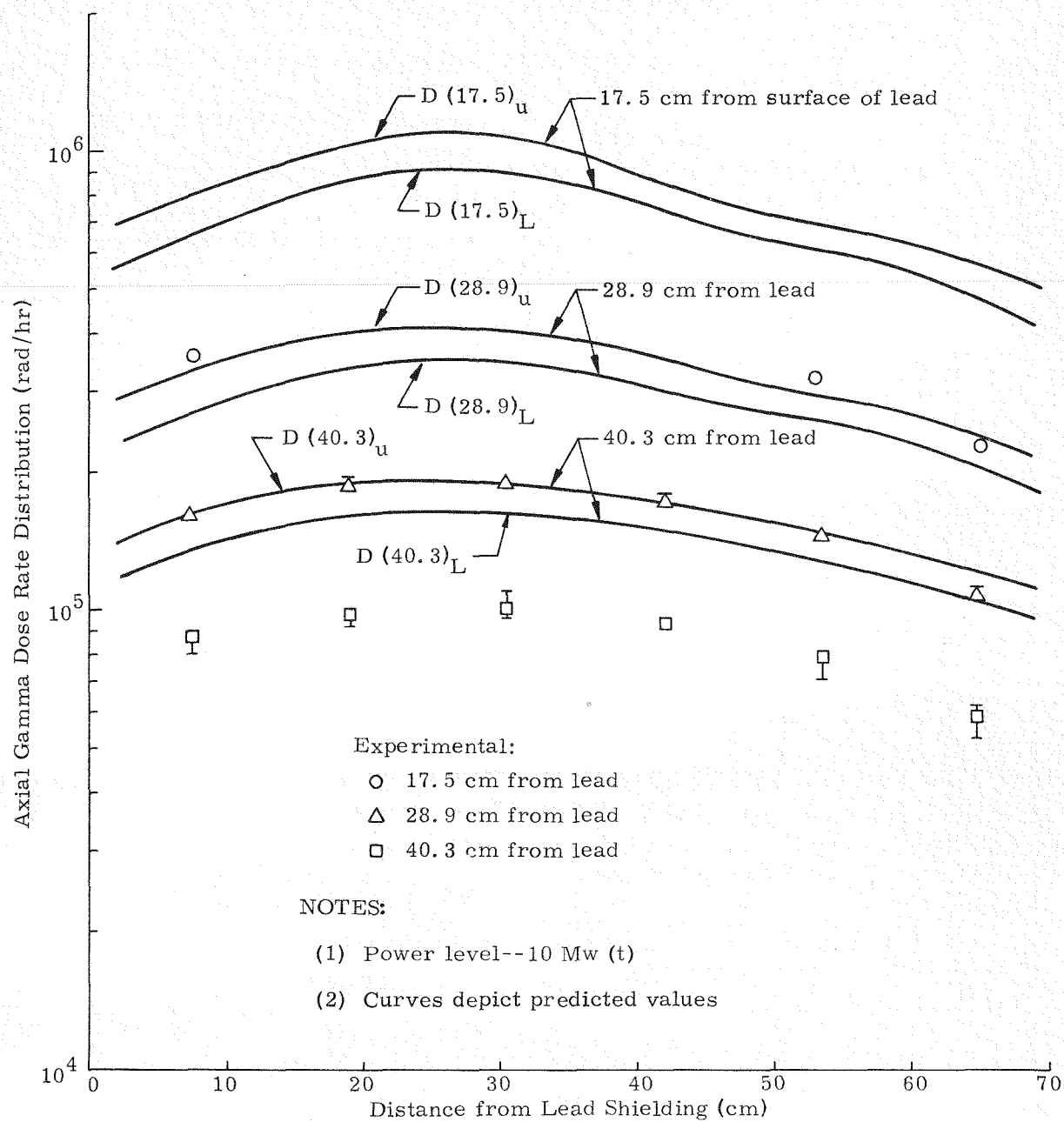


Fig. 31. Axial Distribution of Gamma Dose Rate in the Reactor Tank

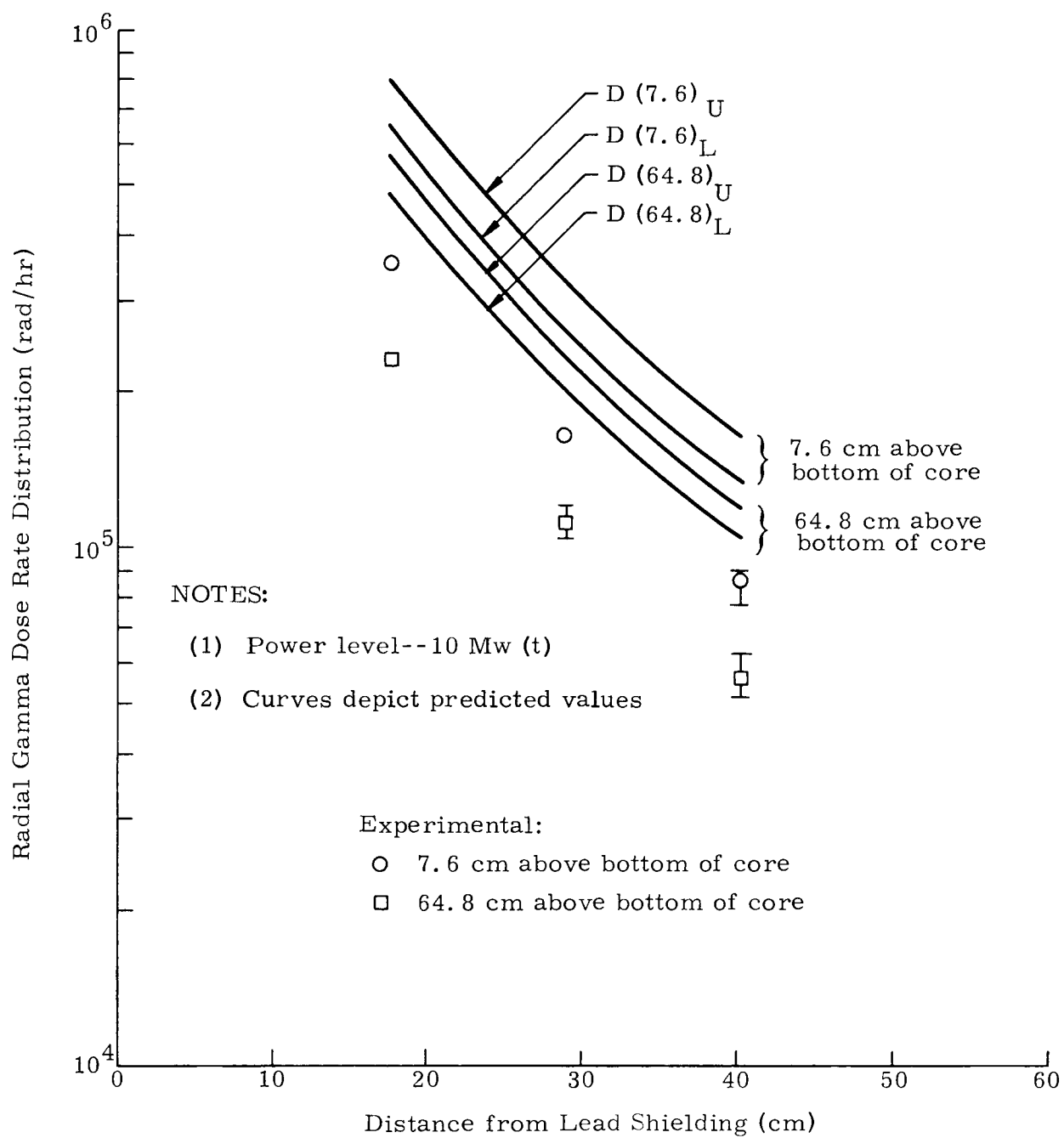


Fig. 32. Radial Distribution of Gamma Dose Rate in the Reactor Tank

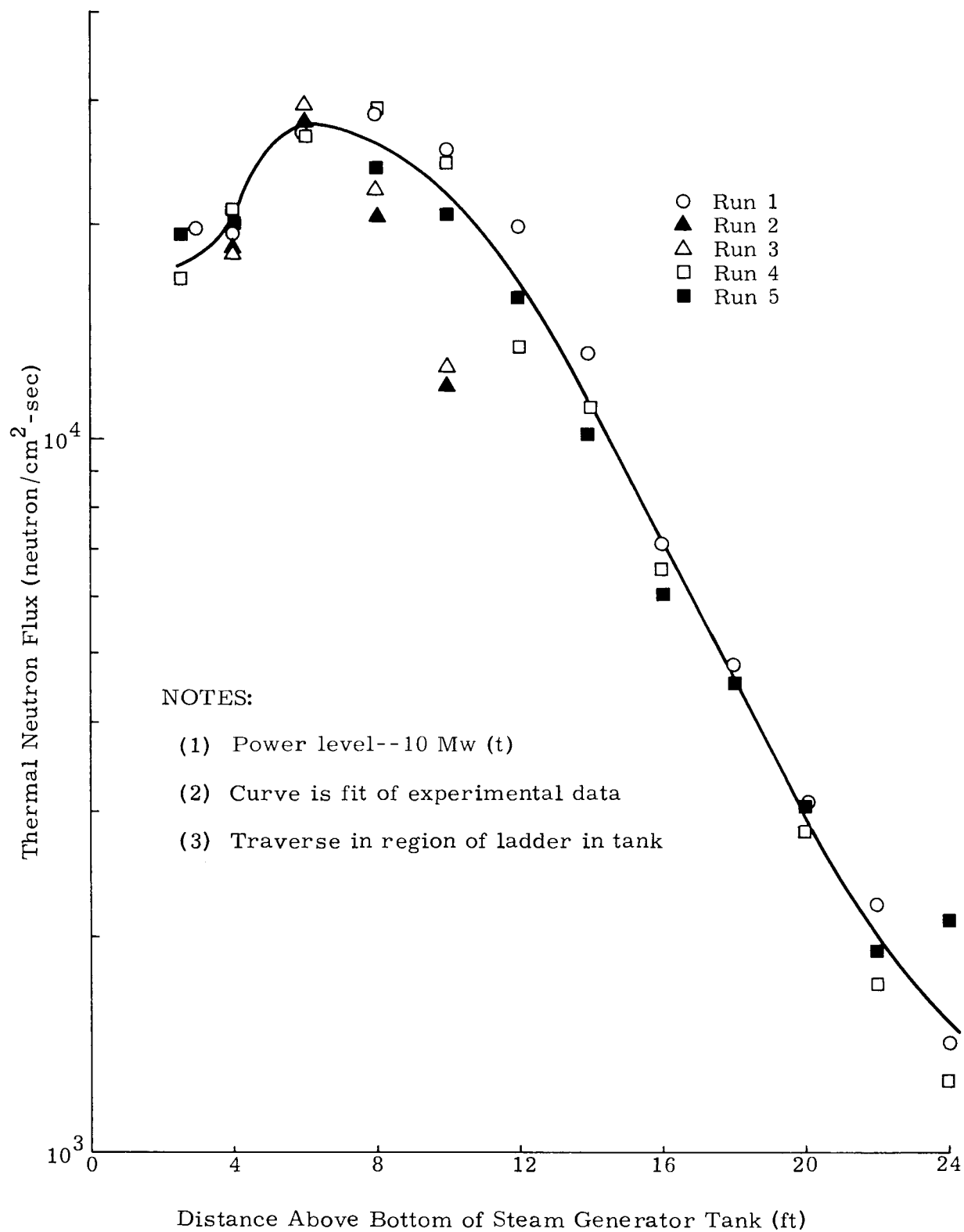


Fig. 33. Axial Distribution of Thermal Neutron Flux in the Steam Generator Tank

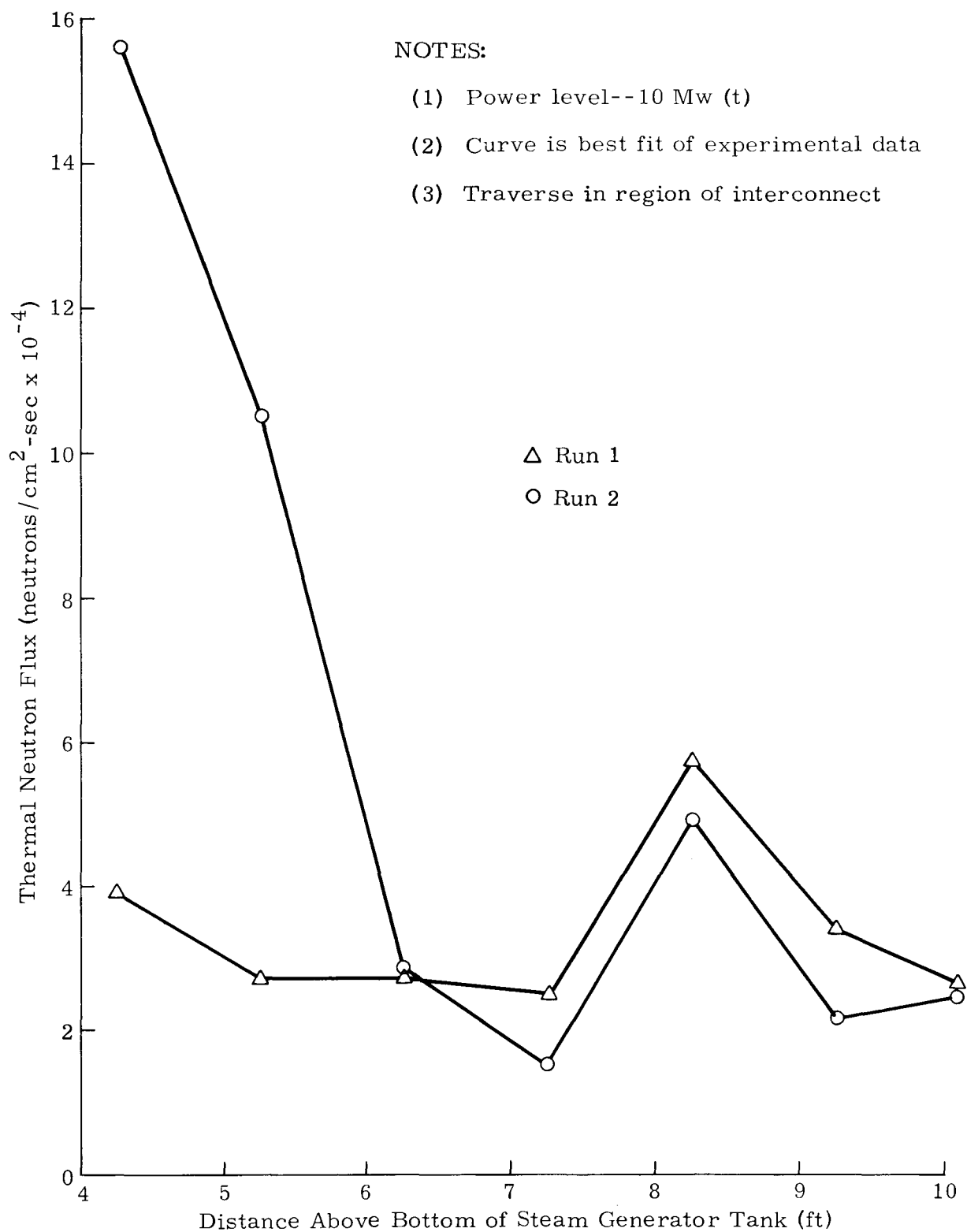


Fig. 34. Axial Distribution of Thermal Flux in the Steam Generator Tank

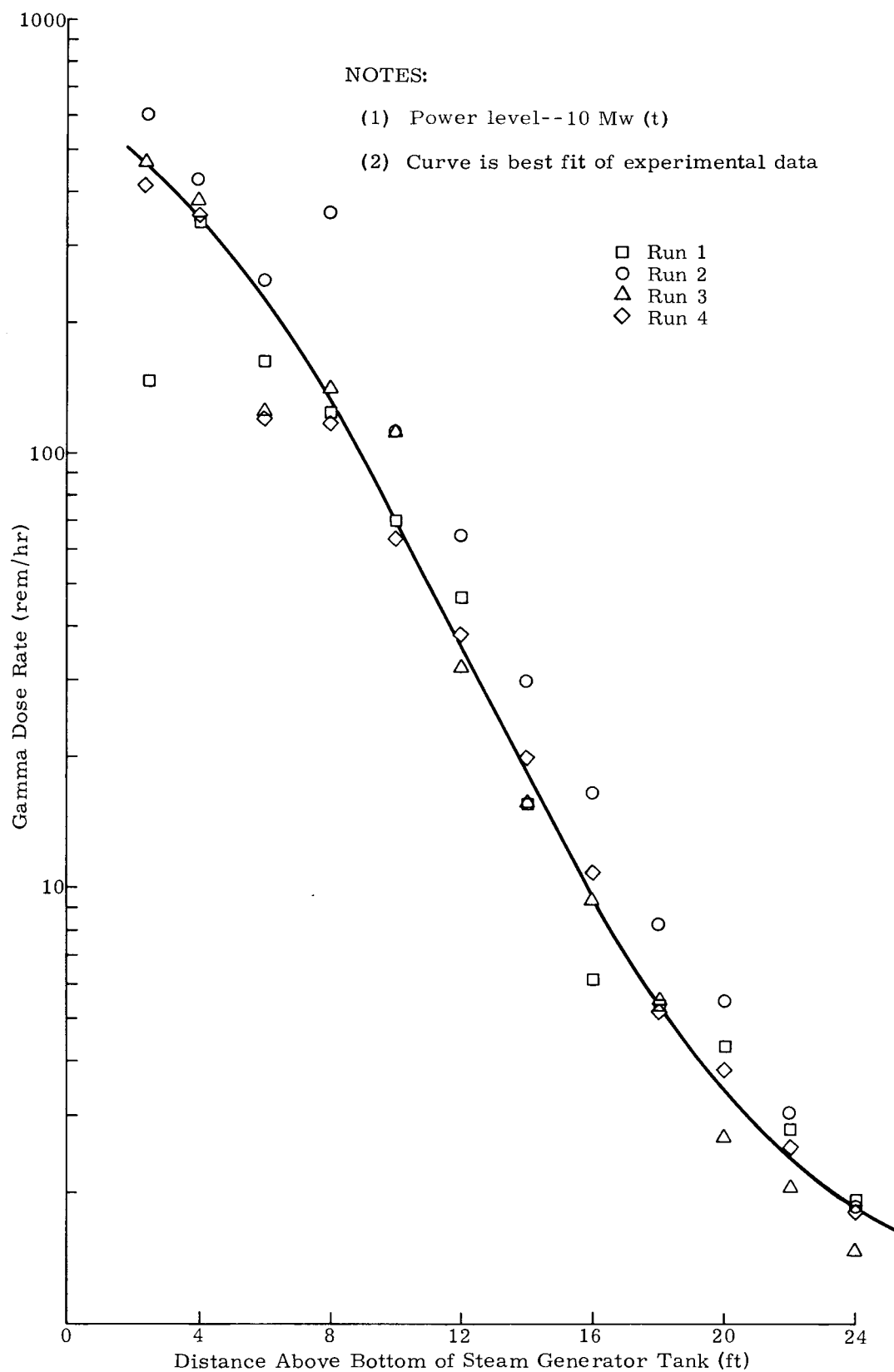
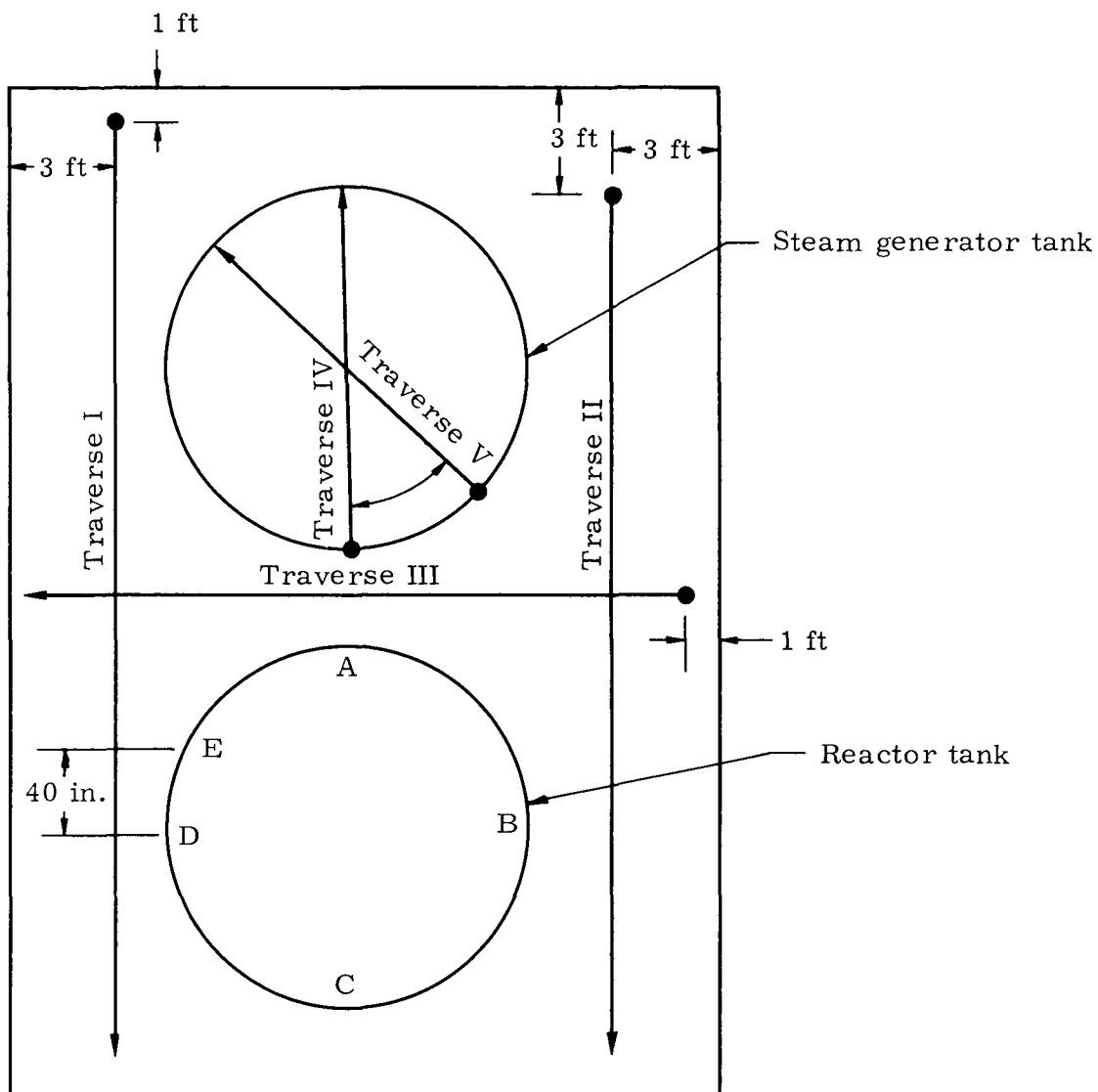


Fig. 35. Axial Distribution of Gamma Dose Rate in the Steam Generator Tank



Traverses I, II, and III taken at 3 in. above deck plates and increments of 12 in. Traverses IV and V taken at 40 in. above deck plates and increments of 10 in. Locations A, B, C, D, and E were on ledge at top of reactor tank double wall.

Fig. 36. Location of Gamma Survey Measurements

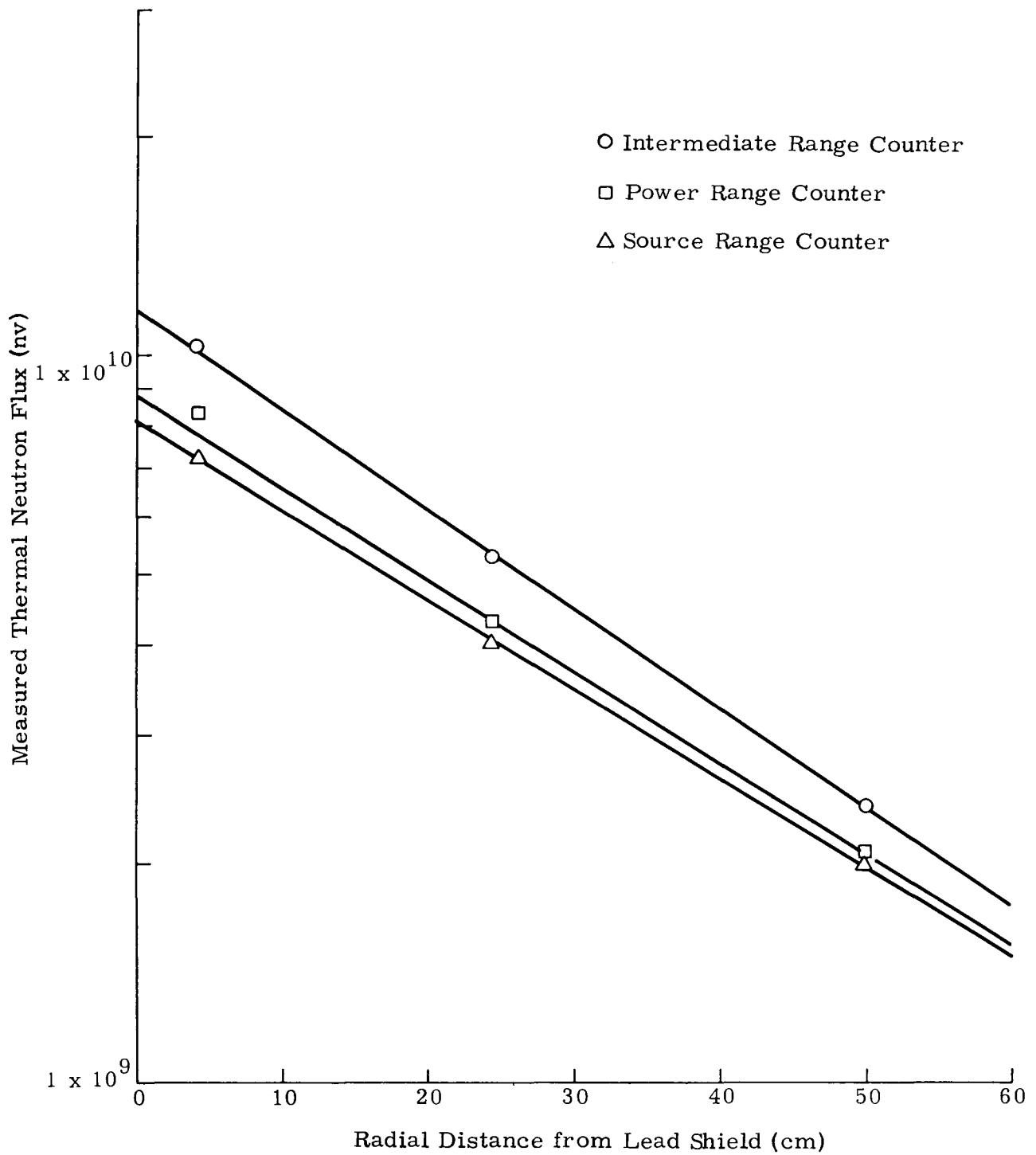


Fig. 37. Average Axial Integrated Flux (nv) As a Function of Radial Distance from Pressure Vessel (based on counter length and position)

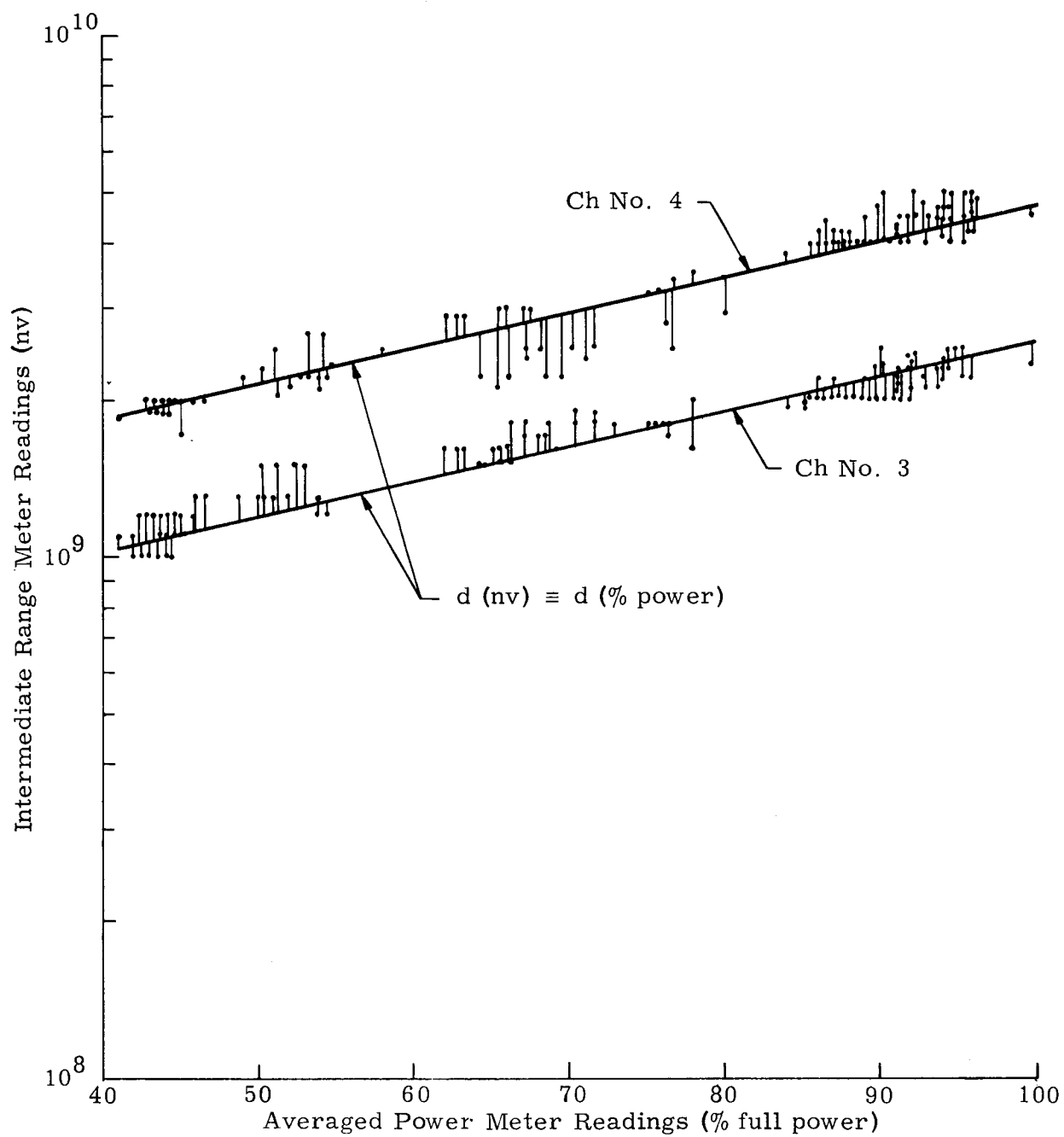


Fig. 38. Intermediate Range Meter Readings as a Function of Power Range Meter Readings

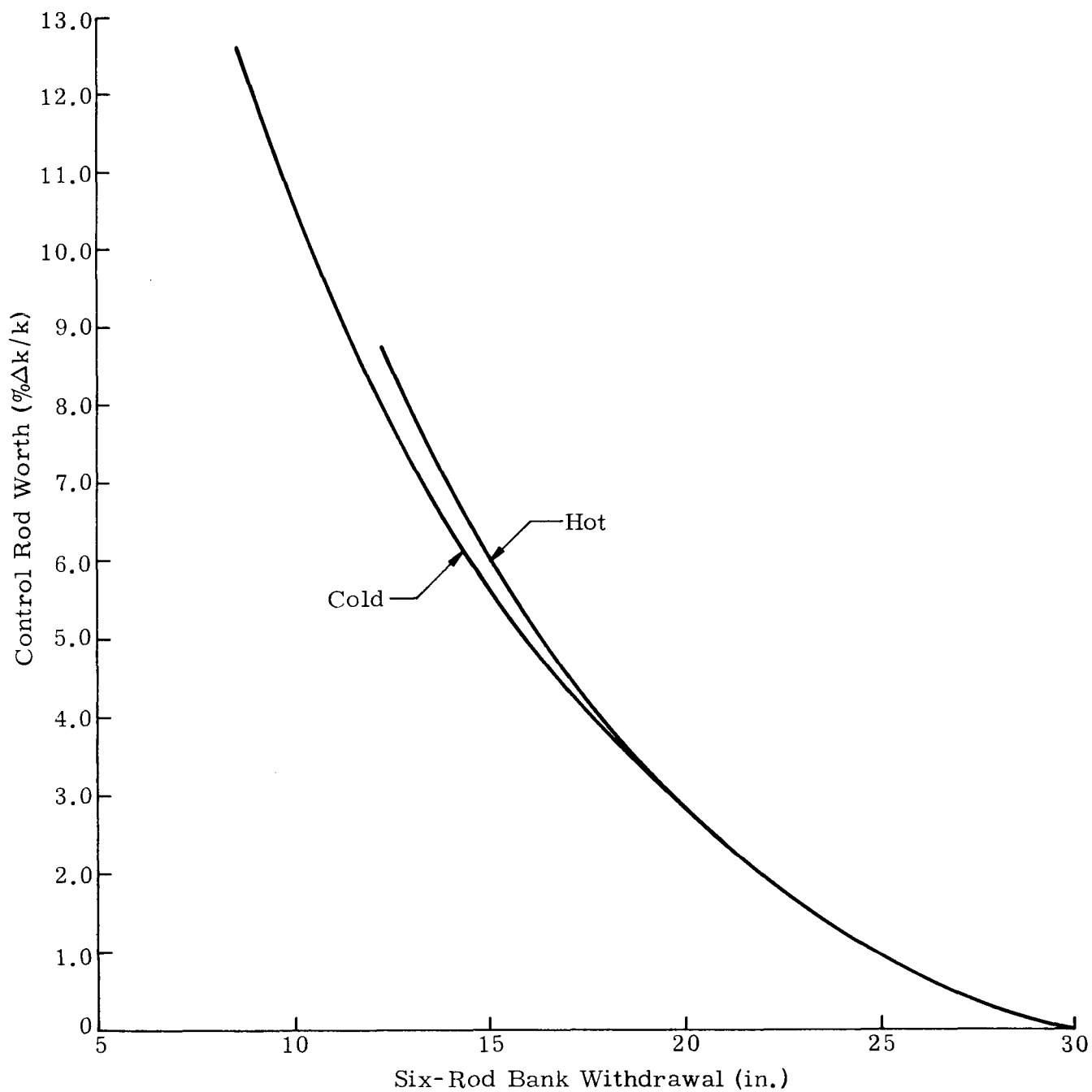


Fig. 39. Control Rod Worth

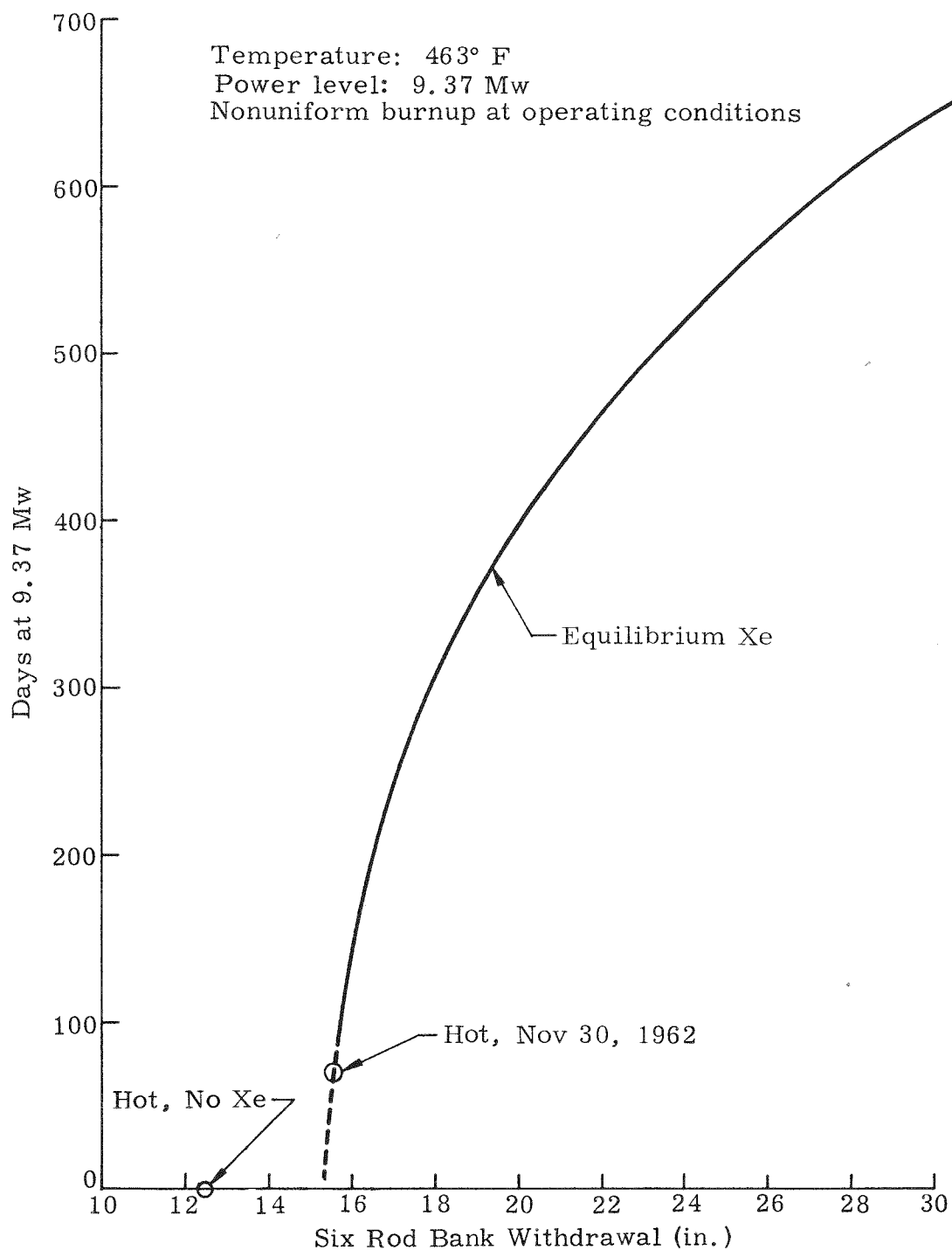


Fig. 40. PM-1 Lifetime Versus Six-Rod Bank Position

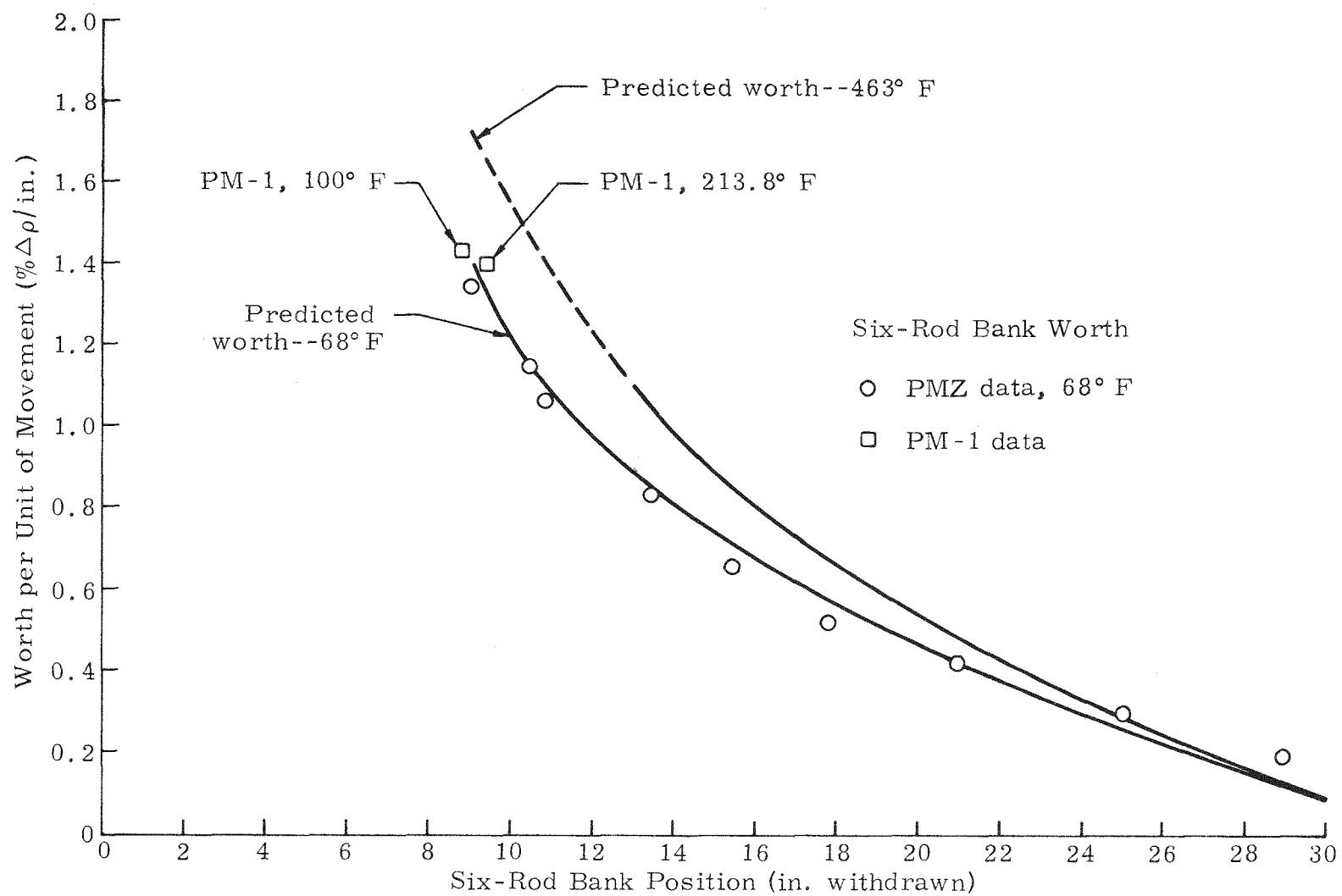


Fig. 42. Predicted and Actual Control Rod Worth

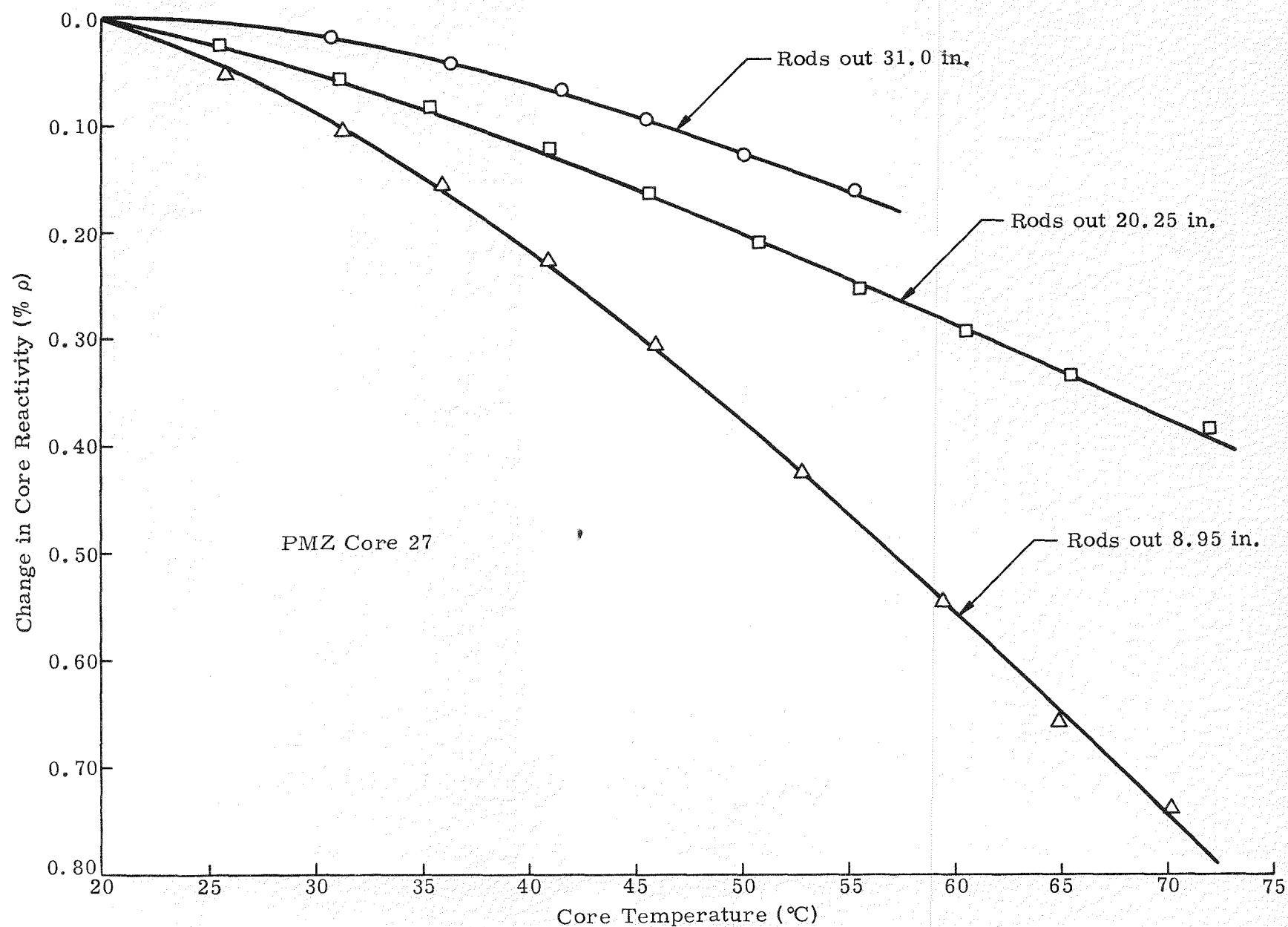


Fig. 43. Experimental Change in Reactivity Versus Moderator Temperature

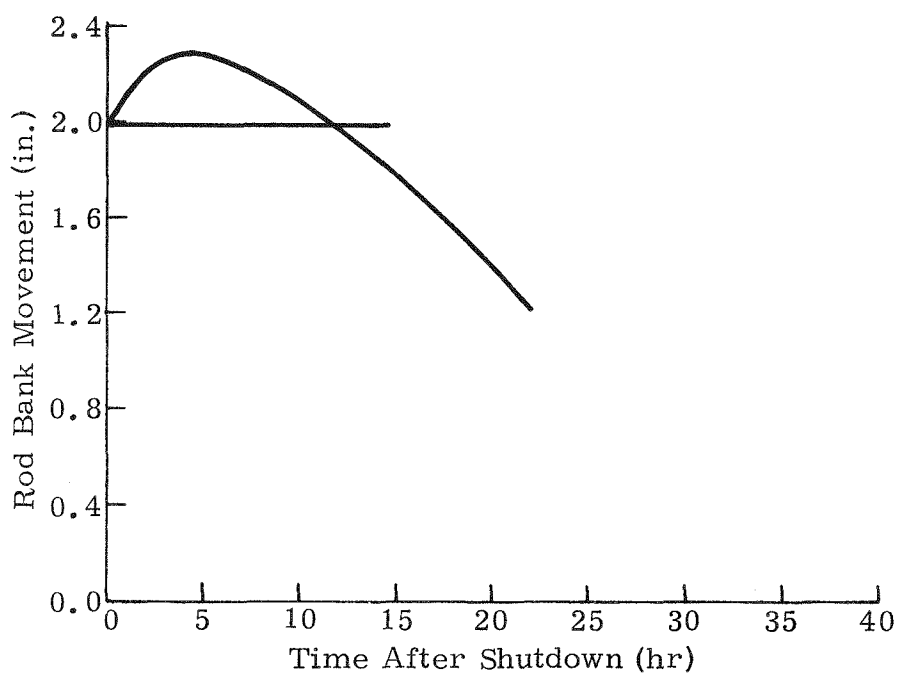
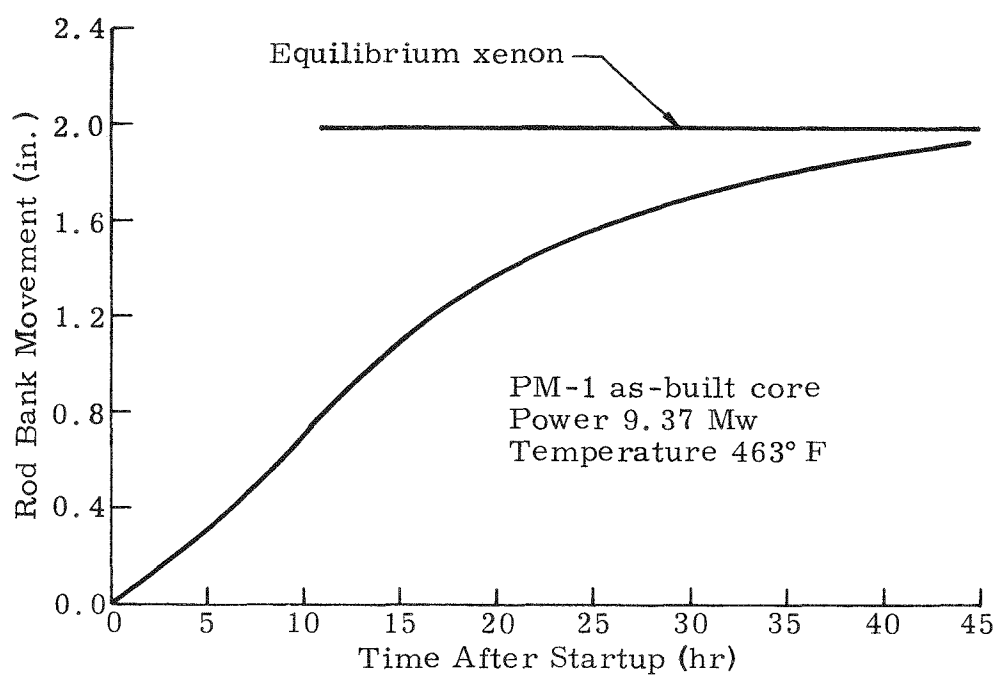
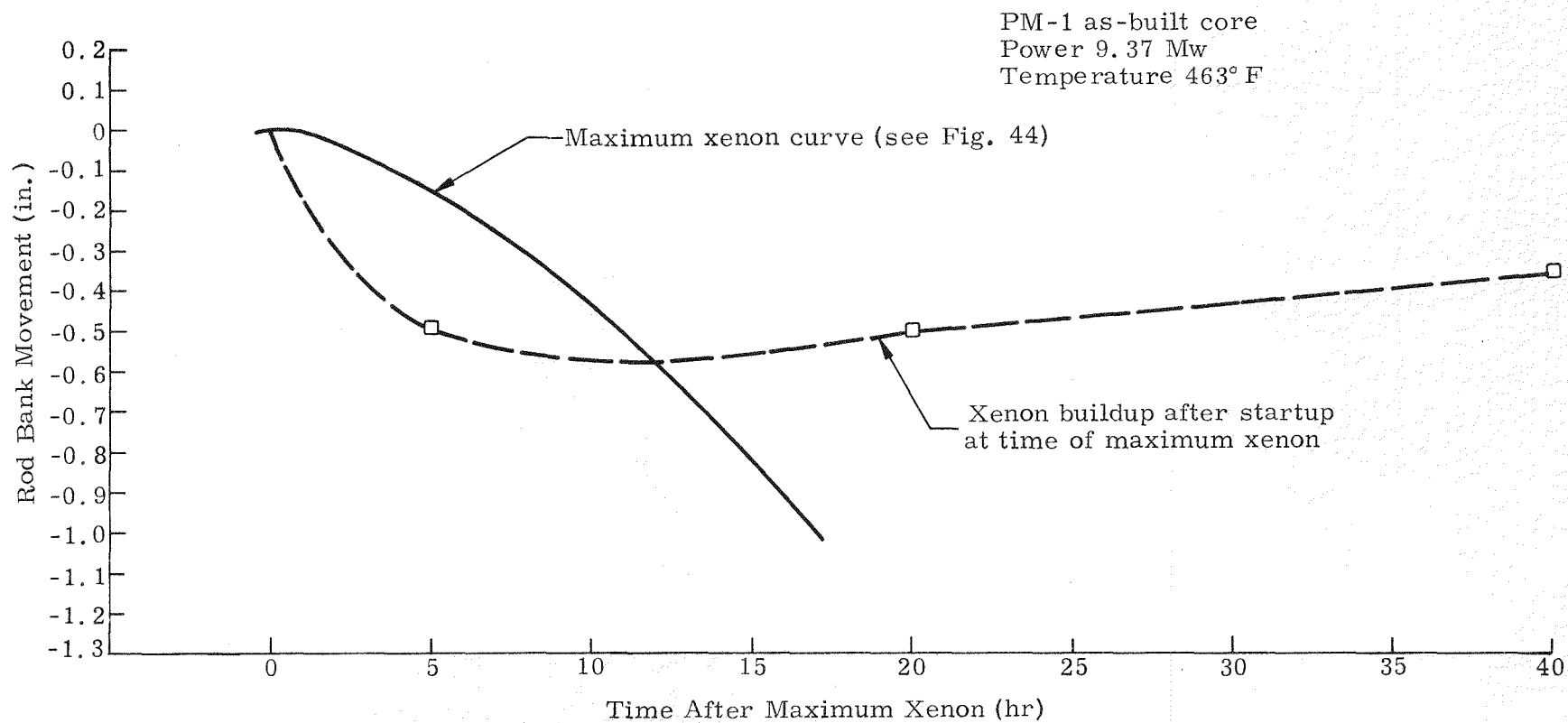


Fig. 44. PM-1 Six-Rod Bank Movement Versus Buildup of Initial Xenon and Buildup of Xenon After Shutdown at Equilibrium Xenon

Fig. 45. PM-1 Six-Rod Bank Movement Upon Restarting at Time of Maximum Xenon



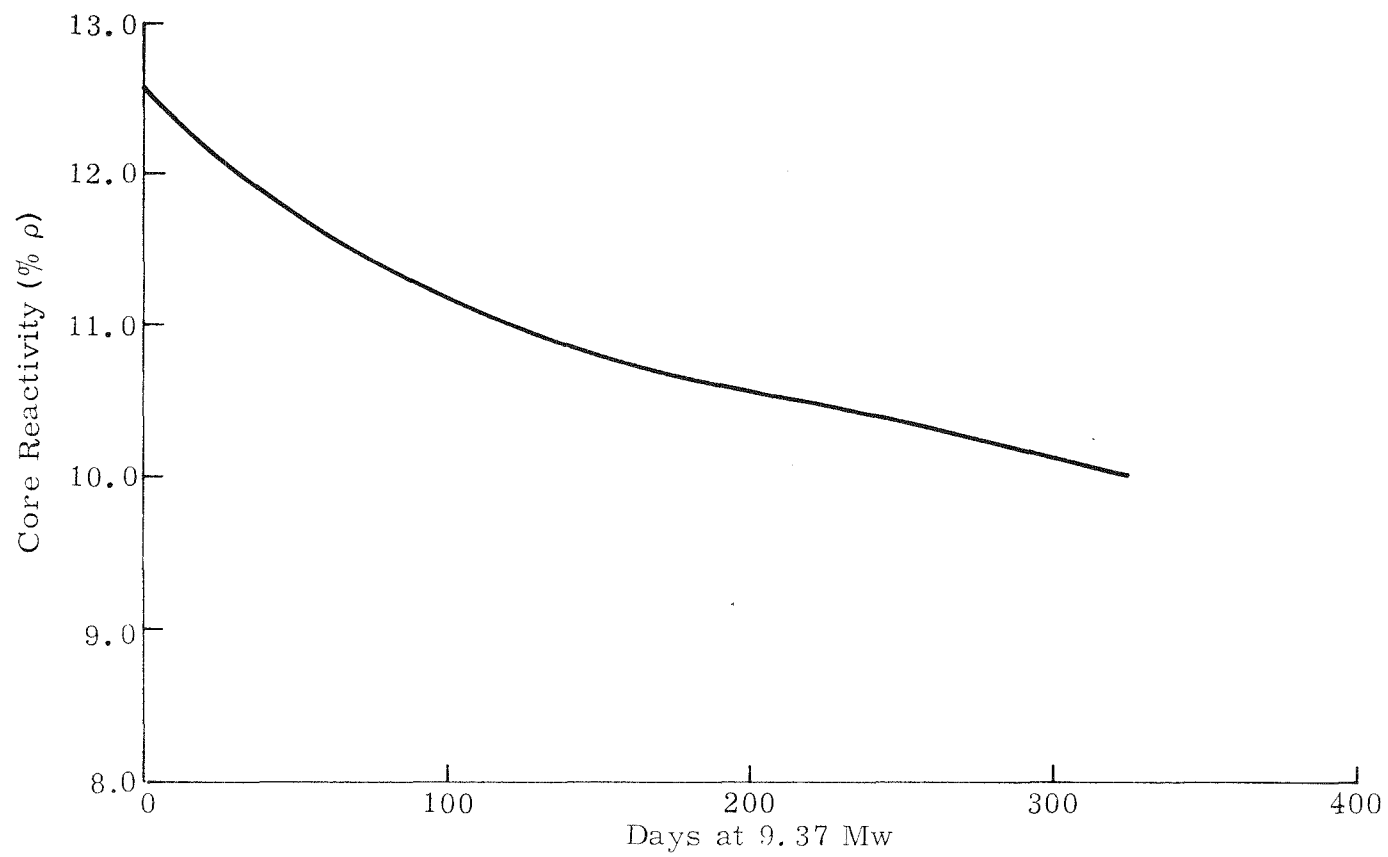


Fig. 46. PM-1 Cold Clean Reactivity Versus Lifetime

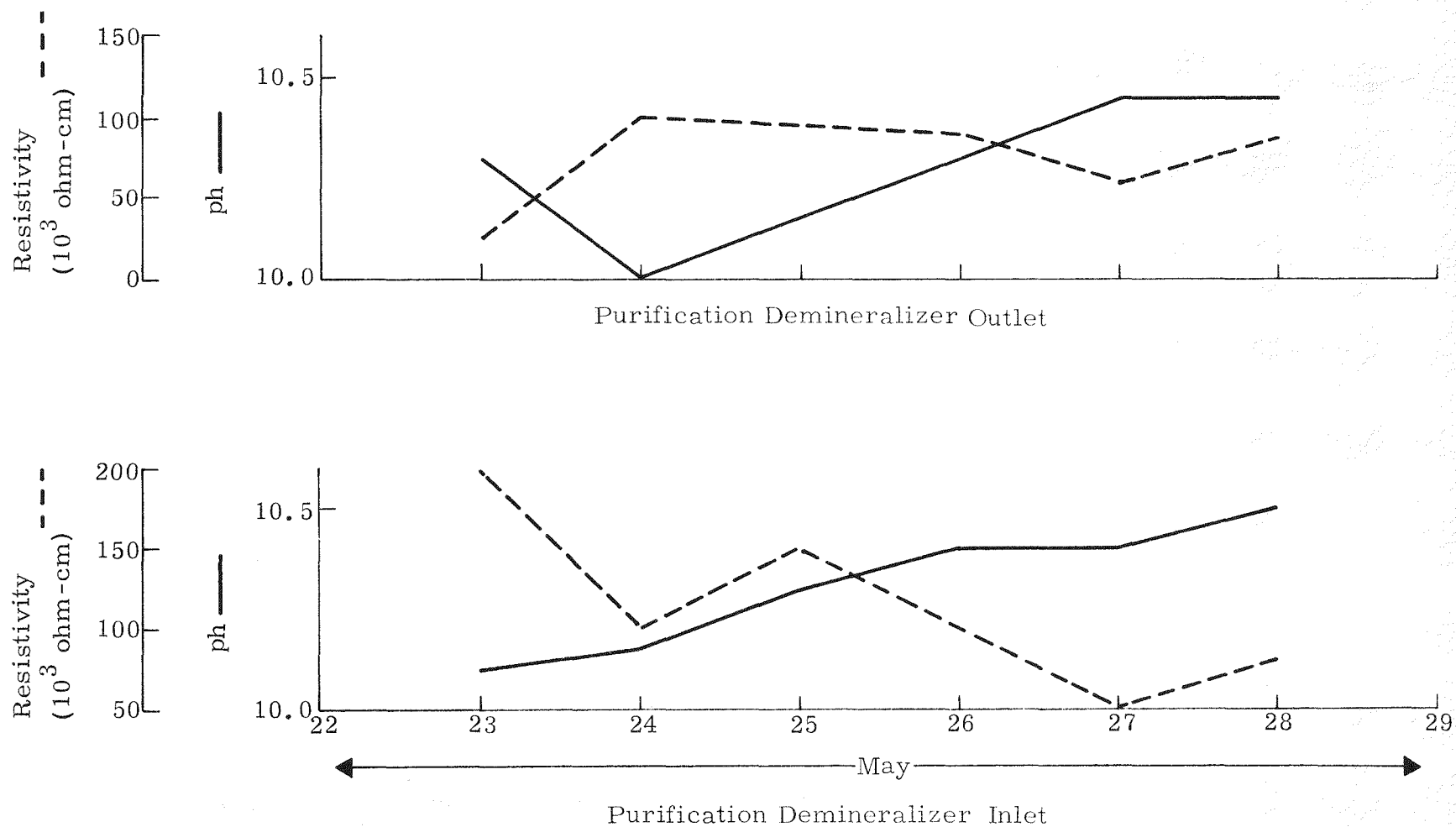


Fig. 47. Primary System Water Chemistry

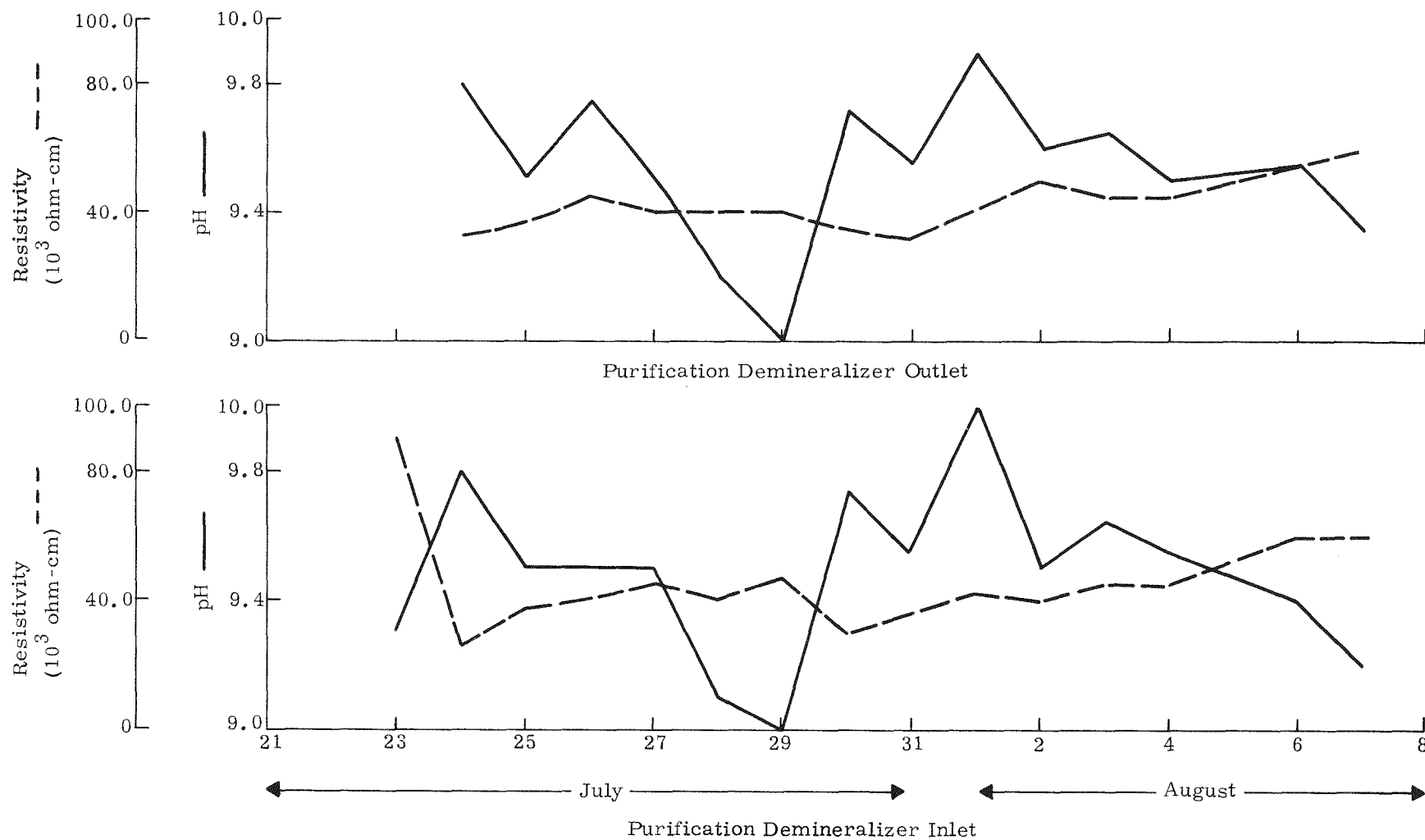


Fig. 48. Primary System Water Chemistry

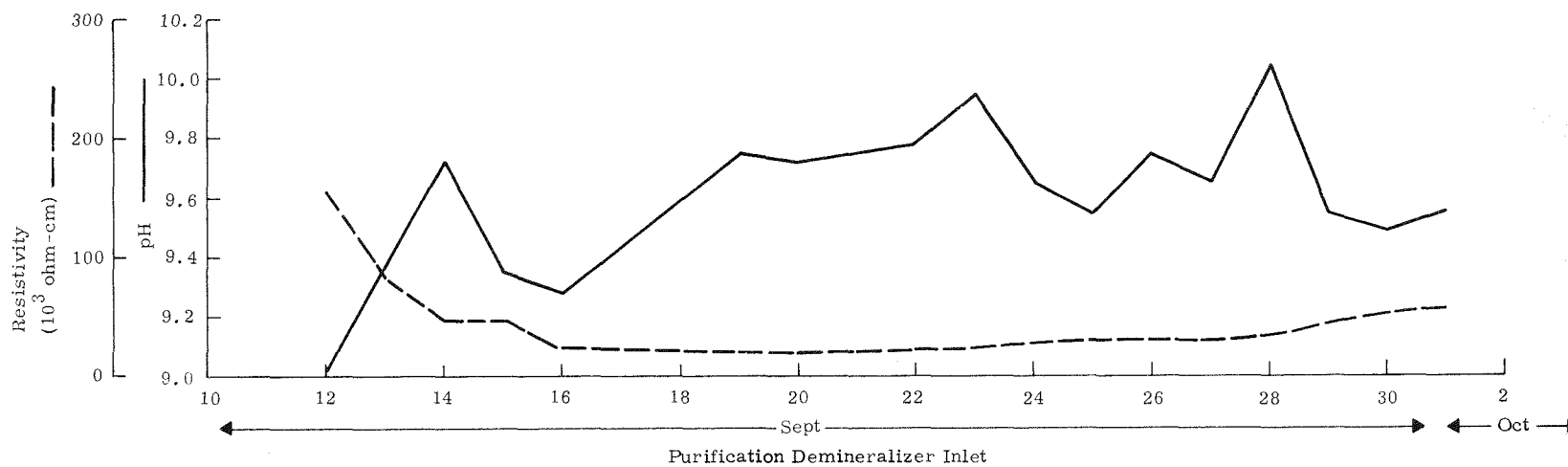
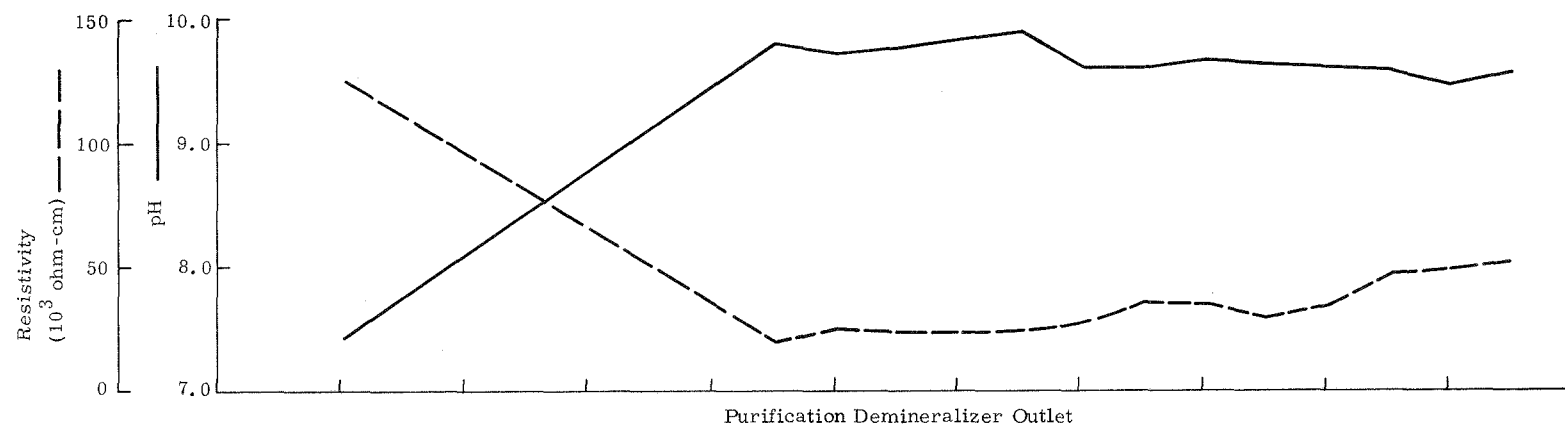


Fig. 49. Primary System Water Chemistry

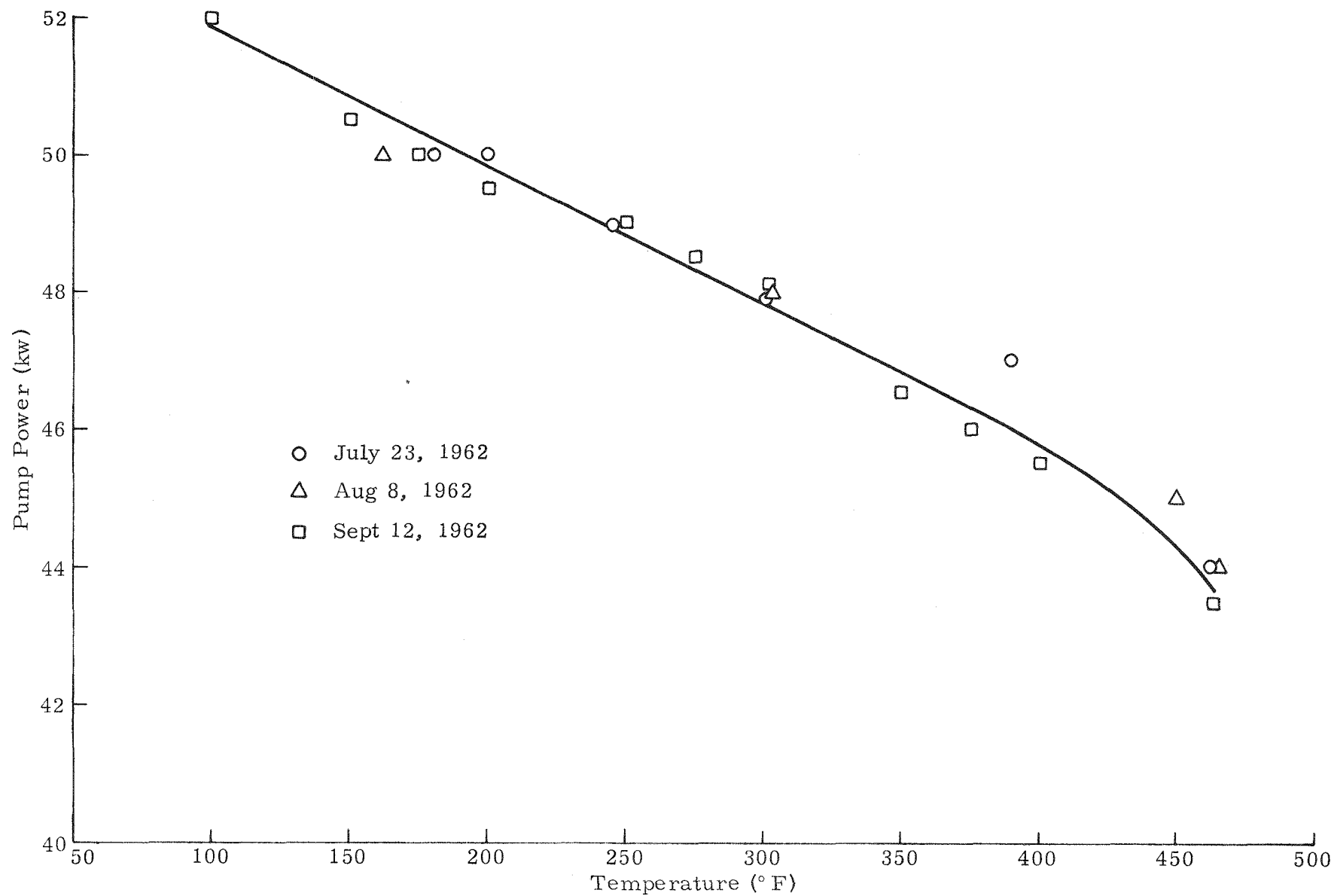
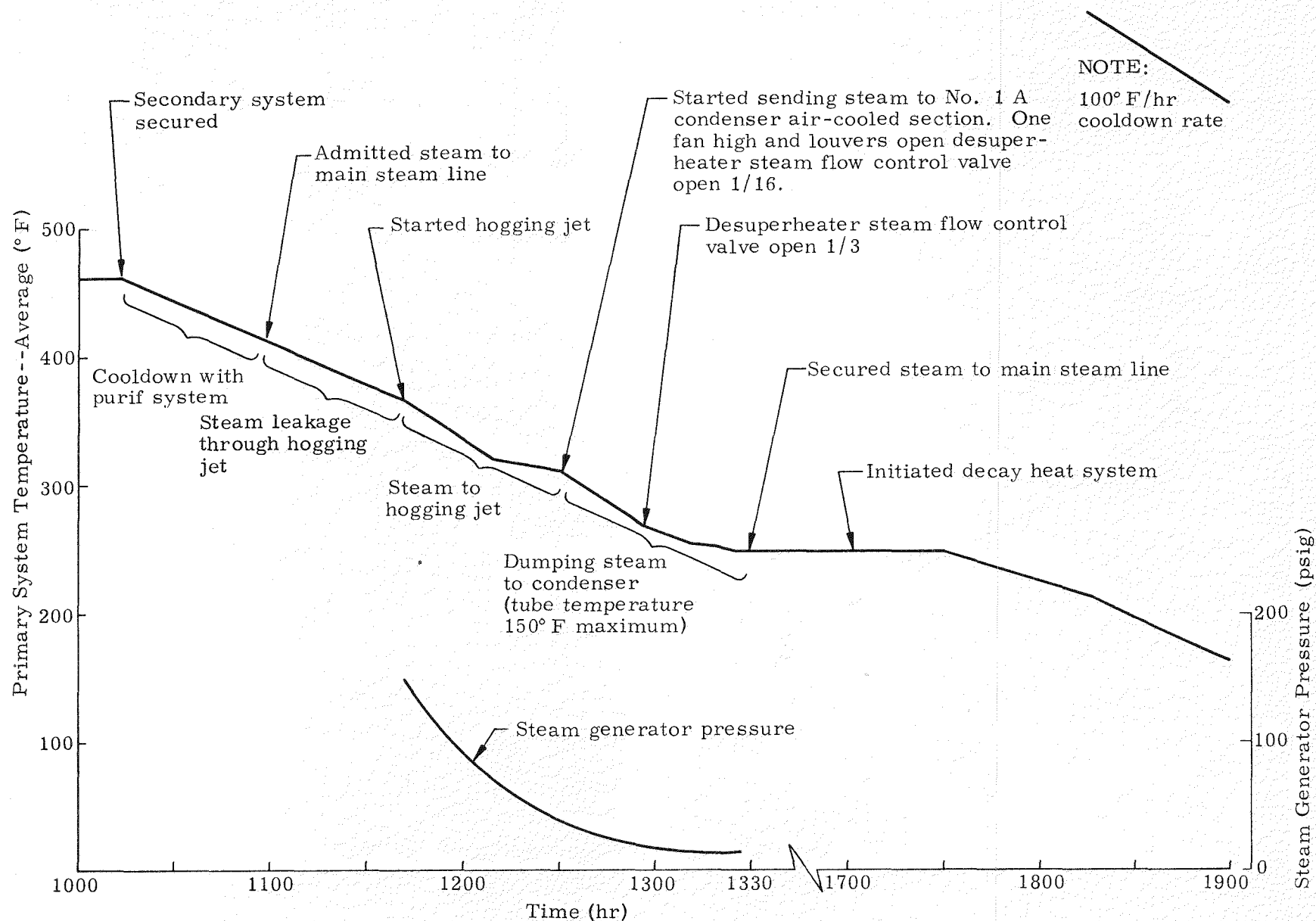


Fig. 50. Primary Coolant Pump Power

Fig. 51. PM-1 Decay Heat Removal Operation



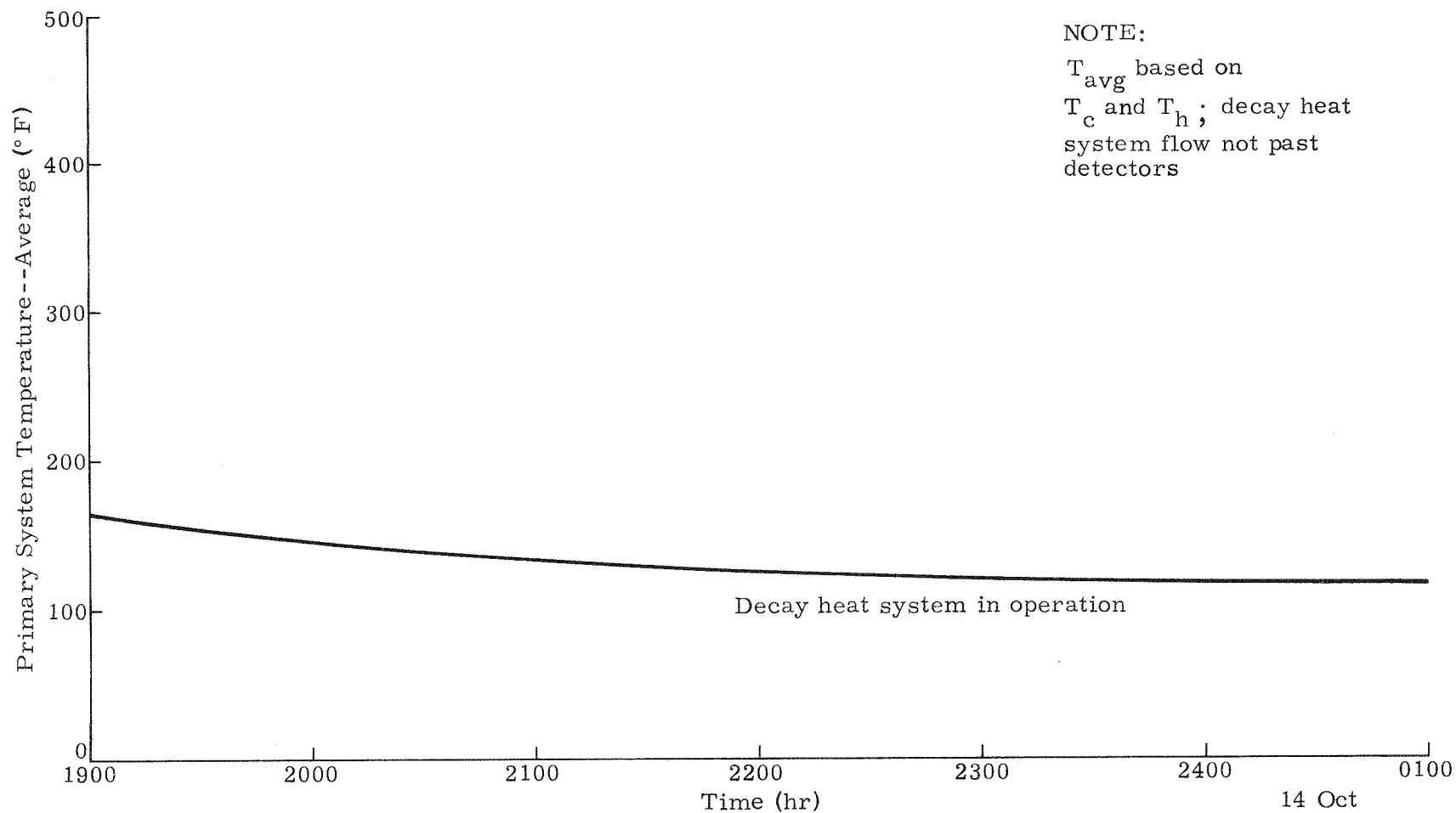


Fig. 51. (continued)

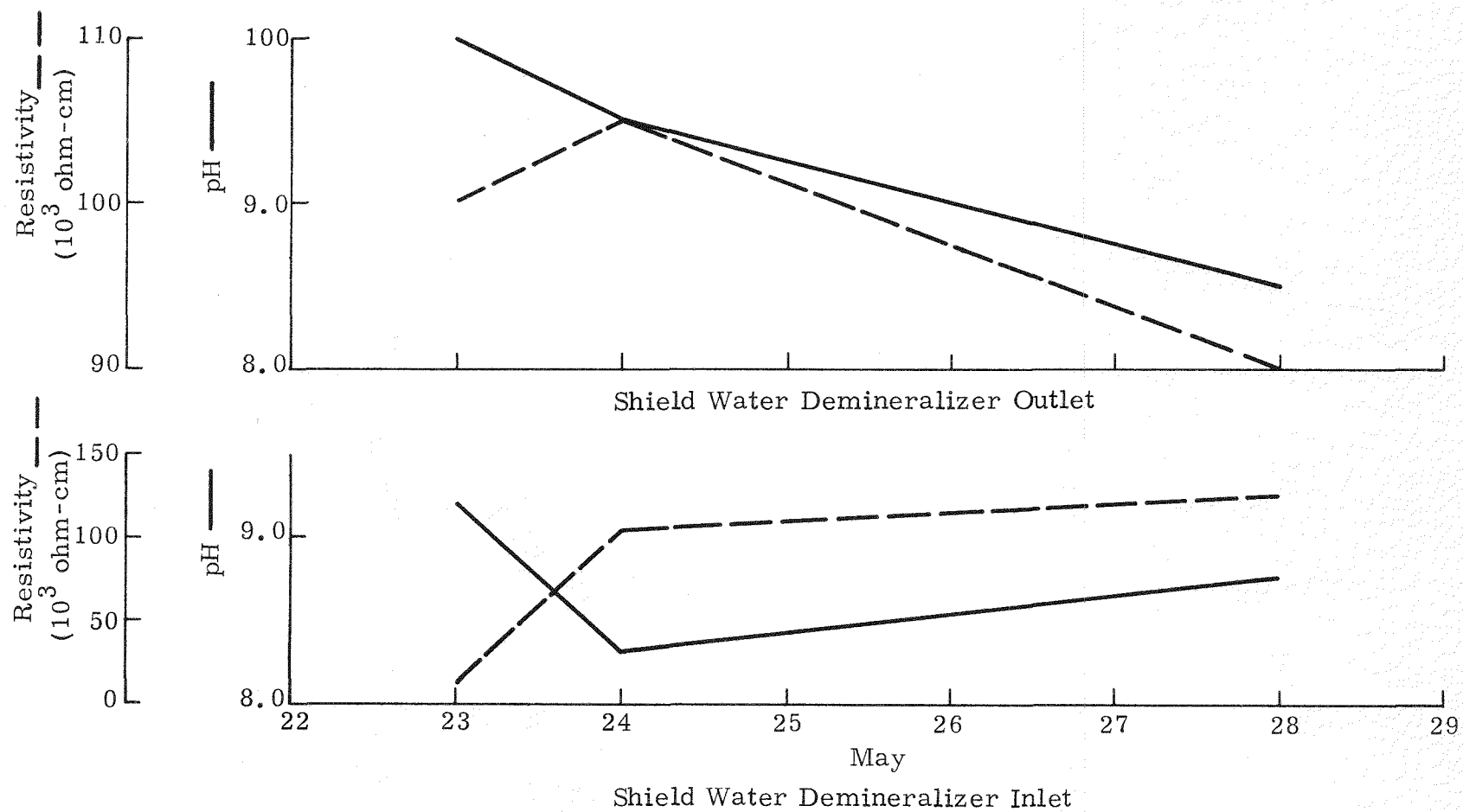


Fig. 52. Shield Water Chemistry

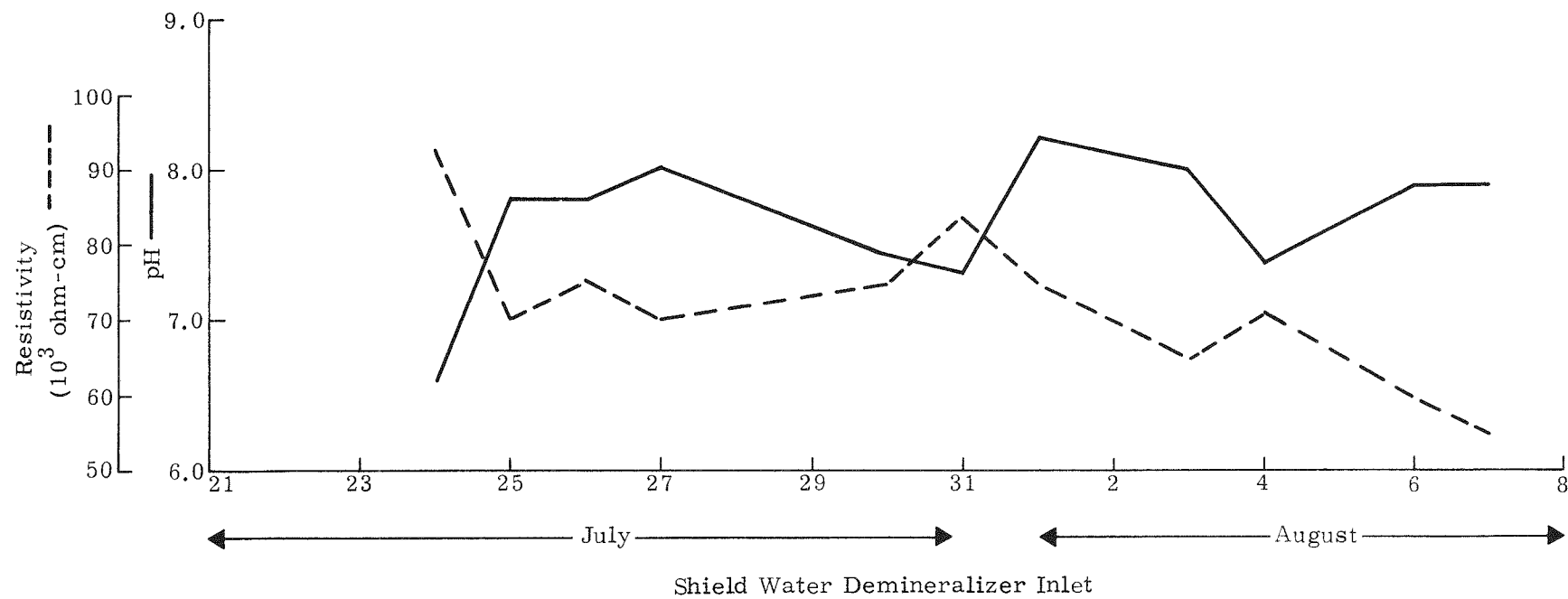


Fig. 53. Shield Water Chemistry

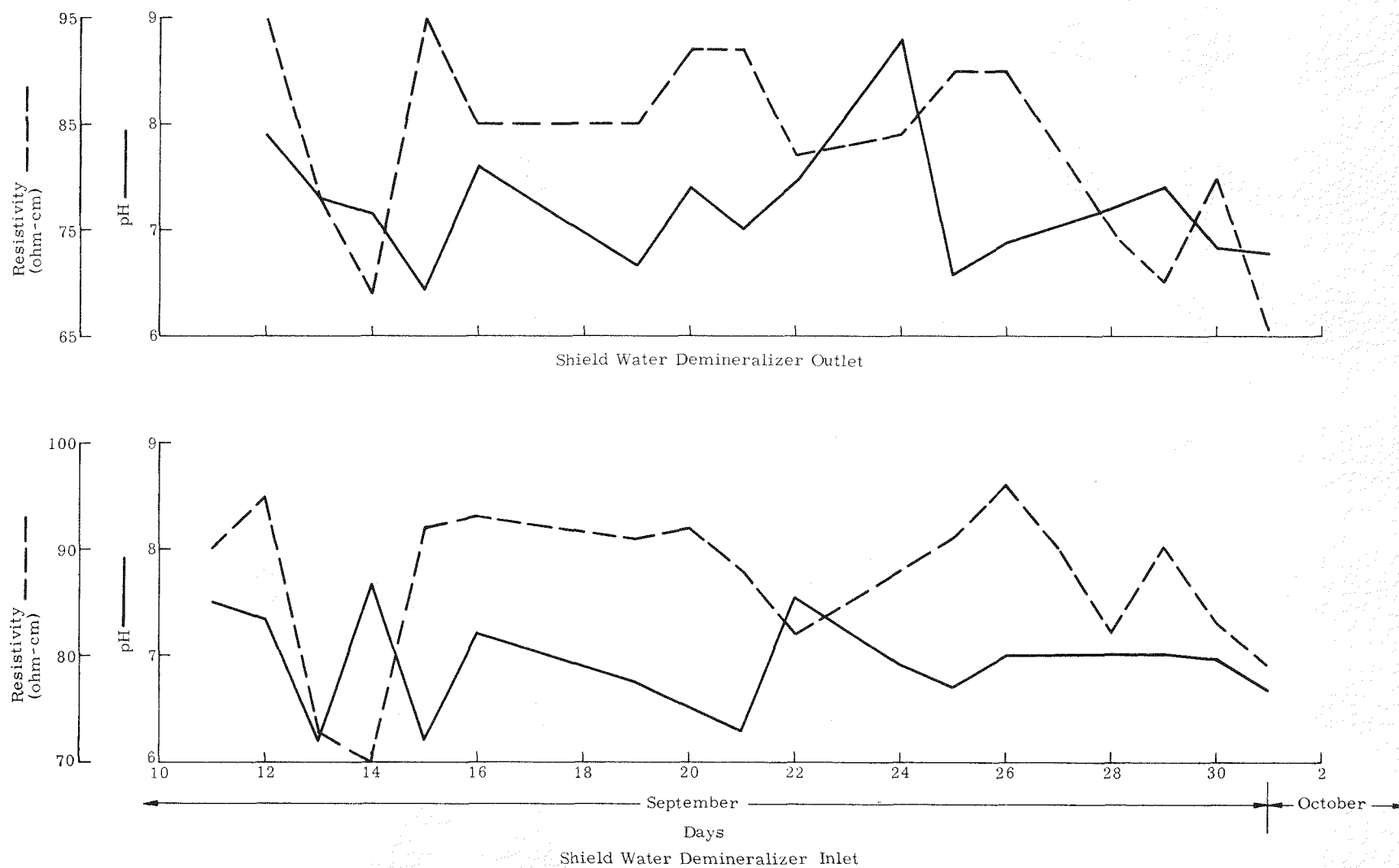


Fig. 54. Shield Water Chemistry

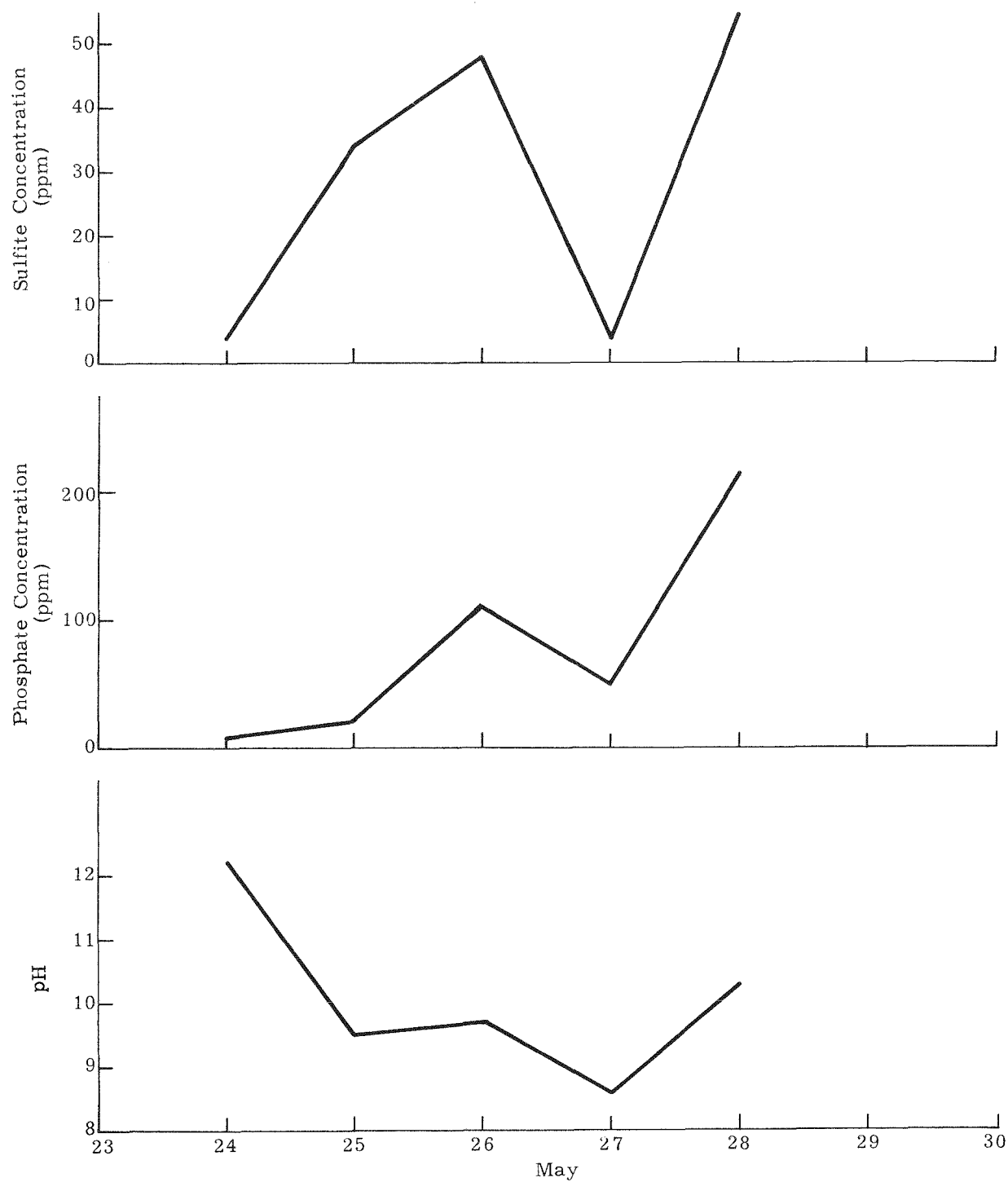


Fig. 55. Secondary Water Chemistry--Steam Generator Blowdown

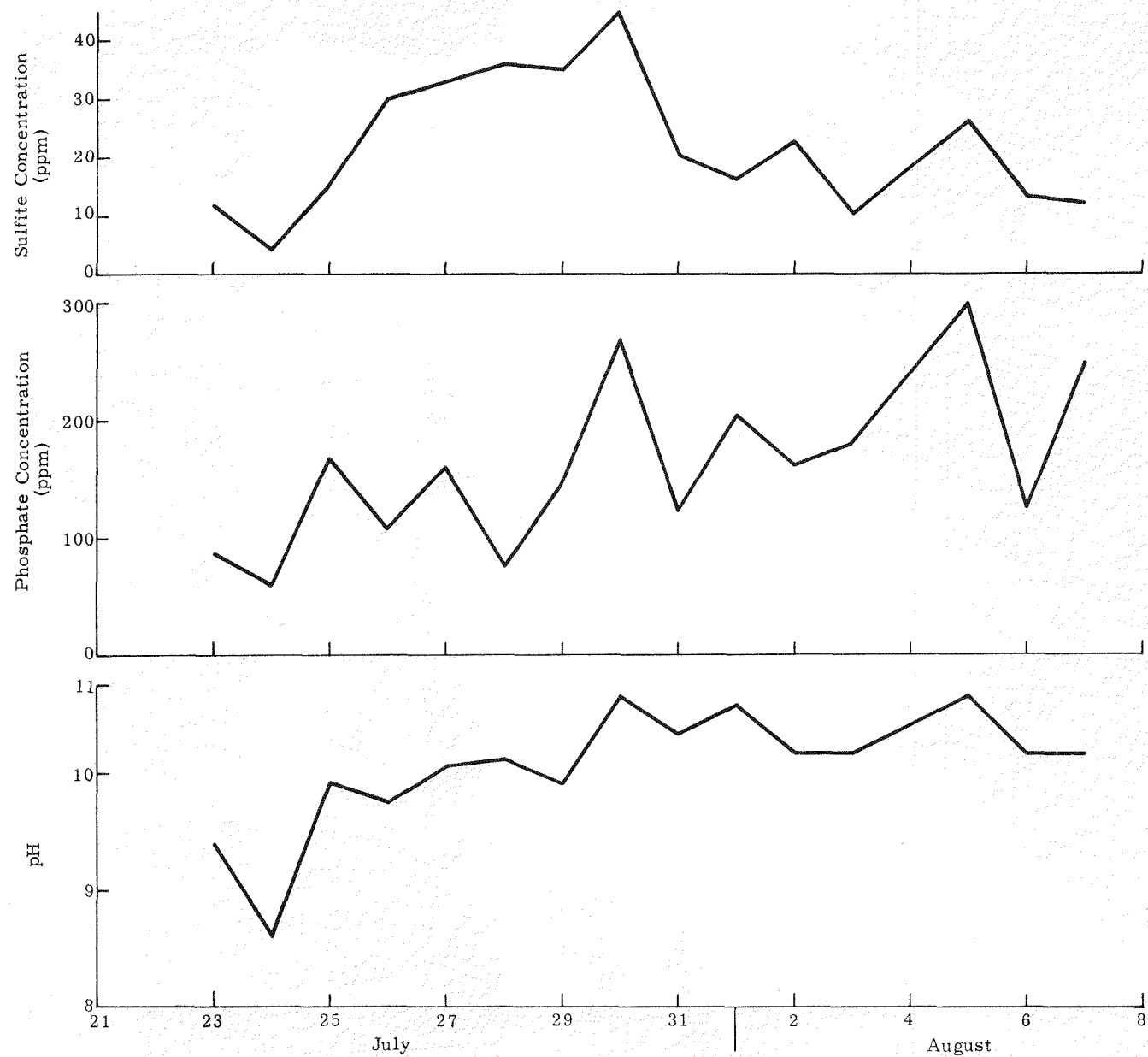


Fig. 56. Secondary Water Chemistry Steam Generator Blowdown

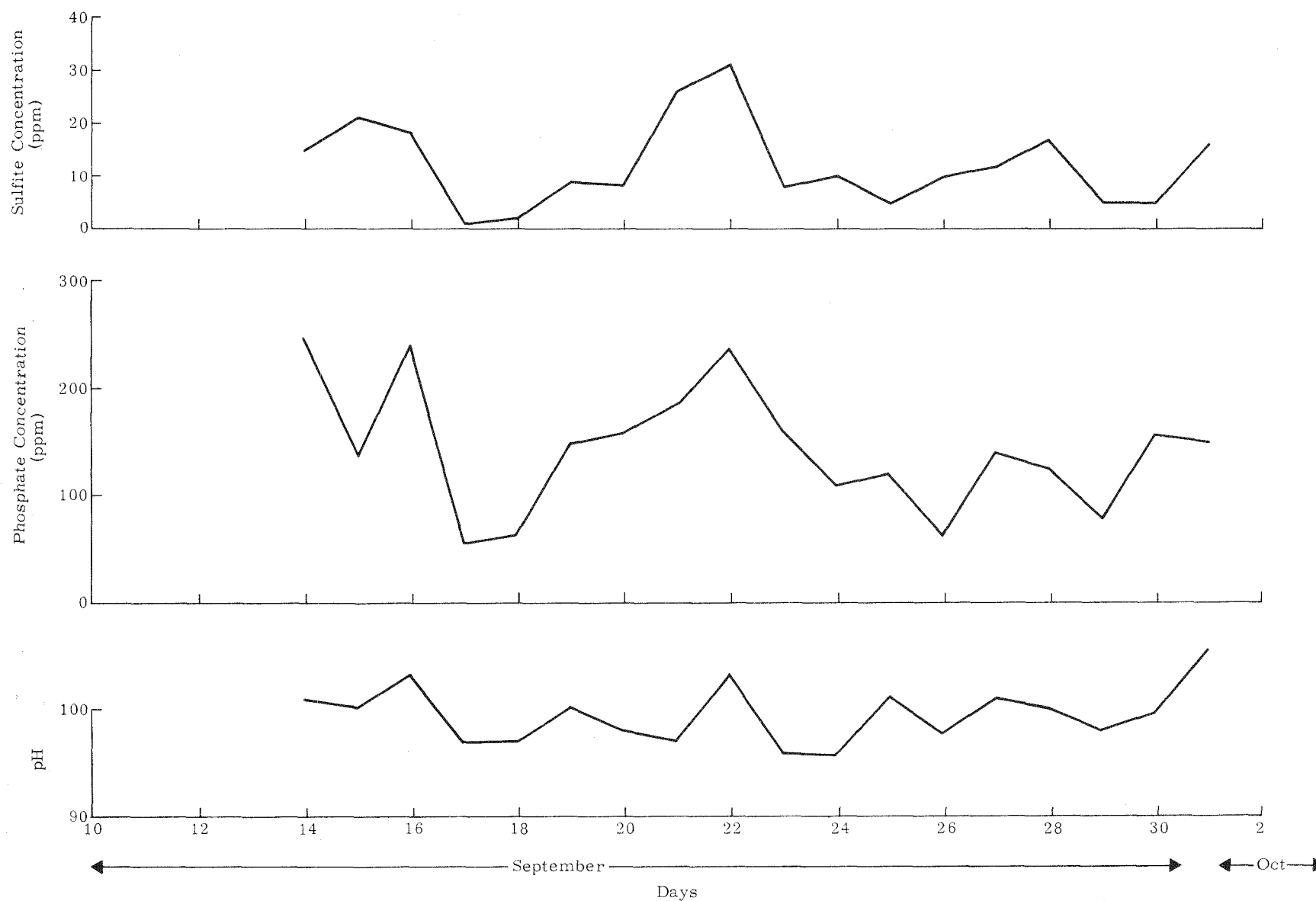


Fig. 57. Secondary Water Chemistry Steam Generator Blowdown

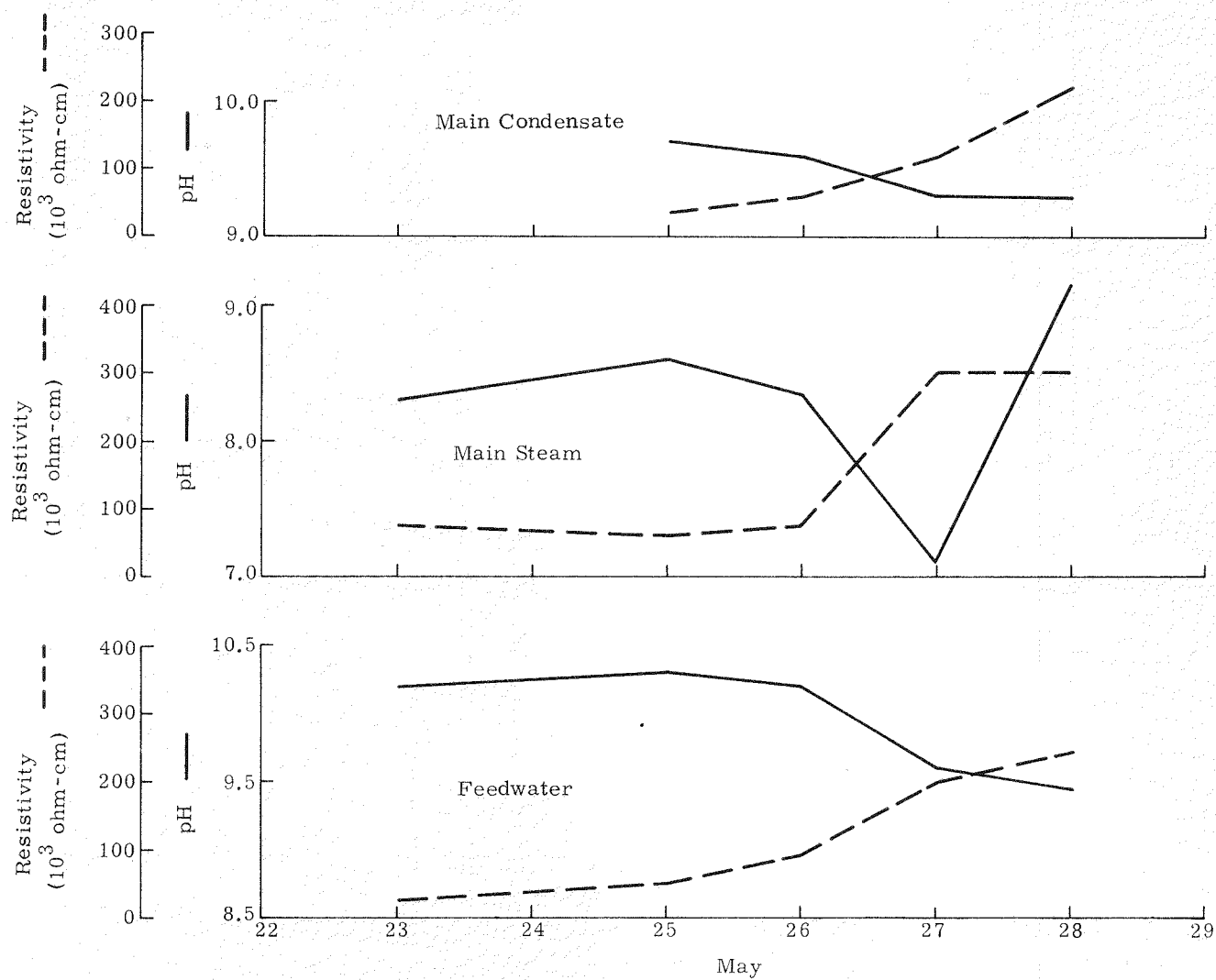
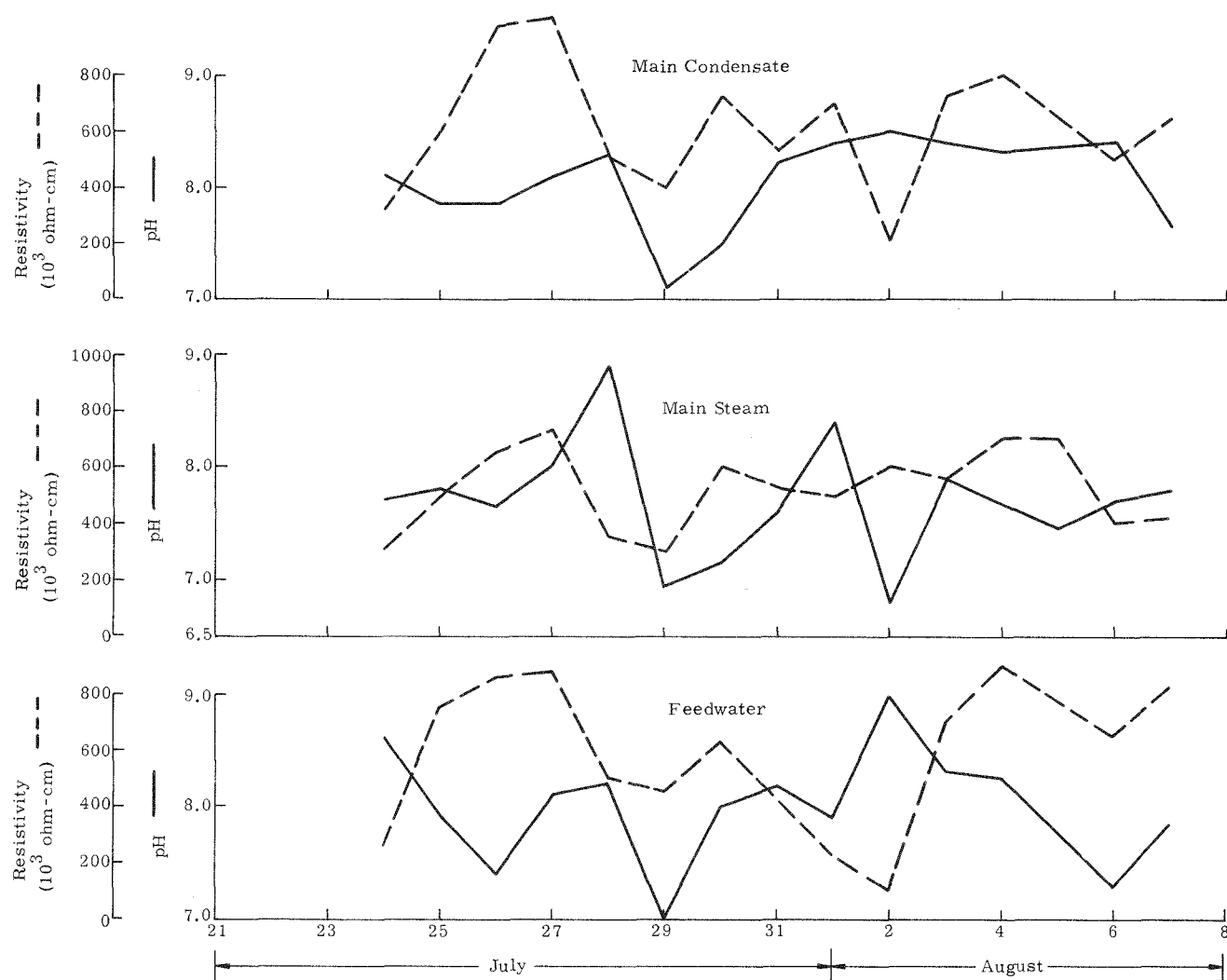


Fig. 58. Secondary System Water Chemistry

Fig. 59. Secondary System Water Chemistry



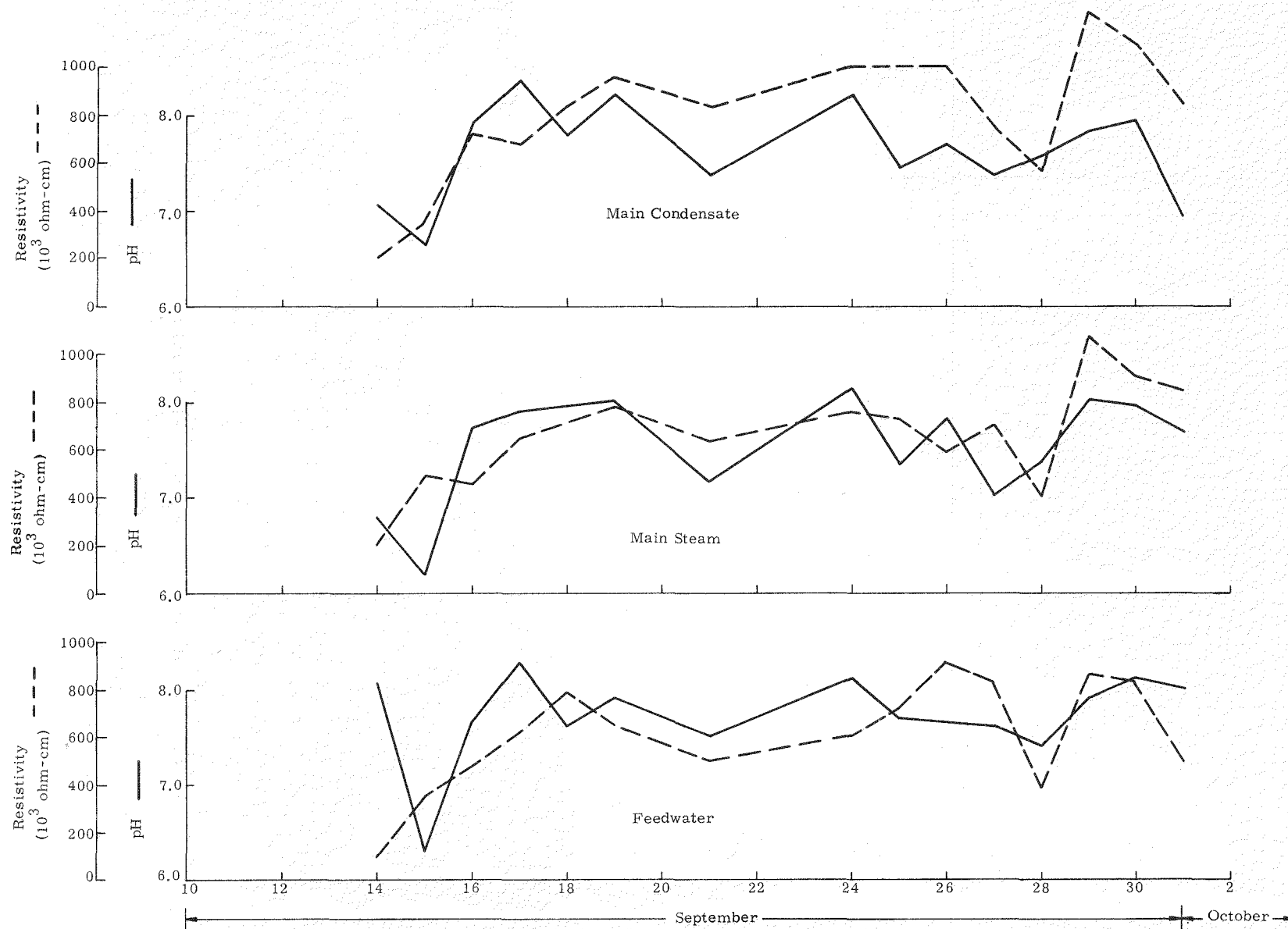
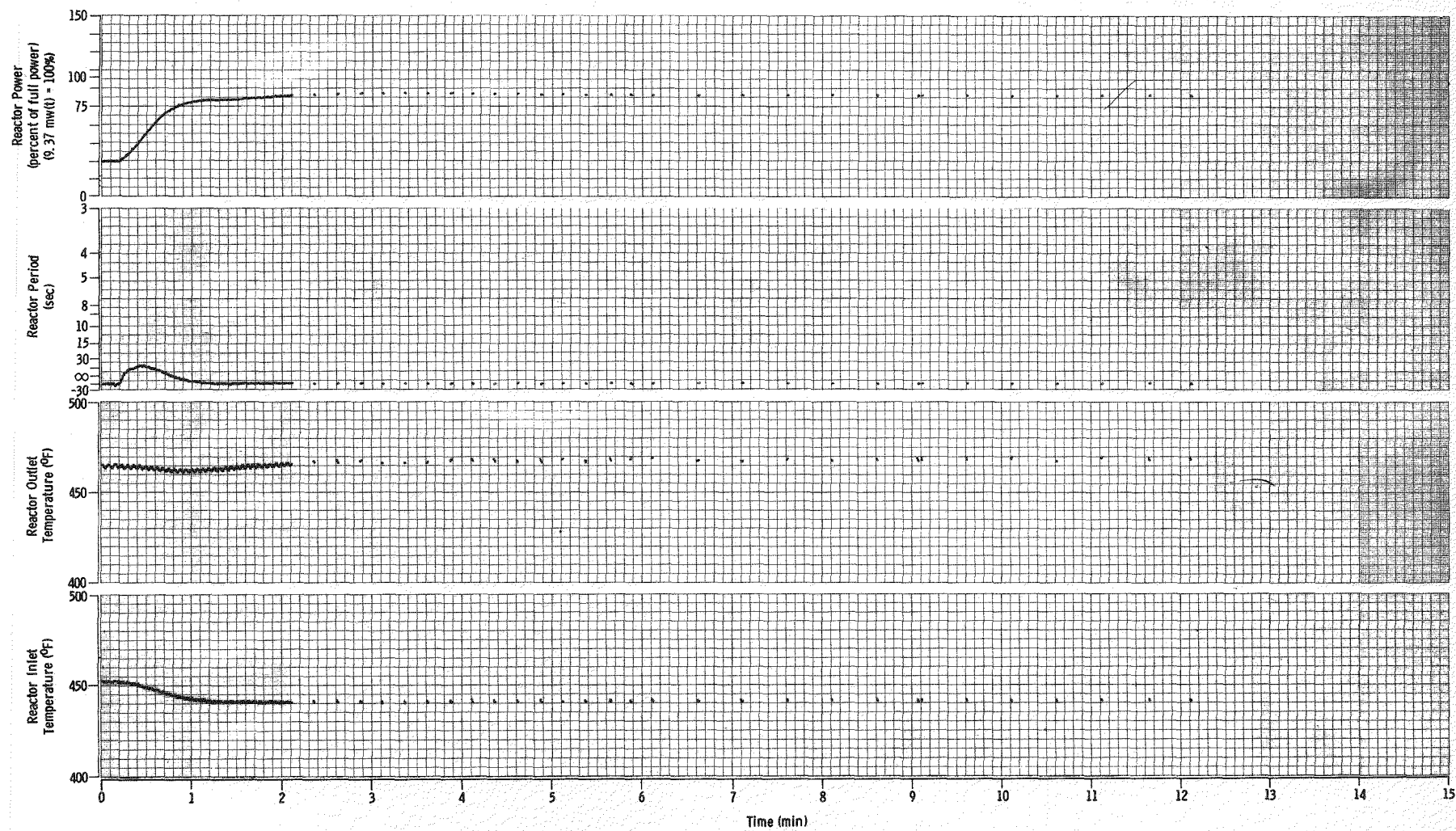


Fig. 60. Secondary System Water Chemistry

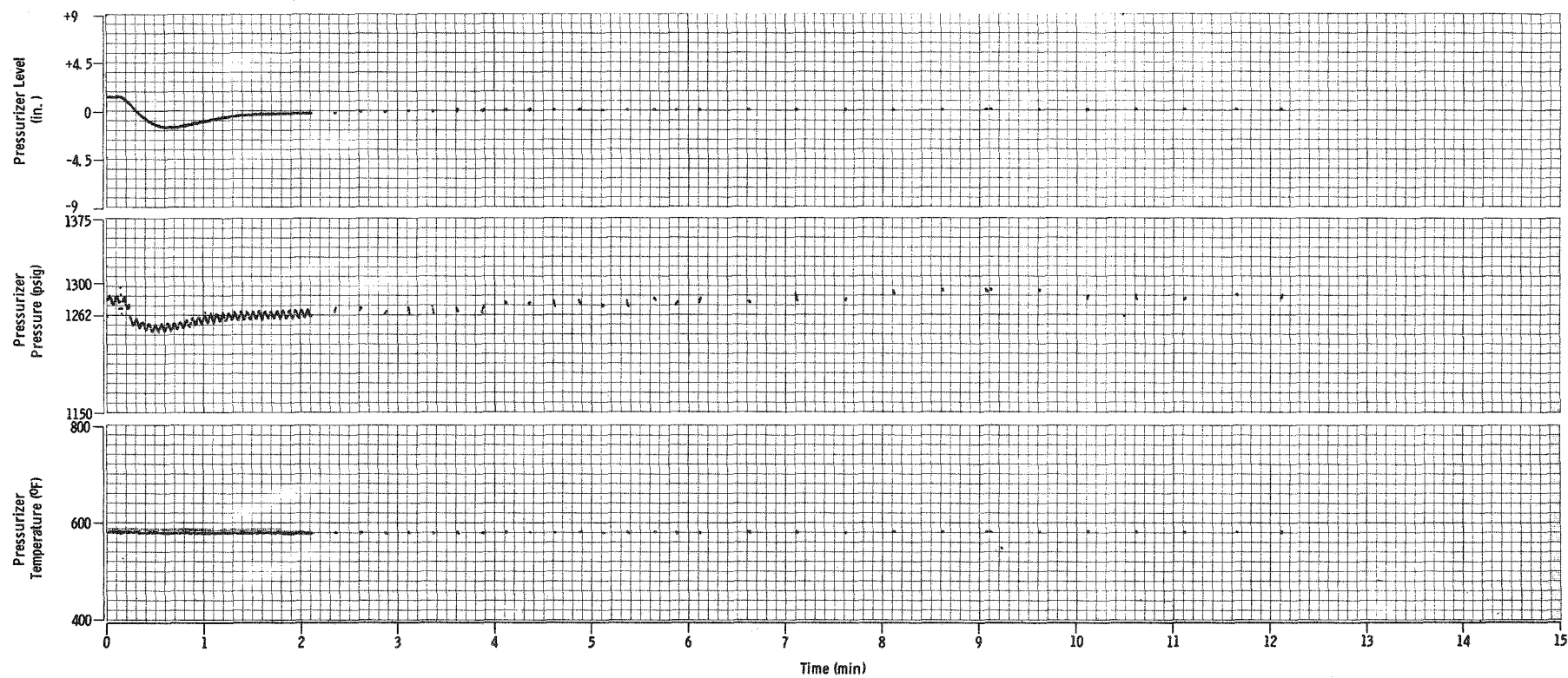
APPENDIX A

PLANT TRANSIENT DATA



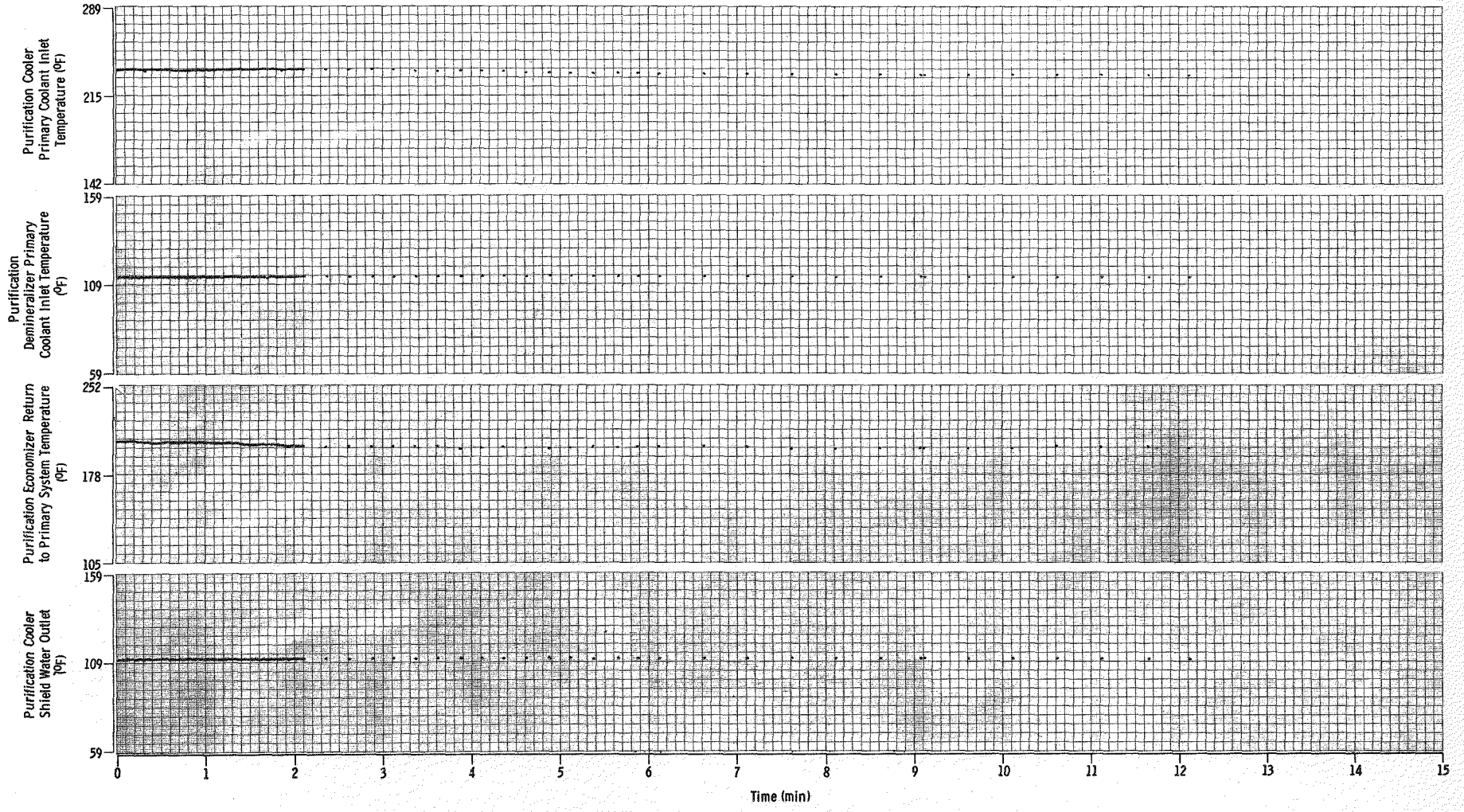
RUN 1--NET LOAD TRANSIENT (0 - 1000 KW)

A-4



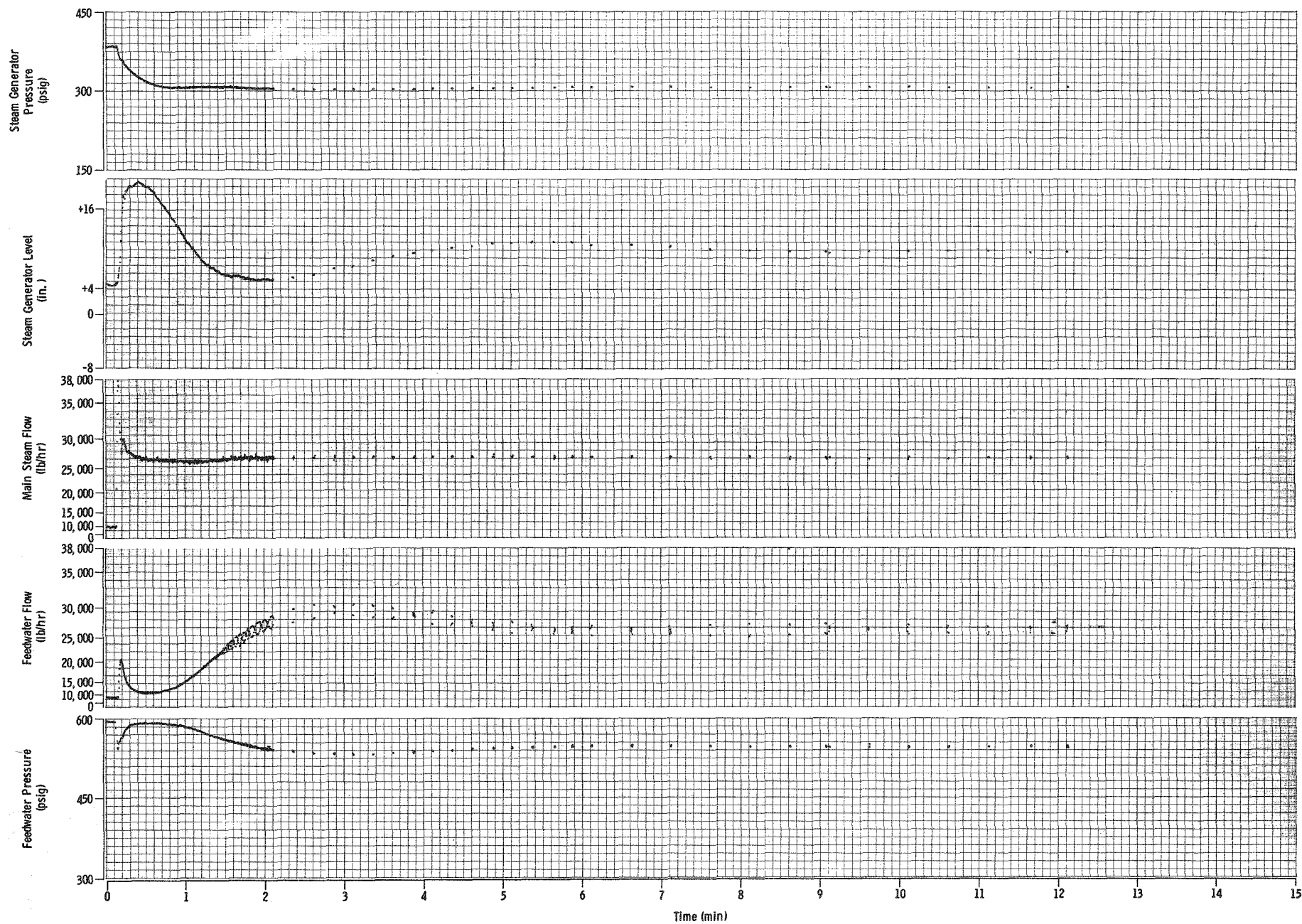
RUN 1--NET LOAD TRANSIENT (0 - 1000 KW) (CONTINUED)

A-5



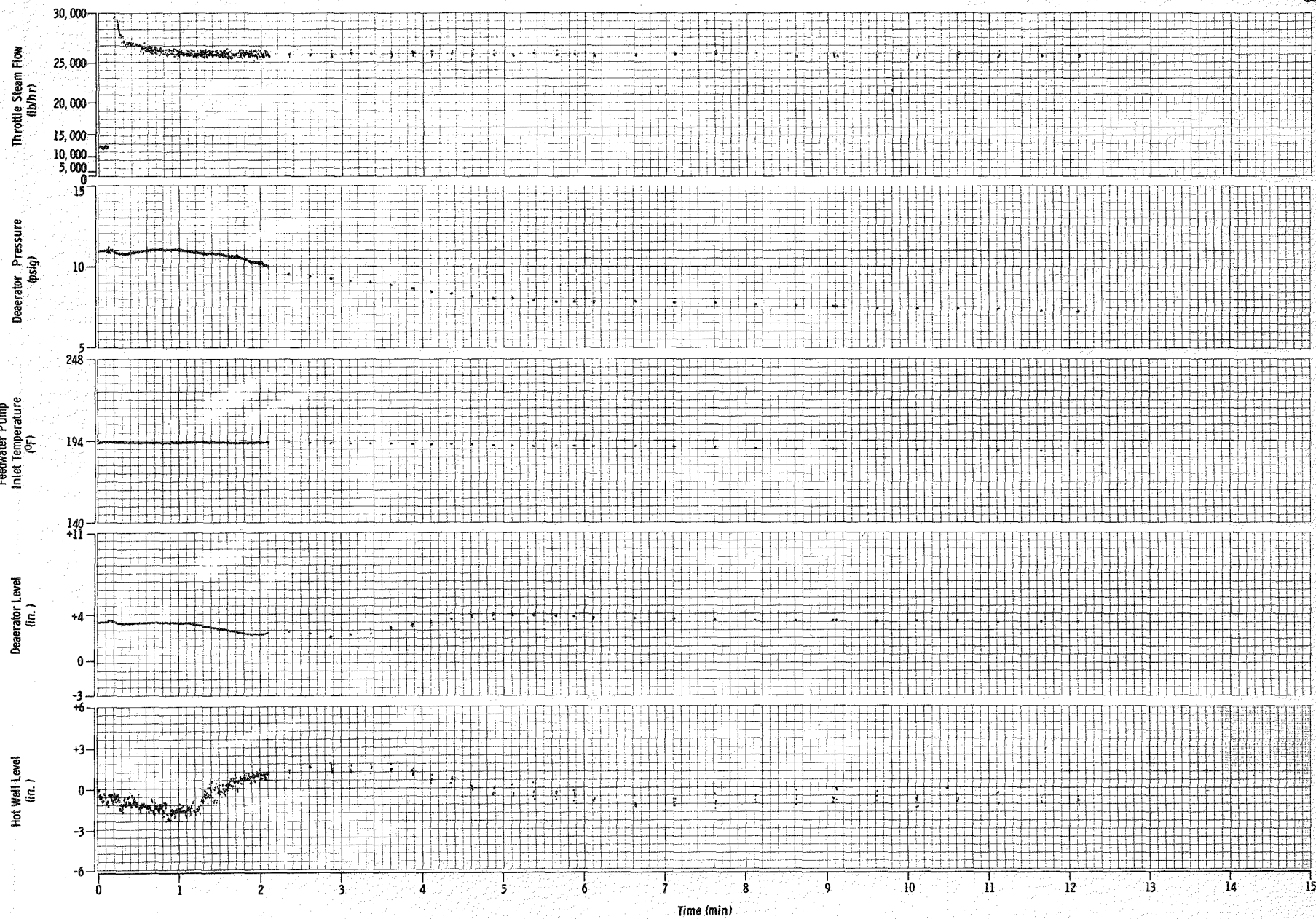
RUN 1--NET LOAD TRANSIENT (0 - 1000 KW) (CONTINUED)

A-6



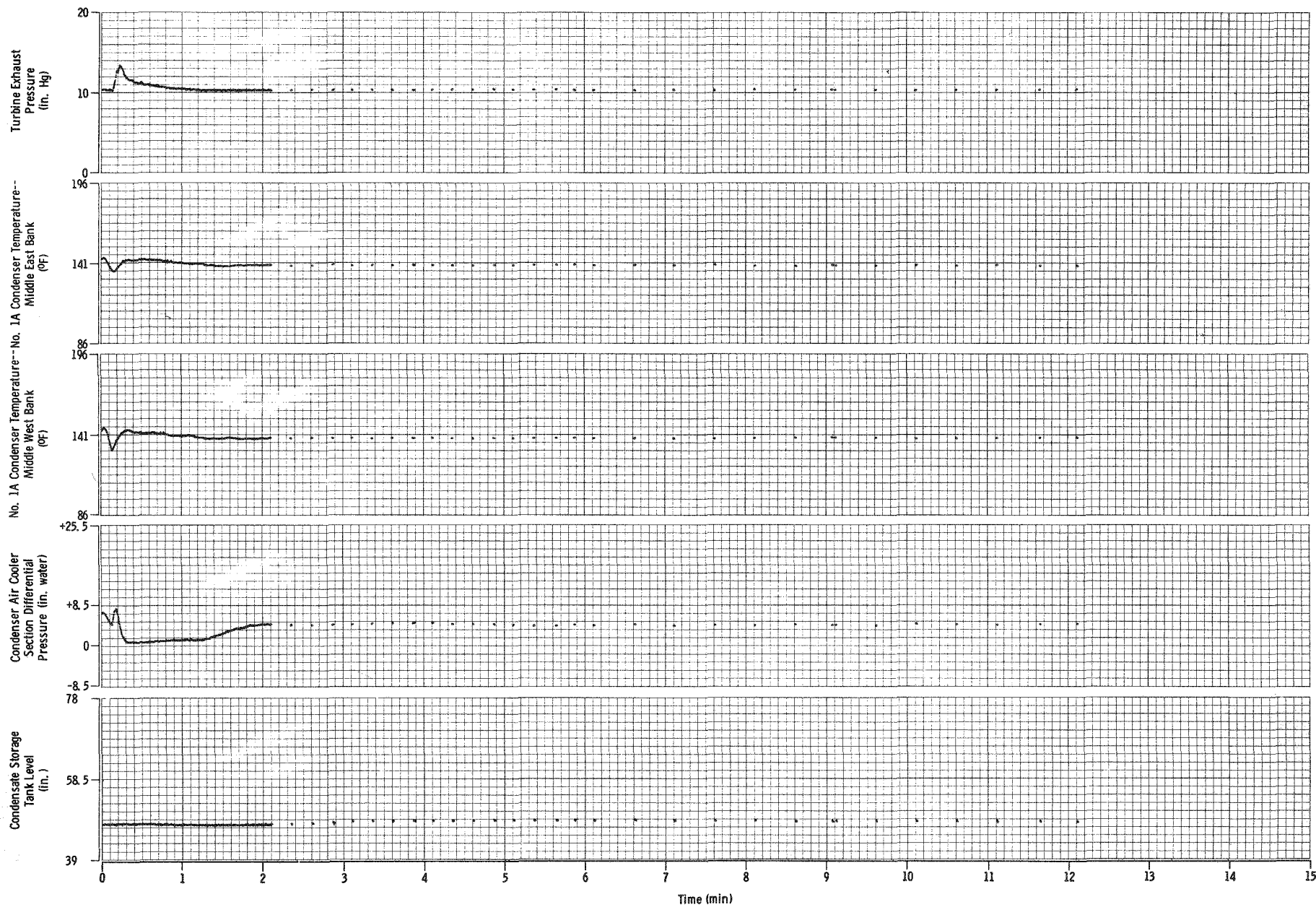
RUN 1--NET LOAD TRANSIENT (0 - 1000 KW) (CONTINUED)

A-7

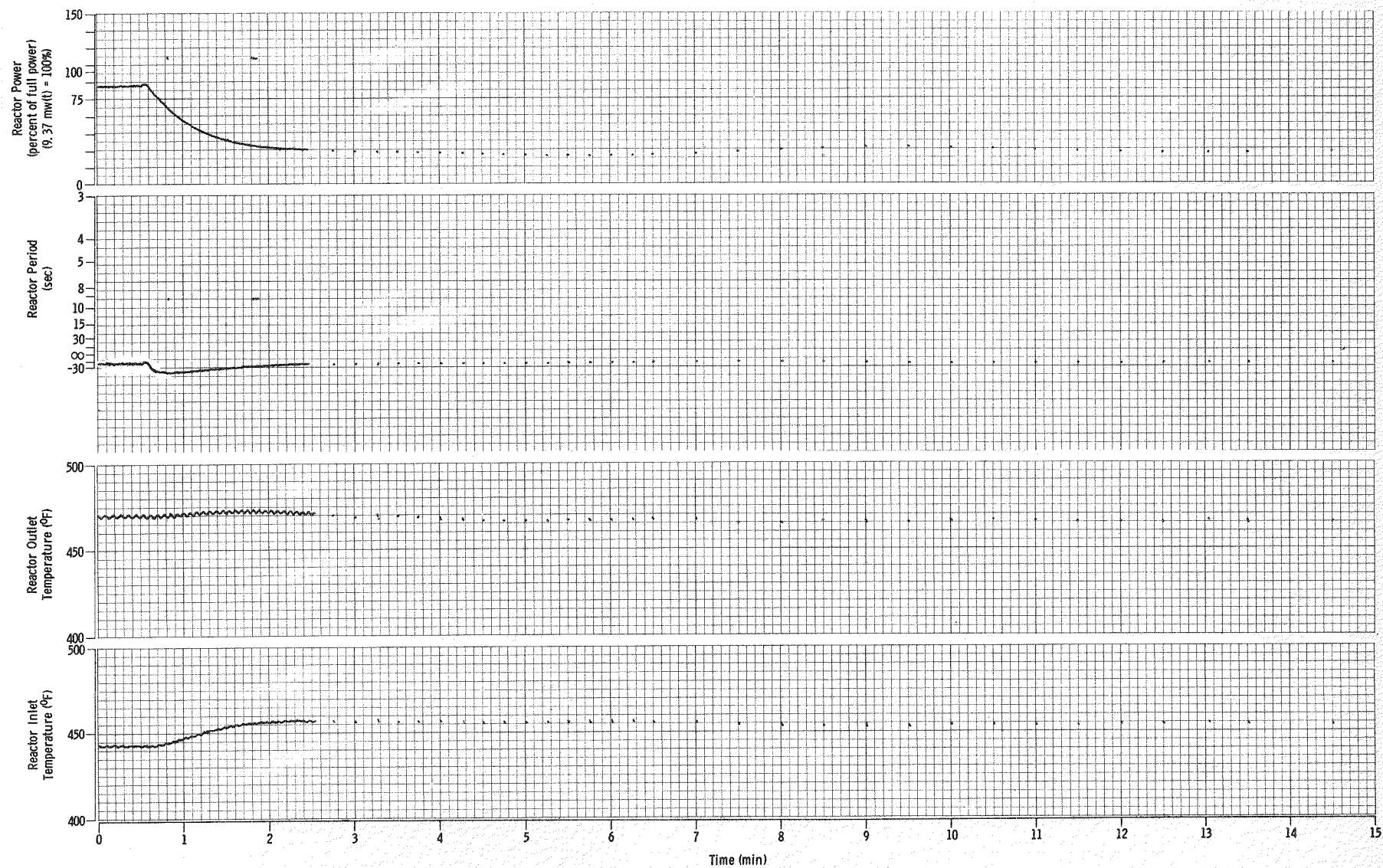


RUN 1--NET LOAD TRANSIENT (0 - 1000 KW) (CONTINUED)

8-V

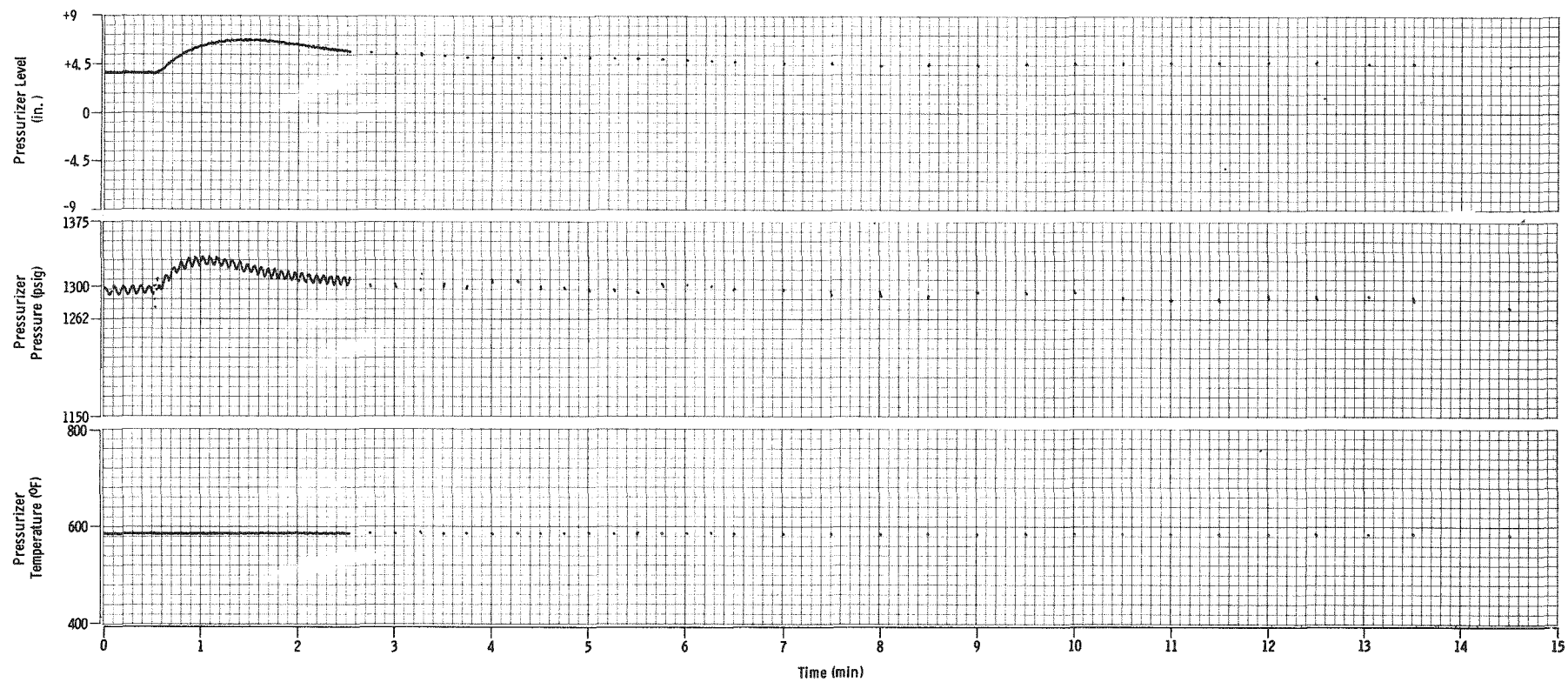


RUN 1--NET LOAD TRANSIENT (0 - 1000 KW) (CONTINUED)



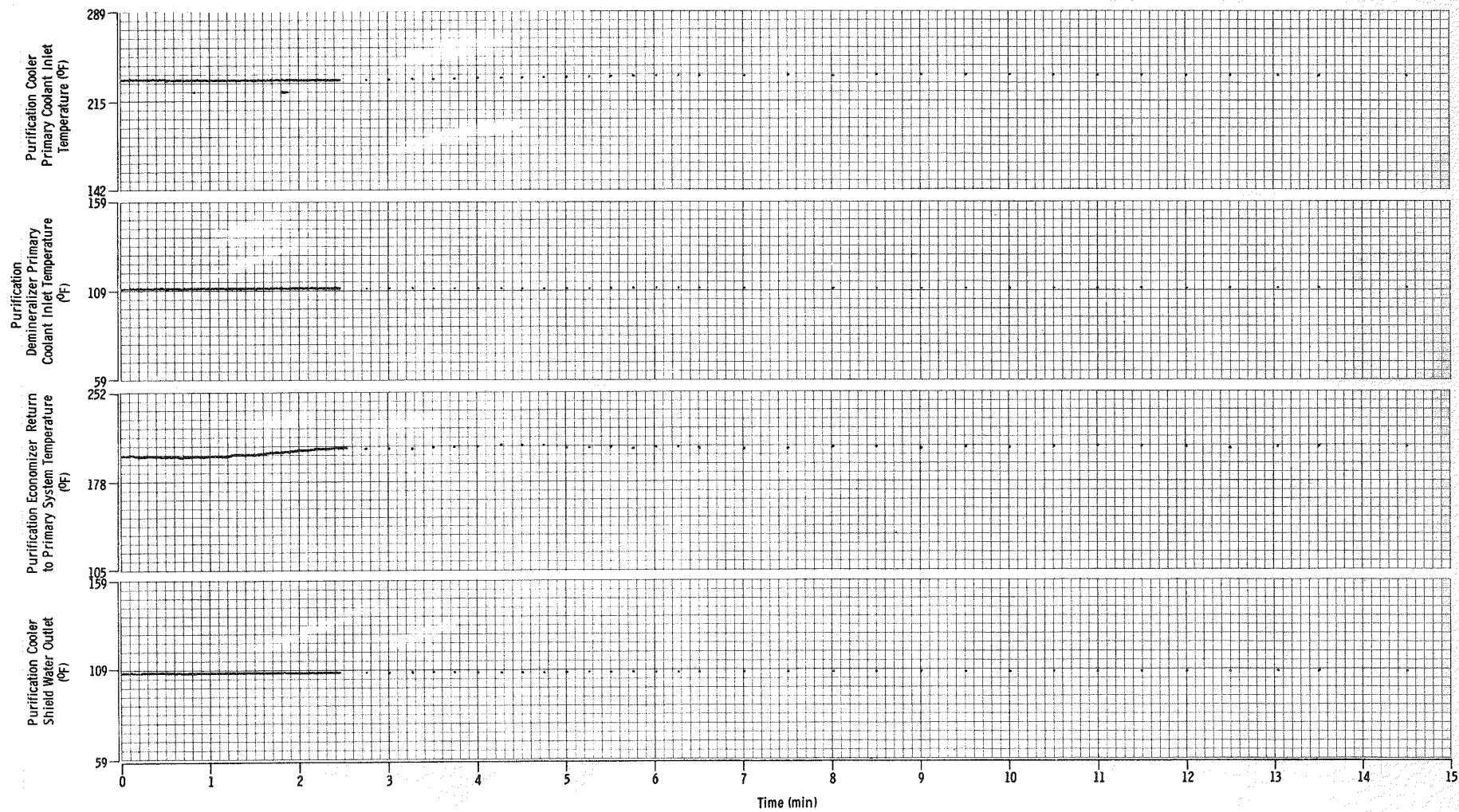
RUN 11--NET LOAD TRANSIENT (1000 - 0 KW)

A-10



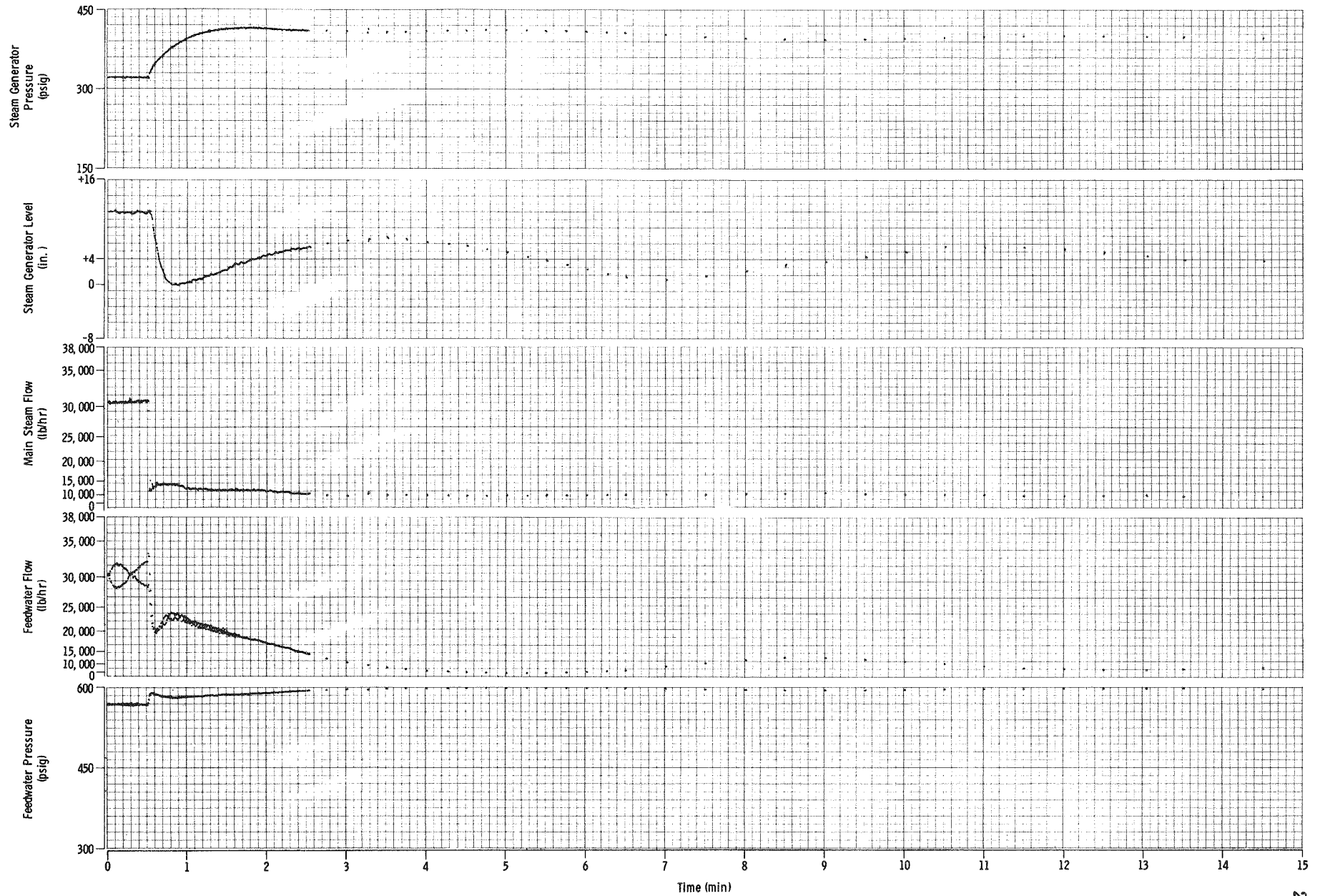
RUN 11--NET LOAD TRANSIENT (1000 - 0 KW) (CONTINUED)

A-11



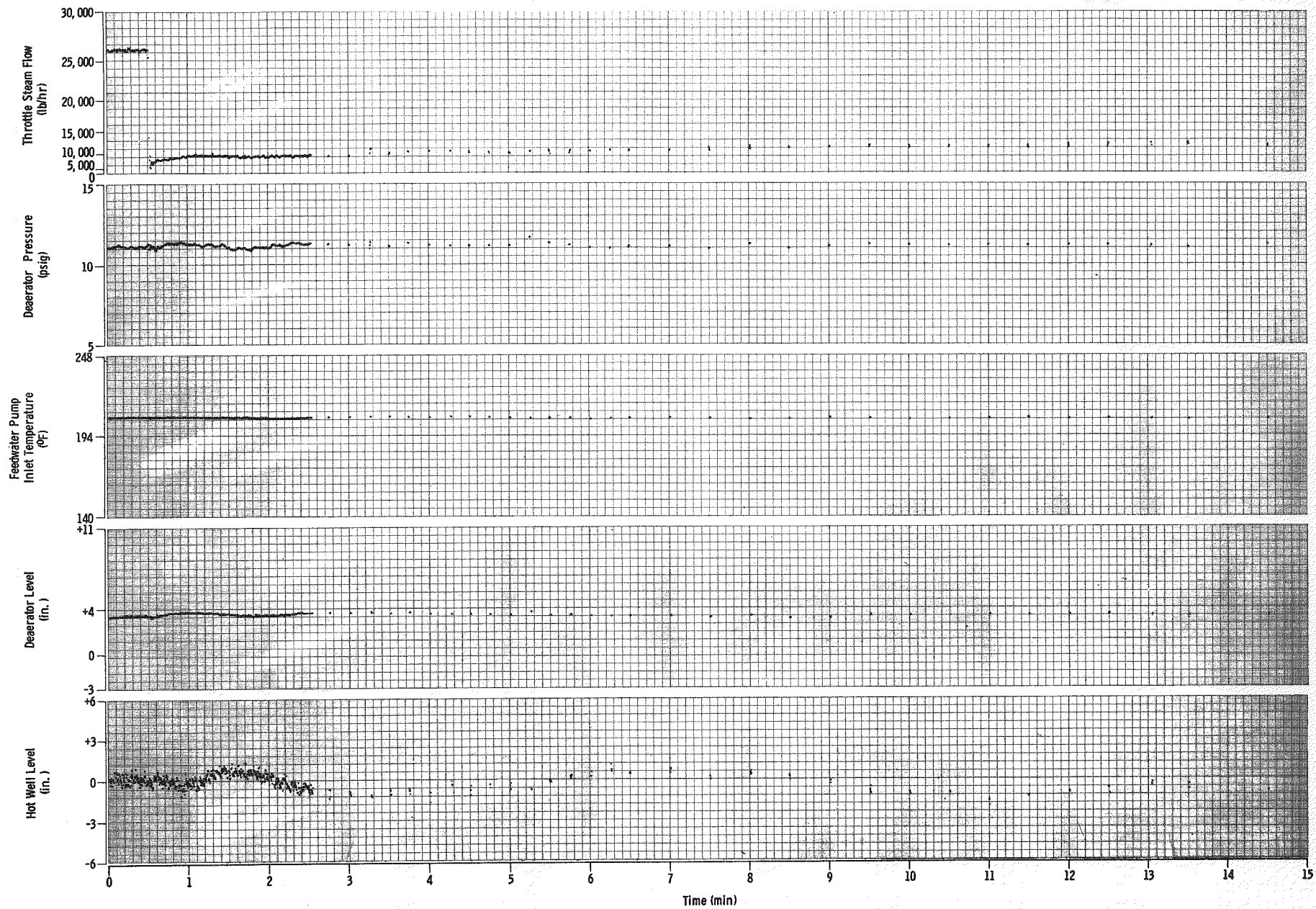
RUN 11--NET LOAD TRANSIENT (1000 - 0 KW) (CONTINUED)

A-12



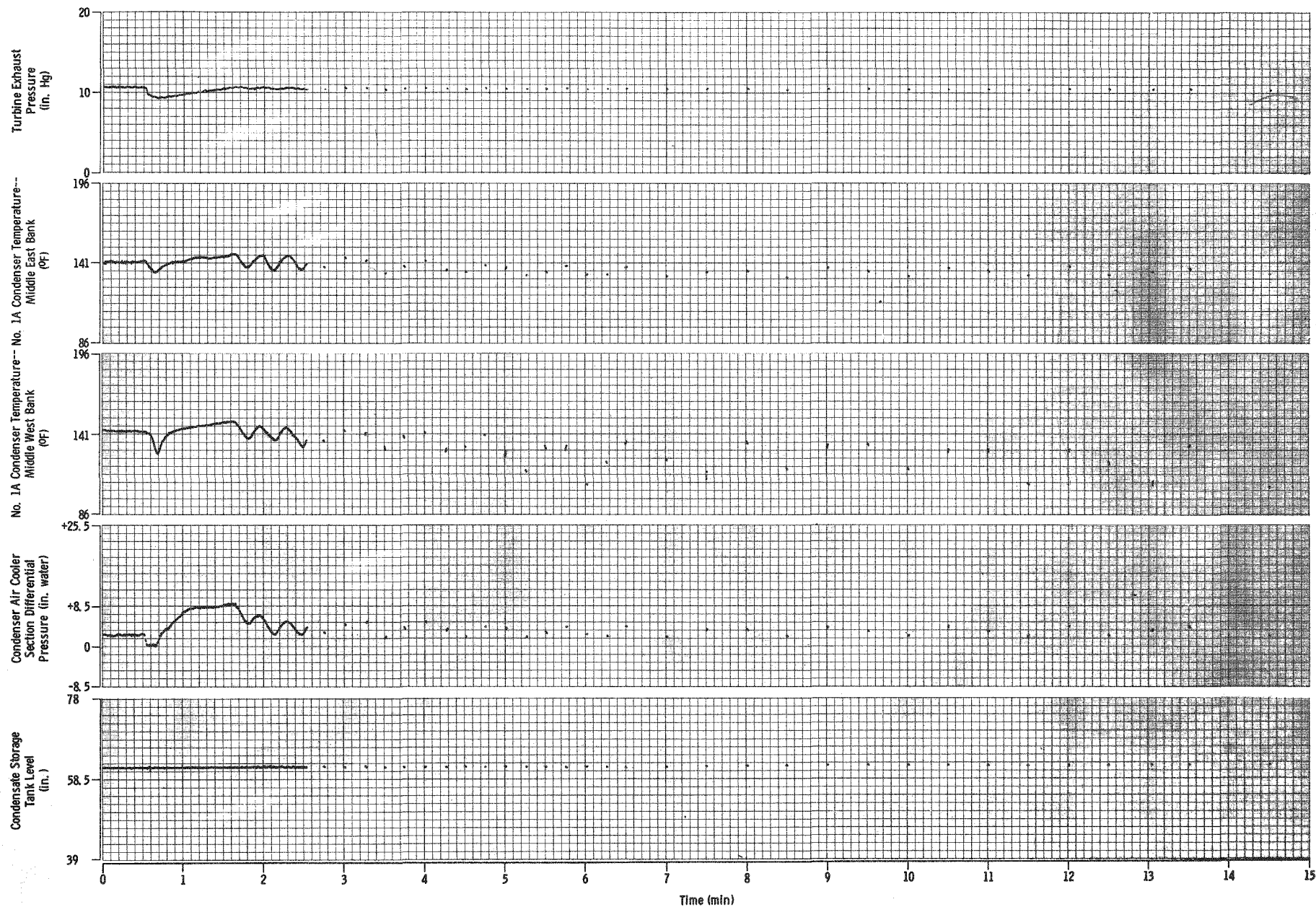
RUN 11--NET LOAD TRANSIENT (1000 - 0 KW) (CONTINUED)

A-13

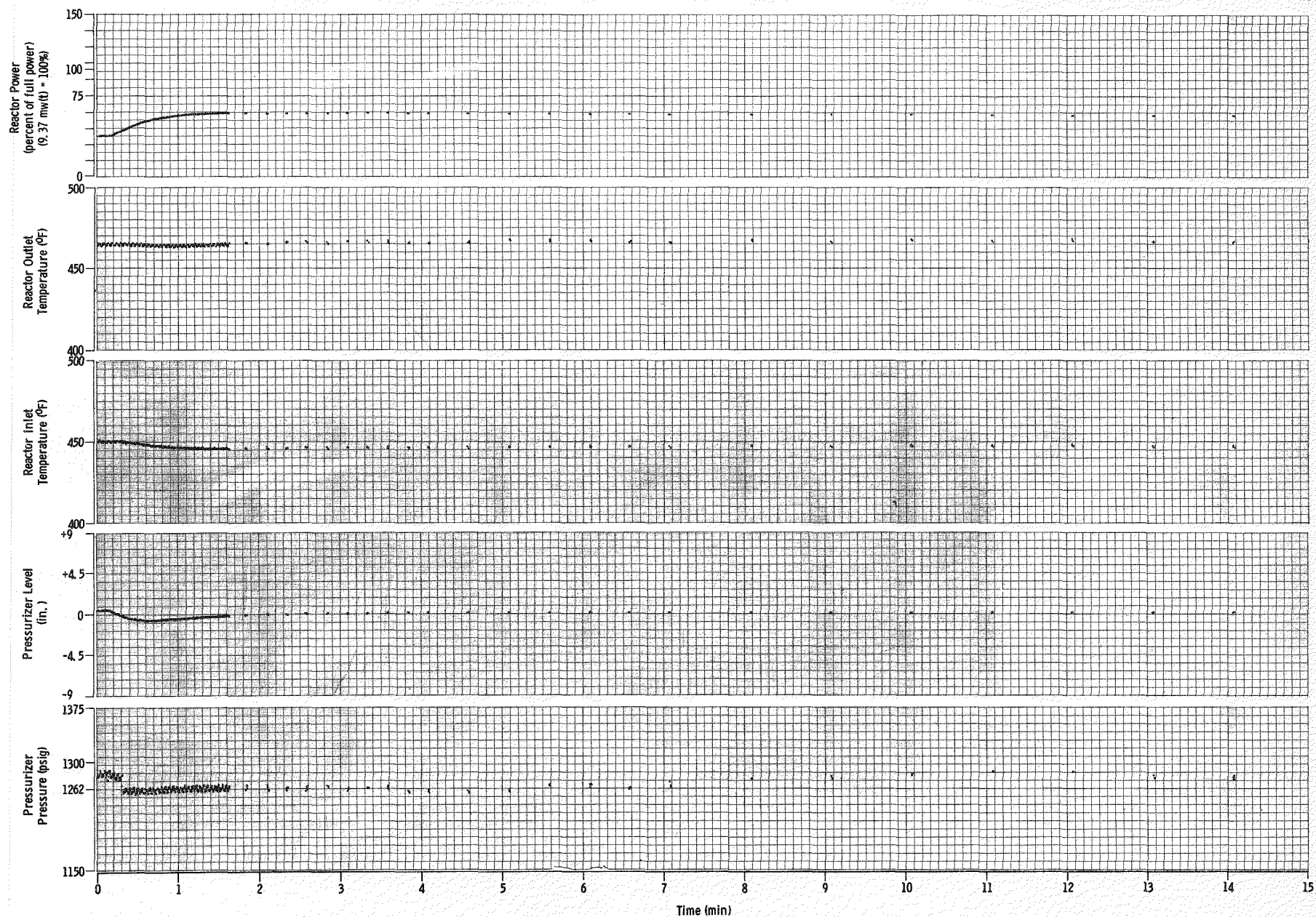


RUN 11--NET LOAD TRANSIENT (1000 - 0 KW) (CONTINUED)

A-14

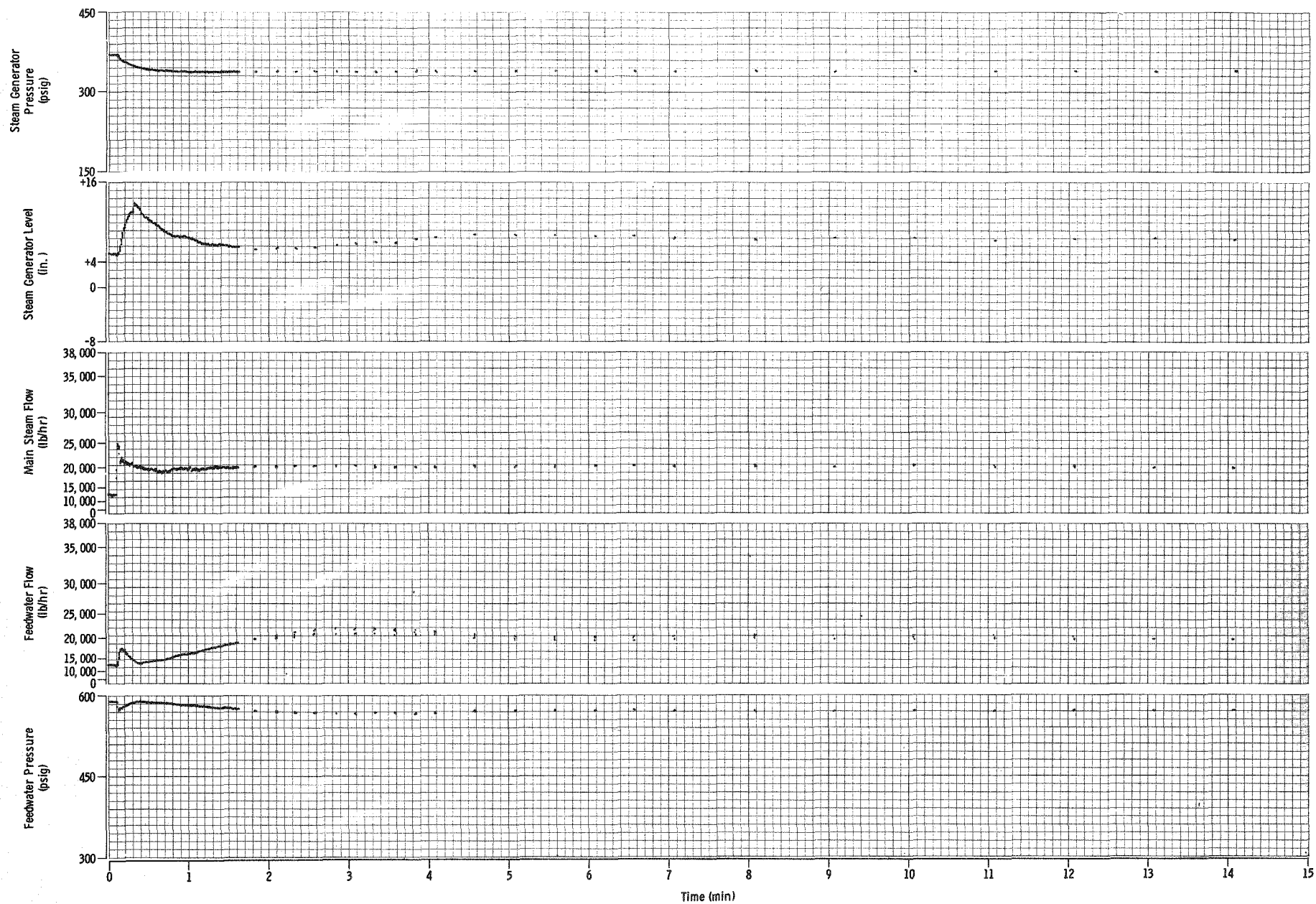


RUN 11--NET LOAD TRANSIENT (1000 - 0 KW) (CONTINUED)

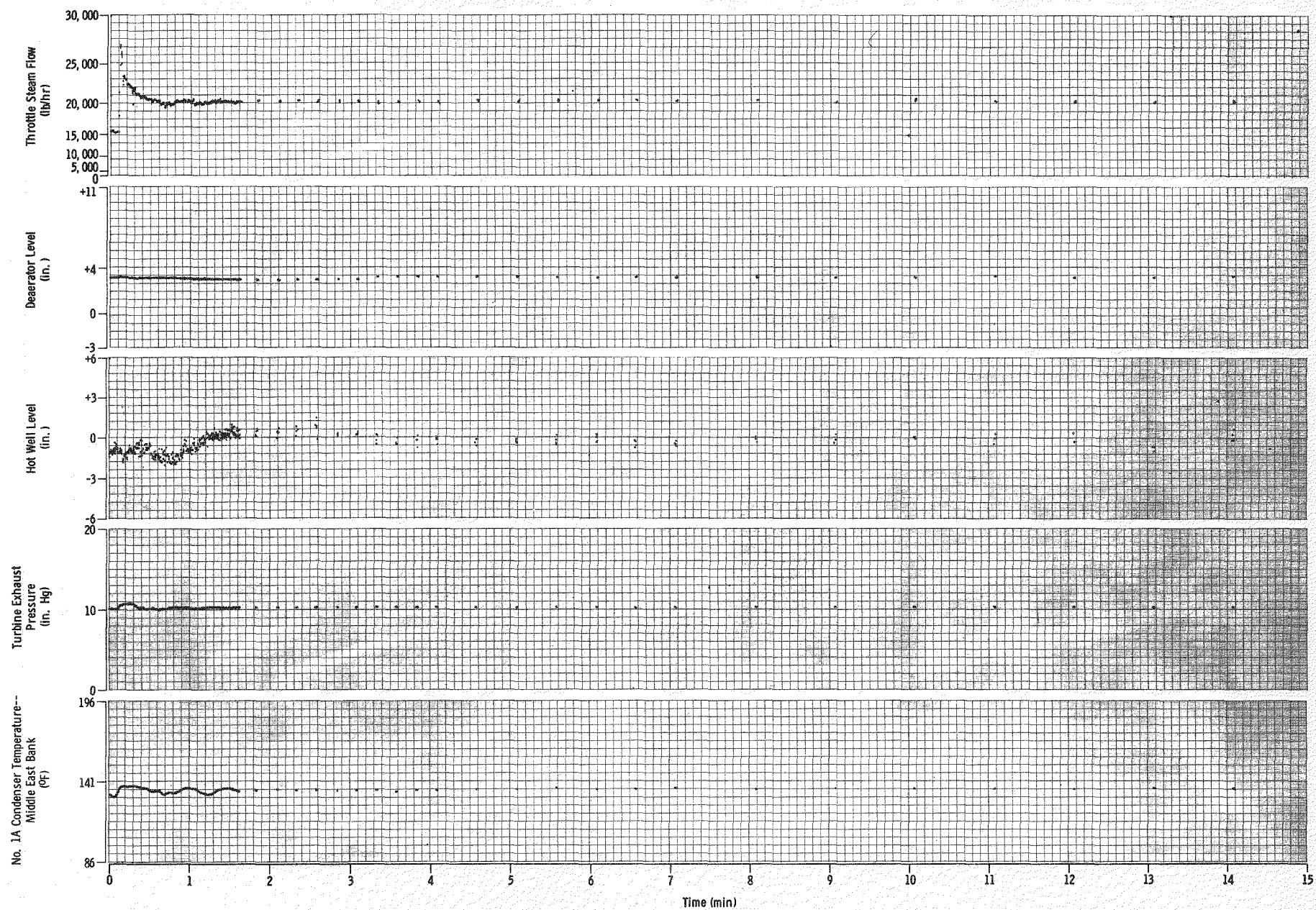


RUN 111--NET LOAD TRANSIENT (200 - 600 KW)

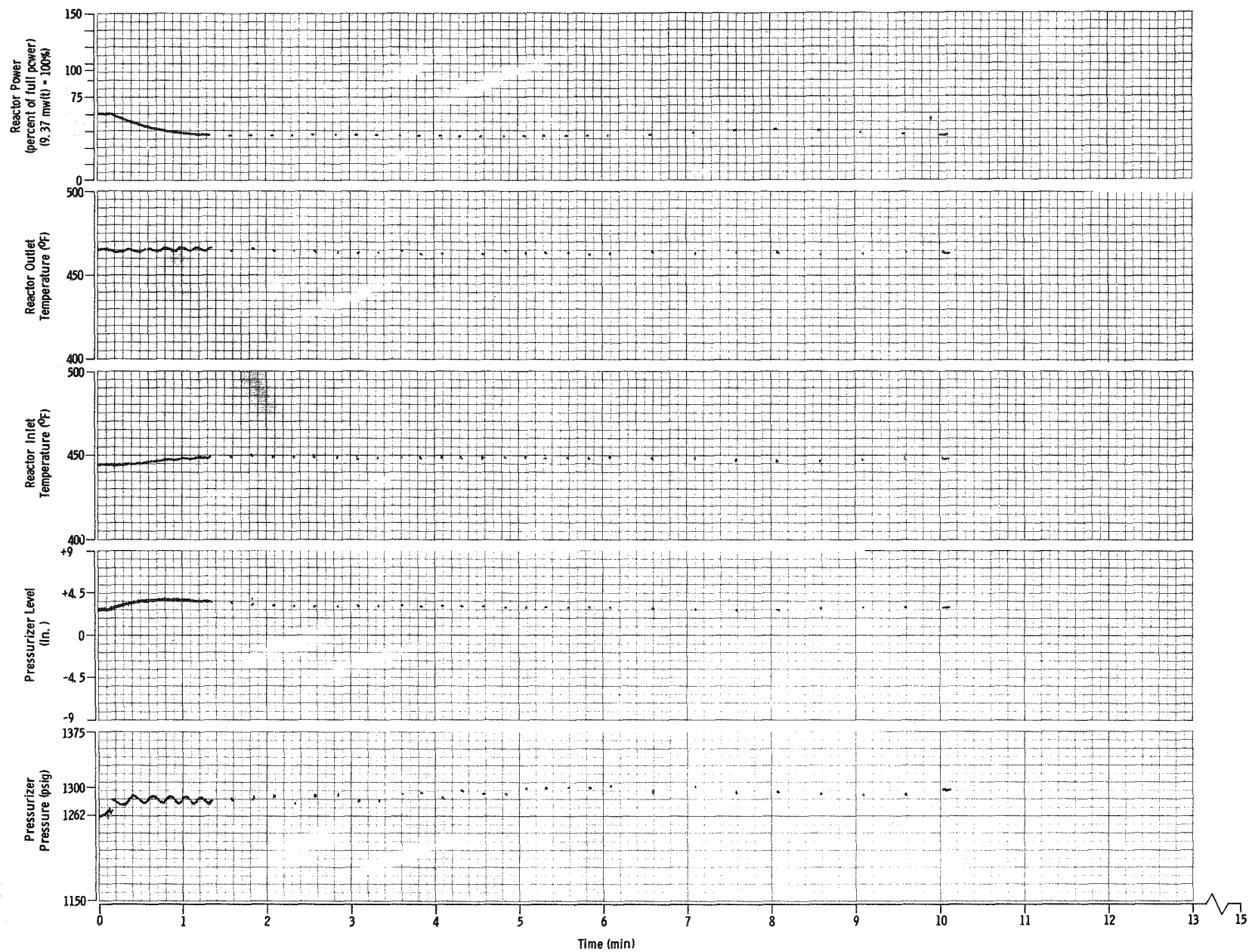
91-A



RUN 111--NET LOAD TRANSIENT (200- 600 KW) (CONTINUED)

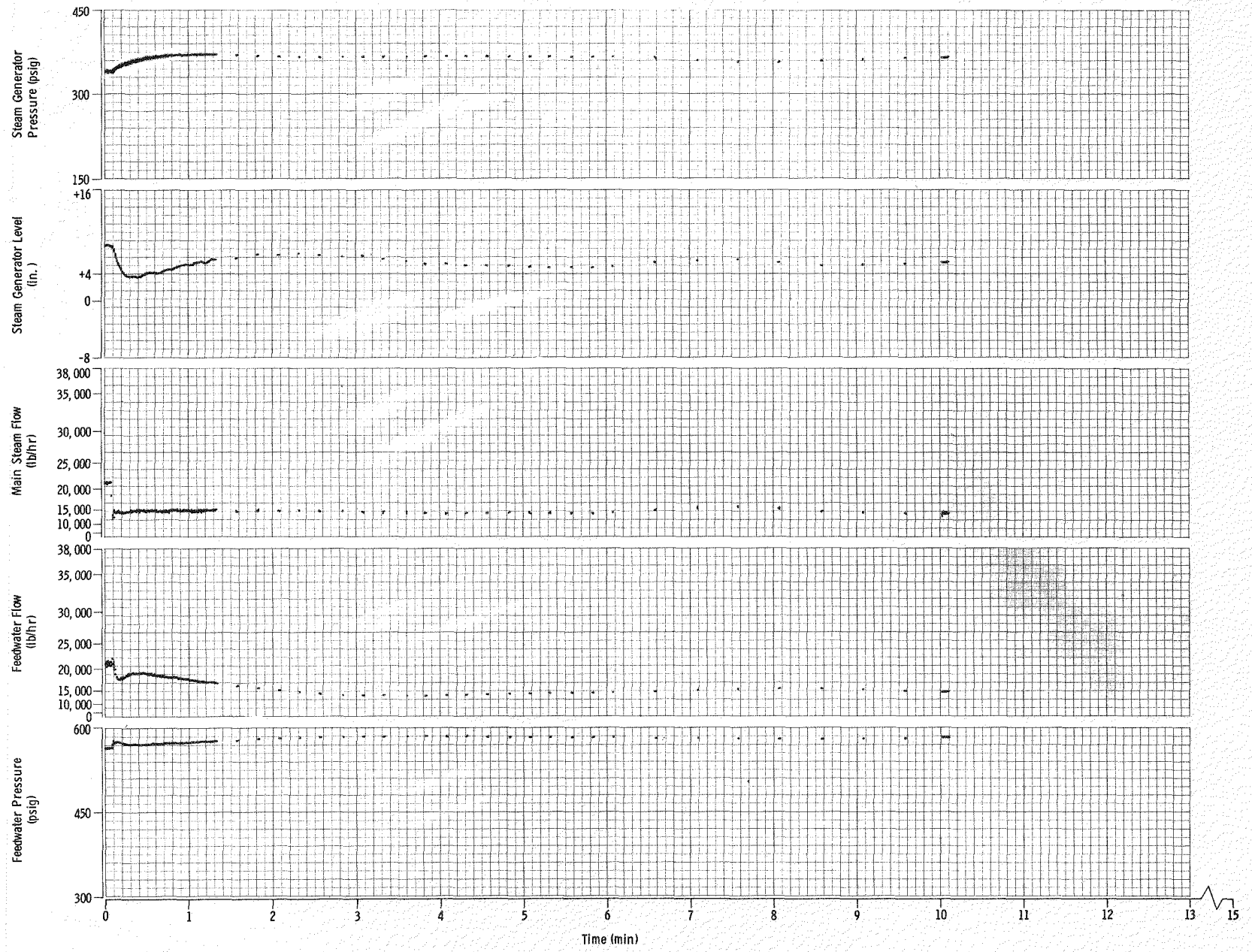


A-18



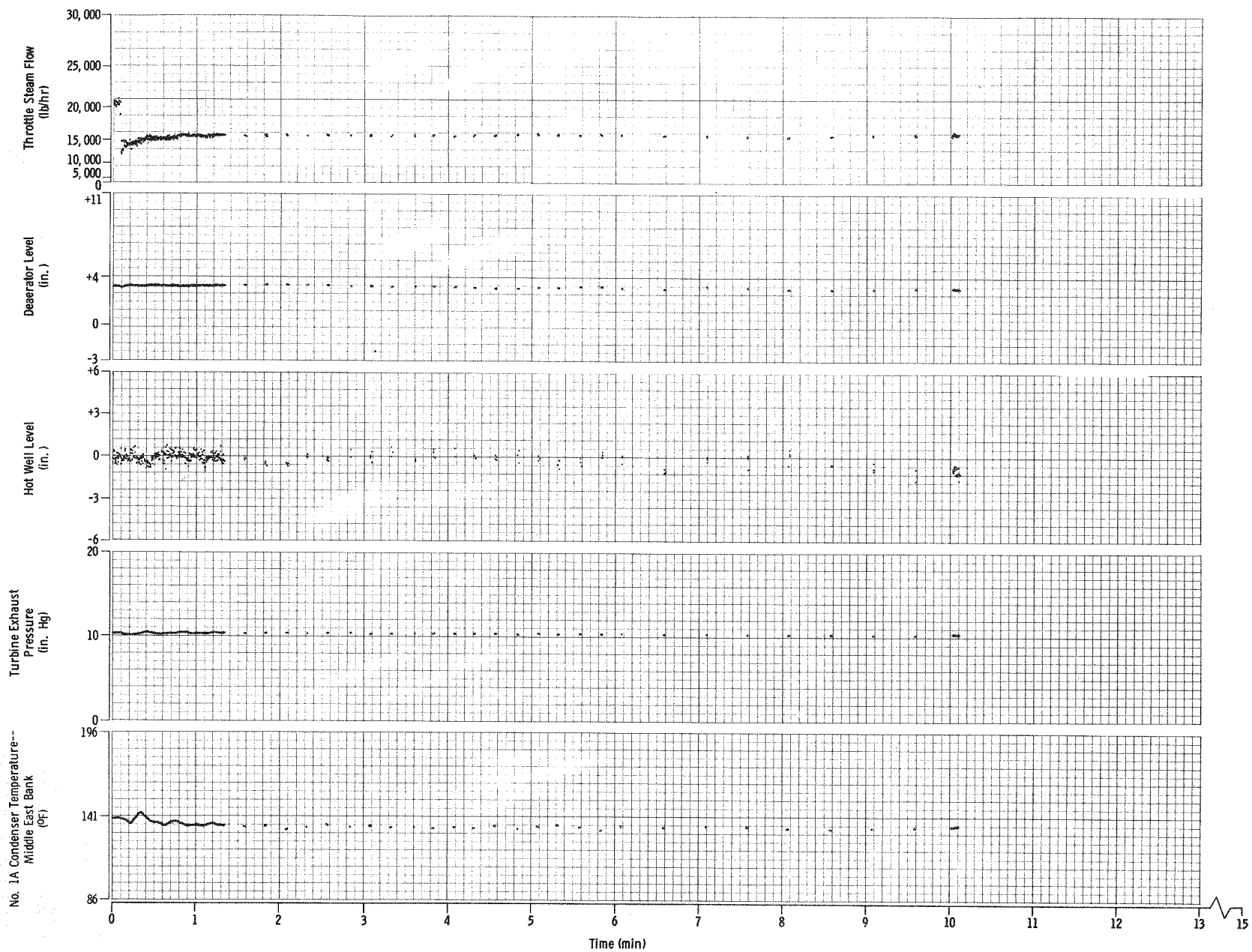
RUN IV--NET LOAD TRANSIENT (600 - 200 KW)

A-19

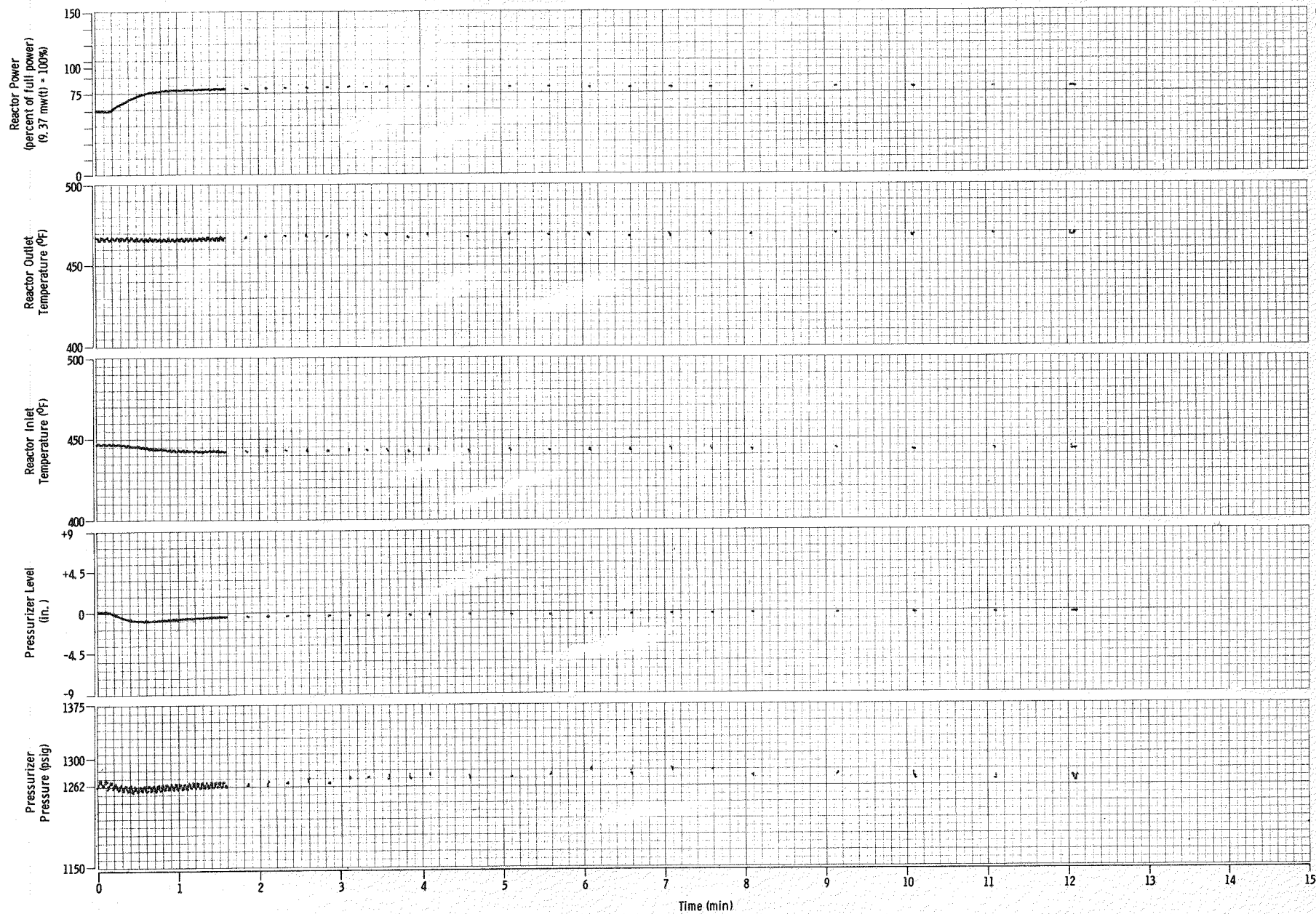


RUN IV--NET LOAD TRANSIENT (600 - 200 KW) (CONTINUED)

A-20

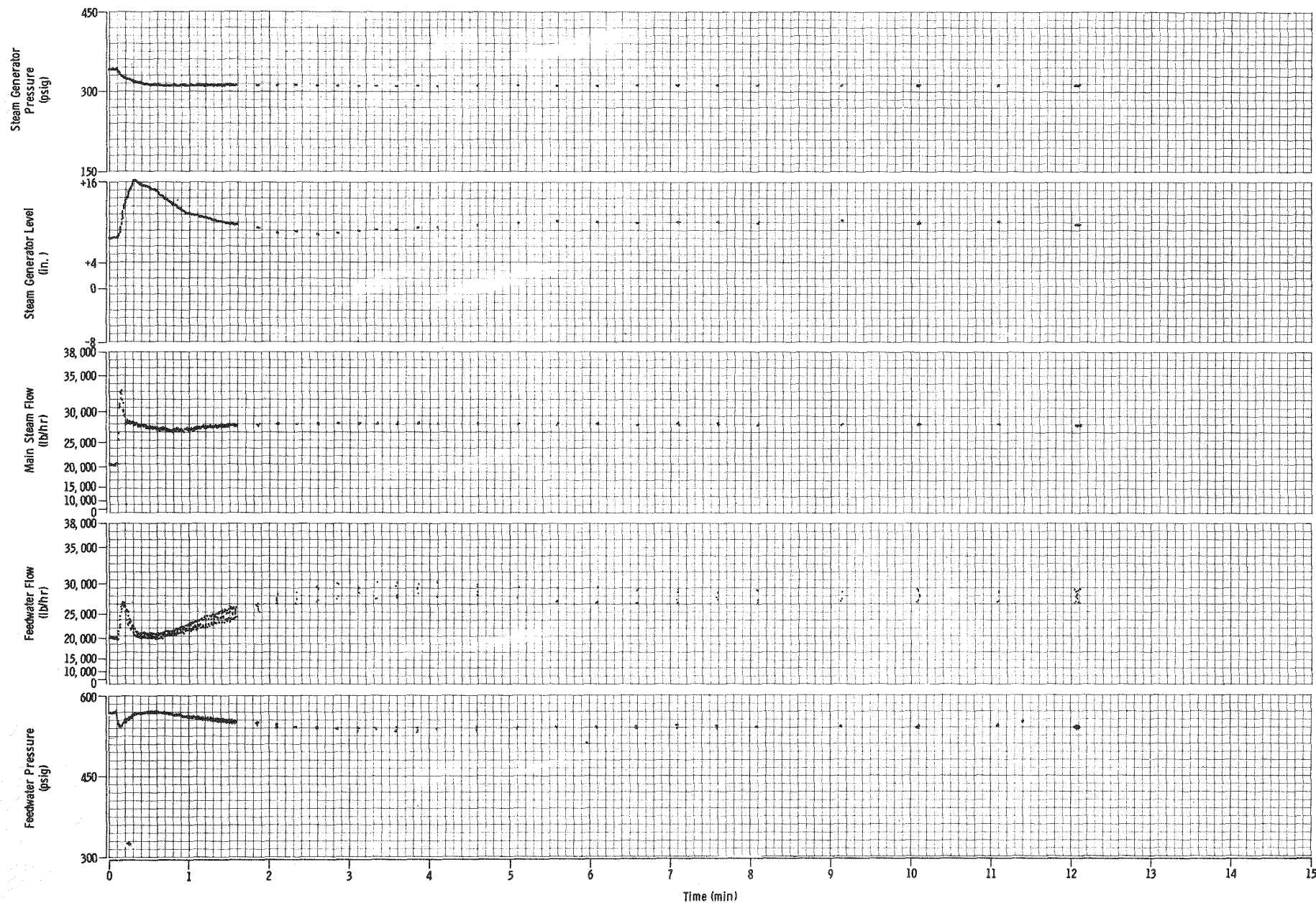


RUN IV--NET LOAD TRANSIENT (600- 200 KW) (CONTINUED)



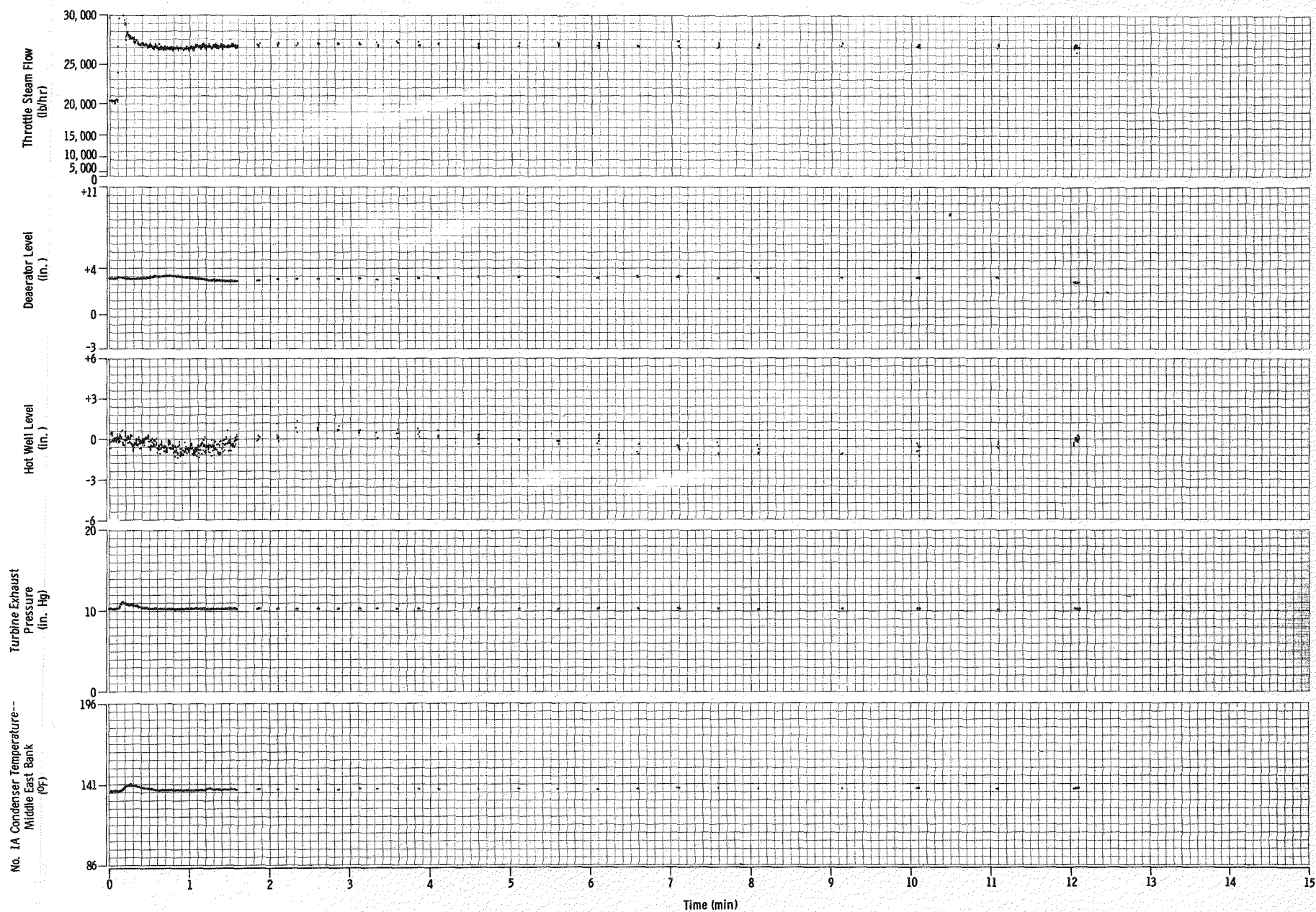
RUN V--NET LOAD TRANSIENT (600 - 1000 KW)

A-22



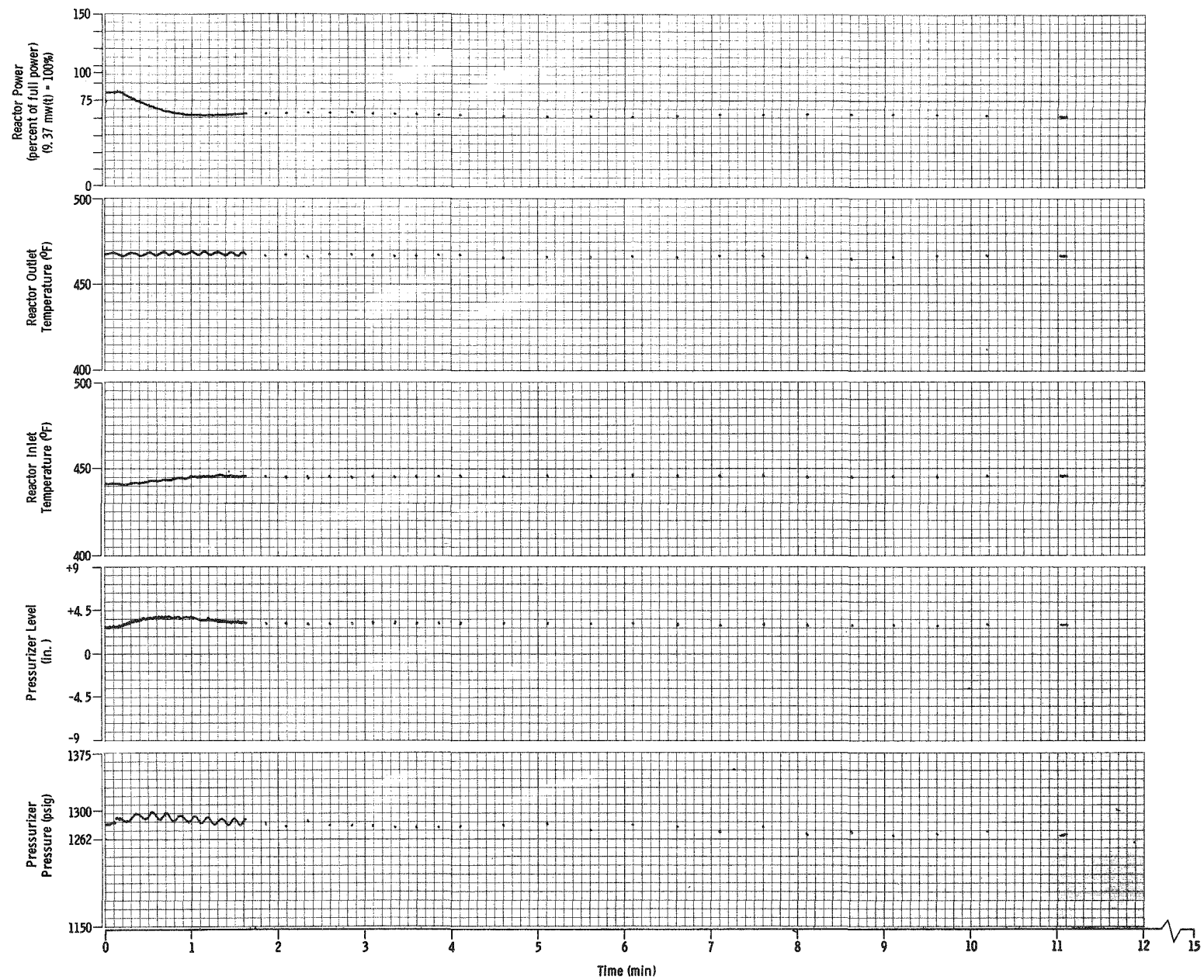
RUN V--NET LOAD TRANSIENT (600 - 1000 KW) (CONTINUED)

A-23



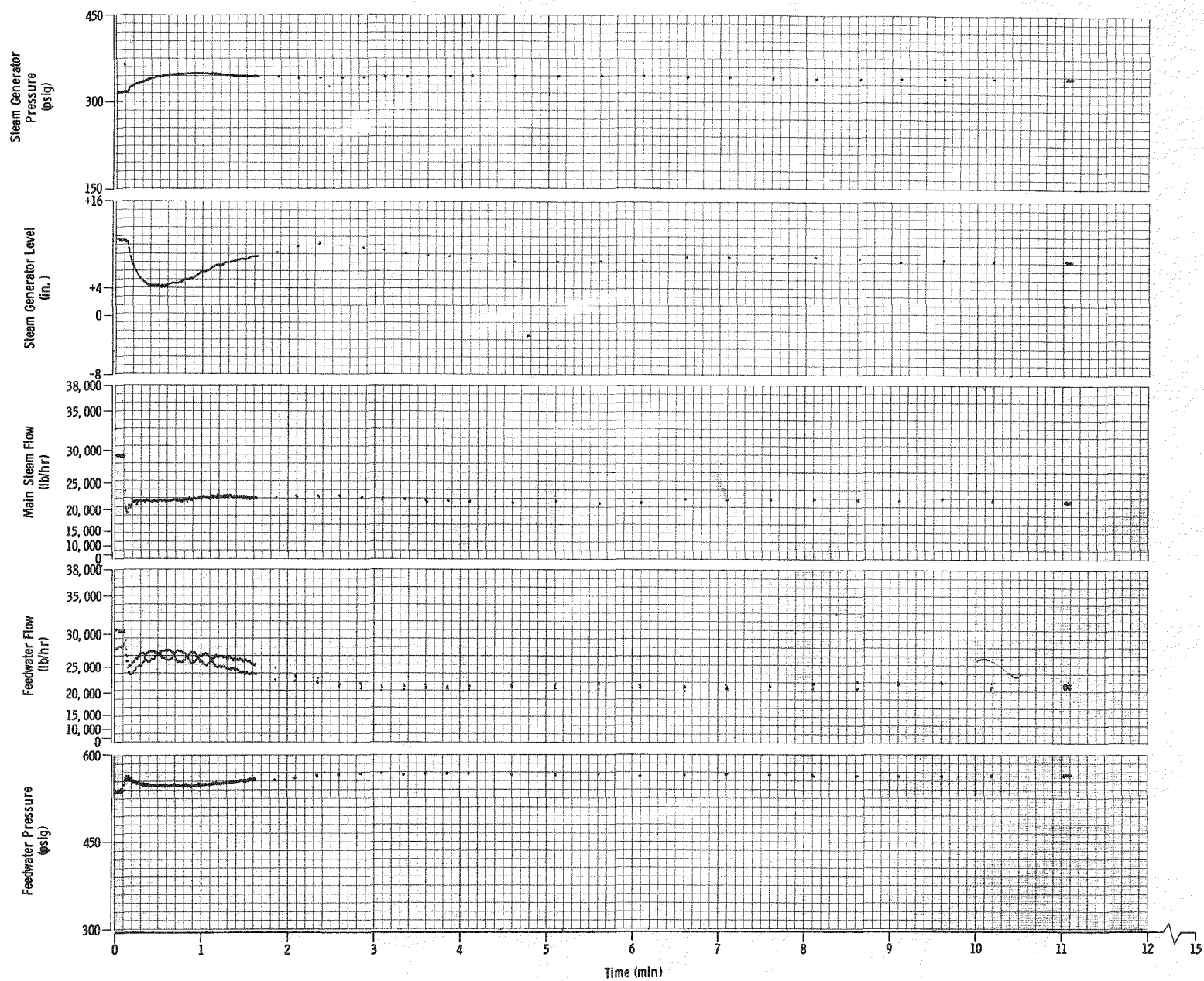
RUN V--NET LOAD TRANSIENT (600 - 1000 KW) (CONTINUED)

A-24



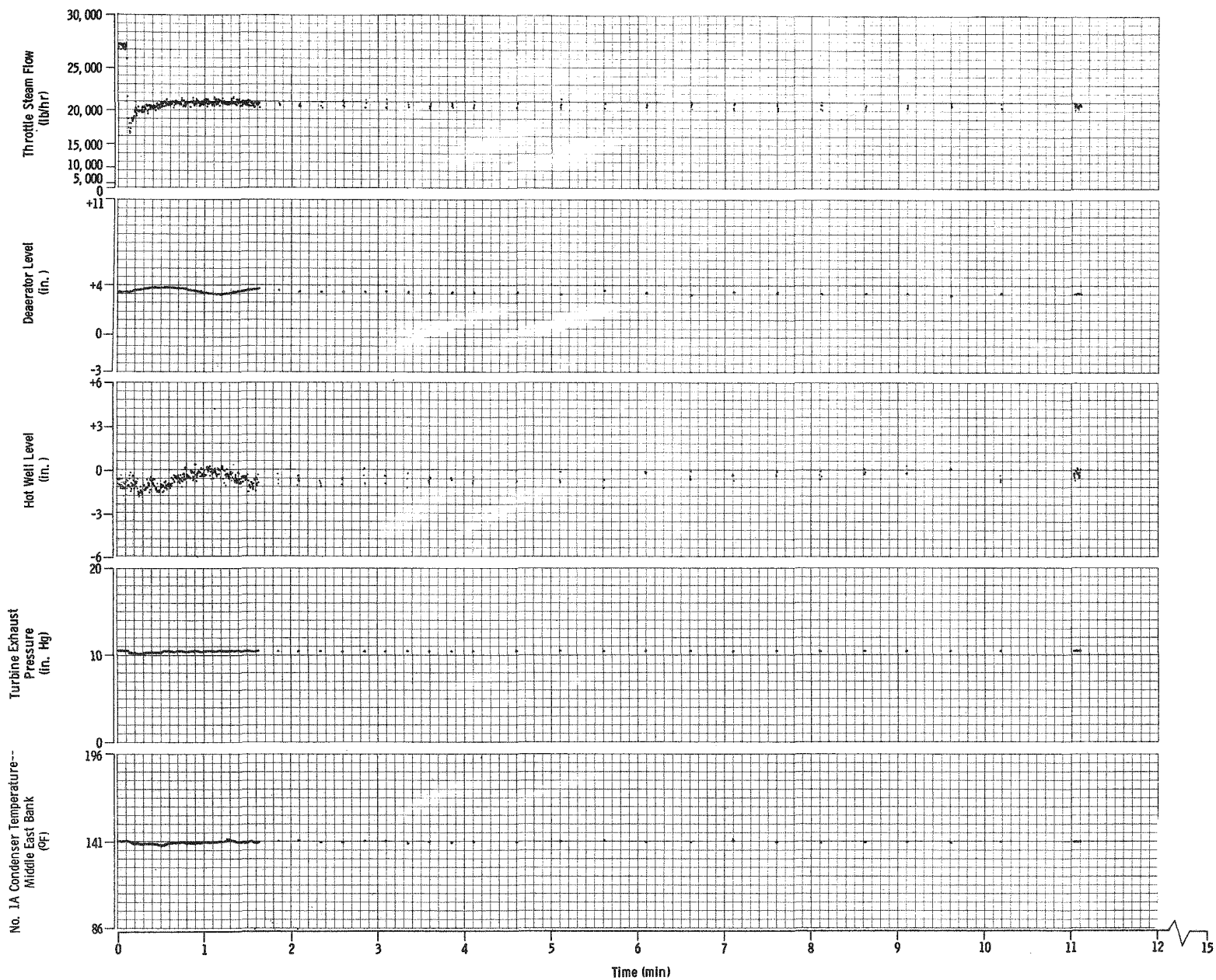
RUN VI--NET LOAD TRANSIENT (1000 - 600 KW)

A-25



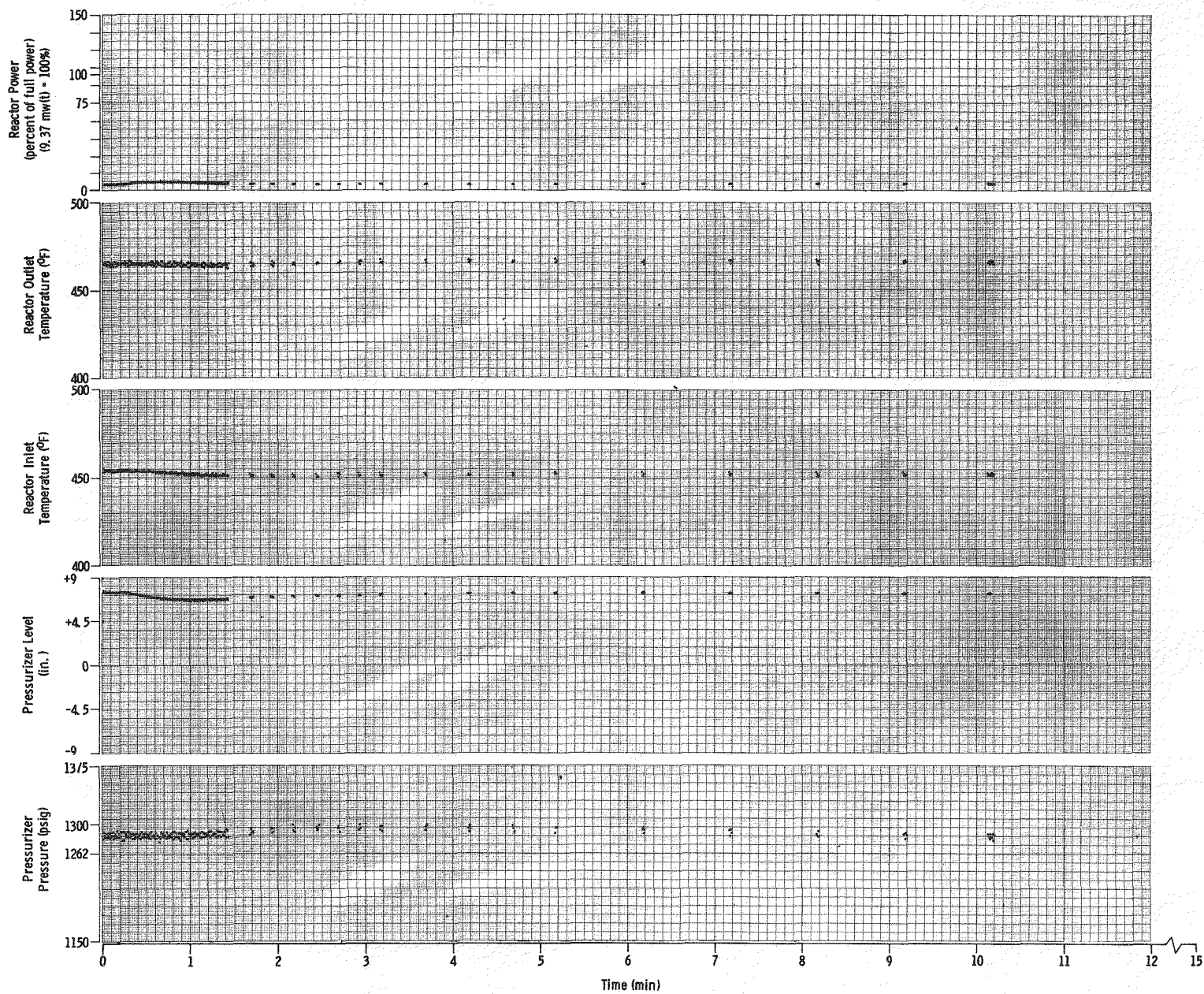
RUN VI--NET LOAD TRANSIENT (1000 - 600 KW) (CONTINUED)

A-26



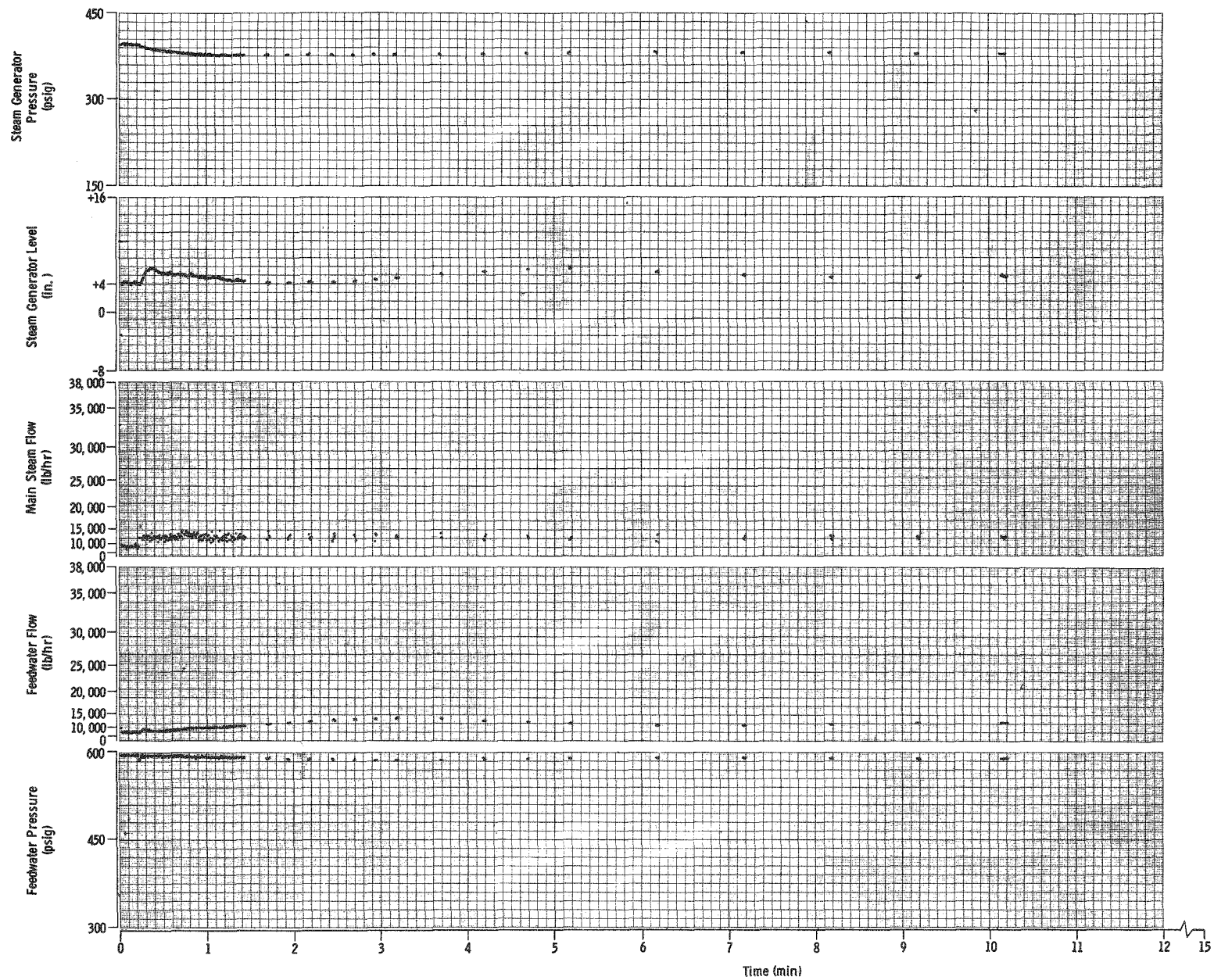
RUN VI--NET LOAD TRANSIENT (1000 - 600 KW) (CONTINUED)

A-27



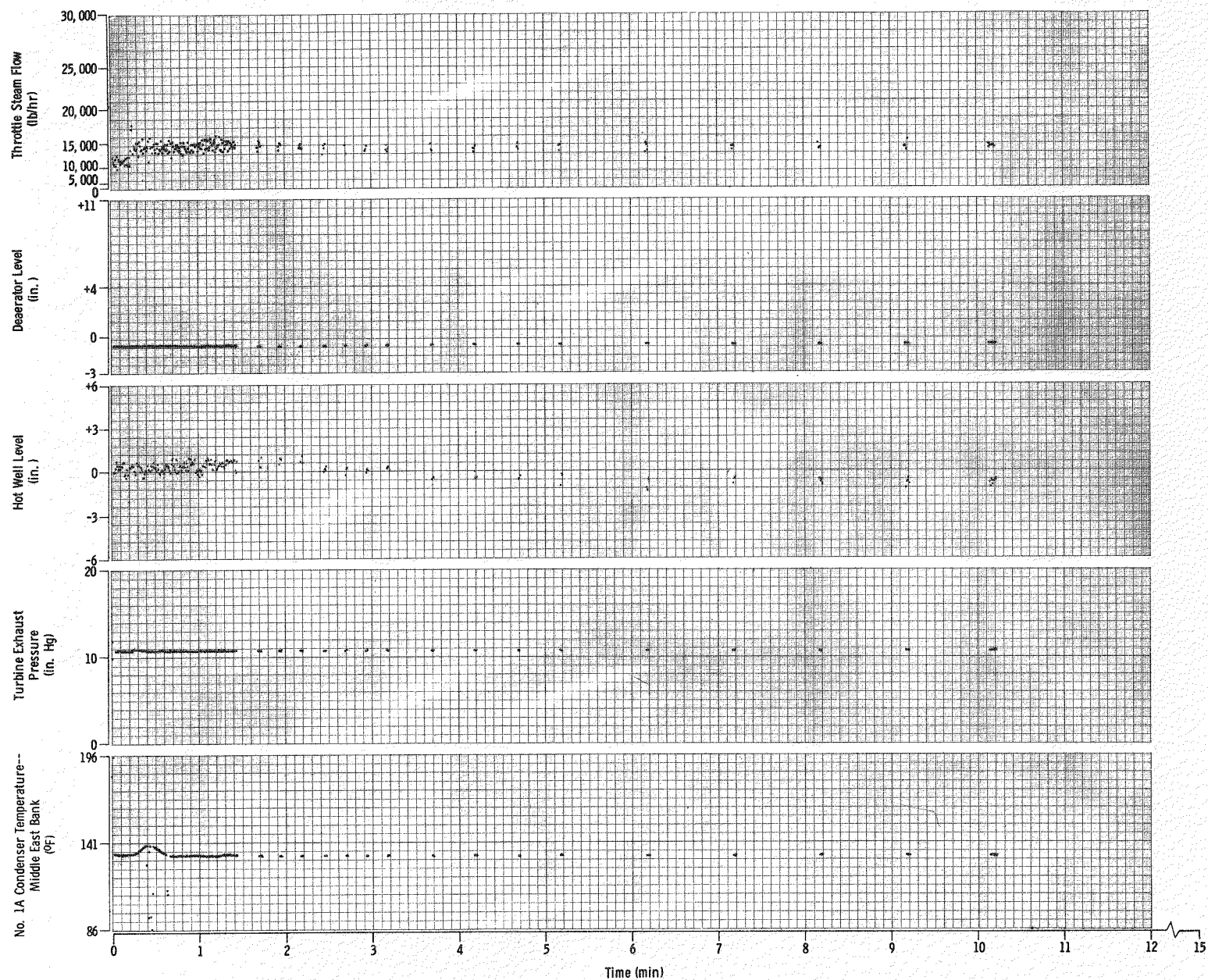
RUN VII--NET LOAD TRANSIENT (0 - 200 KW)

A-28



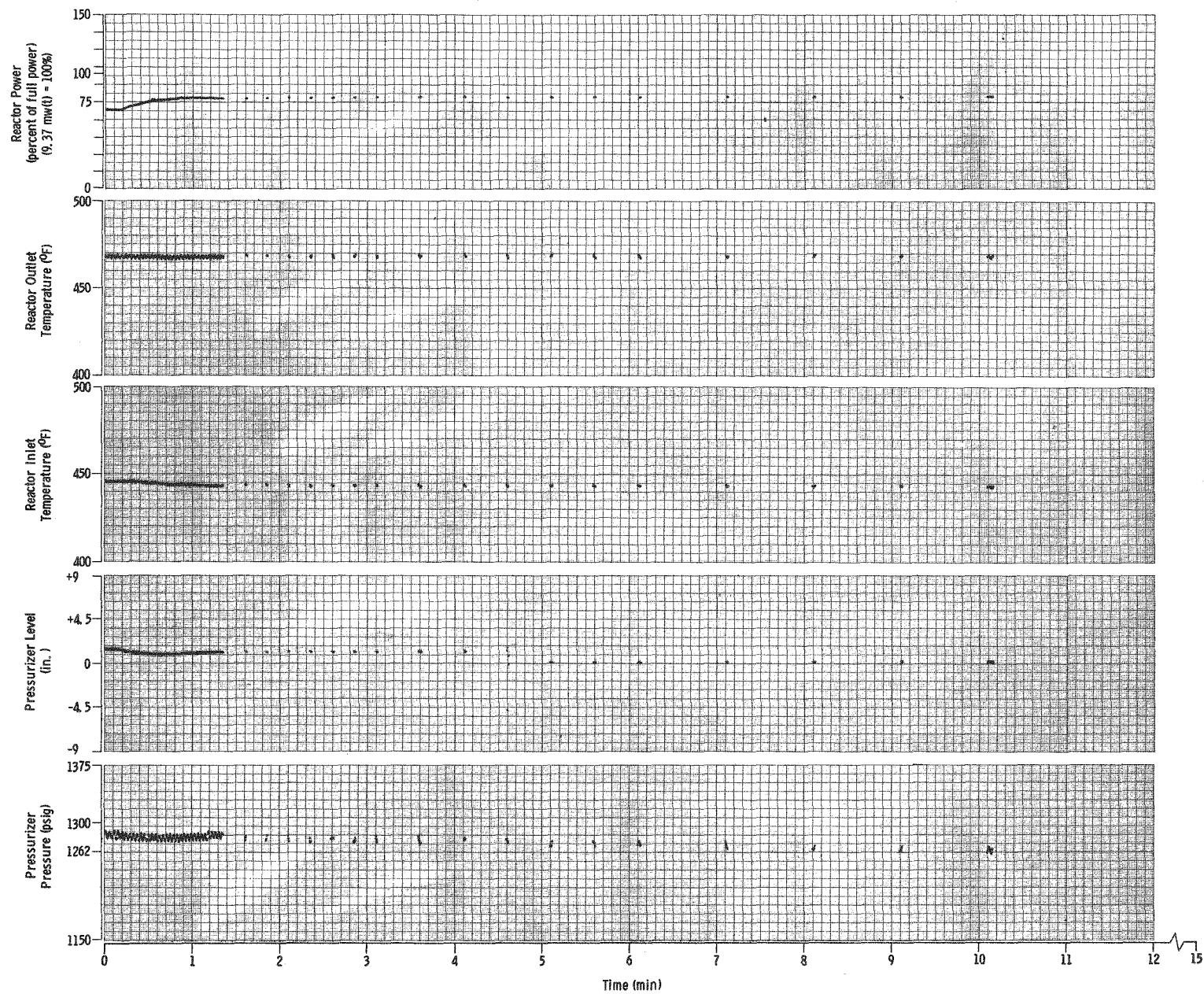
RUN VII--NET LOAD TRANSIENT (0 - 200 KW) (CONTINUED)

A-29



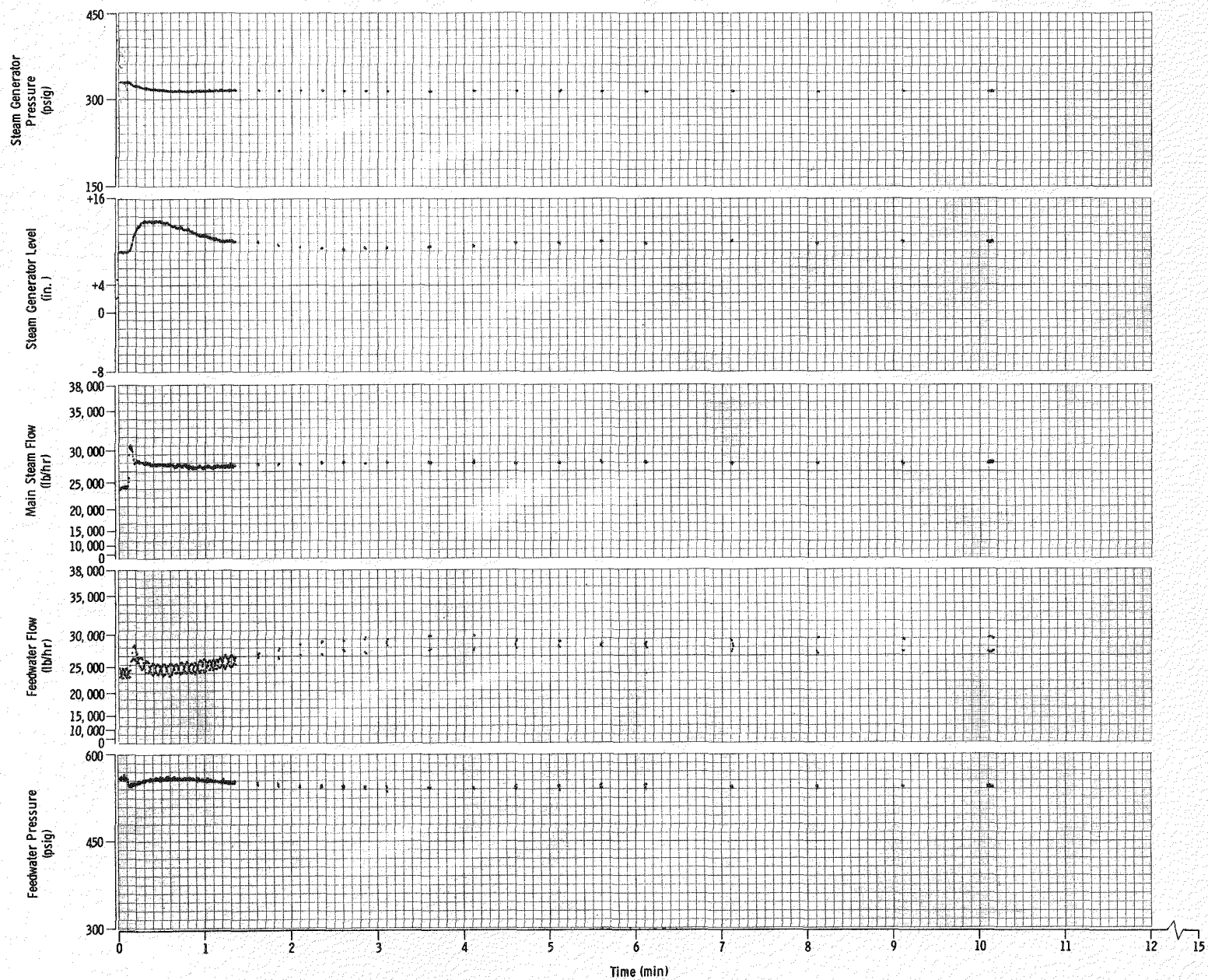
RUN VII--NET LOAD TRANSIENT (0 - 200 KW) (CONTINUED)

05-V



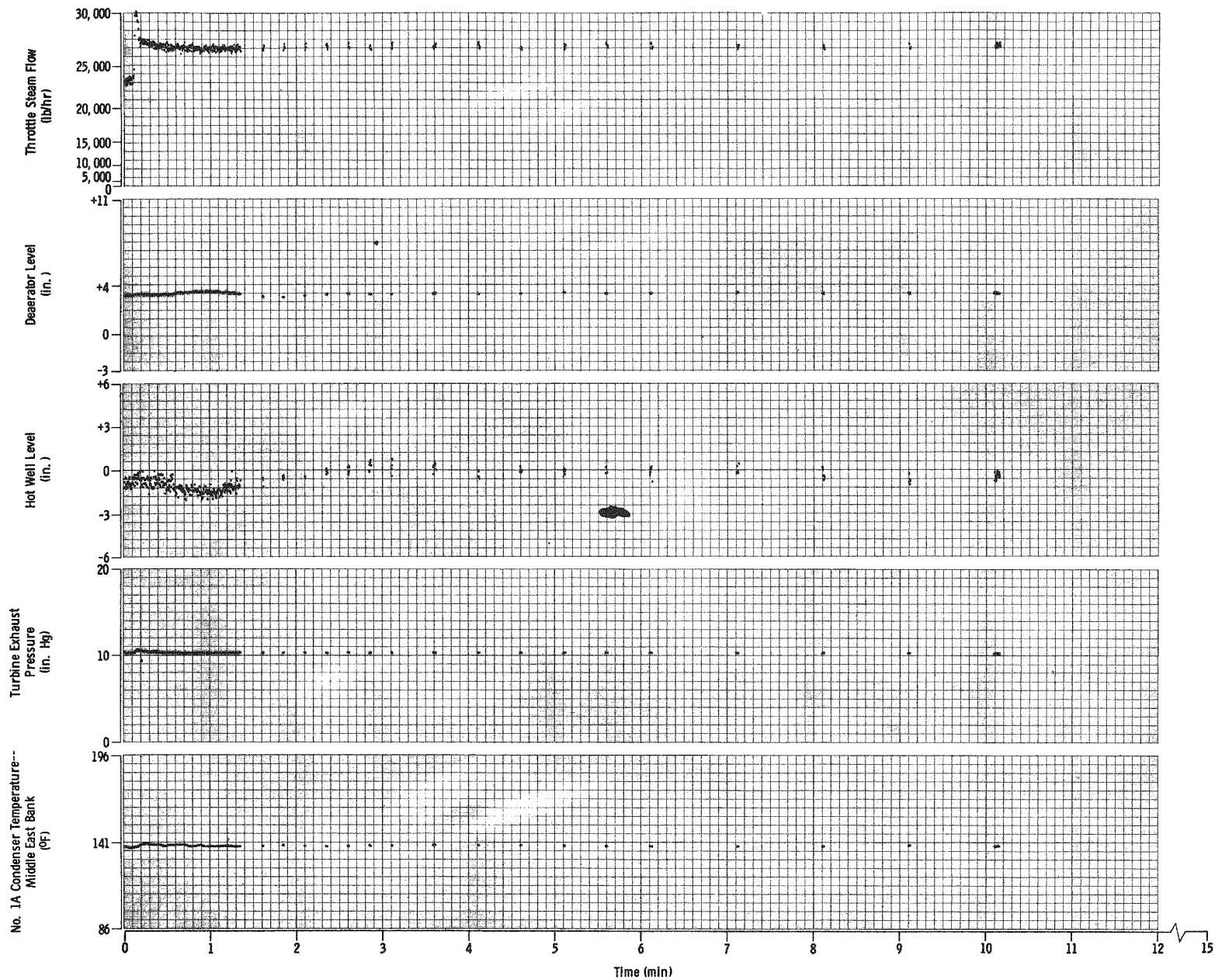
RUN VIII--NET LOAD TRANSIENT (800 - 1000 KW)

A-31



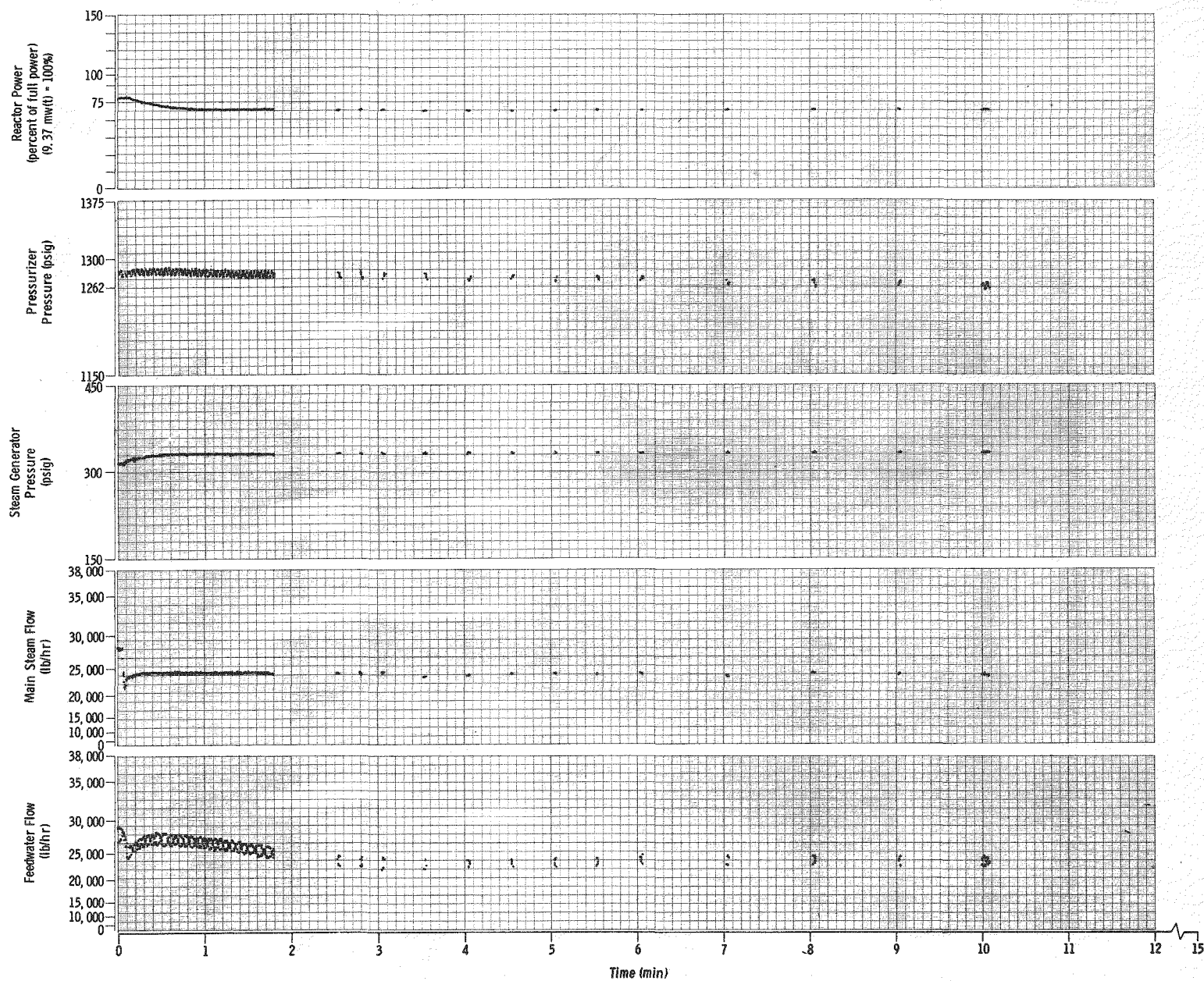
RUN VIII--NET LOAD TRANSIENT (800 - 1000 KW) (CONTINUED)

A-32



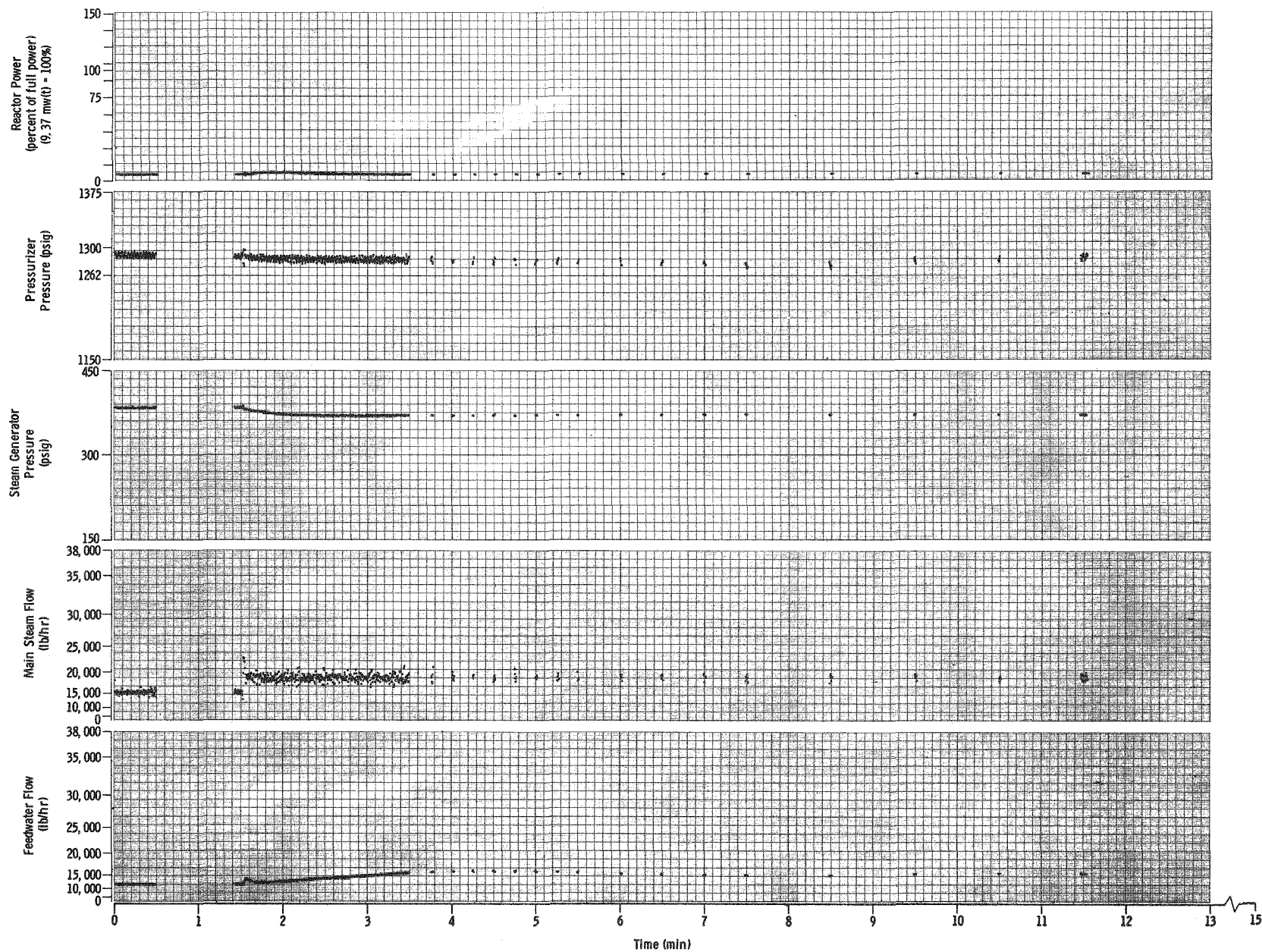
RUN VIII--NET LOAD TRANSIENT (800 - 1000 KW) (CONTINUED)

A-33

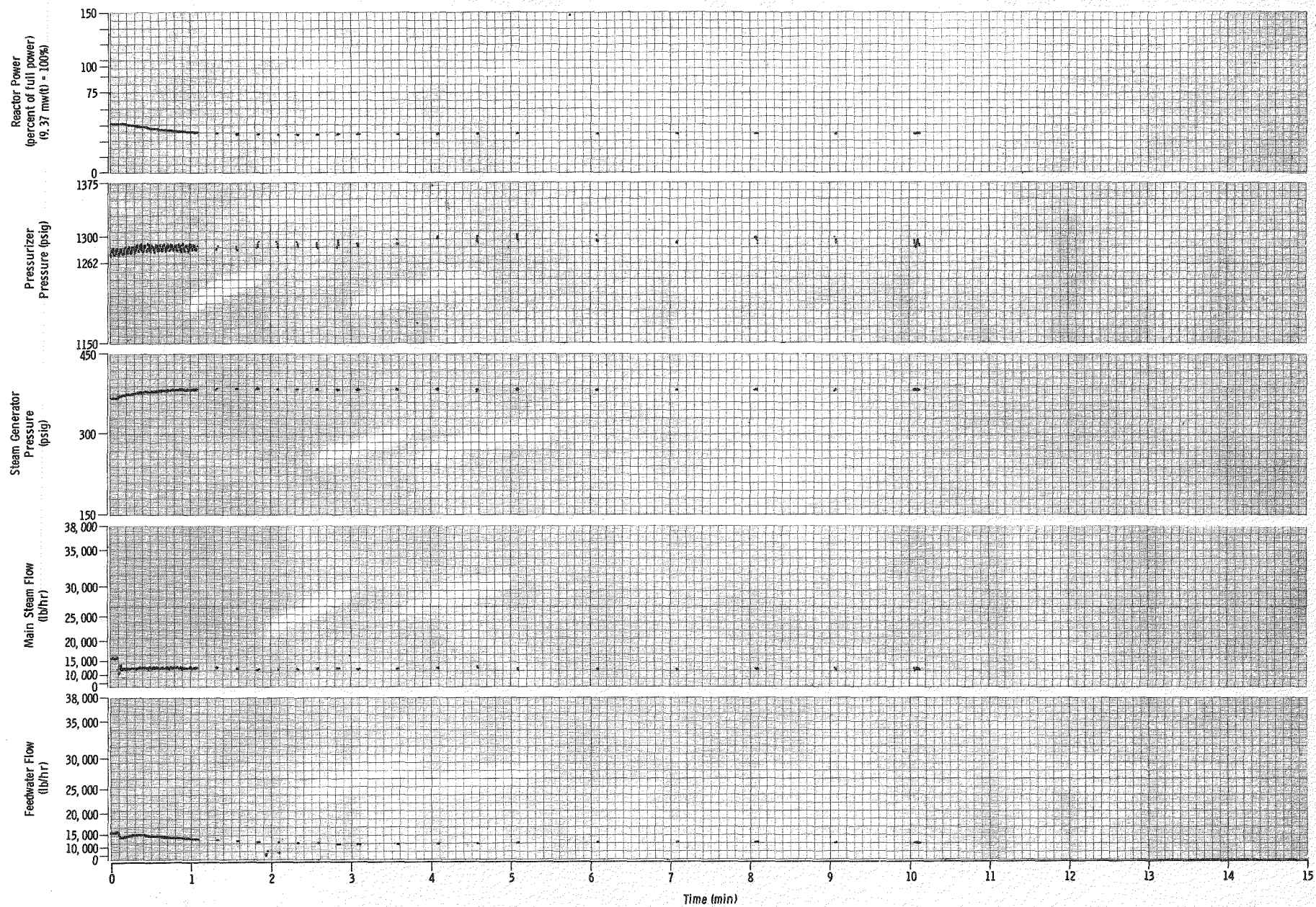


RUN IX--NET LOAD TRANSIENT (1000 - 800 KW)

A-34

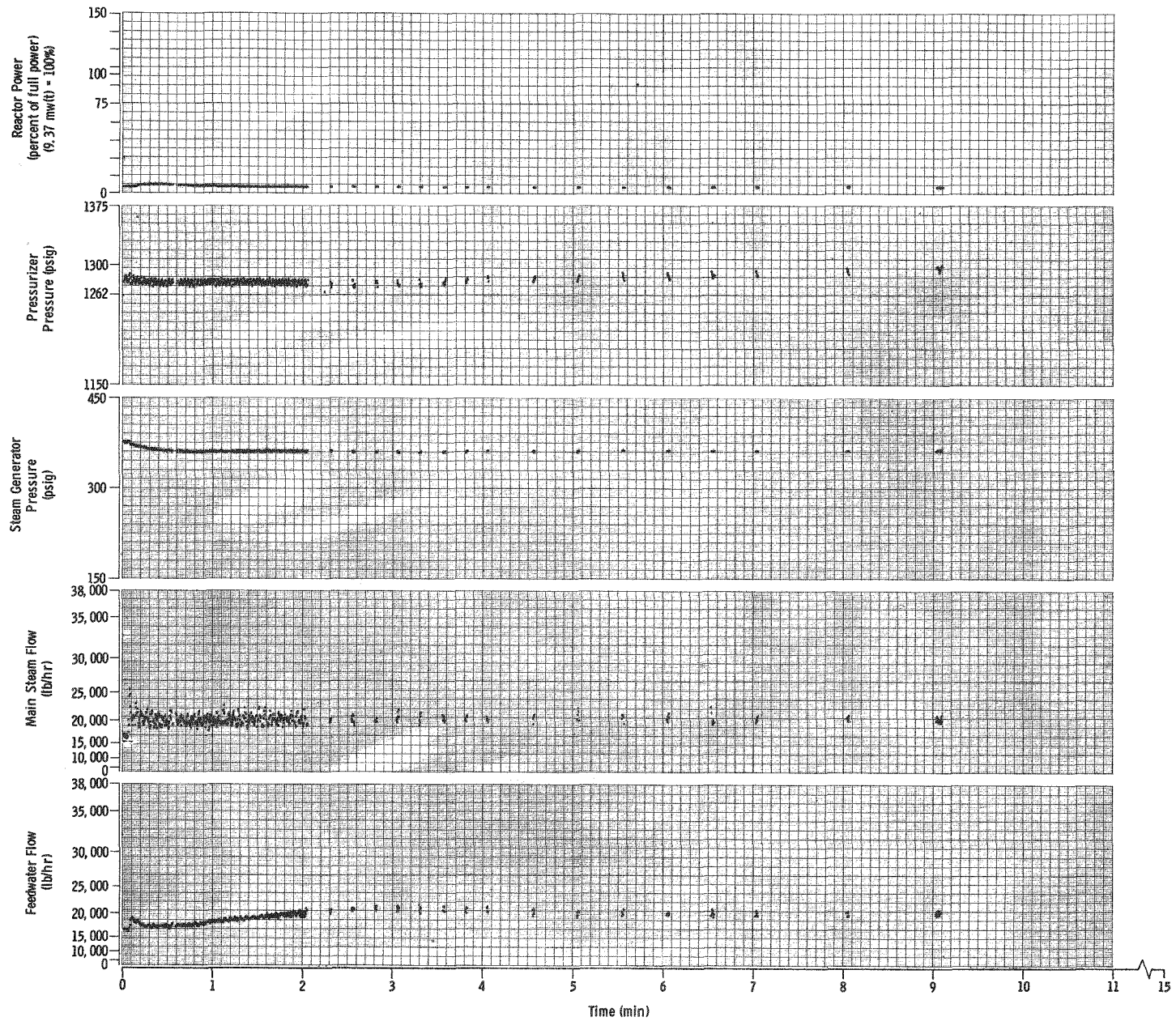


RUN X--NET LOAD TRANSIENT (200-400 KW)



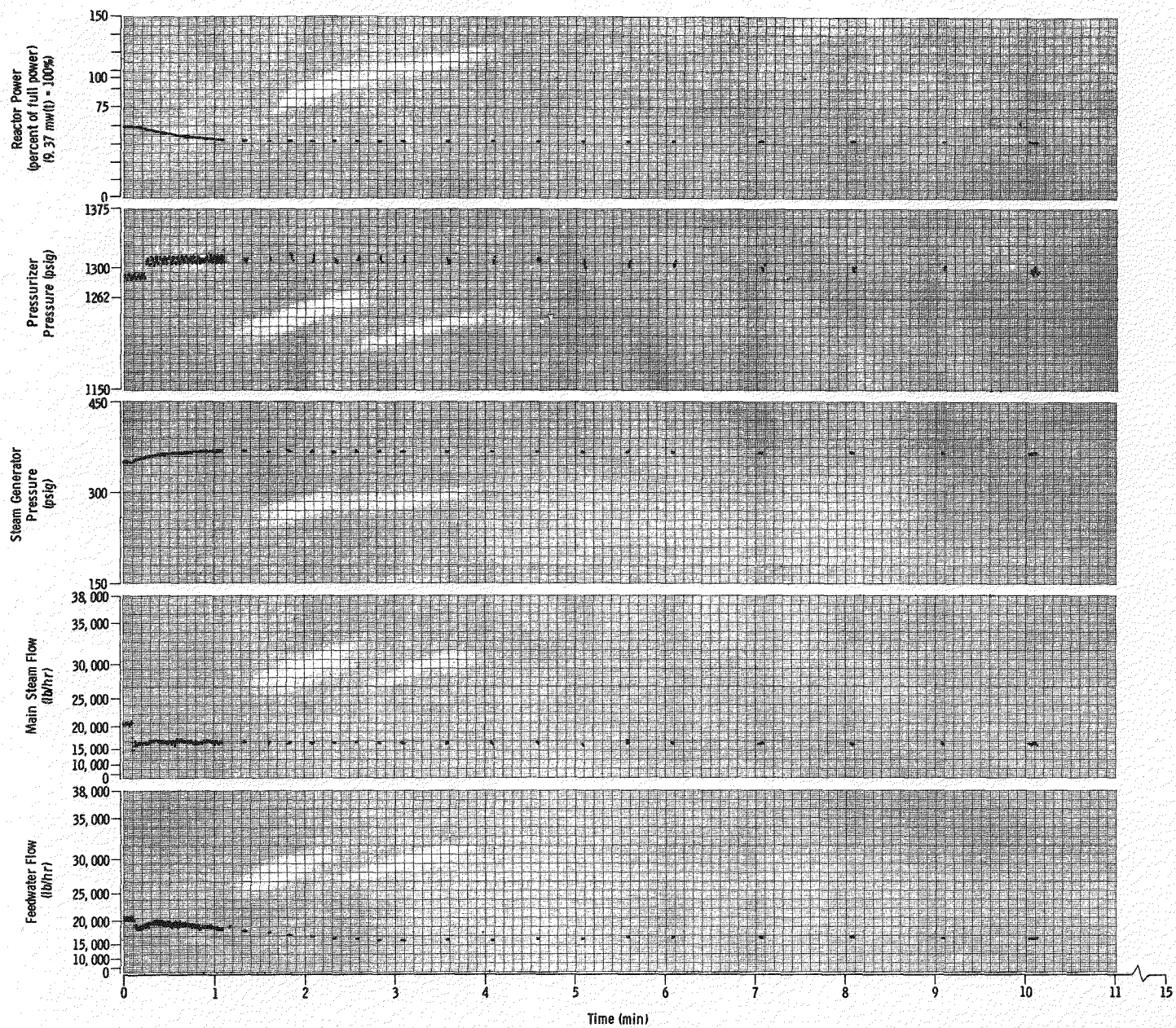
RUN XI--NET LOAD TRANSIENT (400 - 200 KW)

A-36



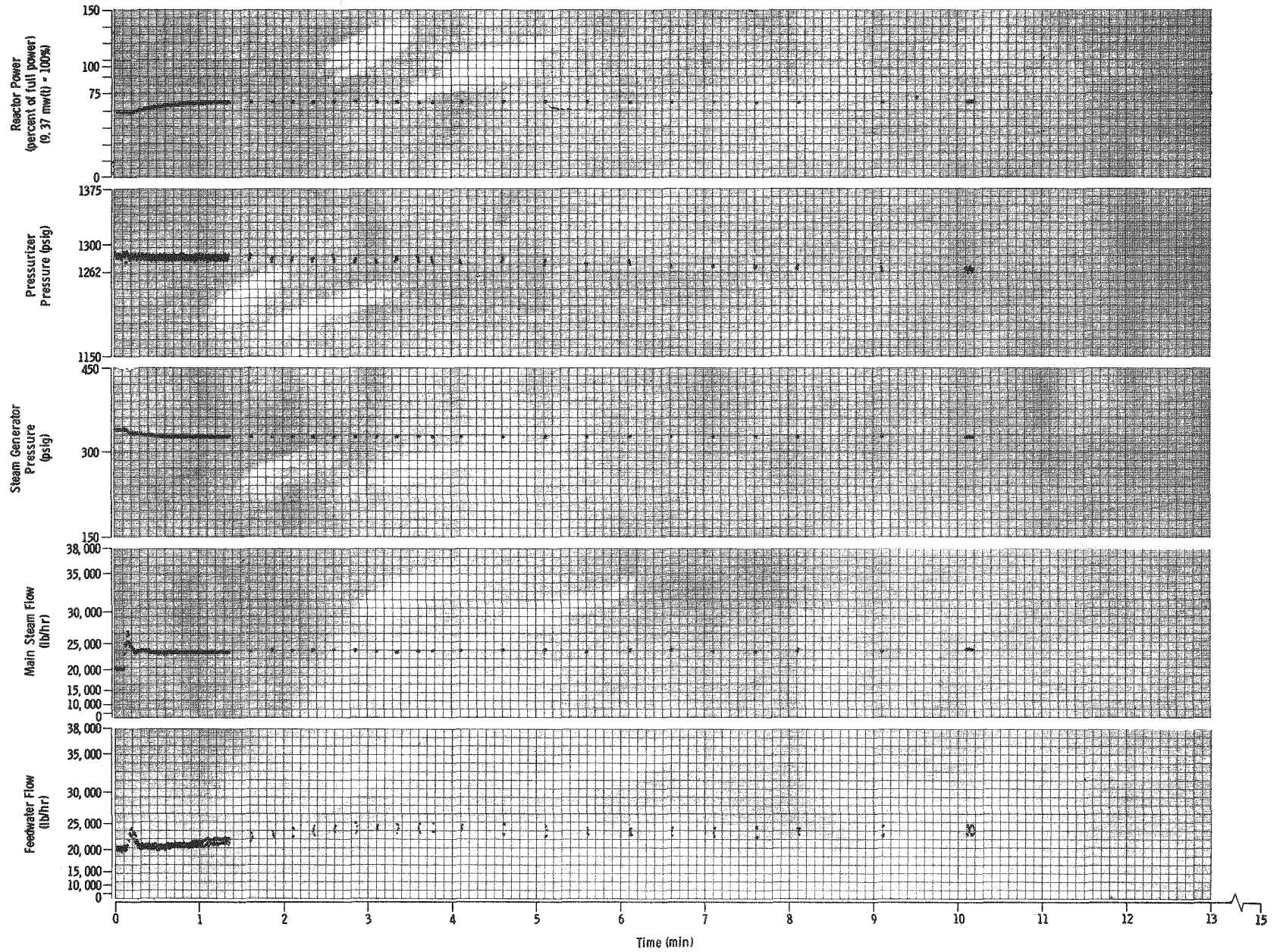
RUN XII--NET LOAD TRANSIENT (400 - 600 KW)

A-37



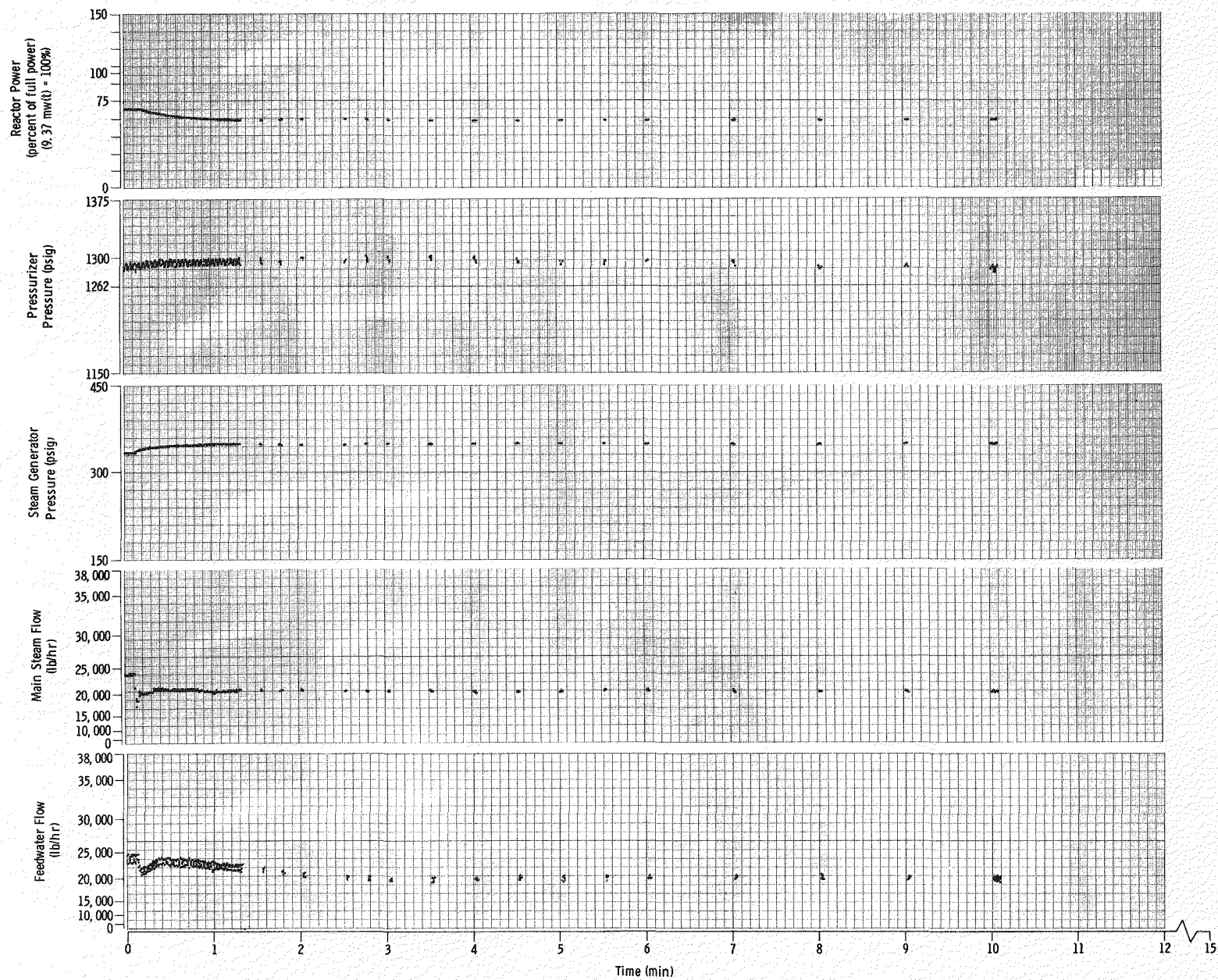
RUN XIII--NET LOAD TRANSIENT (600 - 400 KW)

86-V



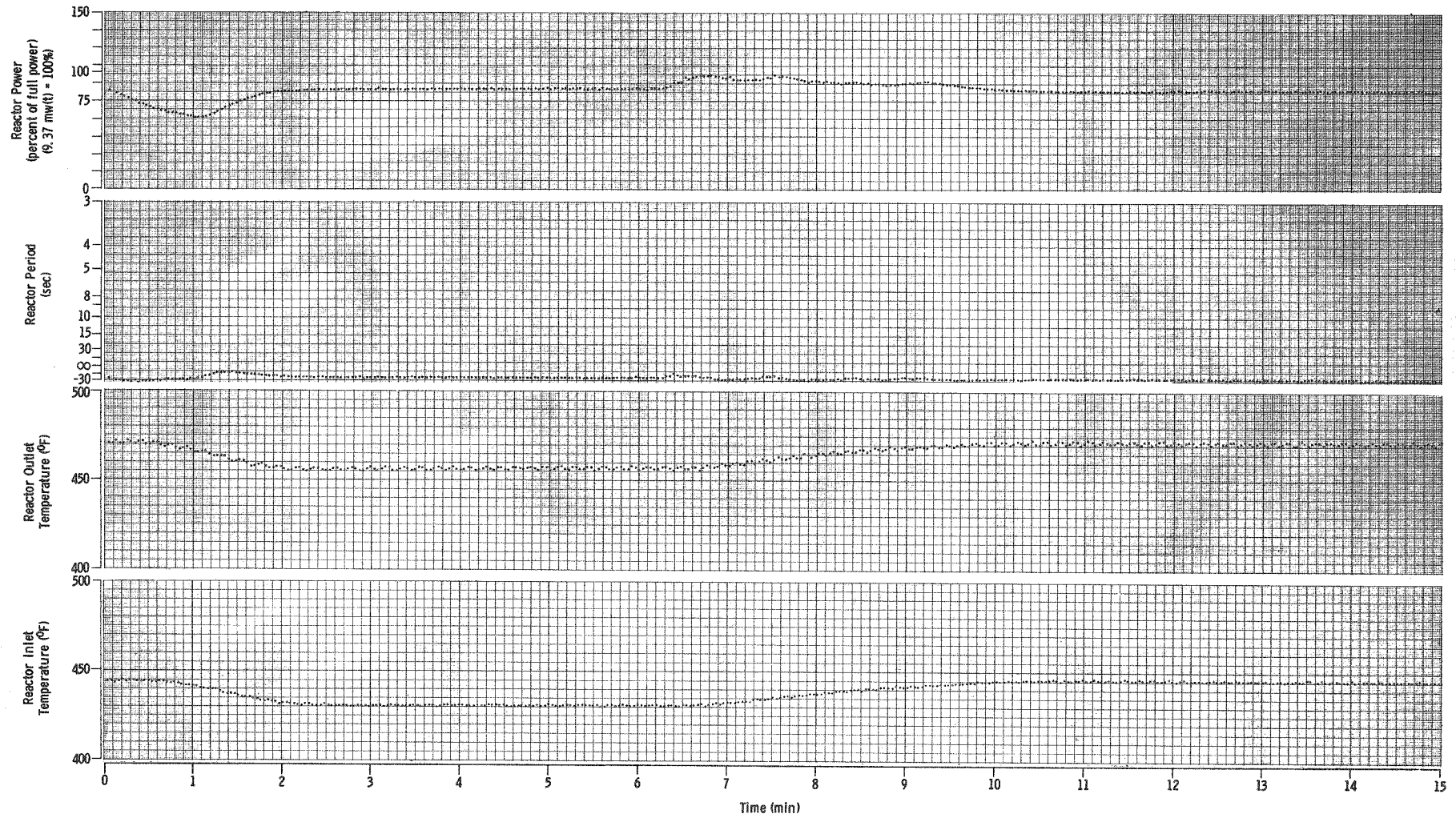
RUN XIV--NET LOAD TRANSIENT (600 - 800 KW)

A-39



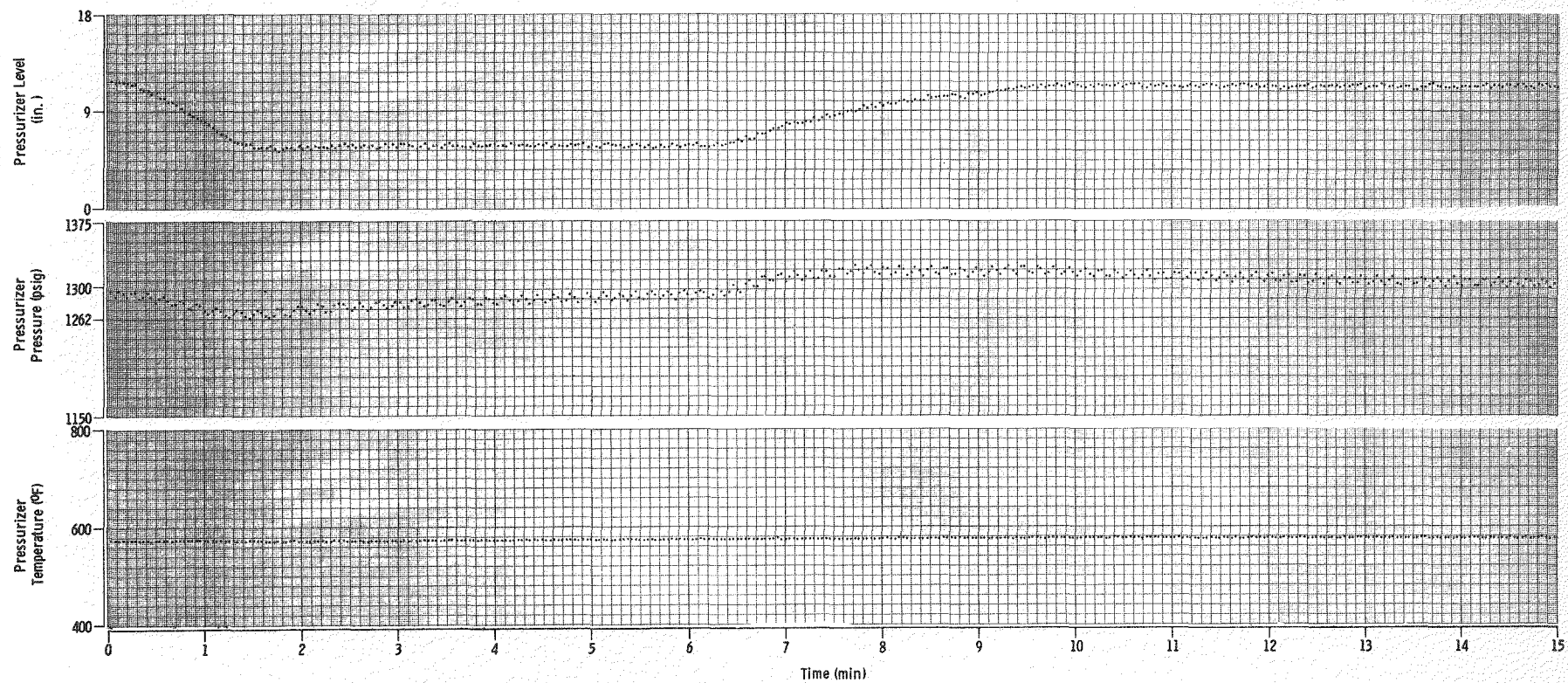
RUN XV--NET LOAD TRANSIENT (800 - 600 KW)

A-40



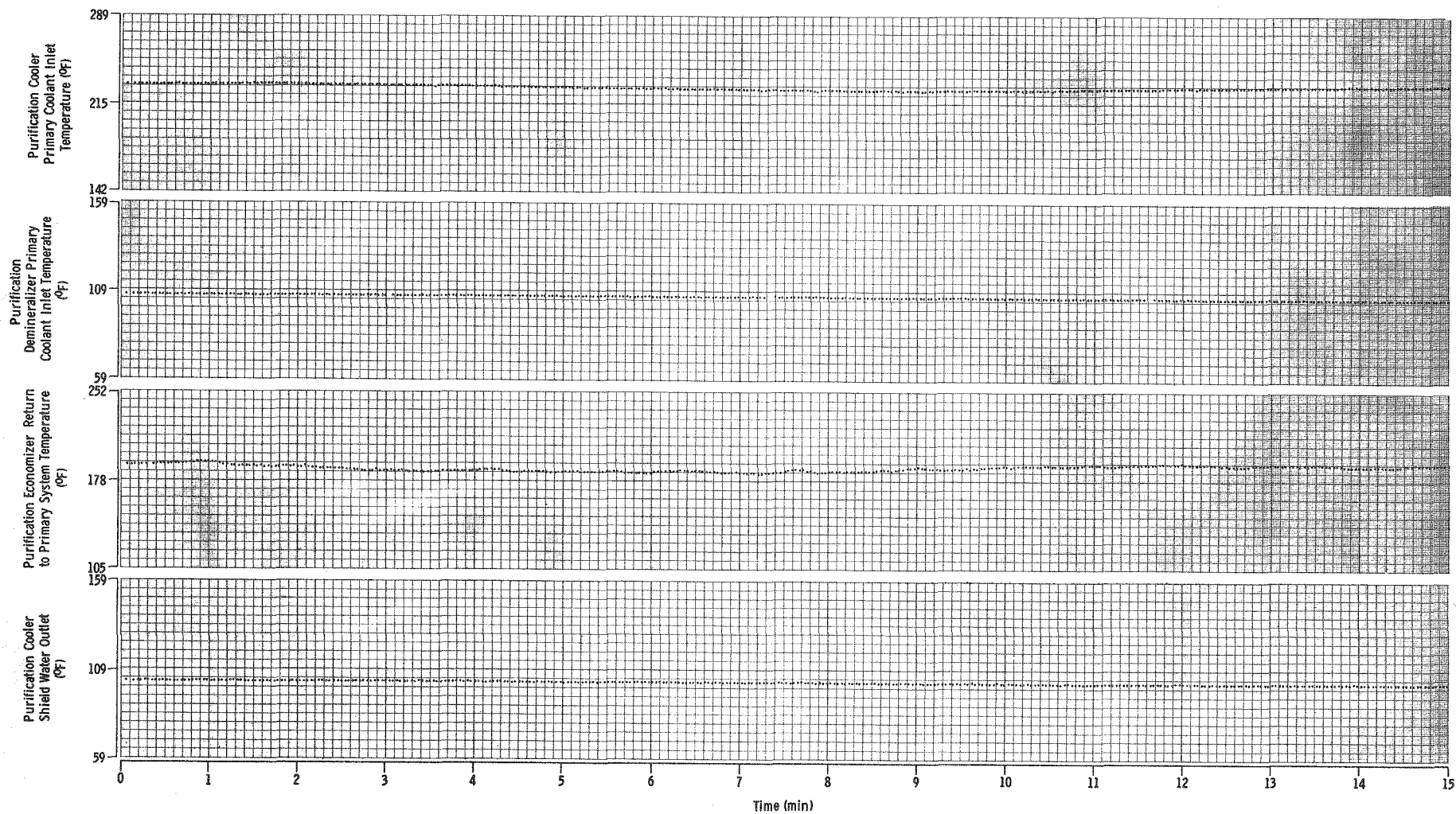
RUN XVI--ONE-ROD INSERTION AT 1000 KW NET (NO. 2 ROD, 2 IN.)

10-V



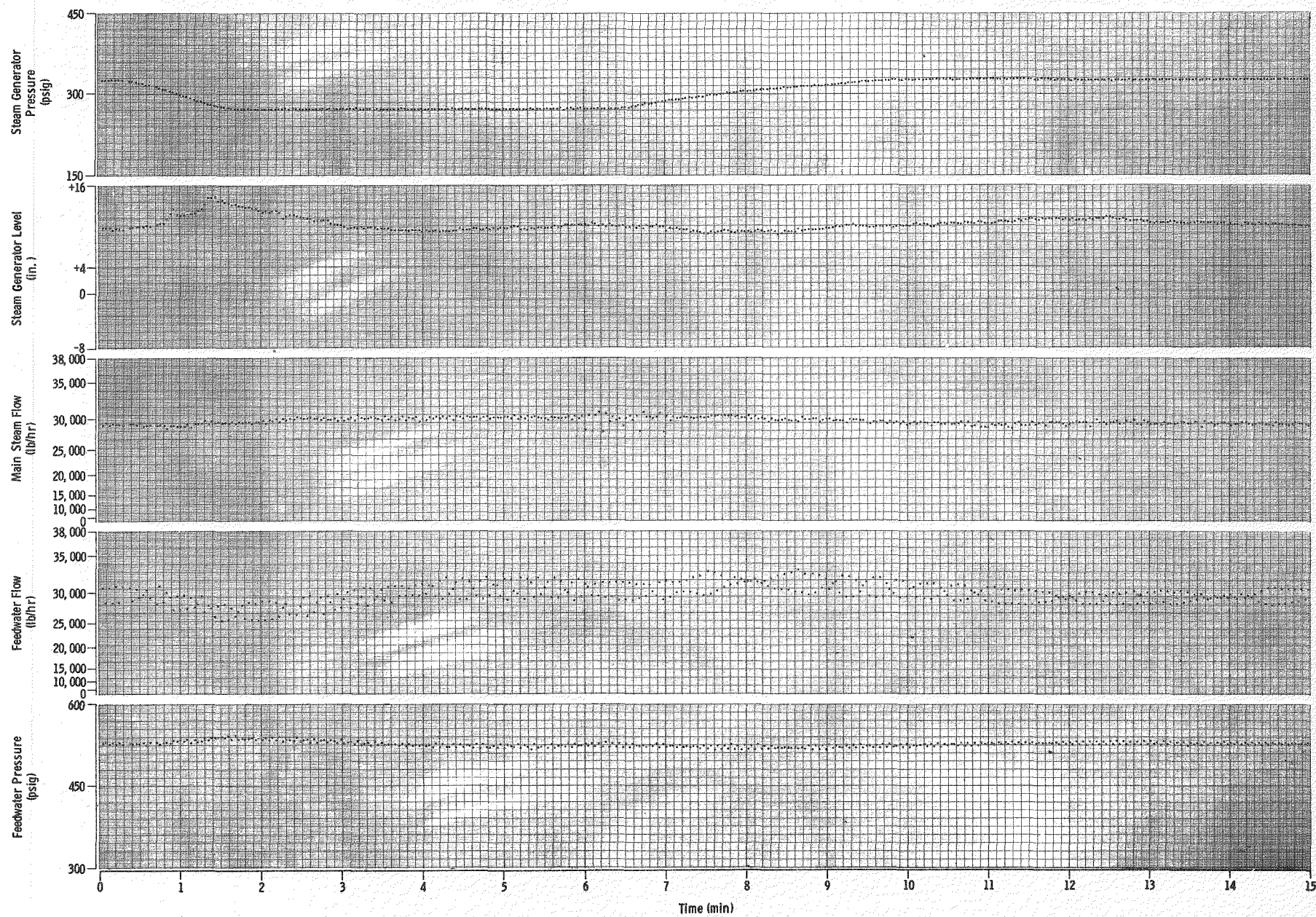
RUN XVI--ONE-ROD INSERTION AT 1000 KW NET (NO. 2 ROD, 2 IN.) (CONTINUED)

A-42



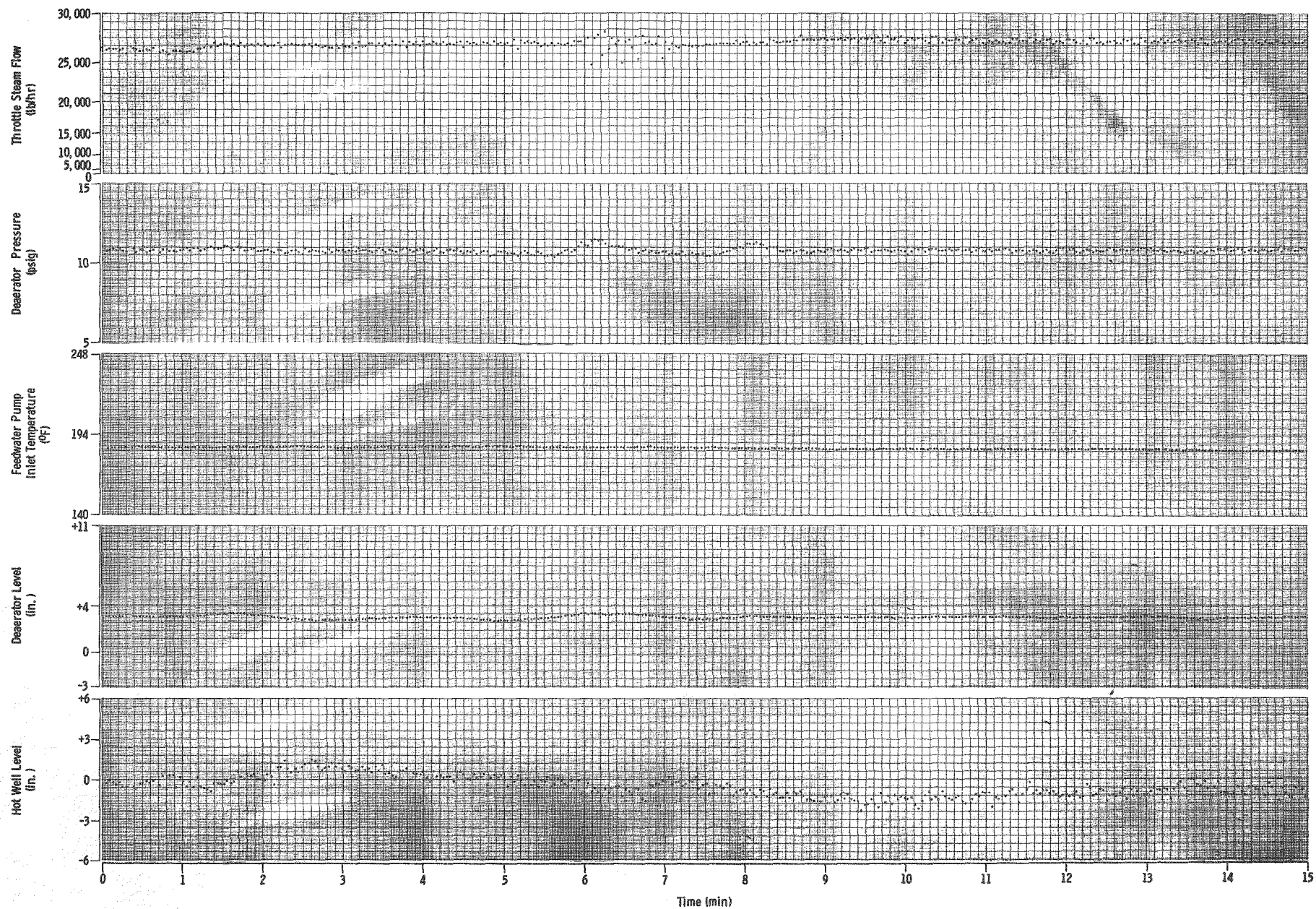
RUN XVI--ONE-ROD INSERTION AT 1000 KW NET (NO. 2 ROD, 2 IN.) (CONTINUED)

A-4



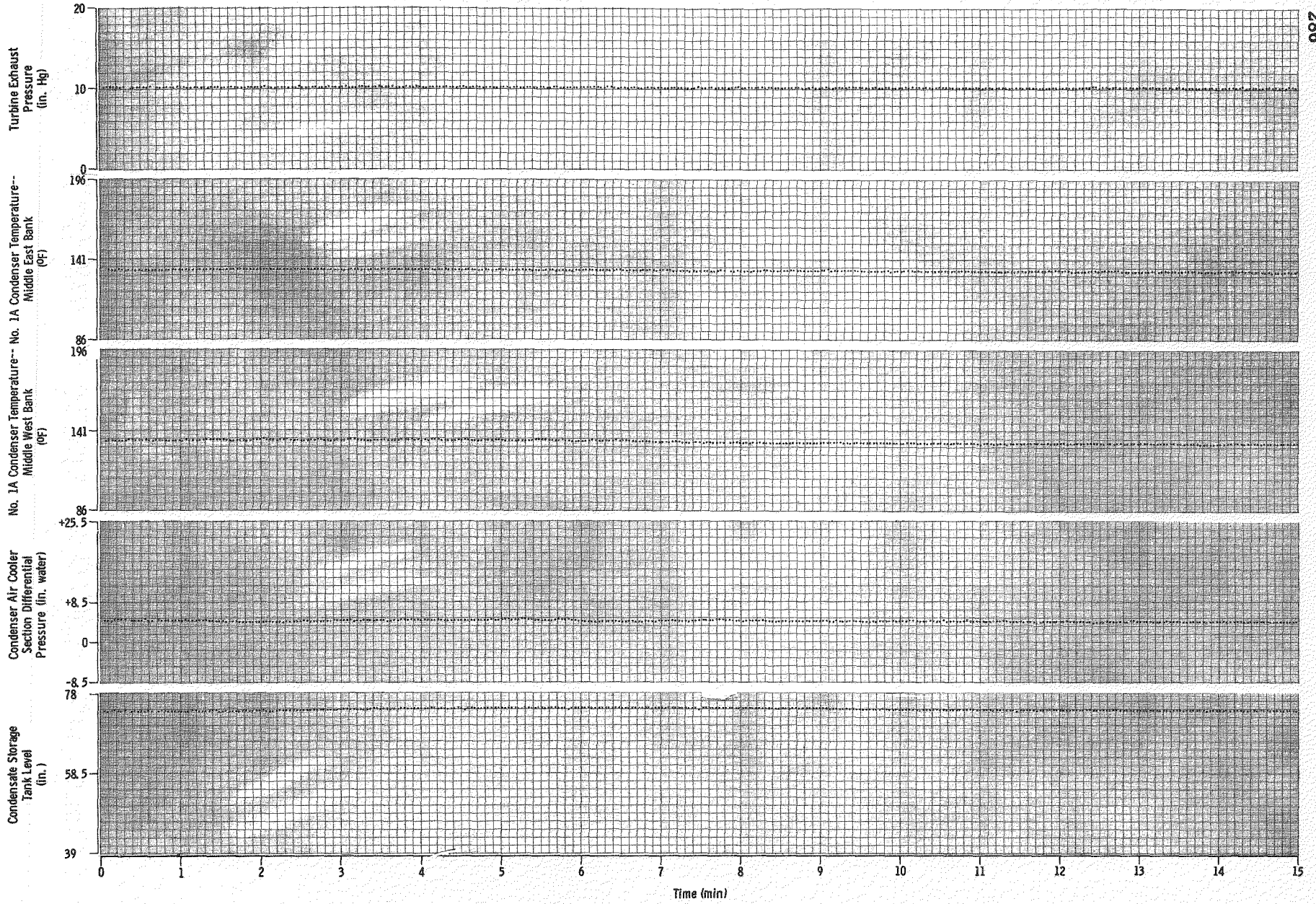
RUN XVI--ONE-ROD INSERTION AT 1000 KW NET (NO. 2 ROD, 2 IN.) (CONTINUED)

A-44



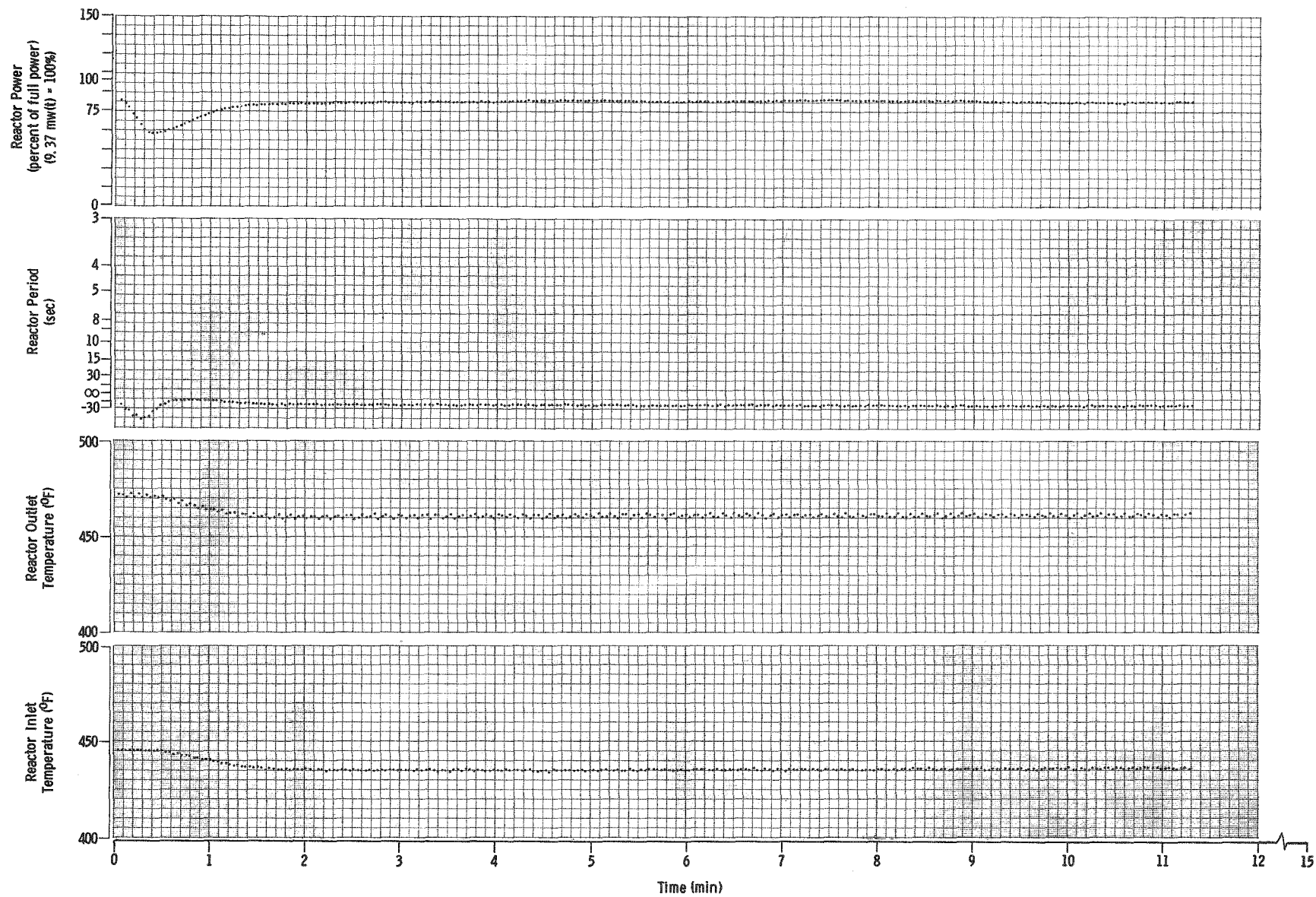
RUN XVI--ONE-ROD INSERTION AT 1000 KW NET (NO. 2 ROD, 2 IN.) (CONTINUED)

A-45



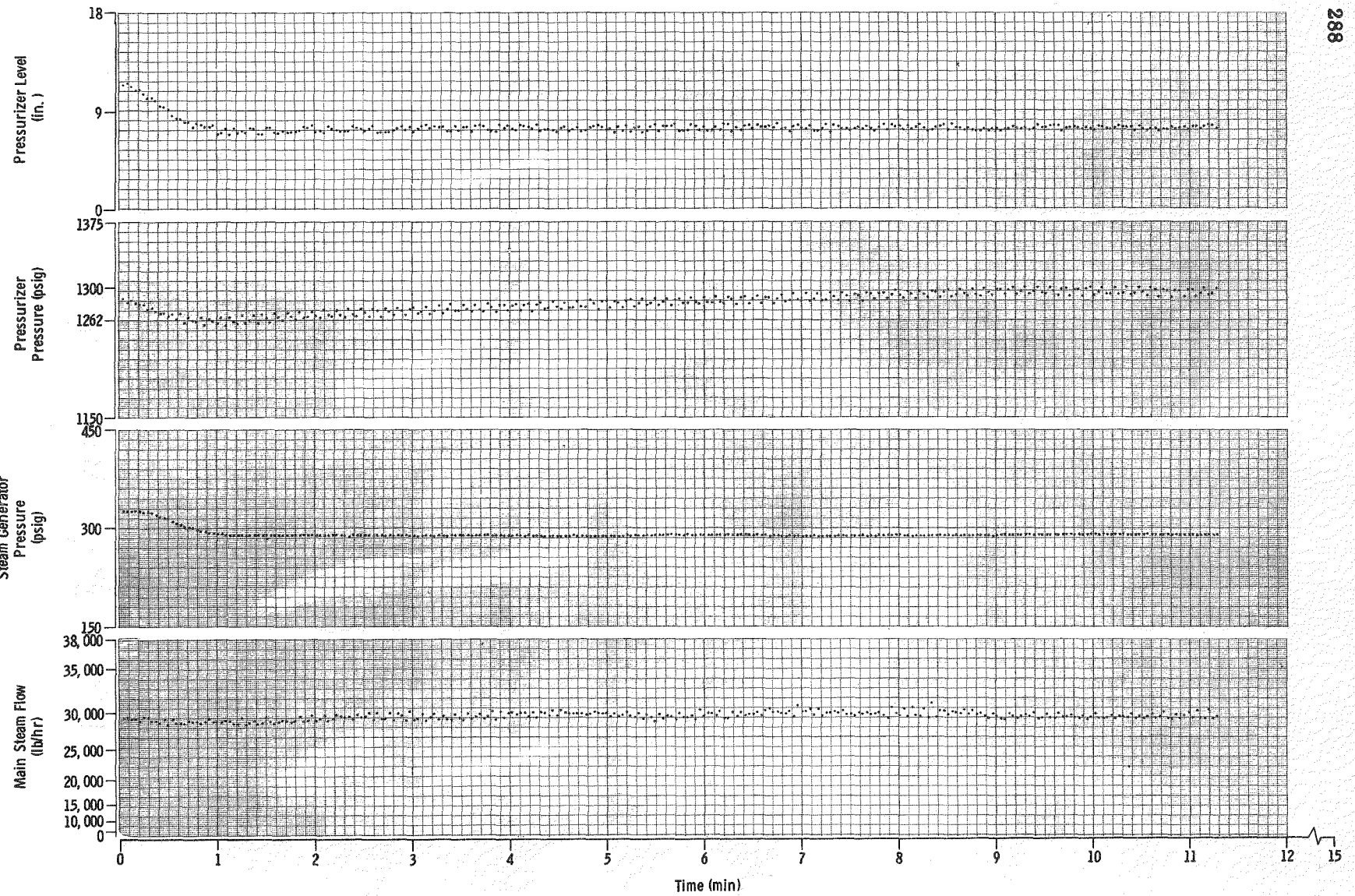
RUN XVI--ONE-ROD INSERTION AT 1000 KW NET (NO. 2 ROD, 2 IN.) (CONTINUED)

A-46



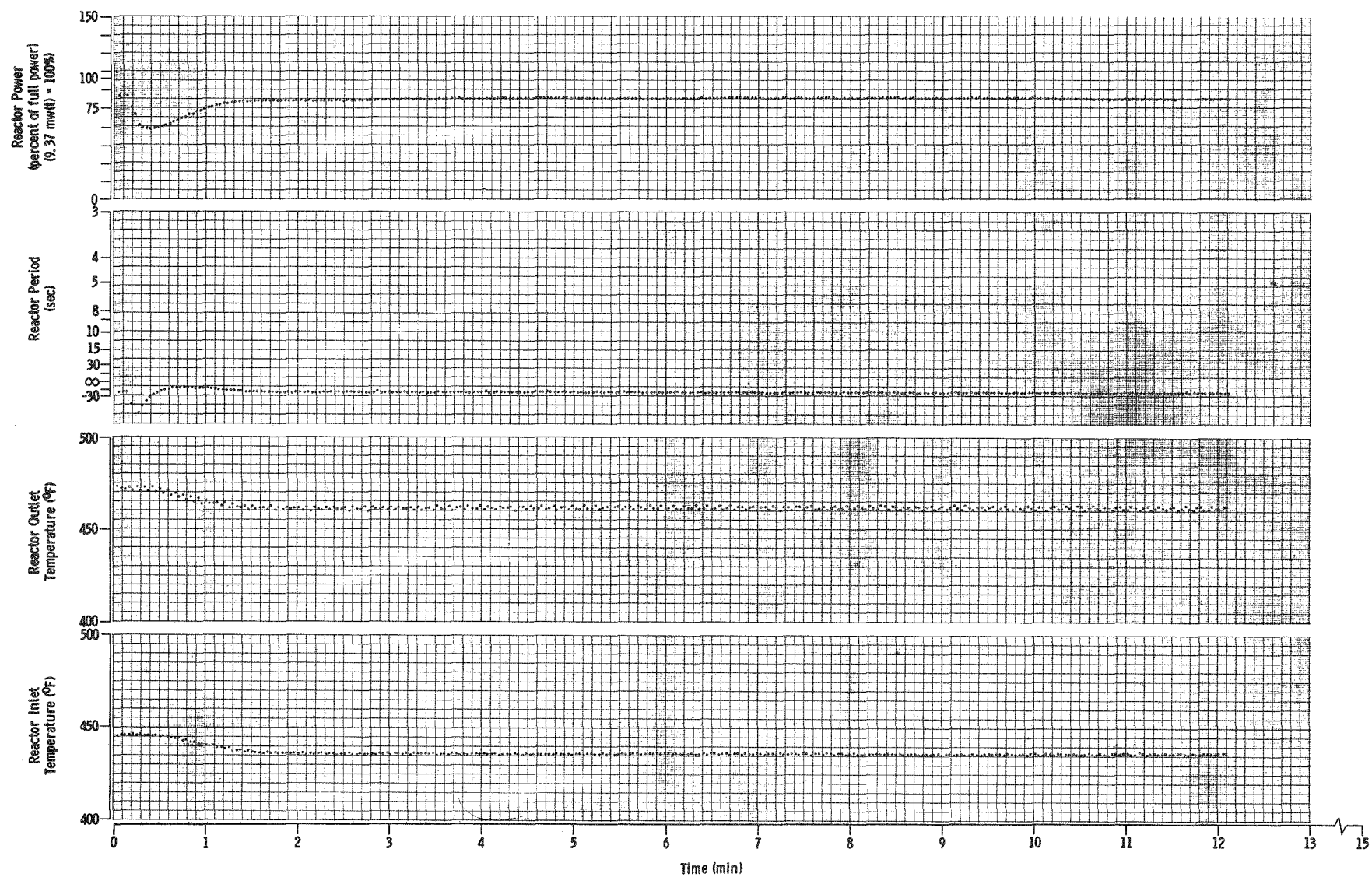
RUN XVII--THREE-ROD INSERTION AT 1000 KW NET (NO. 2, 4, 6 RODS, 1/2 IN.)

A-47



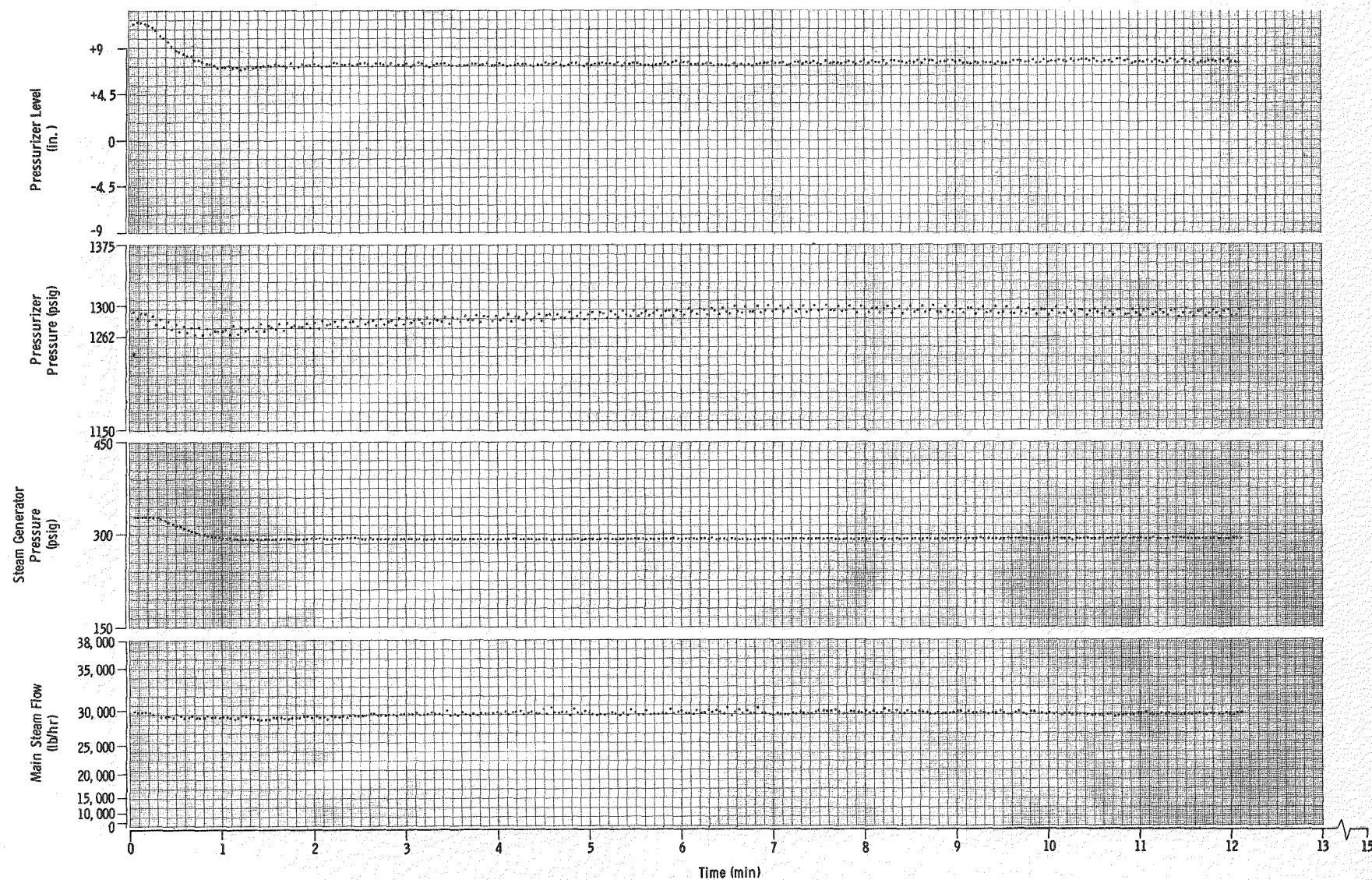
RUN XVII--THREE-ROD INSERTION AT 1000 KW NET (NO. 2, 4, 6 RODS, 1/2 IN.) (CONTINUED)

A-48



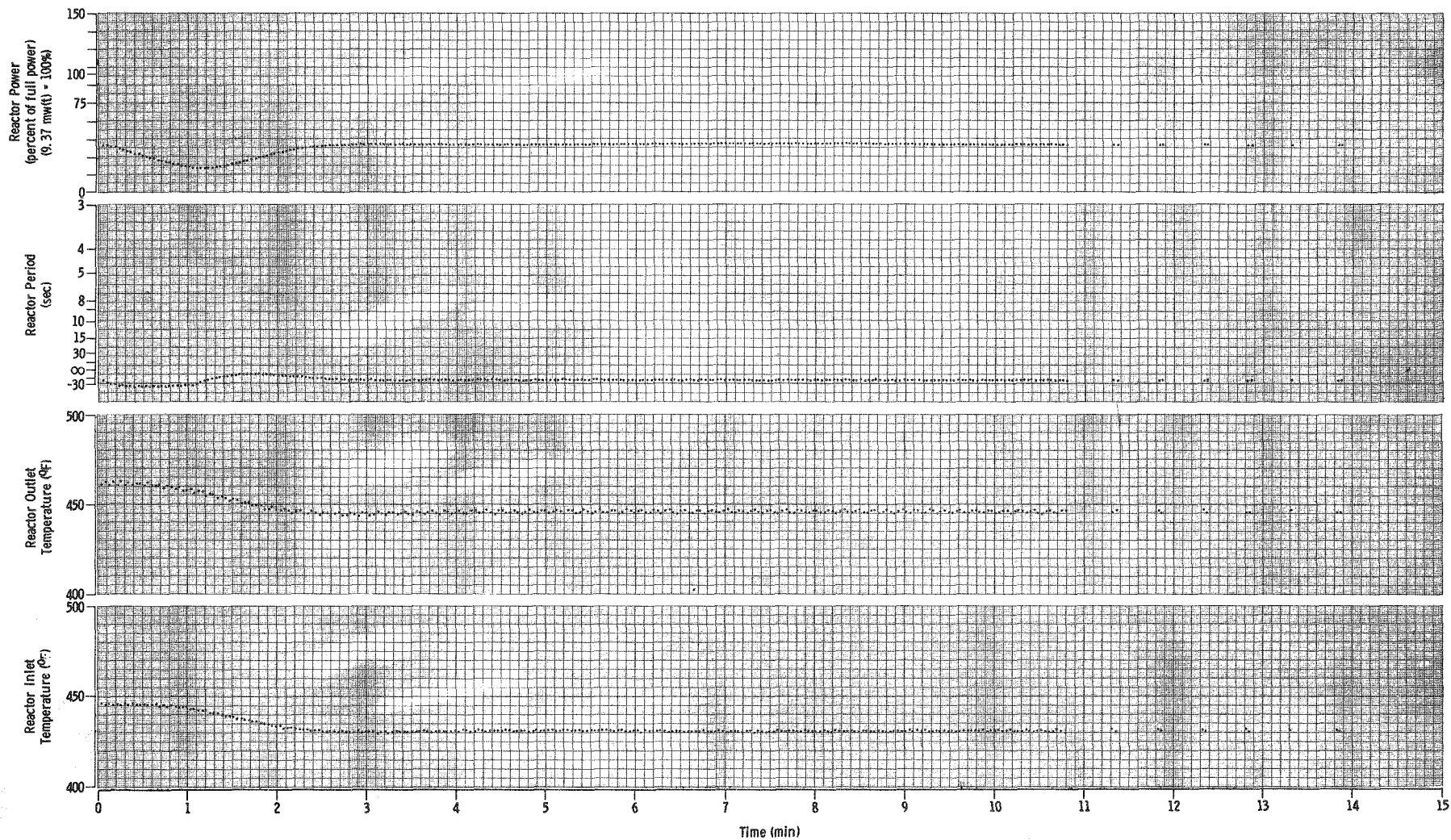
RUN XVIII--SIX-ROD INSERTION AT 1000 KW NET (NO. 1, 2, 3, 4, 5, 6 RODS, 1/4 IN.)

A-49

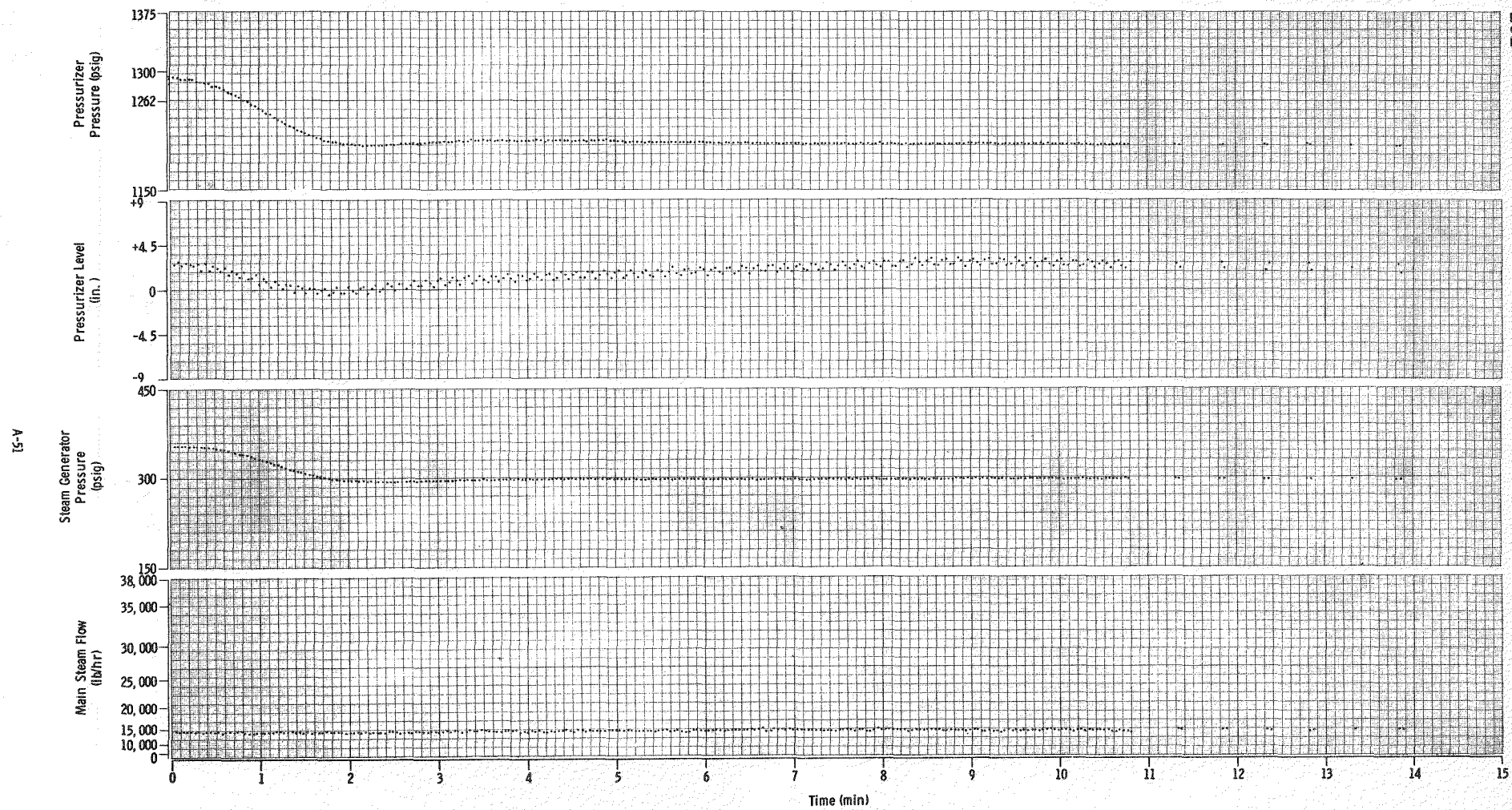


RUN XVIII--SIX-ROD INSERTION AT 1000 KW NET (NO. 1, 2, 3, 4, 5, 6 RODS, 1/4 IN.) (CONTINUED)

A-50

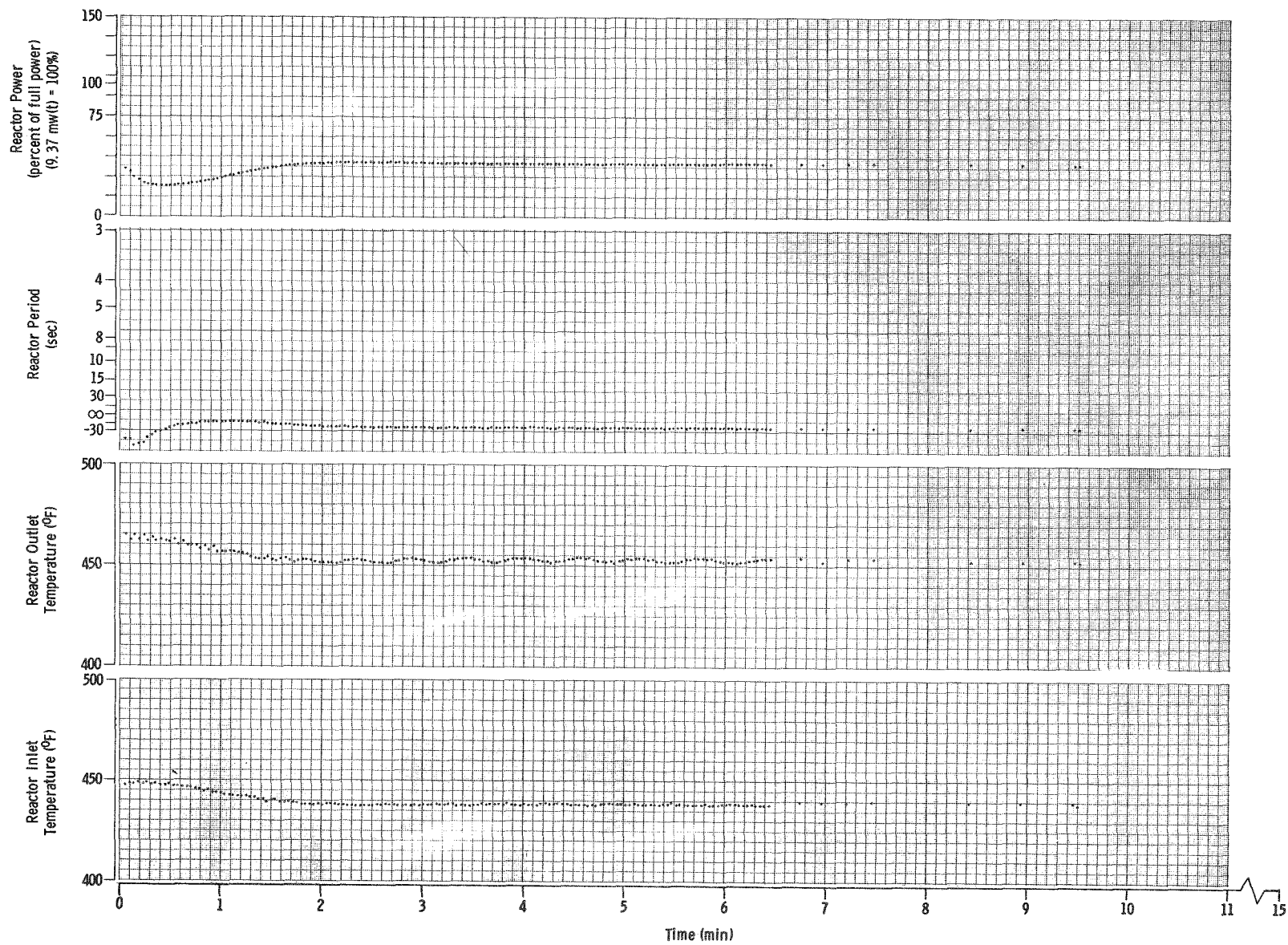


RUN XIX--ONE-ROD INSERTION AT 200 KW NET (NO. 2 ROD, 2 IN.)



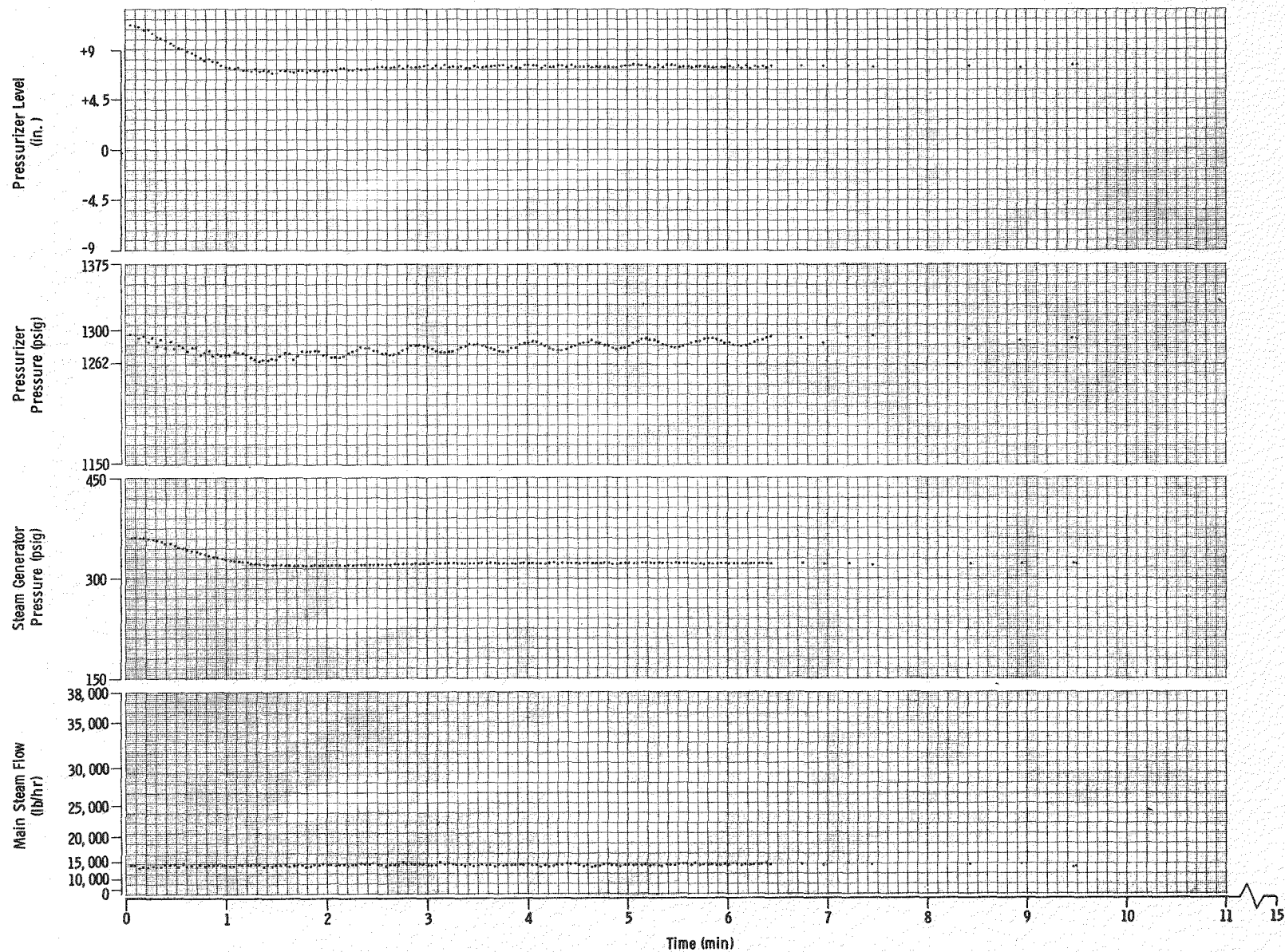
RUN XIX--ONE-ROD INSERTION AT 200 KW NET (NO. 2 ROD, 2 IN.) (CONTINUED)

A-52



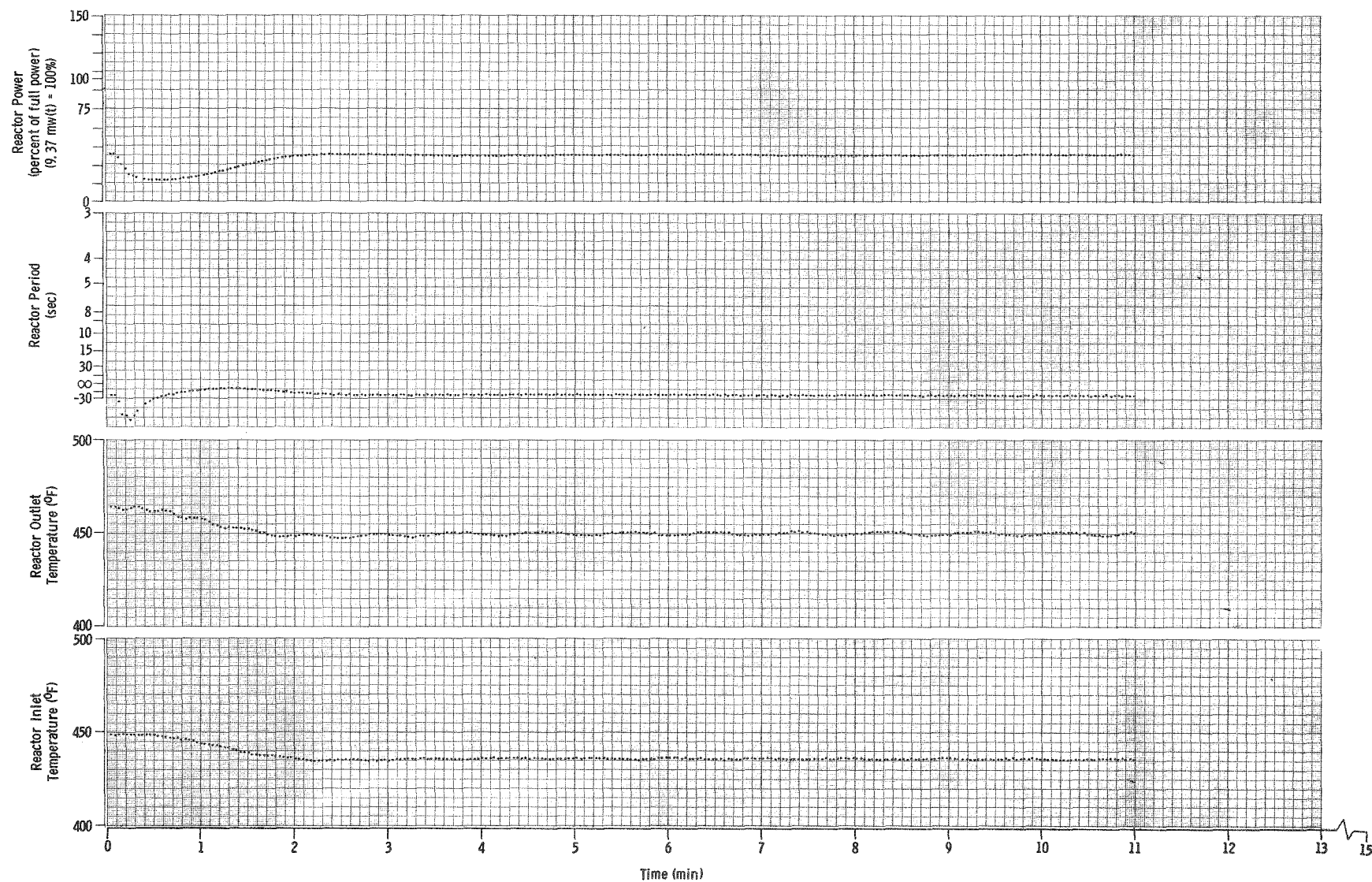
RUN XX--THREE-ROD INSERTION AT 200 KW NET (NO. 1, 3, 5 RODS, 1/2 IN.)

A-53



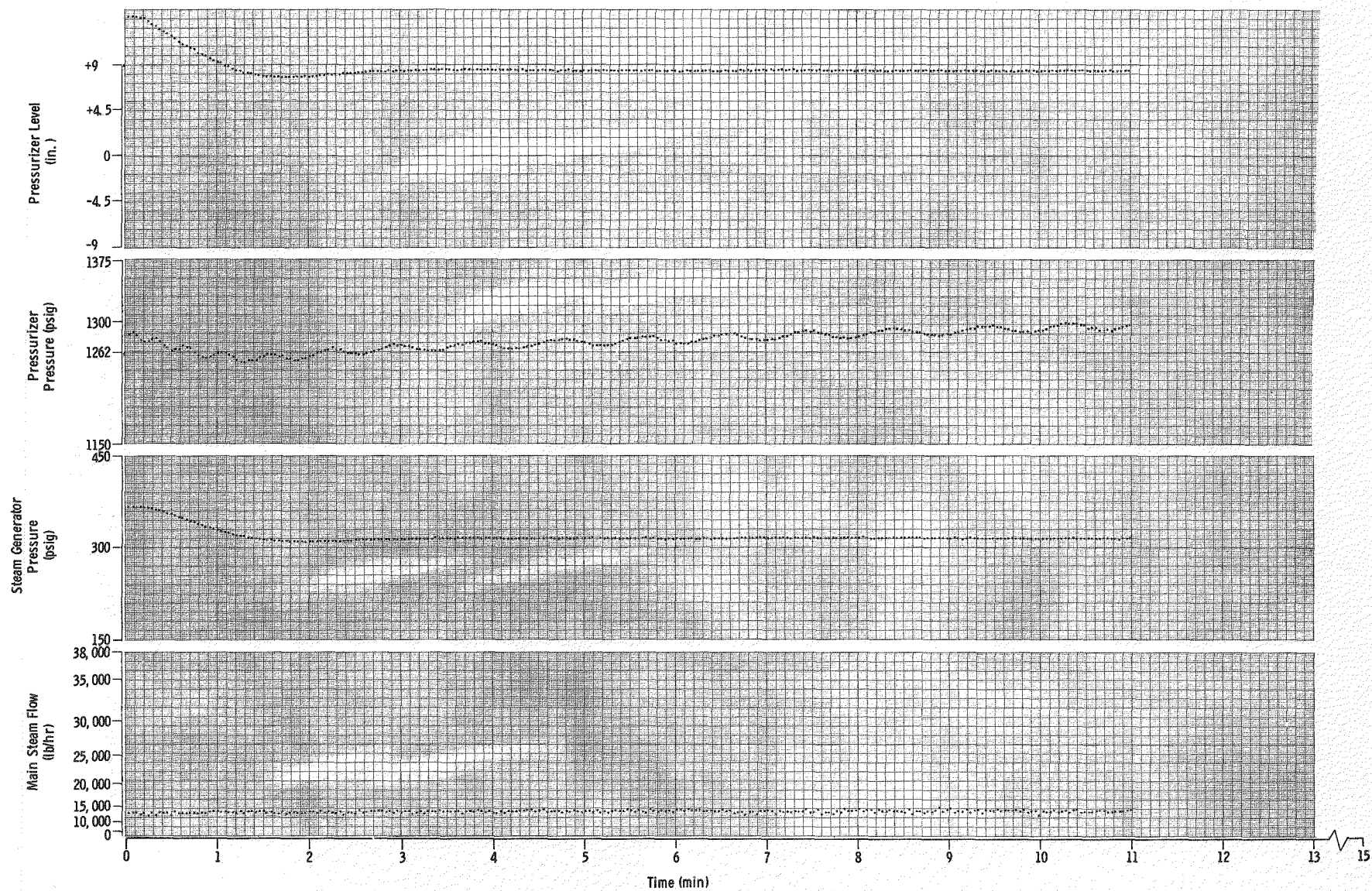
RUN XX--THREE-ROD INSERTION AT 200 KW NET (NO. 1, 3, 5 RODS, 1/2 IN.) (CONTINUED)

A-54



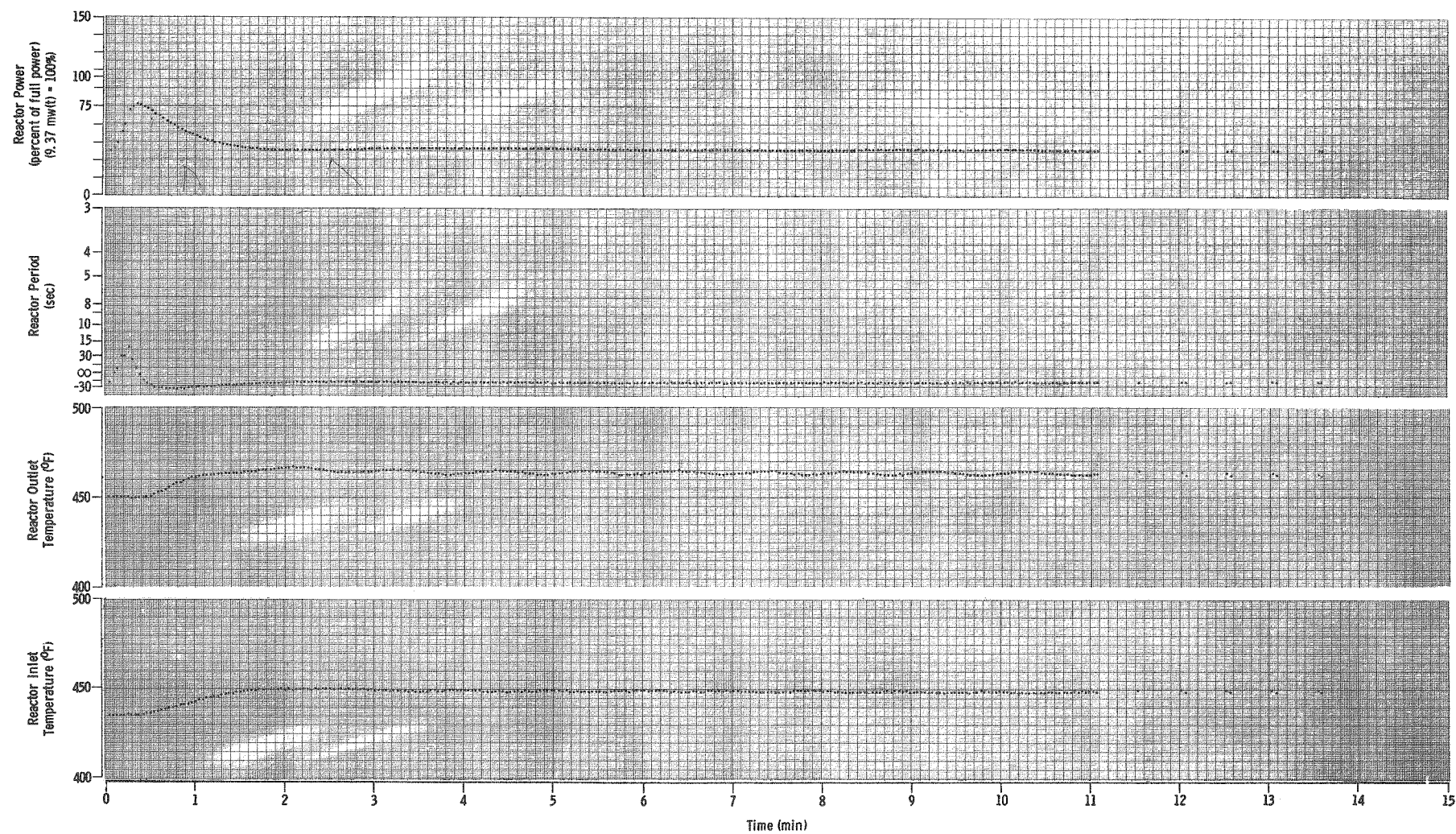
RUN XXI--SIX-ROD INSERTION AT 200 KW NET (NO. 1, 2, 3, 4, 5, 6 RODS, 1/4 IN.)

A-55



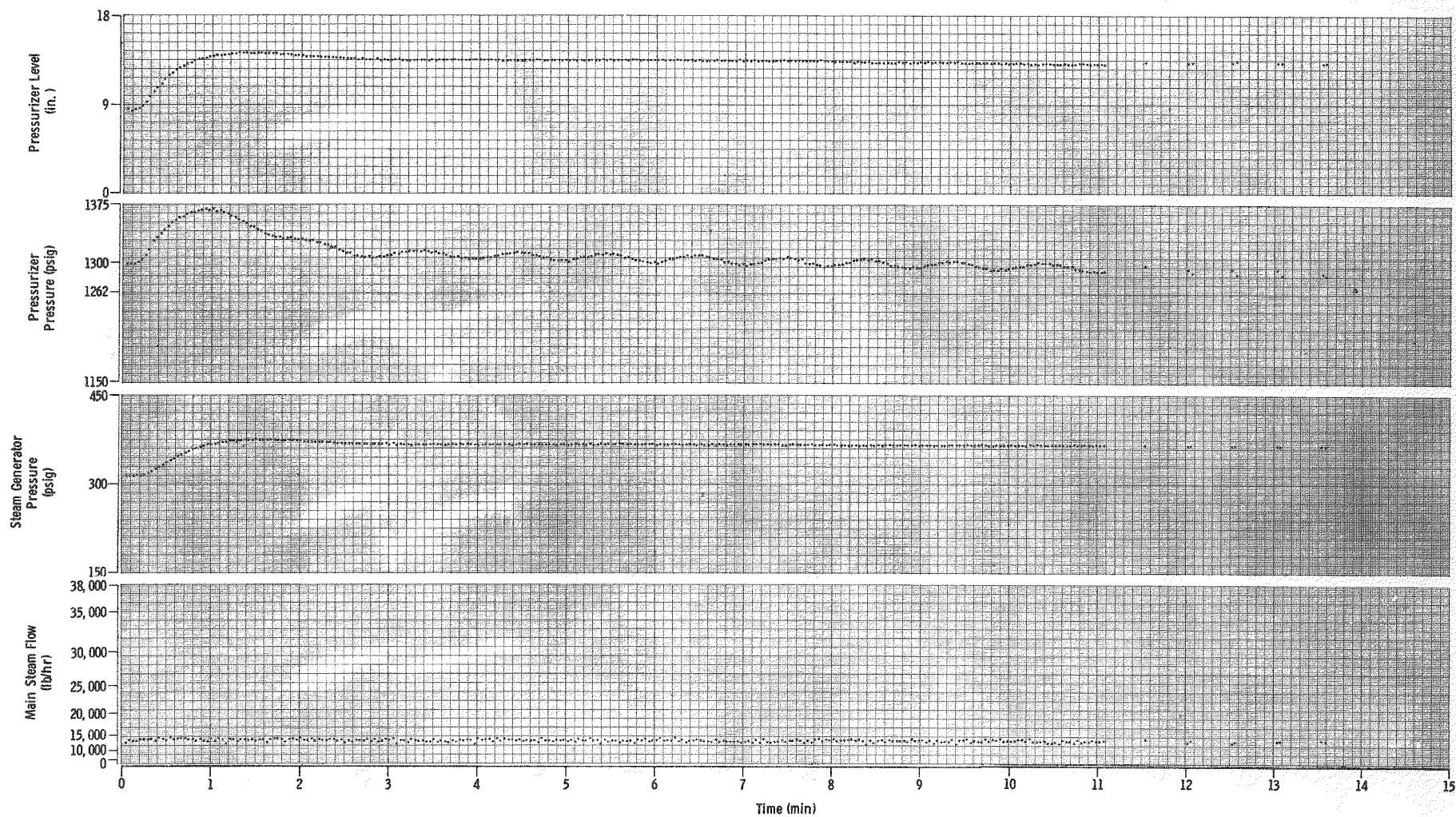
RUN XXI--SIX-ROD INSERTION AT 200 KW NET (NO. 1, 2, 3, 4, 5, 6 RODS, 1/4 IN.) (CONTINUED)

A-56



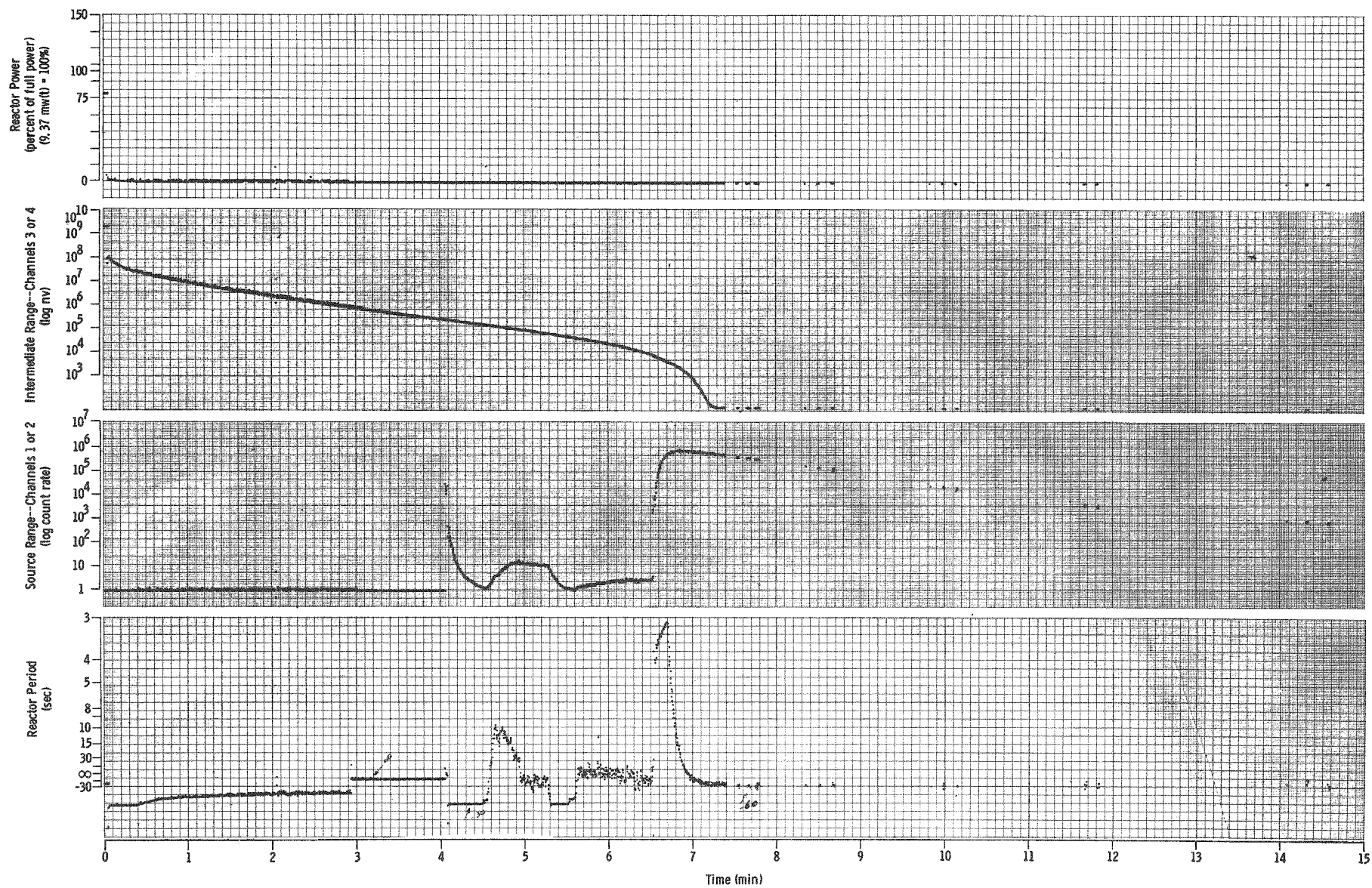
RUN XX11--SIX-ROD EXTRACTION AT 200 KW NET (NO. 1, 2, 3, 4, 5, 6 RODS, 1/4 IN.)

A-57



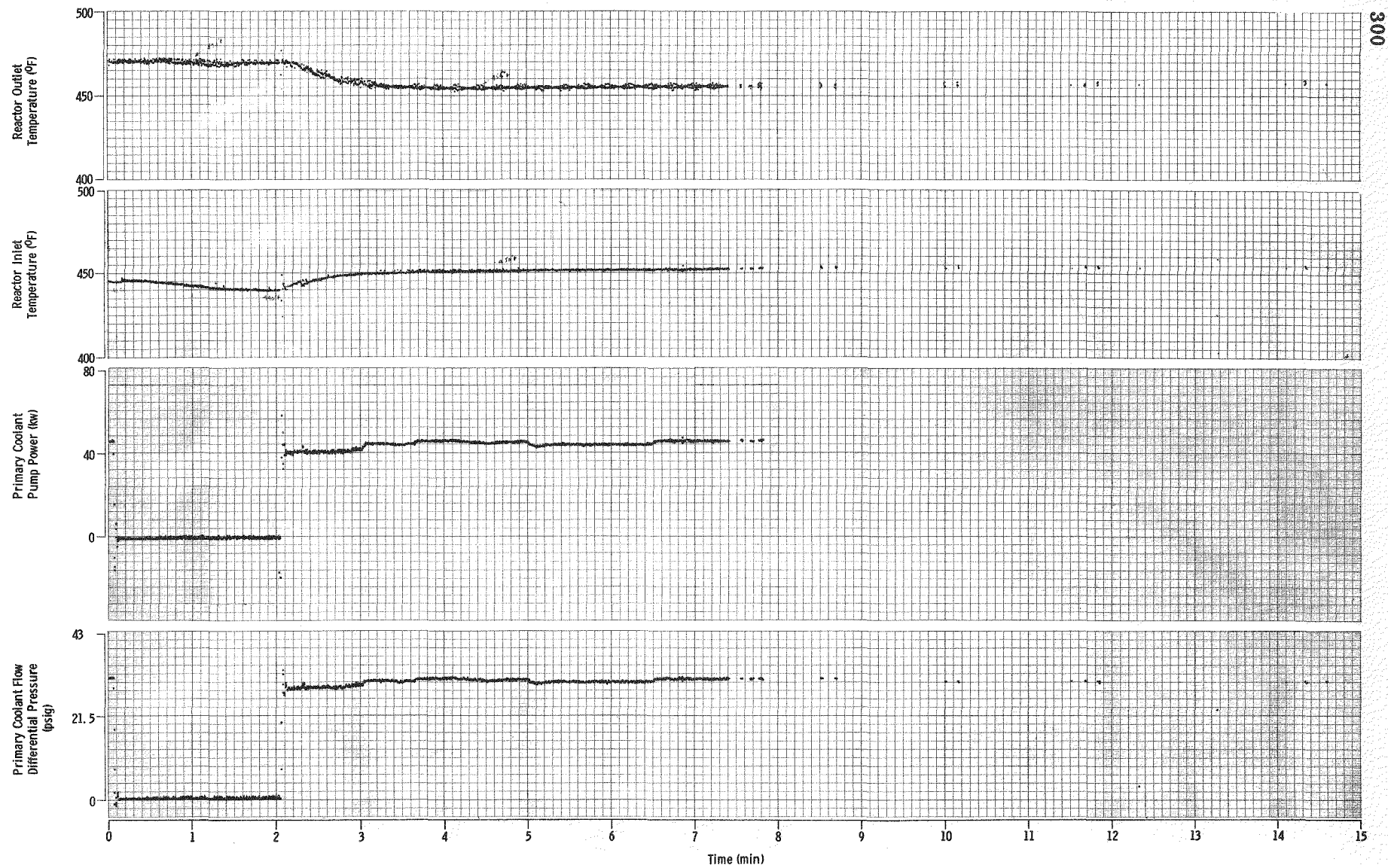
RUN XX11--SIX-ROD EXTRACTION AT 200 KW NET (NO. 1, 2, 3, 4, 5, 6 RODS, 1/4 IN.) (CONTINUED)

A-58



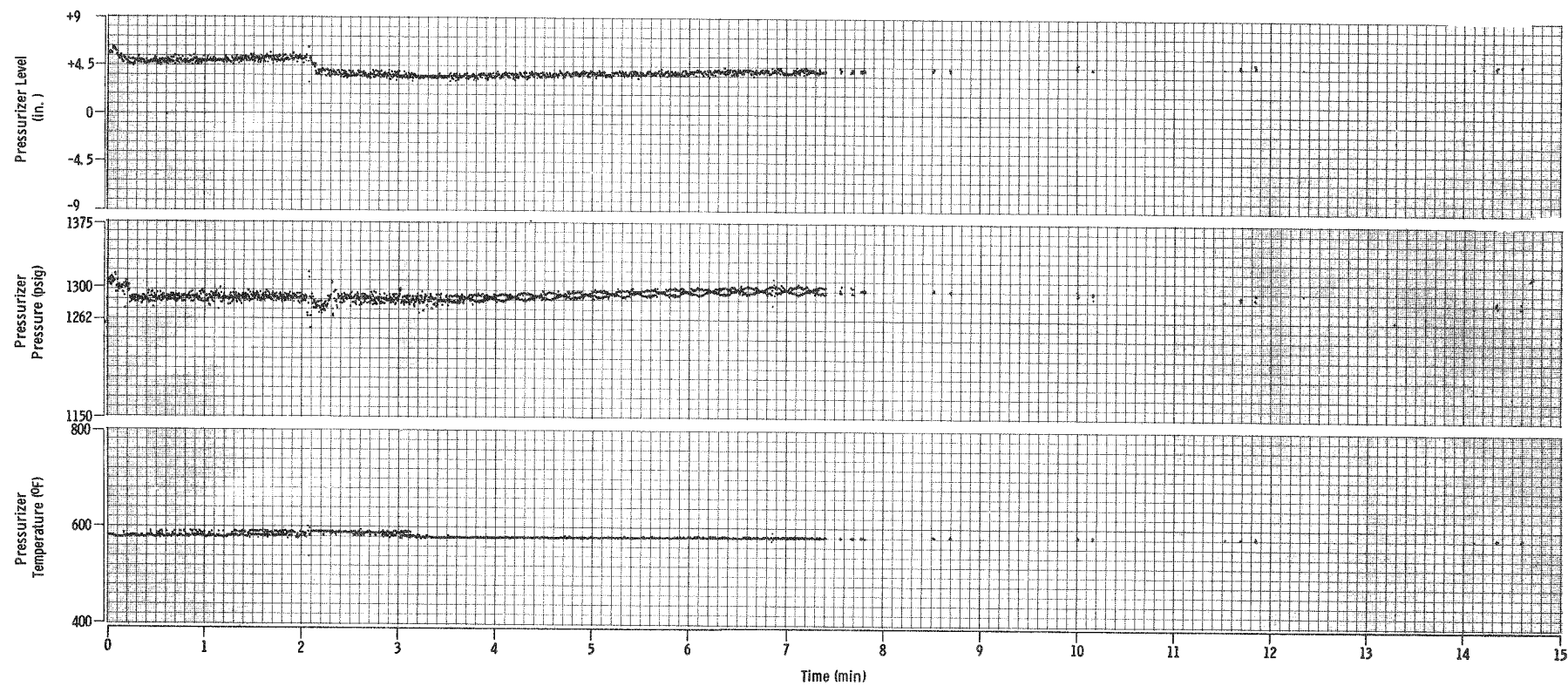
RUN XXIII--SCRAM FROM 80% REACTOR POWER

A-59



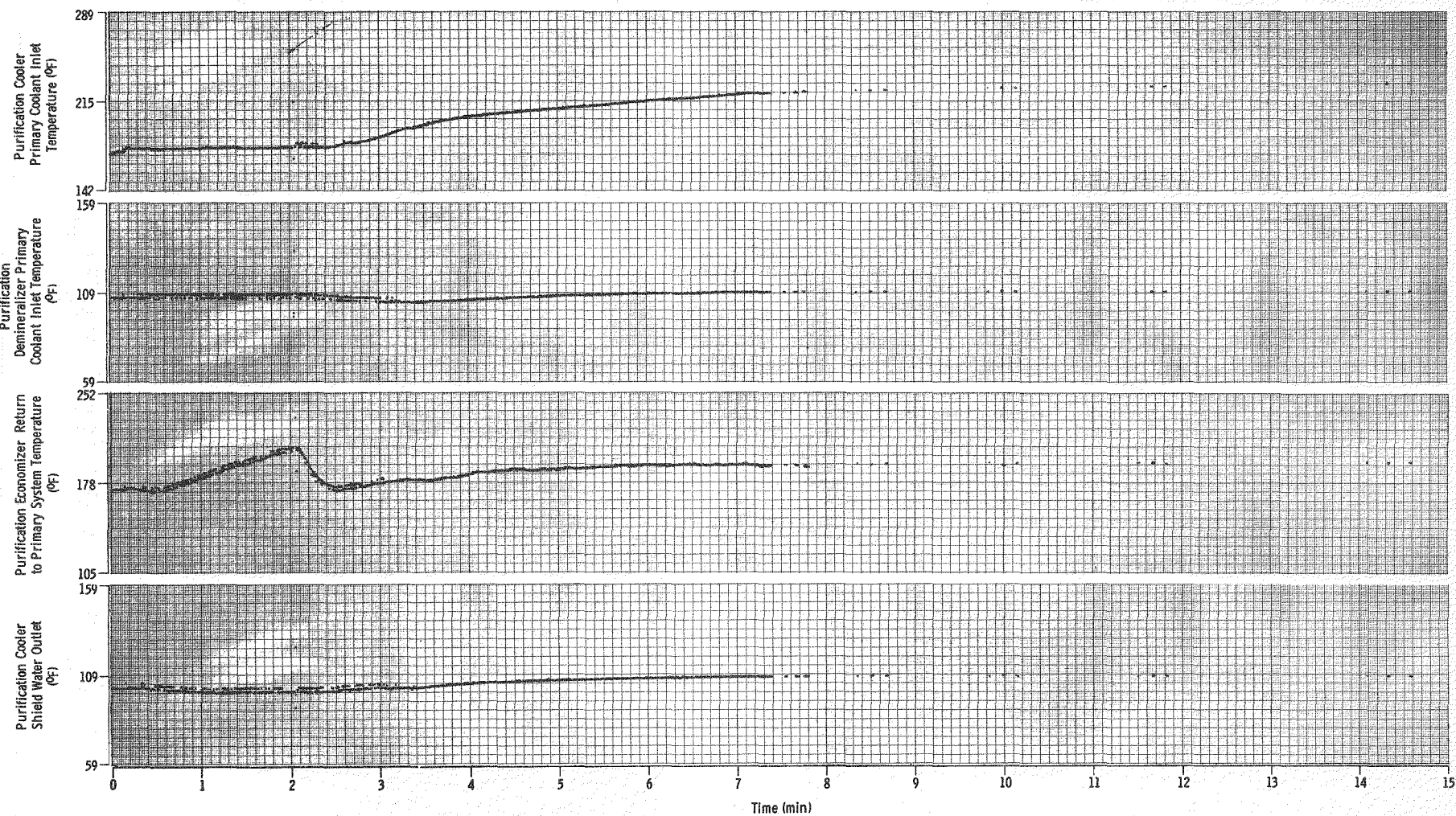
RUN XXIII--SCRAM FROM 80% REACTOR POWER (CONTINUED)

A-60



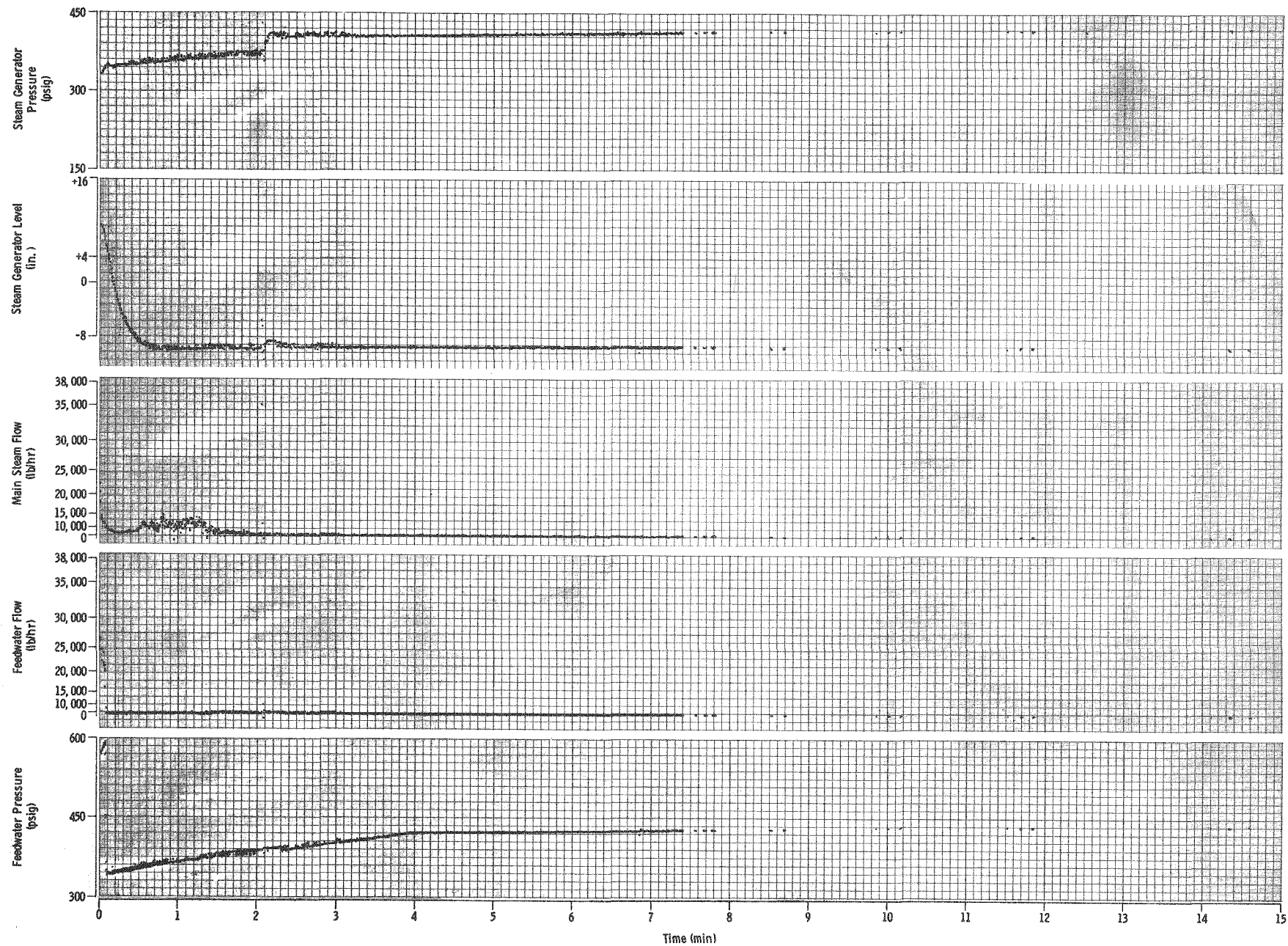
RUN XXIII--SCRAM FROM 80% REACTOR POWER (CONTINUED)

A-61



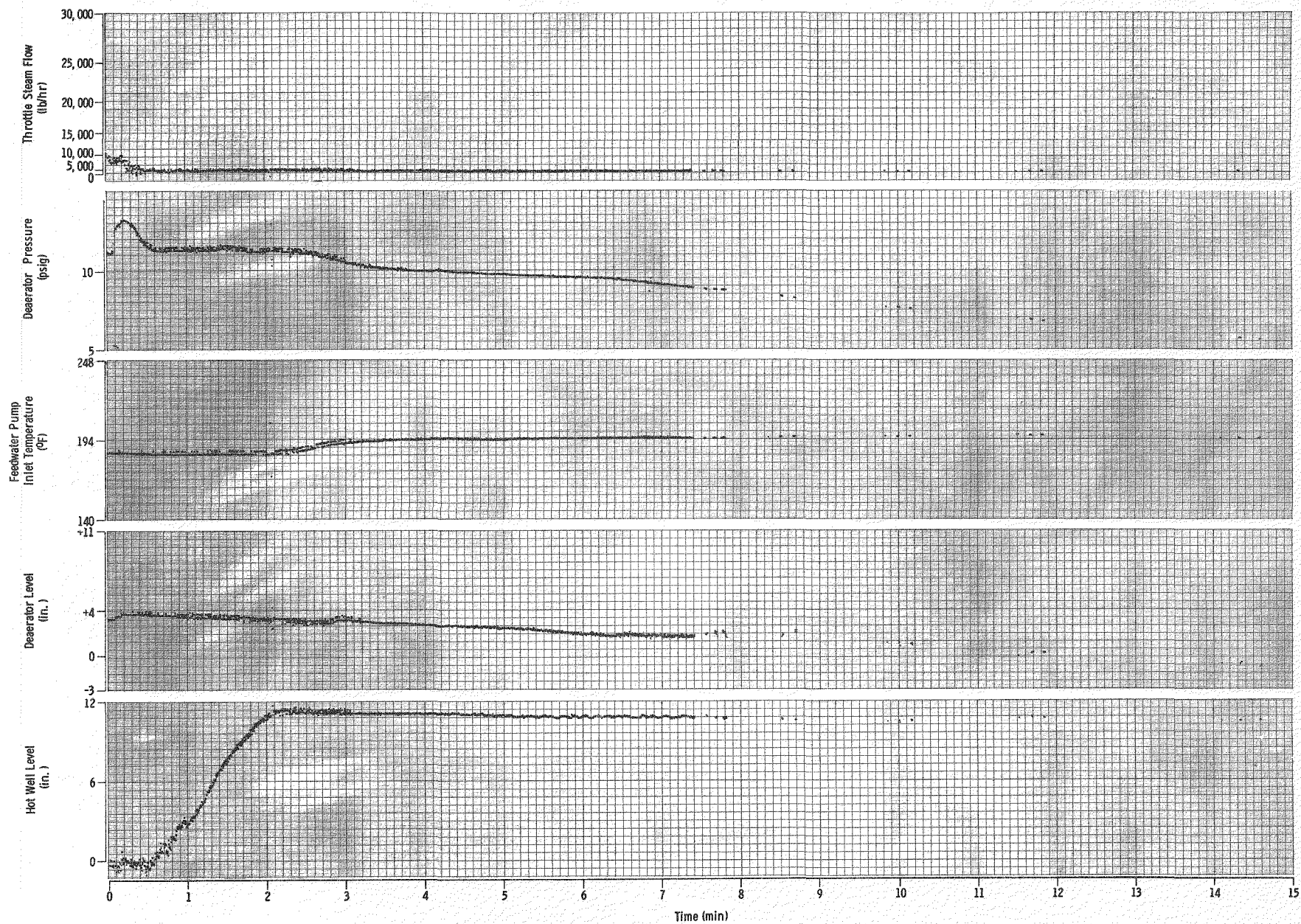
RUN XX111--SCRAM FROM 80% REACTOR POWER (CONTINUED)

A-62



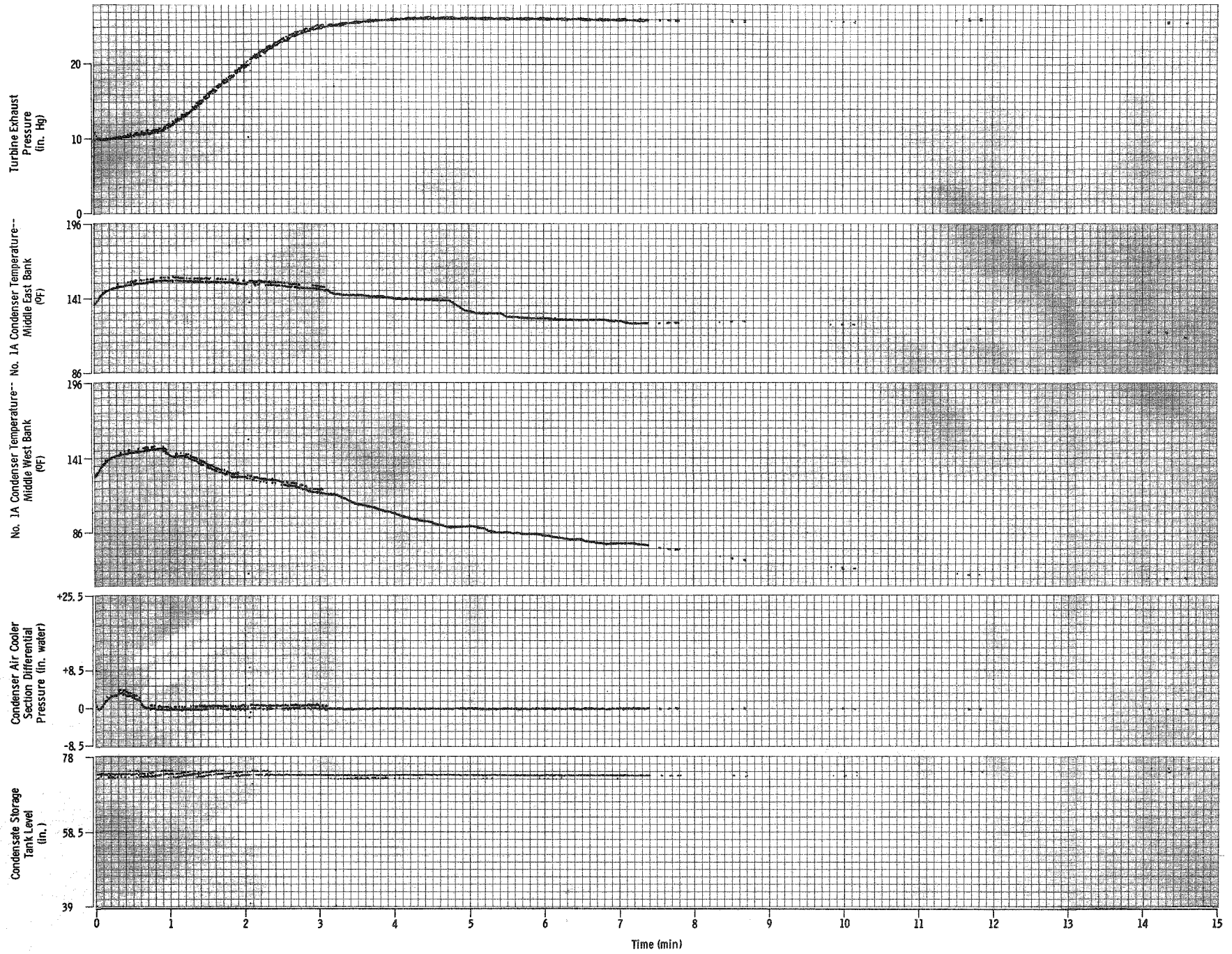
RUN XXIII--SCRAM FROM 80% REACTOR POWER (CONTINUED)

A-63



RUN XX111--SCRAM FROM 80% REACTOR POWER (CONTINUED)

A-64



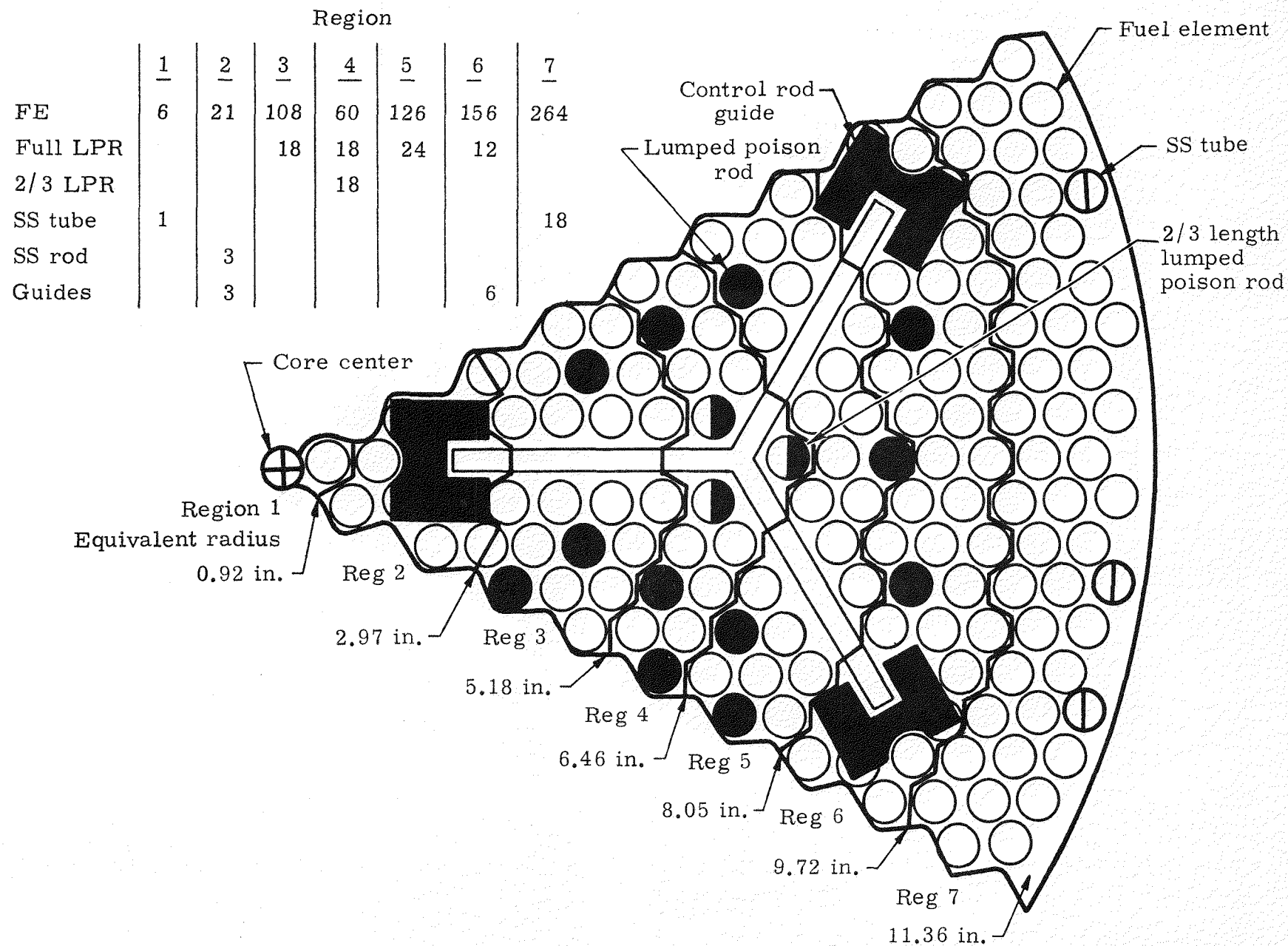


Fig. D-1. Seven Radial Region PM-1 Core Configuration

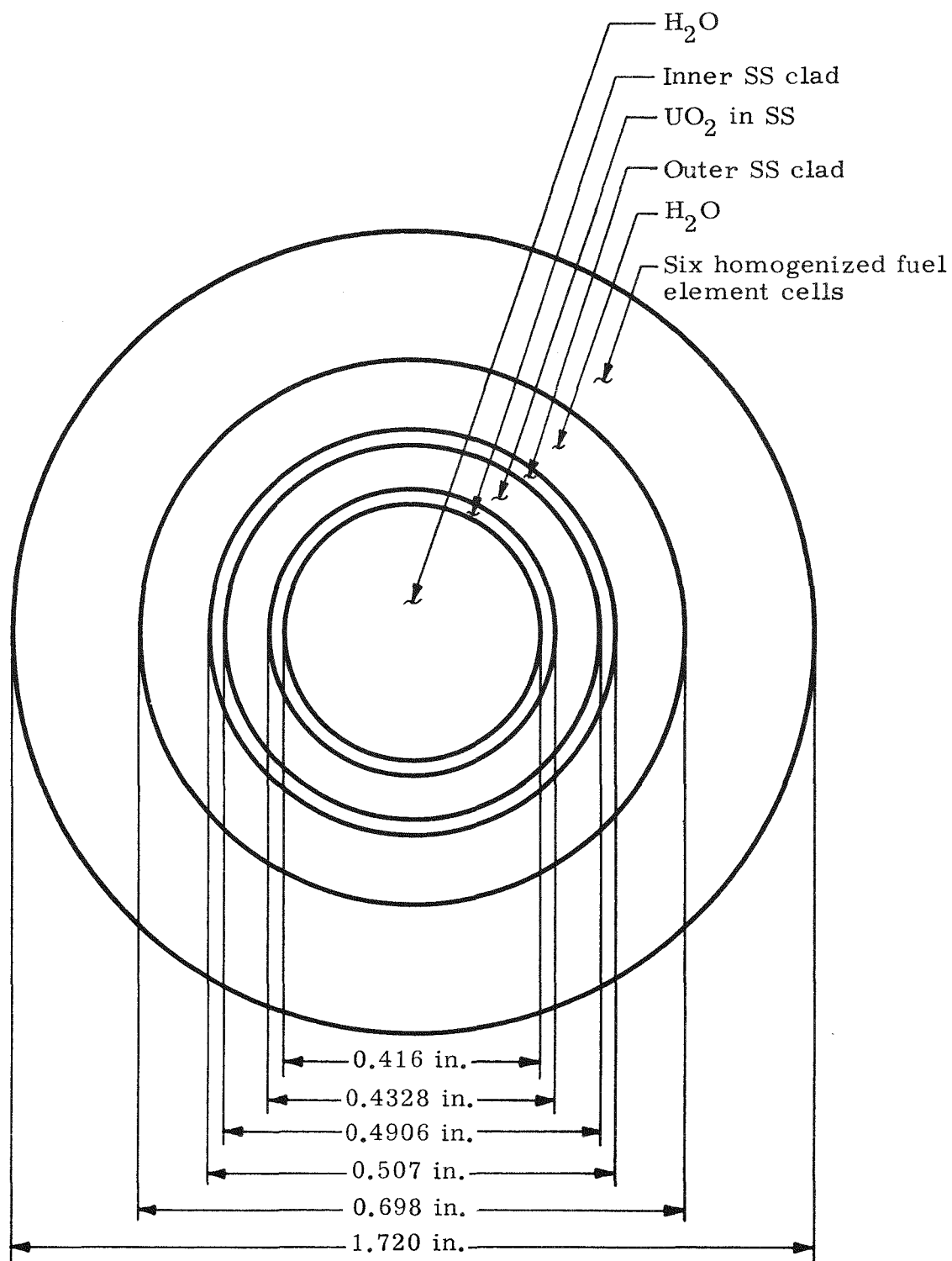


Fig. D-2. Fuel Element Cell

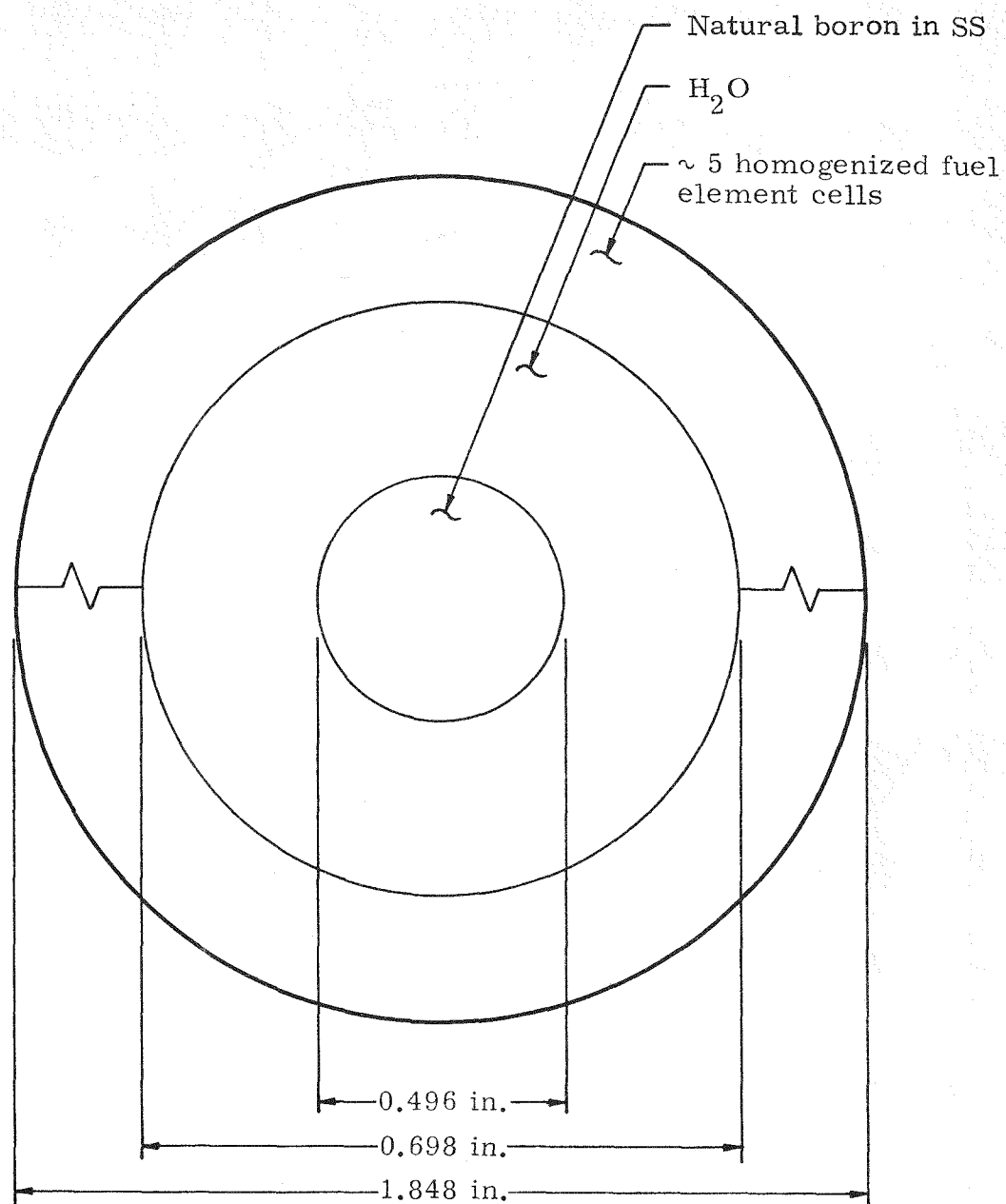


Fig. D-3. Lumped Poison Rod Cell